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<u>*Proprietary Notice*</u>
This letter forwards GNF proprietary information in accordance with 10CFR2.390. Upon the removal of Enclosure 1, the balance of this letter may be considered non-proprietary.

MFN 06-467

Docket No. 52-010

November 29, 2006

U.S. Nuclear Regulatory Commission
Document Control Desk
Washington, D.C. 20555-0001

Subject: **Response to Portion of NRC Request for Additional Information Letter No. 66 Related to ESBWR Design Certification Application – Chapter 4 and GNF Topical Reports – RAI Numbers 4.2-8 through 4.2-10, 4.2-14, 4.3-6, 21.6-86 through 21.6-89**

Enclosure 1 contains GE's response to the subject NRC RAIs transmitted via the Reference 1 letter.

Enclosure 1 contains GNF proprietary information as defined by 10 CFR 2.390. GNF customarily maintains this information in confidence and withholds it from public disclosure. A non proprietary version is provided in Enclosure 2. The affidavit contained in Enclosure 3 identifies that the information contained in Enclosure 1 has been handled and classified as proprietary to GNF. GE hereby requests that the information of Enclosure 1 be withheld from public disclosure in accordance with the provisions of 10 CFR 2.390 and 9.17.

D068

If you have any questions about the information provided here, please let me know.

Sincerely,

Jim Kinsey
for
David H. Hinds
Manager, ESBWR

Reference:

1. MFN 06-377, Letter from U. S. Nuclear Regulatory Commission to Mr. David H. Hinds, *Request for Additional Information Letter No. 66 Related to ESBWR Design Certification Application,* October 10, 2006

Enclosures:

1. MFN 06-467 - Response to Portion of NRC Request for Additional Information Letter No. 66 Related to ESBWR Design Certification Application – DCD Chapter 4 and GNF Topical Reports – RAI Numbers 4.2-8 through 4.2-10, 4.2-14, 4.3-6, 21.6-86 through 21.6-89 – GNF Proprietary Information
2. MFN 06-467 - Response to Portion of NRC Request for Additional Information Letter No. 66 Related to ESBWR Design Certification Application – DCD Chapter 4 and GNF Topical Reports – RAI Numbers 4.2-8 through 4.2-10, 4.2-14, 4.3-6, 21.6-86 through 21.6-89 – Non Proprietary Version
3. Affidavit – Jens G. M. Andersen – dated November 29, 2006

cc: AE Cubbage USNRC (with enclosures)
AA Lingenfelter GNF/Wilmington (w/o enclosures)
GB Stramback GE/San Jose (with enclosures)
eDRFs 0059-9219 for 4.2-8, -9, -10, and -14
0060-7173 for 4.3-6
0059-2459 for 21.6-86
0059-2460 for 21.6-87
0059-2461 for 21.6-88
0060-2923 for 21.6-89

ENCLOSURE 3

MFN 06-467

Affidavit

Affidavit

I, Jens G. M. Andersen, state as follows:

(1) I am Consulting Engineer, Thermal Hydraulic Methods, Global Nuclear Fuel – Americas, L.L.C. ("GNF-A") and have been delegated the function of reviewing the information described in paragraph (2) which is sought to be withheld, and have been authorized to apply for its withholding.

(2) The information sought to be withheld is contained in Enclosure 1 of GE letter MFN 06-467, David H. Hinds to U. S. Nuclear Regulatory Commission, *Response to Portion of NRC Request for Additional Information Letter No. 66 Related to ESBWR Design Certification Application – DCD Chapter 4 and GNF Topical Reports - RAI Numbers 4.2-8 through 4.2-10, 4.2-14, 4.3-6, 21.6-86 through 21.6-89* dated November 29, 2006. The proprietary information in Enclosure 1, *MFN 06-467 Response to Portion of NRC Request for Additional Information Letter No. 66 Related to ESBWR Design Certification Application – DCD Chapter 4 and GNF Topical Reports - RAI Numbers 4.2-8 through 4.2-10, 4.2-14, 4.3-6, 21.6-86 through 21.6-89,* is delineated by double underlined dark red font text and is enclosed inside double square brackets. Figures and large equation objects are identified with double square brackets before and after the object. The superscript notation{3} refers to Paragraph (3) of this affidavit, which provides the basis for the proprietary determination.

(3) In making this application for withholding of proprietary information of which it is the owner or licensee, GNF-A relies upon the exemption from disclosure set forth in the Freedom of Information Act ("FOIA"), 5 USC Sec. 552(b)(4), and the Trade Secrets Act, 18 USC Sec. 1905, and NRC regulations 10 CFR 9.17(a)(4) and 2.390(a)(4) for "trade secrets " (Exemption 4). The material for which exemption from disclosure is here sought also qualify under the narrower definition of "trade secret," within the meanings assigned to those terms for purposes of FOIA Exemption 4 in, respectively, Critical Mass Energy Project v. Nuclear Regulatory Commission, 975F2d871 (DC Cir. 1992), and Public Citizen Health Research Group v. FDA, 704F2d1280 (DC Cir. 1983).

(4) Some examples of categories of information which fit into the definition of proprietary information are:

a. Information that discloses a process, method, or apparatus, including supporting data and analyses, where prevention of its use by GNF-A's competitors without license from GNF-A constitutes a competitive economic advantage over other companies;

b. Information which, if used by a competitor, would reduce his expenditure of resources or improve his competitive position in the design, manufacture, shipment, installation, assurance of quality, or licensing of a similar product;

c. Information which reveals aspects of past, present, or future GNF-A customer–funded development plans and programs, of potential commercial value to GNF-A;

d. Information which discloses patentable subject matter for which it may be desirable to obtain patent protection.

The information sought to be withheld is considered to be proprietary for the reasons set forth in paragraphs (4)a. and (4)b., above.

(5) To address the 10 CFR 2.390 (b) (4), the information sought to be withheld is being submitted to NRC in confidence. The information is of a sort customarily held in confidence by GNF-A, and is in fact so held. Its initial designation as proprietary information, and the subsequent steps taken to prevent its unauthorized disclosure, are as set forth in (6) and (7) following. The information sought to be withheld has, to the best of my knowledge and belief, consistently been held in confidence by GNF-A, no public disclosure has been made, and it is not available in public sources. All disclosures to third parties including any required transmittals to NRC, have been made, or must be made, pursuant to regulatory provisions or proprietary agreements which provide for maintenance of the information in confidence.

(6) Initial approval of proprietary treatment of a document is made by the manager of the originating component, the person most likely to be acquainted with the value and sensitivity of the information in relation to industry knowledge, or subject to the terms under which it was licensed to GNF-A. Access to such documents within GNF-A is limited on a "need to know" basis.

(7) The procedure for approval of external release of such a document typically requires review by the staff manager, project manager, principal scientist or other equivalent authority, by the manager of the cognizant marketing function (or his delegate), and by the Legal Operation, for technical content, competitive effect, and determination of the accuracy of the proprietary designation. Disclosures outside GNF-A are limited to regulatory bodies, customers, and potential customers, and their agents, suppliers, and licensees, and others with a legitimate need for the information, and then only in accordance with appropriate regulatory provisions or proprietary agreements.

(8) The information identified in paragraph (2) is classified as proprietary because it contains details of GNF-A's fuel design and licensing methodology.

The development of the methods used in these analyses, along with the testing, development and approval of the supporting methodology was achieved at a significant cost, on the order of several million dollars, to GNF-A or its licensor.

(9) Public disclosure of the information sought to be withheld is likely to cause substantial harm to GNF-A's competitive position and foreclose or reduce the availability of profit–making opportunities. The fuel design and licensing methodology is part of GNF-A's comprehensive BWR safety and technology base, and its commercial value extends beyond the original development cost. The value of the technology base goes beyond the extensive physical database and analytical methodology and includes development of the expertise to determine and apply the appropriate evaluation process. In addition, the technology base includes the value derived from providing analyses done with NRC–approved methods.

The research, development, engineering, analytical, and NRC review costs comprise a substantial investment of time and money by GNF-A or its licensor.

The precise value of the expertise to devise an evaluation process and apply the correct analytical methodology is difficult to quantify, but it clearly is substantial.

GNF-A's competitive advantage will be lost if its competitors are able to use the results of the GNF-A experience to normalize or verify their own process or if they are able to claim an equivalent understanding by demonstrating that they can arrive at the same or similar conclusions.

The value of this information to GNF-A would be lost if the information were disclosed to the public. Making such information available to competitors without their having been required to undertake a similar expenditure of resources would unfairly provide competitors with a windfall, and deprive GNF-A of the opportunity to exercise its competitive advantage to seek an adequate return on its large investment in developing and obtaining these very valuable analytical tools.

I declare under penalty of perjury that the foregoing affidavit and the matters stated therein are true and correct to the best of my knowledge, information, and belief.

Executed at Wilmington, North Carolina this 29th day of November 2006.

Jens G M Andersen

Jens G. M. Andersen
Global Nuclear Fuels – Americas, LLC

Enclosure 2

MFN 06-467

Response to Portion of NRC Request for

Additional Information Letter No. 66

Related to ESBWR Design Certification Application

RAI Numbers 4.2-8 through 4.2-10, 4.2-14, 4.3-6, 21.6-86 through 21.6-89

Non-Proprietary Version

NRC RAI 4.2-8

Similar to RAI 4.2-5, DCD Tier 2, Rev. 1, Appendix 4C should define specific Tier 2 and Tier 2 control rod design requirements. The current text appears to be an overview of a control rod design change process and should be revised. Section 4C.1 states, "...designs meeting the following acceptance criteria are considered to be approved and do not require specific NRC review". The NRC staff disagrees with this statement. The control rod design employed in the initial core (Cycle 1) in any facility referencing the ESBWR certified design must be specifically reviewed and approved by the NRC if the design deviates from the control blade design approved in the design certification. Accordingly, the staff requests that GE mark the requirements as Tier 2* information in the next DCD revision.*

GE Response

GE agrees to revise the first sentence of paragraph 4C.1 of the DCD Tier 2 to state: "The control rod will meet the following acceptance criteria:". This revised statement does not imply NRC pre-approval of future design modifications. The first paragraph of Appendix 4C is similarly edited to remove any implication of NRC pre-approval of future control rod designs.

DCD Impact

DCD Tier 2 Section 4C.1 and the first paragraph of Appendix 4C will be revised as noted in the attached markup. Section 4C.1 will also be marked as Tier 2*. It is also noted that these same design criteria are contained in the DCD Tier 1.

NRC RAI 4.2-9

DCD Tier 2, Rev. 1, Section 4.2.4.9 states, "Subsequent Marathon designs or absorber section loadings will be within +5%)k/k of the initial ESBWR Marathon design". The control rod design employed in the core in any facility which adopts the ESBWR certified design must be specifically reviewed and approved by the NRC if the design deviates from the control blade design approved in the design certification. Clarify what is meant by "subsequent" and explain the intent in providing this statement in Section 4.2.4.9 of the DCD, Tier 2, Rev. 1.

GE Response

GE agrees to remove the verbiage regarding 'subsequent' control rod designs. Similar to RAI 4.2-8, this removes any implication of NRC pre-approval of future design modifications.

DCD Impact

DCD Tier 2 Subsection 4.2.4.9 will be edited as shown in the attached mark-up.

NRC RAI 4.2-10

DCD Tier 1, Rev. 1, Section 2.9 and Tier 2, Revision 1, Appendix 4C.1, include a control rod design requirement which states that "...lead surveillance control rods may be used". Please clarify what is meant by the phrase "may be used," (i.e., what type or magnitude of design change would warrant in-reactor service prior to batch implementation.) Please revise the design requirement accordingly.

GE Response

GE agrees to remove the lead surveillance control rod criteria from both the Tier 1 and Tier 2 DCDs.

DCD Impact

DCD Tier 1 Section 2.9 and DCD Tier 2 Sections 4C.1 and 4C.2 will be edited as shown in the attached mark-ups.

NRC RAI 4.2-14

DCD Tier 1, Rev. 1, Section 2.9 and DCD Tier 2, Revision 1, Appendix 4C.1 defines principal design criteria for the control rod. One of the design criteria states that the stresses, strains, and cumulative fatigue will be evaluated to not exceed the ultimate stress or strain limit of the material. Certain BWR control rod designs include long axial welds between the square tubes and welds connecting the absorber wings to the handle and connector. In order to set design requirements on material properties, it must be demonstrated that structural properties (e.g. weld regions) are never more limiting than the material properties throughout the expected lifetime of the control rod. Provide evidence (e.g. mechanical testing) to demonstrate that the structural properties would never be more limiting or re-write the design requirement.

GE Response

For clarity, GE proposes to revise the design criteria in Tier 1 DCD Section 2.9 and Tier 2 DCD Section 4C.1 from:

> "…to not exceed the ultimate stress or strain limit of the material."

To:

> "…to not exceed the ultimate stress or strain limit of the material, structure, or welded connection."

GE has performed testing to compare the strength of the welded connections to the strength of the square tube material for Marathon control rods. The first test was a tensile test performed on a test panel of welded square tubes. The test specimen also included welds at top and bottom to plate material, which duplicated the absorber section to fin/handle welds. The test specimen was loaded axially in a tensile test machine. The result of this test was that the material of the square tubes ruptured prior to the failure of any of the welds.

The second test performed was a burst pressure test, testing the pressurization capability of the square tubes. Similar to the tensile test, test panels of welded square tubes were prepared, with welds to plate material at top and bottom to duplicate the absorber section to fin/handle welds. The square tubes were pressurized until rupture. Again, it was found that the material of the square tube failed before any of the welds.

DCD Impact

Tier 1 DCD Section 2.9 and Tier 2 DCD Sections 4C.1 and 4C.2 will be edited as shown above and in the attached mark-ups.

NRC RAI 4.3-6

Provide the effective void reactivity coefficient calculated by PANACEA for BOC, MOC, and EOC at nominal operating conditions.

GE Response

The void coefficient is the ratio of the change in k-effective to the change in void fraction:

$$VODCOF = \frac{1}{k}\frac{\partial k}{\partial(\%VOID)}$$

The void coefficient of reactivity can be calculated with PANAC11 by [[ ]]. [[ ]]

The void coefficient of reactivity is calculated at three exposures and the results are shown in the following table:

[[

]]

DCD Impact

No DCD changes will be made in response to this RAI.

NRC RAI 21.6-86

The isotopic tracking in the PANAC11 code is discussed in NEDC-33239P. Please provide a prototypical calculational model (e.g. the differential equations) for the determination of plutonium content based on the nodal power, exposure, and moderator density history.

GE Response

Isotopic concentration effects are included in the lattice data provided by TGBLA06. PANAC11 captures these effects during the library processing when generating the macroscopic cross sections based on lattice data looked up with exposure E, spectral history corrected historical relative moderator density UHSPH and instantaneous relative moderator density U.

The model described in the paragraph 1.4.7 of NEDC-33239P refers to a prior method of isotopic accounting. PANAC11 tracks isotopics in method directly analog to the method described above for macroscopic cross sections. There is no feedback between the isotopic accounting and nodal parameter. NEDC-33239P will be revised in the next update to delete Section 1.4.7.

DCD Impact

There are no changes to the DCD.

NRC RAI 21.6-87

PANAC11 uses the GEXL correlation to determine critical quality for the purpose of calculating the minimum critical power ratio. Describe how PANAC11 calculates the bundle power where boiling transition occurs.

GE Response

PANAC11 uses the GEXL methodology to determine the Critical Power Ratio for each bundle. GEXL uses the concept of minimum thermal margin, which is based on the local equilibrium quality and critical quality. Given the pressure, flow rate, inlet subcooling, axial power shape, and fuel lattice design and an assumed value for the critical power, local quality and boiling length are computed for each axial node using energy and mass balance. The critical quality is also computed for each node using the GEXL correlation as described in references 1-5 and 1-6 of General Electric Standard Application for Reactor fuel (GESTAR Main), NEDE-24011-P-A-15, September 2005. If at any nodes, the local quality is greater than the critical quality, a lesser value for the critical power is assumed. If the local quality is less than the critical quality at all of the nodes, a greater value for the critical power is assumed. The iteration continues until the local quality is just equal to the critical quality at one node of the nodes and is less at all other nodes. The power for this last iteration is the predicted critical power.

DCD Impact

There are no changes to the DCD.

NRC RAI 21.6-88

The determination of the core flow distribution is described in NEDC-33239P. Describe how the linear interpolation technique is used to determine bundle flow based on the characteristic bundle calculations. Provide a description of the range of flow and power conditions enveloped by the characteristic bundle calculations. If there are cases where bundle flow is determined by extrapolation of parameters beyond the envelope of conditions in the characteristic bundle calculations, provide justification.

GE Response

The goal of the iterative process that determines the flow in each characteristic channel is to obtain an equal channel drop pressure drop for all channels while preserving the total core flow. Therefore, there is no range of power and flow enveloped by the characteristic channel calculations. These calculations are performed at the flow and power conditions input by the user.

Each individual bundle is associated with a characteristic channel by determining which value of each of the five characteristic parameters is closest to the corresponding parameters of the bundle in question. The flow rate in each fuel bundle, which is represented by the characteristic channel IX is set equal to the characteristic channel flow, WCB(IX), modified by corrections which vary linearly with the amount that the actual bundle at (i,j) is removed from characteristic variables, namely radial power, axial power, and crud build-up.

DCD Impact

There are no changes to the DCD.

NRC RAI 21.6-89

Describe the procedure for calculating the detector response kernels that are used for simulated plant instrument response as discussed in NEDC-33239P.

GE Response

Two types of TIPs detector responses are available: gamma and thermal. The TIP detector response, CALTIP, at axial node k in detector string s is an average response based on estimates from four surrounding nodes:

[[ ]]

where P_{ijk} is th nse (TIP correlation) which is dependent on fuel type, exposure, voids, void history, control fraction and detector type.

The equations used to calculate the nodal detector response for thermal TIPs are as follows.

[[ ]]

where

[[ ]]

where Σ_{fg} is the fission cross section, $\hat{\phi}$ the flux at the instrument location and the U235 fission cross sections are given by

[[ )]]

where,

VF is the nodal instantaneous void fraction.

XCU25I are the microscopic fission cross section fit coefficients

ADTIP is the number density of U-235 at instrument location

$\bar{\psi}'_g$ is looked up from the lattice data as a function of fuel type, exposure, void, void history and control fraction.

[[

]]

The LPRM response CALPRM at detector *l* in detector string *s* is calculated using the thermal TIP model and adjusted to account for calibration gain factors and sensitivities:

[[ ]]

where

[[ ]]

[[ ]]

DCD Impact

There are no changes to the DCD.

2.9 CONTROL RODS

Design Description

Control rods in the reactor perform the functions of power distribution shaping, reactivity control, and scram reactivity insertion for safety shutdown response and have the following design features:

- A cruciform cross-sectional envelope shape;
- A connector at the bottom for attachment to the control rod drive; and
- Contain neutron absorbing materials.

The following is a summary of the principal design criteria, which are met by the control rod:

- The control rod stresses, strains, and cumulative fatigue will be evaluated to not exceed the ultimate stress or strain limit of the material, structure, or welded connection~~.~~; [RAI 4.2-14]
- The control rod will be evaluated to be capable of insertion into the core during all modes of plant operation within limits assumed in plant analyses~~.~~;
- The material of the control rod will be compatible with the reactor environment~~.~~;
- The reactivity worth of the control rods will be included in the plant core analyses.~~; and~~
- ~~Prior to use of new design features on a production basis, lead surveillance control rods may be used.~~ [RAI 4.2-10]

Inspections, Tests, Analyses and Acceptance Criteria

No entries for this system.

rod life when absorber burn-up helium gas generation is highest. Absorber tube loads are evaluated during a seismic event near the end of control rod life when absorber burn-up helium gas generation is highest. Absorber section to connector welds and absorber section to handle loads are evaluated during a SCRAM when the absorber helium gas build-up is highest. Per Reference 4.2-8, the ESBWR Marathon does not exceed the ultimate stress or strain limit of the material. Based on the reactor cycles, the combined loads are then evaluated for the cumulative effect of the cyclic loadings in Reference 4.2-8. The fatigue usage is evaluated against a limit of 1.0.

4.2.4.6 Handling Loads

The ESBWR Marathon is designed to accommodate three times the weight of the control rod, Reference 4.2-8.

4.2.4.7 Hydraulics

Inspection experience over 15 years has shown the Marathon control rod is not damaged by the vibrations or cavitations set up by coolant velocities and velocity distributions in the bypass region between fuel channels.

4.2.4.8 Materials

Materials selected for use in the Marathon control rod components are chosen to minimize the component end-of-life radioactivity in order to reduce personnel exposure during handling on-site, and for final off-site shipping and burial. All Marathon control rod materials are less than <0.03 weight percent cobalt. The average niobium content for the handle and absorber section, less boron carbide and hafnium, is < 0.1 weight percent.

4.2.4.9 Nuclear Performance

The nuclear lifetime of the initial ESBWR Marathon control rod type will be established as 10 percent reduction in reactivity worth ($\Delta k/k$) in any quarter axial segment, Reference 4.2-9. ~~Subsequent Marathon designs or absorber section loadings will be within +/-5%Δk/k of the initial ESBWR Marathon design.~~

~~Similar to what has been provided to US BWRs over the last 18 years, additional Marathon control rod designs may be supplied with different absorber configurations allowing higher reactivity worth and larger relative allowable decrease with respect to the initial ESBWR Marathon control rod type's 10 percent reactivity worth reduction.~~[RAI 4.2-9]

4.2.4.10 Mechanical Compatibility

Similar to the control rods supplied to ABWR and BWR/2 through BWR/6, the ESBWR Marathon control rod is designed to be compatible with core and reactor internal interfaces.

The ESBWR Marathon is designed to be compatible with the control rod guide tube (CRGT) cylindrical boundary, to provide a seat with the guide tube base during Fine Motion Control Rod Drive (FMCRD) removal, to provide lower guide rollers for smooth transitions, and to have clearance with the orificed fuel support for insertion and withdrawal from the core.

4C. CONTROL ROD LICENSING ACCEPTANCE CRITERIA

A set of acceptance criteria has been established for evaluating ~~new~~ [RAI 4.2-8]control rod designs. Control rod compliance with these criteria constitutes the basis for NRC acceptance and approval of the design~~ without specific NRC review~~. [RAI 4.2-8] The control rod licensing acceptance criteria and their bases are provided below. Any change to these criteria must have prior NRC review and approval.

4C.1 GENERAL CRITERIA

~~Control rod designs meeting t~~The control rod will meet following acceptance criteria ~~are considered to be approved and do not require specific NRC review~~: [RAI 4.2-8]

- The control rod stresses, strains, and cumulative fatigue shall be evaluated to not exceed the ultimate stress or strain limit of the material, structure, or welded connection. [RAI 4.2-14]
- The control rod shall be evaluated to be capable of insertion into the core during all modes of plant operation within the limits assumed in the plant analyses.
- The material of the control rod shall be shown to be compatible with the reactor environment.
- The reactivity worth of the control rod shall be included in the plant core analyses.
- ~~Prior to use of new design features on a production basis, lead surveillance control rods may be used.~~[RAI 4.2-10]

4C.2 BASIS FOR ACCEPTANCE CRITERIA

The following licensing bases is provided for the acceptance criteria given in Section 4C.1:

Stress, Strain and Fatigue

The control rod is evaluated to assure that it does not fail because of loads due to shipping, handling, normal operation, including the effects of anticipated operational occurrences (AOOs), infrequent incidents and accidents. To ensure that the control rod does not fail, these loads must not exceed the ultimate stress and strain limit of the material, structure, or welded connection. [RAI 4.2-14] Fatigue must not exceed a fatigue usage factor of 1.0.

The loads evaluated include those due to normal operational transients (scram and jogging), pressure differentials, thermal gradients, flow and system induced vibration, and irradiation growth in addition to the lateral and vertical loads expected for each condition. Fatigue usage is based upon the cumulative effect of the cyclic loadings. The analyses include corrosion and crud deposition as a function of time, as appropriate.

Conservatism is included in the analyses by including margin to the limit or by assuming loads greater than expected for each condition. Higher loads can be incorporated into the analyses by increasing the load itself or by statistically considering the uncertainties in the value of the load.

Control Rod Insertion

The control rod is evaluated to be sure that it can be inserted during normal operations including the effects of anticipated operational occurrences (AOOs), infrequent incidents and accidents. These evaluations include a combination of analyses of the geometrical clearance and actual testing. The analyses consider the effects of manufacturing tolerances, swelling and irradiation growth. Tests may be performed to demonstrate control rod insertion capability for conditions such as control rod or fuel channel deformation and vibrations due to safe shutdown earthquakes.

Control Rod Material

The external control rod materials must be capable of withstanding the reactor coolant environment for the life of the control rod. Effects of crudding, crevices, stress corrosion and irradiation upon the material must be included in the control rod and core evaluations. Irradiation effects to be considered include material hardening and absorber depletion and swelling.

Reactivity

The reactivity worth of the control rod is determined by the initial amount and type of absorber material and irradiation depletion. Scram time insertion performance must also be included in the plant core analyses including normal operations, including the effects of anticipated operational occurrences (AOOs), and accidents.

~~*Surveillance*~~

~~Visual inspection of the lead depletion control rod design possessing the new design feature and three additional control rods of such design that are within 15% of the estimated fast fluence of the lead control rod shall be performed. If fewer than three control rods are within 15% of the estimated fast fluence of the lead control rod, only those within 15% shall be inspected. If a control rod with the new design feature reaches analytical end of life, and is visually inspected with no significant issues, the new design feature surveillance program ends. Should evidence of a problem arise, the inspection program is expanded to additional control rods to the extent necessary to identify the root cause of the problem.~~ [RAI 4.2-10]

4C.3 COL UNIT-SPECIFIC INFORMATION

None.