DRAFT

October 3, 1996

Mr. David L. Meyer
Chief, Rules Review and Directives Branch
U.S. Nuclear Regulatory Commission
Washington, DC 20555-0001

Dear Mr. Meyer:

Enclosed are Nuclear Energy Institute (NEI)[1] comments on the "Proposed Generic Communication; Primary Water Stress Corrosion Cracking of Control Rod Drive Mechanism and Other Vessel Head Penetrations," (61 Fed. Reg. 40253, August 1, 1996). These comments were developed by an NEI task force comprised of representatives from utilities, PWR Owners Groups and EPRI. Additionally, these comments were forwarded to the industry for consideration by individual utility licensees in developing plant-specific comments.

NEI will continue to coordinate industry activities in managing primary water stress corrosion cracking (PWSCC) of vessel head penetrations. This coordination will involve EPRI and the PWR Owners Groups to ensure that necessary information is evaluated and communicated to utilities to support their decisions to conduct inspections. NEI continues to believe that the decision to conduct inspections rests with individual utility management after due consideration of susceptibility, evidence of boric acid deposition and economic risk. As in the past, NEI will continue to meet with NRC staff to discuss inspection results as they relate to the Owners Group safety evaluations and inspection criteria, and the NRC's safety evaluation report. NEI believes this approach in managing this issue is appropriate and sufficient given the low safety concern. Therefore, NEI concludes that there is no technical or regulatory basis for this generic letter.

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[1] NEI is the organization responsible for establishing unified nuclear industry policy on matters affecting the nuclear energy industry, including the regulatory aspects of generic operational and technical issues. NEI's members include all utilities licensed to operate commercial nuclear power plants in the United States, nuclear plant designers, major architect/engineering firms, fuel fabrication facilities, materials licensees, and other organizations and individuals involved in the nuclear energy industry.

Comments relating to the general thrust of the draft generic letter are provided in Enclosure 1 and are summarized as follows:

- The draft generic letter essentially requests licensees to define and commit to an augmented inspection program. The stated purpose of the draft generic letter is to determine if augmented inspections are warranted. If augmented inspections are determined by NRC to be necessary, then such inspections should be based on the safety significance of the vessel head penetrations experiencing primary water stress corrosion cracking, not whether or not licensees are currently performing inspections.

- The NRC staff safety concerns have been addressed by the PWR Owners Groups' safety evaluations, which considered the possibility of through-wall cracks.

- The stated scope of the draft generic letter is primary water stress corrosion cracking. The resin intrusion at the Zorita Plant resulted in intergranular stress corrosion which is a different degradation mechanism. Since the Zorita resin intrusion was communicated to utility licensees by Information Notice 96-11, and new concerns have not been identified, it is not clear why the NRC staff is now requesting licensees to submit information on this topic.

Enclosure 2 provides detailed comments on the specific text of the proposed generic letter.

If you have questions concerning these comments, please contact Alex Marion (202-739-8080) or me.

Sincerely,

Ralph E. Beedle

TET/AM/ead
Enclosures

c: C. E. (Gene) Carpenter, NRC/NRR
Brian Sheron, NRC/NRR
Jack Strosnider, NRC/NRR

GENERAL COMMENTS ON THE DRAFT GENERIC LETTER

1. Items 1 and 2. in the Information Requested section, essentially requests licensees to define and commit to an augmented inspection program. The stated purpose of the draft regulatory guide is to evaluate whether or not an augmented inspection program is necessary. The justification for the augmented inspection should be based on the safety significance of the vessel head penetration's (VHP) experiencing primary water stress corrosion cracking, not if licensees are currently performing augmented inspections.

2. On Page 10 of NUREG/CR-6245, Assessment of Pressurized Water Reactor Control Rod Drive Mechanism Nozzle Cracking, it states, "There are two major 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." In the NRC staff's safety evaluation dated November 19, 1993, it states, "The primary safety concern associated with stress corrosion cracking in Alloy 600 is the potential for circumferential cracks. Extensive circumferential cracking could lead to ejection of a CRDM." These safety concerns were considered by the PWR Owners Group safety evaluations submitted to and accepted by the NRC staff. The draft generic letter has not identified any safety concerns that were not previously evaluated and dispositioned. Summaries of these safety evaluations are contained in NUREG/CR-6245 and the NEI's white paper titled, "Alloy 600 RPV Head Penetration Primary Water Stress Corrosion Cracking."

3. The second paragraph of the Discussion section states that the goal of the draft generic letter is to ". . .verify that the margins required by the ASME Code as specified in § 50.55a of Title 10 of the Code of Federal Regulations (10 CFR 50.55a) are met, that the guidance of General Design Criterion 14 of Appendix A to 10 CFR Part 50 (10 CFR Par 50, Appendix A, GDC 14) is continued to be satisfied, ..." These goals are unique and separate from the stated purpose of the first paragraph in the Required Information section which states; "The information requested in Items 1 and 2, below, is required to determine if the imposition of an augmented inspection program is required, ..." Although not stated as such, the Discussions section appears to raise a question of compliance rather than determining if new regulatory requirements (augmented inspections per 10 CFR 50.55a(g)(6)(ii)) should be imposed. Utility licensees are presently in compliance with the requirements identified in the Discussion section based on the following:

- The design and fabrication of the reactor vessel heads satisfy all applicable ASME requirements.
- Only the welds that attach VHPs to the reactor head are within the scope of the inservice inspection requirements (ASME Section XI, Table IWB-2500-1, Examination Category B-E). As noted in NUREG/CR-6245, the VHP surface which could experience PWSCC is not expected to be within the scope of ASME inservice inspections. However, should inservice inspection identify indications, licensees will disposition them per the ASME Code.
- GDC-14 states, "The reactor coolant pressure boundary shall be designed, fabricated, erected, and tested so as to have an extremely low probability of abnormal leakage, of rapidly propagating failure, and of gross rupture." Licensees are meeting GDC-14 because:
  - The reactor vessel head was designed, fabricated, and erected to the ASME Code or other requirements approved by the NRC.
  - The PWR Owners Group safety evaluations, accepted by the NRC staff (dated November 19, 1993), addressed the potential for rapid crack propagation, gross rupture and abnormal leakage. These evaluations determined that PWSCC would either be arrested or would grow very slowly requiring years to obtain a critical length. Axial cracks require many years to obtain critical length. Circumferential cracking requires through-wall leakage and will take significantly more time than the 40-year licensed operating period. One conservative circumferential cracking evaluation estimated that it would take in excess of 90 years before gross failure would occur.
  - Licensees are presently performing GL 88-05 inspections to detect leakage that could occur during operation. If leakage is detected, repairs and corrective action will be performed. In addition, corrective action is required if leakage exceeds the Technical Specification criteria.
  - This approach to GDC compliance is consistent with the leak-before-break criteria applied to other primary piping systems.

4. The Requested Information section asks licensees to summarize the inspections they have performed, define the inspections they plan to perform or justify why inspections are not being performed. The NRC staff witnessed the VHP inspections performed by licensees (five plants) and has received written reports on the results. Hence, this is a redundant request for those licensees who have already performed inspections and requests submittal of information that the NRC already has in its possession. In addition, the NEI white paper discussed the method by which licensees are managing this issue, i.e., future inspections will be performed based on information sharing, predictive methodologies and tools, inspection results, and development of mitigation and repair technologies.

5. The topic of the draft generic letter is "Primary Water Stress Corrosion Cracking of Control Rod Drive Mechanism and Other Vessel Head Penetrations." The inclusion of a different form of degradation (intergranular stress corrosion cracking due to resin intrusion) is not warranted. PWSCC of Alloy 600 is an time dependent degradation mechanism. Intergranular stress corrosion cracking due to resin intrusion is an abnormal operating event. Furthermore, of the over 5200 penetrations inspected worldwide, no evidence has been observed that suggests resin induced intergranular stress corrosion cracking has occurred in any reactor vessel other than Zorita. This is strong evidence that resin induced intergranular stress corrosion cracking is an outlier event that is not generic.

6. The NRC staff issued Information Notice (IN) 96-11, "Ingress of Demineralizer Resins Increases Potential for Stress Corrosion Cracking of Control Rod Drive Mechanism Penetrations," that advised licensees of the Zorita resin intrusion and potential intergranular stress corrosion cracking. It is unclear why the NRC has revised their position concerning a request for submitted information (Required Information, Item 3), since no additional resin intrusions concerns have occurred since IN 96-11 was issued. The extra burden on licensees to respond to Item 3 of the Required Information section is justified.

#	Comment	Resolution
[1]	Requested extension of comment period to October 3, 1996.	Extended Comment Period to October 3, 1996.
[2]	It should be noted in the background information that no CEDM nozzles in any plant worldwide have failed during plant operations. Evidence of cracking has been revealed during planned inspections. As alluded to in the proposal, any through-wall cracking would be slow, result in detectable leakage, and provide an opportunity to take corrective action, because the leak rates of primary systems are tracked during the operation of all nuclear plants. The proposed response period of 90 days may be insufficient given the recognized potential data collection difficulties and the fact that the owners groups may still be completing or updating their susceptibility assessments. We suggest that the NRC Staff coordinate the issuance and response timing of the generic Letter with the owners groups.	Draft GL will be revised to reflect comment Response period will be extended to 120 days.
[3]	Members of the ASME Section XI subcommittee are concerned that additional inspections are being imposed on the industry for what appears to be a non-safety issue. It is our concern that these additional inspection requirements are being justified based on the lack of action by Section XI on this subject.	The NRC Staff neither stated nor implied that the lack of action of the ASME Section XI subcommittee on this issue was a part of the decision reached to issue the generic letter. The Staff's concern is that this issue is a long term safety concern that the industry needs to address.
[4]	The Nuclear Energy Institute (NEI) is providing comments on the proposed generic letter (GL) on behalf of the industry. FPL endorses the NEI comments.	See responses to NEI comments.

[1] By letter dated August 6, 1996, comments from Ralph Beedle, Senior Vice President; Nuclear Energy Institute; 1776 I Street; Washington, DC

[2] By letter dated September 26, 1996, comments from T. L. Patterson, Division Manager; Omaha Public Power District; 444 South 16th Street Mall; Omaha, NR 68102-2247

[3] By letter dated September 30, 1996, comments from George Fecter, Secretary; ASME BPVC Subcommittee on Nuclear Inservice Inspection; 345 East 47th Street; New York, NY 10017

[4] By letter dated October 2, 1996, comments from W. H. Bohlke, Vice President; Florida Power & Light Co.; P.O. Box 14000; Juno Beach, FL 33408

#	Comment	Resolution
	Additionally, the Nuclear Utility Backfitting and Reform Group (NUBARG) is providing comments on the proposed GL. FPL endorses the NUBARG comments.	See Response to NUBARG comments.
[5]	We believe the Staff is required to perform a backfit analysis on the proposed imposition of the development of a plan to inspect the CRDM and other vessel head penetrations. ... Imposition of this new inspection requirement goes beyond simply asking licensees to provide an information response. Instead, the new requirement to develop an inspection program is a modification or addition to the operating procedures resulting from the imposition of a new regulatory Staff position. ... Specifically, by performing a backfitting analysis, the Staff can ensure that the requested periodic inspections and monitoring activities are effective from a safety perspective and are cost beneficial. Moreover, by performing the analysis, the Staff can ensure that unnecessary downtime and adverse schedule impacts are avoided by licensees and that any resulting radiation exposure is assessed and minimized.	The Staff has reviewed the requirements, and has determined that requesting information to determine if additional inspection requirements are necessary is not a backfit. Therefore, no backfit analysis needs be performed. If the Staff determines, after reviewing the requested information, that imposing additional inspection requirements is necessary, a backfit analysis will be performed.
[6]	Items 1 and 2 in the Required Information section essentially requests licensees to define and commit to an augmented inspection program. The stated purpose of the draft regulatory guide is to evaluate whether or not an augmented inspection program is necessary. The justification for the augmented inspection should be based on the safety significance of the vessel head penetration's (VHP) experiencing primary water stress corrosion cracking, not if licensees are currently performing augmented inspections.	The proposed GL is requesting information from the licensees. The GL does <u>not</u> require a commitment to an integrated, long-term inspection program, but rather asks what, if any, periodic inspections licensees are performing, and the bases for concluding acceptability of these plans to (not) perform inspections.

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[5] By letter dated October 3, 1996, comments from Kathryn M. Sutton; Winston & Strawn; 1400 L Street, NW; Washington, DC 20005

[6] By letter dated September 30, 1996, comments from Ralph Beedle, Senior Vice President; Nuclear Energy Institute; 1776 I Street; Washington, DC

#	Comment	Resolution
	On Page 10 of NUREG/CR-6245, Assessment of Pressurized Water Reactor Control Rod Drive Mechanism Nozzle Cracking, it states, "There are two major 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-ahoy steel base metal." In the NRC Staff's safety evaluation dated November 19, 1993, it states, "The primary safety concern associated with stress corrosion cracking in Alloy 600 is the potential for circumferential cracks. Extensive circumferential cracking could lead to ejection of a CRDM." These safety concerns were considered by the PWR Owners Group safety evaluations submitted to and accepted by the NRC Staff. The draft generic letter has not identified any safety concerns that were not previously evaluated and dispositioned. Summaries of these safety evaluations are contained in NUREG/CR-6245 and the NEI's white paper titled, "Alloy 600 RPV Head Penetration Primary Water Stress Corrosion Cracking."	Since (1) the industry has not provided information to the Staff regarding possible primary water contamination that could increase the potential for circumferential cracking, and (2) the intent of the letter is to collect information to understand licensee's plans for inspection and monitoring to assure that the assumptions in their analytic safety evaluations are maintained over the long term, the Staff has determined that the requested information is necessary to determine if the imposition of an augmented inspection program, pursuant to 10 CFR 50.55a(g)(6)(ii), is required.

#	Comment	Resolution
	The second paragraph of the Discussion section states that the goal of the draft generic letter is to "... verify that the margins required by the ASME Code as specified in §50.55a of Title 10 of the *Code of Federal Regulations* (10 CFR 50.55a) are met, that the guidance of General Design Criterion 14 of Appendix A to 10 CFR Part 50 (10 CFR Par 50, Appendix A, GDC 14), is continued to be satisfied, ..." These goals are unique and separate from the stated purpose of the first paragraph in the Required Information section which states; "The information requested in Items I and 2, below, is required to determine if the imposition of an augmented inspection program is required....... Although not stated as such, the Discussions section appears to raise a question of compliance rather than determining if new regulatory requirements (augmented inspections per 10 CFR 50.55a(g)(6)(11)) should be imposed. Utility licensees are presently in compliance with the requirements identified in the Discussion section based on the following:	The commentor misunderstood the second paragraph of the Discussion section, which states, in part: "The NRC Staff considers cracking of VHPs to be a safety concern for the long term .... Therefore, in order to verify that the margins required by the ASME Code, as specified in [10 CFR 50.55a] are met..."
	The design and fabrication of the reactor vessel heads satisfy all applicable ASME requirements.	The quality of the design and construction is not at issue. The construction code does not address the operating environment nor the fact that IGSCC is an age-related degradation mechanism; it only contains a general corrosion statement.
	Only the welds that attach VHPs to the reactor head are within the scope of the inservice inspection requirements (ASME Section XI, Table I@-2500-1, Examination Category B-E). As noted in NUREG/CR-6245, the VHP surface which could experience PWSCC is not expected to be within the scope of ASME inservice inspections. However, should inservice inspection identify indications, licensees will disposition them per the ASME Code.	The Regulations require that ASME Section XI be met for the life of the plant. If no specific requirements are included in Section XI for inspection and evaluation, an augmented inspection program, pursuant to 10 CFR 50.55a(g)(6)(ii), may be required. The proposed GL is requesting information from the licensees to provide adequate assurance that margins and defense-in-depth are being maintained for the long term.
	GDC-14 states, "The reactor coolant pressure boundary shall be designed, fabricated, erected, and tested so as to have an extremely low probability of abnormal leakage, of rapidly propagating failure, and of gross rupture." Licensees are meeting GDC-14 because:	

#	Comment	Resolution
	The reactor vessel head was designed, fabricated, and erected to the ASME Code or other requirements approved by the NRC.	
	The PWR Owners Group safety evaluations, accepted by the NRC Staff (dated November 19, 1993), addressed the potential for rapid crack propagation, gross rupture and abnormal leakage. These evaluations determined that PWSCC would either be arrested or would grow very slowly requiring years to obtain a critical length. Axial cracks require many years to obtain critical length. Circumferential cracking requires through-wall leakage and will take significantly more time than the 40-year licensed operating period. One conservative circumferential cracking evaluation estimated that it would take in excess of 90 years before gross failure would occur.	
	Licensees are presently performing inspections in accordance with NRC Generic Letter 88-05 to detect leakage that could occur during operation. If leakage is detected, repairs and corrective action will be performed. In addition, corrective action is required if leakage exceeds the Technical Specification criteria.	As was reported in IN 86-108, Supplement 1, rapid and severe corrosion caused by boric acid leakage can occur before it is noticed. While the GL 88-05 program will find gross leakage some period of time after it began, the concern of this GL is whether licensees are able to maintain required margins and defense in depth, as discussed above.
	This approach to GDC compliance is consistent with the leak-before-break criteria applied to other primary piping systems.	This concern is directed towards the reactor vessel, not primary piping. Leak before break in the context of the Regulations is only applied to piping that has <u>no</u> active degradation mechanisms. Therefore, the approach is not applicable.

#	Comment	Resolution
	The Required Information section asks licensees to summarize the inspections they have performed, define the inspections they plan to perform or justify why inspections are not being performed. The NRC Staff witnessed the VHP inspections performed by licensees (five plants) and has received written reports on the results. Hence, this is a redundant request for those licensees who have already performed inspections and requests submittal of information that the NRC already has in its possession. In addition, the NEI white paper discussed the method by which licensees are managing this issue, i.e., future inspections will be performed based on information sharing, predictive methodologies and tools, inspection results, and development of mitigation and repair technologies.	Those utilities who have previously submitted the requested information will not need to resubmit; it will be sufficient that they reference their previous submittals in their response letter.
	The topic of the draft generic letter is "Primary Water Stress Corrosion Cracking of Control Rod Drive Mechanism and Other Vessel Head Penetrations." The inclusion of a different form of degradation (intergranular stress corrosion cracking due to resin intrusion) is not warranted. PWSCC of Alloy 600 is an time dependent degradation mechanism. Intergranular stress corrosion cracking due to resin intrusion is an abnormal operating event. Furthermore, of the over 5200 penetrations inspected worldwide, no evidence has been observed that suggests resin induced intergranular stress corrosion cracking has occurred in any reactor vessel other than Zorita. This is strong evidence that resin induced intergranular stress corrosion cracking is an outlier event that is not generic.	The GL addresses the potential degradation of CRDMs and other VHPs. There are various mechanisms (e.g., IGSCC, PWSCC, etc.) by which this degradation does occur, all of which have been seen at least once. The Staff's position has been that there is an increased likelihood of stress corrosion cracking of PWR VHP if demineralizer resins contaminate the RCS, and that licensees will and consider actions, as appropriate, to avoid similar problems. Therefore, to clarify the intent of the GL, the title will be changed to reflect more clearly the degradation concerns of the GL.
	The NRC Staff issued Information Notice (IN) 96-11, "Ingress of Demineralizer Resins Increases Potential for Stress Corrosion Cracking of Control Rod Drive Mechanism Penetrations," that advised licensees of the Zorita resin intrusion and potential intergranular stress corrosion cracking. It is unclear why the NRC has revised their position concerning a request for submitted information (Required Information, Item 3), since no additional resin intrusions concerns have occurred since IN 96-11 was issued. The extra burden on licensees to respond to Item 3 of the Required Information section is not justified.	IN 96-11 did not request specific action nor written response from licensees. Since the PWROGs have already told the Staff during the various meetings referenced in the Draft GL that they have evaluated their plants for susceptibility to a Zorita-type incident, providing that information to the Staff should not be burdensome.

#	Comment	Resolution
	It is unclear why the draft generic letter (GL) did not reference nor discuss in detail the evaluations and conclusions contained in NUREG/CR-6245. This document provides a balanced evaluation of the safety evaluations performed by the PWR Owners Groups.	NUREG/CR-6245 is listed as a reference under Related Generic Communications. The NRC staff's review of the PWR Owners Groups' safety evaluation was documented in the staff's SER dated November 19, 1993. The safety evaluation stated, in part, "...the staff recommends that you 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 ... nondestructive examinations should be performed to ensure there is no unexpected cracking in domestic PWRs. These examinations do not have to be conducted immediately ... 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 that it is prudent for NUMARC to consider the implementation of an enhanced leakage detection method for detecting small leaks during plant operation."
	The phrase "other vessel head penetrations" used throughout the draft generic letter should be clarified to read "other reactor vessel closure head penetrations".	Corrected in final draft of GL.
	Figure 1 and text appear to only discuss CRDMs designed by Westinghouse.	The figure is only for illustrative purposes, it is not considered necessary to show a type of every vendor's CRDMs.
	The first sentence [in the third paragraph of the Background section] states that in 1989 the emerging issue was identified, then the second sentence states leakage has occurred since 1986. This is not chronologically correct and is confusing.	Corrected in final draft of GL.
	The Bugey-3 cracking was discovered in September 1991.	Corrected in final draft of GL.

#	Comment	Resolution
	In the second sentence [in the fourth paragraph of the Background section], it states that the Japanese have "uncovered" VHPs with cracks. We are unaware of any available reference stating that the Japanese have detected PWSCC cracks in their VHPS. A source reference should be provided.	The sentence is poorly worded. It was not the Staff's intent to imply that cracking was found at Japanese plants (the Staff is not aware of any such cracking), but rather that, like the other countries, the Japanese inspected for cracking. Further, in Japan, the three most susceptible vessel heads were scheduled to be replaced because of safety considerations. The sentence will be reworded.
	It would be more precise to state that cracks were "detected" rather than "uncovered" in [the fourth paragraph of the Background section] and throughout the document's text.	Corrected in final draft of GL.
	NUREG/CR-6245 states that the leakage would be detected "long" before significant damage to the reactor vessel head would occur.	This statement is true; however, as was stated in the staff's November 19, 1993, safety evaluation: "...NUMARC should consider methods for detecting smaller [than 1 gpm] leaks to provide defense-in-depth to account for any potential uncertainty in its analyses." Relying solely upon detection of leakage, when the Standard Technical Specifications (TSs) and most facilities' TSs state that no pressure boundary leakage is acceptable, is not consistent with maintaining defense-in-depth. Further, as was reported in IN 86-108, Supplement 1, rapid and severe corrosion caused by boric acid leakage can occur before it is noticed.
	The last two sentences [in the sixth paragraph of the Background section] discuss manual NDE and do not relate to the remainder of the paragraph. The merits of manual NDE and automatic tooling are not the subject of the draft generic letter.	This information was provided to help indicate why the Staff did not pursue immediate action in 1991.
	The purpose of the EPRI NDE demonstration was not to qualify tooling or operators, but was limited to the demonstration of an inspection system's ability to detect and size defects.	Per the staff's discussions with EPRI, the demonstration was not intended to qualify each operator, but rather to qualify the system, which included the operator and equipment.

#	Comment	Resolution
	[The seventh paragraph of the Background section] appears to justify the draft generic letter based on advances in inspection techniques rather than assess the safety significance of PWSCC. This implies that inspections should be required because industry has voluntarily developed improved inspection methods. The paragraph should focus on safety concerns.	NUREG/CR-6245 described the basis for not requiring inspections until such time as methods and equipment could be developed to minimize radiation exposure.
	The description of the Zorita event could be more precise. The Zorita concern was primarily with the response of sensitized material attacked by reduced sulfur species. Inspections at the Zorita plant did not identify circumferential cracks in the J-groove weld, but found a through-wall crack at or near the bimetallic weld.	The Zorita event was described in IN 96-11 and in Westinghouse NSAL-94-028, and further detail is beyond the scope of this GL.
	In the third sentence [of the eight paragraph in the Background section], "resin bed" should be "resin bead."	Corrected in final draft of GL.
	The text would be better understood if the measurements were provided in English as well as metric units, i.e., "liters" and "gallons."	Corrected in final draft of GL.
	It is our understanding that the NRC Staff has Zorita resin intrusion reports and data that are not publicly available. it is difficult to assess the significance of the Zorita resin intrusion without all available information. In previous communications with the NRC Staff, we have been told that these reports have been provided to all PWR Owners Groups. However, inquiries made to the PWR Owners Groups have not supported this. We request the NRC Staff to place all information on the Zorita resin intrusion into the Public Document Room, and provide the opportunity for industry to evaluate.	The Westinghouse staff notified the WOG plants, the B&WOG plants, and the CEOG plants of the Zorita incident by issuing NSAL-94-028. The Staff does not have any information beyond that described in the draft GL and in IN 96-11. Specifically, as was stated in the December 1, 1994, meeting summary (dated December 20, 1994): "3. The Zorita data has been sent to the owners groups. The industry should provide an assessment of the Zorita experience and its implications with regard to their previously submitted stress analysis and safety evaluations."
	To maintain the chronological order of events, the 9th and 10th paragraphs [in the Background section] should be switched.	These paragraphs are grouped by subject matter.

#	Comment	Resolution
	The draft generic letter does not discuss the recent VHPs reinspections performed at Oconee and D.C. Cook, nor the VHP repair at D.C. Cook.	The GL was drafted before the staff had been informed of the results of the recent reinspections; relevant information regarding the results of the re-inspections and the repair at D.C. Cook will be included in the final GL.
	The NRC states that they have not been provided with the WOG resin intrusion review. IN 96-11 does not require any specific action by licensees. Furthermore, Westinghouse NSAL-94-028 did not request licensees to provide a response back to Westinghouse and no WOG report has been prepared.	Hence the need for the information requested by the draft GL. The NRC staff had previously requested information from the industry during the several meetings referenced in the draft GL. Specifically, as was stated in the December 1, 1994, meeting summary (dated December 20, 1994): "2. The NRC considers [PWSCC] in CRDM's a generic issue. The NRC staff requests industry to provide an integrated inspection plan. This should include schedules for [PWRs] that have not completed the CRDM penetration inspections, criteria for reinspections, and scope and method of inspection. The voluntary ad hoc approach suggested by NEI is undesirable in that plant specific submittals and reviews of this generic problem would be resource intensive both for the NRC and industry."
	The citation of Westinghouse, Framatome Technologies, and Combustion Engineering are incorrect. The citations should be the PWR Owners Groups; i.e., WOG, B&WOG, and the CEOG.	Corrected in final draft of GL.

#	Comment	Resolution
	[The fourteenth paragraph in the Background section] states that "(t)he program outlined in the NEI white paper is based on the assumption that the issue is an economic one rather than a safety issue,..." and that the NRC Staff did not agree that the issue was only economic. This is not a correct interpretation of the NEI white paper. The white paper documents the extensive safety evaluations developed by the PWR Owners Groups which addressed all the safety concerns identified by the NRC Staff. The method discussed in the white paper to manage RPV head penetration cracking acknowledges that the issue is not an immediate safety concern and that leak-before-break will occur. Using this knowledge, the management methodology discussed provides a four step approach; of which one step evaluates the economic considerations.	This interpretation is based on both the content of the White Paper and on comments made by NEI during the various referenced meetings with the Staff. The Staff disagrees with NEI that this is not a safety issue, and continues to believe that an integrated, long-term program, which includes periodic inspections and monitoring, is necessary. Further, it is difficult to come to a judgement on this matter since NEI has not provided the various PWR Owners Groups ranking models to the Staff.
	The sentence [in the first paragraph of the Discussion section] starting, "Further, if any significant ..." is an absolute statement which has not been technically justified in his document nor the references. It would be technically correct if the sentence was revised to read, "Further, if any significant resin intrusions have occurred at U.S. plants such as occurred at Zorita, the resultant chemistry condition in combination with stress may be significant."	It is the Staff's technical opinion that, given similar Zorita-type resin intrusions, residual stresses would be sufficient to cause circumferential cracking. This is consistent with the industry's explanation of the Zorita experience in meetings with the Staff.
	The sentence [in the second paragraph of the Discussion section] which starts, "Cracking in the VHPs ..." is potentially misleading. While cracking has occurred in 116 of the 5146 penetrations inspected, it has not been observed in the large majority of VHPS. PWSCC is an age related degradation mechanism which could occur some time in the future, many years beyond the initial or renewed license or never.	The statement is factual and needs no revision. As the comment agrees, cracking in VHPs has occurred and, since PWSCC is an age related degradation mechanism, is expected to continue to occur as plants age.

#	Comment	Resolution
	The [second paragraph of the Discussion section] states that the NRC Staff considers the cracking of VHPs to be a safety concern for the long-term based on the possibility of (1) exceeding the American Society of Mechanical Engineers (ASME) Code for margins if the cracks are sufficiently deep and continue to propagate during subsequent operating cycles, and (2) eliminating a layer of defense in depth for plant safety. These safety concerns are addressed by the PWR Owners Groups safety evaluations. These were summarized on Page 10 of NUREG/CR-6245, "Assessment of Pressurized Water Reactor Control Rod Drive Mechanism Nozzle Cracking," which states that "There are two major 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..." In addition, the NRC Staff's safety evaluation, dated November 19, 1993, states that "The primary safety concern associated with stress corrosion cracking in Alloy 600 is the potential for circumferential cracks. Extensive circumferential cracking could lead to ejection of a CRDM..." Since the PWR Owners Groups safety evaluations evaluated a through-wall crack and ejection of the connected CRDM housing, it appears that the two long term concerns identified by the draft GL are less severe than those already evaluated.	The industry's comment is that compliance with ASME Code margins of safety, as required by 10 CFR 50.55a, and elimination of defense-in-depth, are less severe than the issue elevated in the owners groups safety evaluations; nonetheless, they represent compliance with Federal Regulations.
	The concept of scheduling augmented inspections is inconsistent with the concept of "long term safety concerns." Given that technical safety concerns have been addressed, requesting a "technical basis" for a schedule is unclear.	The Staff is collecting information to determine if augmented inspections need to be required pursuant to 10 CFR 50.55a(g)(6)(ii).
	The required information is unnecessarily prescriptive (e.g., the direction of inspection top or bottom) will not affect the quality of an inspection which a licensee may choose to perform, the presence of thermal sleeves, etc.)	Corrected in final draft of GL.

#	Comment	Resolution
	The first sentence [in paragraph 2 of the Required Information section] states, "... include the susceptibility ranking of your plant and the factors used to determine this ranking." This phrase is redundant with the first part of the sentence which states, "A description of the evaluation methods and results used to assess the susceptibility of the CRDM and other VHFs in your plant to PWSCC, ... " The susceptibility models were not used as input to the PWR Owners Groups' safety evaluations that were submitted and approved by the NRC Staff. The susceptibility models and subsequent rankings may be used by licensees to make economic evaluations, but are not sufficiently precise to be used in a safety assessment that may be submitted to the NRC Staff. In addition, it is unclear how the NRC Staff will use such models to evaluate a safety concern.	The industry has indicated that a model has been developed and is being applied to determine if and when inspections are appropriate. The Staff is requesting the method and results of the licensee's evaluation. Requesting the susceptibility ranking and the factors used to determine this ranking is a very different request. The susceptibility models are being used by the industry to justify establishing vessel head penetration inspection schedules. If they are "not sufficiently precise to be used in a safety assessment," then how can they be precise enough to make a determination of when to inspect? The Staff wants to review the models and monitor their performance in order to assess their precision and reliability.

#	Comment	Resolution
	This requested information implies that the GL 88-05 visual inspection is inadequate to detect boric acid deposits and which could be caused by PWSCC. This implication is not supported by operating history and safety evaluations: • The only through-wall VHPs cracks (Bugey and Zorita) were detected by visual inspections. • GL 88-05 visual inspections are considered acceptable for detecting PWSCC in the remainder of the reactor coolant system. • A conservative definition for "long term safety concern" implied by NUREG-CR-6245 would infer a minimum of nine years after the initiation of a PWSCC through wall leak. Boric acid deposited over this time period would be readily observed using the GL 88-05 visual inspections.	• It is the Staff's understanding that the cracking at Bugey-3 was detected during a hydrotest, but not during normal operations, and the one at Zorita was detected during a refueling outage because of a buildup of boric acid crystals. • As was reported in IN 86-108, Supplement 1, rapid and severe corrosion caused by boric acid leakage can occur for before it is noticed. While the GL 88-05 program will find gross leakage some period of time after it began, the concern of this GL is whether licensees are maintaining defense in depth and Code margins and are able to discover cracking <u>before</u> it becomes severe enough to be noticed under a GL 88-05 program. Further, standard Technical Specifications do not allow pressure boundary leakage. • As stated in the NRC Staff's safety evaluation, dated November 19, 1993: "Once a leak start, 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..." The Staff agreed that "an immediate safety [is not present] as long as the surveillance walkdowns required continue <u>and</u> as long as any leakage is corrected." The staff did not make a determination as to how long it would take for a leak to start, nor how long it would take to take to reduce the wall thickness of the reactor vessel head to below ASME code minimums. However, it should be noted that relying solely upon detection of leakage, when most facilities' technical specifications state that no pressure boundary leakage is acceptable, is not consistent with maintaining defense-in-depth.

#	Comment	Resolution
	The intergranular stress corrosion cracking resulting from a Zorita type resin intrusion is a different mechanism than the primary water stress corrosion cracking (PWSCC). The resin intrusion cracking is a degradation mechanism caused by an abnormal operating event and is not a age-related degradation mechanism like PWSCC. Furthermore, the predictive tools for PWSCC are not capable of predicting resin intrusion. It is noted that the VHP inspections performed on over 5200 penetrations at 87 plants worldwide did not identify any other plant that exhibited intergranular stress corrosion cracking similar to that exhibited at Zorita. The draft generic letter has not provided a basis for supplying information on chlorides, fluorides, oxygen, boron, or lithium. The Zorita experience has been linked to the sulfates, but to our knowledge the other chemistry species have not been linked.	The Staff agrees that resin intrusion can only be identified through review of water chemistry history. Further, the vast majority of inspections that might identify problems resulting from the resin intrusions were not in U.S. plants. The Staff requires the requested information to determine if an augmented inspection program is required.
[7]	..., Southern Nuclear Operating Company is in total agreement with the NEI comments which are to be provided to the NRC.	See responses to NEI comments.
[8]	..., Georgia Power Company is in total agreement with the NEI comments which are to be provided to the NRC.	See responses to NEI comments.
[9]	See Comment #4	See responses to NEI comments.

[7] By letter dated September 30, 1996, comments from Dave Morey, Vice President, Farley; Southern Nuclear Operating Co; PO Box 1295; Birmingham, AL 35201

[8] By letter dated October 1, 1996, comments from C. K. McCoy, Vice President, Vogtle; Georgia Power Company; 40 Iverness Parkwry; PO Box 1295; Birmingham, AL 35201

[9] By letter dated October 2, 1996, comments from W. H. Bohlke, Vice President; Florida Power & Light Co.; P.O. Box 14000; Juno Beach, FL 33408

CRGR REVIEW PACKAGE

PROPOSED ACTION: Issue a generic letter on the primary water stress corrosion cracking of control rod drive mechanism and other vessel head penetrations.

CATEGORY: 2

RESPONSE TO REQUIREMENTS FOR CONTENT OF PACKAGE SUBMITTED FOR CRGR REVIEW

(i) **The proposed generic requirement or staff position as it is proposed to be sent out to licensees. Where the objective or intended result of a proposed generic requirement or staff position can be achieved by setting a readily quantifiable standard that has an unambiguous relationship to a readily measurable quantity and is enforceable, the proposed requirement should merely specify the objective or result to be attained, rather than prescribing to the licensee how the objective or result is to be attained.**

The information requested in items 1 and 2, below, 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 if the imposition of an augmented inspection program, pursuant to 10 CFR 50.55a(g)(6)(ii), is required, while the information requested in item 3 relates to the potential for domestic resin intrusions, such as occurred at Zorita.

Within 120 days from the date of this generic letter, addressees are requested to provide the following information[1]:

1. Regarding inspection activities:

   1.1 A description of all inspections of CRDMs and other vessel closure head penetrations performed to the date of this generic letter, including the results of these inspections.

   1.2 If you have developed a plan to periodically inspect the CRDM and other vessel closure head penetrations:

      a. Your schedule for first, and subsequent, inspections of the CRDM and other vessel closure head penetrations, including the technical basis for your schedule.

      b. Your scope for the CRDM and other vessel closure head penetration 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.

Attachment 2

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[1] Those licensees that have previously submitted the requested information need not resubmit it but should instead reference the appropriate correspondence in their response to this Generic Letter.

1.3 If you have <u>not</u> developed a plan to periodically inspect the CRDM and other vessel closure head penetrations, provide your technical or safety basis for not periodically inspecting your VHPs; or, your schedule for developing such a plan and the basis for that schedule.

2. A description of the evaluation methods and results used to assess the susceptibility of the CRDM and other VHPs in your plant to PWSCC, including the susceptibility ranking of your plant and the factors used to determine this ranking. Other than or in addition to the boric acid visual examination (*see* Generic Letter 88-05, "Boric Acid Corrosion of Carbon Steel Reactor Pressure Boundary Components in PWR Plants," dated March 17, 1988), include 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, and how these models substantiate your susceptibility model. Also, if you are relying on any integrated industry inspection program, provide a detailed description of this program.

3. A description of any resin intrusions in your plant, 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:

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

3.2 What were the durations of these intrusions?

3.3 Do your RCS water chemistry Technical Specifications follow the EPRI guidelines?

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

3.5 Identify any conductivity excursions which may be indicative of resin intrusions, provide your technical assessment of each excursion and your followup actions.

3.6 Provide your 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.

(ii) Draft staff papers or other underlying staff documents supporting the requirements or staff positions. (A copy of all materials referenced in the document shall be made available upon request to the CRGR staff. Any Committee member may request CRGR staff to obtain a copy of any reference material for his or her use.)

(1) Information Notice 90-10, "Primary Water Stress Corrosion Cracking (PWSCC) of Inconel 600," dated February 23, 1990.

(2) NRC staff safety evaluation, "Potential Reactor Vessel Head Adaptor Tube Cracking," dated November 19, 1993

(3) NUREG/CR-6245, "Assessment of Pressurized Water Reactor Control Rod Drive Mechanism Nozzle Cracking," dated October 1994.

(4) Information Notice 96-11, "Ingress of Demineralizer Resins Increases Potential for Stress Corrosion Cracking of Control Rod Drive Mechanism Penetrations," dated February 14, 1996.

(iii) Each proposed requirement or staff position shall contain the sponsoring office's position as to whether the proposal would increase requirements or staff positions, implement existing requirements or staff positions, or would relax or reduce existing requirements or staff positions.

The results of domestic VHP inspections are consistent with the February 1993 analyses by the PWR Owners Groups, the NRC staff safety evaluation report dated November 19, 1993, and the PWSCC found in the CRDMs in European reactors. On the basis of the results of the first five inspections of U.S. PWRs, the PWR Owner's Groups' analyses, and the European experience, the NRC staff has determined that there is a high probability that VHPs at other plants may contain similar axial cracks caused by PWSCC. Further, if any significant resin intrusions have occurred at U.S. PWRs such as occurred at Zorita, residual stresses are sufficient to cause circumferential intergranular stress corrosion cracking (IGSCC).

After considering this information, the NRC staff has concluded that VHP cracking does not pose an immediate or near term safety concern. Further, the NRC staff recognizes that the scope and timing of inspections may vary for different plants depending on their individual susceptibility to this form of degradation. In the long term, however, degradation of the CRDM and other VHPs is an important safety consideration that warrants further evaluation. The vessel closure head provides the vital function of maintaining a reactor pressure boundary. Cracking in the VHPs has occurred and is expected to continue to occur as plants age. The NRC staff considers cracking of VHPs to be a safety concern for the long term based on the possibility of (1) exceeding the American Society of Mechanical Engineers (ASME) Code for margins if the cracks are sufficiently deep and continue to propagate during subsequent operating cycles, and (2) eliminating a layer of defense in depth for plant safety. Therefore, in order to verify that the margins required by the ASME Code, as specified in Section 50.55a of Title 10 of the *Code of Federal Regulations* (10 CFR 50.55a) are met, that the guidance of General Design Criterion 14 of Appendix A to 10 CFR Part 50 (10 CFR Part 50, Appendix A, GDC 14) is continued to be satisfied, and to ensure that the safety significance of VHP cracking remains low, the NRC staff continues to believe that an integrated, long-term program, which includes periodic inspections and monitoring, is necessary. This was the conclusion of the staff's November 19, 1993, safety evaluation, which stated, in part, "...the staff recommends that you 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 ...

nondestructive examinations should be performed to ensure there is no unexpected cracking in domestic PWRs. These examinations do not have to be conducted immediately ... 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 that it is prudent for NUMARC to consider the implementation of an enhanced leakage detection method for detecting small leaks during plant operation." In addition, the NRC staff finds that the requested information is also needed to determine if the imposition of an augmented inspection program, pursuant to 10 CFR 50.55a(g)(6)(ii), is required to maintain public health and safety. Therefore, this request does not increase nor reduce existing requirements. It is a request to obtain information to confirm compliance with existing requirements.

The NRC staff recognizes that individual PWR licensees may wish to determine their inspection activities based on an integrated industry inspection program (i.e., B&WOG, CEOG, WOG, or some subset thereof), to take advantage of inspection results from other plants that have similar susceptibilities. The NRC staff does not wish to discourage such group actions but notes that such an integrated industry inspection program must have a well-founded technical basis that justifies the relationship between the plants and the planned implementation schedule.

(iv) **The proposed method of implementation with the concurrence (and any comments) of OGC on the method proposed. The concurrence of affected program offices or an explanation of any nonconcurrences.**

See attached concurrence page.

(v) **Regulatory analyses conforming to the directives and guidance of NUREG/BR-0058 and NUREG/CR-3568. (This does not apply for backfits that ensure compliance or ensure, define, or redefine adequate protection. In these cases a documented evaluation is required as discussed in IV.B.(ix).)**

Not applicable

(vi) **Identification of the category of reactor plants to which the generic requirement or staff position is to apply (that is, whether it is to apply to new plants only, new OLs only, OLs after a certain date, OLs before a certain date, all OLs, all plants under construction, all plants, all water reactors, all PWRs only, some vendor types, some vintage types such as BWR 6 and 4, jet pump and nonjet pump plants, etc.).**

All holders of operating licenses for pressurized water reactors (PWRs), except those licenses that have been amended to possession-only status.

(vii) **For backfits other than compliance or adequate protection backfits, a backfit analysis as defined in 10 CFR 50.109. The backfit analysis shall include, for each category of reactor plants, an evaluation which demonstrates how the action should be prioritized and scheduled in light**

of other ongoing regulatory activities. The backfit analysis shall document for consideration information available concerning any of the following factors as may be appropriate and any other information relevant and material to the proposed action:

(a) Statement of the specific objectives that the proposed action is designed to achieve;

Not applicable.

(b) General description of the activity that would be required by the licensee or applicant in order to complete the action;

Not applicable.

(c) Potential change in the risk to the public from the accidental release of radioactive material;

Not applicable.

(d) Potential impact on radiological exposure of facility employees and other onsite workers;

Not applicable.

(e) Installation and continuing costs associated with the action, including the cost of facility downtime or the cost of construction delay;

Not applicable.

(f) The potential safety impact of changes in plant or operational complexity, including the relationship of proposed and existing regulatory requirements and staff positions;

Not applicable.

(g) The estimated resource burden on the NRC associated with the proposed action and the availability of resources;

Not applicable.

(h) The potential impact of differences in facility type, design, or age on the relevancy and practicality of the proposed action;

Not applicable.

(i) Whether the proposed action is interim or final, and if interim, the justification for imposing the proposed action on an interim basis;

Not applicable.

(j) How the action should be prioritized and scheduled in light of other ongoing regulatory activities. The following information may be appropriate in this regard:

1. The proposed priority or schedule,
2. A summary of the current backlog of existing requirements awaiting implementation,
3. An assessment of whether implementation of existing requirements should be deferred as a result, and
4. Any other information that may be considered appropriate with regard to priority, schedule, or cumulative impact. For example, could implementation be delayed pending public comment?

Not applicable.

(viii) For each backfit analyzed pursuant to 10 CFR 50.109(a)(2) (i.e., not adequate protection backfits and not compliance backfits), the proposing Office Director's determination, together with the rational for the determination based on the consideration of paragraph (i) and (vii) above, that:

(a) There is a substantial increase in the overall protection of public health and safety or the common defense and security to be derived from the proposal; and

(b) The direct and indirect costs of implementation, for the facilities affected, are justified in view of this increased protection.

Not applicable.

(ix) For adequate protection or compliance backfits evaluated pursuant to 10 CFR 50.109(a)(4)

(a) a documented evaluation consisting of:
(1) the objectives of the modification
(2) the reasons for the modification
(3) the basis for invoking the compliance or adequate protection exemption.

(b) in addition, for actions that were immediately effective (and therefore issued without prior CRGR review as discussed in III.C) the evaluation shall document the safety significance and appropriateness of the action taken and (if applicable) consideration of how costs contributed to selecting the solution among various acceptable alternatives.

Not applicable. The proposed generic letter is a request for information only. The NRC staff is not requesting any new actions from the PWR licensees; rather, the proposed generic letter is requesting the PWR licensees to provide to the NRC information that the PWROGs has

already told the NRC staff it has gathered, but has not shared with the NRC to date.

(x) **For each evaluation conducted for proposed relaxations or decreases in current requirements or staff positions, the proposing Office Director's determination, together with the rationale for the determination based on the considerations or paragraphs (i) through (vii) above, that:**

(a) **Public health and safety and the common defense and security would be adequately protected if the proposed reduction in requirements or positions were implemented, and**

(b) **The cost savings attributed to the action would be substantial enough to justify taking the action.**

Not applicable.

(xi) **For each request for information under 10 CFR 50.54(f) (which is not subject to exception as discussed in III.A) an evaluation that includes at least the following elements:**

(a) **A problem statement that describes the need for the information in terms of potential safety benefit.**

The NRC staff was informed during a meeting on August 24, 1995, that Westinghouse had developed a susceptibility model for VHPs based on a number of factors, including operating temperature, years of power operation, method of fabrication of the VHP, microstructure of the VHP, and the location of the VHP on the head. Each time a plant's VHPs are inspected, the inspection results are incorporated into the model. All domestic Westinghouse PWRs have been modeled and the ranking has been given to each licensee. In addition, the NRC staff was informed that Framatome Technologies, Inc. [FTI, formerly Babcock & Wilcox (B&W)], also developed a susceptibility model for CRDM penetration nozzles and other VHPs in B&W reactor vessel designs. All domestic B&W PWRs have been modeled and the ranking has been given to each B&W licensee. The NRC staff was further informed that Combustion Engineering (CE) had performed an initial susceptibility assessment for the CE PWRs. At present, none of the PWR Owners Groups (i.e., WOG, B&WOG, or CEOG) has submitted its models and assessments to the NRC staff for review.

The results of domestic VHP inspections are consistent with the February 1993 analyses by the PWR Owners Groups, the NRC staff safety evaluation report dated November 19, 1993, and the PWSCC found in the CRDMs in European reactors. On the basis of the results of the first five inspections of U.S. PWRs, the PWR Owner's Groups' analyses, and the European experience, the NRC staff has determined that there is a high probability that VHPs at other plants may contain similar axial cracks caused by PWSCC. Further, if any significant resin intrusions have occurred at U.S. PWRs such as occurred at Zorita, residual stresses are sufficient to cause circumferential intergranular stress corrosion cracking (IGSCC).

After considering this information, the NRC staff has concluded that VHP cracking does not pose an immediate or near term safety concern. Further, the NRC staff recognizes that the scope and timing of inspections may vary for different plants depending on their individual susceptibility to this form of degradation. In the long term, however, degradation of the CRDM and other VHPs is an important safety consideration that warrants further evaluation. The vessel closure head provides the vital function of maintaining a reactor pressure boundary. Cracking in the VHPs has occurred and is expected to continue to occur as plants age. The NRC staff considers cracking of VHPs to be a safety concern for the long term based on the possibility of (1) exceeding the American Society of Mechanical Engineers (ASME) Code for margins if the cracks are sufficiently deep and continue to propagate during subsequent operating cycles, and (2) eliminating a layer of defense in depth for plant safety. Therefore, in order to verify that the margins required by the ASME Code, as specified in Section 50.55a of Title 10 of the *Code of Federal Regulations* (10 CFR 50.55a) are met, that the guidance of General Design Criterion 14 of Appendix A to 10 CFR Part 50 (10 CFR Part 50, Appendix A, GDC 14) is continued to be satisfied, and to ensure that the safety significance of VHP cracking remains low, the NRC staff continues to believe that an integrated, long-term program, which includes periodic inspections and monitoring, is necessary. This was the conclusion of the staff's November 19, 1993, safety evaluation, which stated, in part, "...the staff recommends that you 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 ... nondestructive examinations should be performed to ensure there is no unexpected cracking in domestic PWRs. These examinations do not have to be conducted immediately ... 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 that it is prudent for NUMARC to consider the implementation of an enhanced leakage detection method for detecting small leaks during plant operation." In addition, the NRC staff finds that the requested information is also needed to determine if the imposition of an augmented inspection program, pursuant to 10 CFR 50.55a(g)(6)(ii), is required to maintain public health and safety. Therefore, this request does not increase nor reduce existing requirements. It is a request to obtain information to confirm compliance with existing requirements.

The NRC staff recognizes that individual PWR licensees may wish to determine their inspection activities based on an integrated industry inspection program (i.e., B&WOG, CEOG, WOG, or some subset thereof), to take advantage of inspection results from other plants that have similar susceptibilities. The NRC staff does not wish to discourage such group actions but notes that such an integrated industry inspection program must have a well-founded technical basis that justifies the relationship between the plants and the planned implementation schedule.

(b) **The licensee actions required and the cost to develop a response to the information request.**

The information requested in items 1 and 2, below, 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 if the imposition of an augmented inspection program, pursuant to 10 CFR 50.55a(g)(6)(ii), is required, while the information requested in item 3 relates to the potential for domestic resin intrusions, such as occurred at Zorita.

Within 120 days from the date of this generic letter, addressees are requested to provide the following information[2]:

1. Regarding inspection activities:

   1.1 A description of all inspections of CRDMs and other vessel closure head penetrations performed to the date of this generic letter, including the results of these inspections.

   1.2 If you have developed a plan to periodically inspect the CRDM and other vessel closure head penetrations:

      a. Your schedule for first, and subsequent, inspections of the CRDM and other vessel closure head penetrations, including the technical basis for your schedule.

      b. Your scope for the CRDM and other vessel closure head penetration 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.

   1.3 If you have <u>not</u> developed a plan to periodically inspect the CRDM and other vessel closure head penetrations, provide your technical or safety basis for not periodically inspecting your VHPs; or, your schedule for developing such a plan and the basis for that schedule.

2. A description of the evaluation methods and results used to assess the susceptibility of the CRDM and other VHPs in your plant to PWSCC, including the susceptibility ranking of your plant and the factors used to determine this ranking. Other than or in addition to the boric acid visual examination (*see* Generic Letter 88-05, "Boric Acid Corrosion of Carbon Steel Reactor Pressure Boundary Components in PWR Plants," dated March 17, 1988), include 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, and how these models substantiate your susceptibility model. Also, if you are relying on any integrated industry inspection program, provide a detailed description of this program.

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2 Those licensees that have previously submitted the requested information need not resubmit it but should instead reference the appropriate correspondence in their response to this Generic Letter.

3. A description of any resin intrusions in your plant, 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:

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

   3.2 What were the durations of these intrusions?

   3.3 Do your RCS water chemistry Technical Specifications follow the EPRI guidelines?

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

   3.5 Identify any conductivity excursions which may be indicative of resin intrusions, provide your technical assessment of each excursion and your followup actions.

   3.6 Provide your 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.

All addressees are required to submit a written response with the information requested above within 120 days from the date of this letter.

Any inspection results that do not satisfy the acceptance criteria identified in the NRC staff's safety assessment dated November 16, 1993, should be reported to the NRC staff prior to plant restart.

The public reporting burden for this collection of information is estimated to average 80 hours per response, including the time for reviewing instructions, searching existing data sources, gathering and maintaining the data needed, and completing and reviewing the collection of information.

The cost estimated for the collection of information is estimated to average $8000.00 ($100/hour expended).

(c) An anticipated schedule for NRC use of the information.

The NRC staff plans to make immediate use of the requested information to verify that licensees are monitoring vessel head penetration cracking so as to provide reasonable assurance that existing regulations are being satisfied and to determine if augmented inspection rules need to be developed.

(d) **A statement affirming that the request does not impose new requirements on the licensee, other than for the requested information.**

Because the proposed generic letter only requests information from the PWR licensees, and the requested information has already been collected by the licensees (as stated by the PWR Owners Groups to the NRC during the meeting on August 24, 1995), the proposed generic letter does not impose new requirements on the licensees, other than submission of the requested information.

(xii) **An assessment of how the proposed action relates to the Commission's Safety Goal Policy Statement.**

The NRC staff feels that the proposed Generic Letter has no impact on the Commission's Safety Goal Policy Statement since it is only requesting information.