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

ENCLOSURE 2

November 30, 1993

MEMORANDUM FOR: The Chairman
Commissioner Rogers
Commissioner Remick
Commissioner de Planque

FROM: James M. Taylor
Executive Director for Operations

SUBJECT: STATUS REPORT ON PRIMARY WATER STRESS CORROSION CRACKING OF PWR REACTOR VESSEL HEAD PENETRATION CRACKING

This memo provides an update on primary water stress corrosion cracking of Alloy 600 (Inconel™ 600) in PWR reactor pressure vessel head penetrations. The staff received safety assessments from NUMARC prepared by the Westinghouse Owners Group (WOG), Combustion Engineering Owners Group (CEOG) and B&W Owners Group (B&WOG) on the consequences of postulated cracking of the control rod drive mechanism (CRDM) penetrations or control element drive mechanism (CEDM) penetrations through the reactor vessel head and has prepared a safety evaluation on this issue. The safety evaluation is attached.

The information submitted to the NRC staff to date, including the inspection results from foreign plants, continue to confirm the staff's view that this issue is of low safety significance. The analyses performed by the three owners groups indicate that any cracking which occurs should be predominantly axial in orientation. This is consistent with actual inspection results. All cracks reported to date, with perhaps one exception, are short in length and predominantly axially oriented. The one circumferential crack that was detected appears to be a fabrication rather than a service induced defect. Hence, ejection of a CRDM continues to be an unlikely event. Further, the effects of wastage by borated water on a creviced area, such as between CRDM penetration and the reactor head, have been evaluated on the bases of laboratory testing and similar field experience. The results indicate that any degradation would occur very slowly and, therefore, any boric acid corrosion of the reactor vessel shell would be detected by surveillance walkdowns required by Generic Letter (GL) 88-05 before significant wall thinning could occur. On the basis of the staff's review of the owners groups' safety evaluations and the overseas inspection findings, the staff has concluded that the CRDM/CEDM cracking at the reactor vessel heads is not a significant safety issue at this time. The staff recommended that NUMARC consider 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 industry has committed to conduct nondestructive examinations at three units in 1994. They are:

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

On July 20, 1993, NUMARC submitted to the NRC proposed flaw acceptance criteria to be used in dispositioning any flaws found during CRDM/CEDM examinations. The staff accepted the criteria for axial cracks because the criteria conform to the ASME Section XI criteria. However, based upon information submitted to date and the more serious safety consequences of circumferential flaws, the staff did not accept NUMARC's proposed criteria for circumferential flaws. Circumferential flaws found during CRDM/CEDM inspections will be dispositioned following review by the staff on a case-by-case basis.

The industry is developing remotely operated inspection equipment and repair tools to reduce the radiation exposure associated with the examinations. On the basis of the low probability of the ejection of a CRDM and the low safety significance of CRDM leakage, the staff has concluded that there is sufficient time available for the industry to implement a well-conceived and well-planned inspection, evaluation, and repair program.

James M. Taylor
Executive Director
for Operations

Enclosure:
As stated

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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

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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-pla--ic. 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 ½ 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.

Technical Contacts: Robert A. Hermann (301) 504-2768
William H. Koo (301) 504-2706
James A. Davis (301) 504-2713

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

June 7, 1993

(10 C.F.R. § 2.206)

Mr. John Willis, Coordinator
Nuclear Campaign
Greenpeace International
1436 U Street, N.W.
Washington, D.C. 20009

Dear Mr. Willis:

On behalf of Greenpeace International (petitioner) you wrote on March 24, 1993 to the Chairman of the Nuclear Regulatory Commission (NRC) regarding all pressurized-water reactors (PWRs) now operating in the United States. You request immediate, full inspection of all vessel head penetrations (VHPs) in U.S. PWRs for cracking and publication of the results by the NRC. Because you also request that the NRC shut down affected reactors, whether the cracking is longitudinal or circumferential, your letter has been referred to the NRC staff for consideration as a petition for enforcement action under 10 CFR § 2.206. The petitioner further requests that the NRC staff "re-license" reactors which must be closed due to VHP cracking based on the assertion that repair or mitigation program for such cracks may negatively affect the configuration and effectiveness of safety systems.

The petitioner seeks relief based on allegations that (1) some cracking has been identified in VHPs in PWRs in France, Belgium, Switzerland, and Sweden; (2) testing in France revealed incipient circumferential cracking of some VHPs, which could lead to a through-wall break in the primary pressure boundary without fulfillment of the leak-before-break criterion; and (3) this phenomenon could cause the ejection of the control rod drive mechanism, with resulting loss of control of the reactor. The bases for the petitioner's request are described in more detail in "Vessel Head Penetration Cracking in Nuclear Reactors," Greenpeace International and Greenpeace Sweden, March 1993, which is attached to the petition.

With regard to the request for immediate inspections, the NRC staff understands the petitioner's concern for obtaining information on primary water stress corrosion cracking (PWSCC) through inspection of VHPs in U.S. PWRs; the NRC agrees that data developed by inspections will be needed to confirm the current understanding and evaluation of this issue. However, immediate inspection of all VHPs in U.S. PWRs is not now warranted. This conclusion is based on evaluations of the safety significance of the issue and the potential negative impact of performing

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The staff has reviewed the report, and finds that its results and the recommended inspections, coupled with field experience, provide a sufficient basis to conclude that loss of structural integrity and ejection of components with respect to pressurizers are highly unlikely.

In December 1991, after cracks were found in a control rod drive mechanism (CRDM) penetration in the reactor head at a French plant, an action plan was implemented to address PWSCC at all U.S. PWRs. The staff met with the Westinghouse Owners' Group (WOG) on January 7, 1992, the CEOG on March 25, 1992, and the Babcock & Wilcox Owners' Group on May 12, 1992, to discuss their respective programs for investigating PWSCC of Alloy 600 and to assess the possibility of cracking of CRDM penetrations in their respective plants. Subsequently, the staff asked the Nuclear Utility Management and Resources Council (NUMARC) to coordinate future industry actions because the issue was applicable to all PWRs. Meetings were held with NUMARC and PWR owners on the issue on August 18 and November 20, 1992, and March 3, 1993. Summaries of the meetings are available in the Commission's Public Document Room, 2120 L Street, N.W., Washington, D.C. 20555. In addition, the Electric Power Research Institute (EPRI) is engaging in ongoing research on methods for PWSCC mitigation. EPRI is also developing a demonstration program to ensure that inspections performed on CRDM penetrations will be highly reliable in detecting and measuring flaws because the methods and examiners will have been tested and qualified.

The WOG conducted a safety analysis on Alloy 600 CRDM penetration cracking to support continued operation of its plants. That safety analysis, WCAP-13565, was issued in December 1992 and revised in February 1993. The Westinghouse analysis concludes, and the staff agrees, that the available data do not now present a significant safety concern for Westinghouse plants. During the meeting on March 3, 1993, the WOG reported that one additional small flaw, 3-mm (0.120-in.) long and 2.25-mm (0.090 in.) deep, was found through metallographic examination of the penetration removed from the French plant. This indication was not axially oriented. It was oriented about 30 degrees from the horizontal. The WOG performed an analysis and evaluation of this flaw and concluded that the stresses in the region of the flaw are low; therefore, growth would not be expected. The staff agrees with the conclusions of this analysis. The staff requested that the WOG supplement its safety evaluation with an assessment of the rationale for the proposed inspection schedule and an assessment of the available or additional leak detection systems to ensure technical specifications requirements are met. The other PWR owners' groups are expected to submit separate safety evaluations for their plants in the near future. The staff will complete its

The staff will review your petition in accordance with 10 CFR § 2.206. I will issue a final decision with regard to your petition within a reasonable time. A copy of the notice that is being filed for publication with the Office of the Federal Register is enclosed for your information.

Sincerely,

Thomas E Murley

Thomas E. Murley, Director
Office of Nuclear Reactor Regulation

Enclosure:
As stated

cc: All PWR Licensees
W. Rasin, NUMARC
J. Taylor, EPRI
L.A. Walsh, WOG
J. Hutchinson, CEOG
J. Taylor, BWOG

UNITED STATES NUCLEAR REGULATORY COMMISSION

ALL PRESSURIZED WATER REACTORS

RECEIPT OF PETITION FOR DIRECTOR'S
DECISION UNDER 10 CFR § 2.206

Notice is hereby given that by letter dated March 24, 1993, to Ivan Selin, Chairman of the Nuclear Regulatory Commission (NRC), from John Willis on behalf of Greenpeace International (petitioner), the NRC has received a petition under 10 CFR 2.206 regarding all pressurized water reactors (PWRs) now operating in the United States. The petitioner requests immediate, full inspection of all vessel head penetrations (VHPs) in U.S. PWRs for cracking and publication of the results by the NRC. The letter is being treated as a petition for enforcement action, pursuant to 10 CFR 2.206, because the petitioner also requests that the NRC shut down affected reactors, whether the cracking is longitudinal or circumferential. The petitioner also requests that the NRC staff re-license reactors which must be closed due to cracking based on the assertion that the repair or mitigation program for such cracks may negatively affect the configuration and effectiveness of safety systems. The petitioner seeks relief based on allegations that (1) some VHPs in PWRs in France, Belgium, Switzerland, and Sweden are cracking; (2) testing in France revealed incipient circumferential cracking of some VHPs, which could lead to a through-wall break in the primary pressure boundary without fulfillment of the leak-before-break criterion; and (3)

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GREENPEACE

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Mr Ivan Selin
Chairman
Nuclear Regulatory Commission
2120 C St
Washington, DC, 20555

24 March 1993

Chairman Selin,

As you may be aware, a serious age-related degradation phenomenon has recently appeared in some pressurised water reactors in France and other European countries. Some vessel head penetrations in PWRs in France, Belgium, Switzerland, and Sweden are cracking. This is known mainly on the basis of in-service inspections, not all of which have been complete.

Further testing in France revealed incipient circumferential cracking of some VHPs, a finding that raised the stakes considerably. Suddenly the possibility of a through-wall break in the primary pressure boundary - without fulfillment of the leak-before-break criterion - presented itself. Greenpeace believes that, in some cases, this phenomenon could cause the ejection of the control rod drive mechanism, with resulting loss of control of the reactor.

In the enclosed report to Greenpeace, a group of engineers have summarised the current state of knowledge of the origin, evolution, and possible consequences of VHP cracking around the world. One of the most notable features of their report is the lack of data from the United States, the nation with the most PWRs in the world. The causes of VHP cracking cannot be conclusively judged yet, and therefore a prudent course of action is to adopt conservative assumptions about the vulnerability of US PWRs. Instead, it has been reported that although the NRC recognises the possibility of VHP cracking, it does not judge the phenomenon to be of major safety significance.

Given that cracking in the VHP is an emerging, but still unquantified, risk factor, we request that you and your Commissioners very quickly adopt the following as elements of a comprehensive approach to VHP cracking:

- immediate full inspection of ALL vessel head penetrations in US PWRs and publication of the results;

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- the shutdown of affected reactors, whether the cracking is longitudinal or circumferential.

Inspections in France have revealed that over-pressurisation tests are not sufficient to guarantee complete identification of VHPC. Destructive testing on the Bugey reactors has proved, so far, to be most conclusive method.

Finally, reactors which must be closed due to cracking cannot be restarted without re-licensing, as the repair or mitigation program aimed at VHPC may negatively affect the configuration and effectiveness of safety systems.

We trust that the enclosed report will be of assistance to you in your endeavours. We look forward to a positive reply at your earliest convenience.

Sincerely,

John Willis
Coordinator, Nuclear Campaign
Greenpeace International

include the use of remotely controlled or automatic equipment. Currently, the U.S. nuclear industry is developing such equipment for inspection and possible repairs. Accordingly, requiring immediate nondestructive examinations (e.g., eddy current) at all PWRs could result in significant, unnecessary worker radiation exposures.

Enclosed for your information are copies (Enclosure 2) of the NRC staff's safety evaluation for potential cracking of Alloy 600 VHPs and a memorandum to the Commissioners informing them of the status of the VHP issue.

I will issue a final decision with regard to your petition after the staff has reviewed the findings of the first three inspections at U.S. PWRs scheduled for the spring and fall of 1994. In the interim, the NRC staff will include you on distribution for all external correspondence associated with this issue and will keep you informed of any future meetings on this subject.

Sincerely,

Original signed by

Thomas E. Murley, Director
Office of Nuclear Reactor Regulation

Enclosures:
1. Listing of Documents
2. Commission memorandum
   w/Safety Evaluation attached

cc w/enclosures:

All PWR Licensees
NUMARC
EPRI
CEOG
WOG
B&WOG

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