UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

February 22, 2018

MEMORANDUM TO: George A. Wilson, Director
Division of Materials and License Renewal
Office of Nuclear Reactor Regulation

FROM: David L. Rudland, Senior Level Advisor ***/RA/***
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SUBJECT: CARBON MACROSEGREGATION IN REACTOR COOLANT SYSTEM COMPONENTS MANUFACTURED BY AREVA CREUSOT FORGE – DOCUMENTATION OF THE TECHNICAL DISPOSITION OF THE TOPIC AND SAFETY DETERMINATION

In accordance with the Office of Nuclear Reactor Regulation (NRR) Instruction LIC-504, Revision 4, "Integrated Risk-Informed Decision-Making Process for Emergent Issues," dated June 2, 2014, the staff of the U.S. Nuclear Regulatory Commission (NRC) has performed a risk-informed evaluation of the potential safety significance of carbon macrosegregation (CMAC) in components produced by AREVA Creusot Forge (ACF), as it relates to plants operating in the U.S. This evaluation was prompted by the discovery of CMAC in components produced by ACF for foreign plants. A region of CMAC in a component can locally degrade the mechanical performance of the material. The CMAC discovered in the international components should not be present in U.S. components due to the requirements in Appendix B, "Quality Assurance Criteria for Nuclear Power Plants and Fuel Reprocessing Plants," to Title 10 of the Code of Federal Regulations (10 CFR) Part 50. If there is reasonable information indicating that CMAC may be present in U.S. components, licensees and/or their component suppliers are responsible for performing an evaluation to determine if the CMAC poses a substantial safety hazard, per 10 CFR Part 21. Additionally, 10 CFR Part 21 requires that the NRC be formally notified of information reasonably indicating that a basic component contains defects that could create a substantial safety hazard. The NRC has not received a 10 CFR Part 21 notification associated with the CMAC topic.

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The enclosure to this memorandum contains the NRC staff's assessment of the CMAC topic, as it relates to the safe operation of U.S. plants. The NRC staff evaluated the available CMAC information reported by the U.S. industry, international community, and in the open literature. The NRC staff assessment determined that the safety significance of CMAC is negligible for U.S. plants. As discussed in the enclosure, the staff evaluated the following four options to address the CMAC topic. A reasonable attempt was made to align the options considered with those proposed by public stakeholders through the 10 CFR Part 2.206 petition process (ADAMS Accession No. ML17025A180).

> Option 1 - Monitor and Evaluate: NRC staff are to continue monitoring all domestic and international information associated with the CMAC topic. The staff will evaluate all new information, as it becomes available, to ensure that the conservatism in the staff's final safety determination is maintained. New information may prompt additional actions, if warranted.
>
> Option 2 - Issue a Generic Communication: Issue a generic letter (GL) to licensees operating with forged components produced by ACF requesting that they confirm that their 10 CFR Part 50, Appendix B, quality assurance program verified compliance with applicable NRC regulations and American Society of Mechanical Engineers (ASME) Code requirements. Licensees are required to submit a written response to a GL, in accordance with 10 CFR 50.54(f).
>
> Option 3 - Issue Orders Requiring Inspections: Order the licensees operating with components produced by ACF to conduct nondestructive examinations during the next scheduled outage. Examinations are to verify the condition of the components and carbon levels.
>
> Option 4 - Issue Orders Suspending Operation: Identical to Option 3 except plants are ordered to shutdown immediately and perform inspections. The plants cannot restart until the issue is addressed and approval is granted by the NRC.

The NRC staff recommends Option 1 because: the safety significance of CMAC is negligible for U.S. plants based on the results of deterministic and probabilistic evaluations; the potential extent and practical degree of condition are bounded; and the five key principles of risk-informed decision making (i.e., compliance with regulations, defense-in-depth, adequate safety margins, acceptable risk level, performance measurement strategy) are satisfied.

Options 2 and 3 would provide the NRC staff with varying levels of additional information; however, the staff does not recommend these options. This additional information is not expected to change the conservative assumptions in the existing analyses or conclusion that the safety significance of CMAC is negligible.

Option 4 is not recommended since a Large Early Release Frequency (LERF) of 1E-4 events per year has been established by the NRC for taking immediate actions. The LERF criteria is not exceeded; therefore, the immediate suspension of plant operation is not warranted.

Additionally, the staff has confirmed its preliminary safety determination made in 2016 that: the safety significance of CMAC to U.S. plants is negligible and no immediate action is warranted. The affirmation of the staff's preliminary safety determination is based on supporting information that has been subsequently collected and evaluated. Therefore, the staff's final safety determination remains the same as the preliminary determination.

The enclosure contains details of the staff's evaluation.

Enclosure:
Carbon Macrosegregation in Reactor Coolant System Components Manufactured by AREVA Creusot Forge – Staff Assessment of Potential Safety Significance to Nuclear Power Plants Operating in the United States

CARBON MACROSEGREGATION IN REACTOR COOLANT SYSTEM COMPONENTS MANUFACTURED BY AREVA CREUSOT FORGE - DOCUMENTATION OF THE TECHNICAL DISPOSITION OF THE TOPIC AND SAFETY DETERMINATION

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CARBON MACROSEGREGATION IN REACTOR COOLANT SYSTEM COMPONENTS MANUFACTURED BY AREVA CREUSOT FORGE – STAFF ASSESSMENT OF POTENTIAL SAFETY SIGNIFICANCE TO NUCLEAR POWER PLANTS OPERATING IN THE UNITED STATES

## 1.0 INTRODUCTION

Carbon macrosegregation (CMAC) is a known phenomenon that takes place during the casting of large ingots. 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 are purposefully removed/discarded as part of the material processing. Regions of positive CMAC that are not appropriately removed (i.e., CMAC with carbon levels higher than specified) result in localized regions near the surface of the final component with higher strength and lower toughness, relative to the bulk material.

Regions of CMAC were discovered in European Pressurized Reactor (EPR) reactor pressure vessel (RPV) heads manufactured for a plant in Flamanville, Manche, France [1]. The forgings for the Flamanville upper and lower RPV heads were produced at AREVA Creusot Forge (ACF). The discovery of the CMAC in the heads promoted the French Nuclear Safety Authority (ASN; l'Autorité de sûreté nucléaire) to request that the operator, Electricity of France (EDF; Électricité de France S.A.), conduct a review of inservice forged components at all of its plants to determine the potential extent of condition. The review identified steam generator (SG) channel heads (also commonly referred to as SG primary heads) produced by ACF and Japanese Casting and Forging Corporation (JCFC) as the most likely components to contain a region of CMAC [2]. 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 nondestructive testing of the potentially affected ACF and JCFC SG channel heads be accelerated, requiring the remaining nondestructive testing to be completed within three months [3, 4]. The accelerated schedule was prompted by the discovery of higher than expected[1] carbon values measured on an inservice SG channel head produced by JCFC. The accelerated schedule resulted in plants operated by EDF being shutdown prior to their scheduled outages in order to perform the required nondestructive tests.

The NRC staff conducted a preliminary safety assessment to determine the potential safety significance posed to the U.S. fleet by the CMAC observed in reactor coolant system (RCS) components overseas. The assessment was initiated in the fourth quarter of the 2016 calendar year in response to the unscheduled shutdown of plants in France. The staff's preliminary safety assessment concluded that the safety significance of CMAC to the U.S. fleet is negligible. The basis for this preliminary determination was presented at the NRC Regulatory Information Conference (RIC) on March 16, 2017, in a technical session entitled "Carbon Macrosegregation in Large Nuclear Forgings" [5, 6], which is described in Section 3.1 of this report.

[1] The highest carbon value measured on an ACF component is 0.32 weight percent (wt%). The highest carbon value measured on a JCFC component is 0.39 wt%. The French material specification, RCC-M 16MND5, has a maximum product analysis of 0.22 wt% for carbon.

The objective of this report is to document the staff's final safety assessment of the CMAC topic, as it relates to plants operating in the U.S. The staff's final safety assessment employs the guidance in NRR Office Instruction LIC-504 [7] to the extent applicable. The remaining sections of this report discuss the: background of the CMAC topic; NRC inspection activities associated with the manufacturing of potentially affected components; technical evaluations performed by the international community; and analysis performed by the U.S. nuclear industry. A summary of the major activities and documents supporting the staff's final safety assessment is provided in Table 1. This report also serves to communicate the agency's position on the CMAC topic to U.S stakeholders.

Table 1: Summary of the major activities and documents supporting the staff's final safety assessment.

| **Date of Activity/Report** | Activity/Report Description |
|---|---|
| **4th Quarter CY 2016** | NRC staff conducts preliminary safety assessment |
| **October 19, 2016** | ASN orders plants in France to shutdown prior to their scheduled outages to perform inspects [3, 4] |
| **November 28, 2016** | NRC participates in a Multinational Design Evaluation Program (MDEP) vender inspection at ACF [8, 9] |
| **February 3, 2017** | All forgings produced by ACF for U.S. plants identified [10] |
| **March 15, 2017** | NRC holds a technical session entitled "Carbon Macrosegregation in Large Nuclear Forgings" at the 2017 RIC [5, 6] |
| **March 31, 2017** | NRC conducts vender inspection at the AREVA Inc. facility in Lynchburg, VA. [11] |
| **June 15, 2017** | ASN/IRSN report on the analysis of the CMAC issue discovered in the Flamanville EPR RPV heads Issued [12] |
| **June 2017** | Electric Power Research Institute's evaluation of the risk posed by CMAC in large nuclear forging issued [13][2] |
| **February 2018** | NRC staff final assessment, documented in this report (ADAMS Accession No. ML18017A441) |

## 2.0 BACKGROUND

The topic being addressed in this report is the potential presence of CMAC in isolated regions of RCS components produced by ACF. The influence of postulated regions of CMAC on mechanical performance is evaluated to determine the potential impact on safe operation. Section 2.0 of this report: describes the origin of CMAC; discusses the impact of CMAC on material performance; identifies the component types potentially containing regions of CMAC; and describes the regulations applicable to the CMAC topic.

[2] A revision to this evaluation (i.e., MRP-417, [13]) is being conducted to incorporate the mechanical test data provided in the ASN/IRSN report [12].

## 2.1 Potentially Affected Reactor Coolant System Components

The international community determined that components manufactured from forgings produced by ACF and JCFC have the potential to contain regions of CMAC. As of the date of this report, the NRC has not identified any U.S. plants operating with forgings from JCFC. AREVA Inc. identified all the forgings produced by ACF for U.S. plants in a letter to the NRC dated February 3, 2017 [10]. The AREVA letter [10] identified seventeen U.S. Pressurized Water Reactor (PWR) plants operating with components produced by ACF. The U.S. components produced by ACF are used in assembling RPVs, SGs, and pressurizers, as shown in Table 2. The general location of the components in Table 2 in a PWR RCS are shown schematically in Figure 1.

The components identified in Table 2 have various operational functions, one of which is to provide a fission product barrier as part of the reactor coolant pressure boundary (RCPB). The RPV and pressurizer are part of the PWR primary loop (i.e., RCS). The channel head is on the primary side of the SG. The SG tubesheet has a primary and secondary side. The SG shells and elliptical dome are on the secondary side of the SG and therefore have a negligible source term. In order to simplify the CMAC safety assessment the entire steam SG (both the primary and secondary side) is being treated as a RCS component with the safety function of barrier integrity. This is a conservative assumption because a breach in the secondary side of the SG is not expected to result in a loss of barrier integrity or breach in the RCPB. The effects of a break in the secondary side of the SG are expected to be similar to that of a main steam line break which is an analyzed event and within the design basis of the containment. The plants identified in Table 2 are two, three, or four loop designs with multiple SGs.

Table 2: Forgings produced by ACF for operating U.S. plants [10]. This tables does <u>not</u> indicate that these components contain regions of carbon macrosegregation.

| Assembly | Component Type | Loop | Arkansas Nuclear One | Beaver Valley | Comanche Peak | Farley | | Millstone | North Anna | | Prairie Island | | Sequoyah | South Texas Project | | St. Lucie | Surry | VC Summer | Watts Bar |
|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|
| | | | Unit 2 | Unit 1 | Unit 1 | Unit 1 | Unit 2 | Unit 2 | Unit 1 | Unit 2 | Unit 1 | Unit 2 | Unit 1 | Unit 1 | Unit 2 | Unit 1 | Unit 1 | Unit 1 | Unit 1 |
| Steam Generator | Secondary Head / Elliptical Head | Secondary | | 3 | 4 | | | | | | | | 4 | | | | | | 4 |
| | Upper Shell / Barrel | | | | 8 | 3 | 3 | | | | | | 8 | | | | | 6 | |
| | Conical Shell | | | 3 | 4 | | | | | | | | 4 | 4 | | | | | 4 |
| | Lower Shell / Barrel | | | | 8 | 6 | 6 | | | | | | 4 | | | | | 6 | 4 |
| | Tubesheet | Primary | | 3 | 4 | 3 | 3 | | | | | | | | 4 | | | | |
| | Primary Head / Channel Head | | | 3 | 4 | | | | | | | | 4 | | | | | 3 | 4 |
| Reactor Pressure Vessel | Monoblock Head | Primary | 1[A] | 1 | | | | | | | | | | | | | | | |
| | Closure Head Flange | | | | | | | | 1 | 1 | | | | | | | 1 | | |
| | Vessel Shell | | | | | | | | | | 1[B] | 1[B] | | | | | | | |
| Pressurizer | Upper Head | Primary | | | | | | 1 | | | | | | | | 1 | | | |
| | Lower Head | | | | | | | 1 | | | | | | | | 1 | | | |
| | Upper Shell | | | | | | | 1 | | | | | | | | 1 | | | |
| | Lower Shell | | | | | | | 1 | | | | | | | | 1 | | | |

Note A: Component not installed at the time this report was issued.

Note B: Represents multiple ring, shell, flange, and nozzle forgings to construct the RPV shell

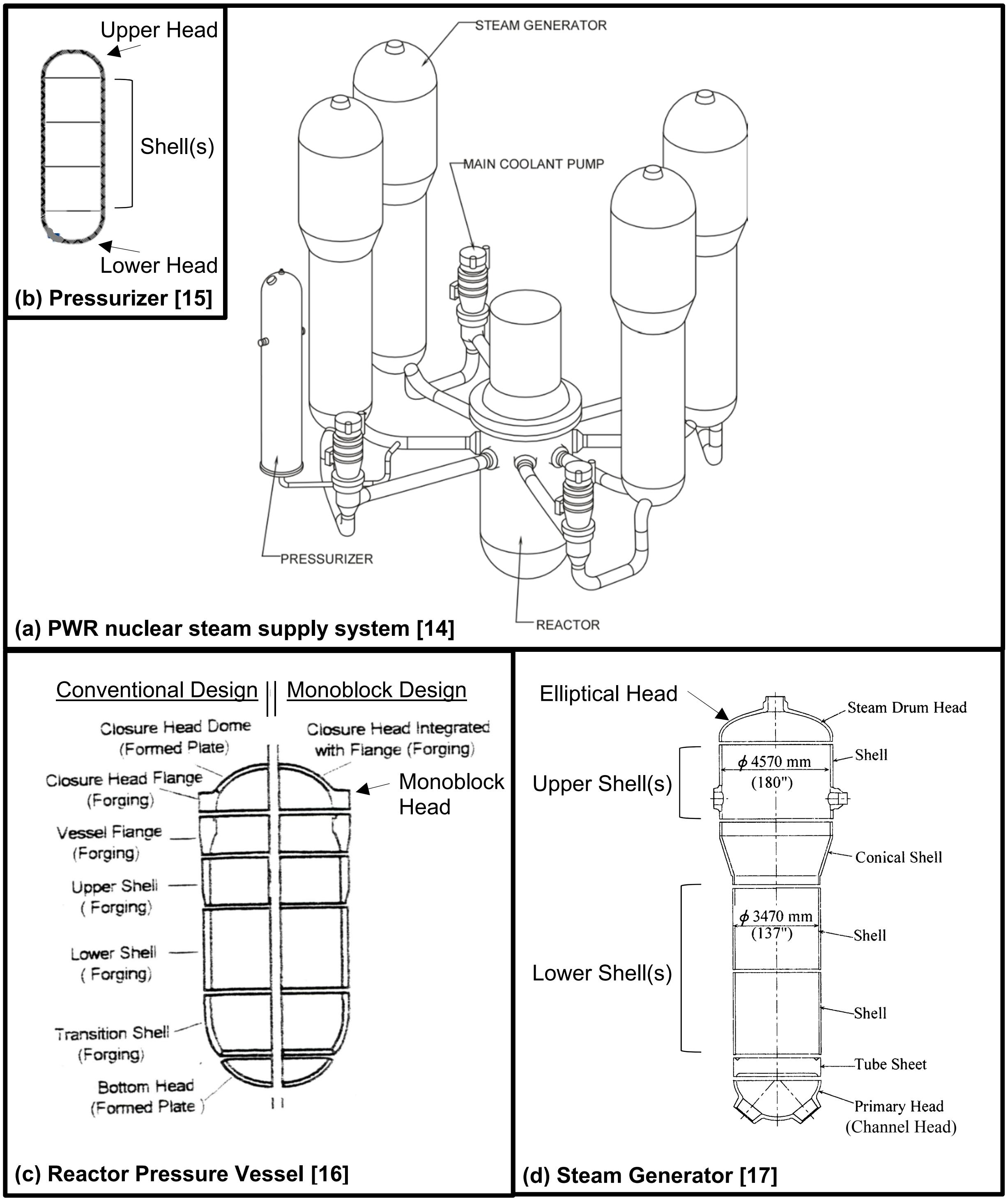



Figure 1: Schematic representation [14] of a PWR nuclear steam supply system (NSSS) identifying the general location of the components in Table 2. (b-d) Schematics of the pressurizer, RPV, and SG showing the generic[3] locations of the forged components [15, 16, 17].

[3] The designs of the NSSS and its components is plant specific. The schematics in Figure 1 are generic and only intended to show the general locations of the ACF forgings in the NSSS assemblies.

## 2.2 Carbon Macrosegregation

The specific type of segregation that resulted in the elevated carbon levels discovered overseas in the forgings produced by ACF is positive "hot-top" macrosegregation. It is necessary to distinguish between different types of segregation to avoid confusion because the potential impact on material performance is not equivalent amongst the types. A high level summary of segregation in steel ingots is provided in this section. Positive carbon hot-top macrosegregation is generically referred to as CMAC in all other sections of this report. Section 2.2 is the only section of this report that will discuss other types of segregation.

The solidification of large steel ingots inherently produces heterogeneities in the resulting cast material. Gradients in the chemical composition, commonly referred to as segregation, are one type of heterogeneity observed in industrial alloys. Segregation can occur during solidification by multiple mechanisms and over different length scales [18, 19, 20]. The solidification process will produce regions of the material that are solute rich (i.e., positive segregation) and solute poor (i.e., negative segregation), relative to the nominal composition. The magnitude of the segregation is described by the following relationship:

$$\frac{\Delta C}{C_O} = \frac{(C_i - C_O)}{C_O}$$

where $C_0$ is the nominal composition of the material, $C_i$ is the composition at a specific location in the material, and $\Delta C$ is the local deviation from the nominal composition in weight percent (wt%) of the element. The nominal chemical composition of an alloy is measured in the molten state during the "heat analysis[4]." The local composition is often referred to as the "product analysis" and may be measured at multiple stages of processing after casting. The product analysis is not expected to be the same as the nominal composition because it is recognized that industrial alloys are not homogeneous throughout the entire bulk of the material. The composition requirements for the product analysis are also often less stringent than those for the nominal composition (i.e., heat analysis) in the material specification. In the body of large fully killed steel ingots, the typical upper range (i.e., $\Delta C/C_0$) of carbon measured in the product analysis is on the order of 20 to 30% [21].

Generally, segregation can be categorized as either microsegregation or macrosegregation. Microsegregation occurs over distances on the order of the microstructure of the alloy. The process of microsegregation is primarily controlled by diffusion in the liquid phase near the solidification front. The degree of microsegregation is dependent on the material composition and cooling rate. Microsegregation is unavoidable during the casting of an alloy. However, it typically does not impact the performance of steel forgings because it is eliminated during the subsequent thermomechanical processing. Microsegregation is not a concern in structural steel forgings.

Macrosegregation occurs over larger distances (e.g., millimeters to meters). The formation of macrosegregation is controlled by multiple interdependent parameters including ingot mold geometry, alloy composition, density driven convection, thermally driven convection, and material shrinkage during solidification. A detailed description of the mechanisms responsible for the various types of macrosegregation is outside the scope of this report. Macrosegregation is unavoidable during the casting of large steel ingots. The amount and magnitude of macrosegregation can be minimized by the melting, refining, and casting methods used to

[4] The heat analysis is often also referred to as the "ladle analysis."

produce the ingot. However, macrosegregation formed during casting cannot be removed by subsequent thermomechanical processing.

In addition to carbon, other alloying and impurity elements segregate to various degrees in the regions of macrosegregation due to the nature of the solidification process in large steel ingots. It is common to focus on carbon levels because the magnitude of carbon segregation is typically greater than other elements in the regions of macrosegregation. The magnitude of other elements are commonly reported as a function of the magnitude of carbon segregation. This report will follow the convention of primarily focusing on carbon macrosegregation, while recognizing that the segregation of other elements is proportional to the segregation of carbon.

There are multiple types of macrosegregation that may occur during the casting of large steel ingots. The most relevant types include positive hot-top, negative cone, and positive channel (i.e., A-type, and V-type), as shown in Figure 2 [19]. The morphology of the macrosegregation and its location within as-cast steel ingots was intensively investigated in the early part of the 1900s [22, 23, 24]. The positive channel macrosegregation forms in the body of the ingot. The regions of positive channel macrosegregation exhibit a rod-like morphology that is typically a few millimeters in diameter and up to a meter in length. The presence of these regions of macrosegregation partially account for the range in carbon (i.e., $\Delta C/C_0$ of -15% to 30%) commonly observed in the body of large steel forgings [25, 26, 27, 28]. It should be noted that the applicable material specification may restrict the degree of chemical variation in the product to a smaller range. When this is the case, appropriate material processing must be used to comply with the material specification requirements. Materials specifications for alloys used to produce nuclear RCS components typically restrict the degree of chemical variation in the product.

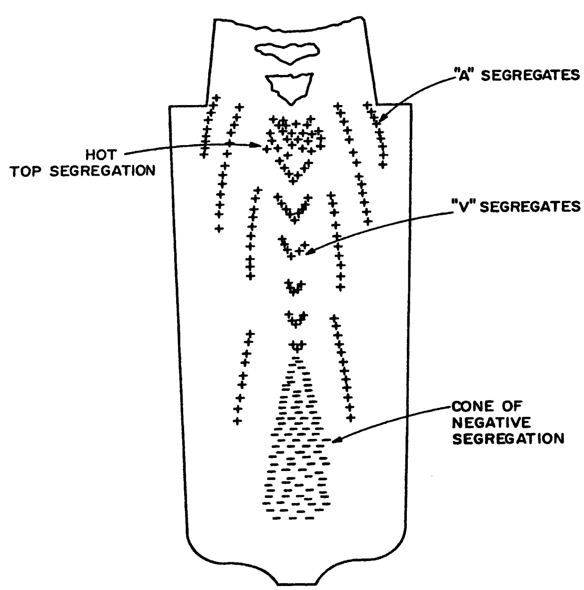



Figure 2: Schematic representation of the typical pattern of macrosegregation in a killed steel ingot. The “+” and “-” symbols represent regions of positive and negative macrosegregation, respectively [19].

The presence of positive channel macrosegregation in large steel forgings and its impact on mechanical performance is recognized by the engineering community. Positive channel macrosegregation results in regions of increased strength [29-30] and decreased toughness in the forging. An increase in strength does not negatively impact the structural integrity of a component; however, a decrease in toughness might. Of particular interest in nuclear applications is the impact on the reference nil-ductility transition temperature ($RT_{NDT}$). The decrease in toughness (increase in $RT_{NDT}$) associated with regions of positive channel segregation has been characterized in forgings produced from large steel ingots [25, 26, 31, 32, 33, 34]. The following observations have been made on forgings applicable to the nuclear industry:

- In addition to carbon; other detrimental elements migrate to the regions of positive macrosegregation, such as phosphorus (P), sulfur (S), and copper (Cu), which are known to degrade toughness;
- Positive channel segregation increases the scattered in mechanical test data;
- The observed shift in $RT_{NDT}$ due to positive channel segregation ranges from approximately 25°C to 75°C;
- The additional shift in $RT_{NDT}$ resulting from radiation damage is not as large in regions of positive channel segregation;
- Local segregation is tolerable if the material was appropriately processed and the surrounding matrix is tough.

The general trend observed in Charpy V-notch (CVN) impact data for RPV steel forgings containing regions of channel segregation is shown in Figure 3 [32]. The Charpy specimen used to generate the data in Figure 3 were extracted from a nozzle cut-out of a PWR shell. The shift in $RT_{NDT}$, at the 68J energy level, for the segregated material cut-out of a PWR shell was reported to be 60°C and 75°C, by two separate laboratories. The effect of channel segregation on the fracture toughness of RPV steel as a function of temperature has also been reported, as shown Figure 4 [33]. The compact tension (CT) specimen used to generate the data in Figure 4 were extracted from a PWR shell. This same material was reported to have a shift in $RT_{NDT}$, at the 56J energy level, of approximately 73°C. It appears that the RCC-M standard curve bounds the majority of the data points. It can also be seen that the typical scatter observed in CVN and CT data as a function of temperature can be significant. This degree of scatter complicates comparisons between properties of unsegregated material and material with segregated regions.

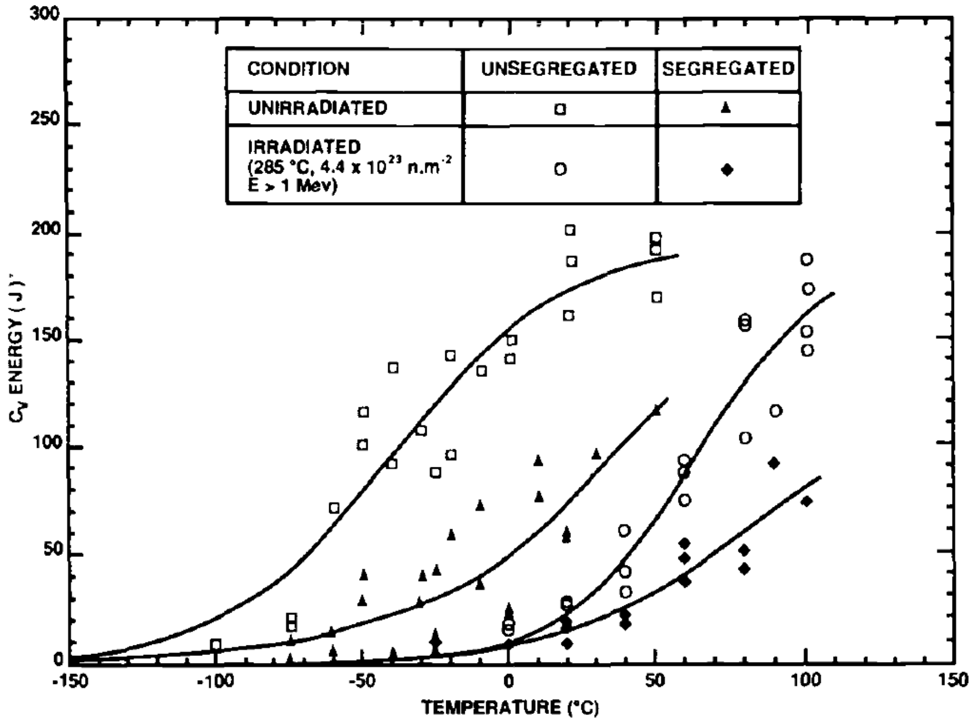



Figure 3. CVN energy ($C_V$) plotted against test temperate for RPV material with and without regions of positive channel segregation [32]. The data exhibits a 75°C shift in the unirradiated $RT_{NDT}$, at the 68J energy level, for the segregated material.

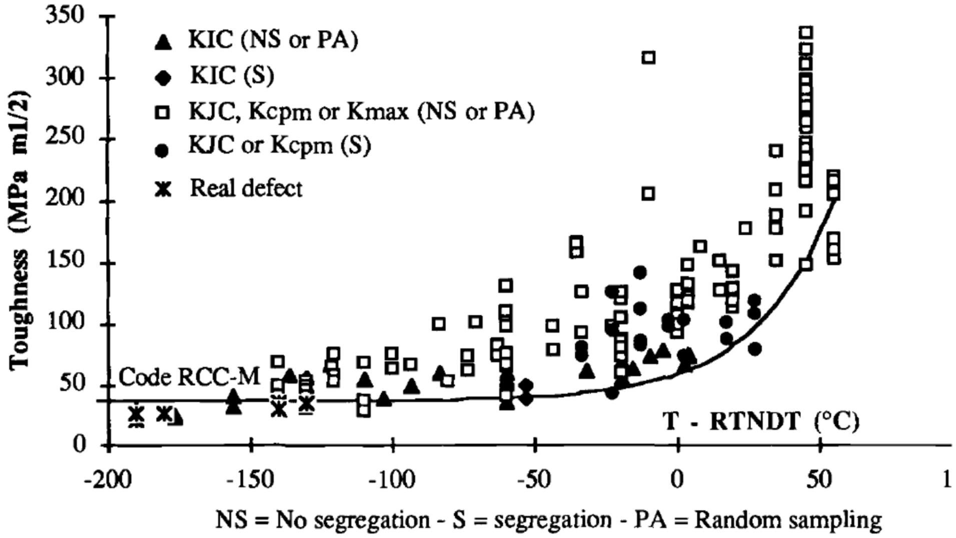



Figure 4. Fracture toughness plotted against the indexed temperature for RPV steel with and without regions of positive channel segregation [33].

Channel macrosegregation contributes to the heterogeneity of structural forgings and their mechanical properties. Material processing should be used to minimize channel macrosegregation and its impact on mechanical properties, to the greatest extent practical.

This form of macrosegregation is tolerable in a structural forging provided that appropriate margins are used during design, the material was appropriately processed, and the surrounding matrix exhibits typical toughness properties. The acceptable degree of heterogeneity in the material is established by the requirements of the applicable material specification and product specification.

The impact on mechanical performance resulting from positive hot-top and negative cone macrosegregation has not been evaluated to the same extent as channel segregation. The reason for this lower level of attention is because the regions of the ingot containing the most significant hot-top and cone macrosegregation are discarded during the material processing. Therefore, the final component should not contain regions of positive hot-top and negative cone macrosegregation that are non-representative of the typical heterogeneity observed in the body of the ingot or in excess of the applicable material specification. Additionally, the lower strength and higher toughness exhibited in the negative cone is of less significance from a structural integrity standpoint.

Positive hot-top macrosegregation is located at the top of the as-cast ingot in the hot-top region, as the name suggests. Hot-top macrosegregation forms as a composition gradient though a volume at the top of ingot. The highest magnitude of carbon macrosegregation is at the top of the ingot and continually decreases as it moves toward the center body of the ingot. The general morphology and magnitude of hot-top macrosegregation is shown in Figure 5 [28]. This illustration shows a large steel ingot produced for a structural nuclear application. The upper-most portion of the ingot hot-top may also contain cavities (i.e., piping) and other defects.

The factors affecting the morphology and magnitude of hot-top macrosegregation are of interest from a steelmaking perspective to minimize the amount of ingot discard necessary [27, 28, 34, 35, 36, 37]. Empirical relationships have been developed for estimating the magnitude of carbon macrosegregation in large steel ingots. It has been reported that the magnitude of carbon hot-top macrosegregation increases with ingot size; although alloys with molybdenum (Mo) and vanadium (V) exhibit less of an increase, as shown in Figure 6 [36]. This data set represents 152 ingots of varying size and production methods that range in nominal carbon composition from 0.10 to 0.54 wt%. The conventional ingots used by ACF to produce the U.S. components contained Mo and V, and had an estimated diameter ranging from approximately 1.7 to 2.5 meters (m). The magnitude of carbon macrosegregation in large steel ingots as a function of nominal carbon composition has also been reported, as shown in Figure 7 [34]. This data set represents approximately 50 ingots of varying size and production methods. The conventional ingots used by ACF to produce the U.S. components were multi-poured (MP) and vacuum-treated with a maximum carbon content of 0.25 wt% specified. The data points in Figure 7 that are applicable to U.S. components are enveloped by the dashed lines.

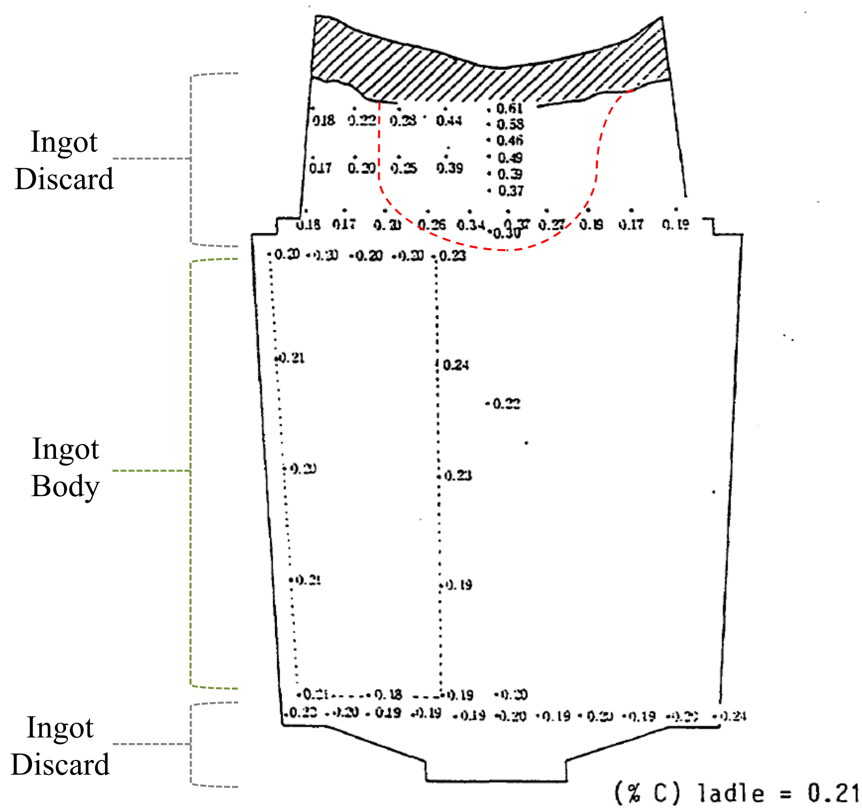



Figure 5. Distribution of carbon content in a 540 ton MnMoNi steel ingot produced for a nuclear application [26]. The broken red line is the approximate region of positive carbon hot-top segregation.

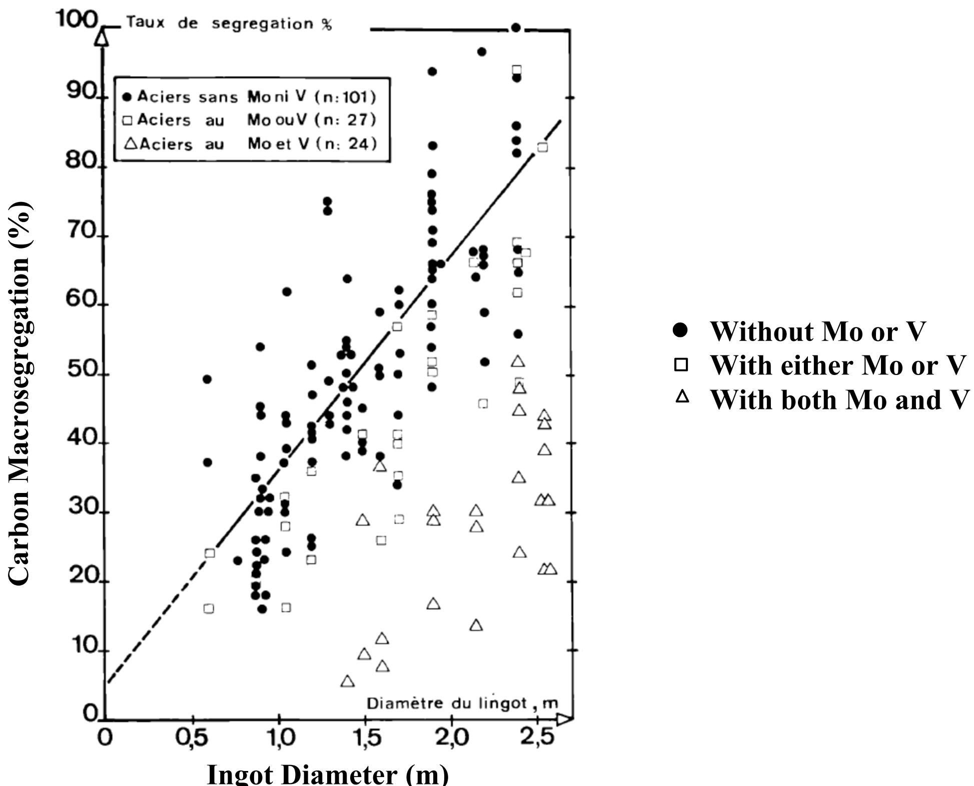



Figure 6. Percent carbon macrosegregation observed in steel ingots as a function of ingot size for alloys with and without molybdenum (Mo) and/or vanadium (V) [36].

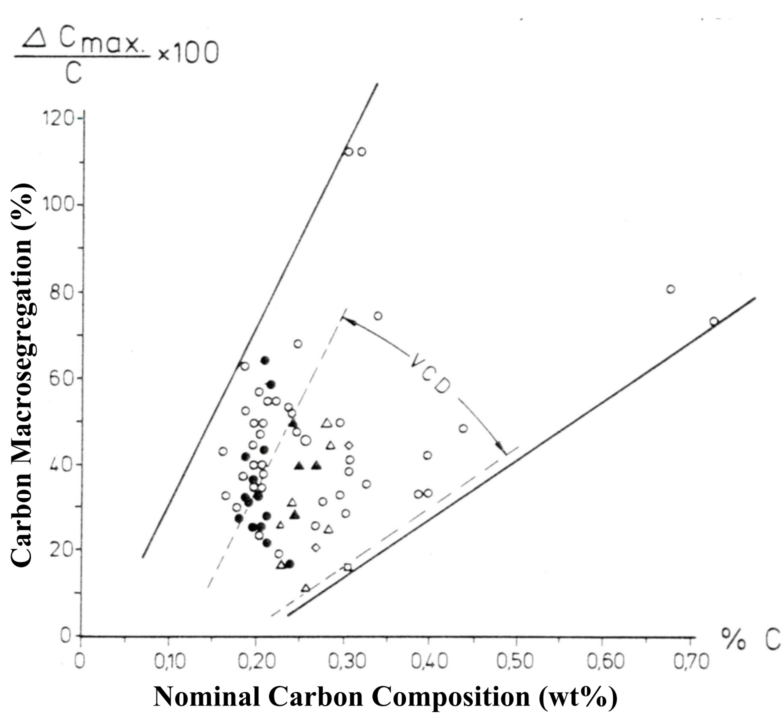



Figure 7. Percent carbon macrosegregation observed in steel ingots as a function of nominal carbon level [34].

The maximum practical level of CMAC postulated in a U.S. component produced by ACF is 60% or a maximum of 0.40wt% carbon. This staff determination is based on the empirical results reported in the open literature, maximum nominal carbon levels for the U.S. components, size of the conventional ingots used to produce the material, and knowledge of the material processing. This determination that 0.40wt% carbon is the maximum level practical to be observed in a U.S. component is also consistent with the information reported by [33], which was collected from forgings for multiple nuclear component types.

## 2.3 Materials and Processing

The U.S. forgings produced by ACF are manufactured to the requirements of ASME Code material specifications SA-508 [37], “Specification for Quenched and Tempered Vacuum-Treated Carbon and Alloy Steel Forgings for Pressure Vessels,” Grade 3 Class 1 or Grade 3 Class 2. SA-508, Grade 3, will simply be referred to as alloy SA-508 for brevity being that it is the only Grade applicable to the U.S. forging produced by ACF. SA-508 is a manganese-molybdenum-nickel (MnMoNi) low-alloy steel (i.e., Unified Numbering System [UNS] K12042). The composition requirements for the heat analysis are provided in Table 3 [37]. The permissible variations in product analysis for killed steels are provided in ASME Code material specifications SA-788 [21], “Specification for Steel Forgings, General Requirements,” although, these variations are only permitted to be applied to Mn, Ni, chromium (Cr), Mo, and V in SA-508. It should be noted that the maximum allowed carbon in SA-508 is higher than the maximum allowed carbon in the French specification, RCC-M 16MND5 (i.e., heat analysis of 0.20% C and product analysis of 0.22% C max).

Table 3: Chemical composition requirements for SA-508 [37] with the permissible variation in product analysis for large killed steel ingots per SA-788 [21].

| Element (wt%) | Carbon | | Manganese | | Phosphorus[1] | | Sulfur[1] | | Silicon | | Nickel | | Chromium | | Molybdenum | |
|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|
| | Min | Max | Min | Max | Min | Max | Min | Max | Min | Max | Min | Max | Min | Max | Min | Max |
| Heat Analysis | -- | 0.25 | 1.20 | 1.50 | -- | 0.025 | -- | 0.025 | -- | 0.04 | 0.40 | 1.00 | -- | 0.25 | 0.45 | 0.60 |
| Variation[2] | -- | 0.05 | 0.09 | 0.09 | -- | 0.015 | -- | 0.006 | -- | 0.06 | 0.03 | 0.03 | -- | 0.06 | 0.08 | 0.08 |
| Product Analysis | -- | 0.25 | 1.11 | 1.59 | -- | 0.025 | -- | 0.025 | -- | 0.04 | 0.37 | 1.03 | -- | 0.31 | 0.37 | 0.68 |

| Element (wt%) | Vanadium | | Columbium | | Copper[1] | | Calcium | | Boron | | Titanium | | Aluminum | |
|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|
| | Min | Max | Min | Max | Min | Max | Min | Max | Min | Max | Min | Max | Min | Max |
| Heat Analysis | -- | 0.05 | -- | 0.01 | -- | 0.20 | -- | 0.015 | -- | 0.003 | -- | 0.015 | -- | 0.025 |
| Variation[2] | -- | 0.01 | -- | 0.03 | -- | 0.03 | -- | -- | -- | -- | -- | 0.050 | -- | 0.010 |
| Product Analysis[3] | -- | 0.06 | -- | 0.01 | -- | 0.20 | -- | 0.015 | -- | 0.003 | -- | 0.015 | -- | 0.025 |

1. Strictor composition limits may apply.
2. Standard product variations specified in SA-788 is not permitted for SA-508 when shaded.
3. Product analysis is the SA-508 heat analysis with the permissible variation applied.

The CMAC discovered internationally in components produced by ACF were manufactured from non-trepanned conventional ingots that were cast in molds with a diameter of greater than 2 meters. An NRC inspection team reviewed the manufacturing processes used by ACF to produce the inservice U.S. components [11]. The NRC inspection team confirmed that the material processing route varied by component type, as would be expected. A summary of the ingot types used by ACF to fabricate the U.S. components is provided in Table 4 [11]. An overview of the typical material processing used to produce a SA-508 SG channel head from a conventional ingot is outlined below. This component and processing route was selected because it is the most applicable to the recent international observation of CMAC.

Table 4: Component and ingot types used by ACF to fabricate inservice U.S. components [11].

| Assembly | Component Type | Ingot Type | | | Trepanned | Ingot Weight[2] (MT) | |
|---|---|---|---|---|---|---|---|
| | | Conventional | LSD[1] | Hollow | | < 150 | ≥ 150 |
| Steam Generator | Secondary Head / Elliptical Head | X | | | | | X |
| Steam Generator | Upper Shell / Barrel | | | X | | | X |
| Steam Generator | Upper Shell / Barrel | | X | | Yes | X | |
| Steam Generator | Conical Shell | | | X | | | X |
| Steam Generator | Conical Shell | | X | | Yes | X | |
| Steam Generator | Lower Shell / Barrel | | | X | | X | |
| Steam Generator | Lower Shell / Barrel | | X | | Yes | X | |
| Steam Generator | Tubesheet | X | | | | X | X |
| Steam Generator | Primary Head / Channel Head | X | | | | | X |
| Steam Generator | Primary Head / Channel Head | | X | | | X | |
| Reactor Pressure Vessel | Monoblock Head | X | | | | | X |
| Reactor Pressure Vessel | Closure Head Flange | X | | | Yes | X | |
| Reactor Pressure Vessel | Vessel Shell | X | | | Yes | X | |
| Pressurizer[3] | Upper Head / Lower Head | X | | | | | X |
| Pressurizer[3] | Upper Shell / Lower Shell | | | X | | | X |

| | |
|---|---|
| 1. | Lingot à Solidification Dirigée (LSD) is French meaning oriented solidification ingot. |
| 2. | Ingot weight can be related to ingot type and mold design. Ingot weight of 150 metric tons was selected to provide an indication of ingot size. |
| 3. | A set of two heads or a set of two shells produced from a single ingot. |

The material processing begins with steel being melted in a basic electric furnace that is vacuum treated during the pouring of the ingot. A multi-pouring (MP) method with two ladles is used to accommodate the size of large ingots and to minimize composition gradients. The molten steel is top-poured into a big-end-up conventional ingot mold. As the molten steel solidifies in the ingot mold a heterogeneous as-cast structure is produced. The as-cast ingot is subjected to a blooming process which produces a more uniform radial cross-section and prepares the material for subsequent processing. It is at this stage of the processing that the cropping of the ingot occurs. Section 4.3.1, "Discard," of material specification SA-508 [37] states: "Sufficient discard shall be made from each ingot to secure freedom from piping and excessive segregation." The objective of the cropping is to discard the top and bottom portions of the ingot that contain defects that cannot be eliminated by subsequent processing, such as: inclusions, voids, discontinuities, and macrosegregation. The discard is measured as a percent of the total as-cast ingot weight. The amount of ingot discard necessary to meet the requirements of the material specification is process dependent. Typically, 13-25% discard is necessary from the top of a conventional ingot and 7-15% from the bottom. After cropping the bloom[5] is subjected to a series of upsetting processes imposed parallel to the longitudinal axes

[5] A bloom is a semi-finished product form. A bloom has been subjected to initial processing steps but the material processioning is not complete.

of the ingot. The upsetting widens the work piece into a disc-like shape. A preliminary heat treatment and rough machining is then performed to prepare the disc for the final forging process. The final forging process utilizes an open die to form the dome (i.e., head) shape. The rough dome is pierced from the inside at the location of the nozzle(s). The forging of the domes at ACF are conducted in an orientation that results in the outside surface of the head corresponding to the upper portion of the ingot. A second rough machining is then performed on the work piece to bring the forging contour closer to the component contour. The final heat treatment to achieve the required mechanical properties is conducted and the final machining is performed to meet dimensional requirements. It should be recognized that multiple inspections are performed throughout the material processing. These inspections include dimensional, visual, magnetic particle, and ultrasonic to ensure that the forging complies with requirements and is free from cracks or other unacceptable indications.

Descriptions of the material processing for nuclear forgings produced from conventional ingots have been described for other components elsewhere [16, 17, 25, 28, 38], including trepanning and drawing for shells. Descriptions of the material processing for nuclear forgings produced from hollow ingots and directionally solidified ingots (i.e., LSD ingots) are also described in the publically available literature [39, 40, 41, 42]. It should be recognized that hollow components (e.g., rings and shells) have a lower probability of containing regions of hot-top carbon macrosegregation, due to their processing. Although, hollow components may still contain some degree of channel macrosegregation. As discussed in Section 2.2 of this report, the magnitude of channel macrosegregation and its impact on mechanical performance is less than that of hot-top carbon macrosegregation. It should also be noted that the focus of this report and the CMAC observed internationally is hot-top carbon macrosegregation, not channel macrosegregation.

## 2.4 Applicable Regulatory Requirements

The U.S. nuclear forgings produced by ACF are required to be manufactured and procured in accordance with all applicable regulations. These regulations include those explicitly stated in the Code of Federal Regulations (CFR) and those incorporated by reference. The regulations most pertinent to the prevention and/or identification of CMAC in regions of RCS components are the ASME Code requirements incorporated by reference in 10 CFR Part 50.55a, “Codes and Standards,” and Appendix B, “Quality Assurance Criteria for Nuclear Power Plants and Fuel Reprocessing Plants,” to 10 CFR Part 50. There are also regulations that specify performance acceptance criteria and design criteria. Additionally, 10 CFR Part 21, “Reporting of Defects and Noncompliance,” requires that the NRC be informed of 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 evaluated, per 10 CFR Part 21.

It is recognized that the applicability of specific NRC regulations and ASME requirements will, in part, be dependent 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 establishes similar fundamental regulatory requirements pertaining to the integrity of the components of interest.

### 2.4.1 10 CFR Part 50, Appendix B, Quality Assurance Criteria

Appendix B to 10 CFR Part 50 establishes quality assurance (QA) requirements for the design, manufacture, construction, and operation of the structures, systems, and components (SSCs) for nuclear facilities. The 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. This necessitates that licensees contractually pass down the requirements of Appendix B via procurement documentation to suppliers of SSCs, as indicated by the following Appendix B criteria:

Criterion IV," Procurement Document Control," states:

> "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," states, in part that:

> "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."

Therefore the licensee is responsible for ensuring that the applicable regulatory and technical requirements are appropriately identified in the procurement documentation and for determining whether the purchased items conform to the procurement documentation.

The technical requirements of ASME Section III are invoked via the procurement documentation. ASME Section III also prescribes that suppliers implement NQA-1, "Quality Assurance Program Requirements for Nuclear Facilities," to satisfy Code quality requirements. The NRC endorses NQA-1, via Regulatory Guide 1.28, as an acceptable method for complying with the requirements of Appendix B. Licensees are responsible for evaluating whether a supplier is capable of meeting the technical and quality requirements for the scope of the items being purchased, including those in ASME material specification SA-508.

The licensee verifies the implementation of technical and quality requirements through the audit process. The audit process provides objective evidence of the suppliers' compliance with the technical and quality requirements specified in the procurement documentation. The requirements of Appendix B and NQA-1 prescribe that measures are established to assure that conditions adverse to quality, such as deficiencies, deviations, defective material, and nonconformances are promptly identified and corrected. Nonconformances identified by the supplier during manufacturing are required to be technically evaluated and dispositioned accordingly. If a nonconformance is identified with the final product, such as the presence of CMAC, an engineering evaluation is performed and the nonconformance is documented on the certificate of conformance (CoC). The licensee is responsible for reviewing the CoC during receipt inspection for acceptance of final product upon delivery.

The implementation of these processes provides reasonable assurance that deviations from technical requirements will be identified, evaluated, and dispositioned in accordance with the requirements of Appendix B to 10 CFR 50. The NRC provides oversight of the nuclear industry's vendor activities by performing inspections of vendors to examine whether the vendor is compliant with the requirements of Appendix B to 10 CFR 50 and other applicable regulations. Additionally, 10 CFR Part 21 requires licenses and their component suppliers to notify the NRC if they become aware of information reasonably indicating that a basic component contains defects that could create a substantial safety hazard. Therefore, U.S. licensees and their suppliers are required to notify NRC if at any point new information related to CMAC in a component could create a substantial safety hazard. All manufacturers of safety-related SSCs are required to implement the reporting requirements of 10 CFR 21.21. The NRC has not received any 10 CFR Part 21 notifications associated with the CMAC topic.

### 2.4.2 10 CFR Part 50.55a, ASME Code Requirements

10 CFR Part 50.55a incorporates by reference ASME Section III, "Rules for Construction of Nuclear Facility Components." Subsections NCA-1220, "Materials," and NCA-3800, "Metallic Quality System Program Requirements," of ASME Section III include requirements applicable to the processes and quality control for addressing CMAC in RCS components.

Subsection NCA-1220 of ASME Section III establishes that metallic materials shall be manufactured to an "SA" specification or other approved material specification. ASME Code, Section II, "Materials" contains the "SA" material specifications. These material specifications contain the mandatory requirements, specific prohibitions, and nonmandatory guidance for material production and procurement. The requirements of material specification SA-508 are provided in ASME Section II. Section 4.3.1, "Discard," of material specification SA-508 states: "Sufficient discard shall be made from each ingot to secure freedom from piping and excessive segregation." Therefore, components produced using SA-508 material shall comply with the alloy composition limits and not contain regions of CMAC, per 10 CFR Part 50.55a.

Subsection NCA-3800 of ASME Section III establishes quality system program requirements for metallic material suppliers. These requirements apply to the production of SA-508, including the following subsections.

Subsection NCA-3851.1 states, in part:

> "The establishment of the Program shall include consideration of the technical aspects and provide for planning and accomplishment of activities affecting quality. The Program shall provide for any special controls, processes, test equipment, tools, and skills to attain the required quality and for verification of quality."

Subsection NCA-3851.2 states, in part:

> "The Quality System Manual shall define the specific activities included in the scope of the work the Material Organization proposes to perform, including any combination of (1) operations performed during the melting and heat analysis, affecting the mechanical properties, conversion from one product form into another product form including applicable dimensional requirements, and certification to the applicable material specification."

Requirements applicable to process control and quality of the material are found in various other subsections of NCA-3800, including NCA-3857. Therefore, using the appropriate process controls, quality controls, testing, and verifications, in accordance with ASME Section III, provides the basis for certifying that an alloy complies with the applicable material specification, per 10 CFR Part 50.55a.

#### 2.4.3 Design and Performance Based Regulations

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 impacted by regions of CMAC. These regulations include:

- 10 CFR Part 50.60, "Acceptance Criteria for Fracture Prevention Measures for Lightwater Nuclear Power Reactors for Normal Operation"
- Appendix A to Part 50, "General Design Criteria (GDC) for Nuclear Power Plants"
    - GDC 1, "Quality Standards and Records"
    - GDC 14, "Reactor Coolant Pressure Boundary"
    - GDC 31, "Fracture Prevention of Reactor Coolant Pressure Boundary"
- Appendix G to Part 50, "Fracture Toughness Requirements"

As discussed in Sections 2.4.1 and 2.4.2 of this report, CMAC is to be addressed prior to a component being put into service. The regulations identified in this section are potentially impacted if a component containing regions of CMAC are put into service. The requirements of 10 CFR Part 50.60; GDC 1, 14, and 31; and Appendix G to Part 50 must be met regardless of presence of CMAC in a component. The regulations listed above would be considered during an evaluation to determine if a defect (e.g., CMAC) poses a substantial safety hazard, per 10 CFR Part 21.

## 3.0 TECHNICAL EVALUATION

The NRC staff conducted a preliminary safety assessment to determine the risk posed to the U.S. fleet by the CMAC observed in components overseas. Subsequently, an NRC inspection team performed a vender inspection to review component-specific information for U.S. components produced by ACF [11]. Following these NRC activities, ASN and the French Institute for Radiological Protection and Nuclear Safety (IRSN: Institut de radioprotection et de sûreté nucléaire) issued a report [12] describing its experimental efforts and summarizing the results of a portion of its activities to address the CMAC issue in France. In parallel, the Electric Power Research Institute (EPRI) performed a probabilistic analysis [13] to assess the potential safety significance of postulated regions of CMAC in U.S. components. Section 3.0 of this report: provides the NRC's preliminary safety assessment; reviews the documents and activities applicable to the ACF CMAC topic; describes the NRC's risk-informed decision making process; reviews the options considered by the NRC to address the CMAC topic in the U.S.; and provides the NRC's final safety assessment. Aspects of the vender inspection report and IRSN/ASN reports that are not directly related to the evaluation of the CMAC topic are outside the scope of this report and not discussed.

### 3.1 Preliminary NRC Safety Determination (2016)

As indicated in Table 1, the NRC staff conducted a preliminary assessment of the safety significance of CMAC during the 4th quarter of 2016. This assessment was based solely on the information available during this time period, including ongoing research and regulatory activities in the U.S., Japan, and France. The preliminary assessment concluded that the safety significance of CMAC to the U.S. fleet is negligible and immediate regulatory action was not warranted. The basis for this preliminary determination is provided below.

### 3.1.1 Basis for Preliminary Assessment

The beltline region in a RPV shell surrounds the active fuel and experiences neutron radiation damage [43, 44]. As a result of this damage, the beltline region of the RPV shell is expected to be the most likely RPV failure location during normal operations or under postulated accident scenarios. Understanding and mitigating failure in this region has been the focus of both research and regulations for over 50 years to provide assurance that the RPV can operate safely [43, 44, 45, 46, 47, 48, 49, 50, 51]. Regulations related to design requirements, preservice fabrication and inspection, inservice inspection and monitoring, and operating restrictions work synergistically to provide reasonable assurance that the likelihood of failure within the RPV beltline region remains acceptably low over the operating lifetime of a nuclear power plant (NPP). Research has been used to demonstrate both the effectiveness of current regulations and the inherent safety of the RPV shell [50].

The assessment conducted a qualitative comparison of the likelihood of failure of components potentially affected by CMAC relative to an RPV shell. The monoblock RPV head and SG channel head were evaluated because they were judged as the component types most likely to be affected by CMAC based on their size and the staff's knowledge of the potential processing paths ACF may have used to produce them. These components are also part of the primary pressure boundary and thus have among the greatest safety significance of all components potentially affected by CMAC. The RPV shell was chosen for this comparison because, it is also part of the primary pressure boundary and, as discussed in the previous paragraph, is expected to be extremely unlikely to fail.

Failure of any component requires a critical combination of the following three attributes: high stresses, a sufficiently deep crack, and sufficiently low material toughness. This qualitative assessment compared each of these three attributes between the RPV shell and the RPV and SG channel heads (referred to as RPV/SG heads through remainder of Section 3.1), which are potentially affected by CMAC. Then, the comparisons of the individual attributes were combined to assess the overall failure likelihood of these components. Failure of the RPV shell is extremely unlikely. Therefore, if the qualitative comparison determined that the likelihood of failure of an RPV/SG head is less than an RPV shell, there would be negligible safety significance of CMAC in U.S. NPPs and no immediate regulatory action would be required.

### 3.1.2 Qualitative Failure Comparison

#### 3.1.2.1 Most Likely Failure Location

Based on the NRC staff's knowledge of ACF component processing and operating/accident conditions, the NRC staff assessed the most probable failure locations of components potentially affected by CMAC. The beltline region of the RPV shell is most likely to fail near the inner surface of the vessel wall. This location is the most likely to fail because radiation embrittlement degrades the material toughness over time and this effect is most significant at the RPV inner wall. Additionally, the highest component stresses can occur at this location during both normal operation and during postulated accident scenarios. Conversely, an RPV/SG head is most likely to fail near the outer surface of the component because this is the area where positive CMAC is most likely to degrade material toughness. Stresses in this location are the highest as the RPV heats up during reactor startup. Therefore, the qualitative evaluation compared the three physical attributes required for component failure (i.e., stress, cracks, and material toughness) near the inner wall of an RPV shell with these attributes near the outer wall of RPV/SG heads.

3.1.2.2 Component Stresses

There are several different contributors to the total stress present in a component. Stresses in RCPB components result from internal pressure (pressure stresses) and the temperature gradient within the component (thermal stresses). Cladded components, like the RPV shell and RPV/SG heads, are also subject to stresses due to differences in the coefficient of thermal expansion (CTE) between the thinner stainless steel cladding and the thicker ferritic steel component. The cladding is applied to the inner surface of these components to prevent corrosion of the ferritic steel. The cladding thermal stresses are highest at the interface between the cladding and ferritic steel and decrease sharply through the thickness of the component such that they are negligible near the outer surface. Finally, welded components are also subject to residual stresses that result from the welding process. These stresses are highest within the weld and adjacent to the weld in the base material. These stresses diminish away from the weld region. It should be noted that CMAC does not induce additional stresses within an affected component. In summary, the following types of stresses may exist in a pressure boundary component: pressure stresses, thermal stresses, cladding stresses, and residual stresses due to welding.

Components are stressed during both normal operations and during postulated accident scenarios. During normal operations, the highest stresses near the inner wall of the RPV shell occur during reactor shutdown (or cooldown) while the highest stresses near the outer wall of the RPV/SG heads occur during reactor start-up (or heat-up). The pressure stresses on these components during these events are similar because ASME Section III prescribes identical design margins. However, the stresses will be slightly less near the outer surface of the RPV/SG heads compared to the inner surface of the RPV shell as the pressure stresses decrease through the thickness. Thermal stresses will be higher in the RPV shell because this component is thicker which leads to greater constraint and higher thermal bending stresses. The RPV shell is also subjected to higher temperatures and a steeper temperature gradient during cooldown as cooler water injected into the vessel streams down the shell wall before mixing with hotter water in the RPV to decrease the temperature inhomogeneity within the water.

As indicated previously, cladding stresses are only significant near the inner wall of these components so these stresses will be present near the inner wall of an RPV shell. Conversely, cladding stresses are insignificant near the outer wall of RPV/SG heads. Additionally, the monoblock RPV/SG heads do not contain through-thickness welds so weld residual stresses are not present. Conversely, RPV shell forgings are joined using circumferential welds which cause significant residual stresses near the weld region. Because the thermal, cladding, and weld residual stresses are higher in an RPV shell, the combined stresses during normal operations near the inner wall of the RPV shell will also be greater than the stresses near the outer wall of RPV/SG heads.

The most significant postulated accident scenarios are more severe cooldown transients than are experienced during normal operations. Such transients will increase the thermal and cladding stresses near the inner wall of the RPV shell while weld residual stresses are not strongly affected. As indicated previously, cladding stresses are insignificant near the outer wall of the RPV/SG heads. Also, for the same reasons as for normal operations, the thermal stresses near the outer wall of the RPV/SG heads will be much less than near the inner wall of the RPV shell. The difference in the combined stresses between an RPV shell and RPV/SG heads will therefore be bigger during postulated accident scenarios than under normal operations. In summary, the stresses experienced by the inner wall of the RPV shell are greater than those experienced by the outer wall of RPV/SG heads during both normal operations and postulated accident scenarios.

3.1.2.3 Probability of Cracks

There are four metal product forms that can exhibit fabrication-induced cracking within RPV shells: plates, forgings, ferritic welds, and stainless steel cladding. Figure 8 [50] illustrates the expected fabrication flaw distribution for each of these product forms. As seen in the figure, welds have a much greater density of cracks and are also more likely to have deep cracks. Deep cracks are typically the ones that challenge component integrity [50]. Plates and forgings have similar flaw densities. The crack density of plates and forgings is less than welds. Plates and forging are also much less likely than welds to have deep flaws. The cladding installation process can cause small flaws in the ferritic steel near the cladding interface. However, the flaw density associated with the cladding process is the smallest of the four product forms.

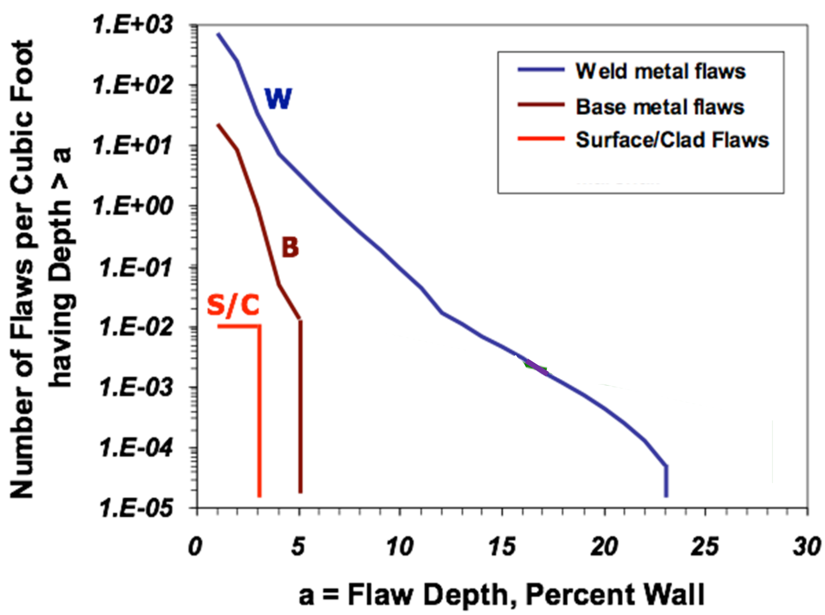



Figure 8: Flaw Distribution in Thick-Section Steels [50]. Base metal refers to plates and forgings.

The fabrication flaw density of regions with postulated CMAC is expected to be the same as regions not affected by CMAC. The fabrication flaw density and sizes are expected to be similar in both forged RPV/SG heads and forged (or plate) RPV shells. However, there will be no cladding induced flaws near the outer surface of the RPV/SG components and, because there are no significant welds, fewer weld flaws exist in these components as well. Therefore, the total flaw density near the inner surface of a welded RPV shell is expected to be much greater than near the outer surface of RPV/SG heads. Further, there is a much greater probability that RPV shell welds will contain deeper flaws that provide the greatest challenge to component integrity.

### 3.1.2.4 Material Toughness

The attribute that is most negatively affected by the possibility of positive CMAC is material toughness. Limited data indicated that a material's $RT_{NDT}$ value increases by up to 33°C for every 0.1% increase in carbon. For a component with an assumed nominal carbon composition of 0.2% and a local region of CMAC with a carbon level of 0.4% (i.e., $\Delta C/C_0$ of 100%), the $RT_{NDT}$ of this local region may be estimated to be as much as 66°C higher than the bulk of the component. Therefore, local regions within a RPV/SG head component affected by positive CMAC will have a lower toughness than the bulk of the component and the magnitude of the toughness decrease (i.e., $RT_{NDT}$ increase) will be directly correlated to the $\Delta C/C_0$ ratio. As indicated earlier, RPV shell components are not expected to be affected by CMAC. Therefore, the initial material toughness of local regions within an RPV/SG head affected by CMAC will be lower than in an RPV shell.

However, the toughness in RPV shell materials decreases due to radiation embrittlement during a reactor's service life. Radiation embrittlement is caused by neutron bombardment and it causes the material toughness to continually decrease over time. Surveillance specimens within RPVs are used to monitor radiation embrittlement to ensure that the toughness of critical materials in individual plants are following expected trends as operation continues. Radiation embrittlement is greatest near the inner wall in the RPV shell beltline region. In contrast, radiation levels in RPV/SG heads do not exceed the thresholds that lead to significant radiation embrittlement. Therefore, material toughness decreases due to radiation embrittlement is not expected in the RPV/SG heads.

The decrease in material toughness due to radiation embrittlement can occur relatively quickly. As shown in Figure 9, the increase in $RT_{NDT}$ (or $RT_{PTS}$ in Figure 9) can exceed 111°C (200°F) after 20 years of operation and it can exceed 167°C (300°F) after 60 years of operation [52]. Every U.S. PWR except for one has operated for more than 20 years and the decrease in material toughness in the RPV shell due to radiation embrittlement at most plants is significantly greater than is conservatively estimated for even the greatest amount of CMAC. In summary, while CMAC could cause the initial toughness of RPV/SG heads to be less than RPV shell material, the toughness in the RPV shell is expected to be lower in most plants after 20 years or more of operation.

After the preliminary assessment was conducted[6], the staff continued to collect information on the effect of CMAC on a material's $RT_{NDT}$ value. The staff conducted a more extensive literature review and found that the increase in $RT_{NDT}$ due to CMAC has been reported to range from 25°C to 75°C, as discussed in Section 2.2 of this report. The staff also reviewed test results that characterized the effect of CMAC on $RT_{NDT}$ for alloys similar to SA-508 during a vendor inspection, as discussed in Section 3.2.1 of this report. These test results indicated that at a carbon level of approximately 0.30wt% the bounding shift in $RT_{NDT}$ due to CMAC is 70°C and that an additional increase in the magnitude of CMAC did not produce a significant additional increase in $RT_{NDT}$.

Most recently, additional data was published from a thorough test program conducted to quantify the effect of CMAC on $RT_{NDT}$ in full-scale forgings produced by ACF. The maximum measured shift in $RT_{NDT}$ is reported to be 45°C, as described in Section 3.2.2.1 of this report.

[6] This paragraph summarized $RT_{NDT}$ data that the staff did not have access to during its preliminary assessment. The staff's estimated shift in $RT_{NDT}$ during the preliminary assessment was 66°C.

The 45°C shift in $RT_{NDT}$ is the value most applicable to the U.S. components produced by ACF and confirms the appropriateness of the 66°C shift in $RT_{NDT}$ used in the preliminary safety assessment as a practical shift for U.S. components. A shift in $RT_{NDT}$ due to CMAC of 75°C for the U.S. components produced by ACF may be considered a bounding value. It should be note that all the shifts in $RT_{NDT}$ due to CMAC reported are less than the shift expected from radiation embrittlement.

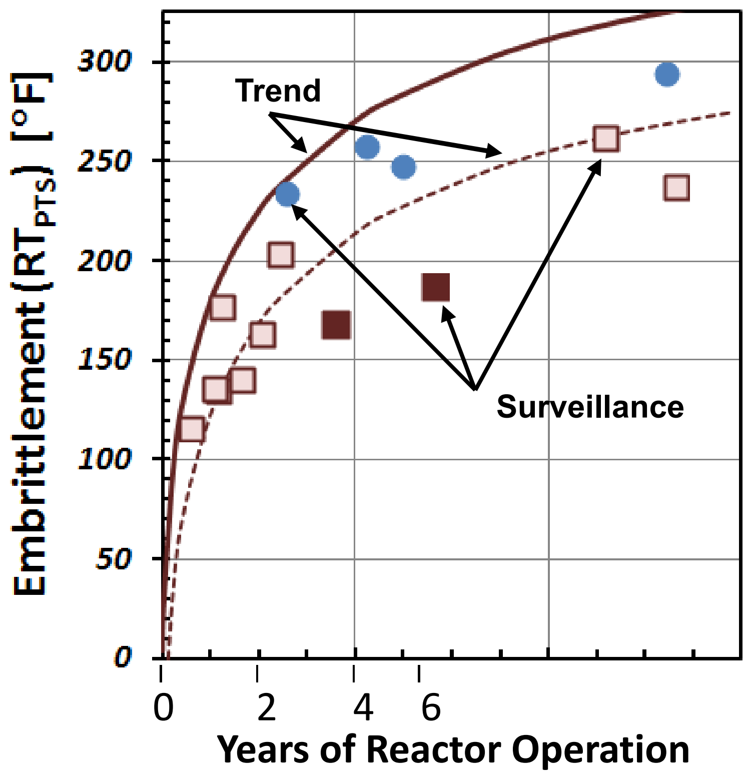



Figure 9: Increase in $RT_{NDT}$ (or $RT_{PTS}$) with Continued Operation [52].

### 3.1.3 Preliminary Assessment Summary (2016)

As discussed, a critical combination of high stresses, a sufficiently large crack (or cracks), and low material toughness are required for a component to fail. A qualitative failure comparison was used 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 may be affected by positive CMAC. The following conclusions were drawn from this comparison:

- The PRV 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 both more flaws and larger fabrication flaws. It is the larger fabrication flaws which are typically necessary to result in component failure;
- While the initial toughness of an RPV shell material may be greater than an RPV/SG head with postulated positive CMAC, the toughness of the shell decrease due to 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 [50]. 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.

## 3.2 Review of Supporting Documentation and Activities

### 3.2.1 Vender Inspection

An NRC inspection team performed a limited-scope vender inspection at the AREVA Inc. facility in Lynchburg, VA. The inspection was conducted March 27-31, 2017 and the inspection report was issued May 10, 2017 [11]. The inspection reviewed: policies and procedures that govern the facility's 10 CFR Part 21 Program for compliance; nonconformance and corrective actions policies and procedures to verify compliance with applicable Criterion of Appendix B to 10 CFR Part 50; and the available documentation associated with 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 [53], "Routine Inspections of Nuclear Vendors," and IP 36100 [54], "Inspection of 10 CFR Part 21 and Programs for Reporting Defects and Noncompliance." The inspection team did not identify any violations or nonconformances during the inspection.

The NRC inspection team reviewed the manufacturing processes used by ACF to fabricate the inservice U.S. components and the resulting mechanical properties. Representative samples for each component type and processing route used by ACF were reviewed at the generic processing level and component-specific level. The NRC inspection activities related to the material processing and properties primarily focused on:

- thermomechanical processing of the forged components to identify processing attributes capable of impacting CMAC;
- component-specific chemistry and mechanical property data to assess the level of conservatism in evaluating the impact of postulated CMAC on U.S. inservice components; and
- preliminary results generated from international test programs characterizing mechanical properties as a function of carbon content for similar alloys to better understand the empirical relationship between CMAC levels and toughness.

Documentation associated with the focus areas outlined above was reviewed, including: quality assurance data packages; manufacturing plans; internal travelers and manufacturing sheets, and certified material test reports (CMTRs). The NRC inspection team also interviewed experts from ACF and was provided with technical presentations.

The NRC inspection team reviewed the results of a generic risk assessment performed by ACF to rank components based on their likelihood of containing regions of CMAC. The ACF generic risk assessment ranked the majority of component types as nil, very low, or low likelihood of containing regions of CMAC. The NRC inspection team found it appropriate to rank these component types as low likelihood of containing regions of CMAC. Components types produced from large non-trepanned conventional ingots with a diameter of greater than approximately 2 meters were ranked “high” in the ACF generic risk assessment. The NRC inspection team did not discover any new processing or testing information pertinent to the likelihood of these component types in the U.S. containing regions of CMAC. The ACF generic ranking does not indicate that a component has a region of CMAC or imply any safety issue with the component.

The ACF generic risk assessment did not consider any component-specific information. The NRC inspection team reviewed component-specific information for a sample of U.S. components. Information from this review was used to produce Table 4 in Section 2.3 of this report. The NRC inspection team determined that the majority (approximately 70%) of components identified in Table 2 have a low likelihood of containing regions of CMAC. The NRC inspection team did not identify any components as having a high likelihood of containing regions of CMAC based on the information reviewed.

The NRC inspection team was unable to make a determination for some SG heads and tubesheets because the documentation reviewed did not contain information relevant to the component regions of interest (i.e., regions where CMAC would be expected if it were present). The documentation reviewed for these SG heads and tubesheets also did not contain information that would indicate the presence of CMAC. Simply stated, the documentation reviewed by the inspection team for these components (approximately 30%) did not independently provide information that facilitated a determination on the likelihood of these components containing regions of CMAC. It should be recognized that the documentation reviewed indicated that this material was procured to ASME material specification SA-508 which contains a requirement to discard regions of excessive segregation.

In addition to reviewing the material processing information for the U.S. components, the NRC inspection team also reviewed component-specific chemistry, tensile test, drop weight test, and Charpy impact test results. The component-specific test results most relevant to the CMAC topic are provided in Table 5. All of the test results reviewed by the NRC inspection team complied with the applicable ASME Code material requirements.

Table 5: Range in carbon content, reference temperature of nil ductility transition, and ingot discard for the component types and processing routes reviewed by the inspection team [11].

| Range | Ladle Chemistry[1] (%C) | Product Chemistry (%C) | $RT_{NDT}$ (°F) | Top Ingot Discard[2] (%) |
|---|---|---|---|---|
| Maximum | 0.21 | 0.23 | -4 | 21.5 |
| Minimum | 0.17 | 0.15 | -30 | 16.4 |

1. Carbon content in the melt is the weighted average of two ladle pours used to produce a single ingot.
2. The top ingot discard range is only for conventional ingots of 150 metric tons or greater.

Additionally, the NRC inspection team reviewed preliminary results generated by international test programs characterizing mechanical properties as a function of carbon content for alloys similar to those used to produce the U.S. components. The preliminary test results indicated that an increase of carbon from 0.18 to 0.29 percent is bounded by a $RT_{NDT}$ shift of 70°C. Additional test results indicated that a further increase in carbon content from 0.29 to 0.40 percent does not result in an equivalent further shift in $RT_{NDT}$ (i.e., the additional $RT_{NDT}$ shift from 0.29 to 0.40 percent carbon content will be less than 70°C).

The primary material processing and property observations in the inspection report are:

- A population of the components produced by ACF have a low or no possibility of containing regions of CMAC.
- Carbon levels and mechanical properties for the components reviewed conformed to ASME requirements.
- Information reviewed did not challenge the NRC's preliminary determination on the CMAC topic: that the safety significance to the U.S. fleet appears to be negligible.

### 3.2.2 Summary of work performed by ASN/IRSN

Over the course of this evaluation, the NRC staff had the benefit of access to three reports [12, 55, 56] written by both IRSN and ASN's Nuclear Pressure Equipment Department on the CMAC issues associated with the Flamanville EPR reactor pressure vessel head. These reports were developed as part of presentations to ASN's Advisory Committee of Experts for Nuclear Pressure Equipment. Each of these reports provides a summary of the issue, and plans for the AREVA analysis and experimental programs to demonstrate the serviceability of the Flamanville 3 reactor heads. The summary contained in Section 3.2.2 of this document will focus on the third report [12], which describes the AREVA program and the ASN findings.

The third report [12] discusses a review of AREVA NPs response to the CMAC issue with regard to the serviceability of the Flamanville 3 EPR RPV upper and lower heads. The AREVA program was mainly an experimental program aimed at demonstrating that the material in the EPR heads have sufficient toughness, with a low probability of fast fracture for the service condition, but also discusses inservice monitoring provisions.

#### 3.2.2.1 Material Properties

In order to characterize the material in the Flamanville 3 heads, AREVA located and procured full-scale production replicas of the heads. The following three replicas were procured:

- an upper RPV head forged for the Hinkley Point EPR in the United Kingdom;
- a lower RPV head initially forged for an EPR in a U.S. plant that was not built;
- an upper RPV head initially forged for the same EPR in the United States. This is the same head that was used in the original qualification process [55] for the Flamanville 3 EPR head where core sections were taken from the center of the head, and the first elevated carbon levels were discovered.

In order to justify the use of the three replicas to characterize the Flamanville head, AREVA compared material processing parameters and resulting mechanical properties to establish equivalency, including:

- Ingot and solidification parameters;
- Forging and forming parameters and data;
- Position of component in the reference bloom;
- Quenching and cooling rate parameters;
- Carbon contents at pouring and from qualification specimens;
- Carbon content measured on the outside surface of the components;
- Mechanical properties in the acceptance zone (outside the area of segregation).

ASN concluded that the comparisons conducted by AREVA were based on more data than is typically used in qualification and appear sufficient for justifying the use of the replica heads. However, they also noted some differences between the heads and concluded that due to these variances, conservative properties must be used in the fast fracture analysis with its conservative design margins.

As part of this test program, AREVA sectioned the three heads consistently to measure the chemistry, strength, and toughness of the three replica domes both along the perimeter of the high carbon area as well as through the thickness. AREVA conducted the following tests:

- 145 tensile;
- 96 drop-weight;
- 574 Charpy;
- 907 toughness (bend bar and CT);
- 1503 chemical analyses.

The maximum (max) carbon levels measured at the outside surfaces of each head ranged from 0.28% to 0.32%. The average (avg) and maximum carbon contents measured through the thickness of each head was:

- 1/4T position[7] = avg: 0.266%, max: 0.279%;
- 1/2T position = avg: 0.248%, max: 0.268%;
- 3/4T position = avg: 0.224%, max: 0.227%.

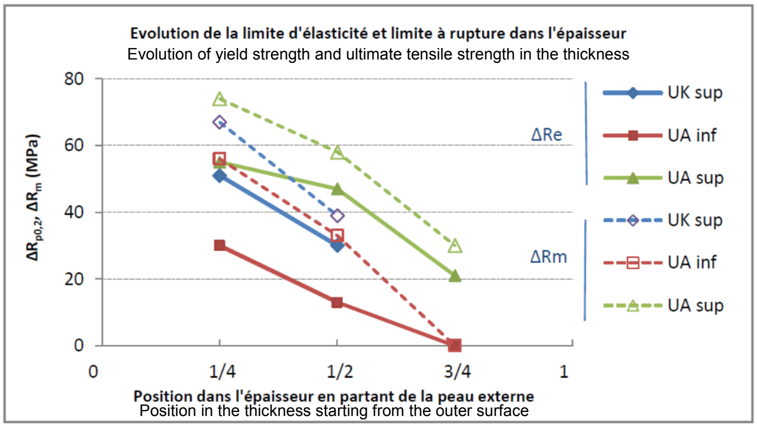
The yield strength and ultimate tensile strength were shown to increase with carbon content. The increase was nearly linear from the 3/4T to the 1/4T position through the wall thickness, as shown in Figure 10. The maximum shift in yield strength was approximately 55 MPa (8 ksi), and the maximum shift in ultimate strength was 75 MPa (10.9 ksi).

From the Charpy and toughness tests, a variety of fracture parameters were calculated, including: $T_{NDT}$, $RT_{NDT}$, $T_O$, $T_{68J}$, $K_J$, $J_{0.2mm}$. The overall trends from the fracture toughness testing suggest that the toughness decreases with increasing carbon content, as shown in Figure 11. However, the effect appears to be non-linear through the material thickness. In fact, the shift in toughness transition temperature due to carbon did not significantly change between the 1/4T and 1/2T location, but increased significantly between the1/2T and 3/4T locations. This behavior was observed in both the CVN and fracture toughness testing. AREVA attributed this behavior to the cooling rate that occurred during the quenching portion of the heat treatment process. No investigation of microstructural changes occurred to confirm this observation.

The maximum shift in $RT_{NDT}$ was 45°C (avg: 35°C), while the maximum shift in $T_0$ was 70°C (avg: 60°C). The minimum toughness curve from the RCC-M code [57] (used in France) did not bound all of the experimental toughness data from this effort. Out of the hundreds of data point developed in this effort, approximately 10 were not bounded by this curve. Shifting this curve by 20°C bound all of the data, as shown in Figure 12. It was this shifted curve that was used in the AREVA fast fracture analysis.

ASN concluded that the fracture mechanisms in the segregated area are not different than expected in ferritic steels, and the carbon content increases the strength and decreases toughness. Since vessel rupture is not postulated in the licensee's safety analysis report, ASN required a conservative analysis based on the properties measured in this program to demonstrate the integrity of the components affected by CMAC.

[7] 1/4T position refers to a location ¼ of the wall thickness from the outer diameter of the head.




$R_{p0.2}$: Tensile Yield Strength
$R_m$: Ultimate Tensile Strength

UK sup, UA inf, and UA sup identify the three full scale heads that specimen were extracted from.

Figure 10: Shift in yield strength and ultimate tensile strength at 330°C in the segregation zone relative to the acceptance zone, as a function of position in thickness of RPV heads [12].

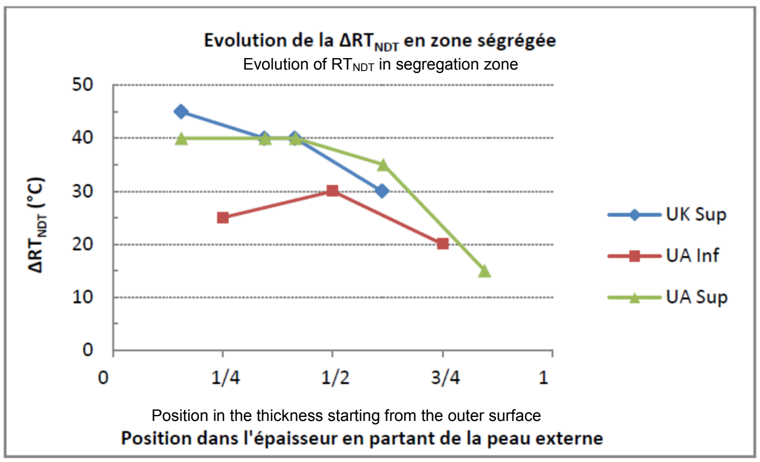



Figure 11: Evolution of the $RT_{NDT}$ in the segregation zone, as a function of position in thickness of full scale RPV heads [12].

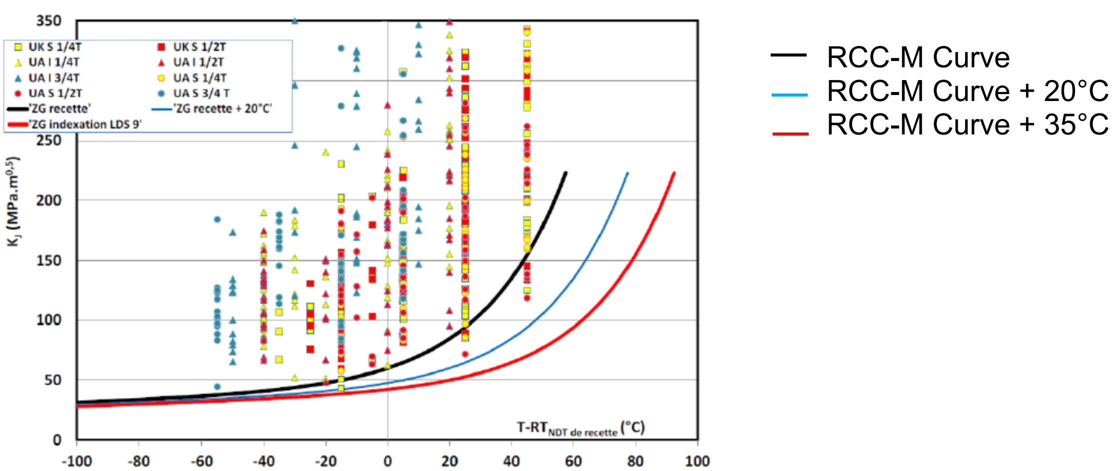



Figure 12: Fracture toughness data from the RPV head test program in relation to the RCC-M curve [12].

### 3.2.2.2 Fast Fracture Analyses

AREVA conducted a variety of fast fracture analyses to bound the problem. They conducted all deterministic analyses in accordance with RCC-M code [57] and applied full safety factors (2 for normal, 1.6 for pressure tests and emergency, and 1.2 for accident). The transient loads they considered included the following cases:

- Plant unit startup from cold shutdown to hot shutdown after refueling;
- Fluctuations between hot and cold shutdown;
- Variety of LOCA situations;
- Primary system overpressure when cold;
- Rupture of steam generator tube;
- Small LOCA;
- Inadvertent opening of PSV;
- Loss of feed water;
- Rapid cooling by secondary system.

The failure criterion used was based on linear elastic fracture mechanics and required the calculation of the crack driving force for a variety of crack sizes and locations. For this study, AREVA postulated defects in 11 locations throughout the upper head and 3 locations in the lower head. They assumed flaw sizes ranging from 10mm x 20mm to 10mm x 120mm, and considered both surface breaking and embedded flaws. For the crack driving force, AREVA used a variety of sources including the RSE-M code [58], Paris-Tada solutions [59], and location specific finite element analyses.

Using the material properties developed in their experimental program and the conditions listed above, the RCC-M code fracture analyses conducted suggested that in all cases, the margin to fast fracture with full safety factors applied was greater than 1. In fact, the margins for fast fracture ranged from:

- 1.9 to 5.8 for normal operating conditions;
- 1.1 to 4.7 for emergency conditions;
- 1.2 to 4.0 for accident conditions.

Because of these margins, ASN ruled out the risk of fast fracture for the Flamanville 3 heads.

### 3.2.2.3 Pre-service and Inservice Inspections

After the CMAC issues was discovered in the Flamanville 3 heads, ASN's Advisory Committee of Experts for Nuclear Pressure Equipment took a position that inservice inspections or other operating measures were needed to ensure defense in depth. ASN then sent AREVA a request to propose "reinforced measures for commissioning oversight, operation and inservice inspection appropriate to the situation encountered and to incorporate them into the equipment instruction manual."

In their response, AREVA stated that inservice inspections are not needed due to pre-service inspection results (no recordable defects), lack of material degradation in the head materials, and low operating stresses. However, AREVA did an inspection feasibility study to determine if it was possible to inspect these heads. They determined that for the Flamanville 3 lower head, inspection equipment is available to inspect for planar flaws to a depth of 20mm. At 36mm deep the performance of this equipment drops considerably. AREVA estimated it could fully develop this procedure in 3.5 months. For the upper head, AREVA investigated a multi-element UT approach in their feasibility study, but did not get acceptable results. AREVA concluded inservice inspection of the upper head is not feasible with the currently available techniques.

3.2.2.4 ASN Position

This section documents the decision the ASN made with respect to the CMAC issue with the Flamanville 3 heads in France. It should be noted that this decision is not relevant to the plants in the U.S., and is only provided here for information. <u>The terms defined in this section and the rationale used in their decision are based solely on the French nuclear regulations</u>.

ASN states that head design is based on four levels of defense-in-depth, per French regulations:

- First level of defense-in-depth is to prevent incidents. It is based on a high level of design and manufacture quality, and must provide a guarantee of that quality.
- Second level of defense-in-depth is to detect occurrence of such incidents, apply measures to prevent them from becoming accidents, and to restore the situation to normal.
- Third level of defense-in-depth is to control any accidents that could not be avoided, prevent them from getting worse, and return the plant to safe conditions.
- Fourth level of defense-in-depth is to manage accidents resulting from failure of first three levels, and to mitigate consequences for persons and environment.

The Flamanville EPR safety case is based on break preclusion. For the RPV heads, no reasonable provisions could be defined to mitigate the consequence of their failure as an initiating event. Therefore, ASN reviewed their defense-in-depth criteria for the heads based on the existence of CMAC. The following defense-in-depth assessment is based on the first two levels, per French regulations:

- First level of defense-in-depth:
    - ASN believes that due to documentation issues at ACF, the quality is not high for these components, and the qualification process should have identified these anomalies.
    - ASN believes that AREVA did not use the state-of-the-art and best available techniques for manufacturing these heads.
    - While ASN recognizes that the margins are sufficient against fast fracture, they are reduced due to regions of CMAC.
    - Because of these issues, ASN believes AREVA needs reinforcement of the second level of defense in depth.
- Second level of defense-in-depth:
    - AREVA demonstrated that no manufacturing flaws exist.
    - However, ASN believes that periodic verification of the absence of flaws is needed to reinforce the second level of defense-in-depth.
    - ASN believes the inservice inspection plan for the bottom head by EDF is sufficient to reinforce the second level of defense in depth.
    - ASN believes that, due to the lack of inservice inspection plan for the upper head, the long-term serviceability of the upper head cannot be confirmed. ASN

considers the use of the upper head beyond a few years of operation is not warranted, unless inservice inspections can be used to reinforce the second level of defense in depth.

ASN recommends that the serviceability of the lower head is acceptable as long as EDF conducts the proposed inservice inspection plan. However, due to the inability to conduct reasonable inservice inspection on the upper head, ASN recommends that the upper head long-term serviceability cannot be confirmed and the head shall be replaced early in the operational life of the plant. Both the upper and the lower Flamanville 3 RPV heads have sufficient margins against fast fracture.

### 3.2.3 Summary of work performed by EPRI

In response to the international CMAC topic, EPRI initiated a research program to address the safety significance of elevated carbon levels due to CMAC. This program was divided into four main tasks each aimed at developing both qualitative and quantitative information in order to make a safety determination. The four tasks included:

A. Extension of RPV probabilistic fracture mechanics (PFM) analyses to qualitatively bound other components;
B. Development of a robust technical basis to support the hypothesis that RPV integrity bounds other components;
C. Quantitative structural analyses to assess whether the results of the PFM analyses of the RPV beltline (Activity "A") bound the other forged components;
D. White paper assessing the impact of CMAC for SG tubesheets, based on expert judgment and experience with fabrication of the tubesheets as large forgings.

As of the writing of this document, Task A is complete[8] and has been publicly released as MRP-417 [13]. The other tasks are still under development with the expected release of the report(s) in the first quarter of 2018. Therefore, this section is a summary of the MRP-417 report.

#### 3.2.3.1 MRP-417 summary

The objective of MRP-417 is to address the structural significance of the potential presence of CMAC in large forged PWR 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 95$^{th}$ percentile through-wall crack frequency (TWCF) of less than 1E-6 per year (yr$^{-1}$) [60] for PTS events and a conditional probability of failure of less than 1E-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 [50].

To conduct these analyses, EPRI used probabilistic fracture mechanics (PFM) techniques to calculate the TWCFs. EPRI used the Fracture Analysis of Vessels-Oak Ridge (FAVOR) PFM computer code, which was used in the development of the technical basis for the 10 CFR 50.61a rule, for its analyses. The analysis approach consisted of the following steps:

- Determine the beltline ring forging with the maximum reference temperature at 80-years (i.e., $RT_{NDT}$ value adjusted for neutron irradiation, assuming 80-years of operation) for the population of U.S. plants.

[8] EPRI is revising MRP-417 to incorporate additional data that has been reported by the French [12].

- Determine representative carbon, copper, and phosphorus[9] macrosegregation distributions for components fabricated from large conventional ingots.
- Construct PFM models using the FAVOR code.
- Apply flaw distributions, transient loading, and normal operating loads consistent with those used in the development of 10 CFR 50.61a.
- Calculate the 95% TWCF over a range of carbon content for 80 years.
- Conduct a sensitivity study on vessel thickness.
- Account for variable uncertainty when appropriate.

In conducting the analyses EPRI incorporated a number of conservative assumptions:

- The macrosegregation distribution is assumed to extend uniformly through the thickness of the components;
- All the components have the same limiting material;
- All the ring forging have the same CMAC distribution;
- All head forgings have the same CMAC distribution;
- Transients and flaw distributions used are conservative, and follow those used in the development of 10CFR50.61a;
- Fluence levels are conservative and bound 80-years of operation.

3.2.3.1.1 Input summary

In order to define the change in toughness as a function of carbon content, EPRI used published data to develop the trends. These data sets were primarily comprised of work conducted by AREVA as part of the Flamanville effort that was available at that time, but also included other published data. In their analyses, EPRI assumed the $RT_{NDT}$ changes per the following expression:

$$RT_{NDT(U)} = RT_{NDT(Uo)} + K\Delta C$$

Where $RT_{NDT(U0)}$ is the unirradiated reference temperature, K is the change in reference temperature as a function of carbon content, and $\Delta C$ is the change in carbon content.

EPRI determined that K=560 F/wt%°C with a standard deviation of 41.7 F/wt%°C was an appropriate value to use based on all of the acquired data. As a comparison, data published in the 1990s [33] suggests a shift in reference temperature of 73°C for a carbon content increase from 0.17% to 0.4% which corresponds to a K=571 F/wt%°C. However, the maximum shift in reference temperature was only measured to be 45°C [12], as described in Section 3.2.2 of this report.

In order to model the CMAC in these large components, EPRI relied on published information on how the positive macrosegregation mapped in these components. Many of the public references were from France, and were related to work done by AREVA. In their study, EPRI defined the increase in carbon content as compared to the nominal ingot value. For the ring forgings, EPRI assumed that the maximum carbon content was 25% higher than the nominal value, while for the closure heads, it was assumed to be 100% higher.

[9] Copper and phosphorous are used in determining the embrittlement behavior of the steel.

The flaw distribution, transient loading, and normal operating load inputs were consistent with those used in the development of 10 CFR 50.61a. In addition, the analyses were conducted for both a thick (~8.45 inch) and thinner (~6.7 inch) walled vessel.

3.2.3.1.2 Analysis results

Prior to conducting detailed PFM analyses, EPRI conducted a series of deterministic analyses to identify the component, load, and irradiation conditions that would have the largest impact on the 95% TWCF. They conducted fracture analyses on the RPV shell, RPV head, pressurizer head, and SG channel head using a variety of typical transient loadings both with and without irradiation effects, similar to those used in the development of 10CFR50.61a. Their results suggest that the RPV shell, RPV head, and the SG channel head may be limiting, therefore they focused the PFM analyses on these components.

In all cases, with the assumed carbon distribution[10], the calculated 95% TWCF and conditional[11] probability of failure (CPF) were less than 1E-6 $yr^{-1}$ and 1E-6, respectively. For the carbon content assumed, the maximum CPF was ~5E-8. Given the low CPF values, EPRI calculated the CMAC values (i.e., $\Delta C/C_0$) needed to exceed the acceptance criterion. The following describes the limiting CMAC values needed to exceed the 95% TWCF/CPF criteria for the shells:

- Thick-wall RPV shells
  - o During normal RPV cooldown the limiting component is the nozzle shell course. A CMAC level of approximately 65%, which is 2.6 times the assumed CMAC value, is needed to exceed the risk criteria.
  - o During PTS the limiting component is the vessel beltline. A CMAC level of approximately 62%, which is 2.5 times the assumed CMAC value, is needed to exceed the risk criteria.
- Thin-wall RPV shells
  - o During normal RPV cooldown the limiting component is the nozzle shell course. A CMAC level of approximately 160%, which is 6.4 times the assumed CMAC value, is needed to exceed the risk criteria.
  - o During PTS the limiting component is the vessel beltline. A CMAC level of approximately 125%, which is 5 times the assumed CMAC value, is needed to exceed the risk criteria.

In addition, the RPV closure, RPV bottom, and SG channel head PFM results suggested there was more margin on the required CMAC values as compared to the vessel shell results. The following describes the CMAC values needed to exceed the 95%TWCF/CPF criteria for the heads:

- Thick-wall RPV heads
  - o During normal RPV cooldown the limiting component is the closure head. A CMAC level of approximately 325%, which is 3.25 times the assumed CMAC value, is needed to exceed the risk criteria.
  - o During PTS the limiting component is the closure head. A CMAC level of approximately 260%, which is 2.6 times the assumed CMAC value, is needed to exceed the risk criteria.
- Thin-wall RPV heads

[10] $\Delta C/C_0$ of 100% in the heads and 25% in shells with $C_0$ of 0.17 wt% C.
[11] Conditional on the assumed transients.

- During normal RPV cooldown the limiting component is the closure head. A CMAC level of approximately 400%, which is 4 times the assumed CMAC value, is needed to exceed the risk criteria.
- During PTS the limiting component is the closure head. A CMAC level of approximately 400%, which is 4 times the assumed CMAC value, is needed to exceed the risk criteria.

- SG channel heads
  - During a normal cooldown event the, a CMAC level of 300%, which is 3 times the assumed CMAC value, is needed to exceed the risk criteria.

EPRI MRP-417 concludes that the results from this work suggest 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 PWRs.

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 CPF and 95%TWCF were identical to that used in the development of 10CFR50.61a;
- Major probabilistic inputs, such as flaw distribution, standard material properties, transients, and normal operating conditions were identical to that used in the development of 10CFR50.61a;
- The CMAC distribution and toughness relationships used were based on historical literature and empirical data;
- Assumptions made using the FAVOR model were consistent with or conservative as compared to those analyses used in the 10 CFR 50.61a development.

The NRC assessment of MRP-417 for this report does not constitute a regulatory endorsement of its contents. The NRC staff will assess the other EPRI report(s) on the CMAC topic in the same manner, as they become available. These EPRI reports/tasks were described in Section 3.2.3 of this report.

## 3.3 Risk-Informed Decision Making Process

NRR Office Instruction LIC-504 [7], Integrated Risk-Informed Decision-Making Process for Emergent Issues," provides the NRC staff with guidance for evaluating and communicating risk-informed decisions. Risk-informed decision making considers risk insights together with other information (e.g., deterministic evaluations, inspections results, operating experience, and expert knowledge) to reach a determination on the safety significance of a topic. The LIC-504 guidance may be employed when a topic being evaluated does not fit into an existing NRC regulatory process. The LIC-504 instruction explicitly states that the process described for decision making is guidance and not intended to be procedural requirements. Additionally, the LIC-504 instruction recommends that the level of analysis and documentation used in the decision making process should be commensurate with the safety significance of the topic under review. The LIC-504 instruction is intended to be tailored to the specific topic being reviewed to increase NRR efficiency and effectiveness.

The evaluation of the CMAC topic does not efficiently fit into an existing NRC regulatory process; therefore, the guidance in LIC-504 is being used as a framework for the staff's decision making process. The five key principles of risk-informed decision making are:

1. Compliance with Existing Regulations;
2. Consistency with the Defense-in-Depth Philosophy;
3. Maintenance of Adequate Safety Margins;
4. Demonstration of Acceptable Levels of Risk;
5. Implementation of Defined Performance Measurement Strategies.

The guidance provided in LIC-504 recommends that the five key principles of risk-informed decision making be used to differentiate among the options being considered. It is also recognized in the LIC-504 instruction that all of the key principles may not be applicable to each option considered. The five key principles of risk-informed decision making, as they relate to the current understanding of the CMAC topic are described below.

*Principle 1 - Compliance with Existing Regulations*:

As discussed in Section 2.4 of this report, the presence of CMAC in RCS components may impact a plants compliance with NRC regulations, including:

- 10 CFR Part 21, "Reporting of Defects and Noncompliance"
- 10 CFR Part 50.55a, "Codes and standards"
- 10 CFR Part 50.60, "Acceptance Criteria for Fracture Prevention Measures for Lightwater Nuclear Power Reactors for Normal Operation"
- Appendix A to Part 50, "General Design Criteria (GDC) for Nuclear Power Plants"
    - GDC 1, "Quality Standards and Records"
    - GDC 14, "Reactor Coolant Pressure Boundary"
    - GDC 31, "Fracture Prevention of Reactor Coolant Pressure Boundary"
- Appendix B to Part 50, "Quality Assurance Criteria for Nuclear Power Plants and Fuel Reprocessing Plants"
- Appendix G to Part 50, "Fracture Toughness Requirements"

The NRC regulations address the integrity of RCS components by: (1) formally endorsing industry Codes/Standards, such as the ASME Code, and requiring compliance with these Codes/Standards as part of the NRC regulations; (2) specifying component properties and performance acceptance criteria that must be met; (3) specifying component/system design criteria that must be met; and (4) requiring that a robust quality assurance program is applied to manufactured components.

The requirement most applicable to the prevention and/or identification of CMAC is 10 CFR Part 50.55a, which incorporates by reference ASME Section III. Subsection NCA-1220 of ASME Section III establishes that metallic materials shall be manufactured to an "SA" specification or other permitted material specification. The requirements of material specification SA-508 are provided in ASME Section II. Section 4.3.1, "Discard," of material specification SA-508 states: "Sufficient discard shall be made from each ingot to secure freedom from piping and excessive segregation." The requirement to discard a sufficient portion of the ingot is directly applicable to the type of CMAC being evaluated.

10 CFR Part 21 requires licensees and their component supplies to notify the NRC if they become aware of information reasonably indicating that a basic component contains defects that could create a substantial safety hazard. A region of CMAC in a component that does not comply with the requirements stated in the material specification and/or procurement documentation may be considered a defect. A substantial safety hazard would include a defect in a component that has a function of barrier integrity of the RCPB.

Licensees and/or their component suppliers are responsible for performing an evaluation to determine if a defect poses a substantial safety hazard, per 10 CFR Part 21. If the evaluation concludes that the defect or deviation does not pose a substantial safety hazard, the evaluation is documented and the process is complete. The documentation associated with this activity shall be available for NRC inspection. If the evaluation cannot be completed within 60 days, the NRC shall be formally notified of the status of the evaluation in an interim report. If the evaluation concludes that there is a substantial safety hazard the NRC shall be formally notified within two days. The NRC has not received a 10 CFR Part 21 notification from a component supplier or licensee associated with CMAC.

*Principle 2 - Consistency with Defense-in-Depth Philosophy*:

A factor for assessing how an emergent issue might degrade defense-in-depth is to see how it affects the balance among the layers of defense. The aspect of defense-in-depth that may be impacted by the potential presence of CMAC in RCS components is "barrier integrity." The RCPB is one of three independent fission product release barriers in a U.S. plant. The NRC has determined that acceptable failure probabilities for RPV integrity are a 95% TWCF of less than 1E-6 $yr^{-1}$ for PTS events [60], and a CPF of less than 1E-6 for normal reactor cooldown events [61]. The acceptable failure probabilities for the RPV bounds all applicable RCS components. The PFM evaluation, summarized in Section 3.2.3 of this report, demonstrated that the 95% TWCF and CPF are less than 1E-6 $yr^{-1}$ and 1E-6, respectively for the conditions evaluated. Therefore, this topic did not significantly change the failure probability of any individual barrier. Additionally, if CMAC is postulated in components produced by ACF, the CMAC would be located in an isolated region near the surface of a component, decrease in magnitude through the affected region as it moves away from the surface, and terminate before reaching half the thickness of the component. As a result, if a crack were to initiate in a region of CMAC, the crack would propagate into material with increasing resistance to crack growth (i.e., increased toughness). It is reasonable to assume that a through-wall crack that initiated in a location of maximum CMAC would arrest, limiting the crack extension. Breaches in the RCPB of the size postulated are bounded by design basis calculations, such as large-break loss-of-coolant accident (LOCA) events and main steam line breaks. Therefore, if the RCPB was compromised by a crack resulting from CMAC, the remaining two independent barriers (i.e., fuel cladding and containment) to the release of fission products would still provide adequate defense-in-depth. Because the CMAC topic does not significantly impact the failure probability of the RPV and the remaining fission product release barriers remain intact, defense-in-depth is maintained.

*Principle 3 - Maintenance of Adequate Safety Margins*:

The presence of positive CMAC in a component would increase the strength and decrease the toughness in the effected region. The decrease in toughness properties ($K_{IC}$, USE, $RT_{NDT}$) would reduce the safety margins. Both deterministic and PFM evaluations have determine that sufficient safety margin is maintained in RPV components with regions of postulated CMAC. The safety margins for the RPV bound all applicable RCS components. The deterministic evaluation performed by ASN/IRSN is summarized in Section 3.2.2 of this report. The PFM evaluation performed by EPRI is summarized in Section 3.2.3 of this report. The deterministic and PFM evaluations concluded that the margins against fast fracture are adequate for the conditions evaluated. Therefore, the potential CMAC issue is not expected to significantly degrade safety margins.

*Principle 4 - Demonstration of Acceptable Levels of Risk*:

The primary purpose of the risk assessment is to determine whether the risk of the potential presence of CMAC in isolated regions of RCS components is sufficient to create an imminent safety concern, such that prompt action must be taken to provide reasonable assurance of adequate protection of the public health and safety. The total change in risk may be determined quantitatively or qualitatively, as appropriate. It is conservative to assume that a through-wall crack will lead to core damage and that core damage will lead to a large early release, since plants are designed to mitigate a loss of coolant accident. This conservative assumption negates the necessity for additional PRA models. A Large Early Release Frequency (LERF) value of 1E-4 $yr^{-1}$ has been established as a guideline for taking immediate actions [7]. Assuming that a through wall crack leads to a large early release results in the LERF value being bounded by the acceptable 95% TWCF value. The PFM evaluation summarized in Section 3.2.3 of this report demonstrated that the 95% TWCF is less than 1E-6 $yr^{-1}$ for the conditions evaluated. Since this failure probability is below the LERF criteria described above, the staff has concluded that no immediate safety concern exists.

*Principle 5 - Implementation of Defined Performance Measurement Strategies*:

There are two basic strategies that could be used to monitor for the presence of CMAC in a component. The first strategy would be to directly measure the carbon levels in the applicable regions of the components that potentially contain CMAC. Carbon levels would need to be measured using a nondestructive method because degradation to the component resulting from a destructive examination technique would exceed that resulting from CMAC, thus making the inspection counterproductive. The challenges associated with nondestructively measuring carbon content on inservice RCS components include: there are no qualified techniques; there are no qualified personnel; and the deviations in carbon content of interest are relatively small (e.g., 0.01 weight percent). The second strategy would be to indirectly monitor for the effects associated with the potential presence of CMAC. The second method would use traditional nondestructive techniques to monitor for cracking. Many of the RCS components are currently monitored for degradation in accordance with inservice inspection (ISI) and aging management program (AMP) requirements.

Given that there is no indication that the U.S. in-service components produced by ACF are noncompliant with the applicable regulations, and there is reasonable assurance that defense-in-depth, safety margins, and risk levels are adequately maintained, the current monitoring programs at the plants are adequate; additional performance measurement strategies are not warranted. The NRC staff, however, will continue to monitor the U.S. nuclear industry and international activities related to the CMAC topic to ensure that any new information is analyzed to determine if additional performance measurement strategies are needed.

### 3.4 Options Considered

Plants in the U.S. are operating with forged components produced by ACF. Internationally, CMAC has been discovered in nuclear components produced by ACF. The four options outlined in Table 6 and described below have been considered to address the potential impact of the international CMAC OE on the U.S. operating fleet. A reasonable attempt was made to align the options considered with those proposed by public stakeholders [62].

Table 6: Overview of the four options considered to address the CMAC topic for the U.S. operating fleet and associated evaluation criteria for each.

| # | Option | Evaluation Criteria |
|---|---|---|
| 1 | Monitor and Evaluate | This option would be appropriate if the issue is not an imminent safety concern and the evaluation determines:<br>• adequate defense-in-depth is maintained;<br>• sufficient safety margin is maintained;<br>• an acceptable level of risk is maintained;<br>• the adequacy of defense-in-depth, safety margin, and risk level have a degree of conservatism that provides reasonable assurance that the potential safety impact of postulated CMAC is bounded. |
| 2 | Issue a Generic Communication | This option would be appropriate if the issue is not an imminent safety concern and the evaluation determines:<br>• adequate defense-in-depth is maintained;<br>• sufficient safety margin is maintained;<br>• an acceptable level of risk is maintained;<br>• additional information is needed to establish that the aforementioned assessments have an adequate degree of conservatism;<br>• additional information is needed to make a regulatory decision. |
| 3 | Issue Orders Requiring Inspections | This option would be appropriate if the issue is not an imminent safety concern but one of the following are true:<br>• there is not reasonable assurance that sufficient/adequate defense-in-depth, safety margin, or level of risk is maintained;<br>• the condition is getting progressively worse over a time period relative to the inspection period. |
| 4 | Issue Orders Suspending Operation | This option would be required if an imminent safety concern were identified, such as the following:<br>• defense in depth is significantly degraded.<br>• there is significant loss of safety margin.<br>• CDF is greater than or on the order of 1E-3 $yr^{-1}$<br>• LERF is greater than or on the order of 1E-4 $yr^{-1}$ |

### 3.4.1 Option 1: Monitor and Evaluate

The first 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 it cannot be concluded that there is reasonable assurance of structural integrity and the potential safety risk is acceptably small, additional action(s) will be considered. The NRC will communicate with applicable stakeholders, as appropriate.

The five key principles of risk-inform regulation, as they relate to the Option 1, are assessed below:

*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, per Appendix B to 10 CFR Part 50. Section 2.4 and Section 3.3 of this report discusses the regulatory and technical requirements pertinent to the CMAC topic in the U.S.

The NRC has not received a 10 CFR Part 21 notification from a component supplier or licensee associated with CMAC. For that reason, it is concluded that the requirements and regulations affected by the potential presence of CMAC in a component have been verified to be in compliance or; alternatively, an evaluation has been performed which concluded that the condition is not adverse to safety. Therefore, Principle 1 is satisfied for Option 1.

*Principle 2 - Consistency with Defense-in-Depth Philosophy*:

As stated in Section 3.3 of this report, the aspect of defense-in-depth of relevance to the potential presence of CMAC in regions of RCS components is "barrier integrity." The RCPB is one of the three principle fission product release barriers in a U.S. plant. A 95% TWCF of less than 1E-6 $yr^{-1}$ [60] and CPF of less than 1E-6 [61] have been established as acceptable failure probabilities, based on the RPV integrity being bounding. The conservative assessment performed by EPRI [13] showed that the probability of compromising the barrier integrity function for the inservice U.S. components of interest are significantly below these acceptance levels, as described in Section 3.2.3.1 of this report. Based on its review of the work performed by EPRI and preliminary safety assessment provided in Section 3.1 of this report, the NRC staff has reasonable assurance that the CMAC discovered internationally in components produced by ACF would not compromise the barrier integrity function of U.S. plants. Additionally, if the RCPB was compromised, the remaining two independent fission product release barriers would still provide adequate defense-in-depth, as discussed in Section 3.3 of this report. The NRC has reasonable assurance that the 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 impact the safe operation of the inservice components being evaluated. The ASN/IRSN work experimentally established relationships between carbon levels and mechanical properties. This empirical data was used in a series of deterministic fracture analysis. The ASN/IRSN evaluation determined that the safety margin against fast fracture is maintained in all conditions analyzed. The EPRI work determined that the CMAC levels necessary to be considered significant to safety are more than 200% of those observed in components. Based on its review of the evaluations performed by ASN/IRSN and EPRI, the NRC has reasonable assurance that the 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*:

It is conservative to assume that a through-wall crack will lead to core damage and that core damage will lead to a large early release, as discussed in Section 3.3 of this report. A result of this conservative assumption is that the LERF value becomes bounded by the acceptable 95% TWCF value of less than 1E-6 $yr^{-1}$. The LERF value is much less than 1E-6 $yr^{-1}$ for the conditions evaluated, as discussed in Section 3.2.3 and Section 3.3 of this report, indicating that there is no immediate safety concern. Therefore, Principle 4 is satisfied for Option 1.

*Principle 5 - Implementation of Defined Performance Measurement Strategies*:

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

### 3.4.2 Option 2: Issue a Generic Communication

The second option involves issuing a generic letter (GL) to the licensees operating with forged components produced by ACF. The objective of the GL is to confirm that the licensees Appendix B quality assurance program verified that the components produced by ACF are compliant with the applicable NRC regulations and ASME Code requirements. The GL would request that the licensees provide: (a) the documentation necessary to confirm that the components in question met all applicable NRC regulations and ASME Code requirements; and (b) a description of how their Appendix B quality assurance program 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. A review of the regulatory requirements and the 10 CFR 50 Appendix B program, as they relate to the CMAC topic, is provided in Section 2.4 of this report. The licensees are required to submit a written response to a GL, in accordance with 10 CFR 50.54(f).

The five key principles of risk-inform regulation, as they relate to the Option 2, are assessed below:

*Principle 1 - Compliance with Existing Regulations*:

The NRC has not received a 10 CFR Part 21 notification from a component supplier or licensee associated with CMAC. Issuing a GL would provide the NRC with information similar to that which would be used in the 10 CFR Part 21 evaluation to determine that either: (a) the CMAC was appropriately addressed during the manufacturing of the components, and therefore, is compliant with no defect or deviation to be evaluated; or (b) the defect or deviation is not safety significant and therefore, the NRC does not need to be notified. 10 CFR Part 21.41 permits the NRC to inspect records and activities applicable to the Part 21 process. The NRC reviews Part 21 evaluations routinely as part of the Revised Oversight Process (ROP). Specifically, Inspection Procedure 71152, "Problem Identification and Resolution" provides guidance to review licensee evaluations to ensure that potential supplier deviations are adequately captured to ensure potential defects are identified and addressed. A review of the Part 21 process is also part of the vendor inspection program.

A GL or focused inspection would both be effective in confirming that the licensees Appendix B quality assurance program verified that the components produced by ACF are compliant with the applicable NRC regulations and ASME Code requirements. Therefore, Principle 1 is satisfied for Option 2.

*Principle 2 - Consistency with Defense-in-Depth Philosophy*:

The GL would request that licensees with components produced by ACF confirm that CMAC has not compromised adequate defense-in-depth. As stated in Section 3.3 of this report, the aspect of defense-in-depth of relevance to the potential presence of CMAC in regions of RCS components is "barrier integrity." The assessment performed by EPRI [13] showed that the potential of compromising the barrier integrity function for the inservice U.S. components of interest are significantly below the acceptance levels of a 95% TWCF of less than 1E-6 $yr^{-1}$ and CPF of less than 1E-6, as described in Section 3.2.3.1 of this report. Based on its review of the work performed by EPRI and preliminary safety assessment provided in Section 3.1 of this report, the NRC staff has reasonable assurance that the CMAC discovered internationally in components produced by ACF would not compromise the barrier integrity function of U.S. plants.

The information collected in the response to a GL would not be expected to be useful for revising the 95% TWCF or CPF calculations. The evaluations performed by ASN/IRSN and EPRI have an appropriate degree of conservatism to bound the potential impact on the barrier function of U.S. components with postulated regions of CMAC. As a result, the defense-in-depth determination would remain unchanged. Additionally, if the RCPB was compromised, the remaining two independent fission product release barriers would still provide adequate defense-in-depth, as discussed in Section 3.3 of this report. Therefore, Principle 2 would be satisfied for Option 2.

*Principle 3 - Maintenance of Adequate Safety Margins*:

The GL would also request that licensees with components produced by ACF confirm that adequate safety margins are maintained. A region of CMAC in a component could reduce the margin against fracture. The ASN/IRSN work used experimentally established data to perform a series of deterministic fracture analysis. The ASN/IRSN evaluation determined that the safety margin against fast fracture is maintained in all conditions analyzed. The EPRI work determined that the carbon levels necessary to be considered significant to safety are more than 200% of those observed in components. Based on its review of the evaluations performed by ASN/IRSN and EPRI, the NRC has reasonable assurance that the U.S. plants with components produced by ACF maintain sufficient safety margins.

The information collected in the response to a GL would not be expected to be useful for revising the margins calculated by ASN/IRSN or EPRI. The evaluations performed by ASN/IRSN and EPRI have an appropriate degree of conservatism to bound the potential impact on the barrier function of U.S. components with postulated regions of CMAC. As a result, the safety margins determination would remain unchanged. Therefore, Principle 3 would be satisfied for Option 2.

*Principle 4 - Demonstration of Acceptable Levels of Risk*:

A GL would request that licensees with components produced by ACF confirm that an acceptable level of risk is maintained at the plant. It is conservative to assume that a through-wall crack will lead to core damage and that core damage will lead to a large early release, as discussed in Section 3.3 of this report. A result of this conservative assumption is that the LERF value becomes bounded by the acceptable 95% TWCF value of less than 1E-6 $yr^{-1}$. It has been established in the evaluation of Principle 2 that the 95% TWCF criteria has been met. Therefore, Principle 4 is satisfied for Option 2.

*Principle 5 - Implementation of Defined Performance Measurement Strategies*:

The GL would request that licensees with ACF components either provide confirmation that existing monitoring/inspection activities are adequate or describe the additional monitoring/inspection activities necessary to ensure that any adverse consequences are detected and corrected, as applicable. Therefore, Principle 5 is satisfied for Option 2.

### 3.4.3 Option 3: Issue Orders Requiring Inspections

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 is 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 adverse to safety or nonconformance to requirements, the plant cannot restart until the issue is addressed and approval is granted by the NRC.

The five key principles of risk-inform regulation, as they relate to the Option 3, are assessed below:

*Principle 1 - Compliance with Existing Regulations*:

CMAC is to be addressed during manufacturing, prior to the component being put into service. As a result, inspections focused on interrogating components for regions of CMAC would be conducted prior to installation. Therefore, there are no regulations and requirements to inspect inservice components for CMAC. Additionally, the NRC has not received a 10 CFR Part 21 notification from a component supplier or licensee associated with CMAC.

As part of an NRC Order, licensees would be required to conduct inspections and analysis to confirm that all components produced by ACF are in compliance with existing regulations, such as: 10 CFR Part 50.55a, 10 CFR Part 50.60; GDC 1, 14, and 31; and Appendix G to Part 50. Therefore, Principle 1 is satisfied for Option 3.

*Principle 2 - Consistency with Defense-in-Depth Philosophy*:

Performing nondestructive examinations of the inservice components produced by ACF would confirm the condition of the applicable locations and potentially the carbon levels. The conservative assessment performed by EPRI [13] showed that the probability of compromising the barrier integrity function for the inservice U.S. components are significantly below the acceptance levels. This assessment was performed using postulated regions of CMAC that are likely to bound any realistic condition found in an inservice U.S. component. Additionally, ASN/IRSN [12] performed destructive testing to characterize the impact of CMAC on mechanical performance. Performing destructive testing on inservice U.S. components would only confirm the thorough testing program conducted in France.

The information collected by performing nondestructive examinations of the inservice components would further confirm the degree of conservatism in the ERPI assessment and reaffirm the data reported by ASN/EPRI. However, the examination would not be expected to impact the defense-in-depth determination. Additionally, if the RCPB was compromised, the remaining two independent fission product release barriers would still provide adequate defense-in-depth, as discussed in Section 3.3 of this report. Therefore, Principle 2 is satisfied for Option 3.

*Principle 3 - Maintenance of Adequate Safety Margins*:

Performing nondestructive examinations of the inservice components produced by ACF would confirm the condition of the applicable locations and potentially the carbon levels. The ASN/IRSN evaluation deterministically concluded that the safety margin against fast fracture is maintained in all conditions it analyzed. The EPRI work determined that the levels of carbon necessary to be considered safety significant are improbably high.

The information collected by performing nondestructive examinations of the inservice components would further confirm the degree of conservatism in the assessments performed by ASN/IRSN and EPRI. However, the examination would not be expected to impact the safety margins determination. Therefore, Principle 3 is satisfied for Option 3.

*Principle 4 - Demonstration of Acceptable Levels of Risk*:

Performing nondestructive examinations of the inservice components produced by ACF would confirm that CMAC has not degraded material condition/performance resulting in an unacceptable increase in the total risk to the plant. It is conservative to assume that a through-wall crack will lead to core damage and that core damage will lead to a large early release, as discussed in Section 3.3 of this report. A result of this conservative assumption is that the LERF value becomes bounded by the acceptable 95% TWCF value of less than 1E-6 $yr^{-1}$. It has been established in the evaluation of Principle 2 that the 95% TWCF criteria has been met. The information collected by performing nondestructive examinations of the inservice components would further confirm the degree of conservatism. Therefore, Principle 4 is satisfied for Option 3.

*Principle 5 - Implementation of Defined Performance Measurement Strategies*:

Nondestructive examinations and analysis required by this order would be used to further confirm the degree of conservatism in the assessments performed by ASN/IRSN and EPRI. Therefore, Principle 5 is satisfied for Option 3.

#### 3.4.4 Option 4: Issue Orders Suspending Operation

The forth option is identical to Option 3, except the NRC issues Orders that require immediate plant shutdown and inspections. The LIC-504 instruction is explicit in stating that, if at any time it is determined that an immediate shutdown of a plant is required, the process shall not interfere with timely action. The LIC-504 guidance defines a risk significant enough to warrant immediate action as a LERF on the order of 1E-4 $yr^{-1}$. It has been conservatively assumed that a through-wall crack will lead to core damage and that core damage will lead to a large early release. A result of this conservative assumption is that the LERF value becomes bounded by the acceptable 95% TWCF value of less than 1E-6 $yr^{-1}$. Therefore, Option 4 is dismissed without a discussion of the evaluation of the five key principles of risk-inform regulation since the potential safety significance of CMAC does not warrant a suspension of operations.

### 3.5 Final Assessment

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

Option 1, "Monitor and Evaluate," provides reasonable assurance that plants operating with components produced by ACF can continue to operate with a negligible safety impact. The follow-on actions for this option involves the:

- U.S. licenses continuing to fulfill all actions currently required by their licensing bases and applicable regulations (e.g., 10 CFR Part 50.55a, 10 CFR Part 50.60, Appendix A, B, and G to 10 CFR Part 50, and 10 CFR Part 21 evaluation and reporting requirements).
- NRC headquarters (HQ) staff continuing to monitor international and domestic activities associated with the CMAC topic to ensure that new information is reviewed to determine if any additional actions or evaluations are needed. The staff is aware of additional work being performed by EPRI[12] and ASN/IRSN[13] on the CMAC topic. The staff will review this information as it becomes available. The staff will also maintain communication with the international community, U.S. industry, and other stakeholders, as appropriate. The staff is in communication with ASME regarding the clarification/interpretation of the text

[12] EPRI is revising MRP-417 to incorporate additional data and completing the tasks identified in Section 3.2.3 of this report.
[13] ASN/IRSN has an ongoing project to evaluate the impact of CMAC on SG channel heads.

in SA-508[14]. The response to these requests will be evaluated and may prompt additional activities within ASME Code committees.

- NRC resident inspectors and Regional Offices continuing the execution of their duties (e.g., ROP activities).

The staff recommends Option 1 based on the: material and processing information reviewed by the staff during the vender inspection of AREVA Inc. [11]; experimental data and evaluation reported by ASN/IRSN [12]; PFM analyses conducted by EPRI [13]; staff review of the open literature on CMAC in steel ingots and its impact on performance; and 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 [6]: that the safety significance of CMAC to U.S. plants is negligible and no immediate action is warranted. 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 re-evaluate the need for additional actions. The additional actions may include aspects of the other options discussed in this document.

Options 2, "Issue a Generic Communication," reinforces the regulatory determination made in Option 1 by issuing a GL requesting the documentation and evaluations performed by licenses and their component supplies to conclude that the components produced by ACF do not have defects or deviations that pose a substantial safety hazard. The information collected in the response to a GL would not be expected to change any of the conclusions reached in Option 1, including those related to defense-in-depth, safety margins, or risk level determination. Additionally, affected licensees have been informed of the CMAC topic by the relevant vendors. Vendors and licensees must meet their Part 21 evaluation and reporting responsibilities should the condition warrant such action. As part of the ROP and vendor inspection program, these evaluations are reviewed for adequacy. Therefore, Option 2 is not recommended.

Option 3, "Issue Orders Requiring Inspections," 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 information collected by performing nondestructive examinations of the inservice components would not be expected to significantly impact the defense-in-depth, safety margins, or risk level determinations made in Option 1. Given that the safety significance of postulated regions of CMAC in U.S. components was determined to be negligible in Option 1, the results of these inspections would not be expected to change the safety determination. Therefore, Option 3 is not recommended.

Options 4, "Issue Orders Suspending Operation," involves the same activities as Options 3; however, it entails the immediate shutdown of the plants operating with components from ACF. The NRC risk criteria to shutdown a plant are not met; therefore, Option 4 is not warranted or recommended.

## 4.0 CONCLUSION

The preliminary safety assessment performed by the NRC staff in 2016 concluded that no immediate action was warranted and the safety significance of CMAC to U.S. plants is negligible. This determination was based on the following considerations, which were affirmed by subsequent information:

[14] ASME interpretation requests 17-1587 and 17-1589.

- Extent of Condition: Internationally, CMAC has only been found in components produced by ACF using a specific processing route. Based on the staff's knowledge of material processing, it was concluded that, only a subset of the U.S. components produced by ACF may have used the processing route that resulted in the CMAC found in international plants. It was later confirmed [11] that many of the U.S. components produced by ACF did not use this processing route. Therefore, a significant population of the U.S. plant components produced by ACF are assessed to have either a low or no possibility of containing regions of CMAC. The staff's review of the manufacturing processes used by ACF to fabricate inservice U.S. components and the resulting mechanical properties is documented in NRC Inspection Report No. 99901359/2017-201 [11] and discussed in Section 3.2.1 of this report. Section 2.3 also provides a more generic discussion of the material processing.
- Degree of Condition: If it is postulated that CMAC is present in a component it will occur in a localized region of the forged component. CMAC is not a bulk material phenomena and does not go through thickness. This staff position is well documented by results published in open literature and was verified by the ongoing international test programs characterizing the CMAC present in components produced by ACF [12,55,56], as discussed in Sections 2.2 and 3.2.2, respectively. Also, based on the staff's knowledge, it was concluded that the highest levels of CMAC observed internationally, to this point, were representative of the magnitude that would be expected in postulated regions of U.S. components. This staff position on the magnitude of the CMAC was affirmed by a review of the open literature which indicated that the maximum CMAC postulated in the components of interest should not exceed 60% or a maximum of 0.40wt% carbon; and that the accompanying shift in $RT_{NDT}$ should not exceed 75°C, as discussed in Section 2.2 of this report. Additionally, international test programs characterizing the impact of CMAC present in components produced by ACF measured a maximum shift in $RT_{NDT}$ of 45°C and an average shift of 35°C [12]. Therefore, the magnitude of the postulated CMAC estimated by the NRC was conservative.
- Qualitative Analysis: An overview of the qualitative failure comparison that was conducted during the preliminary assessment is provided in Section 3.1 of this report. This qualitative analysis reflected the NRC staff's estimates of extent of condition and degree of condition, and also considered the most likely failure locations of the components relative to the portion potentially affected by CMAC. The likelihood of failure for the monoblock RPV head and SG channel head was compared to that of the RPV shell in the beltline. The RPV beltline shell was chosen because its probability of failure is well understood and extremely low. This comparison was performed using the three relevant attributes of component failure: combined stresses, crack size, and material toughness. The qualitative analysis concluded that the probability that failure of a monoblock RPV head and SG channel head are extremely low, and bound by that of the RPV shell, as summarized in Table 8. The conclusion of this qualitative analysis is consistent with the more detailed deterministic analysis conducted by ASN/IRSN [12] and a PFM analysis performed by EPRI [13].

The staff's final safety determination is consistent with its preliminary safety determination: the safety significance of CMAC to U.S. plants is negligible and no immediate action is warranted.

Table 8: Attributes affecting the likelihood of component failure that were assessed during the staff's preliminary safety determination and subsequently affirmed by supporting information.

| | Failure Location | Combined Stresses | Likelihood of Cracks | Material Toughness |
|---|---|---|---|---|
| RPV Shell (Beltline) | Inner Surface | **Higher** | **Higher** | **Lower** |
| RPV Monoblock Head | Outer Surface | Lower | Lower | Higher |
| SG Channel Head | | | | |
| **Bold** text indicates the more conservative attribute | | | | |

Additionally, the staff considered four options to address the potential impact of CMAC on the U.S. operating fleet. These four options were evaluated using the five key principles of risk-informed decision making, in accordance with the LIC-504 instruction. The staff recommends Option 1 "Monitor and Evaluate," as discussed in Section 3.5 of this report.

In conclusion, the staff:

- has determined that the safety significance of CMAC to U.S. plants is negligible and no immediate action is warranted.
- recommends that the CMAC topic continue to be monitored and new information be evaluated, per Option 1.

## 5.0 RECOMMENDATION

The staff recommends Option 1, "Monitor and Evaluate," to address the CMAC topic, as it relates to the safe operation of U.S. plants. The NRC staff will continue to monitor the domestic and international activities associated with the CMAC topic to ensure that new information is evaluated to determine if additional action is warranted. The NRC staff will maintain communication with the international community, U.S. industry, and other stakeholders, as appropriate.

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