August 5, 2009

Mr. T. E. Angerhofer
Division of Naval Reactors
U.S. Department of Energy
Washington, DC 20585

SUBJECT: CERTIFICATE OF COMPLIANCE NO. 9788, REV. 15, FOR THE S5W, SSN 688 CLASS, AND S4G REACTOR COMPARTMENT PACKAGES

Dear Mr. Angerhofer:

As requested by your application dated January 3, 2008, as supplemented by your letter dated July 1, 2009, enclosed is Certificate of Compliance (CoC) No. 9788, Revision No. 15, for the Model Nos. S5W, SSN 688 Class, and S4G Reactor Compartment packages. Changes made to the enclosed certificate are indicated by vertical lines in the margin. The staff's Safety Evaluation Report is also enclosed.

Department of Energy, Division of Naval Reactors, has been registered as a user of the package under the provisions of 49 CFR 173.471. The approval constitutes authority to use the package for shipment of radioactive material and for the package to be shipped in accordance with the provisions of 49 CFR 173.471. Note that the Safety Evaluation Report describes our review of the request for adding the S4G Reactor Compartment Package to CoC No. 9788.

If you have any questions regarding this certificate, please contact me or Chris Staab of my staff at (301) 492-3321.

Sincerely,

/RA/

Eric J. Benner, Chief
Licensing Branch
Division of Spent Fuel Storage and Transportation
Office of Nuclear Material Safety
and Safeguards

Docket No. 71-9788
TAC Nos. L24132

Enclosures:
1. Certificate of Compliance No. 9788, Rev. No. 15
2. Safety Evaluation Report

cc w/encls: R. Boyle, Department of Transportation
J. Shuler, Department of Energy

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NAME	CStaab	CBajwa	JPiotter	GHornseth	DForsyth	
DATE	7/28/2009	7/31/2009	7/31/2009	7/31/2009	7/28/2009	
OFC	SFST	SFST	SFST	SFST	SFST	
NAME	LCampbell	CRegan	MWaters	MDeBose	EJBenner	
DATE	7/30/2009	7/31/2009	7/31/2009	8/4/09	8/5/2009	

C=Without attachment/enclosure E=With attachment/enclosure N=No copy **OFFICIAL RECORD COPY**

SAFETY EVALUATION REPORT
Docket No. 71-9788
Model Nos. S5W Reactor Compartment Package, SSN 688 Class,
and S4G Reactor Compartment Package
Certificate of Compliance No. 9788
Revision No. 15

SUMMARY

By applications dated January 3, 2008 and July 1, 2009, the Department of Energy, Division of Naval Reactors, requested an amendment to Certificate of Compliance No. 9788, to add the S4G Reactor Compartment Package to the certificate allowing for reactor compartment Model Nos. S5W, S4G, and SSN 688 Class Reactor Compartment packages.

Based on the statements and representations in the applications, the Certificate of Compliance has been amended to include the S4G Reactor Compartment package.

1.0 General Information

Changes to the General Information section of the Safety Analysis Report (SAR) include adding information about shipment of the S4G Reactor Compartment, such as:

The S4G Reactor Compartment is shipped as two separate packages: Package 1 and Package 2. The S4G Reactor Compartment packages are approximately 30 feet in diameter, and the containment bulkheads are made of HS steel. Package 1 is approximately 39 feet long and weighs a maximum of 2,280,000 pounds. Package 2 is approximately 40-1/2 feet long and weighs a maximum of 2,080,000 pounds.

All changes to the General Information section are in compliance with 10 CFR Part 71.

2.0 Structural and Materials

The applicant evaluated the S4G Reactor Compartment disposal package for compliance with the structural requirements specified in 10 CFR Part 71, under normal conditions of transport and hypothetical accident conditions. The package will not contain irradiated reactor fuel.

2.1 Structural Normal Conditions of Transport

For normal conditions of transport, the applicant evaluated the package(s) for heat and associated pressures as well as differential thermal expansion, cold and associated pressures as well as differential thermal contraction, reduced and increased external pressure, vibration, water spray, free drop, penetration, and immersion. All methods of evaluation were by means of classical engineering mechanics and were found to be acceptable. Evaluations of deep submergence, corner drop, and compression were not applicable to this package.

2.2 Structural Hypothetical Accident Conditions

The S4G Reactor Compartment package was evaluated for various hypothetical accident conditions including a 9 meter free drop with multiple orientations, hull puncture, reactor puncture, thermal effects and associated pressures, and water immersion. Due to the package characteristics a crush evaluation was not applicable.

Evaluations of the 9 meter free drop were performed for aft and forward end axial drops, side drops of varied orientations, as well as stable corner (i.e. center of gravity over corner) and oblique corner (slapdown) drops. These evaluations were done with classical energy balance methods that produce conservative results with respect to gross damage assessment. Where applicable, shell buckling was evaluated and found to be bounded by the energy balance approach. The reported damage for all of the free drop cases, was consistent with the assumptions used in the thermal, criticality, and shielding evaluations and staff agrees that the methodology used is conservative with respect to overall structural performance.

The hull and reactor vessel were evaluated with classical engineering methods for puncture due to impact on to a 6 inch diameter bar. The applicant noted that the hull evaluation showed a full and complete penetration while the reactor vessel had sufficient material thickness to preclude puncture. The hull puncture was determined to be acceptable as it is the initial condition used for evaluating criticality and shielding requirements.

The thermal evaluation indicated that some loss of lead shielding occurs as well as a complete loss of polyethylene due to fire. As with the puncture evaluation, these results are used as the initial conditions for the shielding evaluation.

With respect to water immersion of 50 feet for all packages, the design bases for both the pressure hull and defueling hull cuts bound the maximum pressure exerted by a fifty foot head of water.

2.3 Structural Conclusions

Based on the statements and representations in the application, the staff concluded that there is reasonable assurance that this package meets the structural requirements in 10 CFR Part 71 under normal conditions of transport and the hypothetical accident conditions, where applicable.

2.4 Materials Brittle Fracture of Containment Boundary and Conclusions

<u>Pressure Hull</u>

The former USS Triton was constructed with 2 reactor plants, each in its own separate reactor compartment. Each reactor compartment pressure hull section forms part of the transportation package containment boundary. The open ends of the reactor compartments are closed with welded-in containment bulkheads. The addition of these welded bulkheads forms a water and air tight containment for the reactor compartment machinery which remains installed inside.

Since the structural materials of the submarine hull, closure bulkheads, and structural welds are ferritic steels, the potential for brittle fracture behavior under extremely cold temperatures must be evaluated.

The Triton pressure hull was fabricated from steel which was the same specification as several previously constructed submarines of that era. This steel was evaluated for its brittle fracture properties, in particular, its Nil Ductility Transformation Temperature (NDTT).

Samples of the Triton hull plates were not directly evaluated by testing. Instead, data obtained from other submarines constructed with the same type of hull plate steel was evaluated. This data was contained in the SARs for submarines with S2Wa, S3W, and S4W reactor types. In all, steel hull plate samples from 5 different retired submarines were tested for NDTT. It was found that the NDTT of the different samples varied as is normal.

The highest measured NDTT, for one sample, was approximately at the lowest normal design operational temperature of minus 20 degrees Fahrenheit, as specified by 10 CFR 71.71(b). Minus 20 degrees Fahrenheit is also the lowest temperature for evaluation of hypothetical accident conditions of transport, as specified by 71.73(b). Since the highest measured NDTT from the various samples was near the specified normal and accident condition of minus 20 degrees Fahrenheit, it was conservatively assumed that brittle fracture at minus 20 degrees Fahrenheit was not necessarily precluded as required by the performance requirements of 10 CFR 71.55.

Measured NDTT for the welds joining the hull plates showed the welds have a significantly lower NDTT and thus do not enter into the evaluation of brittle fracture.

Given the upper limit of the measured hull plate NDTT, and consideration of the practical matters of transport (primarily by barge) for such a large transportation package, an administrative control regarding the lowest allowable temperature for transportation was proposed. This Lowest Service Temperature was conservatively specified as 20 degrees Fahrenheit. This is 40 degrees Fahrenheit higher than the lowest operational and accident condition temperature specified by the Part 71 regulations. It is also roughly 40 degrees Fahrenheit higher than the highest measured value of NDTT for the type of steel employed.

Adoption of this restriction on the lowest permitted ambient temperature for transportation would assure that brittle fracture of the hull would be precluded. Administratively, this condition would be assured and achieved by prohibiting transportation during winter months when the temperature could be lower than 20 degrees Fahrenheit. Suitable climatic data exists to support this administrative control. Weather forecasting would also allow avoidance of adverse conditions.

The staff finds that adoption of this restriction would satisfy the performance requirements of 10 CFR 71.55 and is acceptable.

2.5 Materials Containment Bulkheads and Hull Blanking Plates and Conclusions

The material chosen for the containment bulkheads is another type of ferritic steel with toughness and low temperature properties that are similar to or exceed that of the original hull plate material. The closure bulkheads are welded to the reactor compartment structure with high strength, high toughness filler metals that are similar to the chemical and physical properties of the base materials being joined.

The measured NDTT for the closure bulkhead steel is significantly lower than the NDTT for the hull plate steel. Similarly, the various filler metals employed for welding the various hull blanking plates and the containment bulkheads have NDTTs lower than the hull plate steel. Thus, the hull plates are the most limiting factor with respect to NDTT.

The staff finds that the proposed lowest service temperature of +20 degrees Fahrenheit will provide adequate assurance that brittle fracture of the containment boundary structures cannot occur and is acceptable.

Weld Inspections and Testing

Welds which are used to attach the hull blanking plates and the containment bulkheads are fillet welds. They are sized to withstand all the applicable design accident conditions.

The welds which attach the blanking plates and containment bulkheads are performed using qualified welding procedures and certified welders. Welds are visually examined and magnetic particle tested. The entire containment structure is pneumatically tested to 1.5 times the design maximum operating pressure and examined for leakage. Any weld defects are removed and repaired with qualified procedures and certified welders. Repair welds are visually examined and magnetic particle tested. The containment is then again pressure and pneumatically leak tested.

Welds used to seal equipment and piping internal to the containment structure are executed in a fashion similar to that described for the containment welds.

The staff finds the weld examination and testing program to be acceptable.

Chemical and Galvanic Reactions

Potential chemical and galvanic reactions within/between the reactor compartment contents was assessed for the duration of transportation. The reactor piping systems and equipment are primarily constructed of corrosion resistant alloy steels. The piping systems are sealed off from the interior of the reactor compartment with welded closures. The residual water remaining in the reactor system is not expected to corrode the piping system during the maximum anticipated duration of transportation.

In the unlikely event that water escaped from the reactor piping system into the interior of the reactor compartment, which is constructed of coated ferritic steels, no significant corrosion would occur during the anticipated duration of transportation.

The staff finds that no significant chemical or galvanic corrosion would occur.

3.0 Thermal

The applicant evaluated the S4G Reactor Compartment disposal packages for compliance with the thermal requirements specified in 10 CFR Part 71, under normal conditions of transport and hypothetical accident conditions. The packages will not contain irradiated reactor fuel. The internal decay heat load of this package is generated by irradiated reactor components and is extremely low; therefore, the thermal performance of the package is influenced primarily by external heat sources.

3.1 Normal Conditions of Transport

For normal conditions of transport, 10 CFR 71.43 requires that any exposed surface of the package not exceed 185 degrees Fahrenheit for exclusive use shipments. The applicant performed a 1-D heat transfer hand calculation (with the software package MathCad) for normal conditions and utilized a basic lumped-mass methodology for the transient response of the package for accident conditions. The staff reviewed the calculations and found them acceptable for this application.

The applicant's analysis demonstrated that the maximum package surface temperature will be less than the regulatory limit. The applicant also demonstrated that the maximum component temperatures with full solar insolation applied on the exterior of the package remain within material limits for all components. For the cold condition of -40 degrees Fahrenheit, all components are expected to function as designed. The maximum normal operating pressure and maximum thermal stresses for normal conditions of transport were found to be acceptable and are reported in Chapter 2, "Structural Evaluation," in addition to Chapter 3 of the application.

3.2 Hypothetical Accident Conditions

For hypothetical accident thermal conditions, 10 CFR 71.73 requires a transport package to be evaluated for a 30-minute exposure to a fully engulfing fire environment with a minimum average temperature 1475 degrees Fahrenheit, and an emissivity of 0.9. The applicant reported the results of their thermal analysis, which evaluated the package for these conditions. The surface absorptivity of the package was 0.8, as specified in the regulations.

The applicant utilized two lumped masses, one representing the outer hull of the package, and one representing all the internals of the package, including the reactor vessel, steam generators, etc. The staff reviewed the applicant's calculations and found them to be acceptable.

Two materials, lead and polyethylene, may experience melting during the fire event, especially in areas of the packages where these materials are in direct contact with the package wall. These materials are present in components that provided shielding during reactor plant operation and were not installed to provide radiation shielding during transportation. The applicant accounted for the absence of these shielding materials in the shielding evaluation and the dose remained within regulatory limits, as evaluated in Section 5.

All other materials of the package remained within their service limits, and will continue to function as designed, both during and after the hypothetical accident fire condition. The maximum internal pressure and maximum thermal stresses for the fire accident are reported in Chapter 2 of the application. The staff reviewed the stresses and found them acceptable, as reported in Section 2.

3.3 Conclusions

Based on the statements and representations in the application, the staff concluded that there is reasonable assurance that these packages meet the thermal requirements in 10 CFR Part 71 under normal conditions of transport and the hypothetical accident conditions.

4.0 Containment

4.1 Package Description and Containment Boundary

In contrast to prior applications, the S4G has dual reactor coolant plants. Therefore the application includes two similar packages with similar dimensions and contents. Each package is a defueled submarine reactor compartment including the reactor vessel and reactor plant systems and components. Each package is made by cutting the hull of the submarine at either end of the reactor compartment, and then welding bulkheads at each end. The containment boundary is made up of the bulkheads and submarine hull, along with welded closures over openings. The package is an all welded design. There are no penetrations, mechanical seals, or closure devices in the containment boundary. A leak test of the package is performed to ensure the containment boundaries are well sealed.

A gas generation methodology is presented in the SAR to estimate the hydrogen accumulation between October 2007 and the shipping date. The hydrogen accumulated between September 1969 and October 2007 is considered vented out due to gas sampling, purging and vessel opening. The analysis does not consider the credit of hydrogen/oxygen recombination catalyst. The result shows hydrogen concentration at shipping date is far below the regulatory limit. In addition, a test program was conducted to sample the hydrogen content in representative deactivated plants. Hydrogen/oxygen recombination catalyst and absorbent were added to the test package to satisfy long term burial requirements. Using the preparation procedures described in the application, the measurements taken demonstrated that there would be no potentially explosive mixture of gases or vapors present in the package. Therefore, the package satisfies the containment requirements of 10 CFR 71.43 (d).

The applicant stated the TRITON was docked in October 2007 and sealed in March 2008. The applicant stated after 1.5 years, from the time of sealing (September 2009), the maximum possible volume of hydrogen generated is 0.4 liters, which is equivalent to a concentration of 0.004%. The applicant also stated the total maximum possible volume of hydrogen generated is 2 liters, which is equivalent to a concentration of 0.04%. The possible amount of hydrogen generated is well below regulatory limits.

4.2 Normal Conditions of Transport

For normal conditions of transport, the applicant indicates that there is no loss of radioactive contents since the contents are activated structural materials or crud, neither of which would likely escape through a hypothetical leak. Given the physical form and low quantity of the residual radioactive material, the design inspections, and testing of the all-welded containment boundary, and the results of previous testing conducted on similar packages by the applicant, the staff has reasonable assurance that the package will satisify the requirements of 10 CFR 71.51(a)(1) under normal conditions of transport.

4.3 Hypothetical Accident Conditions

For hypothetical accident conditions, the package damage could include a breaching of the containment boundary. Therefore, the applicant performed the maximum release analysis with assumed release fractions, which demonstrated that the quantity of radioactivity that could be released is below the limit of 10 times an A_2 quantity per week for Kr-85 and one A_2 quantity per week for all other isotopes, as specified in 10 CFR 71.51(a)(2).

4.3.1 Source Term

The reactor is defueled, therefore the radioactivity is limited to the following sources: (1) the activated structural components; (2) crud that was generated by reactor operations, which is present in the reactor vessel, steam generator and in the reactor system piping and components; and (3) residual water that remains within the coolant system and in the reactor vessel. Due to decommissioning activities in 1969, the radioactive materials have gone through long periods of decay and present lower activity levels when compared with prior applications. Essentially no fission products or actinides are present in the package. Activated structural components are not considered releasable. Therefore, releasable radioactivity consists of crud and any residual liquids. The activity contained in the residual liquids was calculated by the applicant and demonstrated to be negligible. Table 1.2-1 in the SAR lists the crud content of reactor compartment at 38 years after End of Plant Life. An average Co^{60} crud concentration determined by radiation level measurement is assumed, as the Co^{60} distribution in reactor plant. Activity estimates for isotopes other than Co^{60} were determined using an average crud distribution. In Table 1.2.1 of the SAR, the activity contained in the ion exchanger is set to zero since the resin was drained during the package preparation. This preparation procedure differs from prior applications, but is acceptable to the staff.

4.3.2 Releasable Activity

The applicant calculated the quantity of radioactivity that may be released into the package using previously-approved methods. Any crud that may be released into the package is assumed to be available for release to the environment. The applicant assumed the following release percentages: 33 percent of the total amounts of crud on the wetted surfaces of all primary systems was conservatively assumed to be loose. Of the loose crud, 20 percent from the reactor vessel and coolant piping and 100 percent in the steam generator is conservatively assumed to be released to the reactor compartment during the accident condition. Then half of the loose crud in reactor compartment is assumed to be released into environment as a result of the hypothetical accident. The staff reviewed these quantities and compared them to previously approved packages. The staff agrees these assumptions are reasonable and more conservative than those approved in the SSN 688 application.

4.4 Conclusions

The staff accepts the assumptions and measurements used to determine the amount of crud and the staff agrees with the assumptions regarding releasable activity. The calculation along with the assumptions proves the quantity of releasable activity in the package remains below the regulatory limits, with reasonable safety margin, for normal conditions of transport and hypothetical accident conditions. Based on the statements and representations in the application, the staff concludes that the package meets the containment requirements of 10 CFR Part 71.

5.0 Shielding

The applicant requested a review of the SAR for the S4G Reactor Compartment package and an amendment to Certificate of Compliance 9788 to include, as authorized contents the S4G Reactor Compartment package.

5.1 Source Term

The applicant used a two-dimensional neutron transport code to calculate the activation source terms based on material composition, actual operating history, and shutdown time. The flux profile used in this calculation was determined from a bounding, conservative model. Sources of activation radiation include the pressure vessel, activated internal components of the pressure vessel, and primary shield tank.

The method for estimating the radioactive source term from the crud deposits included data from measurements taken during the operational history of submarines in service. Radioactive crud exists on the interior surface of the reactor coolant piping, steam generator tubing, and other system components. Even when totaled, crud sources have negligible contribution to the dose rate compared to the magnitude of the activation contribution. The ion exchange resin was removed and not included in this analysis.

5.2 Normal Conditions of Transport

The applicant's analysis showed that the radiation limits for the normal conditions of transport in 10 CFR Part 71 are met following a decay time of 38 years after reactor shutdown. The expected radiation levels are based on actual, measured operating history and prior analyses with a similar reactor compartment with the shortest possible shutdown time. The analysis considered radioactive source terms from the activated hardware under normal conditions of transport. Radiation level calculations were made using the SPAN4 computer code and infinite media iron dose build-up factors.

5.3 Hypothetical Accident Conditions

As in previous analyses, the shielding evaluation for the hypothetical accident conditions considered the results of a 30-foot free drop, puncture, and fire tests. The shielding analysis includes models which consider the movement of the reactor pressure vessel and package boundary to locations which result in higher dose rates. All models result in a conservative estimate of external dose rate.

For this amendment, the applicant considered radioactive material release as a result of hypothetical accident conditions. No ion exchange resin remains in the system to be released. The amount of activated crud in the entire system is below the limits allowed by 10 CFR 71.51.

5.4 Conclusions

The dose rates calculated by the applicant were within the limits specified in 10 CFR 71.47 and 10 CFR 71.51. Based on its review of the methods, analyses, and information presented in the application, staff agrees with the applicant's conclusion that the shielding requirements of 10 CFR Part 71 will be met with the proposed contents and packaging design.

6.0 Criticality

A reactor compartment disposal package is nonfissile, and therefore, does not require a criticality evaluation to comply with the requirements of 10 CFR Part 71.

7.0 Package Operations

Operating procedures remain adequate and conform to the requirements of 10 CFR Part 71.

8.0 Acceptance Tests and Maintenance Program

Acceptance and maintenance testing remain adequate and conform to the requirements of 10 CFR Part 71.

Conclusions

The Certificate of Compliance has been revised as requested by the applicant. The change is to include the S4G Reactor Compartment package as an allowed reactor compartment for disposal per Certificate of Compliance No. 9788.

Condition 5 was revised to include the S4G Reactor Compartment as an allowed reactor compartment to be shipped.

Condition 6 was revised to explicitly state the amount of residual liquids allowed in the S4G Reactor Compartment during shipment.

Condition 8 was revised to explicitly state the Lowest Service Temperature for the S4G Reactor Compartment.

These changes do not affect the ability of the package to meet the requirements of 10 CFR Part 71.

Issued with Certificate of Compliance No. 9788,
Revision No. 15, on August 5, 2009