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Pacific Gas and Electric Company

245 Market Street, Room 836-N9B
San Francisco, CA 94105
Mailing Address
Mail Code N9B
P.O. Box 770000
San Francisco, CA 94177
415/973-0600 Fax 415/973-6567

Lawrence F. Womack
Vice President
Nuclear Technical Services

July 28, 1997

PG&E Letter DCL-97-131

U.S. Nuclear Regulatory Commission
ATTN: Document Control Desk
Washington, D.C. 20555

Docket No. 50-275, OL-DPR-80
Docket No. 50-323, OL-DPR-82
Diablo Canyon Units 1 and 2
120-Day Response to Generic Letter 97-01, "Degradation of Control Rod Drive Mechanism Nozzle and Other Vessel Closure Head Penetrations"

Dear Commissioners and Staff:

Generic Letter (GL) 97-01, "Degradation of Control Rod Drive Mechanism Nozzle and Other Vessel Closure Head Penetrations," was issued to request licensees to describe their program for ensuring the timely inspection of PWR control rod drive mechanism and other closure head penetrations. PG&E's response to GL 97-01 is enclosed.

Sincerely,

Lawrence F. Womack

Subscribed and sworn to before me
this 28th day of July 1997
State of California
County of San Francisco

Notary Public

Attorneys for Pacific Gas
and Electric Company
Roger J. Peters
Richard F. Locke

Richard F. Locke

A075

D. GENA JEE
Commission #1067005
Notary Public — California
SAN FRANCISCO COUNTY
My Comm. Expires JUL 30, 1999

c: Steven D. Bloom
Ellis W. Merschoff
Kenneth E. Perkins
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CONTENTS

ENCLOSURE 1

RESPONSE TO GENERIC LETTER 97-01, "DEGRADATION OF CONTROL ROD DRIVE MECHANISM NOZZLE AND OTHER VESSEL CLOSURE HEAD PENETRATIONS"

ATTACHMENT 1

Diablo Canyon Information Provided in Response to Generic Letter 97-01

Generic Letter (GL) 97-01, "Degradation of Control Rod Drive Mechanism Nozzle and Other Vessel Closure Head Penetrations," was issued to request licensees to describe their program for ensuring the timely inspection of pressurized water reactor (PWR) control rod drive mechanism (CRDM) and other closure head penetrations. Information for the Diablo Canyon Power Plant (DCPP) Units 1 and 2 relative to information requested by GL 97-01 is provided below.

Requested Information

The information requested in Item 1 is needed by the NRC staff to verify compliance with 10 CFR 50.55a and 10 CFR Part 50, Appendix A, GDC 14, and to determine whether an augmented inspection program of the weld between the penetration nozzle and reactor vessel head as well as the portion of the nozzle above the weld is required, pursuant to 10 CFR 50.55a(g)(6)(ii), while the information requested in Item 2 relates to the occurrence of resin bead intrusion in PWRs, such as occurred at Zorita.

Within 120 days of the date of this generic letter, each addressee is requested to provide a written report that includes the following information for its facility:

1. *Regarding inspection activities:*

    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.*

    PG&E Response: Visual examinations of the CRDM head adapter tubes were first performed in 1988 during the Unit 1 second refueling outage (1R2). The examinations were initiated to address indications of canopy seal weld failure, which were subsequently repaired. Similar inspections were accomplished during each refueling outage in both units since 1R2. Canopy seal weld visual examinations are limited to the portion of the head adapter tubes exposed above the upper head insulation. Canopy seal weld leaks are detected by accumulations of boric acid crystals on the affected seal weld and adjacent head adapter tubes.

    Supplemental visual inspections intended specifically to address head adapter tube cracking issues were started in 1993 during the Unit 2 fifth refueling outage (2R5). These inspections include the interface of the reactor vessel head and the insulation adjacent to the peripheral row of head adapter tubes, and they are

performed after removal of the insulation on the lower portion of the head (below the CRDM cooling shroud support ring). Since the peripheral tubes have the highest residual stresses and are the most susceptible to cracking, it is expected that any significant leakage would be detected at this inspection location. This examination is performed on a refueling outage frequency in conjunction with the canopy seal weld exams in accordance with Procedure STP X-CRDM, "Reactor Vessel CRDM Inspection."

DCPP's Inservice Inspection Program Plan specifies surface examination for a portion of the dissimilar metal welds located in the peripheral CRDM housings. These exams are based on ASME Section XI requirements and are scheduled once during the ten-year inspection interval. The first inspection interval examinations were completed in 1R6 and 2R6.

No indication of through-wall head adapter tube leakage (other than canopy seal weld leaks) has been detected in the examinations. Since experience shows that small canopy seal weld leaks are detectable by visual examination and radiation monitoring activities, through-wall leakage in head adapter tubes would likely be detected with current monitoring practices. Additional details of DCPP monitoring practices are provided in the response to GL 97-01 Section 1.3 below.

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.

<u>PG&E Response</u>: DCPP is a participant in the Westinghouse Owners Group (WOG)/Nuclear Energy Institute (NEI) reactor pressure vessel (RPV) head penetration integrated inspection program. This integrated program includes volumetric inspection of head penetrations that have been performed (see Attachment 1, WCAP-14901, Section 1.3) and additional volumetric inspections that will be performed. Present plans call for two CE-design plants and two B&W-design plants to be inspected over the next three years. Additional Westinghouse-design plants are likely to be added to the list over the next few months. There is currently no commitment or firm schedule for performing reactor head penetration volumetric inspections at DCPP; however, this will be further reviewed as the WOG/NEI integrated inspection plan is finalized.

The WOG/NEI integrated inspection plan promotes ALARA by minimizing the cumulative dose associated with inspection activities, including that required for reactor head insulation removal, as well as penetration volumetric inspections. In addition, PG&E believes that the number of plants that have been or will be

inspected under the integrated program is sufficient to address the issue of RPV head penetration cracking.

The need and schedule for reinspection will be based on an evaluation of the inspection results from the integrated inspection program. Part of the WOG/NEI program calls for the plant(s) performing reinspections to keep the NRC staff informed of future reinspection plans.

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>PG&E Response</u>: DCPP Unit 1 has 79 reactor head nozzles and one head vent penetration. The reactor head nozzles are distinguished by the type of nozzle attachment and function. Of the 79 nozzles, 53 of the CRDM nozzles are full-length and contain thermal sleeves. Thirteen nozzles are spares, eight nozzles are for part-length CRDMs (abandoned in place), and five nozzles are for core exit thermocouple columns. The head vent penetration is connected to the reactor vessel level instrumentation system (RVLIS) and head vent, and is thus connected to both narrow- and wide-range level transmitters. One of the part-length nozzles has no drive mechanism and is used as a second vent that feeds the solenoid-operated head vent valves.

DCPP Unit 2 has 78 reactor head nozzles and one head vent penetration. Of the 78 nozzles, 53 are full-length CRDM nozzles with thermal sleeves. Twelve nozzles are spares, eight nozzles are for part-length CRDMs (abandoned in place), and five nozzles are core exit thermocouple columns. The head vent penetration is connected to the RVLIS, and is thus connected to both narrow- and wide-range level transmitters. One of the part-length nozzles has no drive mechanism and is used as a second vent that feeds the solenoid-operated head vent valves.

If PG&E were to perform a reactor head penetration volumetric inspection at Diablo Canyon, it is expected that, at a minimum, the number and distribution of penetrations inspected would follow the guidelines contained in WCAP-14024, "Inspection Plan Guidelines for Industry/Plant Inspection of Reactor Vessel Closure Head Penetration Tubes," for a 4-loop PWR. Specifically, the inspection logic requires inspection of the four outer peripheral rings of penetrations. This includes 26 penetrations (penetration numbers 54-79) at DCPP Unit 1, and 25 penetrations (penetration numbers 54-78) at DCPP Unit 2.

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

PG&E Response: Prior to issuance of GL 97-01, PG&E worked with the WOG, the Electric Power Research Institute (EPRI), and NEI to understand the operational experience and to identify the technical issues, causal factors, relative importance, and potential solutions for managing this issue. One of these tasks was development of a safety evaluation (WCAP-13565, Revision 1, "Alloy 600 Reactor Vessel Adapter Tube Cracking Safety Evaluation") that characterized damage initiation and propagation in reactor head penetration material and assessed the consequences. WCAP-13565, Revision 1, is applicable to DCPP. The NRC reviewed WCAP-13565, Revision 1, and issued a safety evaluation report (SER) on November 19, 1993, which is included herein as Enclosure 2. WCAP-13565, Revision 1, and the NRC SER establish the basis for continued operation of DCPP with respect to this issue.

While DCPP is a participant in the WOG/NEI RPV head penetration integrated inspection program, no augmented inspection is deemed necessary at this time. This conclusion is based on WCAP-13565, Revision 1, and on the NRC SER, which states, "The staff agrees there are no unreviewed safety questions associated with CRDM/CEDM [control element drive mechanism] penetration cracking." The basis for these conclusions is summarized below.

NUREG/CR-6245, "Assessment of Pressurized Water Reactor Control Rod Drive Mechanism Nozzle Cracking," identified two potential safety concerns associated with CRDM nozzle cracking. "First, a crack could eventually lead to a rupture of the nozzle and, if the nozzle is severed, to ejection of the connected CRDM housing. Second, a through-wall crack would allow the borated reactor coolant to come in contact with the vessel head and cause boric acid corrosion of the low-alloy steel base metal." These safety concerns were addressed in the PWR owners group's safety evaluations submitted to and accepted by the NRC staff.

WCAP-13565, Revision 1, contains head penetration stress and crack growth analyses, as well as an assessment of head penetration leakage and vessel head wastage. The WCAP analyses conclude that the stress distribution in the head penetrations results in predicted crack locations (in peripheral penetrations, at circumferential locations coplanar to the vessel centerline) and with orientation (axial) consistent with industry inspection results. Crack growth analyses concluded that growth would, in general, be very slow, would not be expected to change the crack orientation, and would not be expected to grow significantly above the penetration weld, such that leakage of any magnitude, and catastrophic failure of the head penetration, are both extremely unlikely. Since residual stresses diminish rapidly above the nozzle-to-vessel head weld, axial crack propagation would be limited prior to reaching the critical length required for penetration rupture. The critical length is over 20 inches total, 13 of which must extend above the vessel head. The assessment of head material wastage, in the unlikely event of a through-wall leak, concluded that ASME Code

allowable structural margins would be maintained for at least six additional years of plant operation at 100 percent power.

NUREG/CR-6245 is based in part on the results and conclusions of WCAP-13565, Revision 1 and reaches similar conclusions. The NUREG concluded that while PWR nozzles in U.S. reactors are susceptible to PWSCC:

> "CRDM nozzle cracking is not a short-term safety issue. An axial crack is not likely to grow above the vessel head to a critical length and cause a rupture of the nozzle because the stresses present in the nozzle wall are not high enough to support the growth away from the attachment weld. In addition, long before an axial weld reached the critical flaw size, the primary coolant would leak and be detected during periodic surveillance walkdowns for boric acid leakage."

It was also concluded that reactor coolant leaking from a through-wall axial crack could initiate a circumferential crack on the outside surface that could grow through-wall and around the nozzle circumference; however, this would take a long operating time (an analyses performed by Westinghouse concluded this would take over 90 years), well beyond the current license period.

The NUREG also addressed wastage of the reactor vessel head in the event of a leak:

> "Leakage of the primary coolant from a through-wall crack could cause boric acid corrosion of the vessel head. Some analyses results show that it would take at least six to nine years before corrosion challenges the structural integrity of the head, and it is very unlikely that the accumulated deposits of boric acid crystals resulting from such long-term leakage could remain undetected."

Based on the results demonstrated in WCAP-13565, Revision 1, and NUREG/CR-6245, a reactor head penetration leak, such as that seen at Bugey-3, would result in the accumulation and deposition of hundreds of pounds of boric acid crystals before vessel head integrity became a concern. Since the outermost penetrations are most susceptible, a penetration leak would likely result in boric acid accumulation near the head periphery, where it would be detected under the requirements of GL 88-05, "Boric Acid Corrosion of Carbon Steel Reactor Pressure Boundary Components in PWR Plants," as described in the response to Section 1.1. In addition, the boric acid walkdown measures in place at DCPP have been augmented with additional means of leak detection. Specifically, particulate and noble gas containment radiation monitors are used on a routine basis by the DCPP Chemistry Department to detect reactor coolant system (RCS) leakage. Use of these monitors to detect CRDM leakage was formalized in response to CRDM canopy seal leaks in 1988 and 1992. Based on

our experience with canopy seal leaks, PG&E installed three inspection access ports in the DCPP reactor vessel head shrouding, as well as air activity sample taps in the CRDM fan ducts. Grab sample collection via the CRDM fan duct sample taps is formalized in Procedure RCP D-420, "Sampling and Measurement of Airborne Radioactivity." Extensive smear surveys are performed and evaluated following duct removal early in each refueling outage. Radioisotope analysis of the smears and any resultant elevated activities would trigger management attention to assess the potential for a leak. These measures constitute the components of PG&E's Containment Monitoring Program to permit early detection of leaks in the reactor vessel CRDM area.

PG&E is currently evaluating the use of zinc addition to the RCS as a means of mitigating PWSCC of Alloy-600 material. Laboratory tests have shown that the presence of zinc in simulated reactor coolant inhibits the initiation and propagation of PWSCC in Alloy 600 steam generator tubing. The inhibiting effect of zinc on PWSCC is expected to benefit all Alloy 600 applications in the primary system, including the reactor vessel head penetrations. The addition of zinc is tentatively planned for DCPP Unit 1 in the latter part of 1998.

In summary, (1) there are no unreviewed safety questions associated with this issue, (2) a comprehensive monitoring program to address this issue is in place at DCPP, (3) an industry integrated inspection program is underway, and (4) DCPP is evaluating zinc injection, which should provide mitigation for this issue. PG&E will continue to evaluate this issue for relevant impact to DCPP.

1.4 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.

<u>PG&E Response</u>: The data, tests, and methods used in developing the crack initiation and crack growth models on which PG&E's management strategy for addressing the RPV head penetration cracking issue is based, are provided in Attachment 2, WCAP 14901 Sections 2 and 3. The susceptibility model and methodology were developed by Westinghouse for the WOG. Validation of the model included benchmarking against inspection results obtained from Almaraz-1, D.C. Cook-2, and Ringhals-2 as discussed in the WCAP.

DCPP is a participant in the WOG analysis program in which a plant-specific probability analysis using the methodology described in WCAP-14901, Section 4, has been performed. The DCPP plant-specific input parameters to the

analysis are provided in Attachment 3. The analysis results are contained in WCAP-14919, "Probabilistic Evaluation of Reactor Vessel Closure Head Penetration Integrity for the Diablo Canyon Units 1 and 2," and will be incorporated into the WOG/NEI integrated inspection program, for use in the program's development and for consideration of plant-specific inspection recommendations. PG&E supports and is a participant in the integrated inspection program, which includes all three PWR owners groups, EPRI, and NEI. A description of the WOG/NEI program was provided in the response to Section 1.2.

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 durations 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 followup actions.*

    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.*

PG&E Response: PG&E has reviewed plant historical records to determine if any incident of resin ingress similar to those that occurred in 1980 and 1981 at the Jose Cabrera (Zorita) plant has occurred at Diablo Canyon. The period evaluated was from initial criticality, (April 29, 1984 for Unit 1 and August 20, 1985 for Unit 2) through May 31, 1997.

NEI and the WOG, through discussions with the NRC, defined a "significant" resin intrusion event to be an intrusion into the primary coolant system with a threshold of 1 cubic foot. One cubic foot was chosen as a conservative lower bound since it represents less than 15 percent of the estimated volume of resin released into the reactor coolant system during the events at Jose Cabrera. The

data search for Diablo Canyon was structured to identify all resin intrusion events into the primary coolant system with a magnitude greater than 1 cubic foot (30 liters).

For Unit 1, the period between April 29, 1984 and July 1, 1984 reflects plant operation prior to the routine analysis for sulfate in the reactor coolant and results in a data search based on a review of chemistry records relative to specific conductance of the reactor coolant. A 28 micro S/cm incremental increase in specific conductance was the criterion used for indication of cation resin ingress, equivalent to a volume of 1 cubic foot. Beginning July 1, 1984 at Unit 1, and since initial criticality on August 20, 1985 at Unit 2, routine analysis for sulfate has been performed at DCPP. Records for these analyses were reviewed using a sulfate concentration in the range of 15 to 17 ppm as the indicator of cation resin ingress. This concentration is equivalent to a volume of 1 cubic foot.

The data review showed: (1) there have been no incremental specific conductivity excursions greater that 28 micro S/cm with a corresponding increase in sulfate concentration, and (2) though the concentration of sulfate has exceeded Diablo Canyon's administrative limit of 50 ppb at times, RCS sulfate concentrations have never exceeded 0.15 ppm (e.g., following forced oxygenation of the RCS during refueling outages). Therefore, PG&E concludes that there has been no single resin intrusion greater than 1 cubic foot during the time from initial criticality through May 31, 1997. Furthermore, the chemical and volume control system at DCPP is designed with filters downstream of the demineralizers. These filters (initially 25 micron and currently 0.2 micron) would trap resin that escaped from a demineralizer preventing a resin intrusion into the RCS. These filters are changed out upon an indication of high differential pressure.

Had either specific conductance or sulfate increases indicated resin ingress of 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 regarding the incident. In the case of the use of sulfate data as the indicator, specific conductance would also have been included as a confirmatory datum 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 a mixed bed resin (cation plus anion); however, if mixed bed resin leakage to the reactor coolant occurred, the cation portion of the resin would contain the sulfate indicator described above. Detectable dissolved oxygen in the reactor coolant

during power operation with the 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.

DCPP has followed the EPRI PWR Primary Water Chemistry Guidelines since April 1987. The specific needs of DCPP were considered and, therefore, some limits in the DCPP procedures do not match the guidelines.

The following exceptions to the EPRI guidelines exist at DCPP that are potentially related to GL 97-01; however, these are not expected to impact the probability of intergranular attack.

- DCPP does not have formal "Action Levels" as described in the EPRI guidelines. DCPP implements Action Guidelines and Equipment Control Guideline Action Levels.

- The operating limits for dissolved hydrogen are 25-40 cc/kg, while the EPRI Guidelines, Rev. 3, now recommend 25-50 cc/kg. Previously, EPRI recommended the lower limit for plants susceptible to PWSCC. The more relaxed guideline is based on recognition that hydrogen has a minimal impact on PWSCC throughout this broader range.

ATTACHMENT 2

WCAP-14901, Revision 0, "Background and Methodology for Evaluation of Reactor Vessel Closure Head Penetration Integrity for the Westinghouse Owners Group"

This enclosure describes the methodology used in evaluating the susceptibility of PWSCC in the Diablo Canyon Power Plant reactor vessel head penetrations, and which supports a plan for managing this issue at DCPP.

Diablo Canyon RPV Head Penetration Cracking
Predicted Susceptibility Input Parameters

The susceptibility of the Diablo Canyon reactor vessel head penetrations to PWSCC was evaluated using the methodology provided in Attachment 2. The susceptibility analysis is contained in Reference 1. Input parameters to the susceptibility evaluation are included in Tables 1 and 2 for Units 1 and 2, respectively.

The first column in Tables 1 and 2 contains the reactor head "ring number," which identifies a specific ring in a concentric set on the closure head. Nozzles located in the same ring number are at an equal distance from the head dome center and have the same penetration angle. The second column lists the individual penetration numbers (Reference 2 and 3 for Units 1 and 2, respectively). The third column lists the number of nozzles that have the same geometry, material properties, and stress-state; this provides nozzle subgroups and facilitates the analysis. The fourth column provides the penetration setup angle (Reference 4), which is the angle between the nozzle vertical axis and a vector normal to the closure head at the penetration location. The penetration angle determines the amount of nozzle ovality and weld residual stresses, and the angle also affects the crack initiation and growth prediction. The fifth column contains the material heat number for each penetration (References 5 and 6 for Units 1 and 2, respectively). The heat number defines the penetration composition, heat treatment, and yield stress. The yield stress is listed in column 6. This and other specific material properties for each material heat are contained in Reference 7. The grain boundary carbide coverage was determined from a correlation developed for Huntington Alloy tubing (Reference 8) based on carbon, nickel, and manganese contents, ultimate tensile strength, and yield strength. The carbide coverage affects both the crack initiation time and crack growth rate in the susceptibility model; calculated values are shown in column 7 (Reference 1). Penetration temperatures input to the analysis were 589 and 593 degrees Fahrenheit for Units 1 and 2, respectively (Reference 1). The model was based on operating with an overall 85 percent capacity factor. The reactor vessel head CRDM nozzle arrangement for DCPP Units 1 and 2 is shown in Figures 1 and 2, respectively.

The susceptibility model provided in WCAP-14901, and for which these input parameters are provided, has no bearing on either the safety evaluation performed for the Westinghouse Owners Group or the NRC SER on this issue.

REFERENCES

1. WCAP-14919 (Westinghouse Proprietary), "Probabilistic Evaluation of Reactor Vessel Closure Head Penetration Integrity for the Diablo Canyon Units 1 and 2," July 1997

2. Combustion Engineering Drawing E-232-448, "Closure Head Assembly," Contract #23066, August 1967

3. Combustion Engineering Drawing E-234-158, "Closure Head Assembly 173" I.D. P.W.R.," Contract #6268, April 1969

4. WCAP-14024, "Inspection Plan Guidelines for Industry/Plant Inspection of Reactor Vessel Closure Head Penetration Tubes," June 1994

5. Combustion Engineering Drawing E-232-465, "Material Identification Closure Head," Contract #23066, August 1967

6. Combustion Engineering Drawing E-234-171, "Material Identification Closure Head 173" I.D. P.W.R.," Contract #6268, April 1969

7. WCAP-13493, "Reactor Vessel Closure Head Penetration Key Parameter Comparison," September 1992

8. WCAP-13876, Revision 1, "Microstructural Correlations with Material Certification Data in Several Commercial Heats of Alloy 600 Reactor Vessel Head Penetration Materials," June 1997

Table 1
DCPP - 1 Reactor Head Penetration Input Parameters

Head Ring No.	Penetration No.	# Penetrations	Penetration Angle	Heat No.	Yield Stress	GBC %
14	74, 78, 79	3	48.6	NX-7760	38	36.2
	75, 77	2	48.6	NX-5843	44	78.7
	76	1	48.6	NX-5983	36	87.2
13	68, 73	2	45.3	NX-5843	44	78.7
	70-72	3	45.3	NX-7547	40.5	38.2
	69	1	45.3	NX-5983	36	87.2
	66, 67	2	45.3	NX-7760	38	36.2
12	62, 64, 65	3	44.1	NX-7760	38	36.2
	63	1	44.1	NX-5843	44	78.7
11	58-61	4	38.6	NX-7547	40.5	38.2
	56, 57	2	38.6	NX-5843	44	78.7
	55	1	38.6	NX-7280	40.5	42.1
	54	1	38.6	NX-7760	38	36.2
10	50-53	4	36.2	NX-7547	40.5	38.2
9	49	1	35.1	NX-5843	44	78.7
	45-48	4	35.1	NX-7547	40.5	38.2
	42-44	3	35.1	NX-7280	40.5	42.1
8	40,41	2	33.8	NX-7280	40.5	42.1
	38,39	2	33.8	NX-5843	44	78.7
7	30-37	8	30.2	NX-7760	38	36.2
6	26-29	4	26.1	NX-7760	38	36.2
	22-25	4	26.1	NX-7547	40.5	38.2
5	18-21	4	24.7	NX-7547	40.5	38.2
4	14,16,17	3	23.1	NX-7547	40.5	38.2
	15	1	23.1	NX-5836	45	75.2
3	11-13	3	18.1	NX-7760	38	36.2
	10	1	18.1	NX-5836	45	75.2
2	6-9	4	16.2	NX-7547	40.5	38.2
1	2-5	4	11.1	NX-7547	40.5	38.2
0	1	1	0	NX-5965	35	75.2

Table 2
DCPP 2 - Reactor Head Penetration Input Parameters

Head Ring No.	Penetration No.	# Pens	Penetration Angle	Heat No.	Yield Stress	GBC %
14	76-78	3	48.6	NX-1314	52	64.1
	74-75	2	48.6	NX-0688	57.5	55.8
13	72-73	2	45.3	NX-0688	57.5	55.8
	66-71	6	45.3	NX-1314	52	64.1
12	64-65	2	44.1	NX-8251	35	48.2
	63	1	44.1	NX-1314	52	64.1
	62	1	44.1	NX-0688	57.5	55.8
11	61	1	38.6	NX-8699	40	37.3
	57-60	4	38.6	NX-1314	52	64.1
	54-56	3	38.6	NX-0688	57.5	55.8
10	50, 53	2	36.2	NX-8699	40	37.3
	52	1	36.2	NX-0688	57.5	55.8
	51	1	36.2	NX-8623	35	81.3
9	47-49	3	35.1	NX-8699	40	37.3
	45,46	2	35.1	NX-0688	57.5	55.8
	42-44	3	35.1	NX-1314	52	64.1
8	38-41	4	33.8	NX-1314	52	64.1
7	37	1	30.2	NX-1314	52	64.1
	36,30-32	4	30.2	NX-0688	57.5	55.8
	33-35	3	30.2	NX-8251	35	48.2
6	28,29	2	26.1	NX-0688	57.5	55.8
	26,27	2	26.1	NX-8699	40	37.3
	22-25	4	26.1	NX-1314	52	64.1
5	18-21	4	24.7	NX-1408	42.5	46.5
4	14-17	4	23.1	NX-0688	57.5	55.8
3	11-13	3	18.1	NX-1314	52	64.1
	10	1	18.1	NX-0688	57.5	55.8
2	6-9	4	16.2	NX-0688	57.5	55.8
1	3-5	3	11.3	NX-0688	57.5	55.8
	2	1	11.3	NX-8623	35	81.3
0	1	1	0	NX-0688	57.5	55.8

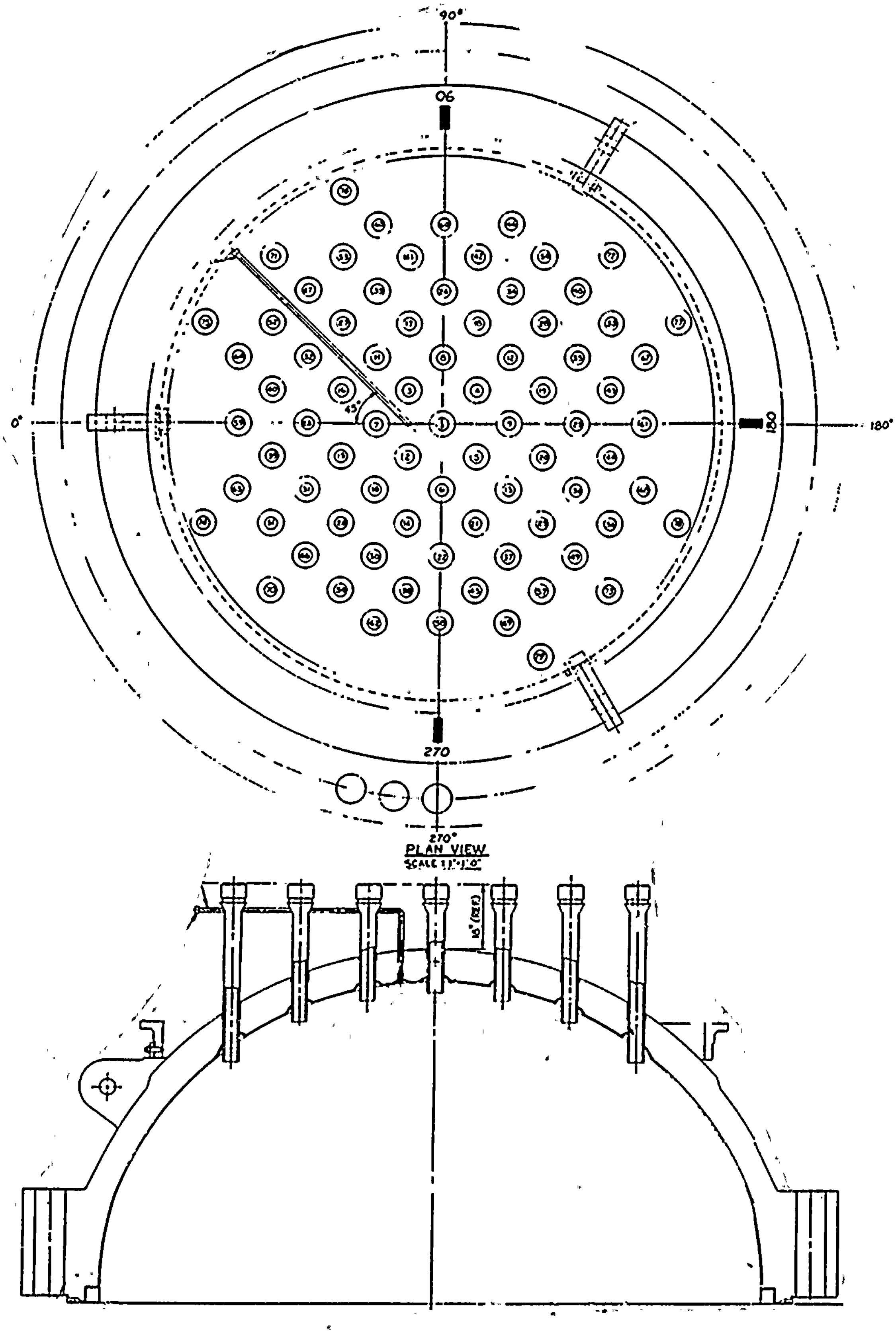

FIGURE 1. DCPP Unit 1 Reactor Vessel Closure Head Penetration Arrangement

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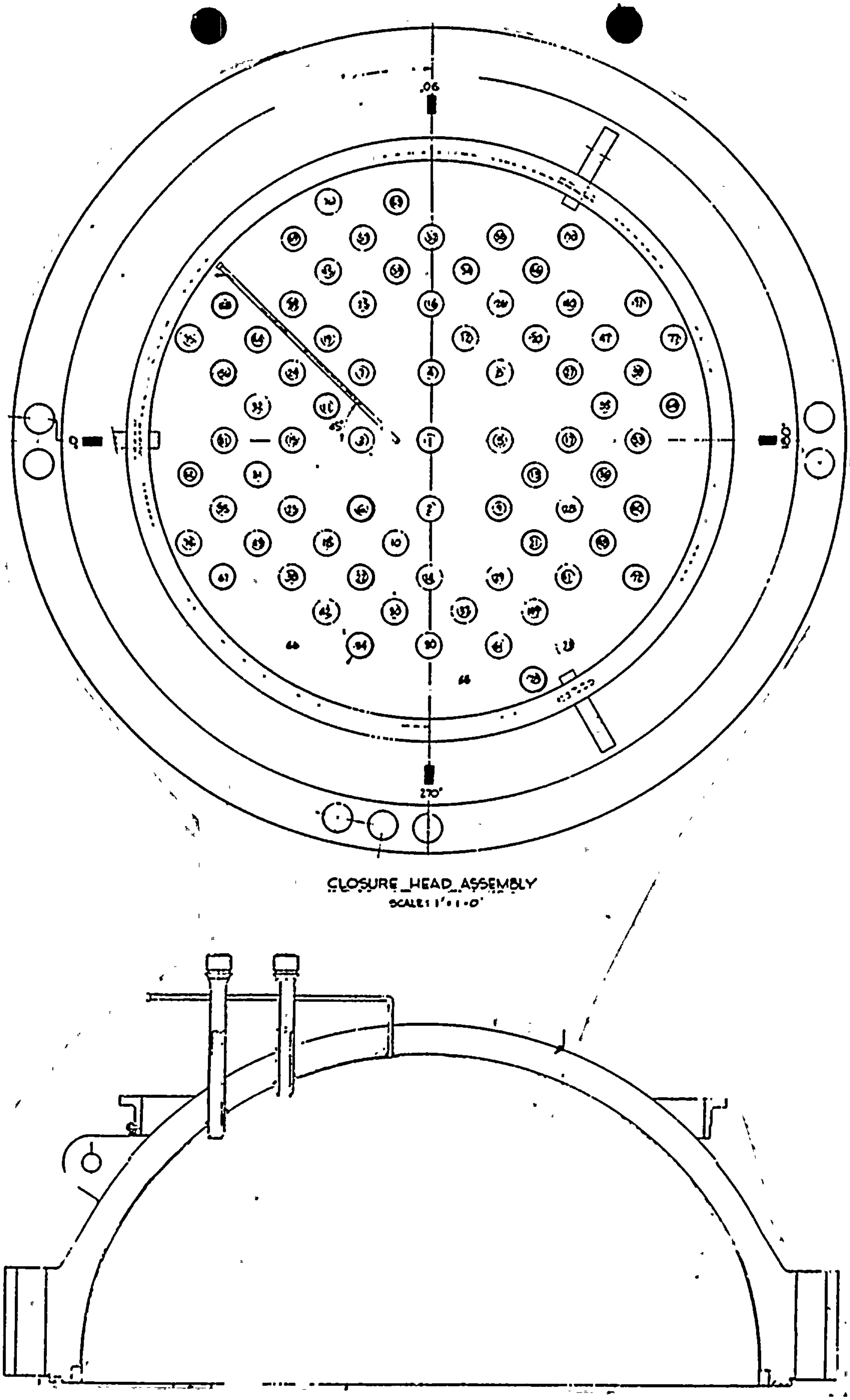

FIGURE 2. DCPP Unit 2 Reactor Vessel Closure Head Penetration Arrangement

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ENCLOSURE 2

NRC SER FOR POTENTIAL REACTOR VESSEL HEAD ADAPTOR TUBE CRACKING

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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

NUS

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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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 Allov 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