# SAFETY EVALUATION REPORT
**Docket No. 71-9788**
**Model No. S5W and SSN 688 Class Reactor Compartment Packages**
**Certificate of Compliance No. 9788**
**Revision No. 13**

## SUMMARY

By application dated March 15, 2002, the Department of Energy, Division of Naval Reactors, requested amendment and renewal of Certificate of Compliance No. 9788, for the Model No. S5W and SSN 688 Class Reactor Compartment packages. Naval Reactors requested that the SSN 688 Class Reactor Compartment package be included as authorized contents to replace the S6G Reactor Compartment package.

Based on the statements and representations in the application, the staff agrees that this change does not affect the ability of the package to meet the requirements of 10 CFR Part 71.

## EVALUATION

### 1.0 GENERAL INFORMATION

### 1.1 Packaging

The package consists of a defueled reactor compartment (including the reactor vessel and internals and other reactor plant components) that has been separated from the remainder of the submarine hull and prepared for shipment by installing bulkheads and attaching handling fixtures. The applicant requested amendment to the certificate to include SSN 688 Class reactor compartments as authorized contents. The SSN 688 Class Reactor Compartment package is approximately 46 feet long and approximately cylindrical, with a maximum diameter of approximately 33 feet. The maximum weight of the package is 3,360,000 pounds.

### 1.2 Contents

The contents are composed of activated structural components associated with the SSN 688 Class reactor vessel complex, plant piping, ion exchanger resin and purification filter media, which may be solidified in place, and small quantities of residual liquids.

### 1.3 Drawings

The package is constructed in accordance with the drawings, figures, and sketches included in the application.

## 2.0 STRUCTURAL

### 2.1 Package Description

The structural evaluation of the amendment request addressed the flat plate containment bulkhead design for the SSN 688 Class Reactor Compartment package. The package consists of the portion of the submarine hull that encloses the reactor compartment plus a containment bulkhead on each end. The reactor compartment consists of the components and structural members between specific submarine bulkheads. The containment bulkhead is a prefabricated flat plate that is welded to the ends of the submarine's pressure hull after the hull cuts are made. The containment bulkheads serve as the forward and aft containment boundaries of the package. They are constructed of 2-inch thick (with 3-inch thick inserts) HS steel plate and are designed to assure that loss or dispersal of the radioactive contents of the package and any increase in external radiation from internal components do not exceed the requirements of 10 CFR 71.51.

### 2.2 General Standards for Packages

The package complies with the general standards for all packages specified in 10 CFR 71.43. There are lifting fixtures and slings for the package; however, they were designed for underwater salvage operation in case of an accident at sea. Due to the size and weight of the package, it could not be lifted inadvertently during normal handling operations. Two support fixtures are attached to the outer surface of the package. These provide a base for attaching each package to the barge. They are shown by analyses to meet the requirements of 10 CFR 71.45(b). Under excessive loads, the support fixture to barge welds will fail first. Thus, the package, with its support fixture, will separate from the barge without reducing the effectiveness of the package.

### 2.3 Normal Conditions of Transport

The containment boundary of the package consists of the submarine's pressure hull and the containment bulkheads. The containment bulkheads are welded flush to the submarine's pressure hull and provide watertight containment for the package. Evaluation of the package with respect to the performance under normal conditions of transport relied primarily upon standard textbook equations. The results demonstrated that the package will remain intact and perform its intended safety function under the tests and conditions in 10 CFR 71.71.

### 2.4 Hypothetical Accident Conditions

The package was evaluated for 30-foot drop tests in the end, top side, bottom side, lateral side, and corner orientations. Because the package is effectively a cylinder laid on its side, the above three "side" drops can be defined as follows: (1) top side drop means that the package is rotated about its long axis by 180° from its normal shipping orientation; (2) bottom side drop means that the package is dropped in its normal shipping orientation; and (3) lateral side drop means that the package is rotated about its long axis by 90° from its normal shipping orientation. Standard textbook equations were used to evaluate these drop orientations. The results indicated that none of these drops produces sufficient stresses on the reactor compartment components to cause their mountings to fail. None of the 30-foot drops causes sufficient damage to the package to cause either the pressure vessel or closure head attachment studs to rupture. However, the outer surface of the package will be crushed during the 30-foot drop accident tests. Consequently, part of the outer seal welds in the top, side, and bottom of the package may fail. Due to the amount of deformation resulting from these drops, some polyethylene shielding contained within some of the deformed structural members is expected to be extruded to either the outside of the package or to other recesses of the package or both. Shielding calculations in the application did not take credit for the shielding effects of the polyethylene that was installed for shielding during reactor operation.

Puncture analyses indicated that the bar can penetrate the submarine hull. The applicant demonstrated that less than an $A_2$ quantity of radioactive material is available for release, as described below (see Section 4.0, "Containment"). The containment evaluation in the application conservatively assumed that both the ion exchanger, or its connecting piping, and the purification filter connecting piping were punctured. The bar cannot puncture the purification filter body. The shielding evaluation in the application considered the possible release and relocation of the resin and media within the package due to the effects of the puncture test.

The thermal evaluation of the damaged package indicated that lead melt and polyethylene degradation may occur where they are located close to the inside surface of the hull. This was acceptable since the shielding analysis in the application did not take credit for the polyethylene or the lead shielding to meet the post-accident dose rate limit. The internal pressure increase resulting from the accident condition fire test may cause the package to rupture at one or more seal welds. The possible failure of these welds was evaluated in the containment and shielding sections of the application. Failure of these welds will not affect the ability of the package to meet the acceptance standards in 10 CFR 71.51. The stresses due to differential thermal expansion and contraction were negligible.

The containment boundary of the package consists of the submarine hull and the containment bulkheads. Both the submarine’s pressure hull and the containment bulkheads are capable of withstanding an external pressure significantly greater than the 50-foot immersion test in 10 CFR 71.73.

## 2.5 Materials

The applicant submitted a revised design for the closure bulkheads that form the containment and structural boundary on each end of the reactor compartment package. The bulkhead material is a high strength steel with toughness and low-temperature properties that are similar to that of the original hull plates that comprise the remainder of the package containment system. The closure bulkheads are welded to the reactor compartment structure with high-strength, high-toughness filler metal that is similar to the chemical and physical properties of the base materials being joined.

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 was considered. The hull plate and closure bulkhead steels and associated welds have been specifically selected to have a ductile-to-brittle transition temperature that is significantly below the design minimum service temperature of -40°F.  This assures that the materials will not be susceptible to brittle behavior under static or dynamic loads at the minimum design temperature of -40°F.

In addition to the ability of the material to withstand extreme cold without being susceptible to brittle fracture, the operating procedures for the transportation of the reactor compartment conservatively preclude shipping if the weather forecast predicts minimum temperatures below the lowest service temperature calculated for the package.  Adoption of this minimum service temperature limit further assures the ductile behavior of the structural and containment system materials of the reactor compartment during transportation.

Potential chemical and galvanic reactions of the reactor compartment contents were assessed for the transport period.  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 will not corrode the piping system during the design maximum duration of transportation (1 year) and well beyond.  In the unlikely event that water escaped into the interior of the reactor compartment, which is constructed of coated ferritic steels, no significant corrosion would occur over the design maximum shipping duration of 1 year.

Radiolytic decomposition of residual water was evaluated to determine if a flammable concentration of hydrogen could be produced during shipping.  Conservative calculations show that a flammable hydrogen concentration would not be generated during a 1 year shipping period.

### 2.6 Conclusions

The applicant has shown that the package containment boundary will not be breached under normal conditions of transport.  The accident conditions tests could result in the package being crushed due to the drop and breached due to the bar puncture, and in the possible loss of polyethylene and lead shielding and containment due to the thermal test.  However, the total amount of radioactive material available for release to the environment due to this damage is less than an $A_2$ quantity, and the external dose rates would remain within the regulatory limits.  Therefore, the staff agrees that the package meets the structural performance requirements of 10 CFR Part 71.

## 3.0 THERMAL

The applicant evaluated the SSN 668 Class Reactor Compartment package for compliance with the thermal requirements specified in 10 CFR Part 71, under normal conditions of transport and hypothetical accident conditions.  The package will not contain irradiated reactor fuel.  The internal decay heat load of the package is generated by irradiated reactor components and is 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°F for exclusive use shipments. The applicant performed a 1-D heat transfer calculation that 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°F, all components are expected to function as designed. The maximum internal pressure and maximum thermal stresses for normal conditions of transport were found to be acceptable and are reported in Chapter 2 of the application.

### 3.2 Hypothetical Accident Conditions

For hypothetical accident conditions, 10 CFR 71.73 requires a transport package to be evaluated for a 30-minute exposure to a fire environment with an average temperature of at least 1475°F and an emissivity of 0.9. The applicant used the REFLEX/THERMAL computer code to perform the package analysis for the fire test condition. The surface absorptivity of the package was 0.8, as specified in the regulations.

The applicant’s analysis did not assess the package with a post-fire solar loading and did not utilize a convective coefficient corresponding to the turbulent nature of a pool fire environment, as specified in the regulations. Given the large size of the package and the relatively small contribution of convective heat transfer to the total heat transfer into the package during the fire event, the differences in temperature presented by accounting for post-fire insolation and a higher convection coefficient would not be significant.

Two materials, lead and polyethylene, may experience melting during the fire event, especially in areas of the package where these materials are in contact with the package wall. These materials are present in components that provided shielding during reactor plant operation and were not installed for purposes of transport. The applicant accounted for the absence of these shielding materials in the shielding evaluation, and the dose remained within regulatory limits, as described below (see Section 5.0, "Shielding").

All other materials of the package remained within their service limits, and would 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 were acceptable and are reported in Chapter 2 of the application.

### 3.3 Conclusions

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

## 4.0 CONTAINMENT

### 4.1 Package Description and Containment Boundary

The package is the defueled submarine reactor compartment, including the reactor vessel and internals. The 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 non-destructive examination of seal welds is performed prior to shipment. Performance of the welds is based on acceptable service during submarine operations, and, for new welds, pressure testing of the package that is performed prior to shipment.

### 4.2 Normal Conditions of Transport

For normal conditions of transport, the applicant has shown that there are no credible events that can penetrate the containment boundary. Therefore the package satisfies the containment requirements of 10 CFR 71.51 under normal conditions of transport.

### 4.3 Hypothetical Accident Conditions

For hypothetical accident conditions, the package damage could include breaching the containment boundary. Therefore, the applicant showed 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 1 $A_2$ 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 ion exchanger, purification filter, and in the reactor system piping and components; and (3) residual water that remains within the coolant system and in the reactor vessel. Essentially no fission products or actinides are present in the package. Activated structural components are not considered releasable. Releasable radioactivity consists of crud and any residual liquids. For the containment analysis, the applicant conservatively assumed that all radioactivity present in crud was Co-60.

The quantity of Kr-85 and H-3 isotopes in all sources (crud, residual water, and activated structural components) is negligible. Residual water that is left in the reactor vessel or internals will be absorbed by an inert absorbent which is added to the package.

The applicant provided four methods that may be used to ensure that the $A_2$ limit for releasable radioactivity would be met: (1) measure the actual amount of Co-60; (2) discharge the ion exchanger and purification filter, which concentrate the crud; (3) protect the ion exchanger and purification filter so that they are not susceptible to damage under puncture test conditions; (4) solidify the ion exchanger resin and purification filter media using epoxy.

### 4.3.2 Releasable Activity

The applicant calculated the quantity of radioactivity that may be released into the inside of the package using previously-approved methods. Any crud that may be released into the inside of the package is assumed to be releasable into the environment. The applicant assumed the following release percentages: 1 percent of the crud from all components, 100 percent of the crud from the ion exchanger and purification filter, and 100 percent of the radioactivity in the residual liquids. If the ion exchanger resins and purification filter media were solidified using epoxy, the applicant assumed release percentages of 10.8 percent and 2 percent, respectively. Radiation measurements were used to determine the contamination levels present on the reactor system components.

### 4.4 Conclusions

The staff agrees with the assumptions regarding releasable activity. The staff agrees that the assumptions and measurements used to determine the amount of crud are acceptable and will provide reasonable assurance that the quantity of releasable activity in the package will remain below an $A_2$ quantity. 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 an amendment to the certificate for shipment of the SSN 688 Class Reactor Compartment package. The major changes related to the shielding evaluation were establishing new curie limits for the ion exchanger and the purification filter when: (1) the media in either of these components are solidified to limit the quantity of radionuclides that can be released under the hypothetical accident conditions, and (2) supplemental shielding is added to the hull to provide additional protection against potential relocation of radiation sources under hypothetical accident conditions.

### 5.1 Source Term

The applicant used the ORIGEN code to calculate the activation source terms based on material composition, operating history, and shutdown time. A bounding operating history was used. Sources of activation radiation included the pressure vessel, core basket, thermal shields, support assemblies, and reflectors.

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 in the ion exchanger resin and purification filter media, and on the interior surface of the reactor coolant piping, steam generator tubing, and other system components.

### 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 365 days after reactor shutdown. The calculated radiation levels are based, in part, on limiting the activity in the ion exchanger resins and purification filter media. The analysis considered radioactive source terms from activated hardware and crud deposits. Radiation level calculations were made using the SPAN4 computer code.

## 5.3 Hypothetical Accident Conditions

As in previous analyses, the shielding evaluation for the hypothetical accident conditions considered the results of the 30-foot free drop, puncture, and fire tests. The model considered the movement of the various reactor plant components to locations which result in higher dose rates. The shielding model considered the maximum hull deflection caused by the 30-foot drop.

For this amendment, the applicant considered the results of several cases where solidified and unsolidified media were released from the ion exchanger and the purification filter under hypothetical accident conditions. The analysis modeled the unsolidified ion exchanger resin and purification filter media as a pile against the hull instead of being spread in an arc around the hull. The volume of the pile remained unchanged in this more realistic configuration. The pile was spread over two frame bays for the release of either the ion exchanger resin or purification filter media; and considered to be spread over three frame bays for the combined release. The analysis also included self-attenuation in the ion exchanger resin and purification filter media piles. Solidified waste was modeled in a single bay for the cases where it was determined that this waste could be released under hypothetical accident conditions.

Using the modeling approach described above for unsolidified ion exchanger resin and purification filter media, the applicant calculated the dose rates after the hypothetical accident conditions when supplemental shielding is applied on the hull exterior around the vicinity of the ion exchanger and purification filter. The supplemental shielding provides additional attenuation of radiation under hypothetical accident conditions, and thus, results in a higher allowable limit on the curie content of the ion exchanger resins and purification filter media. Specifications for the attachment of the supplemental shielding are given in the application to assure that it will remain in place under hypothetical accident conditions.

## 5.3 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 proposed limits on curie content provide reasonable assurance that the design meets the shielding requirements of 10 CFR Part 71.

## 6.0 CRITICALITY

The reactor system is defueled and there are no fissile materials in the package.

## 7.0 OPERATING PROCEDURES

The operating procedures for the SSN 688 Class package are the same as those for previously-approved reactor compartments.

## 8.0 ACCEPTANCE TESTS AND MAINTENANCE PROGRAM

The acceptance tests and maintenance for the SSN 688 Class package are the same as those for previously-approved reactor compartments.

## CONDITIONS

The following conditions have been added to the Certificate of Compliance for the shipment of the SSN 688 Class Reactor Compartment package:

- For SSN 688 Class packages, the Co-60 curie content of the ion exchanger resin and purification filter media shall be limited as described in Table 4-2 of the supplement dated March 15, 2002. For packages that have supplemental shielding as described in Appendix 2.10.14 of the supplement dated March 15, 2002, alternative radioactivity limits for the ion exchanger resin and purification filter media may apply, as specified in Table 2.10.14-1 of Appendix 2.10.14.

- Shipment of the SSN 688 Class packages shall not occur before 365 days after final reactor shutdown.

## CONCLUSIONS

The Certificate of Compliance has been amended to authorize shipment of the SSN 688 Class Reactor Compartment package, subject to the conditions listed above. Reference to S6G reactor compartments has been deleted. The Certificate of Compliance has been renewed for a five-year period that expires on September 30, 2008.

Based on the statements and representations in the application, the conditions listed above, and for the reasons stated in this Safety Evaluation Report, the staff agrees that 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, Rev. No. 13,
on February 28, 2003 .