Pre-shipment Preparations at the Savannah River Site

WSRC's Technical Basis to support DOE's Approval to Ship

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ABSTRACT

In the first four years of the Foreign Research Reactor (FRR) Spent Nuclear Fuel (SNF) Return Program following resumption of the SNF return program with the DOE-EIS ROD in May 1996, 13 shipments involving 77 casks with over 2,600 assemblies have been safely received and stored at the Savannah River Site (SRS). Each fuel type has gone through a rigorous pre-shipment preparation process that includes fuel characterization, criticality safety reviews, and operational reviews, culminating in the Department of Energy's (DOE's) authorization to ship.

Ideally, the authorization to ship process should begin two years in advance of the fuel receipt with an agreement between the Department of Energy – Head Quarters (DOE-HQ) and the research reactor government on the conditions and protocol for the spent nuclear fuel return, with a target of DOE shipment authorization at least two months before facility loading. A visit by representatives from the Department of Energy – Savannah River (DOE-SR) and Westinghouse Savannah River Company (WSRC), DOE's Management & Operations (M&O) Contractor for the SRS, to the research reactor facility is then scheduled for the purpose of finalizing contractual arrangements (DOE-SR), facility assessments, and initial fuel inspections. An extensive effort is initiated at this time to characterize the fuel in a standard format as identified in the Appendix A attachment to the contract. The Appendix A must be finalized in an accurate and timely manner because it serves as the base reference document for WSRC and other involved stakeholders such as the cask owners and the competent authorities throughout the approval process.

With the approval of the Appendix A, criticality safety reviews are initiated to evaluate the unloading and storage configurations. Operational reviews are conducted to allow for necessary adaptation of fuel handling facilities, procedures, and training. WSRC has proceduralized this process, 'Certification to Receive and Store', to provide a disciplined method for ensuring that all of the pre-shipment receipt preparations have been completed.

This paper will provide a detailed description of each of the pre-shipment process steps WSRC performs to produce the technical basis for approving the receipt and storage of spent nuclear fuel at the Savannah River Site. It is intended to be a guide to reactor operators who plan on returning "U.S. origin" SNF and to emphasize the need for accurate and timely completion of pre-shipment activities.

DISCUSSION

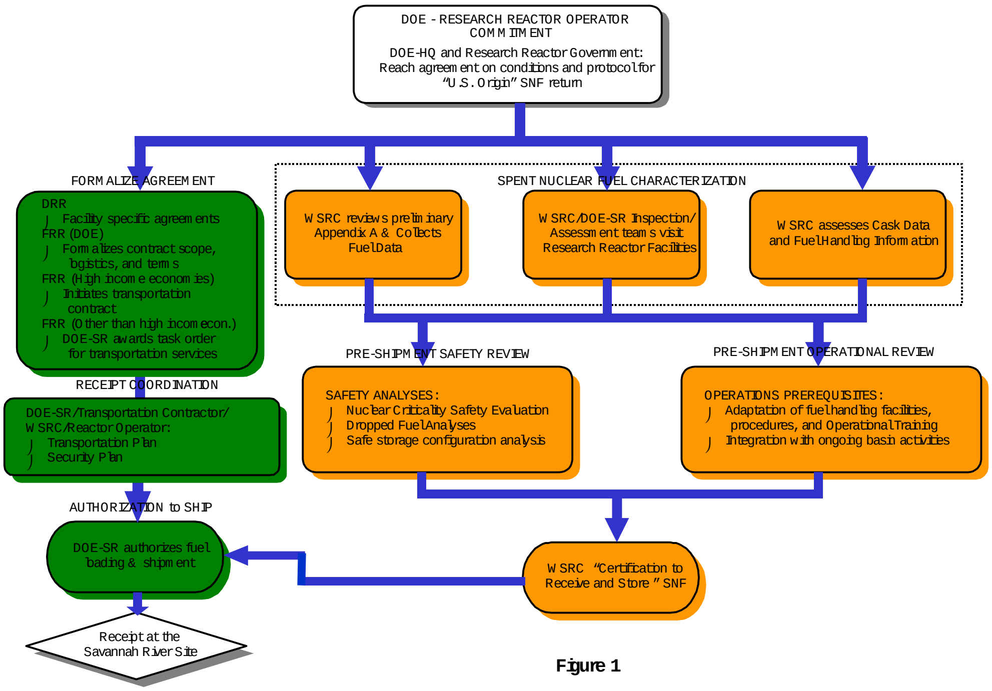

Figure 1

Receipt Preparations

The Department of Energy – Savannah River (DOE-SR) and Westinghouse Savannah River Company (WSRC) have developed a disciplined process for completing the required activities to approve the safe receipt and storage of spent nuclear fuel (SNF) at the Savannah River Site (SRS). Figure 1 depicts the flow of a typical approval process for first time participants. Two years in advance of the planned fuel return to SRS, the pre-shipment receipt preparation process begins with the Department of Energy (DOE) and the research reactor government negotiating the conditions and protocol for the return of "US Origin" SNF. Once an agreement on the terms of the shipment has been reached, the approval process proceeds in two parallel working paths, an Administrative path and a Technical path. The administrative path, DOE-SR's responsibility, is represented by the activities grouped on the left side of the Figure 1 and involves the formalizing of agreements and specific terms of the contracts. The focus of this paper is, however, on the five groups of activities in the center and on the right side of the Figure 1, which is the responsibility of WSRC, and provides the technical bases to support the safe receipt and storage of the fuel.

Spent Nuclear Fuel Characterization

WSRC's first, and most important; step in the receipt preparations process is to secure an accurate characterization of the fuel. Characterization activities are segregated into three groups, which, ideally, are worked in parallel and are completed at least nine months prior to fuel loading (see Figure 2, T-9 months). First, is the characterization of fuel data in the Appendix A attachment to the contract. The second involves a team of DOE-SR and WSRC personnel visiting the reactor facility to inspect and assess the fuel and facility. The third and final fuel characterization activity involves a review of the cask design/certification parameters and how the fuel is to be handled. The following paragraphs will provide a more detailed description of these areas.

Data Collection

Fuel data is collected in a document referred to as the "Appendix A", which is by contract a part of the DOE's agreement with the reactor facility. The reactor operator provides the Appendix A information to WSRC for use in validating compliance with SRS facility operations Safety and Authorization Basis. WSRC uses the reference documents, such as drawings, fuel fabrication reports, reactor safety analysis reports, and others, to review the submitted data. The Appendix A is approved when all of the comments have been resolved in the comment/resolution cycle. Accuracy and timeliness are important factors in this comment/resolution cycle and are essential for the success and most cost effective execution of each shipment.

Thoroughness and accuracy in preparation of the Appendix A is important for several reasons. First, the technical information provided, including drawings and other reference material, is used by SRS as the basis for safety and operational reviews to ensure safe receipt and storage of the fuel in the existing wet storage basins. Secondly, this fuel data provides the basis for cask owner verification and/or alignment to the selected cask license for transport of a particular fuel. Inaccurate fuel data may delay the cask license certification, with the potential for schedule impacts in shipment of the fuel from the reactor facility. Finally, thorough and accurate Appendix A data ensures that fuels will be properly characterized for ultimate disposition in a future permanent repository.

Timely submittal of the Appendix A document is also very important. Ideally, the final Appendix A document should be approved at least thirty-six (36) weeks prior to the scheduled fuel loading (See Figure 2). Historical trends indicate that approximately nine (9) to twelve (12) weeks are required for the SRS review of initial Appendix A fuel data, and involves critical resources of SRS and research reactor operators. The initial Appendix A document should therefore be submitted approximately one year in advance of the scheduled fuel loading. Early finalization of the Appendix A document will allow ample time for SRS to complete its safety bases and operational reviews and implement any new facility modifications, process changes, or special training of basin operations personnel that may be required to safely receive, unload, and store the fuel. Additionally, if cask license reviews and revisions are required for transport of a particular fuel, the cask owner, competent authority and the U.S. Nuclear Regulatory Commission have taken the position that license reviews not begin until the Appendix A has been finalized. Depending on the extent of the evaluations needed to review license submittals, the US NRC/DOT approval process could range from 7 weeks to 12 months. Therefore, late submittals of Appendix A's have the potential to result in significant delays or cancellation of shipments because of licensing issues.

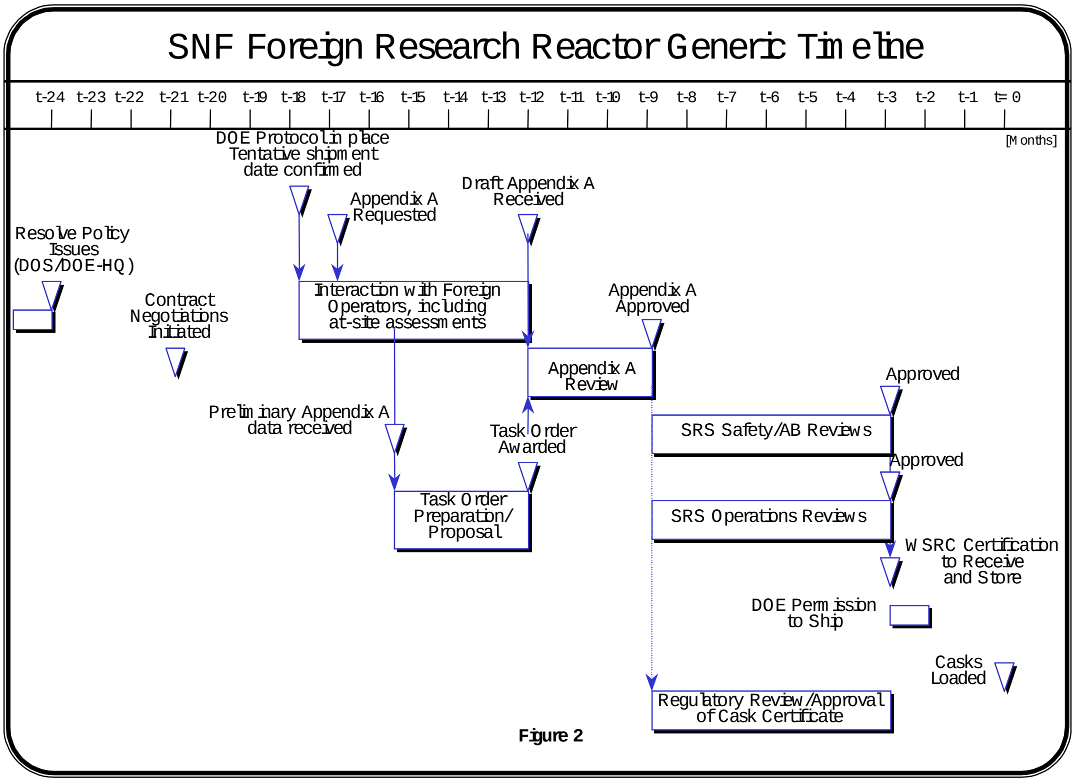

Figure 2

Inspection/Assessment Visits

For those reactor facilities that are preparing for their first shipment, a team of DOE-SR and WSRC representatives will visit the reactor facility. These visits are scheduled to occur 12 to 18 months in advance of the intended receipt date in order to initiate the exchange of technical information and to identify and resolve early concerns. Contracts between DOE-SR and the reactor facility are developed and preliminary fuel return logistics restraints are identified during these visits. WSRC also inspects the fuel at that time for structural integrity, evidence of corrosion, and ease of handling and any other indications that could possibly affect receipt and storage of the fuel in the SRS basins. Facility assessments are conducted on basin water chemistry, cropping techniques, and fuel and cask handling operations for the early identification of issues that may affect fuel handling, storage and schedule. The visit also presents an excellent opportunity for the reactor operator and WSRC representative to discuss Appendix A issues for timely resolution. To date, 34 foreign reactor facilities have been assessed with over 2000 assemblies inspected.

Cask and Fuel Handling Assessment

Once the cask to be used for the shipment has been chosen by the research reactor operator at T-9 months, WSRC initiates an independent review of the various documents that describe the cask, its licensed contents, and its handling. The cask's physical dimensions and handling methods are reviewed against the capabilities of L-Basin and the Receiving Basin for Off-site Fuel (RBOF). Areas of concern are brought to the attention of the SRS Basin Management Team for resolution. The fuel data compiled in the Appendix A document is compared to the licensed contents specified in the Certificate of Compliance to determine if a license revision is required. Ongoing communications are established to discuss any potential discrepancies. The cask owner is immediately contacted if a discrepancy is found. The shape of the fuel and its configuration in the cask basket will dictate the proper handling tool to be used during unloading and movement to the bundling station. Often, assistance from the reactor facility is needed to properly determine the correct fuel handling tool.

Pre-Shipment Safety and Operational Reviews

Once the Appendix A fuel data is finalized, and cask/fuel configuration is approved and the other characterization activities completed, the information is passed on to the SRS criticality safety and operations departments. The criticality engineers perform evaluations to ensure that the fuel can be received, unloaded and stored without the possibility of a criticality incident. An operational review is conducted to ensure that the fuel handling facilities, procedures and training have been adapted for the receipt and storage of the fuel and the fuel is within the limits specified in the facility's Authorization Basis. These reviews are discussed in more detail in the following sections.

Criticality Safety Reviews

Prior to receiving and storing SNF in either SRS's L-Basin or RBOF, a Nuclear Criticality Safety Evaluation (NCSE) is performed to determine the criticality safety limits for fuel storage and handling. The results are summarized on a Nuclear Safety Data Sheet (NSDS) which is the document used by Operations for routine fuel handling and storage.

The methodology for the analysis of fuels has evolved significantly since the first receipt of MTR fuels. In order to maximize the efficiency of our criticality resources, a Uniform Fuel Evaluation Methodology was developed that provides a consistent, well defined, and automated approach to fuel receipt analyses. The primary objective of the methodology is to demonstrate that the incoming fuel is less reactive, in the configurations of interest, than a bounding fuel that has already been shown, via DOE evaluated model calculations, to be safe for the various storage configuration used at SRS. This "reactivity comparison" approach eliminates the need for numerous calculations associated with the evaluation of interactions, involving spacing and tilting, between the incoming fuel and the existing fuels. Efforts are underway, to expand even further, the bounding restraints for these analyses to reduce fuel specific analyses.

For this fuel receipt methodology, an analyzed configuration is considered "safely subcritical" if the following condition is satisfied:

$$k_{eff}+2\sigma<k_{safe}$$

where,

k_{eff} = SCALE/MCNP calculated average effective neutron multiplication factor

σ = statistical uncertainty associated with the Monte Carlo calculation

The value for k_{safe} is determined from the following:

$$k_{safe}= 1 + (\text{bias-bias uncertainty}) – \text{AOA Margin} – \text{MSM}$$
$$k_{be} = 1 + (\text{bias} – \text{bias uncertainty})$$
$$k_{be} = \text{LTL}$$

where,

AOA: = Area of Applicability

MSM: = Minimum Subcritical Margin

k_{be}: = Best estimate of the uppermost calculated neutron multiplication for which a system is not expected to be critical

LTL: = Lower Tolerance Limit

Rearranging the terms gives the following equation:

k_{safe} = LTL – AOA Margin – MSM

Calculated values of $k_{eff}+2\sigma<k_{safe}$ will indicate configurations that are safely subcritical. Configurations with calculated $k_{eff}+2\sigma\geq k_{safe}$ will be considered unacceptable.

Applicable fuel types for the Uniform Fuel Receipt Methodology must satisfy the following criteria:

- The primary fissile isotope is U^{235}.
- The fuel plates/tubes must be aluminum-clad with a 'U-Al type' fuel meat region.
- The fuel must be suitable for homogenization.
    - Enrichment must be greater than or equal to 15%.
    - Fuel meat regions must be thinner than 0.63 cm.

This methodology is appropriate for use with the SCALE code (version 4.4), the MCNP 4B code package and the CASE code. SCALE, which was developed at Oak Ridge National Laboratory for the Nuclear Regulatory Commission, is a multigroup Monte Carlo criticality program used to calculate k_{eff} of 3D systems. MCNP, which was developed at Los Alamos National Laboratory, is a continuous energy Monte Carlo criticality program that is also used to calculate k_{eff} of 3D systems. CASE was developed at SRS using SCALE 4.2.

Fuel types that satisfy the above criteria are evaluated using the Uniform Fuel Receipt Evaluation Methodology described below. For those fuel types that do not meet the above requirements, it is necessary to use explicit fuel models and account for all possible fuel interactions to perform the evaluation.

1. The U^{235} content, enrichment, H/fissile atom ratio and geometry are determined for the incoming fuel type.

2. The incoming fuel (rectangular cross section) is modeled as a homogenized cuboid with a square cross-sectional area equivalent to the actual fuel assembly cross-sectional area. Fuel plates, side plates, and dummy plates are included in the homogenization. The homogenized model will more conservatively model non-rectangular cross-section fuels.

3. The number of assemblies that can be handled and stored in a bundle is determined by calculating the safe number. The safe number of assemblies for the incoming fuel is determined by producing reactivity vs. spacing curves (see Figure 3). The definition of safe number is the maximum number of assemblies that are safe in any configuration in light water. To find the most reactive configuration for a given number of fuel assemblies, $k_{eff}+2\sigma$ is calculated as a function of spacing between multiple (e.g. seven) fuel assemblies starting with the assemblies in contact. Because of the structure of the wet basins, concrete reflection is used in these calculations. The spacing is varied to plot the data points and the number of assemblies is decreased until the peak falls below k_{safe}.

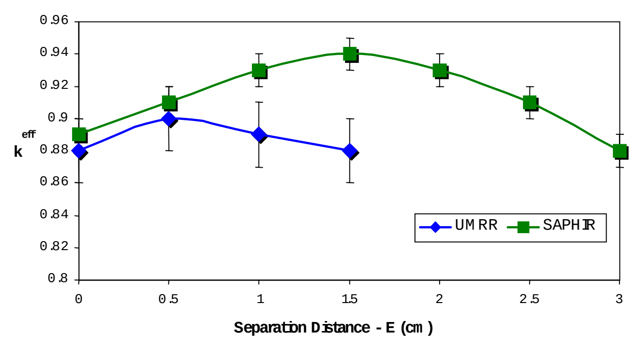

Figure 3
Typical 'Reactivity vs. Spacing Curve"

4. For the storage of the new fuel in the primary storage basin, L-Basin, a reactivity vs. spacing curve is generated comparing the reactivity of multiple (e.g. four) bounding assemblies with the reactivity of the same number or more new assemblies. If the reactivity for the bounding assembly is greater, the new fuel may be stored in any L-Basin storage rack with up to the safe number of assemblies in each bundle. Similar evaluations are performed for the storage racks in RBOF (only one evaluation is needed if the same fuel is used as boundary)

5. The last analysis is the examination of the dropped fuel scenario during cask unloading operations. This analysis is used to demonstrate the dropping an assembly onto the cask will not result in an unsafe

configuration. The reactivity comparison is performed in the same manner as in Step 4 above, except that a bounding fuel assembly for the cask is used. Additional methods of evaluation are discussed in reference 4.

A significant history of analyses has been developed using the above methodology. Statistical models have been generated based upon these past analyses that provide conservative estimates of $k_{eff}+2\sigma$ values for candidate MTR fuels. These statistical models are incorporated into a spreadsheet, referred to as the Candidate Assembly Statistical Evaluation (CASE) program. Model values are compared by the spreadsheet with a k_{safe} of 0.95 to determine whether the tested configurations will be less than k_{safe}.

The criticality evaluations average about four to six weeks to complete, a vast improvement over the old methodology. Accurate Appendix A data is vital so that time intensive analyses do not require to be re-run in order to accommodate newly discovered or corrected data.

Operational Reviews

Operational reviews are conducted in parallel with the criticality safety reviews to ensure that the facility and personnel are prepared to receive and unload the fuel. The fuel is reviewed against the facility's Authorization Basis (AB) to ensure that a design criterion is not compromised. Cask handling issues, such as facility modifications, that were identified in the cask assessment mentioned earlier are implemented. A full set of spare tools is inventoried and staged in order to minimize unloading delays when specific cask tools are not included with the cask. Receipt, unloading, cropping and storing procedures are reviewed and revised to incorporate new casks, fuels or handling methods. Basin supervisors, operators, and engineers are thoroughly trained on these procedures. Information brought back from the reactor facility assessment trips proves valuable in the development of the procedures and training sessions.

In addition to the off-site receipt program, WSRC manages and coordinates a number of fuel transfers on-site for DOE legacy material stabilization. In order to utilize the fuel handling facilities efficiently and minimize cask processing delays, the SRS Basin Management Team was created to integrate all basin activities and promote good communications within the various work groups. This team includes representatives from engineering, operations, health protection, DOE, fuel integration, and criticality. The team actively works together with a joint team meeting at least weekly to develop integrated schedules that incorporate off-site receipts, on-site transfers, facility outages, and routine basin operations activities. It also provides the reactor operators and cask owners with a reliable schedule of when the cask will be processed and returned for its next use.

All of the information collected in the fuel characterization, criticality safety review, and operational review provides the technical bases for WSRC to submit the "Certification to Receive and Store" to DOE. The WSRC "Certification to Receive and Store" process is built around a procedural checklist that contains the technical justification to ensure the safe receipt and storage of the fuel at SRS. DOE-SR grants the "Approval to Ship" to the reactor facility upon receipt of this documentation.

Conclusion

This receipt preparation process provides a good foundation for the success of a shipment and for the Spent Nuclear Fuel Program. The preparations for the safe return of spent nuclear fuel to the Savannah River Site starts well in advance of the actual receipt date. One to two man-years of work at SRS are spent executing the technical and operational reviews, analyses, and evaluations that support WSRC's Certification to Receive and Store. For this reason it is important to maintain a disciplined approach and schedule to ensure that all pre-shipment preparation activities are initiated in a timely manner with accurate data. Failure to do so can result in significant rework and shipment delays in shipment of spent fuel into SRS and elevated costs both to the research reactor operator and the DOE.

References

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