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CENG
a joint venture of
Constellation Energy
eDF

CALVERT CLIFFS
NUCLEAR POWER PLANT

October 2, 2012

U. S. Nuclear Regulatory Commission
Washington, DC 20555

ATTENTION: Document Control Desk

SUBJECT: Calvert Cliffs Nuclear Power Plant
Unit Nos. 1 & 2; Docket Nos. 50-317 & 50-318
Updated Final Safety Analysis Report, Revisions 44 and 45

Enclosed for your use is one copy (CD) of our Updated Final Safety Analysis Report (UFSAR), Revisions 44 and 45. These revisions are submitted within six months after the latest refueling outage in accordance with 10 CFR 50.71(e).

The List of Effective pages are included. This electronic copy is a complete revision.

Attachment (1) provides a description of changes to commitments made to the Nuclear Regulatory Commission.

Should you have questions regarding this matter, please contact Mr. Douglas E. Lauver at (410) 495-5219.

Very truly yours,

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NRR

CERTIFICATION:

I, George H. Gellrich, certify that I am Vice President-Calvert Cliffs Nuclear Power Plant, LLC, and that the information contained in this submittal accurately presents changes made since the previous submittal, necessary to reflect information and analyses submitted to the Commission or prepared pursuant to Commission requirement.

GHG/BJD/bjd

Attachment: (1) Changes to Commitments Made to the Nuclear Regulatory Commission

Enclosure: Updated Final Safety Analysis Report, Revisions 44 and 45 (CD)

cc: N. S. Morgan, NRC

(Without Enclosure)
W. M. Dean, NRC
Resident Inspector, NRC
S. Gray, DNR

ATTACHMENT (1)

CHANGES TO COMMITMENTS MADE TO THE NUCLEAR REGULATORY COMMISSION

ATTACHMENT (1)

CHANGES TO COMMITMENTS MADE TO THE NUCLEAR REGULATORY COMMISSION

Calvert Cliffs has changed the following commitments made to the Nuclear Regulatory Commission (NRC):

Commitment: This engineering test procedure was developed on the basis of vendor recommendations for detecting degradation of neutron-absorbing materials. This program is designed to permit samples of the materials used in the spent fuel pool (SFP) storage racks to be periodically removed from the SFP for examination. Through specific positioning of designated sample packets, both accelerated and long term exposure to gamma radiation and borated water is provided. Sufficient samples are available so that the principal physical properties (i.e., sample weight for the Carborundum material and sample hardness for the Boraflex material) can be determined as a function of exposure on a regularly scheduled basis. Visual condition is assessed on a graded scale, and the results of physical property analyses are compared to historical results. Unacceptable results are documented, reported, and corrected in accordance with the Calvert Cliffs Nuclear Power Plant Corrective Action Program. The program will be modified to: (a) reevaluate the adequacy of the sampling intervals in monitoring Carborundum and Boraflex condition through the period of extended operation; and (b) refine the process for scheduling the sample packet removal from the SFP.

Change: This engineering test procedure was developed on the basis of vendor recommendations for detecting degradation of neutron-absorbing materials. This program is designed to permit samples of the materials used in the SFP storage racks to be periodically removed from the SFP for examination. Through specific positioning of designated sample packets, both accelerated and long-term exposure to gamma radiation and borated water is provided. Sufficient samples are available so that the principal physical properties (i.e., sample weight for the Carborundum material) can be determined as a function of exposure on a regularly scheduled basis. Visual condition is assessed on a graded scale, and the results of physical property analyses are compared to historical results. Unacceptable results are documented, reported, and corrected in accordance with the CCNPP Corrective Actions Program. The program will be modified to: (b) refine the process for scheduling sample packet removal from the SFP.

Justification: We no longer require an aging management program to monitor the condition (loss) of Boraflex in the Unit 2 SFP racks. Boraflex is no longer required to be monitored because it is no longer credited in the Unit 2 SFP Criticality Analysis. The NRC is already aware of this change because it is described in License Amendment No. 246 (Unit 2). The long term coupon surveillance program for Carborundum was submitted to the NRC and approved in Unit 1 License Amendment No. 288.

Commitment: A software upgrade will be installed during the 2008 Refueling Outage to allow the observation of Linear Variable Differential Transformer Field input values and failures.

Change: A software update will be installed during the 2010 Refueling Outage to allow the observation of Linear Variable Differential Transformer Field input values and failures.

Justification: GE was unable to meet the original Deadline to implement the upgrade during the 2008 Refueling Outage.

Commitment: The Fatigue Monitoring Program (FMP) will also assess the effect of the environment using statistical correlations developed by ANL in NUREG/CR-5704. The modified FMP will use the ANL statistical correlations to calculate an effective environmental factor to account for the reduction in fatigue life due to the reactor water environment. This factor will be applied to fatigue loads when the specified threshold criteria for strain rate and temperature have been exceeded. A factor of 1.5 will be used for evaluation of austenitic stainless steel components.

Change: The FMP will also assess the effect of the environment using statistical correlations developed by ANL in NUREG/CR-5704. The modified FMP will use the ANL statistical correlations to calculate an effective environmental factor to account for the reduction in fatigue life due to the reactor water environment. This factor will be applied to fatigue loads when the specified threshold criteria for strain rate and temperature have been exceeded.

Justification: The existing license renewal commitment contains a contradiction with regards to treatment of environmentally assisted fatigue. It first directs that an effective environmental fatigue factor be computed using statistical correlations developed in NUREG/CR-5704. It then states that a value of 1.5 be used for that factor. Originally the industry and NRC believed that a 1.5 factor would be appropriate, but later decided that a component specific evaluation was necessary. The existing commitment contains both versions. Eliminating the 1.5 factor requirement removes this contradiction, and puts us in alignment with latest NRC expectations.

Commitment: A new program will be developed to manage the effects of thermal embrittlement by identifying those components that may be susceptible to the effects of thermal embrittlement. The Cast Austenitic Stainless Steel (CASS) Evaluation Program will: (1) screen components; (2) review operating experience; (3) utilize enhanced VT-1 inspection [a visual examination capable of 1/2 mil resolution] [for Reactor Vessel Internals (RVI) only]; and (4) follow industry programs to evaluate thermal embrittlement and change the program accordingly.

Susceptibility of individual components to thermal embrittlement will be determined based upon the delta ferrite content of the component, the casting method (static or centrifugal), and the molybdenum content. When delta ferrite content is not documented for a component, it will be estimated using actual material data and Hull's equivalent factors. For components that fail the screening and are deemed susceptible to thermal aging embrittlement, the preferred alternative will be a flaw tolerance evaluation and augmented inspection. Fracture toughness properties used for the flaw tolerance evaluation will be estimated using the method in NUREG/CR-6177, "Assessment of Thermal Embrittlement of Cast Stainless Steels," and/or NUREG/CR-4513, Revision 1, "Estimation of Fracture Toughness of Cast Stainless Steels During Thermal Aging in LWR Systems." The intent of the analysis will be to determine if the respective component has adequate fracture toughness, based on its material properties, in order to be capable of performing its pressure boundary function under current licensing basis (CLB) conditions. A second alternative will be to replace the components. The second alternative will be used if a component cannot be qualified for the license renewal term by flaw tolerance analysis, or leak-before-break analysis, or if it is more cost effective to replace rather than perform an analysis. Replacement of the component will make the age-related degradation mechanism (ARDM) non-plausible for the respective component. The corrective actions taken as part of the CASS Evaluation Program will ensure that the CASS components remain capable of performing their pressure boundary function under all CLB conditions.

If the components do not meet the screening criteria described above, then:

1. There will be an augmented inspection combined with a flaw tolerance evaluation; or

2. A full leak-before-break evaluation will be performed to prove that current inspection requirements are adequate to prevent catastrophic failure; or

3. They will be replaced.

If an augmented inspection flaw tolerance approach is chosen, the acceptable flaw size for the inspection will be determined as follows:

- For non-niobium containing components having less than 25% delta ferrite, a component specific J-R curve will be generated. If a component contains nobium or 25% or greater delta ferrite, the actual fracture toughness properties will be determined on a case-by-case basis.

- An elastic-plastic fracture mechanics analysis will be performed for the component to determine the critical flaw size that is stable under all anticipated normal and accident loadings. This analysis may be component specific, or an analysis that bounds a group of components may be referenced.

- The critical flaw size will be used to determine the inspection acceptance criteria. The critical flaw size minus an allowance for flaw growth during operation until the next inspection will equal the allowable flaw size.

If available inspection technology permits, the inspection for Reactor Coolant System components will be conducted using a volumetric examination appropriate to a pressure-retaining weld on an American Society of Mechanical Engineers (ASME) Section XI category B-L-1, B-M-1, or BJ component. If available inspection technology does not permit a volumetric examination, an alternative approach similar to that described in Code Case N-481 will be used.

Change: A new program will be developed to manage the effects of thermal embrittlement by identifying those components that may be susceptible to the effects of thermal embrittlement. The CASS Evaluation Program will: (1) screen components; (2) review operating experience; (3) utilize either a volumetric examination or an EVT-1 visual examination; and (4) follow industry programs to evaluate thermal embrittlement and change the program accordingly.

Susceptibility of individual components to thermal embrittlement will be determined based upon the delta ferrite content of the component, the casting method (static or centrifugal), and the molybdenum content. Delta ferrite content will be determined using Hull's equation. For components that fail the screening and are deemed susceptible to thermal aging embrittlement, the preferred alternative will be either a volumetric examination or an EVT-1 visual examination. A second alternative will be to replace the components. The second alternative will be used if a component cannot be qualified for the license renewal term by either a volumetric examination or an EVT-1 visual examination, or if it is more cost effective to replace rather than perform either a volumetric examination or an EVT-1 visual examination. Replacement of the component will make the ARDM non-plausible for the respective component. The corrective actions taken as part of the CASS Evaluation Program will ensure that the CASS components remain capable of performing their pressure boundary function under all CLB conditions.

If the components do not meet the screening criteria described above, then:

1. There will be a volumetric inspection, if a suitable method is available; or

2. An EVT-1 visual examination will be performed, if a suitable volumetric inspection method is not available; or

3. They will be replaced.

If available inspection technology permits, the inspection for RCS components will be conducted using a volumetric examination appropriate to a pressure-retaining weld on an ASME Section XI category B-L-1, B-M-1, or BJ component. If available inspection technology does not permit a volumetric examination, an alternative approach similar to that described in Code Case N-481 will be used.

Justification: Part of the existing commitment to establish a CASS program is to follow industry programs that evaluate thermal embrittlement and change the CASS program accordingly. Since this commitment was made, the GALL report was issued which contains GALL AMP [Aging Management Program] XI.M12. This AMP provides NRC recommendations for management of thermal embrittlement of CASS components. This revised commitment aligns the Calvert Cliffs program with the current NRC guidance.

The existing commitment requires an augmented inspection combined with a flaw tolerance evaluation for susceptible components. The GALL report recommends either enhanced volumetric inspections or plant-specific or component-specific flaw tolerance evaluations. At the time of the original commitment there were no viable volumetric inspection methods for CASS components. However, recent advances in volumetric inspections indicate that there is a potential for detecting and sizing flaws in CASS components. A flaw tolerance evaluation is not necessary if flaws can be accurately detected and sized. It is anticipated that volumetric inspection methods could be available for examination of the CASS components during the next inservice inspection (ISI) examinations, scheduled for the 2018 and 2019 outages. This would preclude the need for a leak before break analysis. If a volumetric inspection method is not available by the 2018 or 2019 outages, the susceptible CASS components will be subjected to an EVT-1 examination.

Commitment: The Age-Related Degradation Inspection (ARDI) Program is credited for detection and management of the effects of ARDMs for which analysis is not able to demonstrate that an ARDM would not affect the intended function of the components during the period of extended operation. See a discussion of the ARDI Program in Section 4.1

The delta ferrite content will be determined for Calvert Cliffs RVI components made from CASS, and the results will be compared to the acceptable thresholds. Initial investigations revealed that formal calculations should show delta ferrite levels are below the established thresholds for these components. The new calculations are expected to show that thermal aging is not plausible and would not affect the intended function of these components during the period of extended operation. If the new calculations show that thermal aging is plausible, then these components would be managed by the CASS Evaluation Program. This program is described in Section 4.1

These programs are credited with discovering the effects of irradiation assisted stress corrosion cracking on the CEA shroud and bolts capscrews by determining if they have failed by using visual and core scanning techniques. The Preventive Maintenance Program is described in Section 3.1.

Calvert Cliffs will continue to develop data through industry research to determine the susceptibility of RVI components to neutron embrittlement. Until the data and analyses become available confirming that neutron embrittlement should not be considered a plausible ARDM, Calvert Cliffs will perform enhanced VT-1 inspections to detect cracks (if any occur) in the RVI components believed to be potentially most susceptible to neutron embrittlement. The inspections will be performed as part of the 10-year ISI program during the license renewal term. (The ISI Program is described in Section 3.1.) Plant-specific justification will be provided to the NRC in the event the analyses and data support elimination of the inspection. Appropriate revisions to the ISI Program will be made to address the scope, methodology, detection, and acceptance criteria for these new inspections. A fatigue analysis will be performed to show that the stress ranges and expected number of transients for these components will be low enough that thermal fatigue will not impair their intended function during the period of extended operation. The ASME Code's fatigue design rules will be used to demonstrate that the effects of fatigue can be managed adequately for the RVI components. Based on the service loadings for these components, the analysis is expected to show that the fatigue usage factor will be sufficiently low (0.5 or less) and that no further evaluations will be required for the period of extended operation. However, if the analysis shows a cumulative usage factor greater than 0.5 for any specific components, then further evaluations will be performed. For each such component, the evaluation will either provide justification that the component is bounded by other component(s) already monitored in the FMP, or if not bounded, then the specific components will be added to the FMP.

Because the control element assembly (CEA) shroud bolts and core shroud tie rod and bolts (tie rods, nuts, and set screws) are preloaded during initial installation, stress relaxation could affect their structural support function as a loss of preload, which could lead to vibrations and accelerated mechanical fatigue, resulting in cracking. For each of these types of components, an evaluation will be conducted to demonstrate that this ARDM will not occur for the stress levels and radiation conditions. Combustion Engineering's evaluation for stress corrosion for all Combustion Engineering plants will be further developed for stress relaxation of the CEA shroud bolts in the Calvert Cliffs RVI. A plant-specific analysis will be performed to refine the calculated stress levels on these bolts and demonstrate that they are not subject to substantial tensile stress during normal operations, and, thus, would not be subject to loss of preload from stress relaxation at pressurized water reactor operating temperatures. If the analysis does not show the low stress levels expected, an examination of the CEA shroud bolts would be conducted as part of an ARDI Program (see Section 4.1). For the other device type in this group, the core shroud tie rod and bolts, an analysis will be conducted to address the tensile preloads and opposing forces acting on these components during operation. The analysis is expected to demonstrate that the fluence levels and/or stress levels are not sufficient for stress relaxation to occur to the extent where the intended function would be impaired during the period of extended operationx. If the analysis does not show acceptably low stress levels, an examination of the tie rods would be conducted as part of an ARDI Program. An examination developed as part of the ARDI Program would be needed because the tie rods are located within the core shroud and, thus, are not accessible for visual examination.

Change: A new aging management program will be developed to manage the aging effects applicable to RVI components following the recommendations of Section XI.M16A of the GALL report, Revision 2 and Materials Reliability Program (MRP) 227-A. Calvert Cliffs will continue to participate in industry programs through the MRP. Under the guidance of Nuclear Energy Institute 03-08, Calvert Cliffs will incorporate the recommendations for additional inspections and evaluations provided by industry guidelines. These additional industry evaluations and recommendations are documented in MRP-227-A, "PWR Internals Inspection and Evaluation Guidelines." The RVI Program will provide plans for inspection, acceptance criteria and corrective actions.

Justification: Part of the existing commitment is to follow industry programs that evaluate degradation of the reactor internals components and change the program accordingly. GALL AMP XI.M16, Revision 1 through GALL VI recommends (1) participation in industry programs for investigating and managing aging effects of reactor internals; (2) evaluation and implementation of the results of the industry programs as applicable to RVIs; and (3) upon completion of these programs, but not less than 24 months before entering the period of extended operation, submittal of an inspection plan for reactor internals to the NRC for review and approval. The MRP has developed pressurized water reactor internals inspection and evaluation guidelines that are found in MRP-227-A. Materials Reliability Program evaluates all aging effects and aging mechanisms addressed the license renewal application and the associated Safety Evaluation Report (NUREG-1705).