Progress Energy

Crystal River Nuclear Plant
Docket No. 50-302
Operating License No. DPR-72

Ref: 10 CFR 54

October 2, 2009
3F1009-06

U.S. Nuclear Regulatory Commission
Attn: Document Control Desk
Washington, DC 20555-0001

Subject: Crystal River Unit 3 – Response to Request for Additional Information for the Review of the Crystal River Unit 3 Nuclear Generating Plant License Renewal Application (TAC NO. ME0274) – Sections B.2.37-1, 3.3.2.2, and XI.S8

References: (1) CR-3 to NRC letter, 3F1208-01, dated December 16, 2008, "Crystal River Unit 3 – Application for Renewal of Operating License"

(2) NRC to CR-3 letter dated September 2, 2009, "Request for Additional Information for the Review of the Crystal River Unit 3 Nuclear Generating Plant License Renewal Application (TAC NO. ME0274)"

Dear Sir:

On December 16, 2008, Florida Power Corporation (FPC), doing business as Progress Energy Florida, Inc. (PEF), requested renewal of the operating license for Crystal River Unit 3 (CR-3) to extend the term of its operating license an additional 20 years beyond the current expiration date (Reference 1). Subsequently, the Nuclear Regulatory Commission (NRC), by letter dated September 2, 2009, provided a request for additional information (RAI) concerning the CR-3 License Renewal Application (Reference 2). The Enclosure to this letter provides the response to Reference 2.

No new regulatory commitments are contained in this submittal.

If you have any questions regarding this submittal, please contact Mr. Mike Heath, Supervisor, License Renewal, at (910) 457-3487, e-mail at mike.heath@pgnmail.com.

Sincerely,

Jon A. Franke
Vice President
Crystal River Unit 3

JAF/dwh

Enclosure: Response to Request for Additional Information

xc: NRC CR-3 Project Manager
NRC License Renewal Project Manager
NRC Regional Administrator, Region II
Senior Resident Inspector

Progress Energy Florida, Inc.
Crystal River Nuclear Plant
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Crystal River, FL 34428

A035
NRR

STATE OF FLORIDA

COUNTY OF CITRUS

Jon A. Franke states that he is the Vice President, Crystal River Nuclear Plant for Florida Power Corporation, doing business as Progress Energy Florida, Inc.; that he is authorized on the part of said company to sign and file with the Nuclear Regulatory Commission the information attached hereto; and that all such statements made and matters set forth therein are true and correct to the best of his knowledge, information, and belief.

Jon A. Franke
Vice President
Crystal River Nuclear Plant

The foregoing document was acknowledged before me this 2 day of October, 2009, by Jon A. Franke.

Tara Barten

Signature of Notary Public
State of Florida

TARA BARTEN
Notary Public - State of Florida
My Comm. Expires Jan 28, 2013
Commission # DD 856189

(Print, type, or stamp Commissioned Name of Notary Public)

Personally Known ✓ -OR- Produced Identification ______

PROGRESS ENERGY FLORIDA, INC.

CRYSTAL RIVER UNIT 3

DOCKET NUMBER 50 - 302 / LICENSE NUMBER DPR - 72

ENCLOSURE

RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION

RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION

Request for Additional Information (RAI) B.2.37-1

Background

The Generic Aging Lessons Learned (GALL) Report identifies aging effects for stainless steel spent fuel storage racks and neutron absorbing materials, e.g., Boraflex, Boral, or boron-steel sheets, in pressurized-water reactor (PWR) treated water. Aging effects include loss of material or general corrosion and reduction of neutron-absorbing capacity, and further evaluation of a plant-specific aging management program (AMP) for those neutron absorbing materials is recommended. The GALL Report, Revision 1, does not address the specific use of Carborundum, a boron carbide shielding material, as a neutron absorber in spent fuel pools.

In a license amendment that permitted the use of Carborundum in the spent fuel pool at Crystal River Unit 3 Nuclear Generating Plant (CR-3), the applicant implemented a coupon surveillance program in its license amendment commitments to assess degradation of this material in its environment.

Issue

The CR-3 license renewal application (LRA) does not present sufficient specific plant information on how this program will manage reduction of neutron-absorbing capacity or loss of material for Carborundum in the spent fuel pool.

Request

1. To enable the staff to assess the adequacy of the existing plant specific Neutron-Absorber Monitoring and Carborundum (B_4C) Monitoring Program for managing aging effects for Carborundum:

    a. Indicate the installation date of the Carborundum panels in the CR-3 spent fuel pool.

    b. Describe how the loss of material and degradation of material will be monitored or inspected, specifically the methods, techniques, e.g., visual, weight, volumetric, surface inspection, neutron attenuation testing, frequency, sample size, data collection, timing and acceptance criteria.

    c. Describe how the neutron attenuation of the material will be measured. Please include a description of the testing, parameters measured, calculations, and acceptance criteria.

    d. Discuss the correlation between measurements of the physical properties of Carborundum coupons and the integrity of the Carborundum panels in the storage racks.

    e. Identify the subcritical margin used in the criticality analysis. Describe how the program acceptance criteria account for potential degradation between surveillance periods.

f. For the CR-3 Carborundum coupon samples:

i. Identify the quantity and location of coupons relative to the spent fuel racks during the license renewal period.

ii. Describe how the coupons are mounted and whether they are fully exposed to the spent fuel pool water.

g. Discuss whether any coupons that are removed and inspected using non-destructive techniques will be re-inserted in the spent fuel pool for future evaluation.

h. Describe how the results from the inspections of the Carborundum coupons will be monitored and trended, including frequency and sample size, e.g., the number of coupons examined during each surveillance interval.

i. Describe the corrective actions that would be implemented if coupon test results do not meet the acceptance criteria.

Response

1.a The high density Carborundum racks were installed into the borated water environment of Spent Fuel Pool A over a period spanning the first 4 months of 1981.

1.b. A plant procedure details the Carborundum (B_4C) monitoring program at CR-3.

(1) General methodology – Sample coupons of Carborundum material are located in Spent Fuel Pool A. One set resides in a rack holder and the other in a wall holder. This is described in further detail in response to 1.f.ii. On a periodic basis, sample coupons are removed, visually inspected and weighed. Dimensional (length and width) measurements are taken based upon results of the visual and weight results. Gamma dose measurements are also taken when sample coupons are removed, or their location changed.

(2) Techniques

a. Visual (surface) inspection – once the samples have been removed, the first step is a visual inspection. The visual inspection looks for obvious signs of deterioration such as indication of B_4C grain loss, uniformity, spalling, voids, and backing or binder degradation. The sample is determined to be in one of six categories of deterioration.

Samples in Categories 1, 2, or 3 meet acceptance criteria. A Nuclear Condition Report (NCR), and a resulting investigation and evaluation, is initiated for Conditions 4, 5, or 6. (See acceptance criteria on next page.)

b. Weight – After completion of the visual inspections, the sample coupons are dried and weighed. The drying is to remove excess water and return the samples to the state of the original, initial weigh in. Once weighed, the sample coupon weights are compared to the original, initial weights. The change in weight is presented in both actual weight change (gms) and percent weight change.

If the percent weight change does not meet the acceptance criteria, then an NCR is initiated and dimensional measurements (length and width) are taken of the sample coupon. Dimensions are also recorded of any missing material. This data is used to evaluate poison material condition. Additionally, the procedure directs raising the Spent Fuel Pool boron concentration to the refueling level.

(3) *Acceptance criteria*

a. *Visual – sample coupons shall meet the visual inspection criteria for Category 1, 2, or 3:*

1. *B_4C grains intact and surface texture uniform both sides; no visible discoloration as compared to an unexposed control sample (i.e., sample not exposed to environment).*

2. *B_4C grains intact and surface texture uniform both sides; visible discoloration as compared to control sample.*

3. *Minor loss of B_4C grains at surface, either side, but leaving no appreciable craters or voids.*

b. *Visual - sample coupons inspection criteria which result in initiation of a Nuclear Condition Report (Category 4, 5, or 6).*

4. *B_4C grains at surface, either side, noticeably loose and spalling off leaving large craters and/or voids.*

5. *B_4C grains intact and surface texture uniform both sides; however cracks, blisters, or separation of fiberglass backing and binder apparent (unless obviously caused by handling of samples).*

6. *Conditions more severe than categories 4 or 5 above.*

c. *Weight – maximum allowed weight loss is 20%. As will be described in greater detail in response to other questions below, a 20% weight loss corresponds to a 15% boron loss. The criticality analyses model a 15% boron loss in order to provide an acceptance criteria margin for this surveillance.*

(4) *Surveillance Frequency – The surveillance frequency was started on a short interval, increasing to approximately a 5-year interval. The first surveillance was scheduled during the 2^{nd} year of operation, the second surveillance the 3^{rd} year, the third surveillance the 5^{th} year, the fourth surveillance the 10^{th} year. Corresponding to these original surveillance intervals was the expected dose to be received. Originally, the expected dose was conservatively estimated to be 0.5 E +10 rads per year, or 2.5 E +10 rads per 5 years. Measured data demonstrated that the dose rate was actually closer to 0.25 E +10 rads per year. Therefore, when the existing surveillance schedule (which ended in 2002) was extended for the expected life of the racks, the frequency of sample coupon inspections was increased 10 years (still 2.5 E +10 rads per interval) to 2012.*

The rack sample coupon holder is still relocated every refueling to maintain an accelerated dose, as is further described in the response to 1.f.ii.

(5) *Sample size – Generally speaking, the sample size is one sample coupon packet. There are two types of sample packets, A and B.*

a. *Type A sample coupon packets contain 10 samples, measuring roughly 1 inch by 2 inch and weighing about 2.1 to 2.2 gms. To date, all sample surveillances have been with A type packets except for the 1985 (year 3) surveillance which pulled both an A and B sample.*

b. *Type B samples contain 4 larger samples, measuring roughly 2 inch by 10 inch and weighing about 21 to 22 gms.*

Though of different size, the sample coupons are essentially the same, being made of the same material. Both types are sandwiched in metal sample packets that consist of a stainless steel support sheet as a backing, the poison sample, then a stainless steel closure sheet on top. The sample packet has a vent hole in the top closure sheet. The racks themselves likewise have a vent hole near the top of the rack.

(6) *Neutron attenuation testing – Neutron attenuation testing is not part of the surveillance monitoring program for the Carborundum B_4C racks at CR-3.*

Discussion: One of the references for the program implementing procedure is the Handbook of the Effects of In-Pool Exposure on Properties of Boron Carbide-Resin Shielding Materials, June 1981, The Carborundum Company. This handbook is the result of testing performed to ensure Carborundum could stand up to the gamma and borated water environment.

On page 15 of the handbook, it states, "Analysis of the sheet material after exposure to 10^9 gray (10^{11} rad) showed a reduction in boron content of 15% compared to the noted 20% weight loss of the material." As will be discussed in response to 1.e, the criticality analyses assume a 15% loss of boron in the racks. This 15% boron loss correlates to a 20% weight loss. Therefore, since the weight loss provides an indicator of the loss of the neutron absorber, weight loss is used in lieu of neutron attenuation.

1.c *Neutron attenuation is not directly measured by the surveillance program for the Carborundum racks in Spent Fuel Pool A. As noted in Response 1.b(6) above, weight loss is used in lieu of neutron attenuation testing. Since boron is the neutron absorber in B_4C, and since testing has demonstrated a correlation between weight loss and boron loss, weight loss has been considered an acceptable mechanism by which to monitor continued rack poison neutron attenuation capability.*

1.d *The sample coupons simulate the Carborundum material in the racks in that:*

(1) *The sample coupon material is the same as that in the racks. There are two types of Carborundum material. One type is a thick, rigid plate supported by a phenol formaldehyde resin. This is generally referred to as the "plate" type Carborundum material. The second type is chemically and structurally similar, except that it incorporates a woven glass fabric reinforcement. This second type is referred to as boron carbide "sheet" material. Both the rack Carborundum material and the sample coupon material at CR-3 are of the sheet type.*

(2) *The sample coupon packets simulate the rack structure. The sample coupons consist of poison material sandwiched between two stainless steel sheets, much as the rack poison material is situated in the racks. The sample packets also contain a vent hole in the cover sheet of stainless steel, simulating the vent hole at the top of the fuel racks. In this way, the sample coupon packets allow ingress of borated water and egress of gas just as the vent hole does in the rack structures.*

(3) *The sample packets are exposed to the same borated water environment. The gamma sample holder resides in the racks themselves; therefore, it sees the same borated water conditions and temperatures as the boron carbide sheet material in the rack walls. The water sample holder resides some distance from the rack cells, to limit gamma exposure. However, since the spent fuel pool cooling is a forced flow function, the water sample holder can likewise be safely assumed to be in water conditions (boron and temperature) very near that of the rack cells.*

(4) *One set of the sample coupons, those attached to the gamma sample holder, is exposed to the rack gamma flux. These coupons are attached to a fuel shaped holder that is placed within a rack cell location. This location is surrounded by fuel freshly discharged each refueling outage. This ensures the gamma samples are exposed to an accelerated gamma dose, compared to the gamma exposure seen by any single location containing fuel. Therefore, the gamma sample coupons are exposed to an accumulated gamma dose greater than the poison material in the racks. During each surveillance interval, sample coupons are removed from the gamma sample holder.*

Therefore, the sample coupons accurately reflect the conditions and exposure of the rack boron carbide material, and the physical condition of the rack material can be expected to be at least as good as the sample coupon conditions.

Correlation to boron absorption capabilities

A testing program was performed to evaluate the effects of environmental conditions found in the spent fuel pools. The tested materials were subjected to both gamma flux and water chemistry similar to that found in both Boiling Water Reactor and Pressurized Water Reactor pools. Details of the testing program and results can be found in, "Handbook of the Effects of In-Pool Exposure on Properties of Boron Carbide-Resin Shielding Materials," produced by the Carborundum Company. One of the results of this testing was that after 10^{11} rads of exposure, the tested material showed a 15% boron loss for a 20% weight loss. It should also be noted that the report attributes some of this loss to specimen handling and not to in-pool losses. Therefore, since the sample coupons are exposed to larger gamma doses than the rack material in general, and since the handling issue suggests that any coupon weight loss (and consequential boron loss) would bound any rack material loss, the sample coupons present a conservative representation of the condition and continued viability of the rack boron carbide material.

1.e *Criticality analysis assumptions:*

(1) *Margin in the criticality analysis – Though the water in the spent fuel pools is administratively maintained borated to greater than 2050 parts per million (ppm) boron, the criticality analyses at CR-3 have not taken credit for soluble boron for routine storage. Soluble boron is used only to cover fuel misloading concerns.*

Therefore, Spent Fuel Pool A (the pool with the Carborundum racks) are determined to be less than a K_{eff} of 0.95 with unborated water when loaded with the most reactive configuration of fuel.

The various allowable fuel loading patterns in Spent Fuel Pool A Carborundum racks are supported by a series of criticality analyses. All but one of these analyses modeled a 15% boron loss in the Carborundum. The one analysis that did not explicitly model the 15% loss was evaluated and determined to have sufficient margin to support the 15% loss criteria. Therefore, the high density boron carbide racks in Spent Fuel Pool A can lose up to 15% of the boron (20% weight loss) and still support a K_{eff} of less than 0.95 with unborated water.

The additional negative reactivity from the administratively maintained > 2050 ppm boron provides significant additional margin.

(2) The surveillance procedure results are reviewed against the 15% boron loss (20% weight loss). To date, the surveillance results have remained well below the acceptance criteria, yielding approximately 5% weight loss. A spreadsheet is maintained that trends the surveillance results. However, the B_4C Program surveillance procedure is being enhanced to include a provision to formally monitor and trend data for weight loss to ensure the projection meets the acceptance criteria. The weight loss has risen slowly to only about 5% over 20 years. The current trend would not anticipate a 20% weight loss over a 10 year surveillance interval nor a 20% weight loss by the end of the license extension period (2036). With the current trend, and considering the test results documented in the "Handbook of the Effects of In-Pool Exposure on Properties of Boron Carbide-Resin Shielding Materials", it is not likely that a sudden significant change that would challenge the acceptance criteria would occur during the interval between measurements.

Should a significant increase in weight loss occur at one of the test intervals, the schedule can be adjusted to reduce the frequency between measurements.

The minimum 2050 ppm boron provides additional significant negative reactivity margin.

1.f.i Sample coupon quantity and locations:

(1) Rack (gamma) samples – rack samples, also known as gamma samples (see Response 1.f.ii below), are located in the fuel racks. These sample coupons are attached to a fuel assembly shaped holder that is seated in the fuel rack like a fuel assembly. The sample packets consist of poison material sandwiched between stainless steel, modeling the actual rack poison configuration, including a vent hole. There are 5 rack sample packets remaining. These samples will extend the surveillance program through the year 2053, which is beyond the extended period of operation for license renewal.

(2) Water samples – water samples are located well above the plane of the fuel, but still within the borated water environment of Spent Fuel Pool A, and are attached to the side of the pool. The sample packets consist of poison material sandwiched between stainless steel, modeling the actual rack poison configuration, including a vent hole. There are 5 water sample packets

remaining. These samples will extend the surveillance program through the year 2053, which is beyond the license renewal interval.

1.f.ii *Two types of coupons are supplied for the Carborundum surveillance program. These are referred to as water samples and gamma samples and are discussed below. Both sets of samples are exposed to the borated water environment of Spent Fuel Pool A.*

(1) *The water samples are mounted on a flat plate well out of the primary gamma flux from the spent fuel. This plate is mounted on the pool wall well above the fuel and is intended to represent the affect of primarily just borated water on the Carborundum material. The average dose is estimated at 0.01 R/hr or approximately 10 mr/hr.*

(2) *The gamma samples are mounted on a dummy fuel assembly and are placed in a fuel rack cell location in the center of freshly discharged fuel. The gamma sample holder is moved after each refueling to the center of fuel discharged that refueling outage so that it will receive and maintain an accelerated gamma exposure compared to any individual pool rack location. The intent of the gamma samples is to provide a leading indicator of the effect of both borated water and gamma flux. In this way the sample coupons should detect poison material degradation before the material in the racks themselves.*

As of the relocation of the gamma sample holder in the last refueling outage (2007), the accumulated dose estimate on the gamma sample coupons was 4.42 E +10 rads.

1.g *Water and gamma sample coupons removed from Spent Fuel Pool A for surveillance inspections are not returned to the pool for future evaluation.*

1.h *Remaining sample coupon surveillances are established on an approximately 10 year interval. This will result in 5 more surveillances, nominally scheduled for 2012, 2023, 2033, 2043 and 2053. Each surveillance performance removes a gamma sample packet and a water sample packet.*

(1) *The next two surveillance intervals are scheduled to remove B type sample packets. As discussed in Response 1.b(5), the B packets contain 4 samples of larger size than the A type packet.*

(2) *An A type packet is pulled in the middle of the period. A packets contain more samples than B packets (10 versus 4) and are smaller.*

(3) *The final two surveillances consist of a B packet followed by an A packet.*

(4) *Results are trended by CR-3 Reactor Engineering on a spreadsheet tracking weight loss. Failure to meet any acceptance criteria is documented, investigated and evaluated via the Corrective Action Program.*

1.i *Failure to meet the visual inspection acceptance criteria results in initiation of an NCR, which will drive further investigation and evaluation. Failure to meet the weight loss acceptance criteria will result in the following:*

(1) *Raise the spent fuel pool boron concentration to the refueling boron concentration.*

(2) *Measure and record sample coupon dimensions (length and width) as well as the dimensions of any missing material.*

(3) *Initiate an NCR which will drive further investigation and evaluation.*

RAI 3.3.2.2.6-1

Background

The GALL Report identifies loss of material or general corrosion and reduction of neutron-absorbing capacity as aging effects requiring management (AERM) for Boral in PWR treated water, and calls for further evaluation of a plant-specific AMP.

Issue

CR-3 LRA Section 3.3.2.2.6, "Reduction of Neutron-Absorbing Capacity and Loss of Material due to General Corrosion," states that, for Boral spent fuel storage racks exposed to a treated water environment, the aging management review determined that there has been no adverse operating experience at CR-3 with regard to Boral. It further stated that the aging effects for Boral is insignificant and does not require aging management. The LRA does not address applicability of recent adverse operating experience with Boral.

The LRA states that management of loss of material is performed by a plant specific program. However, the CR-3 LRA does not present any specific plant information on how this program will manage loss of material for Boral in the spent fuel pool.

Request

1. The GALL report for neutron absorbing materials cites both loss of material and loss of neutron absorbing capacity as aging effects. Describe how the CR-3 plant specific program addresses each of these effects.

2. If the applicant identifies AERM for Boral, describe the AMPs that will be used. Specifically:

   a. Provide the 10 elements of the AMP for Boral, i.e., scope of program, preventive actions, parameters monitored or inspected, detection of aging effects, monitoring and trending, acceptance, corrective actions, confirmation process, administrative controls, operating experience, including the coupons that will be under surveillance.

   b. Indicate whether the Boral panels in the spent fuel pool are vented or not.

   c. Indicate the installation date of the Boral panels in the CR-3 spent fuel pool.

   d. Describe the surveillance approach that will be used in the cited AMP, specifically, the methods and techniques utilized, e.g., visual, weight, volumetric, surface inspection, neutron attenuation testing, frequency, sample size, data collection, timing and acceptance criteria.

e. Please describe how the neutron attenuation of the material will be measured. Please include a description of the testing, parameters measured, calculations, and acceptance criteria.

f. Discuss the correlation between measurements of the physical properties of Boral coupons and the integrity of the Boral panels in the storage racks.

g. Describe the corrective actions that would be implemented if coupon test results do not meet the acceptance criteria.

3. In September 2003, inspection of Boral test coupons at Seabrook Nuclear Station revealed bulging and blistering of the aluminum cladding. Blistering or bulging on Boral coupons has also been noted at Three Mile Island and Beaver Valley. Blisters or bulges in the Boral panels may impact the ability to insert or remove fuel from cells. In addition, voids caused by blisters or bulges may affect neutron attenuation through flux trap formation.

a. Please discuss the impact that these findings, along with any relevant findings at CR- 3, have on the continued functionality of Boral at CR-3.

Response

1. The LRA indicates that Boral® is used in the spent fuel storage racks (Pool B), but the aging effects for Boral® are insignificant and do not require aging management (refer to LRA Table 3.5.2-2; Table Item 3.3.1-13; Notes I and 528; and Section 3.3.2.6). This was based on no adverse operating experience recorded for Progress Energy's CR-3 or Harris Nuclear Plants and on the results of the NRC staff evaluations of the V. C. Summer Nuclear Station and the Brunswick Steam Electric Plant for these aging effects. The License Renewal Safety Evaluation Reports for the latter two plants have determined the aging effects to be insignificant. When the spent fuel storage racks using Boral® were installed in pool B in 2001, there was no requirement to have a surveillance program for the Boral®; and no coupons were inserted in the pool. The Safety Evaluation Report dated September 13, 2000, in response to the CR-3 License Amendment Request to install the spent fuel racks with Boral® in pool B, evaluated the spent fuel storage racks using Boral® with no surveillance program being required. Also, a letter from the NRC, from Laurence Kopp to Krishna Singh (Holtec International), dated February 15, 1995, stated, "the NRC has no current requirement for in-service surveillance on Boral® in spent fuel storage racks." Since there are no aging effects identified for Boral® and the current licensing basis does not require a plant specific program for Boral®, a plant specific aging management program for Boral® is not in place to manage loss of material or reduction of neutron absorbing capacity. Therefore, LRA Appendix B, Aging Management Programs, did not include an aging management program for Boral.

2.a-g Since there have been no aging effects identified, and an Aging Management Program is not utilized, a response has not been provided for items 2.a through 2.g. However, the following additional information is provided: Per the CR-3 Final Safety Analysis Report (FSAR), Section 9.6.2.4, the Boral® used in the spent fuel racks in Spent Fuel Pool B is a metallic composite of a hot rolled (sintered) aluminum matrix containing boron carbide (B_4C) sandwiched between and bonded to type 1100 alloy aluminum.

3.a The Seabrook operating experience report and the 10 CFR Part 21 notification concerning bulging and blistering of a Boral® test coupon have been reviewed for *impact on CR-3. Although Seabrook identified bulging and blistering in September 2003 by an inspection of their test coupon, the evaluation of this condition concluded that the acceptance criteria for B-10 areal density were met and that there was no impact on the structural integrity of the racks. Based on these conclusions, no safety concerns have been identified related to the CR-3 spent fuel storage racks with Boral®. Additional operating experience related to bulging and blistering of the Boral® aluminum cladding identified at Beaver Valley was also reviewed. Beaver Valley also concluded that blistering did not affect neutron attenuation or the structural integrity of the spent fuel storage racks. Additional operating experience related to bulging and blistering of the Boral® coupons at Three Mile Island (TMI) was also reviewed. It was concluded that there was no loss of boron and no reduction in neutron attenuation tests.*

Based on current industry operating experience reviewed at Seabrook, Beaver Valley, and TMI, as discussed above, it was concluded that a Boral® monitoring program is not needed at CR-3. CR-3 will continue to monitor industry operating experience related to Boral® through the Institute of Nuclear Power Operations (INPO) Operating Experience Program, and any necessary actions will be initiated through the Corrective Action Program. The spent fuel racks containing Boral® should continue to function throughout the period of extended operation.

RAI XI.S8

Background

The GALL Report states that proper maintenance of protective coatings inside containment (defined as Service Level I in Nuclear Regulatory Commission Regulatory Guide [RG] 1.54, Rev. 1) is essential to ensure operability of post-accident safety systems that rely on water recycled through the containment sump/drain system. Degradation of coatings can lead to clogging of strainers, which reduces flow through the sump/drain system.

Issue

The CR-3 LRA does not credit the protective coating monitoring and maintenance program for aging management. Although the licensee does not credit the program for aging management, there needs to be adequate assurance that there is proper maintenance of the protective coatings in containment, such that they will not degrade and become a debris source that may challenge the Emergency Core Cooling Systems performance.

Request

1. Please describe in detail the CR-3 coatings assessment.

    a. Please describe how the program will ensure that there will be proper maintenance of the protective coatings inside containment such that they will not become a debris source that could impact the operability of post-accident safety systems that rely on water recycled through the containment sump or drain system in the extended period of operation.

    b. Please describe the frequency and scope of the inspections, acceptance criteria, and the qualification of personnel who perform containment coatings inspections.

Response

1.a *As described below, the CR-3 Safety Related Coatings Program and the CR-3 ASME Section XI, Subsection IWE, Containment Inspection Program ensure that there will be proper maintenance of the protective coatings inside containment such that they will not become a debris source that could impact the operability of post-accident safety systems.*

The Safety Related Coatings Program and the ASME Section XI, Subsection IWE, Containment Inspection Program are implemented and maintained in accordance with the general requirements for engineering programs. These requirements provide assurance that the programs are effectively implemented to meet regulatory, process, and procedure requirements. The programs specify periodic program reviews and the incorporation of industry and plant-specific operating experience.

The primary purpose of the Safety Related Coatings Program at CR-3 is to ensure that protective coatings inside the Reactor Building do not adversely impact the function of the Emergency Core Cooling System (ECCS). This is done by maintaining the quantity of unqualified or degraded coatings with the potential to be transported to the Reactor Building sump below the calculated design limit for clogging the ECCS suction strainer. Coatings on the surfaces internal to closed components are not included due to their inability to be transported to and ultimately contribute to clogging of the sump strainer. The quantity of coatings is determined through the use of a log of known degraded or unqualified coatings as determined by containment inspections and other engineering evaluations. The inspections are performed jointly by the Safety Related Coatings and ASME Section XI, Subsection IWE program with the ASME Section XI, Subsection IWE program inspecting those components within the scope of that program (Reactor Building liner plate, penetrations, hatches, etc.) and the balance of the inspection performed by the Safety Related Coatings Program. The Safety Related Coatings Program based inspections are performed each refueling outage and the ASME Section XI, Subsection IWE program based inspections are examined three periods during each interval (three refueling outages in 10 years). The log is updated following each refueling outage that involves either new inspection findings or changes to old findings such as repairs. Actions to maintain acceptable sump margin include procedural controls to prevent the addition of unqualified/degraded coatings into the containment structure and maintenance activities to remove unqualified/degraded coatings that are already present.

1.b *As described in Response 1.a above, the Safety Related Coatings Program based containment assessment is performed each refueling outage and the ASME Section XI, Subsection IWE program based containment inspection is performed once each ASME Section XI Interval Period (three refueling outages in 10 years).*

Specific acceptance criteria for the Safety Related Coatings Program are provided in the condition assessment procedure. Coatings acceptance criteria include lack of blistering, cracking, flaking, rusting, checking, insufficient adhesion, and undercutting in accordance with various ASTM standards. Specific acceptance criteria for the ASME Section XI, Subsection IWE program are provided in Nondestructive Examination (NDE) procedures. Coatings acceptance criteria include the lack of blistering, cracking, flaking, rusting, peeling, discoloration, checking, and wear.

The Safety Related Coatings Program assessment inspections are performed by qualified Safety Related Coatings Program Managers or qualified coating inspectors. ASME Section XI, Subsection IWE, component inspections within the Reactor Building are performed by NDE qualified personnel.

Safety Related Coatings and IWE Program Managers are trained and qualified to a specific Progress Energy Training Guides within the INPO accredited Engineering Support Personnel (ESP) Training Program. NDE personnel are qualified for visual examination by the VT-1 or VT-3 Techniques defined by the ASME B&PV Code.