[DOE LETTERHEAD]

The Honorable John T. Conway
Chairman
Defense Nuclear Facilities Safety Board
625 Indiana Avenue, NW
Suite 700
Washington, DC 20004

Dear Mr. Chairman:

In accordance with my November 21, 1996, letter, enclosed is a more detailed response to your November 6, 1996, letter forwarding an August 15, 1996, Board staff report regarding operation of the Savannah River Site for the handling of spent nuclear fuel. This response was developed by supplementing the Department's previous response with further clarification, detail and schedule information. The schedule for delivery of associated documentation to be provided to your staff is also identified.

If you have any further questions regarding this matter, please contact me or have your staff contact John Ford of my staff at 301-903-3782.

Sincerely,

Alvin L. Alm
Assistant Secretary for Environmental Management

Enclosure

cc: M. Whitaker, S-3.1

ENCLOSURE

1. **There is no assurance that makeup water will be available after an accident. This would lead to an unsafe condition including risk to operators from exposure to very high radiation levels.**

   The Department has reviewed the concern related to the potential risk to operators and other workers from exposure to very high radiation levels following a cask drop accident. The cask drop accident scenario analyzed in the recently issued Basis for Interim Operations (BIO) for the L-Reactor Facility assumes that a 70 ton cask is dropped while being transferred, resulting in an unisolable leak in the basin structure. As the water leaks out, the water level would decrease, eventually uncovering the fuel. Although no damage would occur to the fuel, the radiation could potentially present a hazard to facility workers if they were not evacuated in accordance with current procedures.

Since a shipping cask drop has never occurred over the life of the reactors at SRS, the "rare event" theory of probability was applied, resulting in a very conservative frequency of 0.03 events per year. Studies indicate that dropping the 70 ton spent fuel cask could potentially result in a crack which has a maximum leak of 105 gpm. At this leak rate, operators would have a minimum of 6 days to implement mitigative actions before radiation levels began to increase above normal operating levels, at which point all workers in the vicinity would be evacuated. In the case of such an event, operators would respond to the decreasing water levels by implementing various procedures using available systems to restore basin water levels. Assuming the water level dropped to the -18" level, the current L-Reactor Technical Specifications require development of a response plan within 24 hours which would specify the appropriate actions to be taken.

Other accident scenarios which could involve basin water loss, such as a seismic event, have been analyzed in the BIO, although no credible scenario was identified that could cause basin leakage in excess of 1500 gpm. Water leakage of 1500 gpm would lower water levels at a maximum rate of 8 inches per hour, providing a minimum of 10 hours before radiation levels began to increase above normal operating levels and the workers were evacuated.

Because adequate time exists for evacuation of personnel, the relative risk of public or facility worker exposure to increased radiation levels is low. Although detailed pre-planning and demonstration of emergency capabilities are not required for accident scenarios that would allow adequate time for facility workers to respond, an integrated facility response to a basin leak is currently being developed. This response, to be completed in the second quarter of FY 97, will consist of a combination of operational procedures and engineering response plans whose objective is the mitigation of basin leakage.

The following items will be included in the response plan:

- Securing of fuel in transport
- Assessment of available radiation monitoring equipment
- Assessment of leakage rate
- Assessment of available systems and mitigation techniques
- Initiation of basin makeup systems as appropriate
- Implementation of leakage reduction measures

2. **Corrosion is evident along the entire length of the K-Basin cask crane's wire rope and the fatigue life of basin cranes in not known.**

The Department has reviewed the SRS Crane Inspection Program in light of the concerns raised in the DNFSB trip report. SRS has a comprehensive crane inspection program which is based upon a compilation of various national and international codes and standards. This program is designed to provide assurance that adequate safety margins are maintained under all crane operating conditions by establishing inspection criteria and inspection frequencies. Although the inspection program for the 85 Ton Crane is discussed in detail as an example, the same program is applied to all cranes in all facilities at SRS.

The quarterly and annual crane inspections are performed in accordance with Overhead and Gantry Cranes, ASME B30.2, Chapter 2-2. The crane wire ropes are inspected on a monthly frequency pursuant to "Overhead and Gantry Cranes," 29 CFR 1910.179(m), and inspection criteria in accordance with ASME B30.2, Section 2-2.4. Other standards and reference manuals used in the development of the inspection program include Wire Rope Users Manual, Rigging Manual by the Construction Safety Association of Ontario, and Department of Energy Hoisting and Rigging Standard. The Whiting Crane Handbook notes that actual rope life is determined by careful inspection in accordance with the recommendations of ASME B30.2

The inspection procedures for the wire rope include a thorough inspection for signs of fatigue, over-stress, wear, and corrosion, and include inspections for broken outside wire strands, the distribution of any broken strands, changes in wire rope diameter, corroded or broken wires at end connections, corroded, cracked, bent, or worn end connections, and severe kinking, crushing, cutting, or unstranding of hoist cables. The Rigging Manual by the Construction Safety Association of Ontario warns against the potential for internal rope corrosion, and it provides criteria, such as reduction in rope diameter and complete lack of strand gap, as indications of core failure due to corrosion. These inspection criteria are incorporated in the Site's inspection program.

Given the Defense Nuclear Facility Safety Board (DNFSB) concern over internal wire rope corrosion, a supplemental inspection was conduced on November 14, 1996. The inspection consisted of visually examining the wire rope under load as it wound onto the hoist drum. The outer cable strands naturally spread to expose more of the outer cable strand wires and to a lesser extent the perimeter inner core wires. No pitting or corrosion damage was observed, either externally or internally. Although surface oxidation was noted, it was easily removed when wiped. This type of oxidation is expected considering the crane operating environment. We would also note that the wire ropes in question are stainless steel rather that carbon steel.

The Savannah River Technical Center (SRTC) further reviewed the DNFSB wire rope corrosion concerns and concluded that loss of over 25% of the cable wires would be required to reduce the cable strength to a level that would compromise integrity at the operating safety margins. Failure of this many wires by any initiated source, whether fatigue, wear, abrasion, or corrosion, would readily be detected by the crane inspection program. Given the low number of crane load cycles between monthly wire rope inspections, degradation of the magnitude required to compromise cable integrity is highly unlikely.

Similarly, the crane bridge and trolley are inspected for signs of wear, over-stress, and fatigue. The inspection criteria requires verification of structurally sound connections at welds, rivets, bolts, and lack of wear or cracks in the truck wheel assemblies and trolley rails. In addition, the bridge and trolley alignment is verified as an initial indicator of crane overload or excessive truck wheel assembly wear.

The lifting arrangement of the 85-Ton cranes provides redundant load carrying cables and would therefore require simultaneous or sequential failure of two hoist cables to result in a cask drop event. Given the nominal strength of each cable as 69 tons, the static load capacity of the crane is 552 tons. The design factor applied to these cables accommodates static loads and the effect of dynamic loading and material degradation due to fatigue, wear, or abrasion.

Crane inspections are also conducted more frequently than what is required by national standards in terms of crane usage category. Although the 85-Ton crane usage is classified as "heavy service," SRS has conservatively elected to perform inspections on a quarterly, rather than annual, basis commensurate with "severe service" criteria. "Heavy service" is defined by ASME B30.2 as crane usage that involves operating at 85 to 100% of rated load or 10 lift cycles/hr. While "severe service" is defined as crane usage that involves normal or heavy service with abnormal operating conditions. Although the crane does lift nearly the rated capacity when handling a CD cask, the cyclical use of the crane is very low. During future off-site cask handling operations of research reactor fuel, the crane usage would be classified as "normal" since the crane load is approximately 35% of rated capacity with much fewer than 10 lift cycles/hour.

The crane inspections are performed by qualified site inspectors who are independent of the facility. Crane inspection personnel receive specific training provided by crane manufacturers and off-site inspection agencies and are re-certified every three years. They are unbiased and unaffected by normal facility operations and schedule commitments.

Crane inspection program documents for the K- and L-Basins and RBOF will be provided to the DNFSB describing inspection requirements, frequency and criteria, load tests, dye-penetrant tests, etc.

WSRC is in the initial stages of scoping the need for a crane upgrade/ modernization proposal for the DOE. As part of this scoping, a representative from the crane manufacturer will perform a baseline inspection/evaluation of the 85 Ton crane during the second week of December. The vendor will issue a report of its finding within 30 days of the inspection. The crane vendor evaluation will be used as input towards development of a upgrade/ modernization proposal for this and similar cranes if required. All documentation for item #2 will be transmitted within 60 days of the crane manufacturers inspection.

3. **A qualified rigger is not present during fuel cask lifts**

In response to the DNFSB concerns about facility operator qualifications, the Department has reviewed the training and qualification program. The SRS Hoisting and Rigging Manual states that the training for both "a person who rigs as a major part of their job" and an "incidental rigger" shall include the same basics but the depth of detail shall match the job assignment. Pre-engineered lifts are established for routine, repetitive lifting jobs such as cask movement. For these lifts, procedures are established which define the rigging equipment and the process used. These cask handling procedures are reviewed by fully qualified site rigging personnel.

The Facility operators attend "incidental rigger/operator" training. The "incidental rigger" is trained to ensure the rigging equipment has the required capacity and is in good condition, the rigging equipment is utilized per procedure, and that the load path is clear. These qualifications are considered appropriate for the routine pre-engineered operations conducted in the facility. These personnel are therefore classified as "Incidental riggers" in accordance with section 4.2.4.C. of the SRS Hoisting and Rigging Manual and meet the requirements for qualification of production personnel performing pre-engineered lifts in section 3 of the DOE Hoisting and Rigging Standard. In addition, WSRC will continue to conduct an annual review of rigging practices conducted by incidental riggers by a fully qualified rigger, as part of the overall Self-Assessment process for the facility.

4. **There is no disposition for highly radioactive scrap metal located in the basins.**

   We understand the concern of the DNFSB to be the lack of a plan for the disposal of scrap materials in the basins. WSRC, as part of efforts to enhance water quality and reduce hazards, has been pursuing the removal of selected materials from the basins. Activities to date have focused on removal of the more serious hazards, including cadmium control rods, corroded fuel, and excess radioactive sources. Future efforts will be directed at the remaining materials such as irradiated metal and contaminated scrap.

   Past practices at SRS for disposal of highly radioactive scrap involved direct burial in slit trenches. DOE has recently improved methods of disposing waste at SRS. These methods are captured in a waste certification program which ensures that the characteristics of all wastes are identified so that it can be disposed of in the proper container and repository. The Spent Fuel Storage Division is currently revising the RBOF Waste Certification Plan to reflect the new waste disposal process. This plan will encompass all known wastes, including the basin scrap. Implementation schedules will be defined based upon resource and budget availability.

   In the interim, the scrap metal is in the process of being removed from RBOF. This removal would allow the future, potential installation of additional storage racks, if appropriate. The scrap is being stored, in accordance with the Waste Certification Plan, in a safe storage condition which does not present a risk to the public or facility workers. Several controls are in place to protect the facility worker from excess exposure to the material, including radiation monitors, review or work packages, and requirements for oversight by radiological control personnel. The experience gained in removal of scrap from RBOF will be used to ensure the successful removal and disposal of scrap in L-Basin.