SAFETY EVALUATION REPORT

Docket No. 72-1007
Ventilated Storage Cask (VSC-24)
Certificate of Compliance No. 1007
Amendment No. 4

SUMMARY

By application dated March 30, 2001, and as supplemented on July 26, 2001, and April 29, May 16, and August 8, 2002, BNFL Fuel Solutions Corporation (BFS or the applicant) requested an amendment to Certificate of Compliance No. 1007 (certificate) for the Pacific Sierra Nuclear Associates (PSNA) VSC-24 system in License Amendment Request (LAR) 01-01. The certificate is held by PSNA, which is a partnership between Sierra Nuclear Corporation and BFS.

The applicant requested changes to the certificate, including its attachment, and Revision 1 of the Final Safety Analysis Report (FSAR) for the VSC-24 system to address the storage of specific fuel control elements as integral components to fuel assemblies. In addition, the structural calculations were updated to provide bounding analyses for the various design configurations to accommodate the storage of selected fuel control elements.

Based on the statements and representations in the application, as supplemented, the staff agrees that the VSC-24 system will continue to meet the requirements of 10 CFR Part 72.

1.0 GENERAL

The applicant requested changes to the FSAR and Technical Specifications (TS) to allow the storage of specific fuel control elements as integral components to fuel assemblies. Staff revised TS 1.2.1 to limit the allowable fuel burnup to specifically 45 GWd/MTU, which clarified the previous ambiguous terminology. No other previously reviewed and accepted fuel or cask parameters were changed as a result of this amendment. TS 1.1.1, 1.2.1, 1.2.4, and 1.2.6 were amended to address the additional fuel control elements approved for storage, and TS 1.2.10, “Time Limit for Draining the MSB [multi-assembly sealed basket],” was deleted to eliminate redundant requirements for controlling moderator density. The staff’s evaluation is discussed below along with the proposed TS changes.

The staff has reviewed these changes and finds them acceptable. These changes are consistent with or supported by the analyses that have been previously reviewed and approved by the staff. These changes have no adverse impact on the design and operation of the cask and will not affect the ability of the cask to meet the requirements of 10 CFR Part 72.

2.0 STRUCTURAL EVALUATION

2.1 Structural Design Changes

As noted in the LAR 01-01 request, the proposed amendment is to expand the scope of the control elements to be stored in the VSC-24 system. The structural calculations were updated to provide bounding analyses for the various design configurations to accommodate the storage of selected control elements.

The staff's review and evaluation of LAR 01-01 was limited to the revised portions of the structural analyses relating to the storage of additional control elements. This evaluation addresses those items that the staff judged to be of significance whether the change increased or decreased safety, and the associated margins. It should be noted that Tables 2.0-1 through 2.0-4 have been proposed for addition to the FSAR through this amendment. These tables reflect the range of parameters used for the design of the VSC-24 system and Table 2.0-4 specifically identifies the limiting condition/configuration for each component and loading.

The following changes were addressed in the application:

a. For the physical characteristics of the fuel assemblies,
   - Fuel assembly weights can vary from 1110 to 1585 pounds, which is more than the 1576 pound maximum previously identified in the FSAR;
   - The overall length of the assemblies has been increased from 172.0 inches to 178.6 inches with an active fuel region of 141.8 to 150.0 inches, which varies from the previously fixed length of 144 inches;
   - The maximum weight for uranium per assembly is 1037 pounds; which is an increase of 5 pounds over the previous value;
   - There are now three identified MSB types based on the overall height of the cylinder: short - 164.2 inches, standard - 180.3 inches, and long - 192.25 inches.
   - There is a corresponding short, standard, and long fuel storage sleeve/basket assembly for each size MSB type.
   - There are corresponding Ventilated Concrete Cask (VCC) types to accommodate the various lengths of MSBs.

b. The concrete density has been revised from 145 pounds per cubic foot (pcf) to 144 pcf for minimum weight in structural calculations with 140 pcf used in the shielding calculations for conservatism. A value of 150 pcf was used conservatively when considering the reinforced concrete in calculating maximum weights.

c. The original FSAR document contained no significant text information on the ceramic tiles used between the bottom of the MSB and the bottom liner of the VCC. There are now two physical arrangements for the 1.7 x 1.7-inch tiles. These tiles are 0.3-inches thick and serve to create a thermal air gap between the two steel surfaces. One tile arrangement is a new 24 tile configuration with tiles located on the periphery and all tiles on the same radius. The other tile arrangement is the existing 29 tile configuration with tiles distributed evenly over the contact area. Both are now shown in Drawing VCC-24-002, Sheet 2 of 3, Revision 01-01-0. FSAR Section 3.1.1 was also changed to reflect the revised text.

d. The internal diameter of the MSB transfer cask (MTC) was changed to be either 63.0 or 63.5 inches as opposed to the current dimension of 63.5 inches, and the outside diameter was revised to be either 82.0 or 83.5 inches as opposed to the current diameter of 83.5 inches only.

e. The MTC cavity height was revised to be as short as 164.7 inches, whereas the previous minimum dimension was 175.0 inches.

f. The upper limit of the weight range for the empty MTC without the lid was increased from 118,630 pounds to 118,939 pounds;

g. The maximum weight for a fully loaded VCC was revised to be 287,920 pounds, which is an increase from the previous weight of 278,630 pounds;

h. As a result of the physical dimension and weight range changes, there was an additional MTC design presented that included a second design for the shield doors and rails. These two designs are discussed as the standard door assembly and rails, and the light door assembly and rails. The weight changes also cause the trunnions to be the limiting design feature for the two MTC lift requirements.

i. There are two different lifting device sets for the two MPC designs. In particular, there are six lift rings to be used for the lighter MPC with each ring having a capacity of 24,000 pounds, and another set of six lift rings with each ring having a capacity of 30,000 pounds.

j. The tornado design basis wind speeds were transformed to be effective pressure based on ASCE 7-93 instead of the previous ANSI A58.1-82.

k. The design ground snow load was increased from 67.2 psf to 403 psf based on ASCE 7-93. This loading represents the worst-case ground snow load for Alaska.

l. New design conditions were defined for flooding conditions based mainly on the new weights and centers of gravity for the VSC-24 system. The allowable flood water velocity flowing against the loaded VCC has been reduced to 17.7 fps from 25.0 fps and the allowable static head has been increased from 17.75 feet to 120 feet.

m. The case of off-normal pressurization of the MSB was addressed as well as accident condition pressurization with revised calculations.

n. The operational handling accident load condition has been expanded to require that the velocity of downward movement when loading the filled MSB into the VCC storage cask be limited to 0.75 fps so as to control the impact on the bottom of the MSB. This is an additional control for handling operations.

o. The MSB design criteria for accident conditions were revised to be based on ASME Service Level D where plastic behavior is permitted. The maximum stress was limited to 0.9 times the ultimate stress for the material in addition to the existing stress requirements.

p. The MSB internal basket assembly was analyzed by finite element analyses and designed on the basis that under accident conditions and plastic behavior, the fuel assemblies will be retrievable.

q. The MSB helium leak pressure test will be performed at a test pressure of 0.5 atmosphere, which exceeds the ASME Section III, Subsection NC-6000, test requirement of 1.25 times the design pressure. The helium leak rate test will be performed at a pressure greater than required so the minimum test pressure will be met with that test.

2.2 Weights and Center of Gravity

Based on the revised physical dimensions and weights proposed, FSAR Table 3.2-1, "VSC System Weights and Centers of Gravity," was revised to reflect the new conditions with values reported in the form of ranges for the specific configurations. These values were used in supporting calculations to define a "worst case" condition. For example, a calculation to determine if a cask will overturn used the value for the minimum weight combined with the value for the maximum height of the center of gravity.

2.3 Structural Materials

There were no changes to the materials used in the construction of the VSC-24 system and no changes in any of the structural material properties for those materials, except as noted in Section 2.1.b. above for the concrete density.

2.4 Structural Analysis of the VCC and MSB Assemblies

Unless noted, any re-analysis and calculations completed for this amendment and discussed below follow the approach submitted and approved under previous certificate amendments or represent an improved approach.

FSAR Table 3.4-6 identifies the critical section and stress component for the various normal load condition loading combinations for the VCC. This table was updated to reflect the re-analyses performed as a result of the revised dimensions and weights, as well as the configuration of the bounding case of the tiles supporting the loaded MSB around the periphery, and other revised loads like the snow load. The maximum stress or maximum load in each case remains within the allowable stress or capacity. The largest change was reflected in the shear in the concrete at the base-wall intersection that originally showed that only 48% of the allowable capacity was used, but the revision shows 88% of the allowable capacity is used. This is acceptable since the allowable capacity has not been exceeded.

FSAR Table 3.4-7 was revised to reflect a small temperature change resulting in revised stresses in the VCC liner plate and the hooked vertical reinforcing steel in the cylindrical portion of the VCC. These changes had no significant impact on safety margins. The analysis for the VSC bottom lift (with a loaded VCC) was revised based on the new total weight.

For the VCC subject to off-normal conditions arising from severe environmental conditions relative to temperatures, the thermal analysis resulted in minor temperature changes as shown in the table in Section 11.1.1.3 of the application. The resulting thermal gradients across the VCC wall sections were still less than the normal conditions that control the thermal effects design. A second off-normal handling condition was added to the off-normal events to control impact to the MSB, but this has minimal effect on the VCC stresses.

The applicant analyzed the accident condition of the VCC subject to the revised wind and tornado criteria including the new physical dimensions and weight parameter and demonstrated that there was insufficient force to change the results from the original analyses that show there is no sliding or tipover of the loaded VCC. Likewise, the tornado missile strike under the new conditions did not change the original results and conclusions. The applicant re-analyzed the accident condition for a flood for the loaded VCC and demonstrated that the cask remains stable. There was no significant change in the local stresses in the VCC. The seismic response of the loaded VCC under accident conditions was also re-analyzed for a 0.25g design basis earthquake. The applicant's results showed that the VSC would not tipover and demonstrated an adequate safety factor of 1.28 against overturning.

For the MSB under normal conditions, the two hoist ring designs were evaluated in the same manner as the previously approved FSAR analysis with the revised weights and the minimum ring strengths using a dynamic load factor of 10%. The critical elements for the lifting operation are the rings, the structural lid into which the six rings are threaded, and the lid-to-shell weld. The safety factors remained above the required value of 6.0 for the yield stress and above 10.0 for ultimate stress. The applicant re-analyzed the MSB for the revised parameters under normal operating loads. The thermal stresses in the MSB shell were recalculated since there were some revised temperatures. The applicant also reevaluated the thermal clearances due to thermal-induced dimensional changes between the MSB shell and the fuel support sleeve/basket. FSAR Table 3.4-4 was revised to include some minor increases in stresses, but the amount is only several hundred psi. The stresses still remain below the allowable stress limit. The MSB shell and fuel storage sleeve/basket were re-analyzed for the revised conditions, however for the investigation into the bottom support conditions from the bounding support tile configuration a new finite element model was created as shown in Figure 3.4-7 of the application. The summary of the evaluation is contained in Table 3.4-5 of the application. The largest influence from the changes involved in this amendment came from the addition of a second support tile configuration between the bottom the MSB and the liner plate of the VCC. For example, in the bottom plate and the bottom weld the local membrane plus bending stresses were previously approximately 10% of the allowable stress and under the revised calculation were found to be approximately 86% of the allowable stress.

For the MSB under off-normal conditions, several different loading cases were re-analyzed. The off-normal ambient temperature of the MSB shell for the minimum temperature of -40°F was revised as shown in the table in Section 11.1.1.3 of the application, however there was minimal impact on the resulting computed stresses. The off-normal handling load event design bases has been expanded in this amendment to also include a scenario of a loaded MSB being lowered into the VCC and impacting the VCC at a velocity of 0.75fps, which is equivalent to a drop of 0.1 inches. The summarized results are provided in Table 11.1.-1 of the application. The second condition evaluated for off-normal handling resulted in the increase in the computed stresses in the bottom plate of the MSB as well as in the weld between the bottom plate and the MSB shell. These were the most significant increases and for the bottom plate and the bottom weld where the combined local membrane stress plus the bending stress increased from 52% of the allowable stress to 98% of the allowable stress, but the design remains acceptable. The amendment added an off-normal pressurization scenario initiated by the breach of 10% of the fuel rods and control elements in the MSB. The resulting conditions produce a higher temperature in the sleeve/basket than the normal operation condition as well as a pressure increase. These components were analyzed under a condition of 445°F and a 10.0 psig pressure. The result of the analyses are summarized in the revised Table 11.1-2 in the application. The most significant stresses were in the bottom plate and the weld of the plate to the MSB shell, and in the sleeve/basket assembly. The combination of primary local membrane stress, bending stress, and secondary stresses for the bottom plate and weld was approximately 83% of the allowable stress. The same combination of stresses for the sleeve/basket assembly was approximately 88% of the allowable stress. These are acceptable stress levels.

MSB re-analyses were also performed for certain hypothetical accident conditions in this amendment. The drop event was re-evaluated with the new physical data. For the horizontal drop there were generally only increases in stresses in the amount of 10% of the allowable stress which previously had been no more than about 78% of the allowable stress. In the highest increase of stress, related to the weld of the structural lid to the MSB shell, the combination of local membrane stress and bending stress was increased from approximately 78% of the allowable stress to approximately 90% of the allowable stress. For the vertical drop there were greater changes in the percentage of stress allowables which was the result of the physical changes and the new configuration of the MSB support tiles located on the inside of the VCC liner. In this case the MSB shell, the bottom plate, the weld between them and the support ring weld were the elements with the largest increases in stress.

For the MSB shell the primary membrane stress changed from approximately 27% of the allowable stress to 97% of the allowable stress. For the local membrane plus bending the change in stress was from 25% to 82% of the allowable stress. For the bottom plate and the bottom weld of the plate to the shell the combination of the local membrane plus bending stress went from 25% of the allowable to 95% of the allowable. For the membrane stresses in the support ring weld the change was from 32% of the allowable stress to 60% of the allowable stress. The MSB was re-evaluated for its capability under a revised flood accident condition. Since the MSB is contained in the VCC it is the flood static head that is the predominant loading on the shell. The new design pressure for the MSB allows for the static head of water to reach a 120 foot flood depth without exceeding the allowable stresses. The shell buckling strength greatly exceeds the 120 foot flood depth.

The MSB was also re-evaluated for performance under pressurization. The accidental pressurization of the MSB is a result of the release of 100% of the rod and control element volume with all of the fission gases released. The temperature increase was determined to be 434°F, but the design evaluation was based on 440°F resulting in an equivalent pressure of 55.7 psig. This exceeds the original value 36.4 psig. The design pressure was selected to be 60.0 psig. Table 11.2-3 of the application presents the summary of the highest stresses along with each of the major elements in the MSB assembly. The bottom plate stresses and bottom weld stresses for the combination of local membrane plus bending stress changed from about 57% of the allowable stress to about 72% of the allowable stress. This represented the most significant change.

2.5 Structural Analysis of the MTC

The MTC design was re-evaluated under normal conditions of use for the lifting load cases since there were physical changes in weights and dimensions. The original criteria continue to be the design basis for lifted loads. The factor of safety against the yield stress of the material is at least 6.0 and the factor of safety against the ultimate stress of the material is at least 10.0 with the load having a 10% impact factor applied. The bounding MTC trunnion design was re-evaluated against the criteria and was found to fully meet the criteria as identified in Section 3.4.3.3 of the application. The shield door rail and the associated welds were also re-evaluated against the same criteria for the shear stresses and the bending stresses. The criteria were adequately met. The MTC wall capability to carry the loads transferred from the trunnions was also re-evaluated for this amendment. The factors of safety for combined shear and bending exceeded the required values. The MTC top cover plate and bolts were re-evaluated since the plate must resist the weight of the MTC if there were an inadvertent lift made while attempting to remove a loaded MSB. The criteria used for these structural elements were the allowable stresses for the plate and bolting material. The criteria were satisfied.

2.6 Materials Interactions

The significant materials issue in this application is the addition of a variety of control elements to be stored in the VSC-24 system. This means fuel assemblies containing burnable poison rod assemblies (BPRA’s) and thimble plug assemblies (TPA’s) would be added to the approved contents of the VSC-24 cask as shown in LAR 01-01, Table 1, page A-11a. No new fuel types were proposed and all fuel assemblies are limited to a burn-up of 45,000 MWd/MTU.

The BPRA’s proposed for storage contain one of several possible neutron poisons: boron carbide, hafnium, silver-indium-cadmium, borosilicate glass, or aluminum oxide. The BPRA’s may be zircaloy-4 or stainless steel clad. Of these materials, boron carbide, borosilicate glass, and aluminum oxide are all essentially inert under wet loading and dry storage conditions. The remaining metallic neutron poison materials are also inert under storage conditions and can have a slight galvanic reaction under wet loading conditions. As a defense in depth, to avoid any potential galvanic reaction with the MSB internals from the hafnium or silver-indium-cadmium released from BPRA’s, the applicant has specified that no BPRA rod may be loaded if there are known or suspected cladding failures if that rod contains hafnium or silver-indium-cadmium poison material. This requirement was added as Note 1 in Table 1 of the technical specifications.

TPA’s may be either hollow or solid zircaloy-4, stainless steel, or Inconel (a nickel-base material). These materials are identical to other fuel hardware materials and thus are compatible with the storage cask environment, cask materials, and other fuel element components.

2.7 Conclusions

Based on the information provided in the amendment to the FSAR and the supporting documentation, the staff concludes that the VSC-24 system will continue to meet the requirements of 10 CFR Part 72. The application adequately describes all structures, systems, and components (SSCs) that are important to safety and provides drawings and text in sufficient detail to allow evaluation of their structural effectiveness.

The SSCs important to safety are described in the VSC-24 amendment application in sufficient detail to enable staff evaluation of their structural effectiveness and are designed to accommodate the combined loads of normal, off-normal, accident and natural phenomena events. The VSC-24 system is designed to allow handling and retrieval of spent nuclear fuel for further processing or disposal. The staff concludes that no accident or natural phenomena events analyzed will result in damage of the system preventing retrieval of the stored spent nuclear fuel. The VSC-24 system is designed and fabricated so that the spent nuclear fuel is maintained in a subcritical configuration under normal, off-normal and analyzed accident conditions. The configuration of the stored spent fuel remains unchanged for this amendment.

The cask and its systems important to safety were evaluated to demonstrate that they will reasonably maintain confinement of radioactive material under normal, off-normal, and analyzed accident conditions.

The staff finds that the materials of the TPA's and BPRA's are either identical to other materials used in fuel assemblies, are chemically inert under wet loading and dry storage conditions, or are contained within cladding. Consequently, the staff finds that the addition of the specified BPRA's and TPA's are acceptable and that no adverse chemical, galvanic, or corrosion reaction will occur.

The staff concludes that the structural design of the VSC-24 continues to be in compliance with 10 CFR Part 72 and that the applicable design and acceptance criteria have been satisfied. The structural evaluation provides reasonable assurance that the VSC-24 system will continue to safely store spent nuclear fuel. This finding is based on a review that considered the regulation itself, appropriate regulatory guides, applicable industry codes and standards, accepted practices, and confirmatory analysis.

3.0 THERMAL EVALUATION

Staff reviewed the application to confirm that the cask and fuel material temperatures of the VSC-24 system will remain within the allowable values or criteria for normal, off- normal, and accident conditions. Staff verified that the temperatures of the fuel cladding (fission product barrier) will be maintained throughout the storage period to protect the cladding against degradation that could lead to gross rupture. This portion of the review is also to confirm that the thermal design of the cask has been evaluated using acceptable analytical and or testing methods.

3.1 Description of Changes

The proposed changes would allow the storage of fuel assemblies with selected control elements in the VSC-24 system. These proposed changes include a modification to the list of acceptable fuel types identified in the certificate to reflect the selected control elements. No additional fuel types were added. The applicant updated structural calculations to provide bounding analyses of the various design configurations to accommodate the selected control elements. The calculations were based on more conservative assumptions of fuel failures and the bounding physical parameters of the VSC-24 system. The results include a limit on the average gas temperature for the backfill helium. The applicant's use of average helium temperature in the MSB evaluation is a conservative assumption.

3.2 Thermal Analysis

Staff reviewed the impact of the proposed changes on the thermal analysis. The applicant demonstrated that a fuel cladding temperature limit of 712°F for normal and off-normal conditions, which is bounding for all types of fuel approved for storage in the VSC-24, is not exceeded provided the loaded fuel assemblies have a decay heat not greater than 1.0kW and that the uranium loading does not exceed 0.471 MTU per fuel assembly or 3.27 kg/in of fuel in the assembly. The 3.27 kg/in limit ensures that assemblies with fuel length shorter than 144 inches will not produce heat-generation levels exceeding the 6.94 W/in value used in the thermal analyses. The maximum allowable fuel burnup is 45,000 MWd/MTU. Cooling times are prescribed in Table 5.5-1 of the application for fuel burnups up to 52,000 MWd/MTU. The certificate requires a thermal decay heat per fuel assembly of less than or equal to 1kW.

The staff reviewed the reduction in maximum concrete local temperature from 300°F to 225°F. Staff also reviewed a revision to the upper end of the steady state off-normal ambient operating temperature from 125°F to 100°F. However, 125°F was still used in the analysis for an extreme environmental case to demonstrate compliance with the maximum heat load accident case required by ANS-57.9. The thermal analysis presented by the applicant shows that if the cladding temperature limits are not violated, other temperatures (i.e., concrete and RX-277 neutron shielding material) will remain below their limits. The applicant determined a maximum fuel cladding temperature, under off-normal conditions, of 705°F given a maximum decay heat load (24kW) and sustained 24 hour solar loads, and a 100°F ambient temperature. This temperature is bounded by the 712°F maximum fuel cladding temperature limit. For the accident condition, 125°F maximum ambient temperature with 12 hours of full solar load, a maximum sleeve temperature of 721°F was determined. Thus, the short term fuel cladding temperature limit of 1058°F remains bounding.

The staff finds that maintaining a thermal decay heat generation limit of 1.0kW per assembly will not result in damage to any structural materials under normal and off-normal conditions.

3.3 Pressure Analysis

Staff reviewed changes made to the FSAR analyses for off-normal and accident pressurization. The revised pressure analyses included the added gas contribution from off-normal (10% fuel rod and control element failure) and accident (100% fuel rod and control element failure) conditions from the addition of control elements to the fuel assemblies. The total gas content of the MSB, including the helium backfill and released fuel rod and control rod element fill gas was analyzed. The applicant calculated a helium pressure of 14.5 psia (±5psia) during backfilling operations. The applicant assumed that the average helium temperature in the MSB would be greater than 200°F when the MSB is sealed. For pressure considerations, the 14.5 psia and a temperature greater than 200°F assures an upper limit on the number of moles of helium in the MSB. During off-normal and accident conditions the average basket interior temperatures are shown to increase from the normal condition of 434°F to 436°F and 440°F, respectively. To demonstrate added conservatism, the applicant used a gas mixture bulk temperatures of 445°F and 460°F for off-normal and accident conditions. This resulted in maximum basket pressures under off-normal and accident storage conditions of 10.0 psig and 55.7 psig, respectively. The maximum normal operating pressure is 5.23 psig based on an average gas bulk temperature of 439°F. The values determined for off-normal and accident conditions remain bounded by the accident pressure value of 60.0 psig used in the pressure stress analysis.

3.4 Conclusions

The staff concludes that the thermal design of the cask is not negatively affected by the proposed changes to the VSC-24 FSAR, that the design remains in compliance with 10 CFR Part 72, and that the applicable design and acceptance criteria have been satisfied. The evaluation of the proposed changes provides reasonable assurance that the cask will continue to allow safe storage of spent fuel. This finding is reached on the basis of a review that considered the regulation itself, appropriate regulatory guides, applicable codes and standards, and accepted engineering practices.

4.0 CONFINEMENT EVALUATION

The applicant proposed changes to the MSB spent fuel confinement criteria to allow for the storage of control elements. The application addressed these changes and updated the FSAR confinement section. The following calculations were added to the confinement section of the FSAR:

1) Calculations on the activity level for each isotope using ORIGEN2 code.
   - Bounding case analysis was performed for a fuel assembly with 52 GWd/MTU burnup, 5 year cooled, and 5 wt% U-235.
   - Typical case analysis was performed for a fuel assembly with 35 GWd/MTU burnup, 10 year cooled and 5 wt% U-235 as a typical case.
2) Calculations for atmospheric dispersion and leakage rates were performed in accordance with NRC Regulatory Guide 1.145 and ANSI N14.5 (1997).
3) Dose calculations were performed for normal conditions and hypothetical accident conditions using bounding and typical cases.

4.1 Conclusions

The staff determined that although the applicant used 52 GWd/MTU as a bounding case, 45 GWd/MTU is the maximum allowable burnup allowed for the VSC-24. The staff has performed confirmatory calculations and has determined that the confinement section of the application provides reasonable assurance that the confinement boundary will be maintained as required by 10 CFR Part 72.

5.0 SHIELDING EVALUATION

This review included the calculation of the dose rates from both photon and neutron radiation at locations near the cask and at specific distances away from the cask. The regulatory requirements for providing adequate radiation protection to licensee personnel and members of the public include 10 CFR Part 20, 10 CFR 72.104(a), 72.106(b), 72.212(b), and 72.236(d). An overall assessment of compliance with 10 CFR Part 72 dose limits for members of the public is discussed in Section 10 (Radiation Protection) of the application and includes direct radiation, effluent releases, and radiation from other uranium fuel-cycle operations.

5.1 Shielding Design Features

The principle shielding components of the VSC-24 consists of a concrete storage cask, a steel-lead-steel composite transfer cask and a shielded canister. The storage cask is designed to provide both photon and neutron shielding. The principal components of the storage cask radial shielding are the MSB and the thick reinforced concrete body. The shielding at the top of the storage cask consists of carbon steel and RX-277 neutron shielding. The shielding at the bottom of the storage cask consists of carbon steel and concrete.

The transfer cask has different materials to provide gamma and neutron shielding. The radial gamma shielding is provided by a composite structure of steel-lead-steel. The transfer cask radial neutron shielding is provided by RX-277. The shielding at the top of the transfer cask consists of either RX-277 and carbon steel. The shielding at the bottom of the transfer cask consists of a thick steel plate.

5.2 Radiation Source Definition

The radiation source specification is presented in Section 5.2 of the application. Generic photon and neutron source terms were generated with the ORIGEN-2.1 computer code. The gamma source terms were binned into 13 energy groups. The energy spectrum of the neutron source term is based on the Watt-fission spectrum for Cm-244.

The applicant performed source term calculations for a bounding fuel assembly which contains a maximum of 0.5187 MTU, which contains more uranium than any fuel assembly to be stored. The applicant determined the minimum cooling time, for a variety of burnups and enrichments, required to satisfy the surface dose rate limits in TS 1.2.4. The source term includes photons from activated assembly hardware located within the fuel region (i.e., grid spacers and in-core control components).

The applicant proposed removal of the Gamma and Neutron source limits specified in TS 1.2.1 Table 1, “Characteristics of Spent Fuel to be Stored in the VSC-24 System.” The staff finds this is acceptable because the total source term and the source spectrums are provided by Table 5.2-1 and 5.2-2 in the FSAR and can be calculated from the information provided in Table 1.

The applicant limited the maximum amount of cobalt allowed in the fuel region to 46.7 grams for fuel assemblies in any location and 250 grams of cobalt for fuel assemblies located in the inner 12 MSB fuel sleeves. The applicant assumed cobalt impurities in Inconel-718, zircaloy, and stainless steel of 4694, 10, and 800 ppm, respectively. Measured cobalt impurities in Inconel grid spacers from a Westinghouse 14x14 assembly range from 890 to 1490 ppm. Another set of measurements resulted in a range of cobalt impurities from 186 to 3600 ppm. The value of 4694 ppm used to estimate the Co-60 source term for the grid spacers bounds these measured values. The applicant calculated the maximum quantity of cobalt in the active region of a pressurized water reactor (PWR) fuel assembly of 46.7 grams. To account for fuel assemblies which may have been irradiated with elements that have a much higher cobalt quantity, such as a stainless steel rod to replace a leaking fuel rod, the applicant performed an analysis to show that up to 285.3 grams of cobalt could be allowed in the inner 12 fuel sleeves of the MSB. The applicant limited the cobalt quantity to 250 grams.

For non-fuel hardware locations the applicant evaluated total cobalt quantities of 20.6 grams, 10.2 grams and 47.7 grams for the bottom nozzle, gas plenum region and the top nozzle region, respectively. The applicant stated that the cobalt quantities bound all U.S. PWR reactors and include the presence of BPRAs and TPAs. The applicant evaluated the source terms from the cobalt in the end fittings and applied scaling factor of 0.2 for the bottom nozzle and gas plenum zone and 0.1 for the top nozzle region, based on their location from the active fuel region. These scaling factors are consistent with current NRC acceptance criteria.

5.3 Shielding Model Specifications

5.3.1 Model Specifications

The model specifications for shielding are presented in Section 5.3 of the application. The applicant's shielding model for normal and accident conditions consists of a 3-D representation of the VSC-24 cask using the design drawings in Section 1.5. A description of the shielding configurations is presented in Section 5.3 of the application. The storage cask shielding models are depicted in Figures 5.3-1 through 5.3-5. The transfer cask shielding model is depicted in Figures 5.3-6 of the application.

5.3.1.1 Source Configuration

The radiation source is divided into four axial regions: bottom end fitting, fuel, gas plenum, and top end fitting. The relative positions of these source term regions are also depicted in the figures identified above. The fuel assemblies are homogenized within each fuel sleeve in the MSB. The end fittings and plenum regions are modeled as homogeneous regions of stainless steel, Inconel, and zircaloy.

Variation of the maximum peaking factor as a function of axial fuel length is provided in Table 5.2-4 of the application. The neutron profile was used to account for the non-linear buildup of neutron source terms (primarily Cm-244) as a function of burnup. The photon source distributions within the plenum, top end fittings, and bottom end fittings were assumed to be uniform.

5.3.1.2 Streaming Paths and Regional Densities

The storage cask shielding models included streaming paths for the inlet and outlet vents. The cask design eliminates other potential streaming paths. The applicant performed an evaluation of the potential for streaming through the inlet and outlet vents using Monte Carlo N-Particle Transport (MCNP) computer code.

The composition and densities of the materials used in the shielding analysis are presented in Tables 5.3-1 and 5.3-2 of the application. The applicant did not identify any materials that undergo changes in material density or composition from temperature variations.

The applicant explicitly modeled the ducts and materials surrounding them using MCNP, including the cannister and internals in the duct regions. The model dimensions and material specifications are consistent with the drawings in Section 1 of the FSAR and provide the basis for reasonable assurance that the VSC-24 cask was adequately modeled in the shielding analysis. The staff evaluated the SAR shielding models and found them to be acceptable.

5.4 Shielding Analyses

The shielding analyses were presented in Section 5.4 of the application. The applicant used the MCNP code to determine the dose rates from the storage and transfer casks. The applicant used ANSI/ANS Standard 6.1.1-1977 flux-to-dose conversion factors to calculate dose rates in the shielding analysis.

5.4.1 Storage Cask

The applicant performed a shielding analysis for the storage cask and transfer cask. The applicant determined the average surface dose rates on the side and top of the storage cask, along with the dose rates at the inlet and outlet vents. Additionally, MCNP code was used to determine the off-site dose rate. The applicant determined the dose rates on the side, top and bottom of the transfer cask to estimate the occupational doses during loading and storage.

The application presented calculations for normal condition dose rates of bounding and typical analyses for the VSC-24 cask. Calculated dose rates were summarized in Table 5.1-1 of the application. The maximum dose rates on the sides of the VSC-24 system were calculated to be 99 mrem/hr at the location of the top of the fuel assemblies and 86 mrem/hr at the location of the mid-point of the fuel assemblies. The average dose rate over the VSC-24 sides was calculated to be 86 mrem/hr. The maximum and average dose rates on the top of the VSC-24 system were calculated to be 245 mrem/hr and 150 mrem/hr, respectively. The applicant evaluated potential accident conditions around the storage cask in Section 11 of the application. None of the evaluated accidents increase the dose rate due to direct radiation from the cask.

For normal operations, the applicant increased dose rate limits in the proposed TS 1.2.4 in Section 12 of the application from 20 mrem/hr to 100 mrem/hr on the sides; from 50 mrem/hr to 200 mrem/hr at the top; and from 50 mrem/hr at the air inlets and outlets to 350 mrem/hr and 100 mrem/hr, respectively. These limits were increased as a result of revised specifications on the burnup, cool time and allowable decay heat limit. However, dose limits at the site boundary continue to be limited to below 25 mrem per year as determined in Section 10.4.

The applicant performed an MCNP calculation to determine the dose rates at the inlet and outlet vents. The applicant calculated the dose at the inlet and outlet vents of 315 mrem/hr and 69 mrem/hr, respectively. These are below the maximum dose rate criteria in TS 1.2.4 of 350 mrem/hr and 100 mrem/hr for the inlet and outlet, respectively.

Confirmatory calculations for the VSC-24 cask were made with the SAS4 module in the SCALE 4.4 computer code system. The staff homogenized each of the 24 fuel assemblies, similar to the approach taken by the applicant. A comparison between the applicant's results and the staff's confirmatory calculations showed a slight variation in the results, which is expected when two different codes are used for shielding calculations. The staff performed radial dose rate calculations for gamma radiation only, since the applicant found the neutron dose rate to be small compared to the gamma dose rate. The average surface gamma dose rate on the side of the cask was calculated by staff to be 87.7 mrem/hr. The staff also performed additional evaluations to randomly check other points on both cooling tables using the applicant's importance functions. The staff found that dose rates for other points on the cooling table are below the criteria in TS 1.2.4. Overall, the differences between the applicant's and staff's confirmatory results fell within acceptable bounds.

5.4.2 Transfer Cask

The maximum dose rates for the transfer cask are shown in Table 5.1-2 of the application. The maximum dose rates on side of the transfer cask are 805 mrem/hr at the location of the top of the fuel assemblies and 632 mrem/hr at the location of the midpoint of the fuel assemblies. The maximum dose rates on top and bottom surfaces of the transfer cask are 731 mrem/hr and 3758 mrem/hr, respectively.

Confirmatory calculations for the VSC-24 transfer cask were performed with the SAS4 module in the SCALE 4.4 computer code system. The staff's model of the MSB is the same one used in the storage cask calculations. A comparison between the applicant's results and the staff's confirmatory calculations showed a slight variation in the results which is expected when two different codes are used for shielding calculations. The staff evaluated the dose rates for the radial surfaces. The staff's analyzed maximum dose rate on the side of the VSC-24 was determined to be 596 mrem/hr at the midplane of the fuel assemblies. Overall, the differences between the applicant's and staff's confirmatory results fell within acceptable bounds.

5.4.3 Occupational Exposures

Design-basis fuel was used to estimate occupational exposures during cask operations. Section 10 of the application presents estimated occupational exposures using the calculated dose rates for the locations shown in Figures 5.1-1 and 5.1-2.

5.4.4 Off-Site Dose Calculations

Direct-path off-site dose rates are presented in Section 10 of the application for a single cask and an array of casks. Direct-path dose rates for off-site locations are for a bounding fuel loading, level topography, and a 100% occupation time. The applicant determined the off-site dose from a 5x5 array of casks. The applicant evaluated the direct dose from a single cask loaded with fuel assemblies with a burnup of 52 GWd/MTU and a cooling time of 5 years, which provides higher dose rates than those determined in Section 5 of the application. Table 10.4-1 of the application shows the dose for one year exposure (2000 hours) at specified distances from the array of storage casks. Table 10.4-4 shows the dose at various distances from the array of casks from both direct dose and releases of radioactive material. The applicant showed that the VSC-24 array of casks continue to meet the 25 mrem/yr criteria at a distance of approximately 260 meters. The staff has reasonable assurance that continued compliance with 10 CFR 72.104(a) can be achieved.

The general license user of the VSC-24 cask must perform a site-specific evaluation, as required by 10 CFR 72.212(b), to demonstrate operational compliance with 10 CFR 72.104(a). The actual doses to individuals beyond the controlled area boundary depend on site-specific conditions such as cask-array configuration, topography, demographics, and use of engineered shielding features (e.g., berm). In addition, the dose limits in 10 CFR 72.104(a) include doses from other fuel cycle activities in the region such as reactor operations. Consequently, final determination of compliance with 10 CFR 72.104(a) is the responsibility of each general licensee.

The general licensee will also have an established radiation protection program as required by 10 CFR Part 20, Subpart B, and will demonstrate compliance with dose limits to individual members of the public, as required by 10 CFR Part 20, Subpart D, by evaluations and measurements.

5.5 Conclusions

The staff found that the application sufficiently described shielding design features and design criteria for the components important to safety. The VSC-24 radiation shielding features are sufficient to meet the radiation protection requirements of 10 CFR Part 20, 10 CFR 72.104, and 10 CFR 72.106. The staff concludes that the design of the shielding system for the VSC-24 is in compliance with 10 CFR Part 72 and the applicable design and acceptance criteria have been satisfied. The evaluation of the shielding system provides reasonable assurance that the VSC-24 cask will continue to provide safe storage of spent fuel. This finding is based on a review that considered the specifications in the application, the regulations, appropriate regulatory guides, applicable codes and standards, and accepted engineering practices.

6.0 CRITICALITY EVALUATION

6.1 Criticality Design Characteristics and Features

The applicant requested revision of the fuel specifications to expand the types of fuel and control elements suitable for storage in the VSC-24 system. The applicant provided Table 1-a to TS 1.2.1. This table completely describes the fuel assemblies to be stored, including the fuel assembly layout for each assembly type. The applicant also deleted the minimum burnup requirements from TS 1.2.1 and deleted TS 1.2.1 Figure 1, "Fuel Criticality Acceptability Curve." The applicant replaced those requirements with minimum boron requirements in TS 1.2.6, which included Figures 2 through 9 of the Technical Specifications, showing the variation of boron level with enrichment for each fuel assembly type.

6.2 Model Specifications

To support these revised fuel assembly specifications, the applicant performed a criticality analysis using the MCNP4A code. The applicant explicitly modeled the fuel assemblies and the VSC-24 cask as shown in the figures in Section 6. For each fuel assembly class, the applicant varied the amount of soluble boron (ppm) at five different enrichment levels, bounding the maximum enrichment proposed for storage.

6.3 Criticality Analysis

The applicant determined the minimum amount of boron such that the calculated k-eff, including bias and uncertainty, is less than 0.95 for optimum moderation. The applicant performed an evaluation of each fuel assembly type with the respective amount of boration to determine the base case identification of the most reactive set of boron level and enrichment. A sensitivity analysis was performed for each fuel assembly class to show the variation of reactivity for various parameters, such as assembly position within each fuel sleeve, fuel sleeve thickness, MSB position within the MTC and fuel pellet diameter. The applicant determined that the maximum k-eff increase from these sensitivity analyses is 0.0034Δk.

Additionally, the applicant requested removal of TS 1.2.10, “Time Limit for Draining the MSB.” The applicant demonstrated the lack of the need for the TS using a sensitivity analysis with the cask partially full of water. The applicant evaluated the VSC-24 cask partially flooded with borated water at four different levels: 7/8-flooded, 3/4-flooded, 5/8-flooded and 1/2-flooded. The applicant determined that maximum reactivity occurred at fully flooded and that there was a monotonic decrease in reactivity with decreasing water level. The staff found the applicant's analysis acceptable to demonstrate removal of TS 1.2.10.

The staff performed confirmatory criticality calculations using the 44groupndf5 cross section set in the SCALE 4.4a computer code system. The staff performed criticality calculations for the B&W 15x15 fuel assemblies, with and without control components. The staff varied the water density in the analysis to find optimum moderation.

6.4 Benchmark Comparison

The applicant performed a benchmarking analysis for the MCNP computer code system. The applicant chose 38 critical experiments which are described in Table 6.4-11 of the application. The applicant determined the Upper Subcritical Limit (USL) using Method No. 1 from NUREG/CR-6361, "Criticality Benchmark Guide for Light-Water-Reactor Fuel in Transportation and Storage Packages." The applicant evaluated the USL for a number of parameters, such as boron concentration, fuel rod pitch, water/fuel volume ratio, and energy weight loss. The most limiting USL was calculated based on the limiting value for each parameter and the lowest USL was taken to be the bounding value. The applicant determined the bounding USL to be 0.93824.

The staff reviewed the applicant's benchmark analysis and agrees that the critical experiments chosen are relevant to the cask design. The staff found the applicant's method for determining the calculation bias acceptable. The staff also verified that only biases that increase k-eff have been applied.

6.5 Conclusions

The cask and its spent fuel transfer systems are designed to be subcritical in all configurations. The criticality design is based on favorable geometry and soluble poisons in the spent fuel pool. Based on the information provided in the application and the staff's own confirmatory analyses, the staff concludes that the VSC-24 system meets the acceptance criteria specified in 10 CFR Part 72. The staff reviewed the applicant's benchmark analysis and agrees that the critical experiments chosen are relevant to the cask design. The staff found the applicant's method for determining the USL acceptable. The staff also verified that only biases that increase k-eff have been applied.

7.0 CONDITIONS FOR CASK USE AND TECHNICAL SPECIFICATIONS

The proposed certificate changes for this amendment are as follows:

1) TS 1.1.1, “Regulatory Requirements,” Item 5, changed flood condition velocity from 25 fps to 17.7 fps.

2) TS 1.2.1, “Fuel Specification,” Table 1:
   – Changed maximum burnup from 51,800 MWd/MTU to 45,000 MWd/MTU and removed previous Note 1;
   – Added new Note 1 to specify that high cobalt assemblies must not be loaded into the 12 fuel sleeves located around the perimeter of the MSB.
   – Added Note 2 to specify that failed BPRAs or TPAs may be loaded provided that they do not contain Silver-indium-cadmium or hafnium material. BPRAs containing these materials must have intact fuel cladding, with no known or suspected defects beyond hairline cracks and pinhole leaks.
   – to require that silver-indium-cadmium or hafnium bearing poison rods must be intact with no known or suspected gross cladding failures;
   – Clarified allowable fuel cladding as zircaloy-4;
   – Deleted minimum burnup requirements and Figure 1, “Fuel Criticality Acceptability Curve;”
   – Added Table 1-a, “Fuel Assembly Class Characterization Parameters;”
   – Added BPRAs, TPAs, poison clusters, and plugging clusters to authorized fuel class/types;
   – Deleted Gamma Source and Neutron Source requirements;
   – Added FSAR Table 5.5-1 as additional requirement for post irradiation time limit; and
   – Increased assembly weight to account for additional fuel control elements.

4) TS 1.2.4, “Maximum External Surface Dose Rate:”
   – Changed “Limit/Specification” text from “surface dose” to “surface average dose;”
   – Increased dose rate limit values from 20 mrem/hr to 100 mrem/hr on the sides, from 50 mrem/hr to 200 mrem/hr on the top, from 50 mrem/hr to 350 mrem/hr at the air inlets, and from 50 mrem/hr to 100 mrem/hr at the air outlets.
   – Added 25 mrem/yr 10 CFR Part 72 requirement to “Objective.”

5) TS 1.2.6, “Boron Concentration in the MSB Cavity Water,” revised Figure 2 and added Figures 3 through 9 for fuel assembly class minimum required soluble boron curves.

6) TS 1.2.10, “Time Limit for Draining the MSB,” deleted.

In addition, the certificate format has been revised in accordance with the updated NRC certificate format.

The staff has reviewed these changes, as discussed in this Safety Evaluation Report, and have found them acceptable.

CONCLUSION

The certificate is being revised to modify the allowable fuel specifications to include the storage of specific fuel control elements as integral components to fuel assemblies. This change does not affect the ability of the VSC-24 to continue to meet the requirements of 10 CFR Part 72.

Issued with Certificate of Compliance No. 1007, Amendment No. 4,
on January 22, 2003.