SAFETY EVALUATION REPORT
RELATED TO THE TOPICAL SAFETY ANALYSIS REPORT
FOR THE NAC STORAGE/TRANSPORT CASK, REVISION 2,
SUBMITTED BY NUCLEAR ASSURANCE CORPORATION

U.S. Nuclear Regulatory Commission

Office of Nuclear Material Safety and Safeguards

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1.0 GENERAL DESCRIPTION

1.1 Introduction

This Safety Evaluation Report (SER) documents the staff's review and evaluation of the Topical Safety Analysis Report (TSAR) for the NAC Storage/Transport Cask, June 1987 (Reference 1). The TSAR was prepared by Nuclear Assurance Corporation (NAC), using the Regulatory Guide 3.48 (Reference 2) format, as applicable. This SER utilizes the format of Regulatory Guide 3 (CE-306-4) (Reference 3) with some differences in the section numbering.

The staff's review of the TSAR addresses the handling, transfer and storage of spent fuel in a NAC Storage/Transport Cask (NAC S/T) for an at reactor site independent spent fuel storage installation (ISFSI). Such storage in a ISFSI would be licensed under 10 CFR Part 72, "Licensing Requirements for the Storage of Spent Fuel in a Independent Spent Fuel Storage Installation (ISFSI)". In this TSAR a single dry storage cask design, the NAC S/T is presented.

The staff's assessment is based on the proposed design's meeting the applicable requirements of 10 CFR Part 72, found under Subpart E, "Siting Evaluation Factors", Subpart F, "General Design Criteria", and Subpart G, "Quality Assurance", and of 10 CFR Part 20 for radiation protection for on-site receipt and storage of spent fuel in an ISFSI. Decommissioning, to the extent that it is treated in this TSAR, presumes unloading of a NAC S/T cask at the reactor site and subsequent decontamination of the cask prior to its disposition or disposal. Use or certification of the NAC S/T cask under 10 CFR Part 71, for off-site transport of spent fuel, is not a subject of this safety evaluation.

This review also does not address requirements for physical protection under Subpart H, "Physical Protection," of 10 CFR Part 72 or under 10 CFR Part 73, "Physical Protection of Plants and Materials."

1.2 General Description of the Storage Cask

1.2.1 Cask Design Characteristics

The NAC S/T (see Figure 4.2-1) was developed by the Nuclear Assurance Corporation, and is designed for the storage and shipment of irradiated spent fuel assemblies. The NAC S/T cask is a right circular cylinder of multi-wall construction with a 38.1-mm (1.5-in) thick inner shell and a 66.8-mm (2.63-in) thick outer shell of stainless steel separated by 81.3 mm (3.2 in) of lead shielding. The inner and outer shell are connected to each other at each end by an austenitic stainless steel ring and plate. The upper end of the cask is sealed by an austenitic stainless steel bolted closure lid which is 165.1 mm (6.5 in) thick in the edge flange region and has a 25.4-mm (1-in) inner closure plate and a 139.7-mm (5.5-in) outer closure plate. The closure plates are separated by 50.8 mm (2.0 in) of lead shielding. The closure lid utilizes a double barrier seal system with two metallic O-rings forming the seal. A third, optional, closure seal is seal welding the stainless steel cover of the upper solid neutron end cap skirt to the cask body. The lower end of the cask is 152.4-mm (6.0-in) thick austenitic stainless steel with a 25.4-mm (1.0-in) outer closure plate. The bottom end and the closure plate are separated by 45.7 mm (1.80 in) of lead shielding. The overall dimensions of the cask are 4796 mm (188.8 in) long and 2388 mm (94 in) in diameter. The unloaded cask weighs approximately 74 tonne (82 ton). The loaded cask, including stored fuel and contained water, is less than 113 tonne (125 ton).

Neutron emissions from the stored fuel are attenuated by an integral neutron shield located on the outside of the outer shell which contains a 177.8-mm (7.0-in) thickness of borated solid neutron shield material. Neutron emissions from the top of the cask are attenuated during storage by a 76.2-mm (3.0-in) thick solid neutron shield cap encased in stainless steel. This shield cap is placed on top of the cask after fuel loading.

The fuel basket has 26 cavities, each 223 mm (8.78 in) square, to hold the intact design basis fuel assemblies. The fuel cavities are aluminum square tubes which are separated and supported by an aluminum and stainless steel grid of spacers and tie bars to provide water flux traps for criticality control during underwater fuel loading and to transmit loads to the exterior basket aluminum castings. The castings are included to assist in uniform heat transfer from the fuel basket to the cask interior wall and to minimize internal cask free space. Sheets of borated neutron poison material (Boral) are captured along the outer walls of the fuel tubes.

The NAC S/T cask body has six attachment points for bolt-on trunnions. Four of these are located on the top stainless steel forging, spaced 90 degrees apart, and are used for lifting the cask. Two trunnion supports, 180 degrees apart, located near the bottom are used when rotating the cask to or from a horizontal position. They are off-set three inches from the cask centerline to assure proper rotation.

The 152.4-mm (6-in) diameter lifting trunnions are attached to the upper ring of the cask body with ten 44.45-mm (1.75-in) diameter bolts. Eacn lifting trunnion is designed to meet the requirements of NUREG-0612 for a non-redundant lifting fixture. The 127-mm (5-in) diameter rotation trunnions are attached to the lower ring of the cask body with eight 28.6-mm (1.125-in) diameter bolts. The rotation trunnions are designed to support 3.04 times the empty cask weight based on the application of a 3.0 g longitudinal load at the cask cavity center.

The NAC S/T cask has four containment penetrations; one cask cavity drain, one cask cavity vent, one inter-seal test port, and one inter-seal pressure transducer port. Each of these penetrations is in the single lid and utilizes double barrier seal containment.

The cavity drain line penetrates the closure lid and terminates at a sump relief in the bottom of the cask cavity. This is used to drain water from the cask cavity after underwater fuel loading. It is also used during the drying and helium back-filling of the cask cavity. The drain valve is of the quick-disconnect type and not analyzed as part of the primary containment system. A bolted support plate surrounds and protects the valve and provides two metal

O-ring seals as the primary and secondary containment barriers. A second cover plate fits over the support plate. This cover plate is bolted and provides two additional metal O-ring seals.

The cavity vent line penetrates the cavity through the closure lid. The line terminates in a quick-disconnect type valve recessed into the closure lid. The quick-disconnect valve is not analyzed as part of the primary containment. The valve is surrounded and protected by a bolted support plate with two metal O-rings providing the primary and secondary seals. A second cover plate fits over the support plate. The cover plate is bolted and provides two additional metal O-ring seals.

The inter-seal test line penetrates the closure lid to the space between the two O-ring seals. The line terminates in a quick-disconnect type valve recessed into the closure lid. The quick-disconnect valve is not analyzed as part of the primary containment. The valve is surrounded and protected by a bolted support plate with two metal O-rings providing the primary and secondary seals. A second cover plate fits over the support plate. The cover plate is bolted and provides two additional metal O-ring seals.

A single pressure transducer line also penetrates the closure lid and terminates in the space between the two closure lid O-ring seals. The transducer itself is recessed into the lid, but is not analyzed as forming the primary seal. Output wires from the transducer lead through a hermetically sealed feed-through which is part of a bolted support plate with two O-rings which form the primary and secondary containment seals. If the ISFSI operator desires continuous inter-seal pressure monitoring, the output wires then lead through a second hermetically sealed feed-through in a bolted cover plate with two additional metal O-ring seals.

The support skid will be used for shipping the empty cask from the manufacturing facility to the storage site.

1.2.2 Operational Features

The NAC S/T Cask is designed to safely store 26 intact design basis PWR fuel assemblies. Each fuel assembly may have an initial enrichment as high as 3.3 w/o U-235, as much as 35,000 MWd/MTU burnup, a decay time of no less than five years after reactor discharge and generate up to 1 kW of decay heat (total 26 kW per NAC S/T cask).

The heat rejection capability of the NAC S/T cask maintains the maximum fuel rod clad temperature below 380°C (716°F), based on normal operating conditions with a 26 kW decay heat load, 47°C (116°F) ambient air, and full insolence. The fuel assemblies are stored in an inert helium gas atmosphere.

The shielding features of the NAC S/T cask are designed to maintain the maximum combined gamma and neutron surface dose rate to less than 100 mrem/hr under normal operations conditions.

The criticality control features of the NAC S/T cask are designed to maintain the neutron multiplication factor k-effective (including uncertainties and calculational bias) at less than 0.95 under all conditions.

1.2.3 Cask Contents

The type of spent fuel to be stored in the NAC S/T cask is LWR fuel of the PWR type. PWR fuel is made of short cylinders (pellets) or high-fired ceramic uranium dioxide (UO_2). These pellets are 9.4mm (0.37 in) in diameter and 15.2mm (0.60 in) long. A 3658mm (144 in) long stack of 240 of these pellets are loaded and hermetically sealed into a zirconium alloy tube. Fuel rods are assembled into bundles in a square array, each spaced and supported by grid structures. The assembly has a top and a bottom fitting. A PWR assembly consists of a 15 x 15 array of individual rods. The overall dimensions are 214.5mm (8.45 in) square by 4064mm (160 in) long. Each assembly contains about 453 kilograms (999 lbm) of uranium in the form of UO_2. The standard Westinghouse 15 x 15 fuel assembly is used as the reference design in this TSAR.

Design Basis Fuel

Fuel Type - PWR, Westinghouse 15 x 15
- 3.3 w/o U-235 maximum initial enrichment
- 26,000 MWd/MTU minimum burnup for maximum initial enrichment
- 35,000 MWd/MTU maximum burnup
- 1 kW per assembly maximum decay heat
- Approximately 5-year decay time after reactor discharge

Quantity - 26 design basis fuel assemblies per NAC S/T cask

1.3 Identification of Agents and Contractors

Nuclear Assurance Corporation (NAC) provides the design, engineering, analysis and quality assurance for the NAC S/T cask.

NAC is a privately-owned, United States Corporation (Delaware) whose principal office is located at:

6251 Crooked Creek Road
Norcross, Georgia 30092

The NAC S/T cask may be manufactured by one or more qualified organizations.

There are no other agents of contractors involved with the NAC S/T cask.

1.4 Generic Cask Arrays

The ISFSI may include one or more NAC S/T casks. The NAC S/T cask may be stored vertically on its bottom plate or horizontally upon its support skid. The TSAR provides analyses of typical storage arrays (both horizontal and vertical storage) including:

- a single cask
- a four-cask square array
- a ten-cask linear array (two rows of five casks each)
- a 140-cask array (14 rows of 14 casks each with every third row removed).

2.0 PRINCIPAL DESIGN CRITERIA

2.1 Introduction

Subpart F of 10 CFR Part 72 sets forth general design criteria for the design, fabrication, construction, testing and performance of structures, systems and components important to safety in an independent spent fuel storage installation (ISFSI). In this chapter, we discuss the applicability of these criteria to the Nuclear Assurance Corporation Storage/Transportation (NAC S/T) spent fuel storage cask and the degree to which the NAC TSAR is in compliance with these criteria. Section headings in this chapter generally correspond to sub-sections of Subpart F of Part 72.

2.2 Fuel to be Stored

The NAC S/T cask is designed to store in a dry condition irradiated PWR fuel from nuclear power stations. The design basis fuel is UO_2 with an initial enrichment of 3.3 percent U-235 by weight or less, clad in Zircaloy. The design basis fuel is assumed to have been irradiated to an exposure of 35,000 MWd/MTU and cooled for five years. Estimates of the radionuclide activity in spent fuel described above were made using the ORIGEN computer code.

2.3 Quality Standards

Quality standards for structures, systems and components important to safety are required by 10 CFR Section 72.72 (a). Section 3.4 of the TSAR identifies cask components classified as important to safety. A quality standard provides numerical criteria or acceptable methods or both for the design, fabrication, testing, and performance of these structures, systems and components important to safety. These standards should be selected or developed to provide sufficient confidence in the capability of the structure, system, or component to perform the required safety function. Since quality standards are generally embodied in widely accepted codes and

standards dealing with design procedures, materials, fabrication techniques, inspection methods, etc., judgments regarding the adequacy of the standards cited by the NAC S/T TSAR are presented in the sections of this report where the standards are applicable.

2.4 Protection Against Environmental Conditions and Natural Phenomena

Section 72.72 (b) of 10 CFR Part 72 requires the licensee to provide protection against environmental conditions and natural phenomena. Section 3.2 of the TSAR describes the structural and mechanical criteria for tornado and wind loadings, flood potential, tornado missile protection, seismic design, snow and ice loadings, thermal loadings, combined load criteria and structural design criteria.

In this section, the discussion is limited to the adequacy of the criteria for protecting against environmental conditions and natural phenomena. The technical basis for accepting these criteria is defined by the regulatory requirement to consider the most severe of the natural phenomena reported for the site with appropriate margins to take into account the limitations of the data. Since the NAC S/T cask was not designed for a specific site, the regulatory requirement is interpreted to mean that protection against environmental conditions and natural phenomena should be provided for either by the limits specified in the TSAR or for the most severe of the natural phenomena that may occur within the boundaries of the United States.

2.4.1 Tornado and Wind Loading

The TSAR establishes 160.93 m/s (360 mph) in Section 3.2.1.1 as the design basis tornado wind speed. This is in accordance with Regulatory Guide 1.76 (April 1974).

2.4.2 Flood

While no design basis for flood was established, the TSAR provides limits for submergence below which no breach of containment will occur and for current velocity below which no tipover will occur. It remains for the applicant to set the site-specific design criteria for flood and reference the TSAR to show the cask's ability to meet these criteria.

2.4.3 Seismic

A horizontal acceleration of 0.25 g was established as a basis for seismic design in Section 3.2.3. This peak acceleration reflects 10 CFR Part 72.65 for ISFSI sites east of the Rockies. The TSAR analysis interpreted this requirement as referring to only one direction. However, the staff interpreted this requirement to mean that this acceleration should be combined vectorially with a component normal to this direction resulting in a maximum horizontal ground acceleration of 0.35 g. In addition, Regulatory Guide 1.60 requires that the vertical acceleration used be 2/3 of horizontal so that 0.17 g is the acceleration in the vertical direction.

2.5 Protection Against Fire and Explosions

Pursuant to 10 CFR Section 72.72 (c), the licensee is required to provide protection against fires and explosions. In section 3.3.6 the TSAR establishes the design basis fire of 800°C (1475°F) for one-half hour duration. This is a basis established for Type B shipping casks under 10 CFR Part 71, Section 71.73, "Hypothetical Accident Conditions," Subsection 71.73 (a)(3), "Thermal." As such, it constitutes an upper bound that is unlikely to be exceeded within a nuclear power plant site. While no design basis for explosion was established, the TSAR provides maximum allowable external pressures below which no loss of containment will occur. It remains for the applicant to set the site-specific criteria for explosion and reference the TSAR to show the cask's ability to satisfy these criteria.

2.6 Confinement Barriers and Systems

Pursuant to 10 CFR Section 72.72 (h)(1), the licensee must protect the fuel cladding against degradation and gross ruptures. The TSAR provides analyses and cites data in Sections 3.3.1, 3.3.2, 3.3.3 and 4.8.2.4 supporting the case that dry storage of spent fuel does not cause degradation and gross rupture of the cladding.

Section 3.3.9 (Heat Rejection) of the TSAR addresses the issue posed by 10 CFR Section 72.72(h)(1) by acknowledging that, "...fuel cladding integrity shall not be degraded during 20-year normal storage operations". However, the ANS-57.9 and PNL references cited to justify temperature limits as high as 380°C are no longer considered to provide the governing criteria for assuring fuel cladding integrity.

In view of this situation, the reviewers conducted an investigation directed toward determining the adequacy of the cladding under the specified TSAR storage conditions. For protection to be adequate, the design of the cask should be such that degradation after at least a 20-year storage life should not preclude the ability of the cladding to resist gross rupture during normal operations associated with cask unloading and subsequent fuel rod handling operations.

After reviewing the current research relating to spent fuel cladding damage mechanisms, the reviewers concluded that a diffusion controlled cavity growth (DCCG) mechanism was the only mechanism of damage for dry storage applicable to the storage conditions of the fuel rods that could cause degradation and gross rupture of the cladding. Under the influence of stress and temperature, this damage mechanism progresses by the nucleation and growth of cavities along grain boundaries. This damage mechanism is serious since it can progress without external evidence of damage, may not cause pin holes or through cracks to relieve the internal pressure, and manifests itself by a sudden non-ductile type of fracture. The staff has therefore paid particular attention to evaluating the potential for cladding damage from this mechanism for the conditions of storage specified in this TSAR.

The only parameters that the cask designer may control to prevent cladding degradation or gross rupture in an inert environment are the maximum initial temperatures of the fuel rods and their temperature decay characteristics. Both are governed by the quantity, specific power, and age of the fuel assemblies, and by the heat dissipation properties of the cask. The TSAR addresses the general thermal characteristics of the cask in Section 4.8.2. This SER addresses the thermal evaluation in Chapter 4 and fuel cladding integrity in Appendix A.

10 CFR 72.72(h)(3), though specifically referring to ventilation and off-gas systems that are normally associated with an ISFSI, is interpreted to apply to cask storage as a requirement to confine airborne radioactive particulate materials during normal or off-normal conditions. Consequently, closures secured by bolts or other fasteners should be designed to limit leakage to levels that do not exceed Regulatory limits 72.67 and 72.68. The NAC design features a single closure lid incorporating two metallic "O" ring seals. The design criterion for each seal is a leakage rate not exceeding 10^{-6} atm-cm^3/sec of helium for a cavity pressure of 125 psig. The staff considers the leakage rate to be acceptable for maintaining the cask helium atmosphere for projected storage periods of at least 20 years. The design also provides capability to detect seal failure through pressure monitoring. If seal failure should occur, leak tightness can be restored by welding the stainless steel cover with the neutron shield and cap skirt to the cask body. The acceptability of the leak criterion with respect to leakage of airborne radioactive particulate and gaseous materials is addressed in Chapter 7 of this SER.

2.7 Instrumentation and Control Systems

Pursuant to 10 CFR Section 72.72 (i), the licensee must provide instrumentation and control systems that monitor systems important to safety over anticipated ranges for normal and off-normal operation. The NAC S/T cask incorporates a pressure monitoring device which serves as a cask tightness surveillance system. The design criteria and description of this system appears in Section 3.3.3.2 of the TSAR. Considering the passive

nature of cask storage, the staff finds the gauge system acceptable instrumentation for this requirement.

2.8 Criteria for Nuclear Criticality Safety

Section 72.73 of 10 CFR Part 72 requires that spent fuel handling, transfer and storage systems be designed to be maintained subcritical. The margins of safety should be commensurate with the uncertainties in the handling, transfer and storage conditions, in the data and methods used in the calculations, and in the immediate environment under accident conditions. Section 72.73 also requires that the design be based on either favorable geometry or permanently fixed neutron-absorbing materials. Section 3.3.4 of the NAC S/T TSAR addresses nuclear criticality safety criteria. Criticality analysis and prevention are reviewed in Chapter 6 of this report.

The TSAR establishes a maximum effective multiplication factor of 0.95 for all credible configurations and environments for the prevention of criticality. This factor is widely accepted as a criticality prevention limit, and the staff concurs with its application to the NAC S/T cask.

2.9 Criteria for Radiological Protection

Section 72.74 of 10 CFR Part 72 requires that the licensee provide adequate (a) protection systems for radiation exposure control, (b) radiological alarm systems, (c) systems for monitoring effluents and direct radiation, and (d) effluent control systems in a radiological protection program. Section 3.3.5 of the TSAR addresses radiological protection. The detailed evaluation for compliance with the regulation is discussed in Chapters 5, 7, and 10 of this SER.

The principal design features of the NAC S/T cask for exposure control are the inherent shielding capability of the cask and the integrity of the seals at the closure joints. Radiological alarm systems and systems for monitoring effluents and direct radiation are not applicable to the design of

the storage cask. Effluents are not a normal consequence of the passive dry storage operation; consequently, control systems to provide radiological protection for this condition are not applicable. Only provision (a) above is applicable to the cask with respect to shielding capability and the possibility of leakage from seals that may degrade or suffer damage as a result of an accident.

However, it should again be noted, as in Section 2.7 above, that the sealing system of the cask uses a pressure monitoring device as a tightness surveillance system. Leakage past the outer metallic seal will be manifested by a drop in inter-seal pressure.

The shielding capability of the cask for gamma rays relies primarily upon the thickness and attenuation property of the lead and steel cylinder and the lead and steel closure lids which comprise the primary barriers to radiation. The cask must maintain its structural integrity under loadings associated with normal operation, accident events, natural phenomena, and environmental conditions. Of particular concern is the response of the cask to dynamic loading conditions associated with cask drop and/or tip over. It is essential to demonstrate that its fracture toughness is sufficient to resist catastrophic brittle fracture under the assumption that undetected flaws may exist at locations of maximum primary membrane or bending stress. Resistance to brittle fracture is discussed in Section 4.2.1.1.1 of the TSAR, and a review of this topic is presented in Section 3.4.4.1.4 of this SER.

The TSAR also establishes in Section 3.3.5.2 (Criteria) the surface dose limit as 100 mrem/hr. The staff believes that this limit is acceptable provided the distance to the site boundary for a single cask is not less than 250 meters (820 feet) (see Sections 5.2 and 5.4 of this SER). However, in finding these limits acceptable for a 250 meter site boundary distance for a single cask, the staff notes that for site-specific analyses consideration must be given to cumulative dose rate because of reactor operations and to individual residency time at or near the site boundary (The nearest individual has been conservatively assumed in this evaluation to be present continuously at the site boundary).

2.10 Criteria for Spent Fuel and Radioactive Waste Storage and Handling

Pursuant to 10 CFR Section 72.75, the licensee is required to design the spent fuel storage and waste storage systems to ensure adequate safety under normal and accident conditions. These systems must be designed with (a) a capability to test and monitor components important to safety, (b) suitable shielding for radiation protection under normal and accident conditions, (c) confinement structures and systems, (d) a heat removal capability having testability and reliability consistent with its importance to safety and (d) means to minimize the quantity of radioactive wastes generated.

This section of the regulations defines the requirements for the spent fuel storage system within the context of the entire ISFSI. The TSAR presents a summary that addresses only spent fuel loading of the cask in Section 3.3.7. Actually, the entire TSAR serves to demonstrate compliance with the details of this part of the regulations.

2.11 Criteria for Decommissioning

Pursuant to 10 CFR Section 72.76, the licensee is required to design the ISFSI for decommissioning. For dry cask storage, this requirement applies to the cask design itself. Thus, decommissioning provisions should address decontamination of the cask components following removal of the radioactive spent fuel. The quantity of radioactive wastes produced and contamination of equipment should be minimized. The TSAR addresses this requirement in Section 3.5 in detail.

3.0 STRUCTURAL EVALUATION

3.1 Area of Review

This chapter evaluates the structural response of the NAC S/T cask to loadings under normal operating conditions, accident conditions and loads due to environmental conditions and natural phenomena.

The review procedure addresses the assumed loads and material properties, the allowable stress limits and an evaluation of the structural analysis provided in the TSAR for each of the components and systems important to safety. The structural review consists of a review for the storage requirements of 10 CFR 72 only. No review has been made for transportation requirements.

3.2 Acceptance Criteria

The structural integrity of the cask will be deemed adequate if it can be demonstrated that the stresses induced by the loads noted in 3.1 above are lower than the allowable stress limits for the the cask components important to safety. The allowable stress limits are documented in the TSAR in Section 3.2.6.2, Tables 3.2-3 and 3.2-4.

3.3 Review Procedure

The TSAR was reviewed for compliance with 10 CFR Section 72.72(a) which refers to quality standards that govern the characterization of materials, the establishment of stress intensity limits, and the design and analysis methods that provide confidence in the capability of the structure, system or component to perform the required safety function. The TSAR was also reviewed for compliance with 10 CFR Section 72.72(b) which requires that protection against environmental conditions and natural phenomena be demonstrated; for compliance with 10 CFR Section 72.72(c) which requires that protection against fires and explosions be demonstrated; and for compliance with 10 CFR Section 72.72(h) which requires that protection of fuel cladding against degradation and gross rupture be demonstrated.

3.4 Findings and Conclusions

3.4.1 Loads

3.4.1.1 Normal Operating Conditions

The TSAR specifies in Section 4.8.1.3.3 the normal operating pressures of 32.4 psia hot, and 15 psia cold. In Section 4.8.1.4 the trunnion loads are based upon NUREG-0612 for a non-redundant lifting system. The normal loads are further increased by a 1.15 dynamic factor.

3.4.1.2 Loads Due to Environmental Conditions and Natural Phenomena

The design basis loads due to environmental conditions and natural phenomena are summarized in Section 3.2 of the TSAR. In accordance with Section 2.4.3 of this SER, the staff used 0.35 g horizontal acceleration plus an upward acceleration of 0.17 g to determine whether the cask would tip as a result of an earthquake. A maximum horizontal windspeed of 360 mph was adopted.

3.4.1.3 Loads Due to Postulated Accidents

10 CFR Section 72.72 (b)(1) requires that the cask be designed to accommodate the effects of postulated accidents. The TSAR describes these postulated accidents in Chapter 8. The loads due to these accidents arise as a result of impact due to handling accidents, gas cloud explosion, or fire. The handling accidents assumed in the TSAR are a 6-foot end on drop and a tip-over from the vertical standing position. The staff has performed confirmatory analyses which indicate that these accidents will not impair the integrity of the cask body. This is discussed in more detail in Section 3.4.4.1.3 of this SER. The staff therefore recommends that steps be taken to ensure that the cask not be lifted to a height greater than six feet while it is moved vertically from the reactor to the storage pad, and that the cask never be carried horizontally. The staff notes that the TSAR provides analysis for the six-foot drop with a bottom impact limiter attached. Therefore, the bottom limiter must be in place for all handling situations, and must be left in place during storage.

3.4.2 Materials

Materials used for fabrication of the NAC S/T storage cask are listed in Tables 4.2-1 to 4.2-13 of the TSAR. All materials are identified by ASME code designation which are related to ASTM Specifications. These specifications are considered by the reviewers to be quality standards in accordance with 10 CFR Section 72.82(a). However, the structural properties of the neutron shield material is not listed in this table. Since the neutron shield material provides structural support at the fins during the impact loading conditions, the structural material properties for the Bisco should appear in the TSAR. The properties of SA-276, which is used for the primary penetration cover, is also not shown in the TSAR. Since this is 304 stainless steel but in bar form, its properties are similar to SA-240 which is described. Nevertheless, for completeness, the properties of SA-276 should appear in the TSAR.

3.4.3 Stress Intensity Limits

The TSAR lists in Tables 3.2.-3 and 3.2-4 material properties and stress intensity limits for normal operating conditions, as a function of temperature, for all components important to safety. In general, the stress intensity limits are in accordance with the standards established by the ASME BPV Code. Consequently, they conform to the quality standard requirement of 10 CFR Section 72.72 (a)

3.4.4 Structural Analysis

3.4.4.1 Cask Body

3.4.4.1.1 Normal Operating Loads.

The cask body was analyzed for an internal pressure of 32.4 psia using a finite element code as described in Section 4.8.1.3 of the TSAR. The maximum stress was 5400 psi, which is far below the allowable stress intensity limit of 20 ksi for the cask body.

During truck transport, the cask rests on two trunnions (at the upper end of the cask) and a shipping skid (to support the lower end of the cask). There is no analysis provided in the TSAR for this horizontal load condition. A simple beam analysis performed by the reviewers shows that the maximum stress in the cask is well below the stress intensity limit.

During the handling by crane, the cask is supported in a vertical position by two or four trunnions. Either a non-redundant, two-arm yoke or a redundant four-arm yoke may be used to lift and handle the cask. A finite element analysis, described in Section 4.3.1.4.3 of the TSAR, shows that the highest membrane plus bending stress in the cask body is 14,700 psi, which is below the allowable stress of 30,000 psi (1.5 Sm). The combination of pressure, bolt preload, and handling stresses is below the stress intensity limit.

3.4.4.1.2 Environmental Conditions and Natural Phenomena

As a result of the design basis tornado wind loads, the staff concludes that the cask will not suffer a tip over. The TSAR states in Section 3.2.3 (and the staff concurs) that the cask may tip over as a result of the design basis earthquake. The staff concludes that the cask integrity will also be maintained for snow and ice loadings, for flooding conditions and for lightning strikes. For tornado generated missiles see Section 3.4.4.1.6 of the SER.

3.4.4.1.3 Accident Conditions

The TSAR describes analyses of the cask body for accident conditions in Section 8.2. The impact conditions considered in the TSAR are tipover, bottom end drop, and corner drop. A side drop is also discussed, but is provided in the TSAR for comparison purposes only since the analysis includes two side limiters, while the actual NAC S/T design includes only one side limiter. The tip-over analysis is discussed in Section 8.2.3 of the TSAR. A confirmatory finite element analysis was performed by the staff for the tip-over condition based on the revised upper limiter design, provided by letter No TCT/87/64/ETS dated October 19, 1987. The confirmatory analysis shows that the g-loads due

to the tip impact will be less than 20 g's, and that the stresses in the cask body are within the allowable limits.

The bottom end drop is discussed in Section 8.2.4.2.2.1 of the TSAR. A finite element confirmatory analysis was performed by the reviewers for this condition. The results of the confirmatory analysis show that the lead slumps 0.9 inches, and that the maximum stress in the cask body is 22 ksi. This is well below the allowable of 72 ksi (3.6 S_m). (It is not clear why the primary plus secondary accident allowable stress for the cask body is stated in Table 8.2-10 of the TSAR to be 1341 ksi.)

An analysis of a corner drop accident is described in the TSAR in Section 8.2.4.2.2.3. The analysis uses an axisymmetric finite element model with non-axisymmetric loading. The results given in Table 8.2-25 of the TSAR show that the stresses due to this loading condition are well below the allowable stress.

No analysis is provided in the TSAR for a bottom or tip-over condition without the impact limiters attached. Therefore, the cask must be handled and stored with both the bottom end and upper side impact limiters in place.

3.4.4.1.4 Fracture Toughness Evaluation

The austenitic stainless steel material for the cask body is fracture resistant. Consequently, brittle fracture is not a relevant failure mode.

3.4.4.1.5 Cask Thermal Stress Analysis

The thermal stress analysis for the NAC S/T cask was reviewed to ensure that the containment would not fail under the assumed loading conditions. The requirement for structural integrity can be met if, by using ASME code methods, it is demonstrated that the maximum primary plus secondary stress intensity range is less than three times the design stress intensity (3 S_m) and that the fatigue usage factor due to thermal cycling is less than one. The TSAR calculated a maximum stress intensity range of 186MPa (26,975 psi) for a cycle of hot case

(54°C = 130°F ambient temperature with insolation) to cold case (-40°C = -40°F ambient temperature without insolation).

For type 304 stainless steel, 3 Sm is larger than 414 MPa (60 ksi). The margin of safety is thus

$$M.S. = \frac{\text{Allowable}}{\text{Actual}} - 1 = \frac{414}{186} - 1 = 1.23$$

For an alternating stress intensity range of 93 MPa (13,488 psi), the allowable number of cycles is larger than 106 for fatigue (Fig. I-9.2.1. Appendix I of ASME Boiler and Pressure Vessel Code). Assuming a daily stressing for 40 years, the fatigue usage factor is

$$\frac{365 \times 40}{106} = 0.0146 \ll 1$$

Based on the review of the thermal stress analysis in the TSAR, it is concluded that the cask containment will not fail. The thermal analysis in the TSAR may be referenced in a site-specific license application provided that the site environmental conditions are within the thermal cases analyzed.

3.4.4.1.6 Tornado - Generated Missiles

Tornado generated missiles that may damage the cask are described in NUREG-0800. All missiles are assumed to impact the cask at 35 percent of the maximum windspeed, which is defined in Regulatory Guide 1.76 to be 360 miles per hour. Thus the maximum missile velocity is 126 miles per hour. The point of application and orientation of the missile is that which can cause the greatest amount of damage. NUREG-0800 also recommends that 70 percent of the postulated horizontal velocities be used, in this case 88.2 miles per hour, to assess damage caused by vertical impact of missiles, except for the small rigid missile described below. Types of missiles described include:

1. A massive high kinetic energy object that is deformable upon impact with the cask. This may be represented by an 1800 kg automobile.

2. A rigid missile that tests the penetration resistance of the cask as represented by a 125 kg, 20.32 cm (8 in) armor piercing shell.

3. A small rigid object such as a solid steel sphere 2.5 cm in diameter which may pass through any openings in the protective barriers.

In accordance with the criteria for radiological protection described in Section 2.9 of this SER, the cask must maintain its structural integrity under the impact of tornado-generated missiles.

The TSAR addresses the subject of tornado-generated missiles in Section 8.2.8.2. The analysis presented shows that the massive high kinetic energy missile will not cause cask tipover if the cask is hit on its side. The reviewers agree with this conclusion. There may be some damage to the neutron shield due to the impact of the massive missile on the cask; neutron shield damage is addressed in Section 3.4.4.3 of this SER. An analysis is also provided in the TSAR showing that there will be no permanent deformation of the lid due to a massive missile vertical impact onto the lid, although a complete loss of the upper end neutron shield would be possible.

The staff determined that the rigid 125 kg (276 lb) armor piercing shell posed the greatest damage potential to the cask. The TSAR addresses this missile in section 8.2.8.2.2. The analysis shows that the cask will not tip over due to an impact with the missile on the side, and that the cask will not be penetrated by the missile. A confirmatory analysis performed by the reviewers, based on the work of Hagg and Sankey confirms this result; however, it shows that there may be some plastic deformation of both the neutron shield and the outer cask wall during an impact with this missile.

The TSAR addresses the effect of the 2.5 cm solid steel sphere in Section 8.2.8.2.3. The analysis shows that this missile does not cause tip over and does not penetrate the cask. The analysis shows that it might cause some

plastic deformation on the primary cover, but it will not affect the secondary plate. Since there are no openings in the protective barrier represented by the cask body and lids, this small missile will not cause any other damage to the cask.

3.4.4.2 Neutron Shield

3.4.4.2.1 Normal Operating Loads

The analyses of the neutron shield shell are provided in Section 4.8.1.6 of the TSAR. The loads on the neutron shield while the cask is in a horizontal position supported by the skid cradle are considered normal operating loads. The NAC analysis of the stresses on the shield fin due to this load is an incorrect application of the formulation given by Roark such that if correctly applied, the stresses in the fin based on a simple model will exceed the yield strength of the fin material in the region of the cradle support. However, by utilizing the argument that the Bisco is an elastic foundation, the load will be transferred from the shell to the cask by the Bisco as well as the fin, which will reduce the stress in the fin to an acceptable level.

3.4.4.2.2 Environmental Loads and Natural Phenomena

While there are environmental and natural phenomena loads on the neutron shield, the staff does not expect the structural integrity of the neutron shield to be affected by these loads, with the exception of the tornado missiles. The tornado missile loads on the neutron shield are not addressed in the TSAR. There is a good chance that part or all of the neutron shield will be damaged by the a tornado missile; however, the cask integrity will not be affected, as discussed in Section 3.4.4.1.6, of this SER. The shielding analysis for this condition is discussed in Section 11.3.2.1 of this SER.

3.4.4.2.3 Accidents

The analysis of the response of the neutron shield to end drop impact loads has been reviewed to be correct for the specified loads. The neutron shield will not fail under a 55g end drop impact condition.

Most of the energy from a tip-over condition will be absorbed by the upper limiter for the cask. It is possible that a small portion of the lower neutron shield will be damaged by this condition. This will not affect the overall cask integrity. A shielding analysis for the damage neutron shield condition is discussed in Section 11.3.2.1 of this SER.

3.4.4.3 Fuel Basket

3.4.4.3.1 Normal Operating Loads

The TSAR addresses the analysis of the Fuel Basket in Sections 4.8.1.1. Since the loaded NAC S/T cask is in the vertical orientation for all handling and normal operation conditions, the basket members are loaded only by their own weight, and no detailed analysis of the normal operation load condition was performed. This is acceptable to the reviewers.

3.4.4.3.2 Environmental Loads and Natural Phenomena

The only consequence of environmental or natural phenomena on the fuel basket would result from cask tip over. The analysis for tip over is reviewed in the following section.

3.4.4.3.3 Basket Accident Loading

Accident conditions for the fuel basket are addressed in Section 8.2.4.2.3 of the TSAR. The impact conditions considered are bottom end impact, and tip over in three orientations. The stresses due to the bottom end impact are quite low, since the basket is loaded by its own weight times 37 g's only. The tip-over condition is more critical, since the basket components must also support the decelerating load of the fuel assemblies. The analysis in the TSAR for this condition is, in general, complete and coherent. Three tip orientations were considered. The worst orientation from a stress point of view is what the TSAR refers to as the 90° orientation. For this case, the lowest

margin of safety in the basket is 0.05, occurring in the steel spacer between tie bars. The allowable is taken as the yield stress at the operating temperature of the basket.

3.4.4.4 Trunnions and Trunnion Bolts

3.4.4.4.1 Normal Operating Loads

The NAC S/T has six trunnions: four trunnions are used for vertical lifting, spaced at 90° intervals near the top of the cask; two trunnions are used to rotate the cask into a horizontal position for temporary storage or transportation (the cask is not designed for horizontal storage). The rotation trunnions are designed to support a resultant load of 3.04 times the empty weight of the cask without producing stresses anywhere in the trunnion in excess of the material yield strength. Each set of two lifting trunnions is designed to support six times the fully loaded weight of the cask without producing stresses anywhere in the trunnions in excess of the yield strength, and ten times the fully loaded weight of the cask without producing stresses anywhere in the trunnions in excess of the ultimate strength.

The analysis for these conditions is contained in Section 4.8.1.4 of the TSAR. The analysis uses a cask design weight of 250,000 lbs, which is significantly larger than the actual weight of 205,800 lbs. The reviewers question the use of an allowable ultimate shear strength of 0.8 times the ultimate stress of the material. If 0.5 ultimate is used for allowable shear, and 205,000 lbs is used instead of 250,000, then the bolt shear is 62,094 lbs compared to the allowable of 67,500 lbs, see page 4.8-32 of the TSAR. In the analysis for the Lifting Trunnion Box y_c and Ix_c are incorrectly calculated (see page 4.8-37). They should be $y_c = 1.234$ and $Ix_c = 8.783$. Once again, if 205,000 lbs is used for the cask weight, then the stresses in this plate are lower than the allowable.

The analysis of the shear stress in the trunnion base does not account for the effect of bolt holes. By using a design weight of 205,000 lbs instead of 250,000 lbs, the stresses are still lower than the allowable.

The dimensions for the trunnion pick-up point are not consistent between the drawing (Figure 4.2-4) and the sketches on pages 4.8-27, 4.8-33, and 4.8-44. A method for ensuring that the distance from the pick-up points the attachment surface of the cask does not exceed that used in the analysis (3.2 inches) should be described, and implemented in the lifting procedure.

A few other inconsistencies are noted as follows: there should be a minus sign for $Ay^2 = -36.792$ on page 4.8-29. There should be a square root in the second equation for M.S. There should a be a square root in the equation for RTR. The plate thickness of 1.12 in the sketch on page 4.8-40 is inconsistent with the drawing (Fig. 4.2-4). Ay for bolt 5 should be 2.611 rather than 2.677 on page 4.8-42, and Is should be 6.963 on this same page, and on following pages. A square root is missing from the equation for b on page 4.8-45.

3.4.4.5 Upper Side Impact Limiter Attachment and Support Structure

The impact limiter attachments are discussed in Section 8.2.4.2.5 of the TSAR. A letter-transmittal dated October 7, 1987, provides revisions for some of the analysis. The 10/7/87 version of the upper side impact limiter attachment analysis has been reviewed to be correct, and appropriate for the load, which is a 1.5g vertical handling load. The welded attachment of the tabs to the inner ring is not described on the drawing with sufficient clarity (page 4.2-0) to draw unambiguous conclusions about the weld design.

The 10/7/87 version of the upper impact limiter support structure analysis has been reviewed. While the dimensions of the span used in the NAC analysis are inconsistent with the sketch, the use of the correct dimensions show that the upper impact limiter support structure will not fail under the specified loads.

3.4.4.6 Lower Impact Limiter Attachment

The impact limiter attachments are discussed in Section 8.2.4.2.5 of the TSAR. A confirmatory analysis performed by the reviewers shows that the bolts are the weakest link in the attachment system (consisting of bolts, strap, and

weld). While the bolt stresses for the lifting condition are borderline, they are acceptable.

3.4.4.7 Bolted Covers

The TSAR addresses the analysis of the bolted covers in Section 4.8.1.5, while the results of a finite element analysis of the bolts, lid and cask are given in Section 4.1.8.3.3. An independent analysis of the main closure lid system as well as the penetration cover systems was performed to confirm that the stresses in these systems do not exceed the allowable design stress of the materials used.

3.4.4.7.1 Main Closure Lid System

3.4.4.7.1.1 Bolts

The bolt analysis described in the TSAR assumes that the preload on each bolt is 120,000 pounds as specified (page 4.8-40). The primary stress in the bolt is due to the specified preload, plus a design basis accident pressure of 200 psi on the lid. A secondary stress arises from the difference between the thermal expansion of the lid thickness and the bolt from the loading temperature of 70°F and the operating temperature with a 26 kW decay heat source at 130° ambient temperature and full insolance. The sum of these primary and secondary stresses does not exceed the yield strength of the SA-564, Grade 630 bolt material. Operating the storage cask at an ambient temperature of -40°F, with an end of storage, 7.3 kW decay heat source and no insolance, will result in a partial unloading of the bolts. For a pressure load of 200 psig on the lid, the 120,000 lb preload per bolt should still preserve the integrity of the lid seals.

3.4.4.7.1.2 Main Closure Lid

The primary stresses in the lid are from the pressure load and the seal load. The secondary stresses are thermal stresses from the axial and radial temperature profiles given in Figure 4.8-20 of the TSAR. The sum of the primary

loads with a pressure of 150 psi, and an estimate of the thermal stresses is less than the design stress intensity for the SA-204 lid material. For a pressure of 200 psi from an accident, the sum of the primary and secondary stresses is less than 1.5 times the design stress intensity limit.

3.4.4.7.2 Penetrations

The independent stress analysis of the bolts used to attach the penetration covers to the main closure lid of the storage cask confirms that the stresses in the bolts from the seal loads and an accident pressure of 200 psig on the penetration cover does not exceed the yield strength of the SA-103, Grade B6 bolt material.

The primary penetration cover material is listed as SA 276 in Figure 4.2-6 of the TSAR. This material is not given in Table 4.2. The stresses in the primary penetration cover from the seal loads and an accident pressure of 200 psig are less that the design stress intensity limit for the SA 240, 304 stainless steel. The stresses in the secondary penetration cover from the seal loads and a normal operating pressure load of 32.4 psia (17.7 psig) will not exceed the design stress intensity limit of the SA-240, 304 stainless steel specified as the cover material. Using an accident pressure of 200 psig on the secondary penetration cover will result in stresses that do not exceed 1.5 times the design stress intensity limits for accident conditions.

3.4.4.8 Fuel

3.4.4.8.1 Area of Review

In this section, the integrity of the fuel rod cladding is evaluated for compliance with the requirement of 10 CFR Section 72.72(h).

The system reviewed consists of pressurized Zircaloy cylinders in an inert helium atmosphere. The thermal environment is characterized in the TSAR by an initial temperature of 360°C (680°F) which decays over time in accordance with the curve shown in Figure 4.8-27 of the TSAR. However, an independent analysis

performed by the reviewers indicates that the initial storage temperature is 380°C. The reviewers assumed that the decay curve is parallel to the one provided in Figure 4.8-27, but 20°C higher and leveling off at 150°C.

3.4.4.8.2 Acceptance Criterion

The requirements of 10 CFR Section 72.72(h) will be met if it can be demonstrated that, for the design configuration of NAC S/T cask, damage accumulation is negligible at the end of storage life.

3.4.4.8.3 Review Procedure

The integrity of the fuel rods under dry storage conditions was evaluated with reference to the damage mechanisms that are likely to be effective. There are several potential mechanisms for fuel cladding failure which include fracture as the terminal event of stable or unstable crack propagation, stress corrosion cracking induced by fission products, hydriding, stress rupture due to creep, oxidation and diffusion controlled cavity growth. Since the cask is designed to maintain an inert gas (helium) environment for the fuel rods, oxidation is precluded and need not be considered further as a potential damage mechanism. The effect of the remaining damage mechanisms were assessed based upon a review of the available data and conclusions of researchers involved in cladding integrity studies (See also Appendix A).

3.4.4.8.4 Findings and Conclusions

Three fundamental agents contribute to fuel cladding degradation under dry storage conditions: stress, temperature, and an aggressive environment. Under normal conditions the stress in the cladding is due to internal gas pressure in the fuel rod. The major component of this gas is helium which is introduced into the free fuel to moderate the effects of the external pressure while in the reactor core. In the course of time, fission products accumulate in the fuel rod cavities. Besides contributing to the internal pressure of the fuel rod, the fission products may also attack the inner surface of the cladding.

The effect of temperature manifests itself by accelerating the rate of degradation mechanisms activated by both stress and corrosion.

Stress corrosion cracking (SCC) occurs as a result of synergistic combination of a susceptible material, an aggressive environment and high stress. The corrosive environment associated with SCC of fuel rods has been attributed to fission products generated during irradiation. While the specific agent has not yet been identified, iodine, cesium, and cadmium are considered the most likely agents. SCC may also be related to pellet cladding interaction (PCI), but this has only been observed during reactor operation due, in part, to the large external pressure on the fuel rods. The only known cause of cladding failure due to SCC occurred in a reactor during a ramp-up. No other failures from this cause are known to have occurred either during pool storage or under dry storage conditions. One explanation may be the pellet temperatures during dry storage are much lower than those in a reactor. Consequently, the accumulation of fresh fission products at the cladding is slowly reduced during dry storage. Furthermore, the activation of SCC requires stress levels substantially above those that can reasonably be expected to prevail under dry storage conditions. The possibility exists, however, that cracks may be present that were initiated during reactor operation. Under these conditions, the stresses generated at the crack tips may be large enough to cause crack extension. However, should such a crack penetrate the cladding, it is likely that the internal pressure will be relieved and, as a consequence, effectively terminate the progress of the SCC damage mechanism. The staff concludes, therefore, the SCC is not a damage mechanism that can lead to gross rupture of the fuel rod cladding.

Hydrides in Zircaloy have been know to cause cracking by embrittling the cladding. Terminal solubilities of hydrogen in Zircaloy increase with temperature. If the temperature subsequently decreases, hydrides will precipitate in an orientation determined by the stress level. Normally the hydride precipitates in a circumferential direction and is not a problem even at hydrogen concentrations up to 400 ppm. At hoop stress levels of 90 to 95 MPa the hydride will precipitate in a radial direction which can encourage crack penetration. At 400°C (725°F) the hydrogen concentration could be as high as 200 ppm. Brittleness

may be induced as the fuel rods decrease in temperature during dry storage. However, the hoop stresses in the cladding are not expected to be high enough to cause a radial orientation of the hydride and consequent crack initiation. It is remotely possible that pre-existing cracks under stress can induce the diffusion of hydrogen to the crack tips where substantially higher concentrations could precipitate hydride in a manner that would encourage crack extension. However, as is the case of SCC, crack penetration would result in a loss of fuel rod internal pressure and termination of the damage mechanism. The staff concludes, therefore, the delayed hydriding is not a damage mechanism that can lead to gross rupture of the fuel rod cladding.

Creep rupture is a potential failure mode under dry storage conditions. Researchers have demonstrated that using a Larson-Miller approach, temperature limits from 380°C (716°F) to 400°C (725°F) could be tolerated for creep rupture lives well beyond that required for interim storage of spent fuel. The Larson-Miller approach, however, is somewhat empirical since it depends upon the existence of experimental data to establish the appropriate parameter. Practicality limits the duration of creep rupture tests, which are usually conducted at stress levels and temperatures far higher than those that prevail under dry storage conditions. The creep damage mechanisms in the high temperature, high stress regime are different from those that occur at lower temperatures and stresses. Consequently, predictions based on a Larson-Miller mode are clouded with sufficient uncertainty to warrant a more fundamental approach to cladding degradation under creep conditions.

The staff examined this matter to determine potential mechanisms for significant creep damage under dry storage conditions applicable to the case of the NAC S/T cask. The only creep damage mechanism (in fact the only mechanism for any of the failure modes considered above) that the staff found which represented a possible potential for cladding degradation and gross rupture was diffusion controlled cavity growth (DCCG), which is most applicable to the conditions of dry storage. Damage is manifested by the nucleation and growth of cavities at the grain boundaries which, in effect, reduces the area of material available to resist loads. The measure of damage is the fraction of the grain boundary area that undergoes decohesion. The reviewers developed a method to determine the level of damage as a function of time (See Appendix A).

The progress of damage based upon the applied methods indicated the area of decohesion after 20 years of storage would be less than 15 percent. Consequently, an initial storage temperature not exceeding 380°C (716°F) for the design basis fuel is marginally acceptable for meeting the requirements of 10 CFR 72 Section 72.72(h).

Staff evaluation of potential damage mechanisms to spent fuel assemblies stored under conditions specified in the TSAR leads to the conclusion, that for storage at an initial fuel cladding temperature of 380°C (716°F) or less in a helium atmosphere, potential for significant deterioration or degradation of the cladding during storage is small.

4.0 THERMAL EVALUATION

4.1 Normal Conditions

4.1.1 Area Of Review

The thermal analysis presented in the TSAR was reviewed to evaluate the protection provided to prevent fuel cladding degradation and gross ruptures in compliance with 10 CFR Section 72.72(h). The NAC S/T cask basically consists of a 1.5 inch of inner shell and a 2.63 inch of outer shell of austenitic stainless steel separated by 3.2 inches of lead gamma shielding.

The TSAR provided a thermal analysis for transporting and storing 15 x 15 PWR fuel. The maximum heat output of any specific assembly is 1.0 kW just prior to loading into the cask.

4.1.2 Acceptance Criteria

The requirements of 10 CFR Section 72.72(h) can be met if it is demonstrated that, for the NAC Storage/Transport cask operation, the maximum fuel cladding temperature is less than 380°C (716°F), as established in Section 3.4.4.9 of the SER.

4.1.3 Review Procedure

The thermal analysis in the TSAR was reviewed and confirmatory calculations were performed to ensure that the fuel rod cladding temperature is below 380°C (716°F). The steady-state thermal analysis in the TSAR was performed with the finite difference codes HEATING5 and SCOPE, assuming an absorptivity of 0.35 for the cask surface. Since the exact value of absorptivity is difficult to verify, the staff did an independent confirmatory transient analysis with the finite element code TOPAZ2D. In this confirmatory analysis, a power peaking factor of 1.1 was in the hottest region of the cask basket, the absorptivity of the cask surface was assumed to be unity, the insolation a sinusoidal function

of both time and space, and the Wooton-Epstein Correlation was used to calculate the maximum temperature of the fuel cladding.

4.1.4 Findings and Conclusions

The maximum cladding temperature calculated in the TSAR is 364°C (688°F) with a 54°C (130°F) ambient condition and a peak power output of 1.0 kW per assembly, whereas the confirmatory analysis predicted a maximum cladding temperature of 380.2°C (716.4°F). Since the absorptivity of the cask surface is definitely less than unity, it is concluded that the fuel cladding will remain below 380°C (716°F) during storage to prevent cladding degradation and gross rupture in compliance with 10 CFR Section 72.72(h).

The thermal analysis in the TSAR is acceptable for referencing provided that the maximum heat output of any single assembly does not exceed 1.0 kW, and the total heat content stored within the basket does not exceed 26 kW.

Operations during cask loading that occur within the reactor spent fuel pool area are briefly described in Section 5.1.1 of the TSAR, including spent fuel loading, lifting of the cask to the pool surface, primary lid closure and seal testing with drying of the cask cavity region by a vacuum system and a pressurization with helium.

Procedure for cask loading, unloading (including sampling and fuel cool-down), and decontamination, as adapted for site-specific conditions and use, will be described in detail by a license applicant.

4.2 Accident Conditions

4.2.1 Explosion

10 CFR requires an evaluation of 20 psia over-pressure for transportation casks. The TSAR calculated that an external pressure of 104 psia is required for the initiation of the yielding of the cask outside surface. It is concluded

that the NAC S/T cask is structurally adequate to withstand any credible explosive over-pressure.

4.2.2 Fire

4.2.2.1 Area Of Review

The thermal analysis for an accidental fire was reviewed in the TSAR to determine if any radioactive release could occur in violation of 10 CFR Section 72.68. The TSAR assumed that the cask is exposed to a 800°C (1472°F) engulfing fire for 1/2 hour.

4.2.2.2 Acceptance Criteria

The requirements of 10 CFR Section 72.72(c) can be met if it is demonstrated that the fuel rod cladding temperature remains below 380°C (716°F).

4.2.2.3 Review Procedure

The confirmatory analysis on the fire accident was performed with a two-dimensional finite element code, and the fuel rod cladding temperature was calculated with the Wooten-Epstein Correlation.

4.2.2.4 Finding and Conclusions

The TSAR calculated a maximum fuel temperature of 374°C (704°F) under the 10 CFR 71 fire conditions, whereas the confirmatory analysis for the fire plus insolation indicated a maximum fuel temperature of 380.9°C (717.7°F) using an absorptivity of unity. Since the absorptivity is definitely less than one, it is concluded that the maximum fuel rod temperature during and after a fire will be below 380°C and the cask design is structurally adequate to meet the requirements of 10 CFR Sections 72.72(c) and 72.72(h).

5.0 SHIELDING EVALUATION

5.1 Area of Review

Section 72.67(a) of 10 CFR Part 72 requires that during normal operations and anticipated occurrences, the annual dose equivalent to any real individual located beyond the controlled area shall not exceed 25 mrem to the whole body, 75 mrem to the thyroid, and 25 mrem to any other organ as a result of exposure to (1) planned discharges of radioactive materials, radon and its daughters excepted, to the general environment; (2) direct radiation from ISFSI operations; and (3) any other radiation from uranium fuel cycle operations within the region.

Section 72.68 (a) of 10 CFR Part 72 requires that for each ISFSI site, a controlled area shall be established. Since the TSAR is generic in nature no specific controlled area has been established. However, the TSAR has provided a dose rate calculation (see Table 7.4-1 of the TSAR), which for a distance of 260 meters (853 feet) from a single cask, yields an annual dose of less than 25 mrem to an individual assumed to be continuously present at that distance.

In addition to the above, NAC addresses the shielding design criteria in TSAR Sections 1.2.2 (Principal Design Criteria), 3.3.5.2 (Criteria), 7.1.2 (Design Considerations), and 10.1.2.1 (Fuel Characteristic Limit), and a supplemental letter dated October 21, 1987. As stated, the maximum dose rate (neutron + gamma) is 100 mrem/hr at any accessible cask surface. This limit is not the actual value expected for storage of PWR spent fuel assemblies in a NAC S/T cask. NAC calculated values are given in Table 7.3-4 (Combined Gamma and Neutron Dose Rates (26 Assembly 35,000 MWD/MTU 5-year Cooled)) of the TSAR and supplemental letters dated October 21, 1987, November 13, 1987, and December 17, 1987. Maximum values are 75.6, 170.1, and 853.0 mrem/hr at the surface of the cask top, side, and bottom, respectively. The maximum dose rate at the cask side occurs at the bottom of the neutron shield which will be accessible. With regard to the maximum dose rate at the bottom, NAC indicates that no personnel should be at the bottom surface of the cask while attaching the bottom impact limiter, and during storage, the bottom surface of the cask is not accessible.

5.2 Acceptance Criteria

Since the TSAR must be generic in its approach and cannot address site-specific conditions for a license applicant's given array configuration and size, the case of a single cask is examined to evaluate cask shielding design adequacy. Arbitrarily, we have set the minimum distance to the site boundary at 260 meters (853 feet). For the case of a single loaded NAC S/T cask and the conservative assumption that an individual is continuously present, cask shielding is acceptable if it can be shown that the annual dose to an individual at the site boundary does not exceed 25 mrem.

5.3 Shielding Review Procedure

5.3.1 Source Specification

5.3.1.1 Gamma Source

The TSAR addresses the gamma source for the active fuel length of the spent fuel elements in Sections 3.1.1 (Materials to be Stored) 7.2.1 (Characterization of Sources), and 7.3.2.1 (Analysis Source Description). Included in the description is the axial source distribution. The gamma source strength is determined from an ORIGEN2 (LOR-2 version) calculation using the Westinghouse 15 x 15 array fuel assembly described in Table 3.1-1 (Design Basis Fuel, Fuel Physical Parameters) and the irradiation conditions described in a supplemental letter dated November 19, 1987. In this calculation, the average burnup is 35,000 MWd/MTU, the specific power is 30.4 MW/MTU, the initial fuel enrichment is 3.3 percent, the irradiation time is three 383-day cycles with interim shutdown of 50 days, and 453 Kg (997 lbm) per assembly of heavy metal is considered. The cooling time for the spent fuel used in the shielding evaluation is five years. Activation of the cladding material and hardware in the active fuel region is not included in the gamma source strength.

The gamma sources for the upper end fitting and lower end fitting regions of the spent fuel elements are addressed in TSAR Sections 3.1.1 (Materials to be Stored) and 7.2.1 (Characterization of Sources) and supplemental letters dated October 21, 1987, November 13, 1987, and November 19, 1987. Gamma source strengths for the end fitting regions were determined with ORIGEN2 (LOR-2 version) assuming Inconel-718 Co content (4,694 gram/ton) for the SS-304 Co content (800 gram/ton).

5.3.1.2 Neutron Source

The TSAR addresses the neutron source for the active fuel length of the spent fuel elements in Sections 3.1.1 (Materials to be Stored), 7.2.1 (Characterization of Sources) and 7.3.2.1 (Analysis Source Description). Included in the description is the axial source distribution. The neutron source strength is determined from an ORIGEN2 (LOR-2 version) calculation using the Westinghouse 15 x 15 array fuel assembly described in Table 3.1-1 (Design Basis Fuel, Fuel Physical Parameters) and the irradiation conditions described in a supplemental letter dated November 19, 1987. The major input parameters for this calculation are described in Section 5.3.1.1 (Gamma Sources) of this SER. Sub-critical multiplication is automatically included with the neutron transport calculations.

5.3.2 Model Specification

The shielding model is addressed in TSAR Section 7.3.2.3 (Shielding Analysis Models) and supplemental letters dated October 21, 1987, November 13, 1987, and November 19, 1987. NAC assumes one and three dimensional geometries with a homogenized circularized spent fuel array. Vent, drain, lid seal test, and monitoring port penetrations into the side of the closure lid, and ducting through fins at the cask side are not specifically modeled. Neither is the cask with bottom impact limiter in place.

In our evaluation, we have assumed a slightly different source geometry. The top end fitting and active fuel regions are as modeled by NAC. The bottom end fitting region is modeled as extending for 6.86 cm (2.70 in) in height and located 2.49 cm (1.98 in) below the active fuel region. This model is a more

accurate representation of the bottom end fitting and results in an increased density within the source region.

For the normal and accident shield geometries, slightly different models are also assumed. The staff normal geometry model of the radial neutron shield is some 4.45 cm (1.75 in) in height greater in Bisco NS4FR than the NAC model. The difference is due to an NAC decision not to take credit for the material in this region because it will contain an elastic foam filler to account for expansion as the shield heats up. For the loss of neutron shield accident, the staff and NAC models at the cask top and bottom are identical. At the side, the staff has assumed the complete removal of the Bisco NS4FR and the neutron shield shell. NAC keeps the neutron shield shell intact and voids the Bisco NS4FR region. With the lead slump accidents, the staff and NAC models are the same.

For completeness, a model of the cask with the bottom impact limiter in place was also prepared.

5.3.2.1 Description of the Radial and Axial Shielding Configuration

The radial and axial shielding configurations are addressed in Section 7.3.2.2.1 (Shielding Analytical Models). Supplemental information is also provided by NAC under separate letter dated October 21, 1987, November 13, 1987, and November 19, 1987. Dose point locations are at the cask surface. Radial surface locations are: the fuel mid plane (Dose Point 1), the top and bottom of the neutron shield (Dose Points 10 and 5), the top of the lead annulus (Dose Point 12), and the top and bottom edges of the cask (Dose Points 11 and 6). Axial surface dose point locations are at the top and bottom of the cask on the center line (Dose Points 7 and 4).

Figures presented for the radial and axial shielding configurations, when updated with the supplemental information provided under separate letters dated October 21, 1987, and November 19, 1987, appear adequate for the NAC model.

5.3.2.2 Shield Regional Densities

Material densities (gm/cm^3) are addressed in Sections 4.2.1.3 (Properties of Materials), 7.3.2.1 (Analysis Source Description), and 7.3.2.3 (Shielding Analysis Models) of the TSAR. Atom number densities (atoms/barn-cm) are addressed in Section 7.3.2.1 and 7.3.2.3. Elemental and material density data are also supplemented by NAC under separate letters dated October 21, 1987, and November 19, 1987.

Active fuel region element and atom number densities for U, O, Al, and B differ between QAD-CG and XSDRNPM inputs. In addition, Zr has been omitted from the XSDRNPM source composition and stainless steel has been omitted entirely from both the QAD-CG and XSDRNPM source compositions. NAC has given no reason for the density differences. The Zr and stainless steel were conservatively neglected; a procedure the staff cannot endorse.

Shield region element and atom number densities for Fe, Cr, Ni, and B differ between QAD-CG and XSDRNPM. In addition, O has been omitted from the XSDRNPM shield composition, and C has been omitted entirely from both the QAD-CG and XSDRNPM shield compositions. The differences in Fe, Cr, and Ni are an adjustment to normalize QAD-CG to XSDRNPM. The difference for B is unexplained. Omission of the O and C is tied to a NAC decision to consider only those elements that were fixed in the original liquid neutron shield analyses. Conservatism in results is once again claimed. The staff continues to believe that conservatively neglecting materials is an unacceptable practice.

5.3.3 Shielding Evaluation

The TSAR addresses the shielding evaluation in Section 3.3.5.2 (Criteria) and Section 7.3.2.2.2 (Shielding Results). NAC shielding calculations are performed with the QAD-CG, MICROSHLD and XSDRNPM codes; flux-to-dose rate conversion factors are those of ANSI/ANS-6.1.1. Effects of the stainless steel and copper plates in the radial neutron shield are estimated. Effects of the lost neutron shield accident condition are calculated from the model. Effects of the lead slump accident conditions are estimated from half value layer considerations. Supplemental information is also provided under separate letters dated November 13, 1987, and November 19, 1987.

In our evaluation of the NAC calculation, we use the SHIELD and COG Codes. SHIELD is an unbenchmarked code for gamma dose from simple source geometries and is based upon analytical solutions to simple integration kernels. Buildup factors employed in this use of SHIELD are for a point isotropic source in iron. COG (Reference 4) uses the Monte Carlo method to transport both neutrons and gamma rays. With SHIELD, we determined the surface gamma dose at the axial centerline of the top and bottom and along the cask side using a cylindrical volume source with a slab shield at side and end geometries. With COG, we use a three dimensional finite cylinder analysis. We determined the average neutron and gamma dose at the side surface of the cask and the average gamma dose at the top and the bottom of the cask and at the side surface of the impact limiter. Areas used in these average dose rates are those associated with a radius of 20.23 cm (8 in) about the axial centerline for the top and bottom, a height of 20.32 cm (8 in) above and below the active fuel midplane for the side, and a height of 27.94 cm (11 in) for the impact limiter. Flux-to-dose rate conversion factors are also those of ANSI/ANS-6.1.1. Gamma sources used in the SHIELD code computations are the same as those used by NAC. For COG, the NAC 10 fixed neutron and 11 fixed gamma groups are used. Effects of the various penetrations and lost neutron shield and lead slump are evaluated through calculation.

5.4 Findings and Conclusions

NAC's calculated total (neutron + gamma) maximum surface dose rates for normal conditions are 75.6, 170.1, and 853.0, mrem/hr at the top, side, and bottom, respectively. Staff calculations confirm the NAC results.

For computation of the annual dose commitment, NAC and the staff have assumed the dose rate at the active fuel midplane as representative of the cask average. Annual dose commitment at 260 meters from a single cask to an individual, conservatively assumed to be continuously present, is calculated by the staff to be less than 17 mrem/yr calculated by NAC which is less than the 25 mrem/yr allowed under Section 72.67(a). For arrays involving more than one cask, a license applicant will have to assess the conditions for the site concerned and the total number of casks to arrive at a suitable distance.

6.0 CRITICALITY EVALUATION

6.1 Area of Review

In compliance with 10 CFR Section 72.73, the criticality analysis presented in the TSAR was reviewed to determine if the NAC S/T cask is designed to be subcritical and to prevent a nuclear criticality accident.

The NAC S/T cask with its fuel basket is described in Section 1.2 of the TSAR. The cask is a right circular cylinder of multi-wall construction with a 1.5-inch thick inner shell and a 2.63-inch thick outer shell of austenitic stainless steel separated by 3.2 inches of lead gamma shielding. The upper end of the cask is sealed by an austenitic stainless steel bolted closure lid which is 8.5 inches thick and contains a 1-inch thick steel inner closure plate, 2 inches of lead gamma shielding, and a 5.5-inch thick steel outer closure plate. The lower end of the cask is welded to the sides of the cask and is 8.8 inches thick and contains a 1-inch thick lower closure plate and a 6 inches thick upper closure plate of austenitic stainless steel separated by 1.8 inches of lead gamma shielding. Bisco NS4FR neutron shield material that is 7 inches thick surrounds cask outer cylinder for 153 inches, covering the active fuel region axially. This Bisco is held in place by 0.25-inch thick austenitic stainless steel plates that are welded to the cask wall. Bisco NS4FR neutron shield material that is 3 inches thick and 86.75 inches in diameter is used on the top of the lid. This Bisco is held in place by 1.25-inch thick austenitic stainless steel plates that are welded to the lid.

The spent fuel assemblies are supported by a basket of proprietary design. This design provides for support of 26 intact spent fuel bundles and the support of Boral^{TM} neutron absorbing material near the fuel along with flux traps for criticality control during underwater fuel loading and unloading.

The criticality analysis presented in the TSAR was performed with the AMPX cross-section processing codes NITAWL and XSDRNPM and the Monte Carlo criticality code KENO-IV code combined with a 123 group cross-section set based on Gam and Thermos data. These codes and cross-sections are available from the Reactor Shielding Information Center, Oak Ridge, Tennessee.

The criticality analysis in the TSAR was based on the following assumptions: (1) the fuel was enriched to 3.3 w/o ^{235}U in uranium; (2) the fuel was unirradiated; (3) the boron content in the Boral was 0.03 grams $10B/cm^2$; (4) the fuel, clad, fuel-clad gap and moderator inside each assembly formed a homogeneous mixture at 20°C (68°F); and (5) a two-dimensional model was used such that the cask was assumed infinite along the vertical axis (the lid and cask bottom were not modeled).

The criticality analysis in the TSAR considered a cask content model in which all fuel bundle types were centered within each storage location (nominal model).

The calculation method and cross-section values which were used in the criticality analysis in the TSAR were verified by comparison with critical experiment data for assemblies similar to those for which the cask was designed. Seventy critical experiments were analyzed. These experiments considered water moderated, oxide fuel arrays separated by various materials (Boral, steel, lead, and water, for example) that simulate Light Water Reactor (LWR) fuel storage conditions. K-effective results were calculated for the models of each of these critical experiments. From these k-effective results, the bias of the computational tools are determined. See Section 3.3.4.3 of the TSAR.

Because a homogenized fuel-clad-water cell was used in the TSAR criticality analysis, a discrete fuel pin model and a homogenized fuel model were prepared as input to KENO-IV for one cell, and the k-effective values were calculated and compared. Each cell model included the portion of the basket applicable to that cell. The results of this comparison showed the homogenized model to give the higher k-effective value by 0.39 percent in mean value with a standard deviation of 0.32 percent. Hence, a bias correction was not necessary.

6.2 Acceptance Criteria

The requirement of 10 CFR Section 72.73 can be met if it is demonstrated that, for the NAC S/T cask design, the effective multiplication factor is less than 0.95 ($k_{eff} < 0.95$) for all credible configurations and environments.

6.3 Review Procedure

The criticality analysis in the TSAR was reviewed and verification calculations were performed for comparison to ensure that the NAC S/T design is subcritical at all times.

The criticality review was performed with the KENO-Va code combined with a 123 group cross-section set described in CCC-475, RSIC (Reference 5). Criticality calculations using these computational tools were performed on a IBM 3033 mainframe computer at the Oak Ridge National Laboratory (ORNL).

The criticality review was based on the following assumptions: (1) the fuel was enriched to 3.3 w/o ^{235}U in uranium; (2) the fuel was unirradiated; (3) the boron content in the Boral was 0.03 grams $10B/cm^2$; and (4) the fuel, clad, fuel-clad gap and moderator inside each assembly were modeled discretely and at 20°C (68°F). The elements of the proprietary basket design were modeled discreetly. The cask and its contents were modeled in three dimensions.

The criticality review considered one PWR fuel bundle design, the Westinghouse W-STD 15 x 15. This was also the only design analyzed in the TSAR.

The criticality review considered cask content models in which all fuel bundle types were centered within each storage location (nominal model), a model in which the fuel bundles were clumped together toward the center of the cask (fuel-to-center model), and a model in which the fuel bundles were located as close to the lead as possible (fuel-to-lead model).

The calculation method and cross-section values used in the criticality review were verified by comparison with critical experiment data for assemblies similar to those for which the cask was designed. Four critical experiments were analyzed. These experiments included water moderated, oxide fuel arrays separated by Boral plates on two sides of a linear three fuel bundle array for two different UO_2 enrichments (references 6 and 7) at near optimum water moderation, and water moderated, oxide fuel arrays separated by Boral plates on two to four sides of a 3 x 3 fuel bundle array at undermoderated water conditions

(reference 8) close to those found under storage conditions. From these k-effective results, the bias of the computational tools used in the criticality review were determined.

6.4 Findings and Conclusions

The largest k-effective value reported in the TSAR is 0.948 for the nominal model with W-STD 15 x 15 fuel. This is an upper limit value at 95 percent confidence. This stems from a calculated k-effective value of 0.93837 + 0.00185 for 145,642 neutron histories. The correction to 0.948 is due to the application of the bias for the computational tools and the experiment. A 1.645 multiplier on the one-sigma values and root-mean-square averaging of the one-sigma values from calculated results were used, as described in the TSAR.

The largest k-effective value found in the confirmatory analysis was 0.950 for the nominal model with W-STD 15 x 15 fuel. This is an upper limit value at 95 percent confidence with biases applied. This stems from a calculated k-effective value of 0.94097 + 0.00421 for 30,000 neutron histories. The correction to 0.950 is due to the application of the bias for the computational tools. A 1.645 multiplier on the one-sigma values and root-mean-square averaging of the one-sigma values from calculated results were used.

The fuel-to-center model gave an upper limit k-effective result of 0.946 at 95 percent confidence with biases applied. This stems from a calculated k-effective value of 0.92975 + 0.00404 for 30,000 neutron histories.

The fuel-to-lead model gave an upper limit k-effective result of 0.949 at 95 percent confidence with biases applied. This stems from a calculated k-effective value of 0.93255 + 0.00440 for 30,000 neutron histories.

The confirmatory analysis also included calculations of the pure water moderator that was less than full density inside the cask and found the k-effective results to be less than those for a density of 1 gm/cc. Less than full density moderator conditions may occur during cask evacuation following cask loading of fuel and during fuel cool-down operations in preparation for cask unloading.

The confirmatory analysis models of the NAC S/T cask bound the actual fuel and basket configurations and materials. The calculated results for these models show the peak k-effective to be equal to the maximum acceptable design limit of 0.95. The TSAR calculated results show the peak k-effective to be less than the maximum acceptable design limit of 0.95. On the basis of the TSAR evaluation and the confirmatory analysis, the staff concludes that the NAC S/T cask is designed to be maintained subcritical and to prevent a nuclear criticality accident in compliance with 10 CFR Section 72.73 with Westinghouse standard 15 x 15 fuel enriched to 3.3 w/o ^{235}U in uranium in the UO_2.

7.0 CONFINEMENT

7.1 Area of Review

The confinement analysis presented in the TSAR was evaluated to ensure that the annual doses specified in 10 CFR Section 72.67 (a) are not exceeded during normal operations and anticipated occurrences. The NAC S/T cask with its fuel basket is described in Section 1.2 of the TSAR. The cask is a right circular cylinder of multi-wall construction with a 1.5-inch thick inner shell and a 2.63-inch thick outer shell of austenitic stainless steel separated by 3.2 inches of lead gamma shielding. The upper end of the cask is sealed by an austenitic stainless steel bolted closure lid which is 8.5 inches thick and contains a 1-inch thick steel inner closure plate, 2 inches of lead gamma shielding, and a 5.5-inch thick steel outer closure plate. Filling and flushing connections are in the lid. A pressure gauge is installed on the lid. The closure and all the openings in the lid are sealed with flexible metal seals. The space between the two flexible metal seals in the lid sealing the lid to the cask body is used as a gas barrier and pressure monitoring space.

7.2 Acceptance Criteria

The requirements of 10 CFR Section 72.67 (a) can be met if it is demonstrated that the annual doses are within regulatory limits.

7.3 Review Procedure

The confinement analysis in the TSAR was reviewed and confirmatory calculations were performed to ensure that the regulatory dose limits are not exceeded. The cask is loaded with spent fuel in the storage pool. The cask is then removed from the pool, drained and helium dried. The cask is filled with 1.0 atm of helium and leak-checked to 10^{-6} atm-cc/sec at the primary seal. Because of decay heat from the fuel, the pressure in the cask can increase to 2.23 atm under normal storage conditions. The TSAR analysis assumed that fuel clad failures are 1 percent for normal conditions and 100 percent under

accident conditions. Under accident conditions the cask pressure can increase to 4.22 atm because of the release of helium and radioactive gases and vapors from the assumed failed fuel rods. The radioactive gases considered for release are tritium and krypton with the release fractions into the cask cavity obtained from Regulatory Guide 1.25.

An independent analysis was performed to confirm the pressure in the cask for normal and accident conditions. The leakage rates for normal conditions were then determined using the 10^{-6} atm-cc/sec leakage rate. For the accident conditions, we have assumed the instantaneous release of ^{3}H and ^{85}Kr.

7.4 Findings and Conclusions

Assuming the 4.x 10^{-8} std-cc/sec leakage rate at 1 atm and 1 percent fuel rod cladding failure, the expected ^{3}H and ^{85}Kr releases are 1.64 and 91.9 microcuries/year, respectively. The dose consequences of these activities will be less than 10^{-6} mrem/yr at a site boundary of 260 meters. For the accident conditions of 100 percent fuel rod cladding failure and instantaneous release, the expected ^{3}H and ^{85}Kr releases are 458 and 25,657 Ci, respectively. The dose consequences of these activities are discussed in the following section.

7.5 Confinement Requirements for the Hypothetical Accident Conditions

7.5.1 Area of Review

Section 72.15(a)(13) of 10 CFR Part 72 requires, in part, analysis of the potential dose or dose commitment to an individual outside the controlled area from accidents or natural phenomena events that result in the release of radioactive material to the environmental or direct radiation from the ISFSI.

Section 72.68(b) of 10 CFR Part 72 requires that any individual located on or near the nearest boundary of the controlled area shall not receive a dose greater than 5 rem to the whole body or any organ from any design basis accident. The minimum distance chosen is 100 meters to conform with the minimum allowable controlled area boundary distance required in Sections 72.68(b).

7.5.2 Acceptance Criteria

Cask confinement of radioactive material is deemed acceptable if it can be shown that the release of material subsequent to an accident shall not deliver to any individual a dose of 5 rem outside the controlled area.

7.5.3 Review Procedure

The review consists of consideration of: (1) the maximum gaseous activity within the cask, (2) the maximum dose from gaseous activity release.

7.5.3.1 Maximum Gaseous Activity Within the Cask

The TSAR addresses the maximum ^{3}H and ^{85}Kr gaseous activities expected to be found within the cask in Sections 3.1.1 (Materials to be Stored), 7.2.1 (Characterization of Sources), 7.2.2 (Airborne Radioactive Material Sources), and 8.2.6.2.2 (Boundary Dose). Other gaseous sources such as ^{129}I and ^{131}Xe are identified by NAC but not quantified. Volatile isotopes with limited availability such as ^{134}Cs and ^{137}Cs are also identified and quantified. Cladding tube failures of 100 percent are assumed.

7.5.3.2 Maximum Dose From Gaseous Activity Release

The maximum dose expected from gaseous activity release is addressed in Section 8.2.6.2.2 (Boundary Dose) of the TSAR. NAC assumes the available gaseous inventories of ^{3}H and ^{85}Kr are released to the environment. The site boundary is set at 260 meters.

The maximum dose to an individual at the minimum site boundary (100 meters) following an accident in which the available gaseous inventories of ^{3}H and ^{85}Kr are released has been calculated by the staff. In computing the doses due to gaseous activity release, the staff has assumed the following: (1) 100 percent cladding tube failure; (2) the release fractions of Regulatory Guide 1.25; (3) the population weighted inhalation rate of Regulatory Guide 1.109; (4) the inhalation dose and whole body dose factors of Regulation Guide 1.109; and (5) F-stability atmospheric diffusion with a windspeed of 1 meter/sec with plume meander.

7.5.4 Findings and Conclusions

The dose consequence due to gaseous activity release from a single cask following an accident in which the available gaseous inventories of ^{3}H and ^{85}Kr are released is less than 0.41 rem to the whole body at a site boundary of 100 meters. Accident consequences are less than the 5 rem established in 10 CFR Section 72.68(b).

Compliance with 10 CFR Section 72.68(b) is site dependent and depends on the number of casks being stored. Thus, a license applicant must assess conditions for the cask array proposed for his site.

8.0 OPERATING PROCEDURES

The loading procedures for the NAC S/T cask are described in Sections 5.1.1.2 (Fuel Loading) and 7.1 (ALARA) in the TSAR. This SER review is limited to the procedures as presented by NAC in this TSAR. The staff does not, at this time, make any prior judgement on the operating procedures that must be included as part of the license application for an ISFSI storage facility using the NAC S/T cask.

8.1 Area of Review

10 CFR Part 72 states the requirements that must be met by the ISFSI licensee during operations. Section 72.15 covers the technical information required in the license application for an ISFSI. Operational requirements are in Section 72.15(a)(4)(i), 72.15(a)(5), 72.15(a)(8) and 72.15(a)(14). Section 72.75(a) states the requirements for spent fuel storage and handling systems.

10 CFR Part 20 covers the standards for protection against radiation that must be met during the operation of a ISFSI.

Regulatory Guides 8.8 and 8.10 provide guidance to ensure that occupational radiation exposures will be "As Low As Is Reasonably Achievable" (ALARA).

The NAC S/T Cask TSAR addresses the cask receipt, loading, and some on-site transportation procedures at the ISFSI. The procedures for unloading and are not covered in the TSAR. The review covers the inspections, tests and special preparations of the cask for loading spent fuel. Section 5.1.1.2 of the TSAR addresses the loading procedures while Section 7.1 of the TSAR addresses the issue of ensuring that the occupational radiation doses are ALARA.

8.2 Acceptance Criteria

The operating procedures for loading the NAC S/T cask are deemed acceptable for use in the ISFSI license application if it can be shown that the considerations of 10 CFR 72, 10 CFR 20 and ALARA are in compliance with those regulations.

8.3 Review Procedure

TSAR Sections 5.1.1.1 (Initial Receipt) and 5.1.1.2 (Fuel Loading) describe the operational procedures involved with the receipt and loading of the NAC S/T cask at the ISFSI. These two sections describe how the NAC S/T cask is to be handled during the loading operation. Inspections and tests are described as part of the preparation for loading.

TSAR section 7.1 describes the general procedures to be followed to meet the requirements of 10 CFR 20, Regulatory Guide 8.8, and Regulatory Guide 8.10. These general procedures for radiation protection and meeting ALARA limits for occupational exposure apply to the cask loading procedure.

8.4 Findings and Conclusions

The operational procedures for loading the NAC S/T cask at the ISFSI are in compliance with the appropriate guidance and/or regulations. These procedures must be incorporated into the operational procedures for the ISFSI. The procedures for on-site transportation not covered in the TSAR and unloading, must be added to the operational procedures for the ISFSI.

9.0 ACCEPTANCE TESTS AND MAINTENANCE PROGRAM

9.1 Acceptance Tests

The NAC S/T TSAR addresses the subject of acceptance tests in Sections 3.3.2 and 9.2. These two sections refer to acceptance tests for the confinement system, criticality prevention and neutron shielding only. As noted in Section 1.1 (Introduction) of this SER, the NAC S/T TSAR was generated in the format of Regulatory Guide 3.48. In Regulatory Guide 3.48, Chapter 9 "Conduct of Operations," covers such tests under Section 9.2 "Pre-operational Testing and Operation." With the exception of the limited sections cited above, the TSAR treats this as a site-specific matter. This is acceptable to the staff for this TSAR. We note, however, that test procedures are required under 10 CFR 50, Appendix B. A complete set of inspection and test procedures will be required in the license application for the ISFSI.

9.2 Maintenance Program

Maintenance is addressed only briefly in the NAC S/T TSAR. In Section 5.1.3.5 (Maintenance Techniques), NAC states "The NAC S/T cask does not require any maintenance during normal operation conditions." This treatment of maintenance is acceptable for the TSAR. However, for a licence applicant proposing to use an array of casks at an ISFSI, a detailed description of site-specific maintenance activities and procedures will be required.

10.0 RADIATION PROTECTION

10.1 Area of Review

10 CFR Section 72.15(a)(5) requires the licensee to provide the means for controlling and limiting occupational radiation exposures within the limits given in 10 CFR Part 20 and for meeting the objective of exposures as low as is reasonably achievable.

10 CFR Section 72.74(a) states, in part, that radiation protection systems shall be provided for all areas and operations where on-site personnel may be exposed to radiation or airborne radioactive materials.

Guidance is also provided in Regulatory Guide 8.8, "Information Relevant To Ensuring That Occupational Radiation Exposures At Nuclear Power Stations Will Be As Low As Is Reasonably Achievable," and Regulatory Guide 8.10, "Operating Philosophy for Maintaining Occupations Exposures as Low as is Reasonably Achievable."

Our review focuses on those policy, design, and operational considerations associated with occupational exposures as low as is reasonably achievable that are not site specific. In this regard our review is limited. A second area of our review focuses on the estimated on-site dose from direct radiation and gaseous activity release during normal operations.

10.2 Acceptance Criteria

Radiation protection is deemed acceptable if it can be shown that the non-site-specific considerations for occupational radiation exposures as low as is reasonably achievable are in compliance with appropriate guidance and/or regulations, and that the dose from the transporting, storage, and repair of casks are not in excess of Part 20 limits.

10.3 Review Procedure

The review is divided into three main parts: (1) ensuring that occupational radiation exposures are as low as is reasonably achievable, (2) radiation protection design features, and (3) estimated on-site dose assessment.

10.3.1 Ensuring that Occupational Radiation Exposures are as Low as is Reasonably Achievable (ALARA)

Non-site-specific policy, design, and operational considerations are addressed in Sections 7.1.1 (Policy Considerations), 7.1.2 (Design Considerations), and 7.1.3 (Operational Considerations), respectively, in the TSAR.

The TSAR sections cited above describe how the NAC S/T cask is designed to meet the ALARA requirements. These requirements are met through the massive shielding, the passive nature of the system, the ruggedness of design, and the double confinement system utilized.

The objectives of Regulatory Guide 8.8 with regard to access control, shielding, decontamination, and monitoring are also met by the design features.

The staff evaluated the non-site-specific information provided by NAC in comparison with the guidance and/or regulations cited in Section 10.1 of this SER.

10.3.2 Radiation Protection Design Features

Installation design features are addressed in Sections 1.1.2 (General Description of the Installation), 1.2 (General Description of the Installation), 1.3 (General Systems Description), 3.1.1 (Materials to be Stored), 3.3.2 (Protection by Multiple Confinement Barriers and Systems), 3.3.3 (Protection by Equipment and Instrumentation Selection), 3.3.5 (Radiological Protection), 3.5 (Decommissioning Considerations), 4.2.3 (Individual Unit Description), 5.1.1 (Narrative Description), 5.1.2 (Flowsheets), 5.1.3.5 (Maintenance Techniques), 5.2.1 (Component/Equipment Spares), 5.4 (Operation Support Systems), 6.0 (Waste Confinement and Management), 7.1.2 (Design Considerations),

7.2 (Radiation Sources), 7.3 (Radiation Protection Design Features), and 8.1 (Off-Normal Operations). Supplemental information was also provided by NAC under separate letters dated October 21, 1987, November 13, 1987, and November 19, 1987.

TSAR Sections 1.1.2 (General Description of the Installation), 1.2 (General Description of the Installation), and 1.3 (General Systems Description) provide a physical description of the design of the cask. Included in this description are the features pertaining to shielding, the gas containment system, and other features pertaining to radiation protection.

Sections 3.1.1 (Materials to be Stored), 3.3.2 (Protection by Multiple Confinement Barriers and Systems), 3.3.3 (Protection by Equipment and Instrumentation Selection), 3.3.5 (Radiological Protection), and 3.5 (Decommission Considerations) provide information basic to the principal design of the cask. Included are descriptions of the spent fuel characteristics and major source terms; the confinement barriers and sealing procedures; the lid tightness monitoring system; the airborne and direct dose consequences from a generic single cask and 140 cask array; and neutron activation of the cask materials over the storage period.

Section 4.2.3 (Individual Unit Description) contains a description of the cask containment and radiation protection components.

Sections 5.1.1 (Narrative Description), 5.1.2 (Flowsheets), 5.1.3.5 (Maintenance Techniques), and 5.2.1 (Component/Equipment Spares) describe the handling operations in the cask loading and cask storage areas. Included in this description are the estimated number of personnel and their associated exposure periods and locations. Section 5.4 (Operation Support Systems) describes various means for monitoring the cask containment status.

Section 6.0 (Waste Confinement and Management) describes the dose consequences associated with gaseous activity release during normal operations.

Sections 7.1.2 (Design Considerations), 7.2 (Radiation Sources), and 7.3 (Radiation Protection Design Features) provide information basic to radiation protection and shielding. Included are discussions of design considerations, the source terms for the fuel assemblies and airborne radioactive material, and the shielding design features and analyses.

Section 8.1 (Off-Normal Operations) describes the off-normal structural consequences of gaseous activity release into the cask cavity and the dose consequences of gaseous activity release to the environment.

10.3.3 Estimated On-Site Dose Assessment

Information important to the estimate of the on-site collective dose is found in Sections 3.1.1 (Materials to be Stored), 3.3.5 (Radiological Protection), 5.1.2 (Flowsheets), 5.1.3.5 (Maintenance Techniques), 6.0 (Waste Confinement and Management), 7.2 (Radiation Sources), 7.3 (Radiation Protection Design Features), and 7.4 (Estimated On-site Collective Dose Assessment). Supplemental information was also provided by NAC under separate letter dated October 21, 1987.

With the exception of Section 7.4 (Estimated On-site Collective Dose Assessment), the general contents of the above pertinent sections are described in the previous section of this SER. The description provided in Section 7.4 includes an estimate of the direct radiation collective dose associated with various loading, transporting, storage, and repair operations of a single cask; and the airborne and direct collective dose associated with a generic single cask and 4, 10, and 140 cask arrays.

The dose from a single cask to any individual from direct radiation and gaseous activity release during transporting, storage, and repair operations was computed by the staff. Specific operations considered are those grouped under preparation and transfer to ISFSI storage, storage, auxiliary shield, and repair/replace shield. One transport to storage and one return from storage for shield repair/replacement is assumed during the year.

Dose from direct radiation is computed for each operation via two methods with the more conservative result being used. In method one, dose from direct radiation is based on the NAC individual dose rates in Table 7.4-5 (Occupational Doses) of the TSAR. In method two, dose from direct radiation is based on the NAC active fuel midplane side surface dose rates of Table 7.3-4 (Combined Gamma and Neutron Dose Rates (26 Assembly 35,000 MWD/MTU 5-year Cooled)) and the staff predicted dose versus distance effects. For conservatism, beginning of life is assumed for all operations.

Doses form gaseous activity release are based on the release of ^{3}H and ^{85}Kr and are computed under the following assumptions: (1) 1 percent cladding tube failure; (2) the release fractions of Regulatory Guide 1.25; (3) the NAC leakage rate of 4.2×10^{-8} std cc/sec; (4) the occupational inhalation rate of Regulatory Guide 1.25; (5) the exposure times and distances in Table 5.1-4 (Estimated Operation Times and Personnel) of the TSAR; (6) the inhalation dose and whole body dose factors of Regulatory Guide 1.109; and (7) a close-in box model for atmospheric diffusion with a wind speed of 1 meter/sec.

At 100 meters from a single cask, the dose to any individual from direct radiation and gaseous activity release during normal operations (40 hours/wk and 50 weeks/yr period of exposure) was also evaluated.

Dose from direct radiation is based on the NAC active fuel midplane side surface dose rates of Table 7.3-4 (Combined Gamma and Neutron Dose Rates (26 Assembly 35,000 MWD/MTU 5-year Cooled) and the staff predicted dose versus distance effects.

Doses from gaseous activity release are based on the release are based on the release of ^{3}H and ^{85}Kr and are computed under the following assumptions: (1) 1 percent cladding tube failure; (2) the release fractions of Regulatory Guide 1.25; (3) the NAC leadage rate of 4.2×10^{-8} std cc/sec; (4) the occupational inhalation rate of Regulatory Guide 1.25; (5) an exposure time of 40 hours/wk and 50 weeks/yr; (6) the inhalation dose and whole body dose factors of Regulatory Guide 1.109; and (7) F-stability atmospheric diffusion with a wind speed of 1 meter/sec with plume meander.

10.4 Findings and Conclusions

Non-site-specific policy, design, and operational considerations are in compliance with appropriate guidance and/or regulations, and the dose from a single cask to any individual from direct radiation and gaseous activity release during rormal operations is estimated to be less than 408 mrem/yr to the whole body from the transport, storage, and repair operations. At 100 meters from a single cask, the dose to any individual from direct radiation and gaseous activity release for 40 hours/wk and 50 weeks/yr is less than 58 mrem/yr.

Radiation protection is acceptable.

11.0 ACCIDENT ANALYSIS

11.1 Area of Review

10 CFR Section 72.15(a)(13) requires, in part, an analysis of the potential dose or dose commitment to an individual outside the controlled area from accidents or natural phenomena events that result in the release of radioactive material to the environment or direct radiation from the ISFSI.

10 CFR Section 72.67(a) requires that during normal operations and anticipated occurrences the annual dose equivalent to any real individual who is located beyond the controlled area shall not exceed 25 mrem to the whole body, 75 mrem to the thyroid, and 25 mrem to any other organ as a result of exposure to (1) planned discharges of radioactive materials, radon and its daughters excepted, to the general environment, (2) direct radiation from ISFSI operations and (3) any other radiation from uranium fuel cycle operations within the region.

10 CFR Section 72.68(b) requires that any individual located on or near the nearest boundary of the controlled area shall not receive a dose greater than 5 rem to the whole body or any organ from any design basis accident. The minimum distance chosen is 100 meters to conform with the minimum allowable controlled area boundary distance required in Section 72.68(b).

Our review focuses on the dose from direct radiation and activity release associated with postulated off-normal and accident events. In the context of this review, off-normal events are anticipated occurrences. As such, the minimum distance chosen is 260 meters to conform with the minimum distance assumed in the shielding evaluation (see Sections 5.2 and 5.4 of this SER).

11.2 Acceptance Criteria

Cask safety in the event of postulated off-normal and accident events is deemed acceptable if it can be shown that the dose from a single cask to any individual from direct radiation and activity release is not in excess of the applicable values given in Section 11.1 above.

11.3 Review Procedure

The review is divided into two main parts: (1) off-normal operations and (2) accident events.

11.3.1 Off-Normal Operations

11.3.1.1 Event

Three events are identified for off-normal operations: (1) leakage through a containment closure, (2) internal pressure due to leakage of fission product gases from the stored fuel rods, and (3) failure of instrumentation. Causes of the events are addressed in Sections 8.1.1.1 (Postulated Cause of the Event - Leakage through a Cask Closure), 8.1.2.1 (Postulate Cause of the Event - Fission Product Gas Release), and 8.1.3.1 (Postulated Cause of the Event - Failure of Instrumentation). The means of detecting the events are discussed in Sections 8.1.1.2 (Detection of Event - Leakage through a Cask Closure), 8.1.2.2 (Detection of Event - Fission Product Gas Release), and 8.1.3.2. (Detection of Event - Failure of Instrumentation). Analysis of the effects and consequences, and the proposed corrective actions in the case of these three events appear in Sections 8.1.1.3 (Analysis of Effects and Consequences - Leakage through a Cask Closure), 8.1.2.3 (Analysis of Effects and Consequences - Fission Product Gas Release), 8.1.3.3 (Analysis of Effects and Consequences - Failure of Instrumentation), and Sections 8.1.1.4 (Corrective Actions- Leakage through a Cask Closure), 8.1.2.4 (Corrective Actions - Fission Product Gas Release), and 8.1.3.4 (Corrective Actions - Failure of Instrumentation), respectively.

11.3.1.2 Radiological Impact from Off-Normal Operations

Sections 8.1.1.3.2 and 8.1.2.3.1.2 (Boundary Dose) of the TSAR present summaries of the estimated collective doses at 260 meters due to gaseous activity release following the off-normal events of leakage through a containment closure and internal pressure due to leakage of fission product gases from the stored fuel rods, respectively. Section 7.4.2 (Analysis Results) of the

TSAR describes the direct radiation collective dose versus distance for a single cask and the minimum site boundary distances for generic 4, 10, and 140 cask arrays. Supplemental information was also provided by NAC under separate letter dated October 21, 1987.

The radiological impact from off-normal operations only involves gaseous activity release and is computed for an individual outside the 260 meter controlled area. In computing the dose due to ^{3}H and ^{85}Kr gaseous activity release, the staff has assumed the following: (1) the worst case condition of 10 percent cladding tube failure; (2) the release fractions of Regulatory Guide 1.25; (3) a leakage rate of 1.4×10^{-7} std cc/sec from Section 8.1.1.3.1 (Calculated Leakage Rate); (4) the population weighted inhalation rate of Regu,atory Guide 1.109 for the off-site individual; (5) an exposure period of 1 year; (6) the innalation dose and whole body dose factors of Regulatory Guide 1.109; and (7) F-stability atmospheric diffusion with a windspeed of 1 meter/sec with plume meander.

11.3.2 Accidents

11.3.2.1 Accidents Analyzed

TSAR Sections 8.2.1, (Accident-Loss of Neutron Shield) and 8.2.6 (Accident-Cask Seal Leakage) address the worst case situations of the complete loss of the neutron shield from all external surfaces, and the release of the available gaseous inventories of ^{3}H and ^{85}Kr from the cask cavity, respectively. Other gases such as ^{129}I and ^{131}Xe and volatile isotopes with limited availability such as ^{134}Cs, and ^{137}Cs are not evaluated in Section 8.2.6. Sections 8.2.3 (Accident - Cask Tipover) and 8.2.4 (Accident - Cask Drop) describe the axial and circumferential lead slump that may occur from tipover and end drop accidents, respectively. All other accidents described in the TSAR may have one or more of the above as their worst case consequence.

In the staff review of the radiological impact of the instantaneous release of the gaseous contents, we followed the same procedures and made the

same assumptions as those discussed in our review of the maximum dose from gaseous activity release (see Section 7.5.3.2 of this SER).

For the dose consequence for direct radiation, we have assumed the worst case situation for each radiation type. Neutron dose is derived from the loss of neutron shield accident and is computed from the NAC active fuel midplane side surface neutron dose rate of Table 7.3-4 (Combined Gamma and Neutron Dose Rates (26 Assembly 35,000 MWD/MTU 5-year Cooled)) and the staff predicted dose versus distance effects. Gamma dose is derived from the tip-over lead slump accident and is computed from the NAC maximum side surface gamma dose rate in Section 8.2.3.2 (Accident Analysis-Cask Tipover) of the TSAR and the staff predicted dose versus distance curves. A worst case condition of no corrective actions is assumed for the period of one year.

11.4 Findings and Conclusions

The dose consequences due to gaseous activity release from a single cask following off-normal and accident events are less than 7×10^{-7} mrem and 0.41 rem, respectively. The dose consequence due to direct radiation from a single cask following the accidental loss of neutron shield and axial lead slump is less than 3.22 rem/yr.

For the accident events, the total dose from gaseous activity release and direct radiation is less than 3.63 rem. Accident consequences are less than the 5 rem limit established in 10 CFR Section 72.67(b).

Compliance with 10 CFR Section 72.68(b) is site dependent and depends on the number of casks being stored. Thus, a license applicant must assess conditions for the cask array proposed for his site.

12.0 DECOMMISSIONING

12.1 Area of Review

10 CFR Section 72.18 provides requirements for a site-specific decommissioning plan, including financing. Among the items to be addressed is the disposal of residual radioactive materials after all spent fuel has been removed.

10 CFR Section 72.76 provides requirements for decommissioning and states, in part, that the ISFSI shall be designed for decommissioning. Among the items to be addressed under this part are the provisions to facilitate decontamination of equipment, the provisions to minimize the quantity of radioactive wastes and contaminated equipment, and the provisions to facilitate the removal of radioactive wastes and the materials at the time of permanent decommissioning.

49 CFR Sections 173.421, 173.423, and 173.435 provide information on the radionuclide activities that may be transported as limited quantity materials.

10 CFR Sections 30.14 and 30.70 address radionuclide concentrations that are exempt from licensing requirements.

10 CFR Sections 30.18 and 30.71 address radionuclide quantities that are exempt from licensing requirements.

10 CFR Sections 61.55 and 61.56 address radionuclide concentrations for Class A wastes and the characteristics of such waste.

A decommissioning plan for a site-specific ISFSI as required by 10 CFR Part 72.18 is not applicable for a topical report. Therefore our review focuses on the non-site-specific elements of decommissioning and in particular the decommissioning of a single cask.

12.2 Acceptance Criteria

Cask decommissioning is deemed acceptable if it can be shown that regulations cited in Section 12.1 above have been followed as appropriate, and where limits can be applied, these have not been exceeded.

12.3 Review Procedure

The review is divided into two main parts: (1) unloading of the casks; and (2) decommissioning of the cask components.

12.3.1 Unloading of the Cask

A brief description of cask unloading is presented in Section 3.5.1 (Storage Casks) of the TSAR. Both wet unloading (reactor pool) and dry unloading (hot cell) are mentioned as alternatives for cask unloading. Prior to either unloading process, an off-gas system intake will be connected to the cask drain valve and a helium supply line to the cask vent valve to flush the cavity of potential gaseous activity and to lower the stored fuel temperature. For the wet unloading, the cask cavity is flushed by pumping cooling water through the internal cavity via the cask vent and drain valves.

Subsequent to the unloading, the ISFSI site may elect to remove internal cask cavity surface contamination.

For a site-specific license application, the applicant would be expected to develop and commit to detailed procedures for use in unloading the cask.

12.3.2 Decommissioning of the Cask Components

Activation of the cask body, fuel basket, and closure lid are discussed in Section 3.5.1 (Storage Casks) of the TSAR. Supplemental information on the materials and their weights used in the activation calculations was provided in a letter dated November 13, 1987.

Neutron fluxes obtained from the XSDRNPM shielding calculations were used in conjunction with ORIGEN-2 to calculate the activities of ^{25}Na, ^{27}Mg, ^{28}Al, and ^{64}Cu in the fuel basket and ^{52}V, ^{51}Cr, ^{55}Cr, ^{54}Mn, ^{56}Mn, ^{55}Fe, ^{59}Fe, ^{58}Co, ^{60}Co, ^{60m}Co, ^{63}Ni, ^{65}Ni, and ^{209}Pb in the cask body and closure lid at unloading and one year subsequent to unloading. Several of these nuclides have half-lives on the order of minutes (^{25}Na, ^{27}Mg, ^{28}Al, ^{52}V, ^{55}Cr, and ^{60m}Co) and hours (^{64}Cu, ^{56}Mn, ^{65}Ni, and ^{209}Pb). In addition some (^{55}Fe, ^{63}Ni, and ^{209}Pb) emit no gamma rays.

In evaluating the activation products, the staff has assumed a minimum decay period of 60 days. At 60 days, 4.73×10^{-38} Ci of ^{64}Cu remain in the fuel basket, and 5.64×10^{-3} Ci of ^{51}Cr, 1.83×10^{-3} Ci of ^{54}Mn, 2.17×10^{-2} Ci of ^{55}Fe, 3.20×10^{-4} Ci of ^{59}Fe, 2.61×10^{-3} Ci of ^{58}Co, 4.88×10^{-3} Ci of ^{60}Co, and 1.40×10^{-3} Ci of ^{63}Ni remain in the cask body and closure lid.

Materials quantities used by NAC in the activation calculations are considerably larger than those in the cask itself. For conservatism in the calculation of activation product concentrations, the staff has assumed weights more representative to the cask. A weight and volume of 6,278 kgs (13,812 lbs) and 2.32 m^3, respectively, are assumed for the fuel basket. For the cask body and closure lid, a weight and volume of 57,304 kgs (126,068 lbs) and 6.24 m^3, respectively, are used.

With respect to decommissioning of the cask components, their activation product concentrations are such that the cask components at 60 days subsequent to unloading contain license-exempt concentrations of ^{51}Cr, ^{54}Mn, ^{55}Fe, ^{59}Fe, ^{58}Co, ^{60}Co, and ^{64}Cu. Furthermore, the activities or concentrations are such that the cask components may be classed as limited quantity materials for off-site transportation and may be disposed of as Class A waste.

12.4 Findings and Conclusions

The cask design is consistent with the requirements of 10 CFR Section 72.76 that an ISFSI be designed for decommissioning. The actions involved in cask unloading, are also consistent with the requirements of 10 CFR Section 72.18 as feasible elements of a site-specific decommissioning plan.

13.0 OPERATING CONTROLS AND LIMITS

13.1 Area of Review

Each license issued under 10 CFR Part 72 shall include license conditions pursuant to 10 CFR Section 72.33. In addition to the conditions pursuant to 10 CFR Section 72.33(b), each application for a license under 10 CFR Part 72 shall include proposed technical specifications pursuant to 10 CFR Part 72.16 and consistent with 10 CFR Section 72.33(c). The final approved technical specifications will be made part of the license.

The technical specifications of a license define certain features, characteristics and conditions governing operation of an installation. Technical specifications cannot be changed without approval of the NRC.

13.2 Acceptance Criteria

Consistent with 10 CFR Section 72.33(c), the operating controls and limits established in Chapter 10 of the TSAR will be deemed acceptable if they cover, for the cask, all required safety limits, limiting conditions for operation surveillance requirements and design features.

13.3 Review Procedure

Operating controls and limits which may serve as a basis for licensing conditions are derived from the analyses and evaluation included in the TSAR.

13.4 Findings and Conclusions

The staff reviewed the specific operating limits summarized in Chapter 10 of the TSAR. The limits established for these parameters reflect the design criteria upon which the safety analyses were based and are acceptable. With regard to the fuel characteristic limits described in Section 10.1.2 of the TSAR, the maximum enrichment is limited to 3.3 percent. Storage of only Westinghouse W-STD 15 x 15 fuel bundle designs was covered in the TSAR.

In addition, a maximum handling height of six feet with limiters attached should be included as an operating limit. Horizontal storage is not permitted.

A license applicant for an ISFSI must review parameters covered in the TSAR and develop appropriate proposed technical specifications and license conditions for the site-specific conditions.

14.0 QUALITY ASSURANCE

In Chapter 11 "Quality Assurance" of Revision 2 of the TSAR, NAC has committed to apply the NAC Quality Assurance Program, in a graded manner to the NAC S/T cask analysis, engineering, and fabrication. Section 3.4, "Classification of Structures, Components and Systems," of Revision 2 of the TSAR describes the three classes of S/T cask components and lists the components in each class.

The staff has reviewed NAC's commitments for quality assurance given in the TSAR. The staff finds that the NAC TSAR commitments for quality assurance meet the requirements of Subpart G of 10 CFR Part 72 for the NAC S/T cask and are, therefore, acceptable. The TSAR can be referenced without further quality assurance review in a license application to receive and store spent fuel under 10 CFR Part 72, provided that the applicant applies its NRC-approved quality assurance program that meets the requirements of Appendix B to 10 CFR Part 50 to the design, construction, and use of the spent fuel storage installation.

15.0 REFERENCES

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APPENDIX A

ANALYSIS OF DIFFUSION CONTROLLED CAVITY GROWTH (DCCG) DAMAGE TO FUEL CLADDING IN DRY STORAGE

1.0 INTRODUCTION

The only damage mechanism that the staff found with a possible potential for cladding degradation and gross rupture was DCCG. The staff has examined this potential and based on available information has developed a method to determine the level of damage which could occur under dry storage conditions for the NAC S/T cask as a function of spent fuel time in storage.

2.0 REVIEW PROCEDURE

The area fraction of decohesion at grain boundaries in Zircaloy cladding at any time can be ascertained by satisfying the following equation

$$\int_{A_i}^{A_f} \frac{dA}{f(A)} = \int_{o}^{t_f} G(t)dt$$

where A_i is the initial area fraction of decohesion due to the nucleation of stable cavities and A_f is the area fraction of decohesion that occurs over the period of time t_f. Furthermore,

$$f(A) = \{(1-(A_i/A)^{\frac{1}{2}} \sin \alpha)(1-A)\}/\{A^{\frac{1}{2}}[\{\ln(1/A)\}/2-3/4-A(1-A/4)]\}$$

$$G(t) = [32/3\pi]\ [F_B{}^{3/2}(\alpha)/F_v(\alpha)]\ [\Omega\delta\sigma_\infty(t)/k\lambda^3]\ [D_{gb}(t)/T(t)]$$

The terms of expression (2-2) and (2-3) are defined as follows:

α = grain boundary cavity dihedral angle

Ω = atomic volume

δ = grain boundary thickness

σ_∞ = stress on the cladding

k = Boltzman's constant

λ = average cavity spacing

Dgb = grain boundary diffusion rate

T = absolute temperature

$F_B(\alpha)$ = $\pi \sin^2\alpha$

$F_V(\alpha)$ = $[2\pi/3]\ [2 - 3\cos(\alpha) - \cos^2(\alpha)]$

Some of the foregoing terms may be further defined by

α = $\arccos(\gamma_B/2\gamma)$

D_{gb} = $D_{gbo} \exp[-Q/RT(t)]$

D_{gbo} = grain boundary diffusion coefficient

Q = activation energy for grain boundary self-diffusion

R = gas constant

γ = free surface energy

γ_B = grain boundary surface energy

Much of the review effort focused on establishing the values of the parameters in the above expressions. Where there was wide divergence in reported values, the value that led to the most conservative result was selected.

2.1 Grain Boundary Cavity Dihedral Angle, α

For clean surfaces in pure metals Raj and Ashby (Ref. 1) suggest that $\gamma_B = \gamma/2$ so that α is computed to be about 75°. To account for non-ideal conditions, a value for α of 50° was used in the analysis.

2.2 Atomic Volume, Ω

The atomic volume can be estimated from

$$\Omega = A/N\rho$$

where A is the atomic weight = 91

N is Avogadro's number = 6.02×10^{23}

ρ is the specific gravity = 6.55 gms/cc

which gives a value for Ω of 2.31×10^{-29} m^3/atom. This agrees closely with a value of 2.37×10^{-29} m^3/atom reported by Lloyd (Ref. 2). However, Chin, et al., (Ref. 3) used the cube of the Burgers vector, $b = 3.23 \times 10^{-10}$m, which gives an atomic volume of 3.37×10^{-29} m^3/atom. For the sake of conservatism, the value for $\Omega = 3.37 \times 10^{-29}$ m^3/atom was selected for the analysis.

2.3 Grain Boundary Thickness, δ

The grain boundary thickness defines the area through which grain boundary vacancies migrate to the cavity. The disorder that characterizes the structure at the grain boundary is only a few atoms thick. Since grain boundary diffusion rates are many orders of magnitude greater than volume diffusion rates, a grain boundary thickness of 3 Burger's vectors is considered adequate. Consequently, a value of $\delta = 3(3.23 \times 10^{-10}) = 9.69 \times 10^{-10}$ m was selected for the analysis.

2.4 Stress on the Cladding, σ_∞

The cladding stress is due to the fuel rod internal pressure at the storage temperature. There is considerable uncertainty regarding the level of pressure in the fuel rod, either from rod prepressurization, fission gas release, or volume increase due to creep strain. The maximum pressure expected in the fuel rods appears in Table 3.1-1 of the TSAR and is 1,462 psia at 360°C. A pressure of 1,508 psia at 380°C is the value used to start the DCCG damage analysis. Credit is taken for the reduction in pressure with the temperature reduction with time, as discussed in Section 2.7 below. It must be emphasized that DCCG damage is strongly a function of stress in the fuel cladding, and that this analysis is based on the 1,508 psi internal pressure. If the maximum pressure in any rod at the time the rods are put into the NAC S/T cask for storage is greater than the equivalent of 1,508 psi at 380°C, then this analysis is invalid.

2.5 Average Cavity Spacing, λ

The value of this parameter has been particularly difficult to establish. Cavity spacing depends upon the density of nucleation sites and will vary with the type of nucleation mechanism. Experimental work conducted at Cornell University (Ref. 5) indicated a spacing in unirradiated Zircaloy-2 of from 10 to 20×10^{-6} m. This experimental work further established that grain boundary cavities do form at 350°C especially at stresses over 100 MPa. The cavity density appeared to reach a saturation level after about 10 days suggesting a limited number of nucleation sites in the material. Consequently, it is not likely that the intercavity spacing, λ, will decrease during dry storage as a result of further nucleation. Conservatism dictated the use of the lower value of 10×10^{-6} for the analysis.

2.6 Grain Boundary Diffusion Rate, D_{gb}

There are many reported values of volume diffusion rate for α-Zirconium but few with respect to grain boundary diffusion rate. The two values specific for grain boundary diffusion are $6 \times 10^{-10} \exp(-112/RT)$ m^2/sec reported by Chin (Ref. 3) and $5.9 \times 10^{-6} \exp(-131/RT)$ m^2/sec reported by Garde, et al., (Ref. 6). The latter is the more conservative value by about two orders of magnitude and was, consequently, used for the analysis.

2.7 Temperature, T

The temperature dependence of grain boundary decohesion was established using the temperature decay curve provided in Figure 4.8-27 of the TSAR. These temperatures were adjusted upward by 20°C for use in this analysis in the SER, because of the review of the thermal analysis discussed in Section 4.1.4 of this TSAR. See Figure 2-1.

3.0 FINDINGS AND CONCLUSIONS

The progress of damage based upon the methodology and assumed values for the parameters previously described indicates that the area of decohesion at the end of twenty-year storage life to be less than 15% percent. Based upon the degree of conservatism maintained throughout the analysis, it can be concluded that this level of damage is minor and would not be exceeded. Consequently, an initial storage temperature not exceeding 380°C for the design basis fuel in a NAC S/T cask is acceptable for meeting the requirements of 10 CFR 72 Section 72.72(k).

REFERENCES

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2. L. T. Lloyd, "Thermal Expansion of Alpha-Zirconium Single Crystals," ANL-6591, Argonne National Laboratory (1963).

3. B. A. Chin, N. H. Madsen and M. A. Khan, "Application of Zircaloy Deformation and Fracture Maps to Predicting Dry Spent Fuel Storage Conditions," Department of Mechanical Engineering, Auburn University, Auburn, AL.

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6. A. M. Garde, H. M. Chung, and T. F. Kaisner, "Micrograin Superplasticity in Zircaloy at 850°C," Acta Met., Vol. 26, p. 153 (1978).

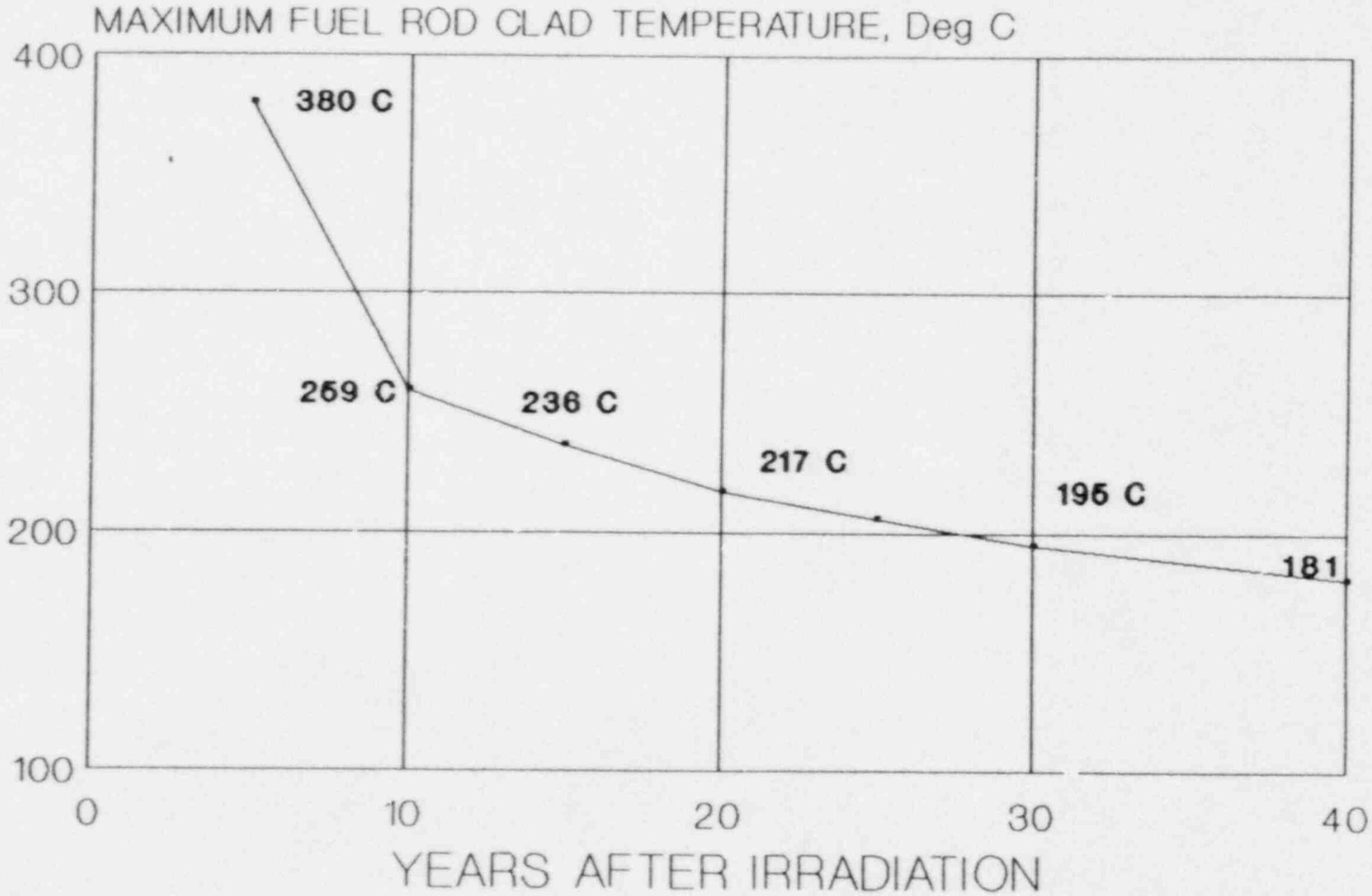

FIGURE 2-1: SPENT FUEL CLAD MAXIMUM TEMPERATURE VS DECAY TIME (NAC S/T)

TSAR CORRECTIONS

p. 4.8-32	Bolt shear. See p. 34 of SER.
p. 4.8-37	Lifting Trunnion Box y_c and Ix_c. See Sec. 3.4.4.4.1 of SER. Analysis of shear stress in trunnion base. See Sec. 3.4.4.4.1 of SER.
pp. 4.8-27, 33 and 44 and Fig. 4.2-4	Dimensional inconsistencies.
p. 4.8-29	Inconsistencies in equations. See Sec. 3.4.4.4.1 of SER.
p. 4.8-40 and Fig. 4.2-4	Plate thickness inconsistency.
p. 4.8-42	Ay for bolt 5 is 2.611, and I_s is 6.963 on this and following pages.
p. 4.8-45	Square root missing from equation for b.
p. 7.3-12 Table 7.3-4	Incorporate as appropriate information in supplemental letters dated October 21, 1987 and November 13, 1987.
Sections 3.1.1, and 7.2.1, and 7.3.2.3	Incorporate as appropriate information in supplemental letters dated October 21, 1987, November 13, 1987, and November 19, 1987. See Secs. 5.3.1.1, 5.3.1.2 and 5.3.2 of SER.