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

**SAFETY EVALUATION REPORT**
**Docket No. 71-9372**
**Model No. TN-B1**
**Certificate of Compliance No. 9372**
**Revision No. 1**

## SUMMARY

By letter dated November 18, 2016, AREVA Inc. (the applicant), requested an amendment to Certificate of Compliance (CoC) No. 9372, for the Model No. TN-B1 transportation package. The applicant provided a consolidated application with changes made to reflect (i) the addition of the ATRIUM 11x11 boiling water reactor (BWR) fuel assemblies or (ii) a maximum of 50 loose fuel rods in two protective cases (25 fuel rods may be claimed together or loaded individually) or (iii) 60 loose fuel rods in two 5-inch pipes, each holding up to 30 fuel rods.

The application was supplemented February 17, August 29, December 21, 2017, and March 27, 2018 in response to staff's requests for information dated May 15 and November 16, 2017.

AREVA, Inc. also requested, by letter dated January 17, 2018, a name change to Framatome, Inc. on this Certificate of Compliance.

NRC staff reviewed the applicant's request and found that the package meets the requirements of 10 CFR Part 71.

## EVALUATION

### 1.0 GENERAL INFORMATION

The applicant did not propose any changes to the packaging design. The Model No. TN-B1 is a rectangular box, comprised of an inner and outer container both made of stainless steel where polyethylene cushioning material, placed inside the inner container, provides protection for the fuel assembly. Wood and honeycomb resin impregnated kraft paper act as shock absorbers.

The Model No. TN-B1 package is used to transport a maximum of two unirradiated Boiling Water Reactor (BWR) fuel assemblies or individual rods (BWR, Uranium Carbide, or PWR fuel rods) contained in a cylinder, protective case, or bundled together and positioned in one or both sides of the inner container. The Model No. TN-B1 package is designed for shipment by either truck, ship, or rail, as either a Type B(U)F or Type AF package.

The applicant defined the cladding of the ATRIUM-11 assembly, and the end caps, as the containment boundary in the structural evaluation during drop scenarios (Report FS1-0025122, Rev. 1). The staff noted that this is an unusual approach for the shipment of fresh unirradiated fuel, with specific materials review guidance not available in the pertinent standard review plan

(NUREG-1609); thus, the staff requested additional details about the design, fuel rod acceptance and qualification tests for the ATRIUM-11 fuel rods and end caps.

The contents of the package are unirradiated fuel with a maximum of two fuel assemblies being placed within the packaging. The quantity of radioactive materials, Type B quantity of radioactive material, and the isotopes and A2 fractions of the contents are described in Tables 1-2, 1-3, and 1-4 of the application. The contents of the package rely on gadolinia loading for criticality control based on enrichment.

The applicant provided updates to (i) Table 1-2, “Quantity of Radioactive Materials (Type A and Type B) to include the Type 11x11 fuel, (ii) Table 1-3, “Type B Quantity of Radioactive Material,” (iii) Table 1-4, “Isotopes and A2 Fractions”, and (iv) Table 1-5 to add Zirconium Alloy to reflect the current fuel assembly design characteristics. Staff noted that, in Table 1-3, the maximum content value of U-238 of 9.23 $10^{-1}$g/gU is based upon a maximum U-235 concentration of 5%. For concentrations less than 5%, the U-238 value will be higher. This is why, in the CoC, the U-238 maximum content is shown as "Balance of Uranium".

Framatome exclusively uses Zircaloy-2 as the base material in the manufacturing of BWR fuel cladding. The applicant states that the fuel rod cladding can optionally have a thin liner on the inside diameter that may have a lower material strength in comparison to non-liner clad. As the fuel rod clad is the primary containment boundary, the least bounding clad design is evaluated as part of the safety analyses.

The liner thickness is dependent upon the fuel rod design For the ATRIUM 11 fuel rod design, the clad liner is 0.09mm thick (approximately16% of the nominal clad wall thickness). The applicant states that this thickness is excluded from the clad ID when calculating the clad hoop stress for verifying compliance to the Table 3-5 footnote in the application.

For the structural analysis, the liner thickness is included as part of the cladding wall thickness. Since the liner clad may have lower the material strength in comparison to non-liner clad, the model takes this into consideration by using the material properties for liner clad.
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The applicant states that the maximum internal pressure of fuel rods, at room temperature, is conservatively limited by an allowable clad stress of 10.18 MPa, and that fuel rod designs must have a maximum pressure times the maximum clad inside radius to thickness ratio - P(r/t) - of 10.18 MPa or less to meet hypothetical accident conditions (HAC) requirements. Thus, all fuel to be shipped must have a maximum P(r/t) of 10.18 MPa or less. This is reflected as a condition in the CoC. The staff notes that the thickness of the liner in liner cladding shall be excluded when determining radius and thickness.

Shipment of 8x8, 9x9 and 10x10 fuel designs from other fuel manufacturers than Framatome is authorized. However, shipment of 11x11 fuel designs is limited to fuel manufactured by Framatome since its qualification report is specific to the fuel rod design, equipment, tooling setup, weld parameters, etc., and its results or limits are not transferable to other designs, equipment or suppliers.

Any 11x11 fuel assembly to be shipped within the Model No. TN-B1 package must meet all of the requirements detailed in the application, including, but not limited to, the requirements of Table 6-1 “TN-B1 Fuel Assembly Loading Criteria” and those detailed in Section 6.12 “Appendix B: 11X11 Fuel Assembly Criticality Analysis”.

## 2.0 STRUCTURAL AND MATERIALS EVALUATION

The objective of the structural review is to verify that the structural performance of the package meets the requirements of 10 CFR Part 71 for normal conditions of transport (NCT) and HAC conditions.

### 2.1 Structural Evaluation

The structural design of the Model No. TN-B1 package was not modified as part of this amendment request. Therefore, there were no changes to the structural analyses of the package.

However, with the addition of a new ATRIUM 11x11 fuel design, the applicant provided two calculation packages: (i) FS1-0015328 (Revision 2.0) "Structural Analyses of the AREVA Atrium-11 LTA Fuel Assembly in the RAJ-II Container during Normal and Accident Transport Conditions", and (ii) FS1-0025122 (Revision 1.0) "AREVA TN-B1 ATRIUM-11 Fuel Assembly Shipping Container Drop Analyses".

The first calculation package is an update from the report, NSA-DAC-AREVA-14-01 “Structural Analyses of the AREVA Atrium-11 LTA Fuel Assembly in the RAJ-II Container during Normal and Accident Transport Conditions”, which was previously reviewed and accepted by the staff for the Atrium-11 LTA shipment, and it includes an appendix to better evaluate the pitch changes of the fuel bundle after a 30-foot end drop. The staff noted that this appendix was used as an input for criticality evaluations. The staff reviewed the FS1-0015328 (Revision 2.0) report and confirmed that the results of the structural analyses, using LS-DYNA, met the minimum factor of safety of 1.4 for the (plastic) structural stability as required by ASME, B and PV Code, 2010, Section III, Appendix F, Section F-1341.4. The staff also confirmed that the ATRIUM 11 LTA fuel bundle met all the structural requirements of 10 CFR 71.71 and 10 CFR 71.73 without any gross deformation that can result in a criticality event.

The second calculation package (FS1-0025122 (Revision 1.0) "AREVA TN-81 ATRIUM-11 Fuel Assembly Shipping Container Drop Analyses") provides structural analyses to demonstrate that the ATRIUM 11 fuel assemblies, within a TN-B1 package, can structurally withstand the free drop requirements of 10 CFR 71.71 and 71.73 without any breach of the containment boundary caused by structural failure or plastic instability. The staff reviewed the structural analyses, performed with LS-DYNA with the same analytical methodology and assumptions as those used in the FS1-0015328 report, and questioned the modeling of the fuel mass in the LS-DYNA finite element analyses. In response to staff’s RAI, the applicant stated that the fuel pellets are not modeled separately and that the weights of the fuel pellets are added to the cladding tubes in the base finite element model. The applicant explained that the fuel cladding is coupled together with the fuel such that the mass of the fuel pellets are uniformly distributed to the cladding surface and that the lumping of the fuel mass in the cladding shell simplified the finite element analysis while making the cladding heavier, thus causing the fuel tube to buckle during and at the end of impact in the analysis.

The staff determined that the buckling load of a fuel rod under inertia loading depends on both the weight of the cladding and the total weight of the fuel regardless of whether the fuel is bonded to the cladding (e.g., spent fuel) or not bonded to the cladding (e.g., fresh fuel). Thus, the staff is confident that the approach the applicant used in the base finite element model is exactly the proper approach and results in the correct solution for the buckling of the fuel rod.

The applicant provided the staff with the LS-DYNA output files (i.e., the d3plot files) of the fuel assembly shipping container drop analysis. The output files of the fuel assembly analysis were reviewed using the LSPrePost software. The results show that the fuel rods buckle, which, in turn, produces a maximum equivalent plastic strain of 2.6% at a location approximately 9 inches above the lower plug where the lateral displacement is a maximum. The lower end of the rod where the cladding is welded to the plug offers little rotational restraint against bending and, as a result, the equivalent plastic strain in this region is less than 1%.

The LS-DYNA fuel assembly model used liner cladding mechanical properties in the analysis. The mechanical properties of the liner cladding are lower than that of a non-liner cladding. Therefore, the staff agrees that the results of the analyses, using liner cladding properties, are conservative for the ATRIUM 11 fuel assembly with respect to the higher strength non-liner clad. The uniaxial failure strain of the cladding is 14%. At the location of maximum equivalent plastic strain (2.6%) where the bending moment in the rod is highest, there exists a biaxial tensile stress field caused by the bending moment and internal rod pressure. The biaxial tensile stress state results in a triaxiality factor of 2.0, which reduces the failure strain at this location from 14% to 7% due to the effects triaxiality. This results in a safety margin of 2.7 (7%/2.6% = 2.7) against cladding failure, which the staff finds to be more than adequate.

The staff finds that the results of the structural analyses for the ATRIUM 11x11 fuel assembly, in the Model No. TN-B1 package, comply with the minimum factor of safety of 1.4 for the (plastic) structural stability, as required by ASME, B and PV Code, 2010, Section III, Appendix F, Section F-1341.4, and that the ATRIUM 11x11 fuel assembly meets all the structural requirements of 10 CFR 71.71 and 71.73.

## 2.2 Materials Evaluation

### 2.2.1 Material Properties and Specifications

The staff verified that the applicant used adequate cladding mechanical properties for the ATRIUM 11x11 design-basis fuel (i.e., Zircaloy-2), per the structural evaluation (Report FS1-0025122, Rev. 1). The applicant confirmed that Zircaloy-2 is exclusively used as the base material in the manufacturing of BWR fuel cladding. The staff further verified the adequacy of stainless steel properties for the fuel cage, container shells, and container frames. The material properties are consistent with those in ASME B&PV Code, Section II, Part D. The applicant clarified that the fuel rod can optionally have a thin liner on the inside diameter, which may have different mechanical properties than the cladding material. However, the applicant conservatively ignored any structural support provided by this liner (Section 1.2.2, Report No. FS1-0014159, Rev. 8).

The staff reviewed the Qualification Summary Report 127-9222576, Revision 1, which the applicant submitted to demonstrate the qualification for the Upset Shape Welding of the ATRIUM 11 assemblies. The staff requested that the applicant justify that the qualification program (parameters, tooling, process limits) in the aforementioned Qualification Summary Report would be adequate for fuel from alternative suppliers. The applicant confirmed that shipment of 11x11 fuel designs from other suppliers is not possible in accordance with the application and supporting documentation submitted.

The applicant did not propose any changes to the materials of construction for the TN-B1 packaging. However, during the staff's review of the thermal performance of the package, the

applicant provided information in its responses regarding the cumulative effects of the 10 CFR 71.73 HAC events, to ensure the package would not exceed allowed releases.

The applicant clarified that the impact of strain experienced by the fuel rods was examined in the structural analysis during normal and accident transport conditions (Report FS1-0015328, Rev. 2), which demonstrated that, at no point, the load experienced by the fuel rods exceeds 70% of the plastic instability load as required by ASME Code Subsection III, Appendix F, Section F-1341.4. This is defined as the acceptance criteria for fuel assembly stability during the 9-m (30-ft) drop accident scenario in the Model No. TN-B1 package.

Further, the applicant clarified that the calculated deformations for the ATRIUM 11 fuel assembly in the TN-B1 drop analyses are consistent with those measured after the HAC cumulative effects of a 9-m (30-ft) drop, puncture drop, and thermal tests that were performed on a GNF-J Certification Test Unit with a 10 x 10 assembly. The applicant clarified that these deformations will result in a small degree of cold work. However, the subsequent maximum fire event temperature will be similar to the annealing temperature for the zirconium alloy tubing; therefore, the thermal performance of the fuel rod cladding would be unaffected by the rod bending.

The staff reviewed the applicant's analysis and verified that the applicant provided a conservative estimate of the maximum degree of strain in the Certification Test Unit's bent fuel rods. The staff also reviewed processing and mechanical property data for zirconium cladding alloys (ASM, 1990) and verified that: (a) the cold work in the bent rods would not significantly reduce the ductility of the cladding, and (b) cladding temperatures during the postulated fire event are similar to those experienced by zirconium alloys to relieve cold work during typical process annealing.

Therefore, because the staff verified that the strain in the fuel cladding in a package drop would have an insignificant effect on the mechanical properties of the fuel cladding during the hypothetical fire, the applicant's analysis of this event is found to be acceptable.

The staff notes that this conclusion is consistent with that of Revision 11 of the Model RAJ-II transportation package (Certificate of Compliance No. 9309, ML17222A010), which has the same design as the TN-B1 package.

### 2.2.2 Prevention of Chemical, Galvanic, or Other Reactions

The potential for chemical, galvanic or other reactions in the package has been previously reviewed, as documented in Section 2.2.2 of the application. The addition of the ATRIUM-11 assembly, as an approved content, does not alter the conclusions.

### 2.2.3 Effects of Radiation on Materials

The effects of radiation on the Model No. TN-B1 package have been previously reviewed, as documented in Section 2.2.3 of the application. The addition of the ATRIUM-11 assembly as approved contents does not alter the conclusions.

## 2.3 Evaluation Findings

Based on a review of the structural and materials sections of the application, the staff concludes that the Model No. TN-B1 package structural design has been adequately described and

evaluated, and has reasonable assurance that the package meets the requirements of 10 CFR Part 71.

## 3.0 THERMAL EVALUATION

The objective of the thermal review was to verify that the Model No. TN-B1 package thermal design, relative to the proposed content changes, was adequately described and evaluated under NCT and HAC conditions, as required per 10 CFR Part 71. Regulations applicable to the thermal review include 10 CFR 71.31, 71.33, 71.35, 71.43, 71.51, 71.71, and 71.73.

### 3.1 General Considerations

The application included changes in Table 3.5 regarding cladding dimensions and design parameters for 9x9 fuel, 10x10 fuel, and the addition of the new ATRIUM 11x11 fuel design. However, there was no change to the packaging thermal design, as part of this amendment request and, as stated in Section 3.1.2 of the application, the decay heat of the unirradiated contents remains insignificant. Thus, the thermal analyses of the package and resulting component temperatures did not change and the temperatures remain below allowable values while package surface temperatures stay below the 122°F non-exclusive use shipment limit.

To supplement the thermal discussion in the application, the applicant provided the calculation package FS1-0024572 (Revision 3.0) “TN-B1 Container Thermal Analysis Applicability”, which discussed a maximum pressure evaluation; results were presented as Table 1, Table 2 (non-liner), and Table 3 (with liner). It is noted that, for ATRIUM 11x11 fuel, the initial fill pressure is 1.1145 MPa (absolute) for non-liner fuel and 0.851 MPa (absolute) for liner fuel. Section 2 of the calculation package stated that results of the pressure evaluation for the ATRIUM 11x11 non-liner clad fuel design were included in Table 3.5 of the application (similar to Table 2 in calculation package FS1-0024572). Table 3.5 stated that the thickness of the liner (for cladding that has a liner) is to be excluded when determining radius and thickness inputs for the allowed pressure/cladding dimension criterion’s calculation, hence the lower allowable initial fill pressure for liner fuel.

The response to the staff’s request for additional information (RAI) and the calculation package FS1-0024572 discussed the applicability of the statement in Table 3.5 that all fuels (with and without liners), including the new ATRIUM 11x11 fuel, must satisfy a fill pressure and cladding dimension criterion. This proposed criterion was that the product [P r/t] of the rod’s fill pre-pressure (P) and ratio of inside cladding radius (r) to cladding thickness (t), must be equal to or less than 10.18653 MPa. The RAI response stated that the maximum allowed cladding stress limit was originally determined by testing and has remained unchanged as rod thickness and diameter have diminished. For example, Table 3-5 indicated that the minimum cladding thickness diminished from 0.0268 inch for the 8x8 fuel to 0.0197 inch for the ATRIUM 11x11 fuel; likewise, the maximum inside radius diminished from 0.2195 inch (0.439 inch diameter) to 0.18 inch (0.36 inch diameter).

As derived in FS1-0024572, the [P r/t] criterion is a design parameter that addresses stress due to the pressure of the helium backfill at static hypothetical accident thermal conditions, but does not explicitly address the effects of damaged rods after dynamic hypothetical accident conditions (e.g., drop, puncture). Staff notes that the structural integrity of the containment boundary is dependent on the dynamic loads of hypothetical accident conditions and the resulting response of deformed rods due to static internal pressure at high temperatures related to the hypothetical accident condition fire test. The integrity of the ATRIUM 11x11 fuel

rod/cladding at the structural loadings during dynamic hypothetical accident conditions (e.g., drop test) was discussed in other areas of the application. For example, recognizing that earlier drop tests for a 10x10 fuel may not be relevant (e.g., bending, rod buckling) for the thinner cladding of the ATRIUM 11x11 rod, calculation package FS1-0015328 described the response of the ATRIUM 11x11 fuel to a 30-ft drop. In addition, Section 7 of calculation package FS1-0024572 described the effect of deformed rods (after a 30-ft drop test) by the subsequent hypothetical accident thermal condition (e.g., 30 minute fire at 800°C). The structural and material sections of this SER discuss the effect of NCT and HAC conditions on the packaging and fuel, including the new ATRIUM 11x11 fuel design.

### 3.2 Evaluation Findings

Based on a review of the thermal chapter of the application, the staff concludes that the Model No. TN-B1 package thermal design has been adequately described and evaluated and staff has reasonable assurance that the package meets the thermal requirements of 10 CFR Part 71.

## 4.0 CONTAINMENT EVALUATION

The staff reviewed the Model No. TN-B1 package to verify that the package containment design, relative to the proposed containment and content changes, has been adequately described and evaluated under NCT and HAC conditions, as required by 10 CFR Part 71. The containment related changes for this amendment, as described in Table 3.5 of the application, included changes to the cladding dimensions and design parameters for the 9x9 fuel, 10x10 fuel, and the addition of a new ATRIUM 11x11 fuel. No modifications to the previously approved packaging design were requested as part of this amendment request.

### 4.1 Description of the Containment System

Drawing FS1-0011596 (Revision 2) provided details of the welded ATRIUM 11 zirconium cladded fuel rod and end caps. Likewise, the specification for the fabrication, inspection, cleanliness, handling, and storage requirements for the welded fuel rods was provided in FS1-0019890 (Revision 1).

Section 4.1.1 and Section 8.2.2 of the application listed closure weld process qualifications, including acceptance criterion for transverse metallographic weld sample examinations and in-process inspections. The in-process inspections included visual weld inspections (with details provided in documents AID-1067, Version 2.0 and AID-10658, Version 2.0) and burst testing to a burst strength greater than 17,400 psi (with details provided in documents SOP-40823, Version 6.0 and SWI-40823 A, Version 5.0).

In addition, Section 4.1.1 of the application stated that overall rod inspections are to include dimensional inspections, fill pressure confirmation, and helium leak tests after fabrication to a 10-7 atm-cm3/sec leak tight criterion, as defined in ANSI N14.5. The fill-pressure criterion was defined as a maximum pressure times the maximum clad inside radius to thickness ratio that is less than or equal to 10.18 MPa; this criterion was discussed in Table 3.5 and calculation package FS1-0024572 (Revision 3.0).

As described in Section 1.2.3 and Section 3.1.2 of the application, the content is unirradiated fuel; a maximum of two fuel assemblies can be placed within the packaging. The quantity of radioactive materials, Type B quantity of radioactive material, and the isotopes and A2 fractions of the content are described in Tables 1-2, 1-3, and 1-4 of the application.

## 4.2 Containment under Normal Conditions of Transport

As described in Section 2.6 and Section 4.3 of the application, the welded containment boundary is not affected by any of the normal conditions of transport.

## 4.3 Containment under Hypothetical Accident Conditions

Calculation package FS1-0015328 (Revision 2.0) presented the HAC 30-ft drop analysis of the ATRIUM 11 fuel rod content; the structural response under HAC conditions is discussed in the SER structural evaluation. In addition, a release calculation was presented in Section 4.4 of the application.

The release calculation was identical to the one submitted for the issuance of the CoC Rev. 0 and was based on a measured leak rate from earlier drop tests of a package with a 10x10 fuel assembly content. As discussed in Section 2.12 of the application, the content in Certification Test Unit 1 (CTU 1) was demonstrated to have a leak rate of less than $10^{-7}$ atm-cm$^3$/sec after the 30-foot lid-down drop test. However, it was found that the content in Certification Test Unit 2 gave a leak rate of 5.5 $10^{-6}$ atm-cc/sec after a 30-foot lower end drop test.

Recognizing that the drop tested 10x10 fuel assembly had 91 fuel rods, the applicant updated the release calculation by increasing the release proportionally to the ATRIUM 112 fuel rods. According to the applicant, the resulting calculated release of 3.3 cm$^3$/week was much lower than the 1,280 cm$^3$/week that would be needed to exceed the 10 CFR 71.51 requirement of one $A_2$/week.

It is noted that Section 7 of the calculation package FS1-0024572 stated that the CTU 2 testing resulted “in no failure of the cladding as verified by leak tests”. However, as discussed above, Section 2.12.1.3 of the application stated that the “assembly from CTU 2 was found to have a He leak rate of 5.5x$10^{-6}$ atm-cm$^3$/s.” In addition, Section 7 of FS1-0024572 stated that the GNF-J CTU 2J certification unit “resulted in no failure of the simulated fuel assembly cladding.” However, the description of the GNF-J CTU 2J certification unit testing, in the application, did not include leak rate data from which a determination of the cladding condition could be made. It is also noted that the GNF-J CTU content was a simulated 8x8 assembly (per Section 2.5.1), with thicker rods than those of an ATRIUM 11x11 assembly.

As discussed above, however, the staff finds that a leak rate of 5.5x$10^{-6}$ atm-cm$^3$/sec adjusted for an ATRIUM 11x11 content, with 112 fuel rods as opposed to 91 fuel rods with a 10x10 content, would continue to meet the containment requirements of 10 CFR 71.51.

## 4.4 Leakage Rate Tests for Type B Packages

As described in Section 4.1.1 of the application, each fuel rod after fabrication, including weld closure, is examined and helium leak tested to a $10^{-7}$ atm-cm$^3$/sec leak tight criterion, as defined in ANSI N14.5.

## 4.5 Evaluation Findings

Based on the review of the statements and representations in the application, the staff concludes that the TN-B1 containment design has been adequately described and evaluated and that the package design meets the containment requirements of 10 CFR Part 71.

## 5.0 SHIELDING EVALUATION

This Chapter is not applicable. The Model No. TN-B1 package is used solely for the shipment of unirradiated fuel.

## 6.0 CRITICALITY EVALUATION

The ATRIUM-11 BWR fuel is made of $UO_2$ and gadolinium poison only, as specified in Section 1.2.3 of the application. The maximum enrichment of the fuel is 5 wt% U-235. For packages containing the proposed two ATRIUM-11 fuel assemblies, a minimal number of $Gd_2O_3$ poisoned fuel rods is required, as specified in the allowable contents section of the CoC.

The purpose of this review is to verify that the package with the proposed contents meets the criticality safety requirements of 10 CFR 71.55 and 71.59.

### 6.1 Description of Criticality Design

#### 6.1.1 Packaging Design Features

The Model No. TN-B1 is a Type B(U)F unirradiated fuel transportation packaging system with two rectangular containers, an inner one and an outer one. The inner container is comprised of a double-wall stainless steel sheet structure with alumina silicate thermal insulator filling the gap between the two walls to reduce heat flowing into the contents in the event of a fire. The outer container is comprised of a stainless steel angular framework covered with stainless steel plates.

The criticality safety design of the package relies on the geometry of the fuel basket, fuel enrichment limit, and neutron poison in the fuel pellet (for fuel assembly payload only).

#### 6.1.2 Summary Table of Criticality Evaluations

The applicant performed criticality safety analyses for the Model No. TN-B1 package containing the ATRIUM-11 fuel assemblies or loose rods in loose rod containers (in this SER, the term container means either protective case or 5” pipe). The applicant provided a summary of the criticality evaluation results in Table 6-59 of the application for a single package and arrays of packages under NCT and HAC.

The results indicate that, under HAC, the array of packages containing two ATRIUM-11 fuel assemblies, with 5% U-235 enrichment, and 13 of the fuel rods containing 2% $Gd_2O_3$, exhibits the maximum reactivity in comparison with the packages with other proposed contents. The maximum reactivity, including two times of the standard deviation, is 0.94068.

The applicant did not provide the $k_{eff}$ value for a package loaded with 50 loose rods because the criticality safety analyses performed for a package containing 60 loose rods bound those for the package containing 50 loose rods. The staff finds this to be acceptable because the latter contains much less fissile materials and, therefore, is expected to be bounded by the analyses of the 60 loose rod packages while the other conditions of the package remain the same.

The results in Table 6-59 indicate that all $k_{eff}$ values, after being adjusted with uncertainty and biases, fall below the acceptance criterion of $k_{eff} \leq 0.95$, as recommended in NUREG-1609. On

this basis, the staff finds that the applicant has provided appropriate criticality safety analyses for the package with its new contents.

6.1.3 Criticality Safety Index

The applicant's analyses considered an infinite array of packages under both NCT and HAC and indicate that the $k_{eff}$ values for these arrays of packages are below the acceptable limit. Based upon the method prescribed in 10 CFR 71.59, the applicant determined that the Criticality Safety Index (CSI) for this package design is 1.0.

The staff reviewed the applicant's criticality safety analyses of arrays of packages under NCT and HAC and finds that the applicant followed the method prescribed in 10 CFR 71.59 to properly determine the sizes of the arrays to ensure subcriticality and the applicant has properly determined the CSI of the package. On this basis, the staff finds it appropriate to assign the CSI value as 1.0 for this package.

6.2 Contents

The applicant requested the addition of the fuel assembly or loose rods of ATRIUM-11 fuel design, a BWR fuel design with burnable poison in some fuel rods. The U-235 content in the proposed ATRIUM fuel is made directly from enriched natural uranium and fuel, made from down blended low enrichment uranium (BLEU) or reprocessed uranium, is not authorized.

The U-235 enrichment of the fuel ranges from 2.9 wt% to 5.0 wt%. Various gadolinium loads per assembly, depending on the fuel U-235 enrichment, are required to ensure criticality safety of the package under NCT and HAC. The gadolinium content is in form of $Gd_2O_3$. The minimal gadolinium load per fuel rod is 2.0 wt%. The application provides fuel assembly enrichment and required poison load for the ATRIUM-11 in Tables 6-56 and 6-57.

For packages loaded with loose fuel rods, polyethylene and plastic wraps in the inner container are used to prevent the contents from sliding under NCT. There is no requirement for gadolinium load for loose fuel rods in freely loose form, strapped form, or in steel pipe. The maximum numbers of fuel rods per package is either 50 (2x25) for individual loose rods or strapped loose rods or 60 (2x30) for loose rods contained in five inches diameter steel pipes respectively. Table 6-58 of the application defines the allowable loose rod configurations for the ATRIUM-11 fuel.

The staff reviewed the description of the new fuel contents and finds that the information provided by applicant is sufficient for the staff to perform a criticality safety evaluation of this package.

6.3 General Considerations for Criticality Evaluations

6.3.1 Model Configuration

The applicant modeled the package containing two fuel assemblies, as well as 60 loose rods in two five inches pipes. For NCT, the applicant assumed that the package cavity is flooded and reflected by 30 cm of water but the pellet/cladding gap remains dry. For HAC, the applicant assumed that the fuel pellet to clad gap is flooded with fresh water.

Although the applicant's assumptions used in the analyses for NCT did not explicitly demonstrate that the package would be subcritical with water leakage into the system, the staff finds that the model for the package under HAC, which assumed flooded pellet to clad gap, provides a bounding scenario to demonstrate the package remains subcritical consistent with the requirements of 10 CFR 71.55(b) and, therefore, is acceptable.

The applicant developed computer models for the package under both NCT and HAC. For a package under NCT, the assumptions include design basis rod pitch (1.195 cm), and dry pellet/cladding gap. For a package under HAC, the assumptions include some deformation of the fuel assembly (i.e., with expanded rod pitch of 1.2548 cm, a 5% pitch expansion) and a flooded pellet/cladding gap.

Other assumptions used in the criticality safety analysis models for a package under NCT include:

- 10.2 kg of polyethylene equivalent mass per fuel assembly. The polyethylene equivalent mass includes all polyethylene/plastic components within the inner compartment, excluding the foam liner.

- Bounding fuel assembly geometry, as shown on Figure 6-52 in the application. The bounding fuel assembly geometry is determined through sensitivity analyses.

- All fuel contains zirconium water channel (i.e., the zirconium channel in the center of the fuel assembly) is modeled as water.

Because some of the ATRIUM-11 fuel assemblies may be shipped with a fuel assembly channel, the applicant also studied the effect of zirconium fuel channel on the reactivity of the package. The range of the fuel assembly channel thicknesses examined varied from 0 to 0.254 cm. This study covers all possible fuel assembly fuel channel configurations, i.e., from no channel to the maximum allowable channel thickness. The results show that the assembly channel has a negative impact on the reactivity of the package and, therefore, the analyses for assemblies without the channels bound the ones with channels. The staff agrees with this conclusion because the channels displace moderator around the fuel and cause the reactivity to decrease.

The 11x11 fuel assembly design has a variable rod pitch along the axial direction because partial length rods are used in the fuel assembly. The upper section of the assembly has a constant 1.195 cm nominal pitch and the lower section of the assembly having rods at slightly varying pitches along the axial direction. However, the variation from the nominal rod pitch in the lower region is less than 1 mm and the overall envelope of the fuel region is unchanged.

Therefore, the H/U-235 ratio for this region is the same as the fully "rodded" (i.e., no water spaces from partial length rods) section of the fuel assembly. For this reason, the rod pitch throughout the entire assembly is modeled with an equivalent nominal pitch of 1.195 cm in the NCT models. The staff finds that this approximation in the model is acceptable because the H/U ratio is more important than the small variation of the rod pitch, as far as criticality safety is concerned: the H/X ratio is the dominating factor for reactivity and is kept constant along the axial direction of the fuel assembly.

For a package under HAC, the fuel assembly is modeled with an expansion of 5 percent (5%) of the rod pitch. The staff finds that this assumption is conservative because the fuel assembly

structural analysis, using a LS-DYNA computer code, shows a rod pitch expansion of 2.6% and the drop test of the FANP 10x10 fuel assembly shows a pitch expansion of 4.1%.

The applicant also discussed the impact of the potential variation of long and short partial length fuel rods in the ATRIUM-11 fuel assembly on the package's reactivity. In the model, the partial length rods were arbitrarily increased by 9.2 cm long to account for the plenum regions and for added conservatism. The models neglect the non-active parts of the fuel assembly and, therefore, assume these parts as water.

The criticality safety analyses for the package containing fuel assemblies took credit for the presence of the gadolinium poison smeared in fuel pellets, but only 75% of the gadolinium load. The minimal number of gadolinium-poisoned rods required is identified in Table 6-57 "TN-B1 11x11 Fuel Assembly Gadolinia Loading Criteria" of the application.

In addition, the model for a package containing two BWR fuel assemblies assumes that the gadolinium-poisoned rods are loaded symmetrically along the major diagonal at a planar view and none of the gadolinium-poisoned rods may be loaded in the periphery of the assembly. These assumptions are captured as conditions of the CoC.

The packaging for each fuel assembly may contain up to 10.2 kg of polyethylene equivalent mass, as defined in Section 6.3.2.2. In the NCT models, the 10.2 kg of polyethylene was homogenized with the water inside the fuel assembly boundary. The staff finds this assumption to be acceptable because it assumes in fact that the polyethylene foam is soaked with water like a sponge that provides the maximum moderation and reflection in this part of the package.

In the HAC models, the 10.2 kg of polyethylene was assumed to melt onto the fuel rods. The polyethylene was smeared into the cladding as one material. During its review, the staff requested the applicant to provide the technical basis for this assumption and demonstrate this treatment of the polyethylene packaging material in the model is conservative for the package under HAC.

In its response to the staff's request for additional information (RAI), the applicant performed an analysis assuming that the burned polyethylene is mixed with water. The result shows that assuming that the burned polyethylene is mixed with water yields slightly higher reactivity and, therefore, is conservative. As a result, the applicant re-performed analyses for all cases of a package under HAC.

The staff reviewed the results presented in the revised application and finds that the analyses are conservative and the results are acceptable because this assumption is consistent with the physical reality that the burned polyethylene is more likely to be soaked or mixed with water than being burned into a mixture with the cladding and the results also showed that the previous assumption was non-conservative.

The applicant also evaluated packages containing loose rods in either freely loose form, strapped form, or in five inches pipe. The calculated $k_{eff}$ values demonstrate that the package containing two BWR fuel assemblies bounds the loose fuel rod package in terms of system criticality safety, in comparison with the acceptance criterion set forth in NUREG-1609.

The staff reviewed the fuel characteristics and the TN-B1 packaging system and determined that the assumptions used by the applicant in the criticality safety analysis models are consistent with the fuel and packaging characteristics (including geometry, fuel loading and

poison loading, and packaging materials) and conservative. On this basis, the staff determined that the applicant's modeling approach for criticality safety analysis is acceptable.

6.3.2 Material Properties

The packaging material properties remain the same as previously reviewed. The fuel materials are the typical BWR fuel compositions, pure $UO_2$ fuel or $Gd_2O_3$ poisoned $UO_2$ materials that have been reviewed and approved for this package. The staff did not perform further review of the material properties used in the models.

6.3.3 Computer Codes and Cross Section Libraries

The applicant used the CSAS5 sequence of the SCALE6.1 computer code and 238 group cross section library that is derived from the ENDF/B-VII cross section database for the criticality safety analysis for this package. CSAS5 is a transport theory based Monte Carlo solution method code and is developed by Oak Ridge National Laboratory for the United States Nuclear Regulatory Commission for safety evaluations. The cross section library is one of the most advanced versions of cross section libraries that support criticality safety analyses for system containing gadolinium. This is consistent with the guidance provided in NUREG-1609. Based on this information, the staff determined that the codes and cross-section sets used in the analysis are appropriate for this application and, therefore, acceptable.

6.3.4 Demonstration of Maximum Reactivity

The applicant performed criticality safety analyses for the TN-B1 package containing the ATRIUM-11 BWR fuel both in form of fuel assembly or loose rods. The applicant also performed parametric studies on packages containing fuel assemblies and loose rods to identify the maximum reactivity. These parametric studies include partial flooding and various rod pitch changes.

The package containing two BWR fuel assemblies with 5 wt% U-235 enrichment and 13 poisoned rods at 2% $Gd_2O_3$ wt% of uranium under HAC is identified as the most reactive configuration for the package with fuel assemblies. The package containing 60 (2x30) loose fuel rods under HAC is identified as the most reactive condition for the loose fuel rod package. The maximum reactivity, including two times of the standard deviation, is 0.94068, which corresponds to the ATRIUM-11 fuel assembly package under HAC.

The staff found that the methods, as discussed above, used by the applicant to search for the maximize $k_{eff}$ are appropriate and the set of parameters used in the analysis is acceptable, based on the guidance provided in NUREG-1609.

6.3.5 Analysis Approach

The applicant analyzed packages for both fuel assemblies and loose fuel rods. The applicant includes fuel and packaging conditions that maximizes the package's reactivity. These conditions include maximum fuel pellet diameter, minimum cladding outside diameter, minimal gadolinium loading in a gadolinium poised fuel rod, 75% credit for the gadolinium poison in the poisoned rods, partial flooding with various polyethylene mass, and full length for all partial length rods.

This approach is consistent with the acceptance criteria provided in NUREG-1609.

## 6.4 Single Package Evaluation

The applicant performed criticality safety analyses for a single TN-B1 package containing fuel assemblies or loose rods of the ATRIUM-11 fuel design. The analyses include criticality safety of the package as loaded, under NCT, and HAC. The assumptions discussed in in Section 6.12.3.1.1, “Fuel Assembly Model”, are used in the models for the appropriate package conditions. Section 6.3.1 of this SER provides details of the staff’s evaluations on these assumptions.

Based on the applicant’s analyses, the maximum reactivity, including two times of the standard deviation, is 0.94068, which occurs with the package containing two fuel assemblies with 5.0 wt% U-235 enrichment and 13 gadolinium-poisoned fuel rods under HAC. This is below the Upper Safety Limit of 0.94094 for this type of fuel assembly design.

The results of the analyses demonstrate that the package meets the regulatory requirements of 10 CFR 71.55(b), 71.55(d), and 71.55(e). Since the criticality safety analyses under HAC assumes that the package and the fuel clad gap are flooded and reflected with water, this analysis satisfies the requirement of 10 CFR 71.55(b), i.e., the package is subcritical if water was to leak into the package.

## 6.5 Evaluation of Arrays of Packages

### 6.5.1 Evaluation of Array of Packages under Normal Conditions of Transport

The applicant performed calculations for an array of the TN-B1 packages under NCT. The applicant used the same assumptions as those used in the evaluation of a single package under normal conditions of transport. The array of package evaluated was 1,512 packages and the $k_{eff}$ is 0.85383.

### 6.5.2 Evaluation of Array of Packages under Hypothetical Accident Conditions

The applicant performed calculations for an array of packages under HAC. The applicant used the same assumptions as those used in the evaluation of a single package under HAC. The array of package evaluated was 100 packages and the $k_{eff}$ is 0.93810.

### 6.5.3 Determination of the Criticality Safety Index (CSI)

Based on the evaluation of array of packages under NCT and HAC, the applicant calculated the CSI following the method as prescribed in 10 CFR 71.59. Since the array of packages under HAC is more limiting, the CSI is determined as 1.0.

The staff reviewed the applicant’s analyses of an array of packages under NCT and HAC as well as the calculation of the CSI. The staff finds that the applicant’s method for analyses to be consistent with the acceptance criteria of NUREG-1609 and the applicant calculated the CSI consistent with the method prescribed in 10 CFR 71.59. On this basis, the staff determined that the CSI value is acceptable.

## 6.6 Benchmark Evaluations

The applicant performed additional benchmarking analyses for the computer code and the selected cross sections using a total of 58 critical experiments from the International Handbook

of Evaluated Criticality Safety Benchmark Experiments. The staff reviewed them and finds that the selected critical experiments provide an appropriate coverage for the range of fuel enrichment, poison loading, rod pitch, and other parameters that are important to the neutronic characteristics of the ATRIUM-11 fuel.

This is consistent with the acceptance criteria provided in NUREG-1609. On this basis, the staff determined that the selected critical experiments are appropriate and acceptable.

Using the results of the code benchmarking analyses, the applicant developed the USL for criticality safety with analyses of trending against important parameters, such as fuel enrichment, rod pitch, H/X ratio, average energy of the lethargy causing fission. The results show a USL of 0.94094 for the ATRIUM-11 fuel assembly and of 0.94047 for loose fuel rods, respectively.

6.7 Confirmatory Analyses

The staff reviewed the information provided in the amendment request. The staff also performed confirmatory analysis for the package under NCT using the SCALE 6.1 computer code and ENDF/B-VII continuous energy cross section library. The result shows a good agreement with the results provided by the applicant. On this basis, the staff determined that the criticality safety analyses performed by the applicant for the TN-B1 package are appropriate and acceptable.

6.8 Evaluation Findings

The staff reviewed the description of the criticality safety features and the safety analyses of the package and concludes that it meets all of the relevant criticality safety requirements of 10 CFR Part 71.

Based on its review, the staff finds that the applicant has performed adequate criticality safety analyses for the proposed new content, i.e., ATRIUM-11 BWR fuel assembly, bundled or individual loose fuel rods, or loose fuel rods in five inches pipe, for the Model No. TN-B1 package. These analyses includes calculations of the $k_{eff}$ values for package under NCT and HAC. The applicant also performed a search for maximum reactivity with partial flooding and changes of fuel assembly geometry under HAC to demonstrate that the package meets the regulatory requirements of 10 CFR 71.55(b), 71.55(d), and 71.55(e). The applicant performed analyses for an array of packages under NCT and HAC to determine the CSI to demonstrate compliance with 10 CFR 71.59. The staff found that the applicant followed the instructions prescribed in 10 CFR 71.59 in determining the CSI of this package.

6.9 Conclusions

Based on review of the statements and representations in the application, the staff concludes that the nuclear criticality safety design has been adequately described and evaluated and that the package meets the subcriticality requirements of 10 CFR Part 71 with the following conditions that are reflected in the CoC: (i) The gadolinium-poisoned rods in each fuel assembly must be loaded symmetrically along the major diagonal axis of the fuel assembly, (ii) the gadolinium-poisoned rods shall not be loaded in the periphery locations of the assembly, (iii) the minimal load of gadolinium per rod is 2.0wt% of fuel, (iv) the minimal number of gadolinium poisoned rods per fuel assembly shall meet the requirements of Table 6-57, “TN-B1 11x11 Fuel Assembly Gadolinia Loading Criteria,” (v) the maximum total polyethylene equivalent mass of

foam used in the package not to exceed 10.2 kg, (vi) the maximum material density for the polyethylene foam shall not exceed four pounds per square feet, and (vii) the ATRIUM fuel cannot be made from down-blended low enrichment fuel or reprocessed uranium.

## 7.0 PACKAGE OPERATIONS

Specific procedures for preparation for loading, loading the fuel assemblies into the TN-B1, loading the loose rods in the protective case into the TN-B1, loading the loose rods in the 5-inch stainless steel pipe, and for preparing the package for transport were not modified for the 11x11 fuel assembly design. Only editorial changes were provided.

The same stands for package unloading and the preparation of an empty package for transport.

## 8.0 ACCEPTANCE TESTS AND MAINTENANCE PROGRAM

Specific inspection techniques used for qualifying and in-process inspection of 11x11 fuel rods were added to this Chapter while others were removed, i.e., X-ray inspection and ultrasonic testing techniques are no longer used to inspect rod closure welds.

The reference to ASTM B811 13 “Standard Specification for Wrought Zirconium Alloy Seamless Tubes for Nuclear Reactor Fuel Cladding, Annex A.1”, applicable to burst strength testing of the cladding, was removed from the section related to the integrity of the closure weld because Framatome, Inc., uses now a proprietary burst test to verify the integrity of sample rod closure welds

The applicant defined the cladding of the ATRIUM-11 assembly as the containment boundary in the structural evaluation during drop scenarios (Report FS1-0025122, Rev. 1). The staff notes that this is an unusual approach to the shipment of fresh unirradiated fuel, and materials review guidance is not available in the pertinent standard review plan (NUREG-1609). Therefore, the staff requested that the applicant supplement the application with additional detail about the design, as well as fuel rod acceptance and qualification tests for the ATRIUM-11 fuel rods and end caps.

The applicant revised Section 4.1.1 and Section 8.2.2 of the application to define the following acceptance test requirements for the ATRIUM-11 fuel rods (also defined in Product Specification FS1-0019890, Revision 1):

- The fuel rods are manufactured under a Quality Assurance program meeting the requirements of 10 CFR 71 Subpart H.

- Welds of the fuel rod end caps to the cladding are to be conducted under a qualified process and verified for integrity using approved inspection procedures performed by qualified inspection personnel.

The critical parameters for welding, current, cladding tube extension, and electrode force are established during the weld qualification process. The closure weld process qualification includes the following:

a. transverse metallographic samples of the welds – the applicant defined quantitative criteria for allowable weld discontinuities,

b. visual inspection of each completed weld to verify that the surface is free of folds, holes, cracks, porosity, and inclusions at a minimum required 1X magnification;

c. burst testing of representative welds on cladding samples shall be conducted at room temperature during initial weld parameter qualification and on in-process samples during production. The applicant defined a quantitative requirement for the burst strength, ( ≥ 17,400 psi (≥ 1,200 bar)), and stated that failure shall not occur along the solid state bond line at the original interface between the cladding and end cap. The applicant defined visual inspection as the method for determining the burst location of the burst tested sample.

d. 100% rod dimensional inspections to the design drawing (example, FS1-0011596 Revision 2.0), and

e. 100% helium leak check and initial fill pressure - each completed fuel rod (of any design) is helium leak tested after fabrication to demonstrate that it is leak tight (< 1 x $10^{-7}$ atm-cc/s).

The applicant further defined welder requirements for burst test frequency: five consecutive at the beginning, one after each repair or change of the welding machine that may impact the process, one after interruption for more than 24 hours, one for every approximately 350 rods during the contract (367 rods maximum between tests), and, one at the end.

The staff reviewed the acceptance test and maintenance requirements and considers them acceptable for shipment of the ATRIUM 11 fuel assembly.

**CONDITIONS**

The following changes have been made to the CoC:

Item No. 3.a has been modified to reflect the name change from AREVA Inc. to Framatome, Inc. which became effective January 1, 2018. The address of the CoC holder has not changed.

Item No. 3.b. has been updated to reflect the latest revision number and title of the application.

Condition No. 5(a)(1) has been slightly edited for clarity, e.g., “a vibro-isolating device between to alleviate vibration occurring during transportation” was replaced with “damping devices to minimize vibrations during transport”.

Condition No. 5.(b)(1) has been updated. Table 1 now includes the 11x11 fuel assembly. Table 2, “Maximum Authorized Concentrations,” has been updated with a value for U-238 mentioned as “Balance of Uranium” because the value of 9.23 x 10-1g/gU in the application is based upon a maximum U-235 concentration of 5%. Condition Nos. 5(b)(1)(iv) and 5(b)(1)(v) have been renumbered to 5(b)(1)(v) and 5(b)(1)(vi), respectively. Additionally, Tables 3A and 4 are now referenced as Tables 4 and 5. Table 3 includes several updates, e.g., fuel rod pitches and $UO_2$ density described in g/cm$^3$ as opposed to theoretical density percentages. Footnotes to this Table 3 have also been updated, e.g., density is based on a pellet modeled as a right cylinder. The fuel rod pitches have also been updated. Table 4 contains the same information as Table 3 but for the 11x11 fuel assembly design in a slightly different format. Table 5 has several updates: for all fuel types, the $UO_2$ and UC density are now described in g/cm$^3$, footnotes were updated, values for the 11x11 fuel rods have been included, loose rod configuration for the non-

BWR fuel types packed in the 5” stainless steel pipe or protective case are now no longer applicable, and freely loose or strapped together fuel rods used to have separate rows but are now combined in the table.

Condition No. 5(b)(2), “Maximum quantity per package,” has been updated to include the 11x11 fuel assembly content.

Condition No. 5(b)(2)(c), “Criticality Safety Index,” has been updated to refer to the correct sub-sections.

Condition No. 7 previously included statements that now are shown as corresponding footnotes to the CoC tables. The new Condition No. 7 requires that all fuel to be shipped meet the maximum P(r/t) criterion–product of the pre-pressure and of the maximum Inside Radius/Thickness- of 10.18653 MPa, similarly to what was required for the certificate of compliance of the Model TN-B1 “sister package”, i.e., the Model No. RAJ-II (ML17222A011). Shipment of 11x11 fuel designs manufactured by other suppliers than Framatome is not authorized. ATRIUM 11x11 fuel shall contain only commercial grade uranium, i.e., cannot be made from down-blended low enrichment fuel or reprocessed uranium to be consistent with what was analyzed in the application and what staff knows would meet the fabrication acceptance criteria mentioned in Chapter 8.

Condition No. 11 authorizes the use of the previous revision of the certificate for approximately one year, i.e., up until its expiration date in this case.

The expiration date of the certificate has not changed.

The references section has been updated to include the March 2018 application.

CONCLUSION

Based on the statements contained in the application, and the conditions listed above, the staff concludes that the changes indicated do not affect the ability of the package to meet the requirements of 10 CFR Part 71.

Issued with Certificate of Compliance No. 9372, Revision No. 1,

on June 21, 2018.