[7590-01-P]

NUCLEAR REGULATORY COMMISSION

[Docket Nos. 50-368, 50-334, 50-445, 50-302, 50-348, 50-364, 50-336, 50-338, 50-339, 50-282, 50-306, 50-327, 50-498, 50-499, 50-335, 50-280, 50-395, 50-390; NRC-2017-0188]

Entergy Operations, Inc.; FirstEnergy Nuclear Operating Company;

Vistra Operations Company, LLC;

Duke Entergy Florida, Southern Nuclear Operating Company, Inc.;

Dominion Nuclear Connecticut, Inc.; Virginia Electric and Power Company;

Northern States Power Company – Minnesota;

South Carolina Electric & Gas Company, Inc.; STP Nuclear Operating Company;

Tennessee Valley Authority

AGENCY: Nuclear Regulatory Commission.

ACTION: Director’s decision under 10 CFR 2.206; issuance.

SUMMARY: The U.S. Nuclear Regulatory Commission (NRC) has issued a director’s decision in response to a petition dated January 24, 2017, filed by Mr. Paul Gunter on behalf of Beyond Nuclear, and representing numerous public interest groups (collectively, Beyond Nuclear, et al., or petitioners), requesting that the NRC take action with regard to licensees of plants that currently rely on potentially defective safety-related components and potentially falsified quality assurance documentation supplied by AREVA-Le Creusot Forge and Japan Casting and Forging Corporation. The

petitioners' requests are included in the **SUPPLEMENTARY INFORMATION** section of this document.

DATES: The director's decision was issued on August 2, 2018.

ADDRESSES: Please refer to Docket ID **NRC-2017-0188** when contacting the NRC about the availability of information regarding this document. You may obtain publicly-available information related to this document using any of the following methods:

- **Federal Rulemaking Web Site:** Go to http://www.regulations.gov and search for Docket ID **NRC-2017-0188**. Address questions about NRC dockets to Jennifer Borges; telephone: 301-287-9127; e-mail: Jennifer.Borges@nrc.gov. For technical questions, contact the individual listed in the **FOR FURTHER INFORMATION CONTACT** section of this document.

- **NRC's Agencywide Documents Access and Management System (ADAMS):** You may obtain publicly-available documents online in the ADAMS Public Documents collection at http://www.nrc.gov/reading-rm/adams.html. To begin the search, select "Begin Web-based ADAMS Search." For problems with ADAMS, please contact the NRC's Public Document Room (PDR) reference staff at 1-800-397-4209, 301-415-4737, or by e-mail to pdr.resource@nrc.gov. The ADAMS accession number for each document referenced (if it is available in ADAMS) is provided the first time that it is mentioned in this document.

- **NRC's PDR:** You may examine and purchase copies of public documents at the NRC's PDR, Room O1-F21, One White Flint North, 11555 Rockville Pike, Rockville, Maryland 20852.

FOR FURTHER INFORMATION CONTACT: Perry Buckberg, Office of Nuclear Reactor Regulation, U.S. Nuclear Regulatory Commission, Washington DC 20555-0001; telephone: 301-415-1383; e-mail: Perry.Buckberg@nrc.gov.

SUPPLEMENTARY INFORMATION: The text of the director's decision is attached.

Dated at Rockville, Maryland, this 7th day of August, 2018.

For the Nuclear Regulatory Commission.

/RA/

Perry H. Buckberg, Senior Project Manager,
Special Projects and Process Branch,
Division of Operating Reactor Licensing,
Office of Nuclear Reactor Regulation.

Attachment – Director's Decision DD-18-03

UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION
OFFICE OF NUCLEAR REACTOR REGULATION

Brian E. Holian, Acting Director

In the Matter of Power Reactor Licensees

Docket Nos.: See Attached List

License Nos.: See Attached List

DIRECTOR'S DECISION UNDER 10 CFR 2.206

I. Introduction

On January 24, 2017,[1] Mr. Paul Gunter submitted a petition on behalf of Beyond Nuclear that represents numerous public interest groups (collectively referred to as the Petitioners) under Title 10 of the *Code of Federal Regulations* (10 CFR) 2.206, "Requests for Action under This Subpart." The Petitioners supplemented their petition by e-mails dated February 16,[2] March 6,[3,4] June 16,[5] June 22,[6] June 27,[7] June 30,[8] and July 5, 2017.[9] The June 16 and June 22, 2017, supplements added the Crystal River Unit 3 Nuclear Generating Plant (Crystal River Unit 3) to the list of plants subject to the petition and requested slightly different enforcement actions. The rest of the supplements did not expand the scope of the petition or request additional actions that

1 See Agencywide Documents Access and Management System (ADAMS) Accession No. ML17025A180.
2 See ADAMS Accession No. ML17052A032.
3 See ADAMS Accession No. ML17068A061.
4 See ADAMS Accession No. ML17067A562.
5 See ADAMS Accession No. ML17174A087.
6 See ADAMS Accession No. ML17174A788.
7 See ADAMS Accession No. ML17179A288.
8 See ADAMS Accession No. ML17184A058.
9 See ADAMS Accession No. ML17187A026.

should be considered as a new petition. The Petitioners asked the U.S. Nuclear Regulatory Commission (NRC) to take emergency enforcement action at U.S. nuclear power plants that currently rely on potentially defective safety-related components and potentially falsified quality assurance documentation supplied by AREVA-Le Creusot Forge (ACF) and its subcontractor, Japan Casting and Forging Corporation (JCFC).[10] Table 1 lists potentially affected components and the at-risk reactors identified in the petition.

Table 1. List of Potentially Affected Components and Reactors

Reactor Pressure Vessels	Replacement Reactor Pressure Vessel Heads	Steam Generators	Steam Pressurizers
Prairie Island, Units 1 and 2 (MN)	Arkansas Nuclear One, Unit 2 (AR) Beaver Valley, Unit 1 (PA) North Anna, Units 1 and 2 (VA) Surry, Unit 1 (VA) Crystal River, Unit 3 (FL)	Beaver Valley, Unit 1 (PA) Comanche Peak, Unit 1 (TX) V.C. Summer (SC) Farley, Units 1 and 2 (AL) South Texas, Units 1 and 2 (TX) Sequoyah, Unit 1 (TN) Watts Bar, Unit 1 (TN)	Millstone, Unit 2 (CT) Saint Lucie, Unit 1 (FL)

10 The petition incorrectly states that JCFC is a subcontractor to ACF.

Specifically, the Petitioners asked the NRC to take enforcement actions consistent with the following:

1. Suspend power operations of U.S. nuclear power plants that rely on ACF components and subcontractors pending a full inspection (including nondestructive examination by ultrasonic testing) and material testing. If carbon anomalies ("carbon segregation" or "carbon macrosegregation" (CMAC)) in excess of the design-basis specifications for at-risk component parts are identified, require the licensee to do one of the following:

    a. Replace the degraded at-risk component(s) with quality-certified components.

    b. For those at-risk degraded components that a licensee seeks to allow to remain in service, apply through the license amendment request process to demonstrate that a revised design basis is achievable and will not render the inservice component unacceptably vulnerable to fast fracture failure at any time and in any credible service condition throughout the current license of the power reactor.

2. Alternatively modify the licensees' operating licenses to require the licensees to perform the requested emergency enforcement actions at the next scheduled outage.

3. Issue a letter to all U.S. light-water reactor operators under 10 CFR 50.54(f) requiring licensees to provide the NRC with information under oath and affirming specifically how U.S. operators are reliably monitoring contractors and subcontractors for the potential carbon segmentation anomaly in the supply chain and the reliability of the quality assurance certification of those components, and publicly release the responses.

The June 16 and June 22, 2017, supplements to the petitions added Crystal River Unit 3, which is currently shut down, and the licensee Duke Energy to the list of facilities for which the Petitioners requested the following fourth NRC action:

a. Confirm the sale, delivery, quality control and quality assurance certification and installation of the replacement reactor pressure vessel head as supplied to Crystal River Unit 3 by then Framatome and now AREVA-Le Creusot Forge industrial facility in Charlon-St. Marcel, France and;

b. With completion and confirmation [of the above Crystal River Unit 3 actions], the modification of Duke Energy's current license for the permanently closed Crystal River Unit 3 nuclear power station in Crystal River, Florida, to inspect and conduct the appropriate material test(s) for carbon macrosegregation on sufficient samples harvested from the installed and now inservice irradiated Le Creusot Forge reactor pressure vessel head [sic]. The Petitioners assert that the appropriate material testing include Optical Emissions Spectrometry (OES).

As the basis of their requests, the Petitioners cited the expert review by Large and Associates Consulting Engineers that identified significant irregularities and anomalies in both the manufacturing process and quality assurance documentation of large reactor components manufactured by the ACF for French reactors and reactors in other countries.[11]

On February 2, 2017,[12] the Office of Nuclear Reactor Regulation (NRR) petition manager acknowledged receipt of the petition and offered an opportunity for the Petitioners to address NRR's 10 CFR 2.206 Petition Review Board (PRB) to discuss the petition. The Petitioners accepted the offer, and the meeting was held on March 8, 2017. The transcript[13] of that meeting is publicly available.

On February 8, 2017, the PRB met internally to discuss the request for immediate actions and informed the Petitioners on February 13, 2017,[14] that no actions were warranted at that time because the NRC has reasonable assurance of public health and safety and protection of the environment. The basis for the PRB's determination included the following:

- **<u>Extent of Condition</u>**. Internationally, CMAC has been found only in components produced by ACF using a specific processing route. Based on the staff's knowledge as of February 2017, only a subset of the plants identified in the

11 See the report titled "Irregularities and Anomalies Relating to the Forged Components of Le Creusot Forge," dated September 26, 2016, Large and Associates Consulting Engineers, London, England (available at http://www.largeassociates.com/CZ3233/Note_LargeAndAssociates_EN_26092016.pdf).

12 See ADAMS Accession No. ML17039A501.

13 See ADAMS Accession No. ML17081A418.

14 See ADAMS Accession No. ML17052A033.

petition contain components that may have used the processing route that resulted in the excess CMAC found in international plants.

- **<u>Degree of Condition</u>**. If CMAC is present in a component, it occurs in a localized region of the forged component. It is not a bulk material phenomenon, does not go through thickness, and is not expected to affect the structural integrity of the component. In addition, based on the staff's knowledge as of February 2017, the highest levels of CMAC observed internationally, if present in the postulated regions of U.S. components, are not expected to alter the mechanical properties of the material enough to affect the structural integrity of the components. Destructive examinations of components containing regions of CMAC have been conducted internationally to determine how CMAC affects mechanical properties and such examinations confirm that structural integrity has not been impacted. A summary of the international investigation is summarized in II.A below, and details of the investigation and its impact on structural integrity are described in the staff's evaluation dated February 22, 2018.[15]

- **<u>Safety Significance</u>**. The staff's preliminary safety assessment concluded that the safety significance of CMAC to the U.S. nuclear power reactor fleet appears to be negligible. The staff based its assessment on knowledge of the material processing, qualitative analysis, compliance of U.S. components with the American Society of Mechanical Engineers *Boiler Pressure and Vessel Code* (ASME Code), and the results of preliminary structural evaluations. The NRC

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15 See ADAMS Accession No. ML18017A441.

subsequently presented the basis for this determination in a technical session, titled "Carbon Macrosegregation in Large Nuclear Forgings," at the NRC-sponsored Regulatory Information Conference on March 15, 2017.[16,17]

On April 11, 2017, the PRB met to discuss the petition with respect to the criteria for consideration under 10 CFR 2.206. Based on that review, the PRB determined that the petition request meets the criteria for consideration under 10 CFR 2.206. On May 19, 2017, the petition manager informed the Petitioners that the initial recommendation was to accept the petition for review but to refer a portion of the petition (i.e., the concern of potentially falsified quality assurance documentation) to the NRC's allegation process for appropriate action.[18] The petition manager also offered the Petitioners an opportunity to comment on the PRB's recommendations. On July 5, 2017, the petition manager clarified the initial recommendation and asked for a response as to whether the Petitioners wanted to address the PRB a second time to comment on its recommendations. The Petitioners did not request a second opportunity to address the PRB. Therefore, the PRB's initial recommendations to accept part of the petition for review under 10 CFR 2.206 and to refer a part to another NRC process became final. On August 30, 2017, the petition manager issued an acknowledgment letter to the Petitioners.[19]

By a letter to the Petitioners which copied the licensees dated June 6, 2018,[20] the NRC issued the proposed director's decision for comment. The Petitioners were asked

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16 See ADAMS Accession No. ML17171A108.
17 See ADAMS Accession No. ML17171A106.
18 See ADAMS Accession No. ML17142A334.
19 See ADAMS Accession No. ML17198A329.
20 See ADAMS Accession No. ML18107A402.

to provide comments within 14 days on any part of the proposed director's decision considered to be erroneous or any issues in the petition that were not addressed. The NRC staff did not receive any comments on the proposed director's decision.

The petition and other references related to this petition are available for inspection in the NRC's Public Document Room (PDR), located at O1F21, 11555 Rockville Pike (first floor), Rockville, MD 20852. Publicly available documents created or received at the NRC are accessible electronically through ADAMS in the NRC Library at http://www.nrc.gov/reading-rm/adams.html. Persons who do not have access to ADAMS or who encounter problems in accessing the documents located in ADAMS should contact the NRC's PDR reference staff by telephone at 1-800-397-4209 or 301-415-4737 or by e-mail to pdr.resource@nrc.gov.

II. Discussion

Under the 10 CFR 2.206(b) petition review process, the Director of the NRC office with responsibility for the subject matter shall either institute the requested proceeding or shall advise the person who made the request in writing that no proceeding will be instituted, in whole or in part, with respect to the request and the reason for the decision. Accordingly, the decision of the NRR Director is provided below. As further discussed below, the petition is denied.

The NRC's policy is to have an effectively coordinated program to promptly and systematically review relevant domestic and applicable international operational experience (OpE) information. The program supplies the means for assessing the significance of OpE information, offering timely and effective communication to stakeholders, and applying the lessons learned to regulatory decisions and programs affecting nuclear reactors. The NRC Management Directive 8.7, "Reactor Operating

Experience Program," dated February 1, 2018, describes the Reactor OpE Program.[21] The NRR Office Instruction (OI) LIC-401, "NRR-NRO Reactor Operating Experience Program," Revision 3, addresses the specific implementation of the Reactor OpE Program.[22]

As reported in internal NRC communications, AREVA notified France's nuclear safety authority, Autorité de Sûreté Nucléaire (ASN), of an anomaly in the composition of the steel in certain zones of the reactor pressure vessel (RPV) upper and lower heads of the Flamanville Nuclear Power Plant (Flamanville), Unit 3, in Manche, France. Both the upper and lower vessel heads were manufactured by ACF. According to ASN, chemical and mechanical property testing performed by AREVA in late 2014 (on a vessel head similar to that of the Flamanville European Pressurized Reactor (EPR)) revealed a zone of high carbon concentration (0.30 percent as opposed to a target value of 0.22 percent), which led to lower than expected mechanical toughness values in that area. Initial measurements confirmed the presence of this anomaly in the Flamanville, Unit 3, RPV upper and bottom heads.

In accordance with the process described in NRR OI LIC-401, the NRC's Reactor OpE Program staff ensured that the appropriate technical experts within the NRC were aware of the issue and were evaluating these issues for relevance to the U.S. industry. In addition, the NRC has strong collaboration with the international community and was separately in contact with ASN to discuss this issue.

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21 See ADAMS Accession No. ML18012A156.

22 See ADAMS Accession No. ML12192A058.

A. Description of the Issue

The CMAC is a known phenomenon that takes place during the casting of large ingots. The CMAC is a material heterogeneity in the form of a chemical (i.e., carbon) gradient that deviates from the nominal composition and may exceed specification limits. Portions of the ingot containing CMAC that exceed specification limits (positive CMAC) are purposefully removed and discarded as part of the material processing. Regions of positive CMAC that are not appropriately removed result in localized regions near the surface of the final component with higher strength and lower toughness relative to the bulk material.

In April 2015, regions of positive CMAC were discovered in EPR RPV heads that were manufactured for the Flamanville plant. The ACF had produced the forgings for the Flamanville upper and lower RPV heads. The discovery of the CMAC in the heads prompted ASN to ask the operator, Électricité de France S.A. (EDF) (Electricity of France), to review inservice forged components at all of its plants to determine the potential extent of the condition. The review identified steam generator (SG) channel heads (also commonly referred to as SG primary heads) produced by ACF and JCFC as the components most likely to contain a region of CMAC. The ASN requested that nondestructive testing be performed on these SG channel heads to characterize the carbon content and confirm the absence of unacceptable flaws.

On October 18, 2016, ASN ordered the acceleration of the nondestructive testing of the potentially affected ACF and JCFC SG channel heads, which required completion of the remaining nondestructive testing within 3 months. The discovery of higher than expected carbon values measured on an inservice SG channel head produced by JCFC

prompted the accelerated schedule. As a result, to perform the required nondestructive tests, EDF had to shut down its plants before their scheduled outages.

AREVA Inc. (AREVA Inc. or AREVA), located in Lynchburg, VA, provides safety-related products and services for U.S. operating nuclear power plants, including replacements for reactor coolant pressure boundary components. On February 3, 2017,[23] AREVA Inc. submitted a list to the NRC of the U.S. reactors that have received components fabricated with forgings from ACF. Operating U.S. plants have no known components from JCFC.

In September 2015, June 2016, and June 2017, ASN convened an Advisory Committee of Experts for Nuclear Pressure Equipment to obtain its technical opinion on the consequences of CMAC for the serviceability of the Flamanville EPR reactor vessel domes. The resulting series of publicly available reports (CODEP-DEP-2015-037971,[24] CODEP-DEP-2016-019209,[25] and CODEP-DEP-2017-019368[26]) justified the continued use of the Flamanville heads. In this effort, AREVA conducted hundreds of mechanical and chemical property experiments on three full-scale replica heads that were manufactured by ACF using the same process as that used for the Flamanville heads. Using these

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23 See ADAMS Accession No. ML17040A100.

24 See ASN/Institut de Radioprotection et de Sûreté Nucléaire (**IRSN**) (Radioprotection and Nuclear Safety Institute) report CODEP-DEP-2015-037971, “Analysis of the Procedure Proposed by AREVA to Prove Adequate Toughness of the Dome of the Flamanville 3 EPR Reactor Pressure Vessel Lower Head and Closure Head,” English translation, dated September 16, 2015. http://www.french-nuclear-safety.fr/Media/Files/00-Publications/Report-to-the-Advisory-Committee-of-Experts-for-Nuclear-Pressure-Equipment.

25 See ASN/IRSN report CODEP-DEP-2016-019209, “Procedure Proposed by AREVA to Prove Adequate Toughness of the Domes of the Flamanville 3 EPR Reactor Pressure Vessel Bottom Head and Closure Head,” English translation, dated June 17, 2016. https://www.asn.fr/content/download/106732/811356/ version/6/file/CODEP-DEP-2016-019209-advisorycommitte24june2016-summaryreport.pdf.

26 See ASN/IRSN report CODEP-DEP-2017-019368, “Analysis of the Consequences of the Anomaly in the Flamanville EPR Reactor Pressure Vessel Head Domes on Their Serviceability,” English translation, dated June 15, 2017. http://www.irsn.fr/FR/expertise/rapports_gp/Documents/GPESPN/IRSN-ASNDEP_GPESPN-Report_pressure-vessel-FA3_201706.pdf.

experimental results, AREVA conducted a variety of code-related fracture and strength analyses that demonstrated that the risk of fast fracture from CMAC was extremely low. Through this effort, ASN concluded that the serviceability of the heads is acceptable as long as EDF conducts the required inservice inspections. However, because of its inability to conduct an adequate inservice inspection on the Flamanville upper head, ASN concluded that the upper head long-term serviceability could not be confirmed and that the head should be replaced after a few years of operation.

B. Initial Actions by the NRC and the U.S. Nuclear Industry

Beginning in December 2016, the NRC staff conducted a preliminary safety assessment to determine the potential safety significance posed to the U.S. nuclear power reactor fleet by the CMAC observed in reactor coolant system (RCS) components overseas and concluded that the failure of an RPV/SG head component has a very low probability, even if the worst practical degree of CMAC occurs within that component. The NRC staff used a qualitative failure comparison to assess the relative likelihood of failure of an RPV shell (which is not expected to be subject to positive CMAC) with RPV/SG head component types that could be affected by CMAC. Based on this comparison, the NRC determined the following:

- The RPV shell experiences higher stresses under both normal operations and postulated accident scenarios.

- The weld region of an RPV shell has a greater likelihood of having more flaws and larger fabrication flaws. The larger fabrication flaws typically have the higher potential to result in component failure.

- Although the initial toughness of an RPV shell material may be greater than an RPV/SG head with postulated positive CMAC, the shell toughness decreases as the result of radiation embrittlement after several years of operation. As a result, the current as-operated toughness of RPV shell material is expected to be lower than the toughness of RPV/SG head material with postulated CMAC. The RPV shell material is known to have adequate toughness for safe operation.

When combining all these individual attributes, an RPV/SG head component with postulated CMAC is much less likely to fail than an RPV shell. Past research and operating experience has demonstrated that failure of an RPV shell under normal operations or postulated accident scenarios has a very low probability of occurrence.[27,28] Therefore, the failure of an RPV/SG head component also has a very low probability, even if the worst practical degree of CMAC occurs within that component. The NRC presented the basis for this preliminary determination in a technical session titled "Carbon Macrosegregation in Large Nuclear Forgings" (cited above) at the March 15, 2017, NRC-sponsored Regulatory Information Conference.

Concurrent with the NRC analyses, the U.S. industry initiated a research program in early 2017, conducted by the Electric Power Research Institute (EPRI), to address the generic safety significance of elevated carbon levels caused by CMAC in

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27 See ADAMS Accession No. ML072830076.

28 See ADAMS Accession No. ML072820691.

the components of interest. This program was divided into the following four main tasks, each aimed at developing both qualitative and quantitative information to make a safety determination:

1. extension of RPV probabilistic fracture mechanics (PFM) analyses to qualitatively bound other components

2. development of a robust technical basis to support the hypothesis that RPV integrity bounds other components

3. quantitative structural analyses to assess whether the results of the PFM analyses of the RPV beltline (Task 1) bound the other forged components

4. a white paper assessing the effect of CMAC on SG tubesheets based on expert judgment and experience with the fabrication of the tubesheets as large forgings

As of the writing of this document, Task 1 has been completed and has been publicly released as Materials Reliability Program (MRP)-417.[29] The other tasks are still under development with the expected release of the report(s) in 2018.

The MRP-417 addresses the structural significance of the potential presence of CMAC in large, forged pressurized-water reactor pressure-retaining components, including the RPV head, beltline and nozzle shell forgings, and the SG and pressurizer ring and head forgings through the end of an 80-year operating interval. The assessment was made using the NRC risk safety criterion of a 95th percentile

[29] EPRI Report No. 3002010331, "Materials Reliability Program: Evaluation of Risk from Carbon Macrosegregation in Reactor Pressure Vessels and Other Large Nuclear Forgings (MRP-417)," issued June 2017 (available at ADAMS Accession No. ML18054A862).

through-wall crack frequency (TWCF) of less than 1×10^{-6} per year (yr^{-1}) (10 CFR 50.61a, "Alternative Fracture Toughness Requirements for Protection against Pressurized Thermal Shock Events") for pressurized thermal shock (PTS) events and a conditional probability of failure (CPF) of less than 1×10^{-6} for normal operating transients. These analyses used many of the same assumptions and inputs as those used in the basis for the 10 CFR 50.61a alternate PTS rule.[30,31] In addition, the analysts approximated the effect of carbon content on the fracture toughness of the steel through a review of the available literature.

The MRP-417 describes the analyses and results for bounding values for the RPV shell, RPV upper head, SG channel head, pressurizer shell, and pressurizer head components based on the analyses assumptions from the alternate PTS rule in conjunction with the effect of the CMAC on the material toughness. The report's deterministic results suggest that the RPV vessel behavior bounds the behavior of the pressurizer components. In addition, the probabilistic results suggest that in all cases, assuming the maximum carbon content observed in the field, the calculated TWCF and CPF were below the NRC risk safety criterion of the 95th percentile TWCF of less than 1×10^{-6} yr^{-1} for PTS events and a CPF of less than 1×10^{-6} for normal operating transients. MRP-417 concludes that there is substantial margin against failure through an 80-year operating interval using the assumed CMAC distributions in the RPV, SG, and pressurizer rings and head forgings in pressurized-water reactors.

In March 2017, an NRC inspection team performed a limited-scope vendor inspection at the AREVA facility in Lynchburg, Virginia, to review documentation from

30 See ADAMS Accession No. ML072830076.
31 See ADAMS Accession No. ML072820691.

ACF and assess AREVA's compliance with the provisions of selected portions of Appendix B, "Quality Assurance Criteria for Nuclear Power Plants and Fuel Reprocessing Plants," to 10 CFR Part 50, and 10 CFR Part 21, "Reporting of Defects and Noncompliance." This inspection focused on AREVA's documentation and evaluation of potential carbon macrosegregation issues in forgings supplied by AREVA for U.S. operating nuclear power plants. Specifically, the NRC inspection reviewed documentation to verify that forgings met the ASME Code requirements for carbon content and mechanical properties. The NRC issued the inspection report on May 10, 2017.[32] The limited-scope inspection reviewed policies and procedures that govern implementation of AREVA's 10 CFR Part 21 program, and nonconformance and corrective action policies and procedures under its approved quality assurance program related to the manufacturing processes used by ACF to fabricate inservice U.S. components and the resulting mechanical properties. The NRC inspection team used Inspection Procedure (IP) 43002, "Routine Inspections of Nuclear Vendors,"[33] and IP 36100, "Inspection of 10 CFR Part 21 and Programs for Reporting Defects and Noncompliance."[34] The inspection team did not identify any violations or nonconformances during the inspection.

The inspection report contains the following primary material processing and property observations:

- A population of the components produced by ACF has a low or no possibility of containing regions of CMAC.

32 See ADAMS Accession No. ML17124A575.
33 See ADAMS Accession No. ML13148A361.
34 See ADAMS Accession No. ML113190538.

- Carbon levels and mechanical properties for the components reviewed conformed to ASME Code requirements.

- The information reviewed did not challenge the NRC's preliminary determination on the CMAC topic (i.e., that the safety significance to the U.S. nuclear power reactor fleet appears to be negligible).

The NRC staff also documented its risk-informed evaluation of the potential safety significance of CMAC in components produced by ACF, as it relates to the safe operation of U.S. plants, and options for addressing the topic using its risk-informed decision-making process in NRR OI LIC-504, "Integrated Risk-Informed Decision-Making Process for Emergent Issues," Revision 4, dated June 2, 2014,[35] to evaluate this issue.

C. Applicable NRC Regulatory Requirements and Guidance

The NRC requires U.S. nuclear reactor components fabricated with forgings from ACF to be manufactured and procured in accordance with all applicable regulations, as well as the ASME Code requirements that are incorporated by reference. The regulations most pertinent to the prevention and identification of CMAC in regions of RCS components are the ASME Code requirements incorporated by reference in 10 CFR 50.55a, "Codes and Standards," and quality assurance requirements in 10 CFR Part 50, Appendix B. In addition to the NRC regulations and ASME Code requirements that are focused on the process and quality controls for addressing CMAC, there are also regulations that focus on performance and design criteria that may be

[35] See ADAMS Accession No. ML14035A143.

impacted by regions of CMAC. These regulations include: 10 CFR 50.60, "Acceptance criteria for fracture prevention measures for lightwater nuclear power reactors for normal operation," Appendix A to 10 CFR Part 50, "General Design Criteria for Nuclear Power Plants," and Appendix G to 10 CFR Part 50, "Fracture Toughness Requirements." The applicability of specific NRC regulations and ASME Code requirements will, in part, depend on the dates that the regulations or requirements became effective relative to a component being put into operation. The plant-specific design basis and current licensing basis address the fundamental regulatory requirements pertaining to the integrity of the components of interest.

Appendix B to 10 CFR Part 50 establishes quality assurance requirements for the design, manufacture, construction, and operation of the structures, systems, and components (SSCs) for nuclear facilities. Appendix B requirements apply to all activities affecting the safety-related functions of those SSCs. These activities include designing, purchasing, fabricating, handling, installing, inspecting, testing, operating, maintaining, repairing, and modifying SSCs. To accomplish these activities, licensees must contractually pass down the requirements of Appendix B through procurement documentation to suppliers of SSCs, as stated in the Appendix B criteria below.

Criterion IV, "Procurement Document Control," of 10 CFR Part 50, Appendix B, states the following:

> Measures shall be established to assure that applicable regulatory requirements, design bases, and other requirements which are necessary to assure adequate quality are suitably included or referenced in the documents for procurement of material, equipment, and services, whether purchased by the applicant or by its contractors or subcontractors. To the

> extent necessary, procurement documents shall require contractors or subcontractors to provide a quality assurance program consistent with the pertinent provisions of this appendix.

Criterion VII, "Control of Purchased Material, Equipment, and Services," of 10 CFR Part 50, Appendix B, in part, states, the following:

> Documentary evidence that material and equipment conform to the procurement requirements shall be available at the nuclear power plant or fuel reprocessing plant site prior to installation or use of such material and equipment. This documentary evidence shall be retained at the nuclear power plant or fuel reprocessing plant site and shall be sufficient to identify the specific requirements, such as codes, standards, or specifications, met by the purchased material and equipment.

The licensee is responsible for ensuring that the procurement documentation appropriately identifies the applicable regulatory and technical requirements and for determining whether the purchased items conform to the procurement documentation.

Criterion XV, "Nonconforming Materials, Parts, or Components," of 10 CFR Part 50, Appendix B, states the following:

> Measures shall be established to control materials, parts, or components which do not conform to requirements in order to prevent their inadvertent use or installation. These measures shall include, as appropriate,

procedures for identification, documentation, segregation, disposition, and notification to affected organizations. Nonconforming items shall be reviewed and accepted, rejected, repaired or reworked in accordance with documented procedures.

Nonconformances identified by the supplier during manufacturing must be technically evaluated and dispositioned accordingly. If the supplier identifies a nonconformance, such as the presence of CMAC in the final product, it must perform an engineering evaluation and document the nonconformance on the associated certificate of conformance. The licensee is responsible for reviewing the certificate of conformance during receipt inspection for acceptance of the final product upon delivery.

Under 10 CFR Part 21, the NRC requires both licensees and their suppliers to evaluate any condition or defect in a component that could create a substantial safety hazard. Regions of CMAC in RCS components suspected of having the potential to create a substantial safety hazard would be an example of a condition that licensees and their suppliers must evaluate. In addition, 10 CFR Part 21 requires the entity to notify the NRC if it becomes aware of information that reasonably indicates that a basic component contains defects that could create substantial safety hazard.

D. Summary of the NRC's Evaluation

The NRC's evaluation of this issue consisted of conducting preliminary safety analyses as described above, reviewing the testing and analyses performed by the French licensee, meeting with French and Japanese regulators to discuss their evaluation, reviewing the nuclear industry's evaluation of the issue, conducting an onsite inspection of manufacturing and procurement records, and determining the final safety

assessment using a risk-informed decision-making process. The staff's evaluation dated February 22, 2018, documents the NRC's full evaluation of the CMAC topics as it relates to plants operating in the United States.

The staff reviewed the publicly available ASN documentation on this issue (CODEP-DEP-2015-037971, CODEP-DEP-2016-019209, and CODEP-DEP-2017-019368) and concluded that, although ASN's decisions and actions are based solely on French nuclear regulations which do not directly correlate to U.S. regulations, the experimental results and the fast fracture analyses can provide direct insight into the expected behavior of postulated CMAC in U.S.-forged components. As concluded by ASN, the analyses demonstrate that the fast fracture of the Flamanville heads from the impacts of CMAC can be ruled out in view of the margins determined by the analyses.

The NRC staff reviewed the technical information in MRP-417 and concluded that it was credible for use in this assessment for the following reasons:

- The risk criteria used for the CPF and 95th percentile TWCF were identical to those used in the development of 10 CFR 50.61a.

- Major probabilistic inputs, such as flaw distribution, standard material properties, transients, and normal operating conditions were identical to those used in the development of 10 CFR 50.61a.

- The CMAC distribution and toughness relationships used were based on historical literature and empirical data.

- The assumptions made using the computational model were consistent with, or were conservative as compared to those used in the analyses for the development of 10 CFR 50.61a.

The NRC assessment of MRP-417 for this report does not constitute a regulatory endorsement of its full contents. The NRC staff will assess the other industry reports on the CMAC topic in the same manner as such reports become available.

Although these evaluations provide useful information to address the impacts of postulated CMAC in forged components in service at U.S. operating reactors, the NRC staff used an analysis approach, leveraging existing PFM results and examining them in the context of the NRC's approach to the risk-informed decision-making process described in NRR OI LIC-504.

Consistent with LIC-504, for this review, the NRC staff considered the following five principles of risk-informed decision-making when considering options for addressing this issue:

- **Principle 1.** The proposed change must meet the current regulations unless it is explicitly related to a requested exemption or rule change.

- **Principle 2.** The proposed change shall be consistent with the defense-in-depth philosophy.

- **Principle 3.** The proposed change shall maintain sufficient safety margins.

- **<u>Principle 4</u>.** When the proposed change results in an increase in core damage frequency or risk, the increases should be small and consistent with the intent of the Commission's safety goals.

- **<u>Principle 5</u>.** Monitoring programs should be in place.

The NRC staff considered the following four options to address the potential impact of the international CMAC OpE on the U.S. nuclear power reactor fleet. Options 2, 3, and 4 align with the Petitioners' requests.

- Option 1: Evaluate and Monitor
- Option 2: Issue a Generic Communication
- Option 3: Issue Orders Requiring Inspections
- Option 4: Issue Orders Suspending Operation

Option 1

This option consists of the NRC staff continuing to monitor all domestic and international information associated with the CMAC topic. The staff will evaluate new information, as it becomes available, to ensure that conservatism in the staff's final safety determination is maintained. Aspects of the staff's safety determination that may be evaluated against new information includes the extent of condition in the U.S., potential degree of CMAC on a generic basis, or data affecting the relationship between CMAC and mechanical performance. This information is to be evaluated to determine if there is reasonable assurance that adequate defense-in-depth, sufficient safety margin,

and an acceptable level of risk is maintained with an appropriate degree of conservatism.

If new information becomes available that warrants evaluation and it is concluded that the staff's safety determination remain appropriately conservative, then no additional actions will be taken. Alternatively, if the staff cannot conclude that there is reasonable assurance of structural integrity, additional action(s) will be considered. The NRC will communicate with applicable stakeholders, as appropriate.

Option 2

The second option involves issuing a generic letter (GL) to the licensees operating with components forged by ACF. The objective of the GL would be to confirm that the licensees' 10 CFR Part 50, Appendix B, quality assurance programs have verified that the components produced by ACF comply with the applicable NRC regulations and ASME Code requirements. The GL would request that the licensees (1) provide the documentation necessary to confirm that the components in question meet all applicable NRC regulations and ASME Code requirements and (2) describe how their 10 CFR Part 50, Appendix B, quality assurance programs verified that the components complied with all applicable NRC regulations and ASME Code requirements, specifically, those related to the manufacturing of the components relevant to the CMAC topic. Section II.C of this Director's Decision provides the regulatory requirements and the 10 CFR Part 50, Appendix B, quality assurance program, as they relate to the CMAC topic. A GL can require a written response in accordance with 10 CFR 50.54(f).

Option 3

The third option involves issuing an order to the licensees operating with inservice components produced by ACF. The order would require licensees with components from ACF to conduct nondestructive examinations of these inservice components during the next scheduled outage. The objective of the examination would be to verify the condition of the components (e.g., no unacceptable flaw or indications) and to verify carbon levels. If the nondestructive examinations reveal a condition that is adverse to safety or does not conform to requirements, the plant would not be allowed to restart until the issue is addressed and until the NRC grants its approval.

Option 4

Option 4 is identical to Option 3, except that the NRC orders would require immediate plant shutdowns to perform the inspections. This Option would be preferable in the case of an immediate safety issue posing a clearly demonstrated significant and immediate risk to an operating plant. NRR OI LIC-504 defines a risk significant condition as significant enough to warrant immediate action if the calculated large early release frequency (LERF) is on the order of 1×10^{-4} yr^{-1}.

Assessment of Options

The NRC staff evaluated the relative merits of the four options discussed in the preceding section. The staff has concluded that any of the four options proposed will adequately address the possible safety impact to the U.S. nuclear power reactor fleet posed by potential regions of CMAC in components produced by ACF. However, all four options are not equivalent or warranted, as discussed below.

Option 1: Evaluate and Monitor

To properly assess this option, the NRC assessed each of the five principles of the risk-informed decision-making process within the context of this option.

Principle 1—Compliance with Existing Regulations

A licensee is responsible for ensuring that the applicable regulatory and technical requirements are appropriately identified in the procurement documentation and for evaluating whether the purchased items, upon receipt, conform to the procurement documentation, in accordance with 10 CFR Part 50, Appendix B. The NRC expects that licensees and vendors subject to NRC jurisdiction affected by the potential presence of CMAC have verified compliance with applicable NRC requirements and regulations for each potentially affected component or, alternatively, performed an appropriate evaluation that concludes that the condition is not adverse to safety. The NRC has not received a 10 CFR Part 21 notification from a component supplier or licensee associated with CMAC. The ongoing evaluations have not yet determined that a deviation exists under 10 CFR Part 21. The NRC confirms licensee and vendor compliance with NRC requirements through submitted reports, routine inspections, and continuous oversight provided by the plant resident inspector. For example, the NRC reviews 10 CFR Part 21 evaluations and the response to operational experience routinely as part of the Reactor Oversight Process (ROP). Specifically, IP 71152,[36] "Problem Identification and Resolution," provides guidance on reviewing licensee evaluations to ensure that potential supplier deviations are adequately captured to identify and address potential defects. A review of the 10 CFR Part 21 process is also part of the vendor inspection

[36] See ADAMS Accession No. ML053490187.

program. Any non-compliances identified through NRC oversight activities are addressed through the enforcement program to ensure compliance is restored. In addition, safety concerns identified through NRC's oversight activities may be escalated, such as to conduct a reactive inspection or to issue a Confirmatory Action Letter or Safety Order. Therefore, Principle 1 is satisfied for Option 1.

Principle 2—Consistency with the Defense-in-Depth Philosophy

The aspect of defense-in-depth of relevance to the potential presence of CMAC in regions of RCS components is "barrier integrity." The reactor coolant pressure boundary is one of the three principal fission-product release barriers in a U.S. plant. Under 10 CFR 50.61a, the NRC established a 95th percentile TWCF of less than $1x10^{-6}$ yr^{-1} and a CDF of less than $1x10^{-6}$ as acceptable RPV failure probabilities. The conservative assessment performed by the industry and described earlier showed that the probability of compromising the barrier integrity function for the inservice U.S. components of interest are significantly below these acceptance levels. If a design-basis accident were to compromise the pressure boundary, the remaining two independent fission-product release barriers (i.e., fuel cladding and containment) would still provide adequate defense-in-depth. The NRC has reasonable assurance that U.S. plants with components produced by ACF maintain adequate defense-in-depth. Therefore, Principle 2 is satisfied for Option 1.

Principle 3—Maintenance of Adequate Safety Margins

A region of CMAC in a component could reduce the margin against fracture. However, it has been shown that this reduction in margin does not affect the safe operation of the inservice components being evaluated. The ASN evaluation described

earlier determined that the safety margin against fast fracture is maintained in all conditions analyzed. Industry determined in MRP-417 that the CMAC levels necessary to be considered significant to safety are more than 200 percent of those observed in components. Based on its review of these evaluations, the NRC has reasonable assurance that U.S. plants with components produced by ACF maintain sufficient safety margins. Therefore, Principle 3 is satisfied for Option 1.

Principle 4—Demonstration of Acceptable Levels of Risk

If it is conservatively assumed that the TWCF equates to the LERF (neglecting mitigating factors), the calculated 95th percentile TWCF for components with CMAC and thus the LERF is less than 1×10^{-6} yr^{-1}. Because this is below the immediate safety determination limit, there is no immediate safety concern. Therefore, Principle 4 is satisfied for Option 1.

Principle 5—Implementation of Defined Performance Measurement Strategies

Because there is no indication that the U.S. inservice components produced by ACF are noncompliant with the applicable regulations and because the NRC has reasonable assurance that defense-in-depth, safety margins, and risk levels are adequately maintained, the current monitoring programs at the plants are adequate, and additional performance measurement strategies are not warranted. However, the NRC staff would continue to monitor the U.S. nuclear industry and international activities related to the CMAC topic to analyze any new information to determine whether additional performance measurement strategies are necessary. Therefore, Principle 5 is satisfied for Option 1.

Option 2: Issue a Generic Communication

This option reinforces the regulatory determination made in Option 1 by issuing a GL requesting that the documentation and evaluations performed by licensees and their component suppliers conclude that the components produced by ACF do not have defects or deviations that pose a substantial safety hazard. The NRC would not expect the information collected in the response to a GL to change any of the conclusions reached in Option 1, including those related to defense-in-depth, safety margins, or risk-level determinations. Therefore, all five principles of risk-informed decision-making would also be satisfied for Option 2. Additionally, the relevant vendors have informed the affected licensees of the CMAC topic. Vendors and licensees must meet their 10 CFR Part 21 evaluation and reporting responsibilities if the condition warrants such action. As part of the ROP and vendor inspection program, the NRC reviews these evaluations for adequacy.

Option 3: Issue Orders Requiring Inspections

This option reinforces the determinations made in Option 1 by performing inspections to confirm that an appropriate degree of conservatism was used in the evaluations of the potential impact of CMAC on U.S. components produced by AFC. The NRC would not expect the information collected by performing nondestructive examinations of the inservice components to significantly affect the defense-in-depth, safety margins, or risk-level determinations made in Option 1. Therefore, all five principles of risk-informed decision-making would also be satisfied for Option 3.

Option 4: Issue Orders Suspending Operation

In evaluating the international, U.S. industry, and NRC safety assessments, the NRC determined that the impact of CMAC on the integrity of the U.S.-forged components in question is small and that the calculated 95th percentile TWCF for PTS and the CPF for normal operating conditions fall below the NRC's safety criteria of $1x10^{-6}$ yr^{-1} and $1x10^{-6}$, respectively. Because the assumption that the TWCF is equivalent to the LERF because of mitigating factors is extremely conservative, the results indicate that the impacts of CMAC would result in a risk of LERF less than $1x10^{-4}$ yr^{-1}. Therefore, because the NRC's risk criterion to shut down a plant is not met, the agency dismissed Option 4 without an evaluation of the five principles of risk-informed decision-making.

Final Assessment

The staff determined that Option 1 was the most appropriate action based on the material and processing information reviewed by the staff during the vender inspection of AREVA, experimental data and evaluation reported by ASN, PFM analyses conducted by the industry, the staff's review of the open literature on CMAC in steel ingots and its effect on performance, and an evaluation demonstrating that Option 1 satisfies all five key principles of risk-informed decision-making. Additionally, this compilation of information reviewed affirms the staff's preliminary safety assessment that the safety significance of CMAC to U.S. plants appears to be negligible and does not warrant immediate action. If new information becomes available that calls into question the conservatism of the evaluations supporting Option 1 or the regulatory compliance of the plants with inservice components from ACF, the NRC staff will reevaluate the need for additional actions. The staff's evaluation dated February 22, 2018, documents the

NRC's full evaluation of the CMAC topics as it relates to plants operating in the United States.

E. Evaluation of the Petitioners' Requests

Petitioners' Request 1: Suspend power operations of U.S. nuclear power plants that rely on ACF components and subcontractors pending a full inspection (including nondestructive examination by ultrasonic testing) and material testing. If carbon anomalies ("carbon segregation" or "carbon macrosegregation") in excess of the design-basis specifications for at-risk component parts are identified, require the licensee to do one of the following:

a. replace the degraded at-risk component(s) with quality certified components, or

b. for those at-risk degraded components that a licensee seeks to allow to remain in-service, make application through the license amendment request process to demonstrate that a revised design-basis is achievable and will not render the in-service component unacceptably vulnerable to fast fracture failure at any time, and in any credible service condition, throughout the current license of the power reactor.

NRC Response:

This request is essentially identical to Option 4 described above. The NRC has determined, through its PFM analyses, that the expected impact of CMAC on the LERF is less than $1x10^{-6}$ yr^{-1}. Therefore, the risk criterion to shut down a plant is not met.

Petitioners' Request 2: Alternatively modify the operating licenses to require the affected operators to perform the requested emergency enforcement actions at the next scheduled outage.

NRC Response:

This request is essentially identical to Option 3 described above. As discussed above, performing nondestructive examinations of the inservice components is not expected to provide information that would significantly affect the defense-in-depth, safety margins, or risk-level determinations that would be provided by continued monitoring and evaluation of new information.

Petitioners' Request 3: Issue a letter to all U.S. light-water reactor operators under 10 CFR 50.54(f) requiring licensees to provide the NRC with information under oath and affirming specifically how U.S. operators are reliably monitoring contractors and subcontractors for the potential carbon segmentation anomaly in the supply chain and the reliability of the quality assurance certification of those components, and publicly release the responses.

NRC Response:

This request is essentially identical to Option 2 described above. As discussed above, the information collected through a 10 CFR 50.54(f) request for information or a GL is not expected to change any of defense-in-depth, safety margins, or risk-level determinations that would be provided by continued monitoring and evaluation of new information. In addition, the relevant vendors and licensees must meet their 10 CFR Part 21 evaluation and reporting responsibilities if the condition warrants such action. As part of the ROP and vendor inspection program, the NRC reviews these evaluations for adequacy.

Petitioners' Request 4: [The Petitioners added Crystal River Unit 3 to the plants for which they requested actions, which include the following]:

a. **Confirm the sale, delivery, quality control and quality assurance certification and installation of the replacement reactor pressure vessel head as supplied to Crystal River Unit 3 by then Framatome and now AREVA-Le Creusot Forge industrial facility in Charlon-St. Marcel, France and;**

b. **With completion and confirmation [of the above Crystal River Unit 3 actions], the modification of Duke Energy's current license for the permanently closed Crystal River Unit 3 nuclear power station in Crystal River, Florida, to inspect and conduct the appropriate material test(s) for carbon macrosegregation on sufficient samples**

harvested from the installed and now in service irradiated Le Creusot Forge reactor pressure vessel head [sic]. The Petitioners assert that the appropriate material testing include OES.

NRC Response:

AREVA did not identify Crystal River Unit 3 as a plant that contained components from ACF,[37,38] and the staff has not confirmed that this unit contained any forgings manufactured from ingots produced by ACF. In addition, Crystal River Unit 3 is currently shut down and in the process of decommissioning. Therefore, the Petitioners' requests 1, 2, 3, and 4(a) do not apply to this plant. However, the acquisition and subsequent testing of irradiated and aged plant material from decommissioned plants could be a valuable research activity that might offer useful scientific information on the progress of aging mechanisms. The harvesting of reactor vessel material from plants that have been permanently shut down can be a complex and radiation-dose-intensive effort. The NRC's Office of Nuclear Regulatory Research has previously obtained samples appropriate for testing from shutdown plants. In regard to this request, the NRC may, in the future, seek to purchase samples. However, the identified facility has ceased operations, and there is no safety concern at those facilities that justifies enforcement-related action (i.e., to modify, suspend, or revoke the license) to give the NRC reasonable assurance of the adequate protection of public health and safety.

[37] See ADAMS Accession No. ML17040A100.
[38] See ADAMS Accession No. ML17009A278.

III. Conclusion

Based on the evaluations provided above, and documented in the February 22, 2018, NRC memorandum, the NRR Director has determined that the actions requested by the Petitioners, will not be granted in whole or in part.

As provided for in 10 CFR 2.206(c), a copy of this Director's Decision will be filed with the Secretary of the Commission for the Commission to review. As provided for by this regulation, the decision will constitute the final action of the Commission 25 days after the date of the decision unless the Commission, on its own motion, institutes a review of the decision within that time.

Dated at Rockville, MD, this 2nd day of August, 2018.

For the Nuclear Regulatory Commission

/RA/

Brian E. Holian, Acting Director,
Office of Nuclear Reactor Regulation.

Attachment:
List of Affected Reactors

List of Power Reactors Affected by the Petition

Plant	Docket No.	Facility Operating License No.
Prairie Island Nuclear Generating Plant, Unit 1	05000282	DPR-42
Prairie Island Nuclear Generating Plant, Unit 2	05000306	DPR-60
Arkansas Nuclear One, Unit 2	05000368	NPF-6
Beaver Valley Power Station, Unit 1	05000334	DPR-66
North Anna Power Station, Unit 1	05000338	NPF-4
North Anna Power Station, Unit 2	05000339	NPF-7
Surry Power Station, Unit 1	05000280	DPR-32
Comanche Peak Nuclear Power Plant, Unit 1	05000445	NPF-87
V.C. Summer Nuclear Station, Unit 1	05000395	NPF-12
Joseph M. Farley Nuclear Plant, Unit 1	05000348	NPF-2
Joseph M. Farley Nuclear Plant, Unit 2	05000364	NPF-8
South Texas Project, Unit 1	05000498	NPF-76
South Texas Project, Unit 2	05000499	NPF-80
Sequoyah Nuclear Plant, Unit 1	05000327	DPR-77
Watts Bar Nuclear Plant, Unit 1	05000390	NPF-90
Millstone Power Station, Unit 2	05000336	NPF-65
Saint Lucie Plant, Unit 1	05000335	DPR-67
Crystal River Unit 3 Nuclear Generating Plant	05000302	DPR-72