December 8, 2000

MEMORANDUM TO: File

FROM: Peter J. Kang, Reactor Systems Engineer
Engineering Section
Licensing Renewal and Standardization Branch
Division of Regulatory Improvement Programs
Office of Nuclear Reactor Regulation

SUBJECT: LIST OF OPEN ISSUES AND ACTION ITEMS FROM NUCLEAR ENERGY INSTITUTE (NEI) ON GENERIC AGING LESSONS LEARNED (GALL) REPORT

On December 7, 2000, Sam Lee received an electronic mail from Mr. Douglas Walters that lists action items and open issues resulted from the staff's November meetings with NEI on the GALL report. The staff plans to use this list when communicating with NEI for the completion of the GALL report.

Project No. 690

Attachment: As stated

cc: PUBLIC

December 7, 2000

LIST OF OPEN ITEMS/ACTIONS
RESULTING FROM NOVEMBER 2000 MEETINGS ON GALL AND SRP

The following reflects discussion and action items.

1. **GALL comment G-IV-4**: Threshold for neutron fluence – NRC has conservatively assigned a value of 10^{17} n/cm^2 whereas industry considers it to be 10^{21}. The fluence level is considered for IASCC, void swelling and synergistic thermal irradiation for CASS components. Within NRC, there was a disagreement on applying this fluence level for all three cases. It was felt that for IASCC and void swelling, this level was low but for thermal irradiation, it seemed reasonable. We indicated that some studies are ongoing to support the industry claim and WOG will present findings when available. NRC did indicate that 10^{17} n/cm^2 was to be used for screening only and inspection is only to be performed for the worst case. Everything else would be bounding.

    NEI ACTION: Provide information to back up the industry number.

2. **GALL comment G-IV-7:** Delta ferrite limit for CASS components – The NRC has limited to 25% delta ferrite the acceptability of existing SAW/SMAW flaw acceptance criteria for CASS components. Industry found that the available data, while sparse, shows good comparison out to 40%. The 25% data is from Argonne Labs. EPRI data available for 40%?

    NEI ACTION: Provide data, if available to NRC**.**

3. **GALL comment G-IV A1-1**: Industry asked NRC to replace BWRVIP 29 (TR-103515) with BWRVIP-79, which is TR-103515, Rev.2. NRC acknowledged they had the revision and will take a look at it.

4. **GALL comment G-IV-A2-16:** NRC wanted to know if calcs done for GL 97-01 were for 40 years and do they need to be considered TLAA and re-done to 60 years? In other words, was the model used time-dependent?

    NEI ACTION: Was the model time dependent? Should revised wordings be provided to the NRC?

5. **GALL comment G-IV-B1-10**: RPV internals separator support ring – Industry comment was to delete it, as it is not safety related. NRC was confused because their lab document called this part as separator support ring in two places and dryer support ring in two places. We explained that the separator and dryer were considered an assembly and are considered non-safety per BWRVIP documents.

    NRC Action: Agree on what the real name is and whether it needs to be in the GALL report.

6. **GALL comment G-V-A-1**: NEI comment asked NRC to delete reference to RG 1.44 as it relates more to design and construction process and not an aging management program. But at the same time NEI comment was to include

design and material controls consistent with RG 1.43. NRC was confused why NEI asked to delete RG 1.44 but include RG 1.43.

NEI ACTION: Does RG 1.43 apply?

7. **GALL comment G-V-D2-1:** this comment and others where Reactor Water Chemistry program is called out including Chapter XI.M11: Industry comment was to delete one-time inspection but consider other options (one-time inspection may be one) for demonstrating that water chemistry, by itself, was adequate to manage aging. NRC asked the industry to provide alternate wording.

   NEI ACTION: Provide alternate wording "in lieu of one-time inspection" for the Chapter XI.M11 reactor water chemistry program**.**

8. **GALL comment G-VB-2:** Charcoal absorber – NEI comment was to delete the entry. NRC wants to leave it as site specific and referred to their consumables letter.

9. **GALL comment G-VD2-2:** NRC agreed to NEI comment to change lower limit in the environment column from 25^{o} C. to 93^{o} C.

10. **GALL comment G-VIIC1-9**: NEI was asked to provide confirmation if the material was aluminum brass or aluminum bronze.

    NEI ACTION: Provide material information**.**

11. **GALL comment XI-M5-2**: BAC program – NEI once again explained why ISI references should be removed from the program description. NRC understood the NEI position and will review it again.

12. **GALL comment XI-M7-1**: AMP for Outer Surface of Above Ground Carbon Steel Tanks – NEI explained why this should be considered a plant specific aging program, since this is not a generic program used by all plants. This program description represents part of what one of the applicants is performing. NRC understood but is not likely to delete it from Chapter XI. They acknowledged that applicants would not necessarily use this program and would be specifying their own program.

13. **GALL comment XI-M8-1**: AMP for Outer Surface of Buried Pipe and Components – same discussion as response to comment XI-M7-1 above.

14. **GALL comment G-VE-7, 8 and XI-M-12-4:** Aging mechanisms/effects for carbon steel bolting – Considerable discussions occurred on why the industry believes that loss of pre-load and cyclic loading stress corrosion cracking are not considered aging degradation but are more either human performance/event related or high yield strength material related. NRC acknowledged our rationale and will review it further. Same comment applies to Chapter VII and VIII.

15. **GALL comment G-VE-2**: Coating program reference to Chapter XI.S8, which is a service level 1 containment coating program. NRC agreed to delete the coating program for mechanical components.

16. **SRP sections**: NRC acknowledged that any changes made to the GALL report should be reflected in the respective SRP sections.

17. **NEI Action** : Conduct survey on industry use of 96-03. Status: In process.

LIST OF OPEN ITEMS/ACTIONS
RESULTING FROM NOVEMBER 2000 MEETINGS ON GALL AND SRP

18. **Issue**: Use of IWE with Appendix J

19. **Issue**: Coatings program

20. **Issue**: Aggressive Chemical Attack

21. **Issue**: Porous Concrete

22. **Issue:** Spent fuel pool liner water chemistry

23. **Issue:** Bolting

24. **Issue:** Fire protection

25. **Issue:** Eliminate A-46

26. **Issue:** Reg Guide 1.160 MR QA Program

27. **Issue:** VT1 vs VT3

28. **ACTION**: send Sam Lee CMAA specification. Status: Complete.

29. **Issue**: Vibration of supports

30. **Issue**: Inaccessible area

Information For Item # 1

Excerpted from letter number OG-99-096, November 24, 1999, Roger Newton (WOG) to R. K. Anand (NRC), **<u>Response to NRC Request for Additional Information on WOG Generic Technical Reports: WCAP-14577, "License Renewal Evaluation: Aging Management For Reactor Vessel Internals"</u>**

RAI #7 AGING EFFECTS AND MANAGEMENT FOR CAST AUSTENITIC STAINLESS STEEL (CASS)

The RVI components fabricated from CASS are potentially subject to a synergistic loss of fracture toughness due to the combination of thermal and neutron irradiation embrittlement. This enhanced loss of fracture toughness is not accounted for within the topical report nor in guidance in revisions to EPRI TR-106092 (Ref. 3). Further, the topical report rules out consideration of thermal embrittlement of RVI CASS components based upon the lack of molybdenum in the materials. The NRC staff does not find this position of considering only thermal embrittlement to be acceptable. A modified screening approach should be used that is similar to that proposed in EPRI TR-106092 (Ref. 3), but also reflecting the potential synergistic effects of neutron irradiation and thermal embrittlement. One acceptable program is outlined below, consistent with the draft SER for the Calvert Cliffs license renewal application (Ref. 2).

The modified approach described in the draft SER for the Calvert Cliffs license renewal application (Ref. 1) consists of either a supplemental (enhanced VT-1) examination of the affected components as part of the applicant's 10-year ISI program during the license renewal term, or a component-specific evaluation to determine the susceptibility to loss of fracture toughness. The proposed evaluation will look first at the neutron fluence of the component. If the neutron fluence is greater than 1×10^{17} n/cm^2 (E > 1 MeV), a mechanical loading assessment would be conducted for the component. This assessment will determine the maximum tensile loading on the component during ASME Code Level A, B, C, and D conditions. If the loading is compressive or low enough to preclude fracture of the component, then the component would not require supplemental inspection. Failure to meet this criterion would require continued use of the supplemental (enhanced VT-1) inspection. If the neutron fluence is less than 1×10^{17} n/cm^2 (E > 1 MeV), an assessment would be made to determine if the affected component(s) are bounded by the screening criteria in EPRI TR-106092 (Ref. 3), modified as described below. In order to demonstrate that the screening criteria in EPRI TR-106092 (Ref. 3) are applicable to RVI components, a flaw tolerance evaluation specific to the RVI would be performed. If the screening criteria are not satisfied, then a supplemental (enhanced VT-1) inspection will be performed on the component.

The CASS components should be evaluated to the criteria in EPRI TR-106092 (Ref. 3) with the following additional criteria:

- *Statically cast components with a molybdenum content meeting the requirements of SA-351 Grades CF3 and CF8 and with a delta ferrite content less than 10 percent will not need supplemental examination.*
- *Ferrite levels will be calculated using Hull's equivalent factors or a method producing an equivalent level of accuracy (±6 percent deviation between measured and calculated values).*
- *Cast austenitic stainless steel components containing niobium are subject to supplemental examination.*

- *Flaws in CASS with ferrite levels less than 25 percent and no niobium may be evaluated using ASME Code IWB-3640 procedures.*
- *Flaws in CASS with ferrite levels exceeding 25 percent or niobium will be evaluated using ASME Code IWB-3640 procedures. If this occurs, fracture toughness data will be provided on a case-by-case basis.*

Components that have delta ferrite levels below the screening criteria have adequate fracture toughness and do not require supplemental inspection. Components that have delta ferrite levels exceeding the screening criteria may not have adequate fracture toughness, as a result of thermal embrittlement, and do require supplemental inspection.

The topical report should be revised to provide a more effective aging management program for cast austenitic stainless steel. An acceptable alternative is the program committed to by the applicant for license renewal of the Calvert Cliffs plant.

<u>RESPONSE</u>

The possible synergistic interaction of thermal and neutron embrittlement of CASS needs to be carefully considered with respect to the possible embrittling mechanisms involved and the available data for both types of embrittlement.

It appears that the fluence level of 10^{17}n/cm^2 was taken from the data for the onset of embrittlement in ferritic pressure vessel steels. This threshold is expected to be much higher in stainless steel weld and base metal materials. The primary mechanism for the embrittlement in these steels is the precipitation of a copper rich phase with possible contributions from nickel and phosphorous. The mechanism of thermal embrittlement below 500°C in the delta ferrite of CASS or of austenitic welds is primarily due to the spinodal decomposition of the chromium rich ferrite to produce variations in chromium content in the ferrite which is referred to as alpha prime embrittlement. There is no copper in these materials. In other words, the mechanisms of embrittlement are quite different for the two different types of ferrite. The data for many welds irradiated at intermediate temperatures (370°C to 430°C), even those with high molybdenum, show that “exposures up to 1 dpa have no significant effect on fracture resistance (Ref. 4).” For the PWR spectrum 1 dpa is 7 x 10^{20} n/cm^2. The literature data on castings is limited. (Ref. 5) That which is available indicates that the fracture toughness of an SA351 CF8 casting with 15% delta ferrite behaves in a similar manner to a 308SS weldment with 7% delta ferrite when irradiated (Ref. 6). Thus, any effects of thermal and neutron embrittlement are not expected until significant fluence at temperature is accumulated.

The major use of CASS in the internals of some of the Westinghouse PWR is the lower core support casting. The fluence at 32 EFPY for this component is typically less than 10^{19} n/cm^{2}and at 48 EFPY will still be less than the 1 dpa fluence cited above. Therefore, if the casting is acceptable based on the guidelines of EPRI TR-106092, no additional concerns should be addressed due to neutron fluence. The temperature of operation of the lower core support casting is expected to be close to the core inlet temperature (~520°F) which is significantly less than that at which the above referenced data was generated leading to a degree of conservatism in the argument. A rough calculation of the effect of lowering the temperature from 370°C where the reference data is cited to the conservatively expected temperature of 330°C (utilizing an estimated activation energy of 30 K cal/mol) on the embrittlement suggests that the susceptibility is lower at least by a factor of 4. The casting will be evaluated in accordance with the guidelines of TR-106092 as modified according to the additional criteria listed in RAI #7.

The only other place where CASS is used in some of the internals of a Westinghouse PWR is as a mixing vane device in the upper internals. It is expected that the loading on these components is sufficiently low that fracture will be precluded. These devices have been determined to not perform any intended function (see Table 2-1) and therefore aging management review is not required.

MODIFICATIONS TO THE TOPICAL REPORT

3.2.8 Thermal Aging

3.2.8.1 Mechanism Description

(no change)

3.2.8.2 Aging Effect Evaluation

The cast austenitic stainless steel lower core support forging is exposed to temperatures that could potentially lead to eventual thermal aging embrittlement, provided that the term of exposure is sufficiently long and that the other factors that control the extent of embrittlement (e.g., casting process, delta ferrite, and material chemistry) are unfavorable. The degradation of cast duplex stainless, if it occurs, is manifested by a decrease in fracture toughness, tearing modulus, and impact strength at room temperature. The fracture toughness, tearing modulus, and impact strength show only a moderate decrease at operating temperatures, 554°F to 617°F.

A review of thermal aging effects shows that cast austenitic stainless steel with ferrite contents as low as 10 percent are susceptible to thermal aging. Further, the structural welds in forged material could be susceptible to thermal aging. As stated above, all the cast duplex stainless steel reactor internals in the Westinghouse-designed NSSS are made from CF-8 or CF-8A.

While CF-8 material is susceptible to thermal aging at operating temperatures (354°F to 617°F), the remaining toughness is high with a Charpy value of 64 ft-lb and fracture toughness values of 750 in.-lb/in.^{2} for J_{Ic} and 3000 in.-lb/in.^{2} for J_{max} at room temperature for material with a high ferrite content (17 percent). Fracture mechanics evaluation of primary piping demonstrates structural integrity with Charpy impact energies as low as 2 ft-lb. Increasing the thermal aging temperature accelerates the thermal aging degradation of the fracture toughness of austenitic cast stainless steels. Using test results, higher temperature thermal aging data can be used to extrapolate to longer periods of time for thermal aging at lower thermal aging temperatures. Using an acceleration factor of 15 (which is conservative) for a thermal aging time for 752°F versus 617°F can project out to 450,000 hours of operation. CF-8 cast stainless steel is expected to have a Charpy value in excess of 28 ft-lb at the end of 60 calendar years or 48 effective full power years (EFPY).

Evaluations of cast internals components demonstrate that the effects of thermal aging for the reactor internals components are not significant and an evaluation or an aging management program for this effect will not be required during an extended period of operation.

REFERENCES FOR THESE RAI RESPONSES

1. References Thomas H. Essig to Lou Liberatori, Safety Evaluation of Topical Report WCAP-15029 “Westinghouse Methodology for Evaluating the Acceptability of Baffle-Former-Barrel Bolting Distributions Under Faulted Load Conditions” (TAC NO. MA1152), November 10, 1998.
2. “Safety Evaluation Report Related to the License Renewal of Calvert Cliffs Nuclear Power Plant, Units 1 and 2,” dated March 1999.
3. EPRI Technical Report TR-106092, “Evaluation of Thermal Aging Embrittlement for Cast Austenitic Stainless Steel Components in LWR Reactor Coolant Systems,” Electric Power Research Institute, September 1997.
4. J. Bernard and G. Verzeletti, “Elasto-Plastic Fracture Mechanics Characterization of Type 316H Irradiated Stainless Steel Up to 1 DPA,” Proceedings of the Twelfth ASTM International Symposium on “Effects of Radiation on Materials, Williamsburg, Virginia, June 18-20, 1984.
5. W. J. Mills, “Fracture Toughness of Irradiated Stainless Steel Alloys,” Nuclear Technology, Vol. 82, September 1988.
6. W. J. Mills, “Fracture Toughness of Type 304 and 316 Stainless Steels and Their Welds,” International Materials Reviews, 1997, V42, #2, pp. 45-82.
7. EPRI Technical Report TR-107521, “Generic License Renewal Technical Issues Summary,” Electric Power Research Institute, April 1998.

December 7, 2000

Information For Item # 6 and Item # 10

Information for Item # 6

RG 1.43 should not be included as a reference as indicated in our prior comment to chapter 5.

Information for Item # 10

The tube material is SB-111-687. The 687 copper alloy # series is 77.5% Cu, 20.5% Zn, and 2% Al. The aluminum is added to improve corrosion resistance particularly in condenser-tube applications.