Response to Request for Additional Information – ANP-10285P "U.S. EPR Fuel Assembly Mechanical Design Topical Report" (TAC No. MD7040)

RAI-4. *Please provide AREVA's fuel rod gas pressure criteria and evaluation specific to the U.S. EPR fuel assembly design; topical report (ANP-10285P).*

Response 4:

Section 5.1.8 of the topical report (TR) provides a description of the fuel rod gas pressure criterion.

The methodology and criterion are those specified in BAW-10183PA (TR Reference 16) and as assessed for M5™ cladding in BAW-10227PA (TR Reference 7). The COPERNIC fuel rod performance computer code was used to determine the bounding rod internal pressure with consideration given to the manufacturing and modeling uncertainties (as detailed in Section 12.1 of BAW-10231PA (TR Reference 17). The applicability of the COPERNIC code to the U.S. EPR fuel rod configuration was reviewed and approved by the NRC in the Final Safety Evaluation Report (FSER) for AREVA NP TR ANP-10263P, "Codes and Methods Applicability Report for the U.S. EPR," [1] (TR Reference 2). Section 6.1 of ANP-10263P addresses the use of M5 cladding and Section 6.3 addresses the fuel rod gas pressure limits for the U.S. EPR fuel rod.

Specifically, Section 6.3 of ANP-10263P states the following criterion for the U.S. EPR fuel rod design:

> "The criterion is as follows: The internal pressure of the peak fuel rod in the reactor is limited to a value below that which would cause (1) the fuel-cladding gap to increase due to outward cladding creep during steady-state operation and (2) extensive departure from nucleate boiling propagation to occur. The report gives a proprietary limit on the difference between the maximum fuel rod internal gas pressure and the nominal reactor coolant system pressure.
>
> The purpose of the first limit is to prevent the opening of the fuel-cladding gap, which could degrade heat transfer and thereby lead to excessive fuel temperatures, excessive internal rod gas pressures due to fission gas release, and excessive cladding stresses and strains. Compliance with this limit is determined by using a fuel rod performance code. The criterion has been approved (Reference 6-4) for use with COPERNIC. When pin pressure exceeds system pressure, the code evaluates whether the rate of outward cladding creep exceeds the rate of fuel pellet swelling."

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[1] See Letter, Getachew Tesfaye (NRC) to Ronnie L. Gardner (AREVA NP Inc.), "Final Safety Evaluation Report for Topical Report ANP-10263(P), 'Codes and Methods Applicability Report for the U.S. Evolutionary Power Reactor (U.S. EPR)' (TAC NO. MD2803)," August 8, 2007.

The U.S. EPR fuel rod design has been evaluated to these criteria and acceptable performance has been established to greater than **[ ]** which exceeds the burnup limit of 62 GWd/mtU in BAW-10186PA (TR Reference 10).

RAI-5. *Please provide AREVA's statistical fuel assembly hold down methodology specific to the U.S. EPR fuel assembly design; topical report (ANP-10285P).*

Response 5:

AREVA NP currently uses two different approaches for evaluation of fuel assembly liftoff – a deterministic approach and a statistical approach. The liftoff analysis performed as part of the design evaluation of the U.S. EPR fuel assembly utilized the deterministic methodology to determine spring deflections and loads. This methodology uses the algebraic sum of extreme variations in core plate separation and fuel assembly length and growth when determining spring deflections. The lift evaluation was performed by comparing the resulting minimum spring forces and the minimum fuel assembly wet weight with the maximum flow lift forces. Using the deterministic methodology, fuel assembly liftoff will not occur during normal operating conditions for a full core of U.S. EPR fuel assemblies. As indicated in Section 5.1.9 of the TR, a minimum of **[ ]** of margin to liftoff was shown to exist at end-of-life (EOL) hot full power (HFP) condition assuming LTL fuel assembly growth, UTL core plate separation, and LTL fuel assembly length based on design tolerances.

The alternative statistical holddown methodology was reviewed and approved by the NRC for PWR fuel designs in BAW-10243PA (TR Reference 22). Applicability of this method to the U.S. EPR fuel assembly design was evaluated and approved by the NRC in the FSER for ANP-10263P. As noted in Section 5.1.9 of the TR, AREVA NP may use this method to evaluate the holddown performance of the U.S. EPR design in the future.

RAI-6. *Regarding Section 3.5 and 4.5, Debris Filter (FUELGUARD™), please provide operational experience and test data on pressure drop of the FUELGUARD™ debris filter.*

Response 6:

Operational data for the FUELGUARD™ bottom nozzle are in TR Tables 4-2 through 4-7, spanning six different fuel assembly designs, which include the following plant type/bundle array: Westinghouse 17x17, 15x15, and 14x14; CE 15x15 and 14x14; B&W 15x15. A total of 6,374 fuel assemblies using the FUELGUARD bottom nozzle have been irradiated worldwide.

The form loss coefficient for the FUELGUARD bottom nozzle was not calculated directly from the Hermes-P pressure drop test data. Pressure drop measurements for the upper and lower nozzles, end grids, and core plates cannot be accurately obtained from full bundle flow tests due to the proximity of these components to each other.

The loss coefficients for the nozzles, upper and lower end grids are calculated using pressure drop measurements for the inlet and outlet groups: each of the groups consists of the core plate, nozzle, and end grid. The outlet group may be configured with a thimble plug, upper structure with CRGT or simply have the thimble and instrument tubes plugged. The 14 foot prototype bundle tested had had a thimble plug component for the test.

The FUELGUARD bottom nozzle loss coefficient is calculated as follows:

The loss coefficient for the inlet group is calculated from the measured pressure drop data; the coefficients for the lower core plate and HMP lower end grid are known values obtained from previous pressure drop testing.

Because the loss coefficient of the bottom nozzle is fairly constant over the Reynolds number range from 200,000 up to 500,000 which is representative of hot operation, a single value obtained from the curve fit to the data is recommended for analysis. Values of the FUELGUARD loss coefficient at Reynolds numbers <200,000 are obtained from a curve fit of the data.

The loss coefficient values of the FUELGUARD bottom nozzle are:

Tube region flow area basis: 38.035 in^2

The FUELGUARD bottom nozzle as a function of Reynolds number is shown in the below figure:

[ ]

Figure RAI 6-1: U.S. EPR FUELGUARD Bottom Nozzle

RAI-7. *Regarding Section 3.6, M5™ Alloy Guide Tube, please provide operational experience and test data of the MONOBLOC™ guide tube design. Identify differences between the MONOBLOC™ and the standard dashpot GT design. Is MONOBLOC™ guide tube design approved for use by the NRC?*

Response 7:

More than 17,000 assemblies with the MONOBLOC™ configuration have been irradiated in 62 reactors worldwide, more than 3,000 of those assemblies use alloy M5. The maximum assembly burnup achieved using the MONOBLOC guide tube design is **[ ]**. In the United States, eight lead assemblies comprised of M5 MONOBLOC guide tubes in a welded cage construction have been inserted at TVA's Sequoyah site. Fuel assemblies using the MONOBLOC guide tube design in conjunction with reduced excess hold-down load exhibit reduced fuel assembly distortion based on Post Irradiation Examination (PIE) inspections, compared to fuel assemblies using the standard swaged guide tube design.

The MONOBLOC guide tube design for the U.S. EPR fuel assembly has undergone various types of tests. Those tests where the effects or influence of the MONOBLOC guide tube design are most apparent in the resulting data are described below.

Regarding rod cluster control assembly (RCCA) drop time, testing was performed using a full-scale mockup as described in TR Section 3.8.7. The full array of 24 guide tubes in a representative fuel assembly cage resulted in a nominal RCCA drop time of **[**

] (CRDM technical specification limit requires trip time to be less than or equal to 3.5 seconds). It should be noted that drop time is a function of the dynamic response of the entire RCCA drive line, which includes the RCCA, drive rod, CRDA (Control Rod Drive Assembly), and control rod drive mechanism (CRDM) and related components.

Other tests performed include FA prototype tests for FA axial and lateral stiffness (CALVA Bench) tests. The axial and lateral stiffness of the U.S. EPR fuel assembly is illustrated in Figure RAI 7-1 and Figure RAI 7-2 below, respectively. The response of these tests include the influence of the MONOBLOC guide tube design as well as the spacer grid attachment method, spacer grid/fuel rod interface, i.e., slip loads.

Figure RAI 7-1: Static Axial Compression Test

Figure RAI 7-2: Static Lateral Bending Test

The primary difference between the MONOBLOC and the standard dashpot GT designs is the exterior configuration (Figure RAI 7-3). The MONOBLOC design reflects a constant outer diameter (OD) and two inner diameters (ID). The upper-section ID is larger to accommodate rapid RCCA insertion. The ID decreases over a short transition zone to the smaller ID in the lower section of the guide tube creating the dashpot region and resulting in an increased wall thickness. The standard swaged GT design, on the other hand, has a constant wall thickness that is mechanically reduced forming a short conical section creating the dashpot region. TR Table 3-4 presents a dimensional comparison between the U.S. EPR MONOBLOC versus the standard guide tube design.

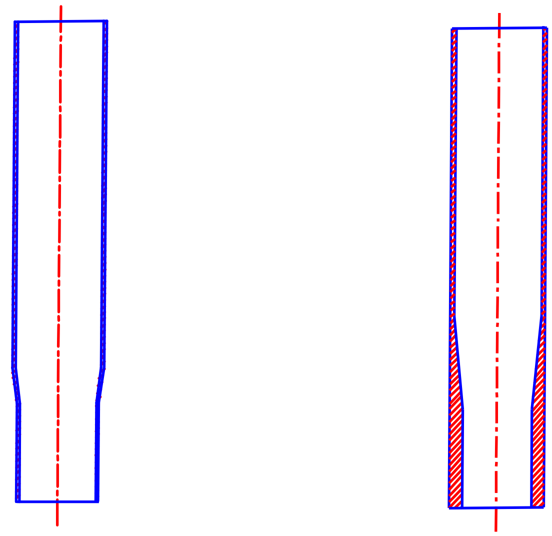

Figure RAI 7-3: Standard Dashpot (left) and MONOBLOC™ (right) Guide Tube Configurations

Per BAW-10227P-A, the use of M5 for structural materials is approved by the NRC. The structural design of the U.S. EPR fuel assembly with the MONOBLOC guide tube design follows NRC approved methodologies; however, no specific geometrical configuration of the MONOBLOC guide tube was explicitly approved by the NRC.

RAI-8. *In Section 5.1.3 of ANP-10285P, it is stated that the fatigue usage factor is well below the limit. Please discuss the relative conservatism of the U.S. EPR fuel fatigue data and methodology relative to the fatigue safety factors in SRP 4.2.II.A1 (b).*

Response 8:

To provide a conservative design, the O'Donnell-Langer design fatigue curve used in the fuel rod cladding fatigue analysis applies a safety factor of two on stress amplitude or 20 on the number of cycles, whichever is most conservative. This conservatism complies with SRP Section 4.2.II.A1(b).

RAI-9. *Regarding Section 5.1.4, Fretting and 14 ft U.S. EPR fuel assembly flow-induced vibration testing:*

a) *Provide justification for the AREVA's assumption that fretting behavior is independent of fuel assembly length;*

b) *Discuss the impact of in-reactor dimensional changes (e.g. fuel rod growth, etc.) on the adequacy of the laboratory testing of un-irradiated samples.*

Response 9:

a) Extensive testing, analysis, and operating experience demonstrate that rod-to-grid fretting and wear is primarily a function of span lengths and grid support properties. As noted in TR Section 5.1.4, “The fretting behavior is independent of fuel assembly length since the key observed parameters governing fretting resistance are associated with rod support characteristics, cross-flow velocities, and lower span distance and materials.”

Assembly vibrations, as measured by grid vibration amplitudes indicating collective motion of all the rods, are usually very small. This is because the random motions of all the rods from turbulence will very rarely coincide at one point in space and time. Individual rod vibrations can be much larger and are at a much higher frequency. It is the relative motion between a rod and its grid supports that has the potential to produce fretting wear. Thus it is the rod vibrations which primarily control or cause fretting wear.

Similar span lengths produce similar rod natural frequencies and mode shapes. However, the mode shapes are well separated between the upper and lower spans of an assembly. For example, an individual rod can vibrate freely in the lower spans without vibrating at all in the upper spans, and vice versa. Such responses are controlled by the fluid flows and turbulence loading in the individual spans. Analytical evaluations and laboratory testing demonstrate that this characteristic behavior of the 12 foot assemblies is also applicable for the longer 14 foot U.S. EPR assemblies. The average span length in the U.S. EPR assembly is shorter, and the bottom span is much shorter, than the corresponding spans in the 12 foot assemblies. Thus, the fretting performance of the U.S. EPR fuel design is expected to be comparable or better than that of the shorter 12’ HTP designs currently in operation.

b) For the above reasons, small changes to the assembly length (less than 1 inch increase due to irradiation effects) do not affect the fretting performance of the design. The small increases in rod lengths due to irradiation growth are negligible with respect to the FIV performance.

The worst case radial dimensional changes would be the situation where the rod diameter might shrink, at some point in time, relative to the grid cell support diameter. This could temporarily leave a gapped support. A gapped condition such as this was specifically tested in the German auto-clave wear test for evaluation of the U.S. EPR rod FIV responses.

Those tests were conducted in an auto-clave **[** **]** The rod was excited by an electro-magnet to amplitudes well beyond any reasonable extrapolation of the actual measured rod responses in the flow tests. For all test cases, the upper **[** **]** grids were gapped to about **[** **]**

in. For the other tests, the bottom [ ] was gapped to about [ ] in. Those gaps were based on the expected worst case rod diameter shrinkage, combined with spacer grid material irradiation relaxation, and are considered to provide bounding conditions. Thus, the laboratory testing performed adequately addresses the expected fuel dimensional changes due to irradiation.

RAI-10. *In Section 5.1.4, it is stated that, "Span average cross-flow velocities shall be less than 2 ft/sec., per Reference 3." Please justify the cross-flow velocities assumption of less than 2 ft/sec for the U.S. EPR fuel design.*

Response 10:

Reference to the "2 ft/sec" criterion will be removed from Section 5.1.4. The first paragraph of Section 5.1.4 will be revised as follows:

"Design Criterion

A full core analysis of the U.S. EPR fuel demonstrated span average cross-flows less than [ ]. Acceptable resistance to fuel rod fretting has been demonstrated historically by preserving a span average cross-flow velocity below 2 ft/sec., approved in Reference 3, in conjunction with traditional 1000 hour endurance testing. The cross-flow velocities were determined using the NRC-approved LYNXT code per Reference 18, which established the flow and pressure drop characteristics of the U.S. EPR fuel assembly for full core implementation."

The current basis for FIV performance and fretting behavior are tests and their supporting benchmarks and dynamic analyses.

- Extensive operating experience in the US with 12 foot HTP assemblies.
- Extensive operating experience in Europe with 12 foot HTP assemblies.
- Extensive operating experience in Europe with 14 foot AFA assemblies.
- Extensive flow testing of these assemblies in the German PETER loop.
- Extensive wear testing of these assemblies in the German auto-clave.

Further details on these tests and their relevance to the verification and validation of the U.S. EPR HTP FIV and fretting performance are provided in TR Section 5.1.4.

RAI-12. *Regarding Section 5.2.2, Cladding Collapse:*

a) Address the impact of burnable poison (Gadolinium) on the predicated creep collapse life of the fuel rod.

b) *Discuss the impact of in-reactor operational and thermal condition changes (e.g. temperature, pressure, stress, etc.) on the adequacy of the CROV input assumptions of un-irradiated fuel.*

c) *Address whether the CROV code predicted fuel rod (with and without Gadolinium) creep collapse lifetime include the 0.9 analysis multiplier.*

Response 12:

[ ]

RAI-14. *Regarding Section 5.3.1, Cladding Rupture, please discuss the requirements on cladding embrittlement for a standard U.S. EPR plant.*

Response 14:

Embrittlement requirements for cladding include the following:

- As the cladding is irradiated, the requirement that the cladding be able to sustain a 1 percent strain without failure as the result of an anticipated operational occurrence (AOO) is one such embrittlement requirement. The ability of M5^{TM} cladding to do so is documented in Section 3.4 of BAW-10227PA. Embrittlement requirements are also contained in 10 CFR 50.46. The ECCS acceptance criteria or limitations of keeping the cladding temperature below 2200°F and the local combined oxidation and corrosion layer below a 17 percent ECR (equivalent cladding reacted) are specific to assuring that the cladding, post LOCA, retain a degree of ductility (i.e., is not brittle). These requirements are imposed by regulation on cladding materials licensed within the United States and specific calculations are performed to assure compliance.

- The failure of cladding at the moderate to low temperatures used in the biaxial burst tests that confirm the claddings ability to support 1 percent circumferential strain is frequently referred to as rupture or burst. Such rupture or burst is prevented for AOOs by calculations of plant responses to AOOs showing that the cladding experiences less than 1 percent strain. Section 5.1.2 of the TR provides this demonstration. The occurrence of rupture during LOCA takes place at relatively high temperatures (1400 ^{o}F and above) at which the cladding is quite ductile. LOCA rupture predictions and modeling are generally coupled with the prediction of clad ballooning and its effect on core cooling. A discussion of these issues is provided to the response to RAI-16.

RAI-15. *Regarding Section 5.3.2, Violent Expulsion of Fuel, please discuss the requirements for violent expulsion of fuel during reactivity accident for a standard U.S. EPR Plant.*

Response 15:

As discussed in TR Section 5.3.2, the requirements on violent expulsion of fuel during a reactivity insertion accident are addressed in the plant-specific safety analyses. Section 1.C.ii and Appendix B of SRP 4.2 discuss the expulsion of fuel as a result of large and rapid deposition of energy in the fuel. Additionally, the methodology for analysis of the control rod ejection reactivity addition accident is described in ANP-10286P, Rev 0, “U.S. EPR Rod Ejection Accident Methodology Topical Report.” This methodology incorporates the criteria in SRP Section 4.2.

RAI-16. *Regarding Section 5.3.3, Fuel Rod Ballooning, please discuss the requirements for fuel rod ballooning for a standard U.S. EPR plant.*

Response 16:

The requirements for fuel rod ballooning for the U. S. EPR are the same as the requirements for U. S. operating plants with M5 cladding.

During a LOCA it is possible for the decrease in system pressure to establish an outward hoop stress that, in combination with the decrease in cladding strength caused by elevated temperatures, can lead to cladding swelling (ballooning) and rupture. The ballooning and rupture behavior of Zirconium based cladding is a complex phenomena. The models incorporated in LOCA predictions are often alloy dependent and based on experimental data. The U.S. EPR fuel pin uses M5 cladding. The swelling and rupture model for M5 utilizes data from the EDGAR facility in Saclay, France. The model follows the form of NUREG-0630 correlations and was originally reviewed and approved in BAW-10227PA. BAW-10227PA documents a description of the EDGAR facility; the testing procedure, measurements and data; and the development of the requisite correlations. The implementation of the M5 swelling and rupture model within the S-RELAP5 small break LOCA evaluation model is documented and approved in BAW-10240PA (TR Reference 8) and EMF-2100P Revision 6, “S-RELAP5 Models and Correlation Code Manual,” (Reference RAI 16.1). The EMF-2103(P)(A), Rev 0, Realistic Large Break LOCA Methodology for Pressurized Water Reactor Reactors (Reference RAI 16.2) does not incorporate a swelling and rupture model.

The impact of clad swelling and rupture on the LOCA transient is encompassed in: flow channel blockage and coolant diversion; adjustments to heat transfer due to increased turbulence, interphase heat transfer, and clad surface area; and the decrease in gap conductance due to the increase in the clad to pellet gap dimension. The impact of these effects differs according to the nature of the LOCA, small break and large break, requiring separate considerations and explanations.

<u>Realistic Large Break LOCA Analysis (RLBLOCA)</u>

For large break LOCA evaluations the U.S. EPR applies the AREVA RBLOCA methodology, EMF-2103PA. The impact of fuel pin swelling and rupture for large breaks is, as reported in EMF-2103PA, beneficial in that the cladding temperatures at and above the rupture location would be reduced if a swelling and rupture model were incorporated. In reality, the swelling and rupture of a single pin in an otherwise undisturbed fuel assembly accomplishes the following:

- At the location of ballooning, the clad pulls away from the pellet, increasing the pellet/clad gap resistance and reducing the energy delivered to the clad until the pellet temperature is increased to offset the effect of the gap change.

- At the location of ballooning, the fuel pin area is increased by the amount of local strain in the cladding, enhancing the heat transfer rate to the coolant.

- At and just down stream of the location of ballooning, turbulence induced by the balloons interruption of the fuel pin coolant channel flow area enhances both convective and, when liquid droplets are present, interphase heat transfer. This provides a direct benefit by lowering the cladding temperature and through a reduction of the coolant vapor temperature. The vapor temperature effect extends downstream of the ballooned location.

- At the location of ballooning, flow from the coolant channel surrounding the pin is momentarily diverted to adjacent channels. This effect reduces the convective heat transfer from the pin.

Swelling and rupture within a fuel assembly initiates at the hottest pin with the swelling and rupture of the remaining pins of the assembly following rapidly. The first two ballooning effects apply generically to each pin. The introduction of increased turbulence in the assembly flow is a stronger effect for a ruptured assembly than for a single pin because the flow has more local blockages with which to interact. Interphase heat transfer is enhanced for the same reason and the effect of liquid droplets interacting with physical blockages is substantially increased. The effect of flow diversion is reduced. Individual pins will not only swell and rupture at different times but, while mostly constrained to one grid span, balloons and ruptures will spread out axially. The net effect is a high level of mixing with flow diverted from one channel replaced quickly by an adjacent balloon just downstream. Thus, within the bundle the beneficial effects of swelling are enhanced and the detrimental effects diminished.

The net impact of swelling and rupture for large breaks has been shown to be beneficial in calculation simulations, (Reference RAI 16.2, Appendix B2) and in the FLECHT-SEASET reflood experiments (Reference RAI 16.3). Therefore, it was considered conservative not to include the effects of swelling and rupture within the RLBLOCA methodology of EMF-2103. Because it is conservative to ignore the effect, there are no swellings and rupture requirements for the evaluation of large break LOCA.

<u>Small Break LOCA (SBLOCA) Fuel Rod Ballooning</u>

The effects of swelling and rupture during a SBLOCA are essentially the same as for large break except that, for the most part, the liquid droplet mechanical interaction which desuperheats steam does not apply. There are two representative causes for cladding temperature increases during SBLOCA; the core uncovery during loop seal clearing and, the core uncovery during core boiloff prior to the accumulator actuation. During both of these periods the coolant flow near the hottest cladding is steam with no water droplets. Without water, there can be no desuperheating of the vapor coolant. The gap effect, heat transfer area increase, turbulence in the coolant, and flow diversion still apply.

The S-RELAP SBLOCA evaluation methodology in EMF-2100P diverts flow from the hot assembly to neighboring assemblies based on the amount of blockage and returns an appropriate flow above the blockage region based on cross flow calculations. The approved model details are in Sections 12.8 and 12.9 (for M5 clad) of EMF-2100P and Section 5.1.14 of TR Reference RAI 16.3.

The effects of swelling and rupture are included within the SBLOCA evaluation as a 10 CFR 50.46 Appendix K requirement. As previously explained the model is experimentally based in the determination of the conditions of rupture and in the determination of the appropriate fuel pin strain used for the prediction of heat transfer and flow diversion. The model follows the form of NUREG-0630 and is based on experimental results presented in BAW-10227PA. NRC approval for the specific S-RELAP implementation was provided in EMF-2100P and EMF-2103PA.

References:

RAI 16.1 EMF-2100P Revision 6, *S-RELAP5 Models and Correlation Code Manual*, Framatome NP Inc., August 2002.

RAI 16.2 EMF-2103(P)(A), Rev 0, Realistic Large Break LOCA Methodology for Pressurized Water Reactor Reactors,

RAI 16.3 M. J. Loftus, et. Al, “PWR FLECHT SEASET 163-Rod Bundle Flow Blockage: Task Data Report,” No. 13, NUREG/CR-3314, EPRI NP-3268, WCAP-10307, October 1983.

RAI-19. *Describe the manufacturing process which will be in place to ensure that Niobium concentration in the M5™ cladding and structure material are within the specified range.*

Response 19:

The current AREVA NP clad and structural tube material specifications limit niobium within the range of **[ ]** percent by weight. The current approved manufacturing process for M5 tubing materials requires each ingot to undergo multiple vacuum arc melting (minimum **[ ]** melts) which provides for a homogeneous distribution of alloying elements throughout the ingot.

The sampling plan for niobium content of cladding must be in accordance with ASTM B350, which requires a minimum of three samples per ingot. Alternatively, the specification allows two analyses on finished tubes per tube lot. The current approved ingot sampling frequency includes five analyzed samples from each ingot, with sampling location distributed evenly from top to bottom along the ingot length. Niobium is analyzed by ICP (inductively coupled plasma) methods.

The current material specifications and approved procedures are maintained through the AREVA NP quality control process and design change process.