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Licensing of Spent Fuel Dry Storage and Consolidated Rod Storage: A Review of Issues and Experiences

February 1990

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LICENSING OF SPENT FUEL DRY STORAGE
AND CONSOLIDATED ROD STORAGE:
A REVIEW OF ISSUES AND EXPERIENCES

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the U.S. Department of Energy
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Pacific Northwest Laboratory
Richland, Washington 99352

ABSTRACT

The results of this study, performed by Pacific Northwest Laboratory (PNL) and sponsored by the U.S. Department of Energy (DOE), respond to the nuclear industry's recommendation that a report be prepared that collects and describes the licensing issues (and their resolutions) that confront a new applicant requesting approval from the U.S. Nuclear Regulatory Commission (NRC) for dry storage of spent fuel or for large-scale storage of consolidated spent fuel rods in pools. The issues are identified in comments, questions, and requests from the NRC during its review of applicants' submittals. Included in the report are discussions of 1) the 18 topical reports on cask and module designs for dry storage of spent fuel that have been submitted to the NRC, 2) the three license applications for dry storage of spent fuel at independent spent fuel storage installations (ISFSIs) that have been submitted to the NRC, and 3) the three applications (one of which was later withdrawn) for large-scale storage of consolidated fuel rods in existing spent fuel storage pools at reactors that were submitted to the NRC. For each of the applications submitted, examples of some of the issues (and suggestions for their resolutions) are described. The issues and their resolutions are also covered in detail in an example in each of the three subject areas: 1) the application for the CASTOR V/21 dry spent fuel storage cask, 2) the application for the ISFSI for dry storage of spent fuel at Surry, and 3) the application for full-scale wet storage of consolidated spent fuel at Millstone-2.

The conclusions in the report include examples of major issues that applicants have encountered. Recommendations for future applicants to follow are listed.

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SUMMARY

The new applicant is faced with numerous licensing issues when applying for approval from the U.S. Nuclear Regulatory Commission (NRC) to place spent light-water (LWR) fuel in dry storage casks or modules at independent spent fuel storage installations (ISFSIs) or for large-scale storage of consolidated spent LWR fuel rods in pools. The issues are identified in comments, questions, and requests from the NRC during its review of the applicants' submittals. Many of those issues have been addressed in earlier licensing reviews by the NRC but are documented in diverse places. The nuclear industry recommended that a report be issued that collects and describes major licensing issues and their resolutions. In response to that recommendation, a study was performed under the Commercial Spent Fuel Management (CSFM) Program, which is sponsored by the U.S. Department of Energy (DOE) and managed by Pacific Northwest Laboratory (PNL), with the objective of preparing such a report.

The results of the study are presented in this report. Discussed in the report are the 18 applications that have been submitted to the NRC for approval of dry storage casks or modules, the three applications that have been submitted for dry storage of spent fuel in at-reactor ISFSIs, and the three applications (one of which was later withdrawn) for large-scale storage of consolidated fuel rods in existing spent fuel storage pools at reactors. For each of the applications submitted, examples of some of the issues (and suggestions for their resolution) are provided. The issues and their resolutions are described in more detail for one example in each of the three subject areas: 1) the application for the CASTOR V/21 dry storage cask (Project M-37), 2) the application for an ISFSI for dry storage of spent fuel at Surry (Docket No. 72-2), and 3) the application for full-scale wet storage of consolidated spent fuel at Millstone-2 (Docket No. 50-336). Comments on each of the three subject areas are presented below.

ISSUES IN APPLICATIONS FOR APPROVAL OF DRY SPENT FUEL STORAGE CASK AND MODULE DESIGNS

In reviewing the applications for approval of dry spent fuel storage casks or modules, the NRC has raised a number of issues. Allowance for burnup credit remains an open issue. The NRC is concerned that, during the typical process of loading and/or unloading spent fuel into a cask or canister in the spent fuel storage pool, there is the potential for introducing nonborated water, which would result in undermoderation because of reduced water density. The NRC has major concern with components (e.g., cask bodies and fuel baskets) that are proposed to be made of nonspecification materials. Such materials must be adequately characterized if they are to be approved by the NRC. Storage of spent fuel in air-filled casks or modules has not been accepted yet by the NRC. However, the NRC Staff does not reject the contention that continued research on this subject may subsequently result in allowance of air as a cover gas for dry storage of spent LWR fuel. A major issue with one application was the criticality design of the fuel basket, which relied on the insertion of a "poison spider" assembly of rods into a fuel assembly. The NRC questioned whether this approach can satisfy the design criteria of 10 CFR Part 72 for at-reactor site storage, at least in the context of a nonsite-specific topical report. The discovery of cracks in the borated stainless steel fuel basket (Project M-37) was of great interest to the NRC. The cracks were caused by constrained thermal expansion. The problem was solved by incorporating a simple design modification that involved eliminating fabrication stitch welds on the first few feet of either end of the basket. An issue with one application (Project M-44) was the use of scale models of the cask in the drop testing program. There is a need for thorough editing of applications. With at least four of the applications, the NRC commented that there appeared to be errors, inconsistencies, and omissions. In some applications, reduced drawings were too small to be legible.

A recent article indicates that the NRC is proposing a rule change that would permit LWR licensees to store spent fuel in NRC-approved dry storage casks (free-standing metal and concrete type) under a general license that would not require additional site-specific reviews by the NRC. So far, four

metal dry storage casks that have been approved by the NRC would come under the general license coverage. The four are CASTOR V/21 (Project M-37), MC-10 (Project M-41), NAC-S/T (Project M-40), and NAC-C28 S/T (Project M-51).

ISSUES IN APPLICATIONS FOR DRY STORAGE OF SPENT FUEL IN AT-REACTOR ISFSIs

A number of issues were involved with these applications. One issue raised by the NRC centered on the question of whether, and to what extent, the spent fuel storage canisters to be used at the ISFSIs will be compatible with the transportation casks ultimately chosen by DOE to transport spent nuclear fuel temporarily stored at reactor sites to permanent repositories. One applicant indicated, given the fact that DOE's design effort for its cask fleet is in its early stages, that no meaningful compatibility review can be conducted at this time. With one application, the NRC stated that the associated Environmental Report did not fully comply with 10 CFR 72. There is a need to show that an ISFSI at a reactor site operates independently (as defined in 10 CFR Part 72) from the reactor. An array of casks/modules needs to be considered in the application.

ISSUES IN APPLICATIONS FOR LARGE-SCALE STORAGE OF CONSOLIDATED FUEL IN EXISTING POOLS AT REACTORS

The NRC raised a number of issues with these applications. With one application (Maine Yankee), an issue involved the need for a procedure for controlling the temperature of the pool water so it does not exceed a specified limit. With two applications (Maine Yankee and Milestone-2), an issue involved controlling/justifying the decay time of the spent fuel. With one application (Maine Yankee), the NRC indicated that the applicant was to avoid lifting a spent fuel shipping cask over the pool until a cask drop analysis was submitted by the licensee and approved by the NRC. Applicants were requested by the NRC to provide a complete description of the consolidation process and an associated safety analysis. Generally, in completing an application, there is a need to eliminate inconsistencies and the potential for unnecessary confusion and incorrect perceptions. With one application (Millstone-2), the NRC raised an issue over the lack of discussion of a testing program involving the spent fuel rack system and the use of the

resulting data in the Safety Analysis Report. The NRC requested the discussion because the testing portion seemed to be an important step in doing a reliable analysis of the spent fuel rack system. An issue with one application (Millstone-2) was the applicant's assumption that no lateral deformation results from a 100,000-lb load impacting the fuel assembly. The NRC stated that the assumption is not justified and seems invalid because it has been determined that the guide tubes yield due to the load. An issue with one application was the discussion about a reactivity meter for determining and verifying fuel burnup and the lack of mention of the meter in the Safety Analysis Report. The applicant's response indicated that the reactivity meter is a state-of-the-art device, which is still under development. Determination and verification of fuel burnup is currently controlled administratively. An issue with one application (Millstone-2) involved NRC's request for information on the method to be used to identify fuel rods that are most likely to rupture. The applicant's response indicated that before consolidation, all candidate fuel assemblies will be inspected with the consolidation system TV cameras to review the condition of the exterior rows of rods. Any fuel assemblies found to have defects will not be consolidated, according to that applicant. In its review of one application (Millstone-2), the NRC recommended that a Technical Specification Surveillance Requirement be incorporated for consolidated fuel to verify the integrity of the fuel and structural elements before movement or placement in the spent fuel pool. The NRC also requested that the applicant provide information on the proposed method for verifying the integrity of the storage canister after it has been loaded with fuel rods.

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(a) NUHOMS is a registered trademark of Nutech Engineers, Inc., San Jose, California.

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1.0 INTRODUCTION

When applying for approval from the U.S. Nuclear Regulatory Commission (NRC) to place spent light-water reactor (LWR) fuel in dry storage or to store a large quantity of consolidated spent LWR fuel rods in pools, new applicants are faced with numerous licensing issues (i.e., comments, questions, or requests from the NRC), many of which have been addressed in earlier licensing reviews by the NRC but are documented in diverse places. The nuclear industry recommended that a report be issued that collects and describes those licensing issues and their resolutions. In response to that recommendation, a study was performed and this report on it was prepared under the Commercial Spent Fuel Management (CSFM) Program, which is sponsored by the U.S. Department of Energy (DOE) and managed by Pacific Northwest Laboratory (PNL). Included in this report are discussions of the issues raised on 1) the 18 topical reports on dry spent fuel storage cask and module designs that have been submitted to the NRC, 2) the three license applications for dry storage of spent fuel at independent spent fuel storage installations (ISFSIs) that have been submitted to the NRC, and 3) the three applications (one of which was later withdrawn) for large-scale storage of consolidated fuel rods in existing spent fuel storage pools at reactors that were submitted to the NRC. For each of the applications submitted, examples of some of the issues (and suggestions for their resolution) are discussed. The issues and their resolutions are also covered in detail in an example in each of the three subject areas: 1) the application for approval of the CASTOR V/21 dry spent fuel storage cask (Project M-37), 2) the license application for an ISFSI for dry storage of spent fuel at Surry (Docket No. 72-2), and 3) the license application for full-scale wet storage of consolidated spent fuel at Millstone-2 (Docket No. 50-336).

2.0 CONCLUSIONS AND RECOMMENDATIONS

The results of the study on licensing issues for dry storage of LWR spent fuel and large-scale storage of consolidated LWR spent fuel rods in existing pools at reactors have led to the conclusions and recommendations shown below in Sections 2.1 and 2.2, respectively.

2.1 CONCLUSIONS

The conclusions from the study of licensing issues associated with applications for a) approval of dry spent fuel storage cask and module designs, b) dry storage of spent fuel in independent spent fuel storage installations (ISFSIs) at reactor sites, and c) large-scale storage of consolidated spent LWR fuel rods in existing pools at reactors are shown below in Sections 2.1.1, 2.1.2, and 2.1.3, respectively.

2.1.1 Conclusions Regarding Licensing Issues Associated with Applications for Approval of Dry Spent Fuel Storage Cask and Module Designs

- Allowance for burnup credit remains an open issue. Several applications (Projects M-43, M-49, M-50, and M-52) have included requests for burnup credit. In 1987, the NRC indicated^(1,2) that the credit-for-burnup issue is being considered on a site-specific basis for licensing-related actions only at this time and that it may remain an open item for future specific vendor/licensee resolution. The NRC Staff's technical concerns relative to allowing credit for burnup in dry storage technology designs are described in a recent paper.⁽³⁾ In that paper, NRC indicates that during the typical process of loading and/or unloading spent fuel into a cask or canister in the spent fuel storage pool, there is the potential for introducing nonborated water, which would result in undermoderation because of reduced water density. Hence, procedures for, and administrative controls on, fuel loading for dry storage must be strictly controlled. The NRC requires that procedures for addressing burnup credit for spent fuel be already in place, or the applicant must provide such procedures and their implementation regarding cask or canister fuel loading and unloading prior to dry storage of spent LWR fuel.
- The NRC has major concern with components (e.g., cask bodies and fuel baskets) that are proposed to be made of nonspecification materials. Several applicants (Projects M-34, M-37, M-40, M-41, and M-42) proposed using nonspecification materials. The NRC requires that such materials must be adequately characterized if

they are to be approved for use. Some applicants (Projects M-34 and M-37) provided the required data; other applicants (Projects M-37, M-40, M-41, and M-42) changed to approved materials.

- Use of air as a storage cover gas (see Project M-46) has not been accepted yet by the NRC. However, the NRC Staff does not reject the contention that continued research on this subject may subsequently result in allowance of air as a cover gas for dry storage of spent LWR fuel.
- A major issue in one application (Project M-42) was the criticality design of the fuel basket, which relies on the insertion of a "poison spider" assembly of rods into a fuel assembly. The NRC questioned whether this approach can satisfy the design criteria of 10 CFR Part 72 for at-reactor site storage, at least in the nonsite-specific context of a topical report. In the case of Project M-42, a significant change in the fuel basket design was made.
- The discovery of cracks in the borated stainless steel fuel basket (Project M-37) was of great interest to the NRC. The cracks were caused by constrained thermal expansion. The problem was solved by incorporating a simple design modification, which involved eliminating fabrication stitch welds on the first few feet of either end of the basket.
- One applicant (Project M-52) expressed interest in obtaining NRC's comments on the applicant's proposal to take credit in the safety analysis for the structural strength of the borated aluminum in the cask's fuel basket. The NRC is currently reviewing the application.⁽²⁾
- Because of the new fee NRC charges for review of topical reports, the applicant under Project M-44 has recently requested the NRC to stop reviewing the application until the applicant has a customer to fund the process.⁽⁴⁾
- An issue in one application (Project M-44) involved the temperatures of the concrete cask. The applicant provided responses and indicated that they believed their answers clarified their use of structural concrete in the cask design and show that this concrete meets the current ACI-349 code.
- In some applications (e.g., Project M-41), the reduced drawings were too small to be legible in some cases. The NRC requested that they be replaced to allow NRC to review them.
- There is a need for thorough editing of applications. In several applications (e.g., Projects M-33, M-36, M-40, and M-43), the NRC noted that there appear to be errors, inconsistencies, and omissions in the topical reports.

- In one application (Project M-46), the NRC indicated that it appeared that some aspects of the design (e.g., load combination analyses) were not fully analyzed and documented in a form that would allow the NRC to continue its review.
- A recent article⁽⁵⁾ indicates that the NRC is proposing a rule change that would permit LWR licensees to store spent fuel in NRC-approved dry storage casks (free-standing metal and concrete type) under a general license that would not require additional site-specific reviews by the NRC. So far, four metal dry storage casks that have been approved by the NRC would come under the general license coverage. The four are CASTOR V/21 (Project M-37), MC-10 (Project M-41), NAC-S/T (Project M-40), and NAC-C28 S/T (Project M-51). According to the NRC, it is likely that concrete modules (NUHOMS^(a)-07P, Project M-39; NUHOMS^(a)-24P, Project M-49) would later be eligible for general license consideration. The NRC also stated that the proposed rule is geared to mobile casks.

2.1.2 Conclusions Regarding Licensing Issues Associated with Applications for ISFSIs for Dry Storage of Spent Fuel at Reactor Sites

- An issue that arose during NRC's review of ISFSI applications (Docket No. 72-4 and Project M-49⁽⁶⁾) centered on the question of whether, and to what extent, the spent fuel storage canisters to be used at the ISFSI will be compatible with the transportation cask ultimately chosen by DOE to transport spent nuclear fuel temporarily stored at reactor sites to permanent repositories. In a response⁽⁷⁾ to that issue, an applicant indicated, given the fact that DOE's design effort for its cask fleet is in its early stages, that no meaningful compatibility review can be conducted at this time. In one case (Project M-49⁽⁶⁾), the utility is offering a cash incentive to the dry storage cask vendor to encourage such direct offsite shipment of dry shielded canisters (DSCs) to foster compatibility by direct offsite disposition of spent fuel to DOE at the termination of storage.
- An issue with one application (Docket No. 72-3) was that the associated Environmental Report (ER) did not fully comply with 10 CFR 72.20.
- An issue with one application (Docket No. 72-4) involved the design basis deceleration levels for the cask drop case. The NRC requested additional information to confirm the levels as the NRC indicated that it is possible that the deceleration levels presented by the storage system vendor could be higher than stated in the topical report (NUH-002, Rev. 1).

(a) NUHOMS is a registered trademark of Nutech Engineers Inc., San Jose, California.

- There is a need to show that an ISFSI at a reactor site operates independently from the reactor. Regarding one application (Docket No. 72-3), the NRC indicated⁽⁸⁾ that the outstanding issues remaining to be closed include final quality assurance commitments and completion of the utility's internal plant review to confirm operations of the ISFSI and the reactor have no significant safety impact on one another, that no changes to the reactor's operating license are needed for the ISFSI, and that the ISFSI operates independently. The utility concluded⁽⁹⁾ that the ISFSI is "independent" as defined in 10 CFR Part 72.
- An array of casks/modules needs to be considered in the application. An issue with the Environmental Report associated with one application (Docket No. 72-3) involved the onsite collective dose commitment to workers constructing additional spent fuel storage modules.⁽¹⁰⁾
- With one application (Docket No. 72-4), an issue raised by the NRC was that the information requested earlier by NRC was not supplied or was not supplied in sufficient detail, and that some calculations that were provided did not appear to correspond to the storage module design.
- In one application (Docket No. 72-2), safeguards information was inadvertently included.⁽¹¹⁾ Later the NRC requested that the applicant completely resubmit the required safeguards plan because the initial application lacked specifics.⁽¹²⁾
- With one application (Docket No. 72-2), an issue concerned license and technical conditions.⁽¹³⁾ The specific areas questioned by the NRC included station health physics procedures, station administrative procedures, station training program, cask handling procedures to prevent a cask from tipping over, and the cask pressure switch monitor.
- An issue involved in one application (Docket No. 72-2) was the need to increase the calculated dose rates shown in the topical safety analysis report to account for the effects of neutron subcritical multiplication.^(14,15)
- With one application (Docket No. 72-2), some of the issues involved seismological, geological, and structural bases for the dry cask ISFSI,^(16,17) and soil liquefaction analysis for the ISFSI.⁽¹⁸⁾ Another issue involved clarification of dose calculations.⁽¹⁹⁾

2.1.3 Conclusions Regarding Licensing Issues Associated with Applications for Large-Scale Storage of Consolidated Spent LWR Fuel in Existing Pools at Reactors

- It can be important to control pool water temperature, to control the fuel decay time, and to submit a cask drop analysis. Acceptance of one application (Maine Yankee) was contingent on resolution of these three open issues:⁽²⁰⁾
 1. The licensee must have procedures stating that during full core discharges and after the first third of the core has been discharged, pool bulk water temperature will be monitored following the insertion of each additional assembly. Should this temperature exceed 154°F these procedures must direct that recently discharged fuel be returned to the reactor vessel until pool bulk water temperature drops to or below 154°F.
 2. A limit must be added to the plant's Technical Specifications that requires that fuel decay at least 120 days from shutdown before it may be consolidated.
 3. The license must be conditioned to preclude lifting a spent fuel shipping cask over the pool until a cask drop analysis is submitted by the licensee and approved by the staff.

These open issues were closed after the utility provided the appropriate responses.⁽²¹⁾

- An issue in one application (Millstone-2) was the proposed requirement that candidate fuel assemblies (for consolidation) must have decayed for at least 5 years.^(22,23) The NRC requested that the applicant provide a justification for the proposed decay time in terms of the consolidation process. Also, the applicant was requested to provide a complete description of the consolidation process and an associated safety analysis.
- There is a need to eliminate inconsistencies and the potential for unnecessary confusion and incorrect perceptions. In connection with one application (Ginna), the utility indicated that in its review of the NRC's Safety Evaluation Report (SER),⁽²⁴⁾ two statements were noted in the NRC Staff's analysis that were not consistent with the amendment or the utility's responses to NRC's requests for additional information. These statements concerned the use of fuel with burnable poisons and the Auxiliary Building crane. The utility requested review and clarification by NRC of the SER because the inconsistencies present the potential for unnecessary confusion and the perception that plant operation may exceed the bounds of the licensing basis.

The utility also noted⁽²⁵⁾ that in the SER, the NRC Staff analysis of canister handling appears to conclude that because the Auxiliary Building crane meets the single failure proof criteria of NUREG-0612 and handling of the canister uses the 5-ton hook of the Auxiliary Building crane, the handling of the canister will satisfy the single failure proof guidelines and is therefore acceptable. The NRC Staff references the SER on the modification to the Auxiliary Building crane dated October 1, 1984. This SER, and the utility's submittals on which this SER was based, never addressed any modification to the crane that would result in a load being transported by the 5-ton hook in a single-failure proof mode. Rather the radiological effects of a canister drop were evaluated and found to be acceptable. The utility has modified the 40-ton hook on the Auxiliary Building crane to be single failure proof to meet other heavy load handling concerns that are separate from use of the 5-ton hook for canister movements.

- With one application (Ginna), two changes to the basis for verifying that the criteria for the pool k_{eff} of less than or equal to 0.95 would be met were not incorporated into the specification that was approved by the NRC's December 1985 Safety Evaluation Report.⁽²⁵⁾ One change involved the use of burnable poisons and the other change involved the effect of storing less than 179 fuel rods per half canister.
- The NRC pointed out that the application from Millstone-2 includes a discussion of a rather extensive test program to be used in conjunction with the analysis of the Spent Fuel Rack System. One issue with that application was the lack of discussion of the testing program and the use of the resulting data in the Safety Analysis Report.⁽²⁶⁾ The NRC requested that the applicant provide the information as the testing portion seemed to be an important step in doing a reliable analysis of the Spent Fuel Rack System. The applicant's response included a statement that the consolidated fuel storage box (CFSB) local stiffness test results showed that the behavior of the CFSB is complicated under a transverse line load and that it exhibits a great variation in stiffness values. At high loads local yielding or buckling of the box walls occurs.
- An issue raised by the NRC in one application (Millstone-2) involved the assumption that no lateral deformation results from a 100,000-lb load impacting the fuel assembly.⁽²⁷⁾ The NRC stated that the assumption is not justified and seems invalid since it has been determined that the guide tubes yield due to the load.
- An issue raised by the NRC was the discussion in the application of a reactivity meter to be used for determination and verification of burnup and the lack of mention of the meter in the Safety Analysis Report's discussion of reactivity determination for storage of spent fuel.⁽²⁸⁾ The applicant's response indicated that the reactivity meter is a state-of-the-art device, which is still under

development. The engineering design reviews are still ongoing. The determination and verification of burnup is currently controlled administratively and the described reactivity meter is not required for determination of reactivity for the storage of spent fuel.

- In its review of one application (Millstone-2), the NRC was concerned that because many fuel rod movements will have been involved in the reconstituted fuel assembly, it is likely activated crud will be released from some fuel pins and adhere to some other pins. The NRC requested that the applicant show how an increase in dose rates from such crud will be precluded when consolidated assemblies are returned to storage in the spent fuel pool. The response from the applicant included three reasons why the potential dose rate consequence from crud redepositing on fuel surfaces is insignificant.
- Fuel rod integrity can be an important consideration. With one application (Millstone-2), the NRC requested information on the method to be used to identify fuel rods that are most likely to rupture.⁽²⁸⁾ The applicant's response indicated that prior to consolidation, all candidate fuel assemblies will be pre-inspected with the consolidation system TV cameras to review the condition of the exterior rows of rods. Any fuel assemblies found to have defects will not be consolidated.
- The NRC indicated in 1987 to one applicant (Millstone-2) that issuance of the requested amendment would be delayed until NRC Staff resolution of BNL 87-05 (Generic Issue 82, "Beyond Design-Basis Accidents in Spent Fuel Pools").

2.2 RECOMMENDATIONS

The recommendations from the study of licensing issues associated with applications for a) dry spent fuel storage cask and module designs, b) dry storage of spent fuel in ISFSIs at reactor sites, and c) large-scale storage of consolidated spent LWR fuel rods in existing pools at reactors are indicated below in Sections 2.2.1, 2.2.2, and 2.2.3, respectively.

2.2.1 Recommendations Concerning Licensing Issues with the Applications for Approval of Dry Spent Fuel Storage Cask and Module Designs

- In its review of several applications (Projects M-34 and M-35), the NRC stated that the topical report should stand as a document that can be referenced in site-specific license applications. The

topical report should be in the format of a safety analysis report and, though not site specific, it should envelope site parameters sufficiently well to encompass an adequate range of sites within the United States.

- The topical report should focus on the single cask itself. For purposes of demonstrating how an array of casks should be addressed in a safety analysis report, NRC suggested (see Projects M-34 and M-35) that a line array and a representative square array of casks be included.
- To avoid the problem of having to provide data fully characterizing a nonspecification material, consider use of alternate materials (see Projects M-37, M-40, M-41, and M-42) that have been accepted by the NRC.
- The selection of a topical report number should not lead to a problem of potential confusion. Such a problem arose in Projects M-39 and M-49 when the topical reports for each project were assigned the same number, NUH-001.
- Referencing and/or verification of computer codes used and of data in calculations would be helpful (see Projects M-33 and M-36).
- In referencing standards and codes, the applicant should be specific as to how these are used and specify the exact revision and date of these (see Projects M-33 and M-36).
- When values are stated in the text of the topical report, justification for their selection should be provided or their bases identified. In two applications (Projects M-33 and M-36), values were given in the text that were not immediately justified or their bases identified. While this oversight may be rectified in a subsequent chapter, where the value is cited again, this still creates a problem for reviewers.
- In commenting on one application (Project M-34), the NRC stressed the need for inclusion of data from tests (as referenced), compatibility of referenced data and analyses, and quality assurance.
- There is a need to furnish legible drawings to the NRC in a timely manner to avoid delays in licensing reviews.
- In comments on several applications (e.g., Projects M-33 and M-36), the NRC stated that thorough editing of topical reports is needed to eliminate confusion, errors, inconsistencies, and omissions.

2.2.2 Recommendations Concerning Licensing Issues with the Applications for ISFSIs for Dry Storage of Spent Fuel at Reactor Sites

- The NRC has stated⁽²⁹⁾ that the Environmental Report (ER) should particularly include a statement of the purpose, or need, for the proposed action and a discussion of alternatives; also, the ER should address the siting evaluation factors contained in 10 CFR 72 (specifically 72.61 through 72.70). The NRC has prepared⁽³⁰⁾ a list of questions [see table in Section 3.2.1.2,c)], which are intended only as a guide, to help applicants prepare an ER.
- The NRC has indicated⁽¹⁰⁾ that a detailed ISFSI site plan drawing and aerial photographs would be most helpful in their reviews.
- The NRC has stated (see Section 3.2.1.1 of Appendix C) that it would be helpful if each supplement to a safety analysis report were 1) dated and identified by a supplement number, 2) accompanied by a dated and numbered supplement index listing the pages to be removed and the pages to be inserted, and 3) identified by number and/or date on each new page containing new material and the newest material on a page be identified by a vertical margin bar.

2.2.3 Recommendations Concerning Licensing Issues with the Applications for Large-Scale Storage of Consolidated Spent LWR Fuel Rods in Existing Pools at Reactors

- In its review of one application (Millstone-2), the NRC recommended that a Technical Specification Surveillance Requirement be incorporated for consolidated fuel to verify the integrity of the fuel and structural elements before movement or placement in the spent fuel pool.⁽³¹⁾ The NRC also requested that the applicant provide information on the proposed method for verifying the integrity of the storage canister after it has been loaded with fuel rods.⁽³¹⁾

3.0 LICENSING ISSUES FOR DRY STORAGE OF LWR SPENT FUEL AND LARGE-SCALE STORAGE OF CONSOLIDATED LWR SPENT FUEL

Dry storage of spent fuel has emerged as a key near-term method for future spent fuel management in the United States. Rod consolidation^(a) is one of the leading candidates for achieving more efficient use of existing space in spent fuel storage pools.⁽³²⁾ Dry storage capacities may also be enhanced by storing consolidated fuel rods. The issues (i.e., comments, questions, or requests from the NRC) and their resolutions (i.e., responses from the applicants) associated with reviews of topical reports on dry storage systems, applications for licensing of dry storage facilities, and applications for large-scale storage of consolidated fuel rods in existing pools are described below.

3.1 TOPICAL REPORTS ON DRY SPENT FUEL STORAGE CASK AND MODULE DESIGNS

Topical reports (i.e., topical safety analysis reports) for dry spent fuel storage cask and module designs that have been submitted by various organizations to the U.S. Nuclear Regulatory Commission (NRC) are chronologically listed in Table 3.1. Planned or proposed topical report submittals that are anticipated are shown in Table 3.2.

3.1.1 General Nuclear Systems, Inc. (GNSI): CASTOR Ic, Project M-34

GNSI is a partnership between Chem-Nuclear Systems, Inc. and Gesellschaft für Nuklear-Service mbH (GNS) of the Federal Republic of Germany. The GNSI dry cask designs involve thick-walled nodular cast iron. The GNSI CASTOR Ic cask has a capacity of 16 BWR spent fuel assemblies in a helium-filled plenum.⁽³³⁻³⁵⁾ GNSI submitted their topical report on CASTOR Ic in January 1982.⁽³⁶⁾ The submittal was docketed under Project M-34 by the NRC in June 1986.⁽³⁵⁾ The chronological listing of the licensing activities is provided in Part 3.1.1 of Appendix A. Examples of some of the

(a) Rod consolidation involves mechanically removing all the fuel rods from the fuel assembly hardware (i.e., the structural components) and placing the fuel rods in a close-packed array in a canister without spacer grids.

TABLE 3.1. Chronological Listing of Topical Reports for Dry Spent Fuel Storage Cask and Module Designs Submitted to and Docketed by the NRC

Organization	Cask/Module Design Designation	Project No.	Comment(s)	Reference No(s).
• General Nuclear Systems, Inc. (GNSI)	Castor Ic	M-34	Modular cast iron cask (holds 16 BWR fuel assemblies); topical report approved; use of this cask in U.S. is no longer expected because of cask's small capacity.	33, 34, 37
• Combustion Engineering, Inc. (C-E)	--	M-35	In May 1983, C-E indicated its plans for proceeding were uncertain. In June 1984, NRC terminated this project review.	38-40
• Mitsubishi Heavy Industries Limited (MHI) ^(a)	REA 2023 (BWR) MSF-IV ^(b)	M-33 M-36	Metal cask; review suspended. Metal cask; review suspended.	34, 41 34, 41
• General Nuclear Systems, Inc. (GNSI)	Castor V/21	M-37	Modular cast iron cask (holds 21 PWR fuel assemblies); topical report approved (basket material can be stainless steel or borated stainless steel); such casks are deployed at Surry.	2, 33, 34, 37, 42-44
• Nutech Engineers, Inc. (NUTECH)	NUHOMS-07P ^(c)	M-39	Concrete module, which contains a stainless steel canister; holds 7 PWR fuel assemblies; topical report approved; such modules are deployed at Robinson-2.	2, 34, 45, 46
• Nuclear Assurance Corporation (NAC)	NAC S/T ^(d) (26 PWR)	M-40	Metal cask; topical report approved.	2, 34
• Westinghouse Electric Corporation (W)	MC-10 (dry cask design)	M-41	Metal cask; topical report approved.	2, 34, 37, 47
• Transnuclear, Inc. (TN)	TN-24P	M-42	Metal cask; topical report approved (basket material is stainless steel) for spent fuel storage in U.S. and spent fuel transport in France.	34, 48-50
• Combustion Engineering, Inc. (C-E)	Dry Cap-P24 and Dry Cap-B60 (dry cask designs)	M-43	Metal cask; topical report being reviewed by the NRC.	2, 34
• FW Energy Applications, Inc. (FWEA)	MVDS	M-46	Modular concrete vault; topical report approved for nitrogen cover gas only.	2, 34, 51
• Nuclear Assurance Corporation (NAC)	Modified S/T	M-51	Metal cask designed for consolidated fuel (28 canisters, with total capacity for fuel rods from up to 56 PWR fuel assemblies); topical report approved.	2, 37, 52
• Nutech Engineers, Inc. (NUTECH)	NUHOMS-24P	M-49	Concrete module, which contains a stainless steel canister; holds 24 PWR fuel assemblies; design relies on allowance for burnup credit; topical report being reviewed by the NRC.	2, 37, 45
• Nuclear Packaging, Inc. (NuPac)	CP-9	M-44	Concrete cask; review suspended.	2, 4, 34

TABLE 3.1. (contd)

Organization	Cask/Module Design Designation	Project No.	Comment(s)	Reference No(s).
• General Nuclear Systems, Inc. (GNSI)	Castor X	M-50	Modular cast iron cask design that relies on allowance for burnup credit; topical report being reviewed by the NRC.	2, 37
• Nuclear Assurance Corporation (NAC)	NAC-31 S/T	M-52	Metal cask designed to rely on burnup credit (capacity: 31 PWR fuel assemblies); topical report being reviewed by the NRC; NAC requested Project M-54 be reviewed before Project M-52.	2, 37, 53
	S/T	M-54	Metal cask designed for 28 intact PWR fuel assemblies; topical report being reviewed by the NRC; a cask is already at a customer's site.	53
• Pacific Sierra Nuclear Associates (PSN)	Ventilated Storage Cask (VSC)	M-53	Concrete cask; topical report being reviewed by the NRC.	54, 55

- (a) Since 1984, MHI has purchased the rights to the Ridihalgh, Eggers, and Associates (REA) cask designs.
 (b) Originally named REA 2023 (PWR).
 (c) NUTECH Horizontal Modular Systems (NUHOMS) design.
 (d) Storage/transport (S/T).

TABLE 3.2. Planned or Proposed Submittals of Topical Reports for Dry Spent Fuel Storage Cask and Module Designs

Organization	Cask/Module Design Designation	Project No.	Comment(s)	Reference No(s).
• B&W Fuel Company (BWFC)	CONSTAR-28P	--	BWFC plans to submit the topical report to NRC in late spring or early summer of 1990. The dry storage cask consists of a steel inner shell and a concrete outer shell. Cask capacity: 28 intact PWR fuel assemblies. The cask is to be dry-loaded using a shielded transfer cask (capacity: 4 PWR or 9 BWR fuel assemblies at a time) to transport the fuel from the spent fuel pool.	56
• Nuclear Assurance Corporation (NAC)	Modified S/T ^(a)	--	NAC plans to modify its dry storage S/T cask so that it can also be used to transport spent fuel. Under current schedules, NAC will be the first U.S. vendor to submit a topical report to the NRC for a dual-purpose cask.	57

- (a) Storage/transport (S/T)

the issues involved (and suggestions for their resolution) are listed in Table 3.3 (placed at the end of Section 3.0 because of its length). Initially, the NRC indicated⁽³⁶⁾ that the submitted topical report did not accomplish the objective of standing as a document that can be referenced in site-specific license applications; the topical report should be in the form of a safety analysis report and, though not site specific, it should envelope site parameters sufficiently well to encompass an adequate range of sites within the United States. The NRC suggested that a revised GNS topical report focus on the single cask itself. For purposes of demonstrating how an array of casks should be addressed in a safety analysis report, NRC suggested that a line array and a representative square array of casks be included.⁽³⁶⁾ The NRC also suggested that the report should consider only an at-reactor site ISFSI scenario.⁽³⁶⁾ In June 1983, inclusion of data from tests, as referenced, compatibility of referenced data and analyses, and quality assurance in revising the topical report were stressed by NRC.⁽⁵⁸⁾ In December 1983, NRC provided detailed comments and some general comments.⁽⁵⁹⁾ The general comments involved material (nodular cast iron) qualification and structural soundness of the cask design and were as follows: 1) the nodular cast iron material used for the cask should be adequately characterized (see detailed comments on Section 4, "Installation Design"), 2) flaw detectability levels and procedures for flaw detection should be detailed and specified, 3) adequate stress analysis should be performed, and 4) assurance of quality control in the casting and fabrication of the cask should be documented in this topical report. The topical report for that cask was approved by the NRC; however, use of the cask is no longer anticipated because of its small capacity.⁽³⁴⁾

3.1.2 Combustion Engineering, Inc. (C-E): Project M-35

C-E submitted a topical safety analysis report entitled "Dry Storage Cask for an Independent Spent Fuel Storage Installation," CEND-270, that was docketed by the NRC on November 24, 1982, under Project M-35.⁽³⁸⁾ Chronological listing of the licensing activities is provided in Part 3.1.2 of

Appendix A. Examples of some of the issues involved (and suggestions for their resolution) are listed in Table 3.4 (placed at the end of Section 3.0 because of its length). Initially, the NRC indicated⁽³⁸⁾ that the submitted topical report did not accomplish the objective of standing as a document that can be referenced in site-specific license applications; the topical report should be in the format of a safety analysis report and, though not site specific, it should envelope site parameters sufficiently well to encompass an adequate range of sites within the United States. The NRC also suggested, for purposes of demonstrating how an array of casks should be addressed in a safety analysis report, that a line array and a representative square array of casks be included.⁽³⁸⁾ The NRC also suggested that the topical report address only an at-reactor site ISFSI scenario.⁽³⁸⁾ In May 1983, C-E indicated that its plans for developing the cask design were uncertain and that no schedule had been developed for revising the topical report.⁽⁶⁰⁾ In June 1984, the NRC informed C-E that in view of the elapsed time with no further action, the NRC had terminated this project review from its record system.⁽⁴⁰⁾

3.1.3 Mitsubishi Heavy Industries Limited (MHI): MSF-IV [formerly REA 2023 (PWR)], Project M-33

Ridihalgh, Eggers and Associates (REA) submitted a topical safety analysis report entitled "Topical Report for the REA 2023 Dry Storage Cask for BWR Spent Fuel" (Revision 0) to the NRC in April 1983.⁽⁶¹⁾ The topical report was docketed by the NRC under Project M-33 on May 5, 1983. Chronological listing of the licensing activities is provided in Part 3.1.3 of Appendix A. Examples of some of the issues involved (and suggestions for their resolution) are listed in Table 3.5 (placed at the end of Section 3.0 because of its length). In July 1983, the NRC indicated⁽⁶²⁾ that their initial evaluation found the topical report to be generally satisfactory and provided detailed comments. In October 1983, NRC transmitted⁽⁶³⁾ additional detailed comments and a few general comments. Those general comments were as follows:

1. There appear to be errors and omissions in criticality, thermal and structural analyses presented in this report. While these may not affect the basic integrity of this cask design, for a successful completion of this safety review these must be corrected.
2. To avoid confusion and eliminate error in analysis, a thorough editing of the report, for discrepancies in parameters given for the spent fuel and the cask design and for misreferenced or missing information, is needed. It appears that this report has not been verified in accordance with Appendix B of 10 CFR Part 50. The information in this submittal is used in audits by NRC inspectors. In all cases the parameters specified in this report form the bases for future audits of the design, fabrication, and operation of this type of cask.
3. Referencing and/or verification of computer codes used and of data in calculations would be helpful.
4. In referencing standards and codes used, please be specific as to how these are used and specify the exact revision and date of these.
5. In some instances values are given in the text that are not immediately justified or their bases identified. While this oversight may be rectified in a subsequent chapter, where the value is cited again, this still creates a problem for reviewers.

The rights to the REA cask designs for BWR and PWR fuel and the associated REA topical report submittals were purchased (effective date was April 5, 1985) by MHI.^(34,64) MHI indicated that it was initially seeking approval of the PWR cask design (Project M-36).⁽³⁴⁾

3.1.4 Mitsubishi Heavy Industries Limited (MHI): REA 2023 (BWR), Project M-36

REA submitted a topical safety analysis report entitled "Topical Report for the REA 2023 Dry Storage Cask for PWR Spent Fuel," which was docketed by the NRC under Project M-36 on June 10, 1983.⁽⁶⁰⁾ Chronological listing of the licensing activities is provided in Part 3.1.4 of Appendix A. Shifting of the rights to the REA cask designs for BWR and PWR fuel from REA to MHI is described in Section 3.1.3. Examples of some of the issues involved (and suggestions for their resolution) are listed in Table 3.6 (placed at the end of Section 3.0 because of its length). In November 1983, NRC transmitted⁽⁶⁰⁾ detailed comments and few general comments on the topical report. The general comments (they are identical to the ones NRC sent to REA under Project M-33) were as follows:

1. There appear to be errors and omissions in criticality, thermal and structural analyses presented in this report. While these may not affect the basic integrity of this cask design, for a successful completion of this safety review these must be corrected.
2. To avoid confusion and eliminate error in analysis, a thorough editing of the report, for discrepancies in parameters given for the spent fuel and the cask design and for misreferenced or missing information, is needed. It appears that this report has not been verified in accordance with Appendix B of 10 CFR Part 50. The information in this submittal is used in audits by NRC inspectors. In all cases the parameters specified in this report form the bases for future audits of the design, fabrication, and operation of this type of cask.
3. Referencing and/or verification of computer codes used and of data in calculations would be helpful.
4. In referencing standards and codes used, please be specific as to how these are used and specify the exact revision and date of these.
5. In some instances values are given in the text that are not immediately justified or their bases identified. While this oversight may be rectified in a subsequent chapter, where the value is cited again, this still creates a problem for reviewers.

MHI indicated to the NRC that it was initially seeking approval for the PWR (REA 2023) cask design; however, MHI renamed it MSF IV and substantially changed the cask basket and the neutron shield part of the cask.⁽³⁴⁾ In May 1986, MHI informed the NRC that they were forced (because of budgetary considerations) to temporarily suspend work on Project M-36.⁽⁴¹⁾

3.1.5 General Nuclear Systems, Inc. (GNSI): CASTOR V/21, Project M-37

The "Topical Safety Analysis Report for the CASTOR V Cask Independent Spent Fuel Storage Installation (Dry Storage)" submitted to the NRC on December 16, 1983, by GNSI was docketed by the NRC on January 23, 1984, under Project M-37.⁽⁶⁵⁾ On February 15, 1985, GNSI indicated that the designation for the CASTOR V (Vb) Cask had been changed to CASTOR V/21.⁽⁶⁶⁾

Chronological listing of the licensing activities is provided in Part 3.1.5 of Appendix A. Examples of some of the issues involved (and suggestions for their resolution) are listed in Table 3.7 (placed at the end of Section 3.0 because of its length). Some of the issues (comments, questions, or requests from the NRC) and the detailed responses from the applicant are listed in

Appendix B. In reviewing the topical report, the NRC transmitted to the applicant some general comments in addition to the detailed comments.⁽⁶⁵⁾ The four general comments involved material (nodular cast iron) qualification and structural soundness of the cask design and were as follows: 1) the nodular cast iron material used for the cask should be adequately characterized (see detailed comments on Section 4, "Installation Design"), 2) flaw detectability levels and procedures for flaw detection should be detailed and specified, 3) adequate stress analysis should be performed, and 4) assurance of quality control in the casting and fabrication of the cask should be documented in this topical report. Included among the specific concerns of the NRC with the GNSI cask designs were 1) the characterization of the nodular cast iron as an acceptable material for use in a dry storage cask, 2) the maximum temperature of the spent fuel cladding in the cask, and 3) the ability of fuel to be stored dry under cask conditions for at least 20 years without sustaining any significant deterioration.⁽³⁴⁾ The discovery of cracks in the borated stainless steel fuel basket of the CASTOR V/21 cask^(44,67,68) at the Idaho National Engineering Laboratory (INEL) on September 30, 1985, was of great interest to the NRC (see Table 3.7 and Appendix B). In response to NRC's concern, GNSI provided results of the investigations of the basket cracking and evaluations and analyses to validate the use of borated stainless steel in the fabrication of the welded basket for the CASTOR V/21 cask.⁽⁶⁹⁻⁷²⁾ The cause of the cracking was determined to be constrained thermal expansion.⁽⁷¹⁾ The problem was solved by incorporating a simple design modification, which consisted of eliminating fabrication stitch welds on the first few feet of either end of the fuel basket. GNSI also proposed using an all stainless steel fuel basket. The NRC approved the CASTOR V/21 cask design for use with either an all stainless steel basket or a borated stainless steel basket.⁽⁴²⁻⁴⁴⁾ GNSI CASTOR V/21 casks are deployed at the Surry ISFSI. In June and August 1987, NRC was notified^(73,74) by GNSI that some material used in fabricating two cask baskets (for CASTOR V/21 Casks 008 and 009) contained a boron content that was not within the tolerance established in the material specification as approved by the NRC. The NRC independently reviewed this issue and verified that such use did not compromise safety.⁽⁷⁴⁾

3.1.6 Nutech Engineers Inc. (NUTECH):^(a) NUHOMS-07P, Project M-39

The dry storage concrete module independent spent fuel storage installation submittal, "Topical Report for the NUTECH Horizontal Modular Storage System for Irradiated Nuclear Fuel" (NUH-001), was docketed by the NRC on December 19, 1984, under Project M-39.⁽⁷⁵⁾ In the NUTECH Horizontal Modular Storage System (NUHOMS), spent fuel assemblies are sealed inside stainless steel canisters that are then placed in concrete modules.⁽⁷⁶⁾ The original module that was approved by the NRC has a capacity of seven PWR fuel assemblies that had an initial enrichment of 3.5 wt% uranium-235, have a maximum burnup of 33,000 MWd/MTU, and have decayed for five years. Chronological listing of the licensing activities is provided in Part 3.1.6 of Appendix A. Examples of some of the issues involved (and suggestions for their resolution) are listed in Table 3.8 (placed at end of Section 3.0 because of its length). Initially, the NRC requested additional information (e.g., design details, descriptions of computer models, and computer calculations) in three design categories: 1) thermal-hydraulics (6 issues), 2) criticality and shielding (5 issues), and 3) structural (13 issues).⁽⁷⁷⁾ Later, the NRC indicated that NUTECH would need to place effort in the areas of shielding penetration estimates and criticality calculations.⁽⁷⁸⁾ The NRC said that a significant effort by NUTECH was needed on criticality and shielding analysis.⁽⁷⁹⁾ The topical report was accepted by the NRC as a reference on March 28, 1986.⁽⁴⁶⁾ A modified form of the NUHOMS design has been authorized for use at the ISFSI at Robinson-2. A NUHOMS design is also intended for use at the ISFSI at Oconee.⁽⁸⁰⁾

(a) NUTECH was purchased by Pacific Nuclear Systems, Inc. in the fourth quarter of 1988.⁽⁸¹⁾ NUTECH is a wholly owned subsidiary of Pacific Nuclear Systems, Inc.; Pacific Sierra Nuclear Associates (PSN) is jointly owned by Pacific Nuclear Systems, Inc. and Sierra Nuclear.⁽⁴⁾

3.1.7 Nuclear Assurance Corporation (NAC): S/T (26 PWR), Project M-40

NAC submitted its "Topical Safety Analysis Report for the NAC Storage/Transport Cask for Use at an Independent Spent Fuel Storage Installation" to the NRC on December 13, 1984.⁽⁸²⁾ The topical report was docketed by the NRC under Project M-40 on December 19, 1984.⁽⁸³⁾ The cask has a stainless steel and lead body with a solid neutron shield. The cask capacity is 26 PWR spent fuel assemblies, which must have decayed for five years, have a maximum initial enrichment of 3.3 wt% uranium-235, and have a maximum average burnup of 35,000 MWd/MTU.^(33,37) Chronological listing of the licensing activities is provided in Part 3.1.7 of Appendix A. Examples of some of the issues involved (and suggestions for their resolution) are listed in Table 3.9 (placed at the end of Section 3.0 because of its length). During the course of the review of the topical report, 16 questions^(84,85) related to the quality assurance program were directed by NRC to NAC. At meetings of the NRC and NAC, the discussions involved thermal, criticality, and structural analysis questions and cask tipover and drop analyses.^(86,87) Also discussed at the meetings were assumptions used in the topical report and fire and explosion criteria for an at-reactor site case. The NRC indicated to NAC that there were inconsistencies and omissions in the revised topical report and that close attention should be paid to technical editing in revising the report.^(88,89)

One major issue involved in the NAC design was the use of an aluminum alloy as a storage cask basket material (the material must be qualified for such use in the cask design).⁽⁹⁰⁾ A different cask basket design was submitted in NAC's revised topical report; the design will accommodate 26 PWR spent fuel assemblies without reliance on burnup credit with respect to fuel basket criticality design.⁽⁹¹⁾ Changes to impact limit design and shielding material use were also being considered. The topical report for the NAC cask design has been approved by the NRC for referencing in site-specific license applications for dry storage at reactor sites under 10 CFR Part 72.⁽²⁾

3.1.8 Westinghouse Electric Corporation: MC-10, Project M-41

The topical report entitled "Topical Safety Analysis Report for the Westinghouse MC-10 Cask for an Independent Spent Fuel Storage Installation

(Dry Storage)" was docketed by the NRC on February 13, 1985, under Project M-41.⁽⁹²⁾ The Westinghouse MC-10 cask design involves a forged steel body with a solid neutron shield.⁽³⁷⁾ The cask's capacity is 24 PWR spent fuel assemblies, with a maximum initial enrichment of 3.7 wt% uranium-235, a maximum average burnup of 35,000 MWd/MTU, and a decay period of 10 years.⁽³⁷⁾ Chronological listing of the licensing activities is provided in Part 3.1.8 of Appendix A. Examples of some of the issues involved (and suggestions for their resolution) are listed in Table 3.10 (placed at end of Section 3.0 because of its length). In reviewing the cask materials, the NRC and Lawrence Livermore National Laboratory (LLNL) raised major concern relative to the basket material.⁽⁹³⁾ Specifically, the NRC's major concern was that the basket material was not an ASME Code Class I material. NRC recommended that Westinghouse change the basket material (aluminum) to an ASME Code Class I material or provide further evaluation to demonstrate the acceptability of the aluminum basket. Westinghouse resolved the issue by changing the basket material to stainless steel and indicating that several engineering reanalyses would be performed to reflect the new material.^(93,94) The NRC also indicated to Westinghouse that existing drawings continue to be too reduced in size to be legible in some cases and should be replaced to allow review of them.⁽⁹⁴⁾ The NRC completed its safety review of the topical report for the MC-10 metal cask and approved the topical report as a reference in 10 CFR Part 72 site-specific license applications.⁽³⁷⁾ In September 1987, the NRC issued⁽⁴⁷⁾ the letter of approval and the safety evaluation report.

3.1.9 Transnuclear, Inc. (TN): TN-24P, Project M-42

On August 30, 1985, TN submitted their TN-24 Dry Storage Cask Topical Report.⁽⁹⁵⁾ The report was docketed by the NRC under Project M-42. The cask design has a ferritic steel body, a solid resin neutron shield, and a capacity of 24 PWR spent fuel assemblies. In the topical report it is indicated that the TN cask employs borated stainless steel (a nonspecification material) as a cask basket material. Chronological listing of the licensing activities is provided in Part 3.1.9 of Appendix A. Examples of some of the issues involved (and suggestions for their resolution) are listed in Table 3.11 (placed at the end of Section 3.0 because of its length). The NRC

indicated that a satisfactory specification must be provided to assure that borated stainless steel is qualified for use in a storage cask basket.⁽³³⁾ Another major issue raised by the NRC involved the criticality design of the fuel basket, which relies on insertion of a "poison spider assembly" of rods into a fuel assembly.⁽⁹⁶⁾ The NRC questioned whether this approach can satisfy the design criteria of 10 CFR Part 72 for at-reactor site storage, at least in the nonsite-specific context of a topical report. At meetings of the NRC and TN, some key items discussed (in addition to the poison spider assembly) included design drawings, missiles, seismic loading, cask drops and tipping, basket loading criteria, containment boundary definition, basket material characterization, and cobalt-60 source characterization.⁽⁹⁶⁾ A revised topical report was submitted to the NRC in February 1988.^(2,48,97) A significant change in the fuel basket design has been made.⁽⁹⁸⁾ The topical report was approved by the NRC in April 1989.^(48,49) The dual-purpose cask is approved for spent fuel storage in the United States and for spent fuel transport in France.⁽⁵⁰⁾

3.1.10 Combustion Engineering, Inc. (C-E): Dry Cap-P24 and -B60, Project M-43

On December 19, 1985, C-E submitted its "Topical Safety Analysis Report for the Combustion Engineering Dry-Cap Cask for an Independent Spent Fuel Storage Installation (Dry Storage)" to the NRC; the NRC docketed the report under Project M-43.⁽⁹⁹⁾ The C-E cask design (Dry Cap) has a ferritic steel body, a solid resin neutron shield, and a capacity of 60 BWR spent fuel assemblies or 24 PWR spent fuel assemblies. Chronological listing of the licensing activities is provided in Part 3.1.10 of Appendix A. Examples of some of the issues involved (and suggestions for their resolution) are listed in Table 3.12 (placed at the end of Section 3.0 because of its length). One general comment made by the NRC was that there were numerous references in the topical report to 10 CFR Part 71 requirements--NRC indicated it is reviewing the report under Part 72, not Part 71.⁽¹⁰⁰⁾ Also, NRC indicated⁽¹⁾ to C-E that a number of discrepancies and omissions were noted in the topical report. The C-E basket designs include allowance for burnup credit. In June 1987, the NRC indicated^(1,2) that the credit-for-burnup matter is being considered on a site-specific basis for licensing-related actions only at

this time; it may remain an open item in this safety review for future specific vendor/licensee resolution. In December 1987, the NRC indicated⁽¹⁰¹⁾ that the NRC staff's technical concerns relative to credit-for-burnup application to dry storage are expressed in a recent paper.⁽³⁾ C-E's revised topical report is currently being reviewed by the NRC.⁽²⁾

3.1.11 FW Energy Application, Inc. (FWEA): MVDS, Project M-46

The topical report^(102,103) by FWEA (a Foster Wheeler Company) for dry modular vault storage of spent fuel in air and/or nitrogen was submitted to the NRC on September 15, 1986. The "Topical Report for the Foster Wheeler Modular Vault Dry Store (M.V.D.S.) for Irradiated Nuclear Fuel," FWEA-08-27164 Revision 0, was docketed by the NRC under Project M-46 on September 30, 1986.⁽¹⁰⁴⁾ The modular concrete vault design can accommodate BWR and/or PWR spent fuel assemblies; a single module has a capacity of 150 BWR or 83 PWR spent fuel assemblies.⁽³⁷⁾ The FWEA design assumes two modules and the capability of expanding to five modules.⁽³⁷⁾ Chronological listing of the licensing activities is provided in Part 3.1.11 of Appendix A. Examples of some of the issues involved (and suggestions for their resolution) are listed in Table 3.13 (placed at the end of Section 3.0 because of its length). In reviewing the topical report, the NRC indicated⁽¹⁰⁵⁾ to FWEA that it appeared that some aspects were not fully analyzed and documented [e.g., items not provided in the structural design were load combination analyses for the bridge and trolley for the fuel handling machine and for the bridge, trolley, and crane in the transfer bay (the latter items may be site specific, and if so, it needs to be stated)] in a form that would allow NRC to continue its review. The NRC also included comments about lacking information on cover gas, ventilation, and depression system designs and also indicated the need for specificity in addressing the potential for fire and explosion.⁽¹⁰⁵⁾ In August 1987, the NRC indicated that it was still evaluating the proposed use of air as a cover gas in conjunction with or in place of nitrogen.⁽¹⁰⁶⁾ In September 1987, storage of spent fuel in air remained an unresolved issue.⁽¹⁰⁷⁾ In March 1988, the NRC issued⁽⁵¹⁾ its letter of approval for Revision 1 of FWEA's topical report; however, the NRC staff did not accept the use of air as a storage cover gas for the MVDS at that time, but did not reject the contention that continued research in this area may subsequently

result in allowance of such use. The NRC indicated that the MVDS design is relatively complex in comparison to other modular, passive dry spent fuel storage designs that the NRC staff has evaluated.⁽⁵¹⁾

3.1.12 Nuclear Assurance Corporation (NAC): Modified S/T (for 28 canisters of consolidated fuel rods), Project M-51

NAC submitted its "Topical Safety Analysis Report for the NAC Storage/Transport Cask Containing Consolidated Fuel for Use at an Independent Spent Fuel Storage Installation"⁽¹⁰⁸⁾ to the NRC; the topical report was docketed by the NRC on February 24, 1988, under Project M-51.⁽¹⁰⁹⁾ The proposed stainless steel and lead cask design was a modified version of NAC's storage/transport (S/T) cask design; the basket holds 28 canisters of consolidated fuel rods from up to 56 PWR fuel assemblies.⁽³⁷⁾ This design was associated with a 10 CFR Part 72 license amendment application planned by Virginia Power for its Surry Power Station ISFSI.⁽³⁷⁾ Chronological listing of the licensing activities is provided in Part 3.12 of Appendix A. Examples of some of the licensing issues involved (and suggestions for their resolution) are listed in Table 3.14 (placed at the end of Section 3.0 because of its length). The NRC staff met with NAC in January 1988 to discuss the submittal of this topical report; technical issues involving criticality analysis, thermal analysis, shielding, and dose rates were included in the discussion.⁽¹¹⁰⁾ Questions raised by Lawrence Livermore National Laboratory (LLNL) in July-September 1988 involved the neutron shield material (BORAL^{™(a)}), the fuel basket, the revised lifting trunnion analysis, and the annual dose rates from a single cask.⁽¹¹¹⁻¹¹³⁾ On September 29, 1988, the NRC issued a letter⁽⁵²⁾ stating its acceptance of Revision 1 of NAC topical report for the cask design as a reference in 10 CFR Part 72 site-specific license applications.

(a) BORAL is a trademark of AAR Brooks & Perkins, Livonia, Michigan.

3.1.13 Nutech Engineers, Inc. (NUTECH): NUHOMS-24P, Project M-49

On February 26, 1988, NUTECH^(a) submitted the "Topical Report for the NUTECH Horizontal Modular Storage System for Irradiated Nuclear Fuel" (NUH-001, Revision 2, February 1988).⁽¹¹⁴⁾ The topical report was docketed by the NRC under Project M-49. This report describes the design features and safety analyses of the NUHOMS-24P design, which is similar to the earlier system (i.e., NUHOMS-07P under Project M-39) that has been reviewed and approved by the NRC. The principal changes from the previously approved design are:

1. An increase in the canister and storage module capacity to 24 PWR assemblies (the assemblies are to have decayed for 10 years).
2. The utilization of credit-for-burnup assumptions in the design of the canister and in the associated criticality analyses.
3. The addition of design details and safety analyses for an on-site transfer cask used to safely transport the fuel canisters from the fuel building to the horizontal storage module.

Chronological listing of the licensing activities is provided in Part 3.1.13 of Appendix A. Examples of some of the licensing issues involved (and suggestions for their resolution) are listed in Table 3.15 (placed at the end of Section 3.0 because of its length). In March 1988, the NRC indicated⁽¹¹⁵⁾ that a serious problem of potential confusion was raised by NUTECH's use of the same report number, NUH-001, for the topical report under Project M-39 (NUHOMS-07P design) and under Project M-49 (NUHOMS-24P design). The NRC requested that NUTECH resolve the problem by re-examining the report number used and revise it. NUTECH responded⁽¹¹⁶⁾ in March 1988 to the NRC's request and redesignated the topical report for the NUHOMS-24P design as NUH-002. In October 1988, the NRC requested information on 12 items in connection with their review of Revision 1 of NUH-002. Three of the 12 comments/questions were as follows:

(a) NUTECH was purchased by Pacific Nuclear Systems, Inc. in the fourth quarter of 1988.⁽⁸¹⁾ NUTECH is a wholly owned subsidiary of Pacific Nuclear System, Inc.; Pacific Sierra Nuclear Associates (PSN) is jointly owned by Pacific Nuclear Systems, Inc. and Sierra Nuclear.⁽⁴⁾

1. The assurance of nuclear criticality safety of the DSC(a) design depends in part on the burnup of the fuel. A relationship between irradiated fuel reactivity and equivalent fresh fuel enrichment was determined through a series of detailed criticality calculations that are referenced in the topical report. Although there is no doubt that the reactivity of irradiated fuel is reduced from that of unirradiated fuel, there are a number of questions regarding the accuracy with which the reactivity of irradiated fuel can be estimated. An alternative to providing assurance of nuclear criticality safety, which avoids these questions, is to consider loading the DSC in a high neutron absorbing solution and to base the criticality safety analysis on fresh fuel enrichment. If nuclear criticality safety can be assured during loading of fresh fuel into the DSC and subsequent drying operations, and if assurance can be provided to preclude reflooding of the DSC, there appears to be a basis for accepting the design.

Therefore, what is the boron concentration required to provide assurance of nuclear criticality safety during the loading of unirradiated 4-wt% enriched fuel into the DSC? Can you identify other design or operational issues that preclude this option? Describe procedures that will assure maintenance of the required boron concentration in solution in the DSC during fuel loading and unloading.

2. There are several questions or uncertainties centering on the variation in burnup along the length of the irradiated fuel (axial burnup dependence). For a given irradiated fuel assembly, how accurately is the isotopic composition of fuel known? What are the uncertainties in fuel composition, and how do these uncertainties translate to uncertainties in reactivity of a finite array of irradiated fuel assemblies? If the isotopic composition of each fuel assembly is known with precision, what errors result from the approximate models of the axial variation in isotopic composition? Can it be demonstrated that the model selected is conservative (in the sense of reactivity)?

Finally, there is the question regarding the availability or applicability of benchmarks for reactivity calculations with irradiated fuel. Please defend that the benchmarks used for validation are representative of the irradiated fuel assemblies to be placed in the DSC, and that the method bias determined from these benchmarks can be appropriately used in the criticality safety analysis.

3. The statement given on the top of page 1.3-3 of the topical report indicates that the maximum temperature of 570°C during short-term operational and postulated accident conditions will not affect fuel cladding integrity. The references cited indicate that cladding failure was not found. However, cladding creep can and does occur at this temperature. Please provide a response that confirms that cladding creep will not adversely affect cladding integrity.

(a) Dry Shielded Canister (DSC).

The NUHOMS-24P design is currently being reviewed by the NRC in association with Duke Power Company's license application for an ISFSI at its Oconee Station site.⁽²⁾ Baltimore Gas and Electric (BG&E) also has selected the NUHOMS technology (specifically the NUHOMS-24P design) for dry concrete module storage at its Calvert Cliffs Station.^(2,6,117) A site license application is currently being prepared by BG&E and is expected to be submitted to the NRC within a few months.⁽⁴⁸⁾

3.1.14 Nuclear Packaging, Inc. (NuPac): CP-9, Project M-44

In November 1987, NuPac submitted to the NRC the "Topical Safety Analysis Report for the NuPac CP-9 Concrete Storage Cask (TP-08)."⁽¹¹⁸⁾ The NRC docketed the topical report on March 23, 1988, under Project M-44. The capacity of the cask is nine PWR fuel assemblies and the cask may be loaded dry.⁽³⁷⁾ Chronological listing of the licensing activities is provided in Part 3.1.14 of Appendix A. Examples of some of the licensing issues involved (and suggestions for their resolution) are listed in Table 3.16 (placed at the end of Section 3.0 because of its length). One of the issues concerned NuPac's drop testing program with scale models of the CP-9 cask design. On the report submitted by NuPac on drop testing program, the NRC⁽¹¹⁹⁾ had the following six comments:

1. The report appears to reflect an appropriate understanding of scale model design and test design suited to experiments with subscale reinforced concrete structures.
2. The report, however, provides insufficient information on the actual prototype and model designs, concrete design, and potential handling scenarios to be replicated by test (i.e., elements of your test program) to support concurrence in the approach.
3. The role(s) and relationship of the model testing to your analytical model or code should be explained. Also, explain these with respect to your design analysis, design verification, and/or validation of proof against accidental impact. In conclusion, the objectives of the test program are not yet clear and must be carefully developed.
4. The report does not address the thermal environment, with the exception of describing potential use of a deliberately weak concrete. The analysis of thermal effects on the potential structural response and design suitability of the actual cask will be closely examined in the topical report. It should be noted that the NRC requires a high degree

of proof prior to accepting designs exceeding the limits of the appropriate codes, regulations, standards (such as ACI 349-75). The potential thermal environment and temperature-provided metamorphosis that may occur with your design may require such proof of design suitability.

5. Be aware that additional potentially critical impact events may be postulated on the basis of the actual design and proposed usage scenarios, when developed.
6. The acceptance criteria indicated should reflect both comprehensive review of potential failure modes corresponding to the hypothesized accident, and adequate measurement and/or tests to verify meeting the criteria.

Another of the issues involved the concrete temperatures of the CP-9 storage cask. Pacific Sierra Nuclear Associates (PSN) provided⁽¹²⁰⁾ responses to NRC's questions on the CP-9 storage cask concrete temperatures and indicated that PSN believed that answers clarify their use of structural concrete in the cask design and show that this concrete meets the current ACI-349 code. It was reported⁽²⁾ in January 1989 that the NRC is continuing its review of NuPac's concrete cask design model CP-9. Because of the new fee NRC charges for review of topical reports, the applicant has recently asked NRC to stop reviewing the CP-9 concrete cask until the company has a customer to fund the process.⁽⁴⁾

3.1.15 General Nuclear Systems, Inc. (GNSI): CASTOR X, Project M-50

On July 1, 1988, GNSI submitted⁽¹²¹⁾ its "Topical Safety Analysis Report for the CASTOR X Cask for an Independent Spent Fuel Storage Installation (Dry Storage)" to the NRC. The topical report was docketed by the NRC under Project M-50. The design of the CASTOR X cask is very similar to the CASTOR V/21 (Project M-37) that has been previously approved by the NRC.⁽¹²¹⁾ The major difference is the assumption of increased fuel decay time, which leads to increased capacity of fuel assemblies stored. The nodular cast iron cask is designed to store 10-year decayed, intact PWR spent fuel assemblies either with (in a 33-assembly fuel basket) or without (in a 28-assembly fuel basket) allowance for fuel burnup credit in criticality design.^(2,122) Chronological listing of the licensing activities involved is provided in Part 3.1.15 of Appendix A. The NRC is currently reviewing the topical report.⁽²⁾ The topical report was submitted for initial use at Virginia Power's Surry

Nuclear Station. Virginia Power intends to obtain an amendment to their license (SNM-2501) for burnup credit. NRC indicated⁽²⁾ that they expected Virginia Power's amendment application in December 1988.

3.1.16 Nuclear Assurance Corporation (NAC): NAC-31 S/T, Project M-52

On July 20, 1988, NAC submitted⁽¹²³⁾ its "Topical Safety Analysis Report for the NAC S/T Cask Containing 31 Intact Fuel Assemblies" to the NRC. The topical report was docketed by the NRC under Project M-52. The capacity of the borated aluminum basket is 31 Westinghouse PWR spent fuel assemblies; the stainless steel and lead cask is designed to rely on allowance for burnup credit. Maximum initial enrichment is limited to 1.9 wt% uranium-235 for storage of startup core fuel.⁽¹²⁴⁾ Earlier (March 1988), NRC met with NAC to discuss structural aspects of the basket design, as those aspects may affect cask impact limiter design.⁽¹²⁵⁾ Chronological listing of the licensing activities is provided in Part 3.1.16 of Appendix A. NAC expressed interest in obtaining NRC's review comments regarding taking credit in the safety analysis for the structural strength of the borated aluminum in the fuel basket of this cask.⁽⁵³⁾ The NRC is currently reviewing this topical report;⁽²⁾ however, NAC requested⁽⁵³⁾ that the Project M-54 application be reviewed prior to the Project M-52 application (see Section 3.1.17, below).

3.1.17 Nuclear Assurance Corporation (NAC): S/T (for 28 intact PWR fuel assemblies), Project M-54

NAC submitted⁽⁵³⁾ its "Topical Safety Analysis Report for the NAC Storage/Transport Cask Containing 28 Intact Fuel Assemblies for Use at an Independent Spent Fuel Storage Installation" to the NRC on December 14, 1988. The topical report was docketed by the NRC under Project M-54. The same cask design was previously submitted for storage of consolidated fuel (Project M-51). Under Project M-54, the NAC S/T cask with a basket with 28 positions will be used for intact fuel assemblies rather than consolidated fuel rods. Initial fuel assembly enrichment is 1.9 wt% uranium-235 for startup core use; however, fuel burnup is a maximum of 45,000 MWd/MTU, with a fuel decay of at least 10 years (should burnup credit become available in design analysis, the cask criticality analysis could be performed for a higher initial enrichment). Chronological listing of the licensing activities is provided in

Part 3.1.17 of Appendix A. Virginia Power owns a cask of this type. NAC requested⁽⁵³⁾ that NRC review the Project M-54 application prior to the Project M-52 (NAC-31 S/T cask) application since a Project M-54 (NAC-28 S/T) cask is already at a customer's site.

3.1.18 Pacific Sierra Nuclear Associates (PSN): VSC, Project M-53

On November 2, 1988, Pacific Sierra Nuclear (PSN)^(a) informed the NRC that it planned to submit a topical safety analysis report on their Ventilated Storage Cask (VSC) System by February 1, 1989.⁽¹²⁶⁾ PSN submitted its topical safety analysis report, PSN-89-001.^(54,55) The report is being reviewed by the NRC. This concrete cask storage system was docketed by the NRC under Project M-53. PSN plans to apply the Pacific Nuclear Systems, Inc., Quality Assurance Program (as described in the Quality Assurance Manual, Revision 15, and letter number QA-86-2, Revision 1, dated February 18, 1987) that was filed as part of the CP-9 cask information under Project M-44; PSN requested that NRC review this material and notify PSN if it is acceptable.⁽¹²⁶⁾ Consumers Power Company has selected PSN's VSC to expand on-site storage at its Palisades plant; the VSC can accommodate 24 PWR or 56 BWR fuel assemblies.⁽¹²⁷⁾ Wisconsin Electric Power Company has also selected PSN's VSC for its Point Beach station.⁽⁵⁴⁾

3.2 LICENSE APPLICATIONS FOR DRY STORAGE OF SPENT FUEL

The three organizations (utilities) that have submitted license applications for dry storage of spent LWR fuel in independent spent fuel storage installations (ISFSIs) are listed in Table 3.17; those submittals are discussed below in Section 3.2.1. The three utilities are Virginia Power (VP), which was previously known as Virginia Electric and Power Company (VEPCO), Carolina Power and Light Company (CP&L), and Duke Power Company (Duke). The applications were submitted in October 1982, February 1985, and April 1988, respectively. Planned or proposed license application submittals are shown

(a) Pacific Sierra Nuclear Associates (PSN) is jointly owned by Pacific Nuclear Systems, Inc. and Sierra Nuclear; NUTECH is a wholly owned subsidiary of Pacific Nuclear Systems, Inc.⁽⁴⁾

TABLE 3.17. License Applications for Independent Spent Fuel Storage Installations for Dry Storage of Spent Fuel

<u>Organization</u>	<u>Site or Plant</u>	<u>Docket No. (a)</u>	<u>Comments(s)</u>	<u>Reference No.</u>
• Virginia Power	Surry Power Station	72-2	GNSI's (b) Castor V/21 casks are to be used.	33,37
• Carolina Power and Light	H. B. Robinson Steam Electric Plant, Unit 2	72-3	It is planned to use NUTECH's (c) NUHOMS-07P (d) modules.	33,37
• Duke Power Station	Oconee Nuclear Station	72-4	It is planned to use a modified version of NUTECH's NUHOMS-24P (d) modules.	2,33,37, 45,128

(a) Docket No. 72-1 is associated with General Electric Company's ISFSI (the GE-Morris Operation) and is for wet storage of spent fuel.

(b) General Nuclear Systems, Inc. (GNSI).

(c) NUTECH Engineers, Inc. (NUTECH).

(d) NUTECH Horizontal Modular Storage System (NUHOMS); the NUHOMS-07P and NUHOMS-24P designs accommodate 7 and 24 spent PWR fuel assemblies, respectively.

in Table 3.18 and discussed in Section 3.2.2. The application for licensing an ISFSI includes the License Application (LA), Safety Analysis Report (SAR), and Environmental Report (ER).

3.2.1 License Applications Submitted

3.2.1.1 Virginia Power Company (Surry, Docket No. 72-2)

a) Current Status. Metal dry storage casks are deployed at the ISFSI located on the Surry Power Station site (a) in Surry County, Virginia. (33,34,37) The General Nuclear Systems, Inc. (GNSI), nodular cast iron cask design has been approved (42) for use with a borated stainless steel basket for the storage of spent fuel with an initial enrichment of ≤ 3.5 wt% uranium-235 and a decay period of at least five years. Earlier, the cask

(a) The site is 17 miles northwest of Newport News, Virginia.

TABLE 3.18. Planned or Proposed Submittals of License Applications for Independent Spent Fuel Storage Installations for Dry Storage of Spent Fuel

Organization	Site or Plant	Docket No.	Comments(s)	Reference No.
Virginia Power Co. (VP)	Surry	50-280 and 50-281	VP expects to submit the application for the license amendment in December 1988; it is associated with both the GWSI ^(a) CASTOR X and the NAC ^(b) modified S/T designs involving burnup credit.	2
Carolina Power and Light Company (CP&L)	Brunswick	50-324 and 50-325	NRC expects the license application in 1989; CP&L plans to store Robinson-2 spent fuel in a dry modular system at Brunswick. CP&L plans to use the NUTECH ^(c) NUHOMS ^(d) -07P design employed at Robinson-2 (Docket No. 72-3). The module design provides for storage of 7 PWR spent fuel assemblies in a stainless steel canister in a concrete module; up to 44 modules are planned for construction.	2
Carolina Power and Light Company (CP&L)	Robinson-2	72-3 and 50-261	NRC indicates that an application could be filed in 1989; CP&L is interested in providing additional storage capacity at the Robinson-2 site.	2
Baltimore Gas and Electric Company (BG&E)	Calvert Cliffs Station	50-317 and 50-318	BG&E plans to submit a license application in May 1989 for dry concrete module storage; BG&E plans to use NUTECH's NUHOMS-24P design. BG&E's schedule calls for construction of an initial set of 24 modules; a total of 120 modules is planned. The BG&E spent fuel is Combustion Engineering (C-E) 14x14 PWR fuel. Proposed limits of 4.3 to 4.5 wt% uranium-235 enrichment and 47,000 MWd/MTU burnup have been mentioned. Use of C-E fuel would in general lead BG&E to downsizing the NUHOM-24P, as presently designed (see Section 3.1.13, Project M-49), although the stainless steel dry shielded canister (DSC) is not expected to change. At a meeting of NRC, BG&E, and NUTECH in November 1988, NUTECH indicated there was potential for designing a transfer cask to meet the requirements of 10 CFR Part 71 for direct offsite shipment of loaded DSCs at the end of licensed storage; BG&E indicated they have provided a monetary incentive in the contract with NUTECH on this matter in an effort to foster compatibility with DOE's efforts on receiving and storing the fuel.	2, 6, 117, 129, 130
Consumers Power Company (CP)	Palisades	50-255	CP has been examining dry spent fuel storage and in-pool consolidated fuel rod storage as options. The NRC expects CP to act on the dry storage option in 1989. CP recently selected Pacific Sierra's concrete VSC (ventilated storage cask; see Section 3.1.18, Project M-54) for use at Palisades. The VSC can accommodate 24 PWR or BWR spent fuel assemblies. CP plans to initially order eight VSCs.	2, 127, 131
Northern States Power Company (NSP)	Prairie Island	50-282 and 50-306	NSP is expected to decide on a dry storage technology late in the summer of 1989.	54
Indiana and Michigan Electric Company	Cook	50-315 and 50-316	The utility is considering using dry storage, rod consolidation, or a combination of super high-density spent fuel storage racks and rod consolidation.	54

TABLE 3.18. (contd)

Organization	Site or Plant	Docket No.	Comments(s)	Reference No.
Jersey Central Power and Light Company	Oyster Creek	50-219	The utility expects to decide by the end of 1989 whether it will use dry storage or rod consolidation.	54
Wisconsin Electric Power Company	Point Beach	50-266 and 50-301	The utility has selected Pacific Sierra Nuclear's concrete VSC for its dry storage system.	54
Vermont Yankee Nuclear Power Corporation	Vermont Yankee	50-271	The utility is considering the possibility of acquiring spent fuel storage casks in the early- to mid-1990s.	54

(a) General Nuclear Systems, Inc. (GNSI).

(b) Nuclear Assurance Corporation (NAC).

(c) Nutech Engineers, Inc. (NUTECH).

(d) NUTECH's Horizontal Modular Storage System (NHHOMS).

design was approved for restricted use with a stainless steel basket to accommodate spent fuel with an initial enrichment of ≤ 2.2 wt% uranium-235 and a decay period of at least five years.

b) Background Information. VEPCO submitted the LA, SAR, and ER for the dry cask ISFSI at the Surry Power Station in October 1982.⁽¹³²⁾ The license (Docket No. 72-2, Materials License SNM-2501) for dry cask storage of spent fuel at Surry was issued (for a 20-yr period) on July 2, 1986.⁽¹³³⁾ This was the first dry storage ISFSI licensed in the United States. Chronological background information on the licensing activities is provided in Part 3.2.1.1 of Appendix C.

c) Issues: Questions/Requests and Responses. The issues (i.e., NRC's questions or requests for information) and their resolutions (i.e., responses from the utility or storage system vendor) for the Surry ISFSI are listed in Table D.1 (Appendix D). Only two questions^(a) pertained to the LA: one involved the number of casks at the site, the other concerned cost estimates.

(a) The same question may also apply to one or more of the three documents (i.e., LA, SAR, and ER).

The distributions of the questions associated with the SAR and ER are shown in Tables 3.19 and 3.20, respectively. There were 60 questions^(a) that involved the SAR.

In December 1982, the NRC indicated⁽⁹⁰⁾ that details of the cask(s) must be provided before licensing action can be completed. The questions listed in NRC's letter of September 9, 1983,⁽¹⁶⁾ addressed three principal areas of review: 1) site-related issues; 2) requests for additional information on the basic characteristics of ISFSI components, so the detailed review of the ISFSI system and its operation could continue; and 3) an elaboration on the radio-logical information associated with the ISFSI. The questions were generated from the NRC's reviews of the ISFSI's ER and SAR. Questions originating from the ER were identified by an "E" following the question number in Table D.1.

Safeguards information was inadvertently included in the license application.⁽¹¹⁾ Later, the NRC indicated⁽¹²⁾ that because of the lack of specifics in the initial application the applicant needed to completely resubmit the safeguards plans. The applicant responded to three issues raised by the NRC; the three concerned seismological, geological, and structural bases for the Surry dry cask ISFSI.^(16,17) Another issue involved the soil liquefaction analysis for the Surry ISFSI.⁽¹⁸⁾ The NRC made a site visit to discuss proposed license and technical conditions and to discuss questions they had in these areas: station health physics procedures, station administrative procedures, station training program, cask handling procedures to prevent a cask from tipping over, and the cask pressure switch monitor.⁽¹³⁾ The applicant also responded⁽¹⁵⁾ to an NRC request⁽¹⁴⁾ to increase the calculated dose rates shown in the topical safety analysis report to account for the effects of neutron subcritical multiplication.

3.2.1.2 Carolina Power and Light Company (Robinson-2, Docket No. 72-3)

a) Current Status. Dry storage of spent fuel in a concrete module and stainless steel canister system at the ISFSI at the H. B. Robinson Steam Electric Plant, Unit 2 (Robinson-2) site^(a) in Darlington County,

(a) The site is five miles northwest of Hartsville, South Carolina.

TABLE 3.19. Distribution of Questions/Requests Associated with the Safety Analysis Report for the Independent Spent fuel Storage Installation at Surry

Safety Analysis Report		Title	Questions/Requests		
Section	Subsection		No.	Subtotal	Total
3.25	2.0	THE SITE CHARACTERISTICS			19
	2.1	Geography and Demography of Site Selected		5	
	2.1.3	Population Distribution and Trends	5		
	2.2	Nearby Industrial, Transportation, and Military Facilities		2	
	2.2.3	Effects of Potential Accidents:			
		--Pipelines	1		
		--Waterborne Traffic	1		
	2.3	Meteorology	5	7	
	2.3.2	Local Meteorology	2		
	2.6	Geology and Seismology	4	4	
	2.7	Summary of Site Conditions Affecting Construction and Operating Requirements	1	1	
	3.0	PRINCIPAL DESIGN CRITERIA			3
	3.1	Purpose of Installation	1	2	
	3.1.1	Materials to be Stored	1		
	3.3	Safety Protection System	1	1	
	4.0	INSTALLATION DESIGN	1	1	2
	4.2	Storage Structures		1	
	4.2.3	Individual Unit Description	1		
	5.0	OPERATIONS SYSTEMS			2
	5.2	Fuel Handling Systems	2		
	7.0	RADIATION PROTECTION			21
	7.1	Ensuring that Occupational Radiation Exposures Are As Low As Is Reasonably Achievable (ALARA)	1	1	
	7.2	Radiation Sources	1	1	
	7.3	Radiation Protection Design Features	9	9	
	7.4	Estimated Onsite Collective Dose Assessment	4	4	
	7.6	Estimated Offsite Collective Dose Assessment	6	6	

TABLE 3.19. (contd)

Safety Analysis Report		Title	Questions/Requests		
Section	Subsection		No.	Subtotal	Total
8.0		ACCIDENT ANALYSIS			1
	8.2	Accidents		1	
	8.2.10	Cask Drops	1		
9.0		CONDUCT OF OPERATIONS			7
	9.1	Organizational Structure	2	2	
	9.3	Training Programs	1	1	
	9.4	Normal Operations	1	1	
	9.5	Emergency Planning	1	1	
	9.6	Decommissioning	2	2	
10.0		OPERATING CONTROLS AND LIMITS			3
	10.1	Technical Specifications	3	3	
11.0		QUALITY ASSURANCE			2
	11.1	Quality Assurance Program Description-Virginia Power	2	2	

TABLE 3.20. Distribution of Questions/Requests Associated with the Environmental Report for the Independent Spent Fuel Storage Installation at Surry

Environmental Report		Title	Questions/Requests		
Section	Subsection		No.	Subtotal	Total
2.0		THE SITE AND ENVIRONMENTAL INTERFACES			8
	2.1	Geography and Demography	1	5	
	2.1.2	Population Distribution	4		
	2.3	Meteorology	3	3	
3.0		THE FACILITY			9
	3.5	Radwaste Systems and Source Terms	9	9	
4.0		ENVIRONMENTAL EFFECTS OF SITE PREPARATION AND FACILITY CONSTRUCTION			7
	4.4	Radioactivity	7	7	
5.0		ENVIRONMENTAL EFFECTS OF FACILITY OPERATION			6
	5.2	Radiological Impact from Routine Operation	3	3	
	5.6	Other Effects		1	
	5.6.2	Climatological Impact	1		
	5.8	Decommissioning and Dismantling	2	2	
6.0		EFFLUENT AND ENVIRONMENTAL MEASUREMENTS AND MONITORING PROGRAMS			3
	6.2	Proposed Operational Monitoring Programs	3	3	
9.0		ALTERNATIVE ENERGY SOURCES AND SITES			1
	9.1	Alternatives to the Proposed Action	1	1	
10.0		FACILITY DESIGN ALTERNATIVES			1
	10.1	Site Alternatives	1	1	

South Carolina has been authorized⁽¹³⁴⁾ and deployed. The system, a modified form of the NUTECH Horizontal Modular Systems (NUHOMS) design, will accommodate spent fuel with a maximum burnup of 35,000 MWd/MTU.⁽³³⁾ A NUHOMS module contains a single stainless steel canister, which is sealed by a double-welded shield plug at each end. The canister, holds seven PWR fuel assemblies of initial enrichment of ≤ 3.5 wt% uranium-235.

b) Background Information. CP&L submitted the LA (included the ER as Chapter 12) and SAR to the NRC in February 1985.⁽¹³⁵⁾ The license (Docket No. 72-3, Materials License SNM-2502) for dry storage of spent fuel in concrete and stainless steel modules was issued on August 13, 1986.⁽¹³⁴⁾ This was the second dry storage ISFSI licensed in the United States. Chronological background information on the licensing activities is provided in Part 3.2.1.2. of Appendix C.

c) Issues: Questions/Requests and Responses. The NRC letter of May 3, 1985,⁽²⁹⁾ indicated that CP&L's license application⁽¹³⁵⁾ (specifically the ER) did not fully comply with 10 CFR 72.20. The NRC letter of June 28, 1985,⁽³⁰⁾ to CP&L provided a list (see Table 3.21, which is at the end of Section 3.0 because of its length) of environmental questions (intended only as a guide) that might be helpful in preparing the ER. Examples of some of the issues involved in the SAR for the Robinson-2 ISFSI are listed in Table 3.22 (placed at the end of Section 3.0 because of its length). The issues and their resolutions for the Robinson-2 ISFSI are listed in more detail in Table E.1 (Appendix E). An issue associated with the LA involved NRC's request for revised security documentation. The distribution of the questions/requests from the NRC in the various sections of the SAR is shown in Table 3.23 (placed at end of Section 3.0 because of its length). The distribution of the 99 questions^(a) (in decreasing order of the number of questions) in the various sections of the SAR is as follows: "Introduction and General Description of the Installation"-20, "Operations Systems"-18, "Principal Design Criteria"-14, "Analysis of Design Events"-12, "Radiation Protection"-10, "Site Characteristics"-9, "Operating Controls and Limits"-8, "Installation Design"-8, and "Waste Confinement and Management"-1, and

(a) Some questions pertain to more than one section.

"Quality Assurance"-1. One of the issues involved the dry shielded canister (DSC) enveloping load combination and the maximum DSC stresses for an 8-foot bottom drop accident.⁽¹³⁶⁾ The applicant provided responses to other issues raised by the NRC, including the assumptions⁽¹³⁷⁾ on fuel enrichment and burnup used in the ORIGEN-2⁽¹³⁸⁻¹⁴¹⁾ analyses, the lubricant to be used on the DSC,⁽¹³⁷⁾ and a description of the analysis of the thermal load on the concrete foundation.⁽¹⁴²⁾ An NRC memorandum⁽⁸⁾ indicated that the outstanding issues remaining to be closed included the final quality assurance commitments and completion of the utility's internal plant review to confirm operations of the ISFSI and the reactor have no significant impact on one another, that no changes to the plant's operating license are needed for the ISFSI, and that the ISFSI operates independently. The applicant provided responses, including the conclusion that the ISFSI is "independent" as defined in 10 CFR Part 72.⁽⁹⁾ A question associated with the ER involved the onsite collective dose commitment to workers constructing the five additional modules. The questions associated with CP&L's request for an amendment to Materials License No. SNM-2502 pertained to the poison material and to the spacer disk structural analysis.

3.2.1.3 Duke Power Company (Oconee, Docket No. 72-4)

On March 31, 1988, Duke Power Company (Duke) transmitted⁽¹⁴³⁾ a letter to the NRC with proposed amendments to the Technical Specifications for Facility Operating Licenses DPR-38, -47, and -55 (Oconee-1, -2, and -3, respectively) to permit Duke to operate an Independent Spent Fuel Storage (ISFSI) it plans to construct at the Oconee site.^(a) At that time, Duke stated that the ISFSI will utilize the NUTECH Horizontal Modular Storage (NUHOMS-24P) Dry Storage System (see Project M-49 for details). Duke is directly involved in the criticality analysis, which involves allowance for burnup credit, and in the design of the concrete horizontal storage module for the NUHOMS-24P.^(37,144) A general comment made by the NRC was that the applicant needs to revise the application to account for (e.g., provide dose rate information) the installation and use of up to 88 Horizontal Storage Modules (HSM); the applicant provided that information.⁽¹⁴⁵⁾ Duke is hoping

(a) The site is located 30 miles west of Greenville, South Carolina.

to have the ISFSI available for operation by January 1, 1990.⁽¹⁴⁴⁾ The capacity of the ISFSI is to be approximately 2000 fuel assemblies.⁽¹⁴⁶⁾ Duke's plans for storage of spent fuel in the dry concrete module and stainless steel canister system at the Oconee Nuclear Station site^(a) in Oconee County, South Carolina, were described in a recent INMM paper.^(147,148) Chronological listing of the licensing activities is provided in Part 3.2.1.3 of Appendix C.

An issue that arose during NRC's review of the License Application for the ISFSI centered on the question of whether, and to what extent, the spent fuel storage canisters to be used at the Oconee ISFSI will be compatible with the transportation cask ultimately chosen by DOE to transport spent nuclear fuel temporarily stored at reactor sites to permanent repositories. In their September 1988 response⁽⁷⁾ to that issue, Duke indicated, given the fact that DOE's design effort for its cask fleet is in its early stages, that no meaningful compatibility review can be conducted at this time. Duke began discussions with DOE in May 1988 on the interface issue.⁽⁷⁾ At a meeting of Duke and DOE in August 1988,⁽⁷⁾ joint agreement was reached on a number of items, including these: 1) in the view of Duke and DOE, the Duke NUHOMS system is presently compatible with the planned DOE system configuration by transferring spent fuel assemblies from the storage canister to a transport cask in the spent fuel pool, 2) Duke will address the feasibility of modifying the storage canister shielding design to reduce canister length, and 3) DOE will encourage standardization of canisters to the extent practical to optimize and facilitate future handling, including potential transportation.

Examples of some of the issues involved in the ISFSI's Environmental Report are shown in Table 3.24 (placed at the end of Section 3.0 because of its length). Examples of some of the issues involved in the ISFSI's Safety Analysis Report are listed in Table 3.25 (placed at the end of Section 3.0 because of its length). In August 1988, NRC requested⁽¹⁴⁹⁾ additional information from NUTECH because some of the information requested by NRC in May 1988 was not supplied or was not supplied in sufficient detail, and some calculations that were provided did not appear to correspond to the NUTECH design. In November 1988, the NRC requested⁽¹⁵⁰⁾ additional

information to confirm the design basis deceleration levels for the cask drop case (the NRC indicates that it is possible that the deceleration levels presented by NUTECH could be higher than stated in the topical report, NUH-002, Revision 1).

The NRC is currently continuing its review of the licensing submissions.⁽²⁾

3.3 LICENSE APPLICATIONS FOR LARGE-SCALE STORAGE OF CONSOLIDATED SPENT FUEL

The status of rod consolidation in the United States is described in a recent report, PNL-6603.⁽¹⁵¹⁾ As indicated in that report, there have been a number of demonstrations of rod consolidation with PWR spent fuel. Three utilities have applied for approval for large-scale storage of consolidated fuel rods in spent fuel storage pools.

3.3.1 Maine Yankee

3.3.1.1 Current Status

The plant is authorized to consolidate spent fuel rods from no more than 20 fuel assemblies.

3.3.1.2 Background Information

On September 18, 1979, a request to increase the spent fuel storage pool capacity at Maine Yankee from 953 to 1545 fuel assemblies by using a rod consolidation technique was submitted to the NRC.⁽¹⁵²⁾ The NRC completed their review of the application and issued a Safety Evaluation and Environmental Impact Appraisal in 1982.⁽²⁰⁾ However, the state of Maine ordered⁽¹⁵³⁾ the utility to withdraw its application to the NRC for licensing of large-scale rod consolidation at Maine Yankee. The state would allow, and the NRC authorized,⁽²¹⁾ the plant to consolidate only up to 20 spent fuel assemblies. The chronological listing of the licensing activities is provided in Part 3.3.1 of Appendix F.

3.3.1.3 Issues: Questions/Requests and Responses

In its evaluation⁽²⁰⁾ of the application, the NRC noted that acceptance of the proposed changes was contingent on resolution of these three open items:

1. The licensee must have procedures stating that during full core discharges and after the first third of the core has been discharged, pool bulk water temperature will be monitored following the insertion of each additional assembly. Should this temperature exceed 154°F these procedures must direct that recently discharged fuel be returned to the reactor vessel until pool bulk water temperature drops to or below 154°F.
2. A limit must be added to the Maine Yankee Technical Specifications that requires that fuel decay at least 120 days from shutdown before it may be consolidated.
3. The license must be conditioned to preclude lifting a spent fuel shipping cask over the pool until a cask drop analysis is submitted by the licensee and approved by the staff.

The utility provided responses and the NRC issued a Supplemental Safety Evaluation on October 22, 1982, that closed the open issues.⁽²¹⁾

3.3.2 Ginna

3.3.2.1 Current Status

The NRC issued⁽²⁴⁾ Amendment No. 12 to Ginna's Facility Operating License on December 12, 1985. It authorized storage of consolidated fuel canisters in the existing spent fuel pool and use of the Auxiliary Building crane to move consolidated fuel canisters. The theoretical storage capacity of the pool would be increased to 2032 fuel assemblies (2 x 1016); however, due to the limitation on the heat removal capability of the spent fuel pool cooling system, the storage capacity is limited to 1016 fuel assemblies.^(24,25)

3.3.2.2 Background Information

Rochester Gas and Electric Corporation submitted an application to NRC on February 27, 1985, for an amendment to Ginna's Facility Operating License to allow storage of consolidated fuel. The chronological listing of the licensing activities is provided in Part 3.3.2 of Appendix F.

3.3.2.3 Issues: Questions/Requests and Responses

At the meeting on June 4, 1985,⁽¹⁵⁴⁾ the NRC requested a change on one page (3.11-2) of the application⁽¹⁵⁵⁾ that involved incorporating previous changes that had been submitted but for which approval had not yet been received. The utility provided a response.⁽¹⁵⁶⁾ It indicated that Section 3.11.3 was revised to read "A load in excess of one fuel assembly and its handling tool shall never be stationed or permitted to pass over storage racks containing spent fuel." It also indicated that Section 3.11.5 had been revised to read "The restriction of 3.11.3 above shall not apply to the movement of canisters containing consolidated fuel rods if the spent fuel rack beneath the transported canister contains only spent fuel that has decayed at least 60 days since reactor shutdown." The response also indicated that the requirement of 3.11.5 ensures that should a handling accident occur during movement of a consolidated fuel canister, the dose at the exclusion area boundary would satisfy the requirements of 10 CFR 100.

In December 1986, the utility sent a letter⁽²⁵⁾ to the NRC. The utility indicated that in its review of the NRC's Safety Evaluation Report,⁽²⁴⁾ two statements were noted in the NRC Staff's analysis that were not consistent with the amendment or the utility's responses to NRC Staff's requests for additional information. These statements concern the use of fuel with burnable poisons and the Auxiliary Building crane. The utility requested review and clarification by NRC of the Safety Evaluation Report (SER) because the inconsistencies noted below present the potential for unnecessary confusion and the perception that plant operation may exceed the bounds of the licensing basis. The letter⁽²⁵⁾ indicated that these issues were discussed with the NRC Staff after Amendment 12 was issued.

In Section 2.1 of the December 1985 SER it is stated that "Rochester Gas & Electric (RG&E) does not have nor do they contemplate to use fuel assemblies with burnable poisons,...".⁽²⁵⁾ The reference given was RG&E's response to Staff questions dated July 11, 1985. In the response to question 4 of that letter, RG&E included a provision that the applicability of the burnup-enrichment curves of Figure 5.4-2 of the Technical Specifications would be verified if burnable poisons were used. The verification would

assure that the criteria for the pool K_{eff} of less than or equal to 0.95 would be met. The provisions for this evaluation were incorporated into the proposed changes to the basis of the specification which were included in RG&E's response. These same proposed changes to the basis included a statement requested by the NRC Staff concerning the effect of storing less than 179 fuel rods per half canister. Neither of these changes to the basis were incorporated into the specification approved by the SER.

In Section 2.3.2 of the SER, the NRC Staff analysis of canister handling appears to conclude that because the Auxiliary Building crane meets the single failure proof criteria of NUREG-0612 and handling of the canister uses the 5-ton hook of the Auxiliary Building crane, the handling of the canister will satisfy the single failure proof guidelines and is therefore acceptable.⁽²⁵⁾ The NRC Staff references the SER on the modification to the Auxiliary Building crane dated October 1, 1984. This SER, and the RG&E submittals on which this SER was based, never addressed any modification to the crane that would result in a load being transported by the 5-ton hook in a single-failure proof mode. Rather the radiological effects of a canister drop were evaluated and found to be acceptable. RG&E has modified the 40-ton hook on the Auxiliary Building crane to be single failure proof to meet other heavy load handling concerns that are separate from use of the 5-ton hook for canister movements.

3.3.3 Millstone-2

3.3.3.1 Current Status

Northeast Nuclear Energy Company (NNECO), a part of Northeast Utilities (NU), has received approval from the NRC for full-scale storage of consolidated spent fuel at Millstone-2.⁽¹⁵⁷⁾

3.3.3.2 Background Information

On May 21, 1986, NNECO submitted to the NRC a request for an amendment to its operating license for Millstone-2 to allow storage of consolidated fuel.⁽¹⁵⁸⁾ The chronological listing of the licensing activities is provided in Part 3.3.3 of Appendix F.

3.3.3.3 Issues: Questions/Requests and Responses

The issues and their resolutions for the large-scale storage of consolidated spent fuel rods at Millstone-2 are listed in Table G.1 (Appendix G). The distribution of questions/requests from the NRC by subject matter is shown in Table 3.26.

In 1987, the NRC indicated that the issuance of the requested amendment to the plant's operating license would be delayed until NRC Staff resolution of BNL 87-05 (Generic Issue 82, "Beyond Design-Basis Accidents in Spent Fuel Pools").

One of the issues involved in the application was the proposed requirement that candidate fuel assemblies (for consolidation) must have decayed for at least 5 years.^(22,23) The NRC requested a justification for the proposed decay time in terms of the consolidation process. The NRC also requested the applicant to provide a complete description of the consolidation process and an associated safety analysis. Another issue involved dose rates from crud.⁽²⁸⁾ The NRC indicated that because many fuel rod movements will have been involved in the reconstituted fuel assembly, it is likely activated crud will be released from some fuel rods and adhere to some other rods. The NRC requested the applicant to show how an increase in dose rates from such crud will be precluded when consolidated assemblies are returned to storage in the spent fuel pool. The applicant's response⁽²⁸⁾ gave three reasons why the potential dose rate consequence from crud redepositing on fuel surfaces is insignificant. One of the NRC's requests was to describe the method for identifying the fuel rods that are most likely to rupture.⁽²⁸⁾ The applicant indicated that prior to consolidation, all candidate fuel assemblies will be pre-inspected with the consolidation system TV cameras to review the condition of the exterior rows of rods. Any fuel assemblies found to have defects will not be consolidated. Another of the NRC's requests was for the applicant to provide all data (mathematical models, parameters, codes and techniques used) that were used in the calculational model devised to determine the increase in dose rates in the spent fuel pool area due to a buildup of radionuclides in the pool water.⁽²⁸⁾

TABLE 3.26. Distribution of Questions/Requests Associated with the Application for a License Amendment Concerning Rod Consolidation at Millstone-2

<u>Subject Matter</u>	<u>No. of Questions/ Requests^(a)</u>
Nuclear Considerations	15
Thermal-Hydraulics	4
Fuel Integrity or Condition	4
Materials	3
Process/Procedures	8
Accident Analyses	3
Radiation Aspects	4
Fuel Rod Breakage	2
Debris (crud, "heavy" ^(b) debris, etc.)	2
Schedule	1
Rack Test Program	1
Fuel from Other Millstone Units	1
Future Plans (dry storage and consolidation)	1

(a) Some questions/requests pertained to more than one subject matter.

(b) For example, parts (fuel pellets, cladding pieces, etc.) from broken or damaged fuel rods.

The NRC noted that the application included a discussion of a rather extensive test program to be used in conjunction with the analysis of the Spent Fuel Rack System but the testing program and the use of resulting data are not addressed in the Safety Analysis Report.⁽²⁶⁾ The NRC requested a discussion of the testing portion as it seemed to be an important step in doing a reliable analysis of the Spent Fuel Rack System. In the applicant's response,⁽²⁶⁾ one of the statements indicates that the results from the consolidated fuel storage box (CFSB) local stiffness test showed that the behavior of the CFSB is complicated under a transverse line load and that the CFSB exhibits a great variation in stiffness values. At high loads, local yielding or buckling of the box walls occurs. Another issue raised by the NRC was the applicant's assumption that no lateral deformation results from a 100,000-lb load impacting the fuel assembly.⁽²⁷⁾ The NRC indicated that this

assumption is not justified and seems invalid since it has been determined that the guide tubes are yielded due to the load. The applicant's evaluation of August 11, 1987, indicates that a 3 x 3 temporary storage rack is to be used for both intact and consolidated spent fuel.⁽¹⁵⁹⁾ This rack was not included in the NRC's review of consolidated spent fuel storage authorized in the recent technical specification change approval; hence, the NRC requested justification for use of this rack. In another issue, the NRC Staff recommended that a Technical Specification Surveillance Requirement be incorporated for consolidated fuel to verify the integrity of the fuel and structural elements before movement or placement in the spent fuel pool.⁽³¹⁾ The NRC also requested information on the method the applicant proposes to use to verify the integrity of the storage canister after it has been loaded with fuel rods.⁽³¹⁾

**TABLE 3.3. Examples of Some Issues^(a) Involved in Project M-34
(GNSI, Castor Ic)**

<u>Section of Topical Report</u>	<u>No. of Issues</u>	<u>Items Involved</u>
1.0 Introduction and General Description of Installation		
1.1 Introduction		
1.2 General Description of Installation		
1.2.2 Principal Design Criteria	4	<ul style="list-style-type: none"> - Criteria in this section versus in Tables 3.1-11 and 3.2-1 (criteria inconsistent). - Cask helium pressure (not stated in table). - Cavity pressure (involves partial vacuum and only helium?; corresponding temperature?). - Maximum temperature of hottest fuel rod (less than 400°C?).
1.2.4 Structural Features	2	<ul style="list-style-type: none"> - Third lid (dimensions and materials?; how secured to cask?; relationship to protective cover?; modified protective cover required also?). - Dimensional specifications (discrepancies in 3 lengths and 3 widths).
1.3 General System Description	1	<ul style="list-style-type: none"> - Location of at-reactor storage installation.
3.0 Principal Design Criteria	1	<ul style="list-style-type: none"> - Maximum design pressure in storage cask (what is it?).
3.1 Purpose of Installation		
3.1.1 Material to be Stored		
3.1.1.2 Characteristics of Spent Fuel	2	<ul style="list-style-type: none"> - Statement on conservative fuel enrichment level (justify or remove). - All tables and figures "should alone" and include design bases information (e.g., burnup, decay time, etc.).
3.2 Structural and Mechanical Safety Criteria		
3.2.4 Snow and Ice Loading	1	<ul style="list-style-type: none"> - Resistance to fatigue damage (needs evaluation).
3.2.5 Combined Load Criteria	2	<ul style="list-style-type: none"> - Criteria relevant to storage casks (include in this section). - Source of design criteria in table (ASME Code or Regulatory Guide 7.6 or both?).
3.3 Safety Protection System		
3.3.2 Protection by Multiple Confinement Barriers and Systems	2	<ul style="list-style-type: none"> - Third lid design criteria (need discussion here). - Need discussion of criteria (accident protection, backup confinement, achieving lowest practical release levels).
3.3.2.1 Confinement Barriers and Systems	2	<ul style="list-style-type: none"> - Sealing system (sealing criteria?; maximum allowable leakage rate?). - Replacement or substitute seals (sealing criteria?)

(a) From Reference 59. Additional issues are in References 36, 58, 165, and 166. Responses are in References 167-170.

TABLE 3.3. (contd)

Section of Topical Report	No. of Issues	Items Involved
3.3.2.2 Activity Release	6	<ul style="list-style-type: none"> - Leakage rate (correct molecular part of Knudsen formula; provide leakage analysis using appropriate parameters including 100% fuel rod failure). - Leakage rate of double seal system (provide formula derivation). - Figure shows leak rate increases with time (equation says leak rate decreases; equation implies leak rate zero at time zero, which is not the case). - Release fraction data (bases?; information should be in text and not in restricted appendix; consider using Regulatory Guide 1.1.25). - Castor 1c seal system (include and explain analysis and test data and procedures; include test data on nodular cast iron). - Annual activity releases (how computed?; time integrated releases and 50-year storage life used?; was decay considered?).
3.3.3 Protection by Equipment and Instrumentation Selection		
3.3.3.2 Instrumentation	1	<ul style="list-style-type: none"> - Switching pressure (what is it?).
3.3.4 Nuclear Criticality Safety		
3.3.4.3 Verification Analysis	8	<ul style="list-style-type: none"> - Neutron cross section for criticality analysis (16-group is inadequate, 123-group is state of the art). - GANTEC and HAMMER codes (what are they?; benchmarked?). - Criticality model (provide complete model; include benchmark calculations). - Criticality analysis for Castor 1c geometry and materials (provide input and output data). - Castor 1c storage cask (moderation ratio?). - Results for 8x8 BWR fuel (where are they?; water rod included?). - Criticality model (discrete fuel rods or homogeneous water-fuel mixture assumed?). - Basket drawing (dimensions are illegible).
3.3.4.3.2 Structural Data	1	<ul style="list-style-type: none"> - Discrepancies in densities for steel and nodular-graphite cast iron (what are correct values?).
3.3.5 Radiological Protection		
3.3.5.2 Shielding	5	<ul style="list-style-type: none"> - Experimental verification of calculated dose rates (were same "representative burnup and decay time" used in shielding calculations?). - Location bases for handling operations dose rates (what are they?; determination of dose rate variations?; distance assumed in shielding calculations?). - Distances and occupancy times in table (how determined?; likely distribution?; include 5% and 95% confidence bounds). - Data in two tables (are they conservative?; uncertainties in dose rates?). - Direct radiation and sky-shine results (not useful unless input data included; all tables and figures need sufficient design bases information to stand alone).

TABLE 3.3. (contd)

Section of Topical Report	No. of Issues	Items Involved
3.3.7 Materials Handling and Storage		
3.3.7.1 Spent Fuel Handling and Storage	8	<ul style="list-style-type: none"> - Potential cladding failure at design temperature of 400°C (not adequately addressed). - Maximum external contamination limits (where do these come from?). - Statement that "...maximum cladding temperature in dry storage casks should be comparable with values during reactor operation" (not valid; provide cladding integrity data). Inclusion of Appendix 4 from Gorleben Safety Analysis Report (permissible for information purposes only; further information on fuel cladding integrity needed; detailed explanation of 425°C as acceptable temperature). Fuel cladding integrity tests (provide details). Exxon GAPEX-2 code and calculations with it (provide details). Gas (helium) in cask prevents external corrosion of cladding (procedures for removing impurities and water?; limiting values?; demonstrate corrosion is insignificant). Stress corrosion cracking (provide detailed discussion). Propagation of fuel cladding cracks (discuss in detail). Potential mechanisms for fuel cladding failure (need to address; establish design bases to avoid cladding failure; demonstrate that design bases met). - Fuel cladding storage temperature limit (provide description and calculational and/or test results; provide calculational details to allow adequacy evaluation). - Surface contamination levels (what are these?; do references to "transport regulations" imply 49 CFR Part 173?). - Storage of defective fuel assemblies (in U.S., NRC says no need to put known defective spent fuel in dry storage; hence no known defective fuel should be accepted for cask storage). - Justification of cladding design temperature of 400°C (description of data and substantiation of conclusions needed; relate stress, temperature, and creep rupture life for cladding material; include variation of cladding pressure or stress with time; provide analysis to support 1000-year cladding life without rupture). - Spent fuel cladding creep rupture life (should be stated; design criteria?). - The 95% confidence level for 1000-year lifetime at 400°C (need supporting analyses and data).
3.4 Classification of Structures, Components, and Systems	3	<ul style="list-style-type: none"> - Discrepancy in table showing flushing and drainage connections (they must not leak during storage period; did analysis in Section 3.3.2.2 include these two potential leakage sources). - Flushing and drainage connections (type of metal seal?). - Discrepancy in table showing cask trunnions (they must not fail when cask lifted).
3.5 Decommissioning Considerations		
3.5.1 Storage Casks	1	<ul style="list-style-type: none"> - Thermal neutron flux assumptions (how determined?; activation thicknesses assumed for cask wall, bottom, and cover?; how were ORIGEN activation calculations performed?; how was variation in flux with time taken into account?).

TABLE 3.3. (contd)

Section of Topical Report	No. of Issues	Items Involved
4.0 Installation Design		
4.2 Storage Structures		
4.2.1 Structural Specification	12	<ul style="list-style-type: none"> - Section III of ASME Code cited (none of calculations conform to this specification). - Stress intensity limits for nodular cast iron (no limits stated in Section III of ASME Code). - Reference to lesser of one-third ultimate or two-thirds yield (only applies to ductile steel). - Yield strength of bolts (temperature associated with stated value?; is this highest temperature for bolts?). - Bolt material stress limits for bending (need to state). - Primary and secondary cover materials stress intensity limit (high range of materials?; base on minimum properties?). - S_m for trunnion material (20,000 psi, not 120,000 psi). - GGG-40.3 specifies 12% minimum elongation (Castor Ic material does not conform). - Allowable stress limits (from ASME Code?; if so, source?). - Cask body material does not meet GGG-40.3 standards (provide material specification; provide data to support properties used in analysis). - X 5 Cr Ni 13 4 (what is it?; identify it with equivalent ASME or ASM material specification). - Primary and secondary cover (identify material with equivalent ASME or ASM material specification).
4.2.1.1 Calculation of the Cask Body	5	<ul style="list-style-type: none"> - Cask internal pressure during normal storage (what is it, assuming 3 fuel rod failure rates and cask buried in debris?; provide calculations showing design limit not exceeded under accident conditions). - Calculation of safety factors or limiting pressure levels (use of formula for extrapolation not acceptable). - All stress intensities for normal conditions (provide data; use ASME criteria and rules for calculations). - Incorrect sentence structure in one paragraph. - Subscript discrepancy in one figure.
4.2.1.2 Calculation of the Primary Lid	4	<ul style="list-style-type: none"> - Primary lid (thin plate theory not valid for thick-walled lids; provide ASME Code analysis and include thermal stresses). - Use of maximum distortion energy theory (deviates from ASME Code's rules). - Page 4.2-8, "foreseen?" - $F_{TB} = 6.25 \times 10^4$ N?
4.2.1.4 Calculation of the Primary Lid Bolts	2	<ul style="list-style-type: none"> - Statement that material yield strength must exceed calculated maximum stress (incorrect; design stress intensity limit must not exceed calculated maximum stress). - German thread standards (NRC reviewers not familiar with them; provide data).

TABLE 3.3. (contd)

Section of Topical Report	No. of Issues	Items Involved
4.2.1.7 Material Properties	10	<ul style="list-style-type: none"> - Section title (make "Ductile Cast Iron Material Properties" or provide materials properties for each major component). - Qualifying cask against brittle failure (relationship of material ductility?). - Fracture mechanics (describe role; provide fracture toughness data). - Statement regarding "positive tests with prototype CASTOR cask" (what is reference?). - Statement that prototype cask values apply to mass-produced casks (assurance that test results help support this?). - Statement concerning "positive drop test results" (what is reference?). - The 9-m free drop test (description of test?). - Qualification of cask against brittle fracture (relationship of material parameters?). - Factor of safety of one or margin of safety of zero (adequate for safety?; assurance that mass-produced casks will not fail?). - Statement regarding "more than 120 materials tests" (source?).
4.2.2 Installation Layout	2	<ul style="list-style-type: none"> - Appendix 5 (Wrong reference). - Discrepancy in optimum distance for center-to-center spacing (correct value?).
4.2.3 Individual Unit Description		
4.2.3.3 Design Bases and Safety Assurance	1	<ul style="list-style-type: none"> - Cask trunnions (include in list).
5.0 Operation Systems		
5.1 Operation Description		
5.1.1 Narrative Description		
5.1.2 Flow Sheets	4	<ul style="list-style-type: none"> - "Function check of the monitoring system" (why not include as Task 8?). - Dose rate at one meter (0.005 rem/hr instead of 0.005 mrem/hr?). - Total exposure (man-rem instead of mrem/hr?). - Dose variation with distance for personnel exposure (explicit function derived or assumed?).
5.1.3 Criticality Prevention	1	<ul style="list-style-type: none"> - Statement that criticality control during storage is not necessary (should not be stated).
5.1.3.6 Heat Transfer Capability of the Cask Body	3	<ul style="list-style-type: none"> - Emissivity coefficient (influence of environmental effects?). - Heat transfer calculation for cask bottom (unconservative assumption used). - Fin heat dissipation (provide calculations for one fin).
5.1.3.7 Fuel Rod Temperatures	4	<ul style="list-style-type: none"> - Equilibrium fuel rod temperature discrepancy involving MIT analysis and GNS data (cause of difference?). - Benchmarking by four full-scale experiments (provide test data; provide analysis). - Half-scale experiments (provide further detail). - Equilibrium fuel rod temperature (use bounding value for temperature increase; determine temperature; provide analysis calculations).

TABLE 3.3. (contd)

Section of Topical Report	No. of Issues	Items Involved
		- Handling operations (health risks?; provide temperature information).
5.3 Other Operating Systems		
5.3.2 Component/Equipment Spares	2	- Evolving? - Installation of spare or alternative equipment (discuss or reference other sections).
6.0 Waste Confinement		
6.5 Radiological Impact of Normal Operations	1	- Dose calculation [how performed?; determination of diffusion factor?; flux-to-dose conversion factors used? (if so, how established; if not, why not); provide sample calculation].
7.0 Radiation Protection		
7.1 Ensuring That Occupational Radiation Exposures Are As Low As Is Reasonably Achievable (ALARA)		
7.1.1 Policy Consideration	3	- Cask safety conclusions (need to substantiate). - Restoration of double containment (reference to Section 5 appears incorrect or insufficiently specific). - "Periodic optical inspection" policies (need details).
7.1.2 Design Considerations	2	- Seal replacement (discussion inconsistent with that in Section 8.1.1.4). - Figure in last sentence (what and where is it?).
7.2 Radiation Sources	3	- "Weight factor" (what is it and how used?). - "Conversion factor" (units and source?). - Energy bounds (needed in two tables).
7.2.1 Characterization of Sources		
7.2.2 Airborne Radioactive Material Sources	3	- Release fractions (bases?). - Cladding failure (is 100% failure assumed in two tables?). - Design bases information (more data needed in one table).
7.3 Radiation Protection Design Features		
7.3.1 Installation Design Features	3	- Regulatory position 2 of Regulatory Guide 8.8 (GNS should address each section). - All sources of radioactivity (provide four types of data). - Radiation dose rates during maintenance, repair and surveillance activities (provide data).
7.3.2 Shielding		
7.3.2.1 Shielding Design Features	4	- Cask wall thickness inconsistency (why the difference?). - Homogenized fuel and basket (material properties?). - Effective moderator thickness (how determined?; discuss method). - Tables and figure (need to stand alone).

TABLE 3.3. (contd)

Section of Topical Report	No. of Issues	Items Involved
7.3.2.2 Shielding Analysis		
7.3.2.2.1 Shielding Calculation	7	<ul style="list-style-type: none"> - Effective moderator thickness calculation (discuss). Polyethylene density (inconsistency in values?; effect on calculation?). - End-piece calculations and source terms (where described?). - Mean neutron spectrum in Appendix 2 (how determined?; need input data for analyses). - Cavity length (inconsistent values). - "Maximum dose rate" in last paragraph (misleading statement; should be "maximum for average dose rate"). - "Shielding effect of the fins" (should state it is taken into consideration). - Shielding calculations (uncertainties?).
7.3.2.2.2 Experimental Verification	2	<ul style="list-style-type: none"> - Experimental versus shielding calculations (same fuel conditions?). - Lid surface (source of "factor of 10..."?).
7.4 Estimated Onsite Collective Dose Assessment	5	<ul style="list-style-type: none"> - Time and dose for routine optical inspection (inconsistency in data; which values are correct?). - Collective dose for cask confinement control (inconsistency in values). - Cask confinement control, including tightness testing (not included in two tables). - Dose rate for individuals (where is it?). - Tightness testing (time allowed?; procedure?; number of people?).
8.0 Accident Analysis		
8.1 Off-Normal Operations	1	<ul style="list-style-type: none"> - Each event (discuss separately; include description, cause, detection method, effects and consequences, and corrective actions).
8.1.1 Event		
8.1.1.2 Detection of Event	1	<ul style="list-style-type: none"> - Hypothetical failure of pressure gage components or interruption of monitoring system electrical energy supply (this event not mentioned here).
8.1.1.3 Analysis of Effects and Consequences	3	<ul style="list-style-type: none"> - Last sentence, first paragraph (does not make sense). - Activity release (comments in Section 3.3.2.2 apply here also). - Activity calculations (need details in table; are certain releases truly undetectable?).
8.1.1.4 Corrective Actions	4	<ul style="list-style-type: none"> - Third lid sealing system (how accomplished?; required criteria?). - Time periods for action steps (need data in figure). - Corrective actions for cask tipping (what are actions for cask drop and cask collision?; same as for tipping?; summaries same as in figure?). - Adding third lid (relevant handling information?).
8.1.2 Radiological Impact from Off-Normal Operations	3	<ul style="list-style-type: none"> - Details of calculations (not understood). - Dose units for secondary cover seal replacement (man-rem, not mrem; substantiate dose exposure statement). - Third lid addition (include dose data).

TABLE 3.3. (contd)

Section of Topical Report	No. of Issues	Items Involved
8.2 Accident		
8.2.1 Accidents Analyzed		
8.2.1.2 Accident Analysis		
8.2.1.2.2 Tornado Missiles	2	<ul style="list-style-type: none"> - Wind speed and missile speed assumptions (inconsistency in data). - Reference to Appendix 9 (be more specific).
8.2.1.2.3 Cask Tip-Over Incident (Case I)	5	<ul style="list-style-type: none"> - Cask loading is independent of distance of fall as cask tips over (assumed in analysis?). - Impact force as a result of a fall (assumptions in analysis do not appear realistic or conservative). - If area of impact is top of cask 45° away from trunnion (higher stresses may result). - Cask failures modes and failure criteria (not described; provide either conservative bounding values or flawed-cask drop test results). - Case II (similar remarks apply).
8.2.1.2.3 Cask Drop Accidents	1	<ul style="list-style-type: none"> - Cask drop accident analysis (has same deficiencies as cask tipping analyses).
8.2.1.2.6 Thermal Loads	6	<ul style="list-style-type: none"> - Experimental test results (provide additional information). - First test with Castor Ic (provide data on five parameters). - Cast iron metallurgical changes due to fire (provide evaluation; limitation on subsequent handling?). - Thermal isolation test (cask size?; internal heat load?). - Last sentence in paragraph (has ambiguities; provide activity release calculation). - Summary on activity releases (Regulatory Guide 3.48 requires discussion of dose consequences following each accident).
8.2.1.2.7 Overall Summary of Accident Analysis	8	<ul style="list-style-type: none"> - Analyses (summaries not included). - Fracture mechanics analysis (associate with component structural analysis). - Dynamic analysis of fuel basket (include structural analysis; provide material properties). - Cask outer surface stresses in drop accident (contrary to statement, high stresses can exist; should use finite element analysis; discuss thermal fatigue). - Details in Appendix 6 (define more precisely in main report). - K_{IC} values (cannot be justified by figure in Appendix 6; NRC insists that material properties be furnished). - Detecting maximum observable flaw sizes (type of capability that exists?). - Tipover and drop analyses (analyses questionable; derived K_{IC} values also questionable).
10.0 Operating Controls and Limits		
10.1 Proposed Operating Controls and Limits		
10.1.1 Contents of Operating Controls and Limits	2	<ul style="list-style-type: none"> - Cask environment helium pressure (acceptable limits?). - Partial pressure of air within cask environment (limits?).

TABLE 3.3. (contd)

Section of Topical Report	No. of Issues	Items Involved
10.1.2 Bases for Operating Controls and Limits		
10.1.2.1 Cask Surface Temperature Limit	1	- Cask surface temperature (shouldn't it be based on cask array rather than single cask?).
10.1.2.2 Cask Surface/Off-Surface Dose Rate Limit	1	- Cask shielding dose rate design values (state surface dose rate limit here and in table; correct distance for off-surface limit?).
10.1.2.3 Cask Tightness Control and Leakage Limits	1	- Helium leak rate (inconsistency in stated values).
10.1.2.4 Fuel Characteristic Limits	3	- Specific power (inconsistency in stated value). - Reference to "an average of the hottest fuel rod" (average of what is not defined). - Fuel rod maximum burnup limits (specify).
10.1.2.5 Siting Limitations	1	- Statements in this section (inadequate; need to address temperature limitations).
10.2 Development of Operating Controls and Limits		
10.2.2 Limiting Conditions for Operations		
10.2.2.2 Technical Conditions		
10.2.2.2.2 Cask Exterior Dose Rate Limits	1	- Single cask values provided (provide cask array data).
10.2.6 Operating Controls and Limits		
10.2.6.1 Tightness Monitoring	1	- Procedure in Step 5 (Action) here versus that under Corrective Actions of Section 8.1.1.4 (inconsistent).
Appendix II Quality Assurance Handbook for Castor Dry Storage Casks		
General	1	- QA handbook (many typing errors; need to correct).
III. Design Control		
III.4 Checking of Documents		
III.4.1.1 Checking Offices and Criteria to Check	2	- Criteria (need to include federal regulations). - Computer codes (need to be benchmarked and verified).
XII. Control of Measuring and Test Equipment	1	- Measuring and test equipment (describe methods for assuring confirmation to requirements).

TABLE 3.4. Examples of Some Issues^(a) Involved in Project M-35 (C-E)

Section of Topical Report	No. of Issues	Items Involved
General Comments	4	<ul style="list-style-type: none"> - Following of format in Regulatory Guide 3.48 (report is not sufficiently substantive in many areas including description of cask and safety-related appurtenances; failure criteria and analysis details; procedures for filling, sealing, lifting, and transporting cask and illustrations of models). - Cask description and analyses (expand report). - Report addressed only single cask (provide generic treatment of cask arrays). - Cited references (submit copies with report).
1.0 Introduction and General Description	1	<ul style="list-style-type: none"> - Purpose and general description of installation (too little information provided).
1.1 Introduction	1	<ul style="list-style-type: none"> - Principle design features (need additional information).
1.1.1 Type of Dry Storage Mode	1	<ul style="list-style-type: none"> - Dry storage mode (describe in general terms).
1.1.4 Description of Spent Fuel to be Stored	1	<ul style="list-style-type: none"> - Identify in general terms.
1.1.5 Waste Product	1	<ul style="list-style-type: none"> - Specify, if any.
1.2 General Description of the Installation	1	<ul style="list-style-type: none"> - Installation description (no information provided; provide information on principal design criteria, operating systems, structural features, etc.).
1.2.1 Principle Characteristics of the Site	1	<ul style="list-style-type: none"> - Principal site characteristics (provide range of parameters to envelope variety of sites).
1.2.2 Principal Design Criteria	1	<ul style="list-style-type: none"> - Design criteria (provide brief discussion).
1.2.3 Operating Systems	1	<ul style="list-style-type: none"> - Applicable to cask only.
1.2.4 Structural Features	1	<ul style="list-style-type: none"> - Structural features such as fuel supports, trunnions, etc. (provide short description).
1.2.5 Passive Decay Heat Dissipation System	1	<ul style="list-style-type: none"> - Need to provide summary description.
1.2.6 Fuel Handling	1	<ul style="list-style-type: none"> - Fuel handling is a site-specific activity (provide general description).
1.3 General System Description	3	<ul style="list-style-type: none"> - General description (should involve only at-reactor storage; provide detailed discussion of transportation and cask placement; provide generic at-reactor site layout). - Safety related systems (provide detailed information). - Cask heat transfer aspects (provide more information; provide details on heat transfer paths and primary design criteria).
1.4 Identification of Agents and Subcontractors	1	<ul style="list-style-type: none"> - Sources of data (identify consultants, testing laboratories, etc., if used).
1.5 Material Description By Reference	1	<ul style="list-style-type: none"> - Supportive data (list and supply all available documented sources).
2.0 Site Characteristics	1	<ul style="list-style-type: none"> - Site-specific parameters (provide range of values that envelope credible conditions).

(a) From Reference 38; the issues were also included in an attachment to Reference 39.

TABLE 3.4. (contd)

Section of Topical Report	No. of Issues	Items Involved
3.0 Principal Design Criteria		
3.1 Purpose of Installation	1	- Provide description.
3.1.1 Materials to be Stored	2	- Radiological characteristics of fuel to be stored (no information provided; is ORIGEN the data source?). - Spent fuel characteristics (for each parameter, need maximum or minimum allowable value; need base calculations on limiting parameter values).
3.1.2 General Operating Functions	1	- Decontamination of cask (will wastes be generated?; discuss onsite transportation of cask).
3.2 Structural and Mechanical Safety Criteria	2	- Various natural conditions affecting structural and mechanical integrity (need design bases completely and concisely stated). - Cask design (objective is safe confinement <u>and</u> prevention of degradation and gross rupture of spent fuel).
3.2.1 Tornado and Wind Loadings		
3.2.1.1 Applicable Design Parameters	1	- Actual design parameters for tornado and wind loadings (specify; provide justification where needed).
3.2.1.2 Determination of Forces on Structures	1	- Tangential and linear wind velocity assumptions (explain; provide analysis to support conclusions).
3.2.1.3 Ability of Structure to Perform Despite Failure of Structures not Designed for Tornado Loads	2	- Any other structure associated with the SSSC ^(a) (any in this category?). - Tornado and other missile effects (must be addressed; consider loss of fluid neutron shield).
3.2.2 Water Level (Flood) Design	1	- Envelope of credible flood conditions (assess effect of resulting forces on SSSC).
3.2.3 Seismic Design	1	- Vertical and horizontal ground accelerations (will SSSC topple unless tied down?; provide safety analysis if SSSC withstands this type of loading and if not, provide nontoppling case calculations).
3.2.4 Snow and Ice Loadings	1	- Envelope of credible snow and ice loadings (any effect on SSSC performance?).
3.2.5 Combined Load Criteria	1	- Loads and loading combinations on SSSC (define in detail; include load factors for each load component).
3.2.6 Thermal Loading	1	- Criteria (describe; include consideration of ambient air temperature range, solar insolation range, and temperature transients).
3.3 Safety Protection Systems		
3.3.1 General	1	- Long-term safe and secure containment (identify items that require special consideration; discuss vulnerability of safety-related components).
3.3.2 Protection by Multiple Confinement Barriers and Systems	1	- Multiple confinement barriers (describe criteria for design; for external natural phenomena and accidents, is zero radioactivity release a criterion?).

(a) Sealed surface storage cask (SSSC).

TABLE 3.4. (contd)

Section of Topical Report	No. of Issues	Items Involved
3.3.2.1 Confinement Barriers and Systems	2	<ul style="list-style-type: none"> - Primary and secondary confinement barriers (address criteria; discussion and figures provided are inadequate. Seals (how made and tested?; consequences of seal failure?; no system included to monitor seals on continuing basis; how is seal integrity verified?). - Fuel cladding as first barrier to release of fuel material (inspection provisions to assure that defective or waterlogged fuel is not placed in dry storage?).
3.3.3 Protection by Equipment and Instrumentation Selection		
3.3.3.1 Equipment	2	<ul style="list-style-type: none"> - The O-ring seal (design criteria for release of radioactivity?). - Thermocouple for monitoring cask temperature (itemize design criteria).
3.3.3.2 Instrumentation	1	<ul style="list-style-type: none"> - Seal integrity monitoring capability (needs to be added).
3.3.4 Nuclear Criticality Safety	1	<ul style="list-style-type: none"> - Mist density assumptions (provide explanation; if undermoderated case results in maximum k_{eff}, provide details).
3.3.5 Radiological Protection		
3.3.5.2 Shielding	2	<ul style="list-style-type: none"> - Collective doses (in manrem) per year in each area and for various operations (need to provide; also, provide detailed descriptions of repair and maintenance operations; consider direct radiation; provide cask surface source field information). - For generic use, single cask considered, as recommended by NRC (NRC recommends that detailed calculations for inline and square arrays of casks also be included).
3.3.6 Fire and Explosion Protection	3	<ul style="list-style-type: none"> - Criteria for 30-minute exposure to ambient temperature of 220°F (this should be in Section 3.2.6). - Loading criteria due to explosion (basis for shock wave assumption?). Meeting 10 CFR 72 criteria (provide calculations). Exposure to fire (criteria?). - Pressure relief valve (design criteria?; pressure level for relief?; conditions that would prevent functioning of valve?; was some release of water and ethylene glycol mixture considered?).
3.3.7 Materials Handling and Storage		
3.3.7.1 Spent Fuel Handling and Storage	1	<ul style="list-style-type: none"> - Cask contamination control (discussion needed; if contamination not anticipated, so state and provide support).
3.3.7.2 Radioactive Waste Treatment	1	<ul style="list-style-type: none"> - Need to provide discussion (if no radioactive waste anticipated, so state and provide support).

TABLE 3.4. (contd)

Section of Topical Report	No. of Issues	Items Involved
3.5 Decommissioning	3	<ul style="list-style-type: none"> - Neutron activation of cask materials (need discussion; consider activation products and expected activities; address cask decontamination and disposition; if cask released for unrestricted use, provide explanation and support). - Removal of fuel assemblies from sealwelded casks (describe method); cask internal surfaces (maintain smoothness?; minimize cracks and crevices?). - Removal of fuel assemblies at end of storage life (address possibility of having degraded fuel present); cask unloading (assuring that boron concentration in water is sufficient?); lowering of fuel and cask interior temperatures (include description of such measures).
4.0 Installation Design	1	<ul style="list-style-type: none"> - Conformance with design bases (provide detailed description of cask design and analyses; report lacks sufficient design and analysis data).
4.1 Summary Description	1	<ul style="list-style-type: none"> - Generic treatment of casks in multiple arrays (should be considered).
4.2 Structural Specifications	2	<ul style="list-style-type: none"> - Use of American Institute of Steel Construction Material of Steel Construction for cask design and fabrication (explain; ASME Code's rules appear more appropriate; are AISC requirements compatible with ASME requirements?). - Welding requirements (American Welding Society is an authoritative source; cite applicable AWS codes).
4.2.1.1 Materials of Construction	1	<ul style="list-style-type: none"> - Materials (describe all materials used).
4.2.1.2 Fabrication and Inspection	1	<ul style="list-style-type: none"> - Fabrication and inspection methods (discuss here rather than in Section 4.2.3.3.2).
4.2.1.3 Quality Assurance	1	<ul style="list-style-type: none"> - Quality assurance activities (itemize in tabular form).
4.2.2 Installation Layout	1	<ul style="list-style-type: none"> - Essential features of cask (provide engineering drawings; include detailed drawings of cask body and closure head, pressure tap locations, thermocouples, valves, trunnions, etc.).
4.2.3 Individual Unit Description	1	<ul style="list-style-type: none"> - SSSC (provide detailed description; describe its functions and design details).
4.2.3.1 Function	1	<ul style="list-style-type: none"> - Description in Section 1.3 (inadequate).
4.2.3.2 Components	1	<ul style="list-style-type: none"> - Safety-related components (describe those requiring analysis to determine design configuration; reference to Figure 4.2-1 with no further discussion is unacceptable; discuss safety aspects and limits).
4.2.3.3 Design Bases and Safety Assurance	1	<ul style="list-style-type: none"> - Structural elements that are important to safety (provide analyses summary; include component or structure description, magnitude and direction of applied forces, analysis method, maximum forces and deflections, applicable material properties, and safety margins).

TABLE 3.4. (contd)

Section of Topical Report	No. of Issues	Items Involved
4.2.3.3.1.1 (CE) General	2	<ul style="list-style-type: none"> - Flooding of cask with unborated mist (no details provided; state assumptions for mist densities and fuel temperatures used in criticality calculations); flooding of cask with unborated water (more details needed). - Credit for soluble boron in pool water (need to provide criticality calculations; reconsider need for additional neutron poison in cask storage racks).
4.2.3.3.2 (CE) Shielding	1	<ul style="list-style-type: none"> - Radiation shield analysis (how was analysis performed?; were specific neutron and gamma dose rates used?; if so, state and support rates).
4.2.3.3.3 (CE) Structural		
4.2.3.3.3.1 (CE) Design	5	<ul style="list-style-type: none"> - Pressure tap connections (confusing statements; are there different taps?); cask surface exposure rates [comments would be more relevant in Section 4.2.3.3.2 (CE)]. - Spent fuel storage containers (isn't AWS D1.1-1981 more applicable than AWS D14.1?). - Cask structural analysis (discuss possible failure modes and failure criteria; analysis must show that containment breach will not occur under assumed loading). - Cask body and welded seals (is brittle fracture possible failure mode?; if so, is material's fracture toughness adequate if structural imperfections exist?; if brittle fracture mode not considered, justify exclusion). - Cask drop and cask tip-over events (discuss effect on spent fuel in the cask; degree of fuel damage in such events?; if damage such as rupture of fuel cladding occurs, will cask be unloaded?; if significantly damaged fuel is stored, what is effect on thermal and criticality analyses in report?).
4.2.3.3.3.2 (CE) Fabrication	1	<ul style="list-style-type: none"> - Pressure tap connections (report states they are seal welded with closure plug, which is incompatible with other statements).
4.2.3.3.3.3 (CE) Seals	1	<ul style="list-style-type: none"> - Testing of volume between primary and secondary seal (how tested?; how is monitoring done if tap welded?); cask leak rate (what is it?; justify).
4.2.3.3.4 (CE) Thermal	1	<ul style="list-style-type: none"> - Table 4.2-3 (excludes solar isolation provision).
4.2.3.3.4.1 (CE) Bases	4	<ul style="list-style-type: none"> - Off-normal events for cask (provide justification for design base 1, is nonrupture of cask water jacket implied?). - Design base 3 says fuel cladding not damaged by temperature of 450°C (provide analysis and/or justification). - Ambient surroundings (add design base for it; address solar isolation effects). - Single cask addressed (add array of casks; if nitrogen used, reactivity with materials?; if inadequate nitrogen data available, use inert gas; expected cask pressure levels?).

TABLE 3.4. (contd)

Section of Topical Report	No. of Issues	Items Involved
4.2.3.3.4.2 (CE) Methodology		
4.2.3.3.4.2.1 (CE) Normal Conditions	1	- Analysis (provide additional information; include heat transfer between assemblies, maximum cladding temperature, etc.; 24 or 21 assemblies?).
4.2.3.3.4.2.2 (CE) Off-Normal	9	<ul style="list-style-type: none"> - This section should be incorporated in Section 8. - Analysis (describe in more detail; include major assumptions; need analysis and consequences of not maintaining water in neutron shield; describe consequences of fuel damage or verify fuel remains intact). - Clarify statement regarding falling of cask and rupture of water jacket. - Water-ethylene glycol mixture (extreme ambient temperature?; proportion of glycol required?). - Possibility of explosion (need to address). - Lightning (protection provided?; if not provided, need basis). - Loss of ethylene glycol (effect on fluid-to-outer jacket heat transfer?; thermal analysis should consider actual range of fluid mixtures). - Ethylene glycol vapor in presence of air (effect of possible spark?); accident analysis (discuss and substantiate breach of outer shell followed by explosion, if credible). - Recovery from off-normal conditions (provide details and analyses for events where design temperatures are exceeded).
4.3 Auxiliary Systems		
4.3.7 Communications and Alarm Systems	1	- Monitoring system for cask integrity (NRC concerned about lack of system; if seal failure occurs, how will operating personnel know accidental radioactivity release possible or imminent?).
4.3.9 Maintenance Systems	1	- Storage cask maintenance (needs to be considered; what happens in seal failure event?).
5.0 Operation Systems		
5.1 Operation Description		
5.1.1 Narrative Description	4	<ul style="list-style-type: none"> - Cask design (describe aspects that facilitate cask lifting, moving and emplacement). - Seal weld integrity (can it be checked?; if so, describe). - Leakage and structural deterioration (describe means for evaluating during storage). - Commitment to action (describe; include limits).
5.1.2 Flowsheet	3	<ul style="list-style-type: none"> - Sequence of operations and control (provide description). - Cask emplacement and periodic inspections (show sequence of operations; include radiation source terms). - Fuel acceptance and delivery operations (describe).
5.1.3 Identification of Subject of Safety Analysis		
5.1.3.1 Criticality Prevention	1	- Discussion under Section 4.2.3.3.1 should be here.

TABLE 3.4. (contd)

Section of Topical Report	No. of Issues	Items Involved
5.1.3.4 Instrumentation	2	<ul style="list-style-type: none"> - Use of pressure taps to monitor leakage (type of instrumentation required?); monitoring cask internal temperatures (describe in more detail; instrumentation required?; comply with Regulatory Guide 3.48). - Permanently installed instrumentation (NRC not convinced that it is not required; without instrumentation, how is seal failure detected?; seal integrity checked only in accident event?; if only periodic surveillance used, then add seal weld integrity checks to procedures).
5.2 Fuel Handling Systems		
5.2.2 Spent Fuel Storage	1	<ul style="list-style-type: none"> - At-reactor storage (describe operations).
5.2.2.1 Safety Features	1	<ul style="list-style-type: none"> - Commitment to action (describe; include limits).
5.3 Other Operating Systems		
5.3.1 Operating Systems		
5.3.1.3 Design Description	1	<ul style="list-style-type: none"> - Minimizing personnel exposures to radiation during operations and maintenance (provide description appropriate to single cask).
5.3.2 Component/Equipment Spares	1	<ul style="list-style-type: none"> - Maintenance and repair operations (describe here; include bases; address off-normal events and accidents; if above items not considered, so state and support).
5.4 Operation Support System		
5.4.1 Instrumentation and Control Systems	1	<ul style="list-style-type: none"> - Temperature and neutron shield monitoring systems/procedures (discuss in some detail); neutron shield radiation monitoring equipment (if specific equipment used, identify type and operation range).
5.6 Analytical Sampling	1	<ul style="list-style-type: none"> - Contingency for accidental release (need to consider); breach of cask integrity (discuss air sampling that could be performed).
6.0 Waste Confinement and Management		
6.1 Waste Sources	1	<ul style="list-style-type: none"> - Accidental release (discuss sources and contamination)
6.5 Radiological Impact of Normal Operations	1	<ul style="list-style-type: none"> - Gaseous effluents (need summary; assess gaseous effluents from a release).
7.0 Radiation Protection		
7.1 Ensuring that Occupational Radiation Exposures Are As Low As Is Reasonably Achievable (ALARA)		
7.1.1 Policy Considerations	1	<ul style="list-style-type: none"> - ALARA consideration during periodic inspection/surveillance and maintenance operations (discuss; how will regulatory guides be followed?).
7.1.2 Design Considerations	1	<ul style="list-style-type: none"> - Generic design considerations (describe; show how design minimizes occupational exposure; indicate how regulatory position 2 of Regulatory Guide 8.8 will be followed).

TABLE 3.4. (contd)

Section of Topical Report	No. of Issues	Items Involved
7.1.3 Operational Considerations	1	- Ensuring exposure is ALARA (the C-E discussion is incomplete; mention of at-reactor decontamination prior to storage appears relevant and needs to be added; references to regulatory position 4 of Regulatory Guide 8.8 and Regulatory Guide 8.10 are not relevant).
7.2 Radiation Sources		
7.2.1 Characterization of Sources	1	- Characterization of radiation sources (incomplete discussion; present all material).
7.2.2 Airborne Radioactive Material Sources	1	- Sources (some discussion needed; provide detailed discussion of gaseous effluent in release event).
7.3 Radiation Protection Design Features		
7.3.1 Installation Design Features	1	- Design features (discussion definitely needed; provide information on all sources stated in Section 7.2).
7.3.2 Shielding	1	- Shielding (insufficient discussion; substantiate stated specific surface dose rates and claims of reduced dose rates due to shield design; address shielding for all radiation sources; provide information about calculations).
7.4 Estimated Onsite Collective Dose Assessment	1	- Dose assessment (discussion definitely needed; address annual collective doses associated with periodic surveillance and repair or maintenance operations; provide bases, models, and assumptions for each dose assessment).
8.0 Accident Analysis		
8.1 Off-Normal Orientations	1	- Off-normal events (incomplete discussion; consider temperature monitoring system failure; consider seal leakage possibility; address radiological consequences associated with corrective or repair actions; if seal failure considered, provide data on effluent that might escape).
8.1.2 Radiological Impact from Off-Normal Operations	1	- Off-normal operations (some discussion needed; demonstrate capability of installation to operate safely within range of off-normal conditions).
8.2 Accidents	2	- Accident potential for proposed ISFSI ^(a) (provide rigorous analysis; identify accidents and causes and provide analyses of effects; describe recovery and steps to mitigate accidents). - Accident resulting in dose of 25 mrem or less beyond control area (if such accident not credible, demonstrate this fact).
8.2.1 Cask Drop		
8.2.1.1 Cause of Cask Drop	1	- Circumstances leading to cask drop (discuss).

(a) Independent Spent Fuel Storage Installation (ISFSI).

TABLE 3.4. (contd)

Section of Topical Report	No. of Issues	Items Involved
8.2.1.2 Accident Analyses of Cask Drop	1	- Cask drop analyses (describe here rather than in Section 4.2.3.3.1; level of damage to cask in accident (provide details; follow Regulatory Guide 3.48 procedure); damage to loaded spent fuel from cask drop (provide details and analysis; effect on its continued storage?; provisions for eventual removal of damaged fuel?; how would fuel be removed?; would it be canned after removal?).
8.2.2.2 Cask Tipping		
8.2.2.2.1 Cause of Cask Tipping	1	- Circumstances leading to cask tipping (discuss).
8.2.2.2 Accident Analyses for Cask Tipping	1	- Cask tipping (comments relating to Section 8.2.1.2 apply here).
8.2.4 Pressure Shock of 1 PSID	1	- C-E's statement regarding maintaining cask integrity (provide support).
8.2.8 Leak from Neutron Shield	1	- Radiological consequences of leak (substantiate statements).
8.2.9 Cask Drop	1	- C-E's statement concerning maintaining cask integrity (provide support; if result purely analytical, were codes benchmarked and have codes been accepted and used in previous NRC licensing reviews?).
8.2.11 Seal Leakage	1	- Seal leakage (comments above apply here also; leakage only possible following an accident?; if so, state and support).
10.0 Operating Controls and Limits		
10.1 Proposed Operating Controls and Limits	1	- Selection of probable subjects in this section (identify and justify; identify temperature limits for thermocouple measurements, describe criteria for limits, and consider adequacy of thermocouple location; specify cask outer surface temperature limits; specify cask surface dose rate limits).
10.1.1 Contents of Operating Controls and Limits	1	- Operating controls and limits (include both technical and administrative matters).
10.1.2 Bases for Operating Controls and Limits	1	- Areas identified in Section 10.1.1 (describe bases).
10.1.2.1 Fuel Characteristic Limits	2	- Fuel characteristic limits (specifically reference here for defining license conditions; provide specific power for reference fuel assembly). - Limit on maximum fuel cladding temperature in storage (establish limits on fuel characteristics and related site parameters that can be measured or otherwise verified).
10.1.2.2 Site Parameter Limits	1	- Site parameter limits used in cask design analysis (maximum and minimum ambient temperature and other data (provide information; summarize operating limits and relate to Section 4.2.3.4.2.1 in CENPD-270).
10.1.2.3 Cask Surface Temperature Limit	1	- Maximum cask surface temperature limit (provide data).
10.1.2.4 Cask Surface/Off-Surface Dose Rate Limit	1	- Limits of 10 CFR Part 72.67 (limits must be met, reference this CFR section).

TABLE 3.4. (contd)

Section of Topical Report	No. of Issues	Items Involved
10.1.2.5 Cask Tightness Control	1	- Requirements (10 CFR Parts 72.72 and 72.74 requirements referenced here should be so stated).
10.2 Development of Operating Controls and Limits	2	- Activities and conditions requiring operating controls and limits (refer to Section 72.33 of 10 CFR Part 72 for guidance on categories). - Areas identified under Section 10.1.1 (provide description under Sections 10.2.2.1, 10.2.2.2, and other pertinent subsections of 10.2).
10.2.1 Functional and Operating Limits, Monitoring Instruments and Limiting Control Settings		
10.2.2 Limiting Conditions for Operations		
10.2.2.1 Equipment	1	- Level of performance for all pertinent systems (establish lowest acceptable level).
10.2.2.2 Technical Conditions and Characteristics	1	- Allowable quantities associated with pertinent limiting conditions (specify; provide specific information for limiting condition definition)/
10.2.2.2.1 Fuel Characteristics	2	- Spent fuel assemblies being stored (provide data on initial enrichment, uranium weight, maximum burnup, and minimum decay time). - Fuel condition prior to storage (provide procedures and limits; provide procedures and limits for rejecting defective and/or waterlogged spent fuel; and provide procedures and limits on acceptance of fuel damaged in loaded cask).
10.2.2.2.2 Site Parameter Limits	1	- Site parameter limits (list applicable items).
10.2.2.2.3 Cask Surface Temperature	1	- Conditions (describe for SSSC only; state time to thermal equilibrium for cask after fuel loaded); cask surface temperature (measured value should not exceed predicted value; provide predicted values in tabular form).
10.2.2.2.4 Cask Surface/Off-Surface Dose Rate Limits	1	- Limits (provide data for single cask and representative cask arrays; predicted values should not exceed measured values).
10.2.3 Surveillance Requirements	1	- Operating controls and limits on periodic inspections of storage cask (need to address).
10.2.4 Design Features		
10.2.5 Administrative Controls	1	- Controls (site specific; provide only generic information).
11.0 Quality Assurance	1	- Reference to quality assurance program defined in CENPD-210-A Rev. 03 (acceptable).

TABLE 3.5. Examples of Some Issues^(a) Involved in Project M-33 (MHI/REA)

Section of Topical Report	No. of Issues	Items Involved
1.0 Introduction and General Description of Installation		
1.2 General Description of Installation	3	<ul style="list-style-type: none"> - Stated weight of cask (loaded or unloaded?). - Cask surface design dose rate (is 20 mrem/hr the maximum?). - Basket length (why shorter than cavity?).
1.3 General Systems Description	1	<ul style="list-style-type: none"> - What ASME BPV^(b) Code used?
1.3.2 Storage Arrays	2	<ul style="list-style-type: none"> - Effect of factors on cask array spacing? - Spacing optimization procedures?
1.3.3 Internal Environment	2	<ul style="list-style-type: none"> - Helium design pressure? - Possibility of air?
1.3.4.1 Drawings	2	<ul style="list-style-type: none"> - Provide 3 additional detail drawings (secondary cover, cask body weldment, removable trunnion). - Expunge proprietary stamp on 5 drawings.
1.4 Identification of Agents and Contractors	1	<ul style="list-style-type: none"> - Include 2 more contractors.
3.0 Principal Design Criteria		
3.1 Purposes of Installation		
3.1.1 Material to be Stored		
3.1.1.1 General Description	7	<ul style="list-style-type: none"> - Some fuel assemblies may exceed characteristics for acceptable storage. - Why is 8x8 BWR fuel excluded in table? - Principal design criteria (why isn't maximum decay heat rate for casks in an array included?). - Subcritical multiplication (excluded from table; if considered, include here; provide ORIGIN results); initial fuel enrichment (identify in Figure 3.1-5). - How was Fig. 3.1-4 obtained? - Justify or reference the basis for two equations. - Is neutron emission rate plot correctly drawn?
3.1.2 General Operating Functions	2	<ul style="list-style-type: none"> - Fuel claddings (how are they protected against degradation and gross rupture; what are design criteria to assure compliance?).
3.2 Structural and Mechanical Safety Criteria		
3.2.1 Tornado and Wind Loadings		
3.2.1.2 Determination of Forces on Structures	2	<ul style="list-style-type: none"> - Vulnerability of free-standing cask to tipping; response of cask to tornado versus seismic loadings?
3.2.1.3 Ability of Structures to Perform	2	<ul style="list-style-type: none"> - Mislabeled section numbers. - "Acceptable limits" for fuel rod temperature and surface dose rate (need to define).

(a) From References 62 and 63. Comments from DOE are in Reference 61; responses from MHI/REA are in Reference 171.

(b) Boiler and Pressure Vessel (BPV).

TABLE 3.5. (contd)

Section of Topical Report	No. of Issues	Items Involved
3.2.2 Water Level Flood Design	1	- Cask response to a range of water heights (would help in estimating safety margins).
3.2.5 Combined Load Criteria	2	- Method used to combine design loads, accident loads, and loads from natural phenomena? - Referenced section does not exist.
3.3 Safety Protection Systems		
3.3.2 Protection By Multiple Confinement Barriers and Systems		
3.3.2.1 Confinement Barriers and Systems	1	- Define "unacceptable release of radioactive products."
3.3.2.2 Ventilation-Offgas	2	- Define "no credible" conditions for radioactive release. - Define "very small magnitude" gas release.
3.3 Protection By Equipment and Instrumentation Selection		
3.3.3.2 Instrumentation	3	- What "features to provide testability" are provided? - How is pressure transducer system tested? - Describe the remote monitoring system.
3.3.4 Nuclear Criticality Safety	6	- Section needs revision and clarification (e.g., active fuel length, criticality calculational model). - Why analysis only done for 7x7 BWR fuel? - Was analysis for 8x8 BWR fuel done? - Significant difference between calculated and measured K effective: why, and is model validity reduced? - Water temperature and density assumptions used? - Explain benchmarking (measured rods but analyzed homogeneous water-fuel mixture).
3.3.5 Radiological Protection		
3.3.5.2 Shielding	7	- Determination of distances and occupancy times in Table 3.3-12 (how done?). - Are data in Table 3.3-11 conservative? - Dose rates in Table 3.3-12 (uncertainties; post-irradiation time assumption?). - Collective dose consequences associated with cask repair, maintenance and decommissioning? - Distribution of occupancy times, in particular for 5% and 95% confidence bounds? - Annual inspection exposure (number of inspections assumed?). - Personnel exposure units (rem as stated, or person-rem?).
3.3.5.3 Radiological Alarm Systems	2	- Cask malfunction alarms? - Internal cask pressure alarm (is there one?; if not, how are personnel alerted?).
3.3.6 Fire and Explosion Protection	1	- Was ethylene glycol in neutron shield considered?
3.4 Classification of Structures, Components, and Systems	1	- Provide safety and quality assurance classifications.
3.5 Decommissioning Considerations	3	- Consideration of neutron activation of cask and fuel basket materials? - Activation products and activities? - Dose-commitment for removal of spent fuel?

TABLE 3.5. (contd)

Section of Topical Report	No. of Issues	Items Involved
4.0 Installation Design	1	- Inadequate reference to cask and component drawings.
4.2 Storage Structures	1	- Does skid slightly tilt cask from horizontal orientation? Explain.
4.2.3 Individual Unit Description	1	- Safety functions of each component (provide description for each component noted in Section 3.4).
4.2.3.2 Components	5	- Is neutron shield solution borated? - Burst disk maintenance period? - Qualified life of pressure transducer? - Function and lifetime of silicon rubber O-rings? - Boron-10 content.
4.2.3.2.2 Basket	2	- Zero clearance for basket/inner wall implied? - Thermal effects if no clearance exists?
4.2.3.2.4 Internal Environment	2	- What gas to be used (helium or air)? - Helium pressure?
4.6 Cathodic Protection	1	- Corrosion?
5.0 Operation Systems		
5.1 Operation Description		
5.1.1 Narrative Description	1	- Cask condition (how is it routinely evaluated?).
5.1.1.2 Cask Removal from Skid	2	- Lifting yoke description and drawing? - Trunnion bolt torque?
5.1.1.4 Primary Cover Removal	1	- Damage to O-ring grooves?
5.2 Fuel Handling Systems		
5.2.2 Spent Fuel Storage		
5.2.2.1 Safety Features	1	- Safety-related features for normal and abnormal conditions (what are they?; action commitment limits?).
5.3 Other Operational Systems		
5.3.2 Component/Equipment Spares	2	- Spare or alternative equipment installation provisions? - Include general handling procedures.
7.0 Radiation Protection		
7.1 Ensuring That Occupational Radiation Exposures Are As Low As Reasonably Achievable (ALARA)	2	- Substantiate compliance with 10 CFR Part 72.3. - Radiological protection (what makes cask design ALARA?).
7.2 Radiation Sources		
7.2.1 Characterization of Sources		
7.2.1.1 General Description	4	- ORIGIN output results for 5-year postirradiation time (why not presented?; need to list gamma and neutron strengths and energies by isotopes). - References for flux-to-dose response functions? - Radiation sources from cladding and structural materials (included in Table 7.2-1?; if not, so state and justify).

TABLE 3.5. (contd)

Section of Topical Report	No. of Issues	Items Involved
		<ul style="list-style-type: none"> - Gamma spectrum in Table 7.2-3 (is it indicated that it is for specific time after irradiation?; what are boundaries for gamma energy groups?).
7.2.2 Airborne Radioactive Material Sources	1	<ul style="list-style-type: none"> - Expected gaseous effluents in accidental release?
7.3.2 Shielding		
7.3.2.1 Radiation Shielding Design Features	5	<ul style="list-style-type: none"> - Time-dependent items in Table 7.3-1 such as dose rate, heat generation rate, and temperature (identify when reported value applies). - Neutron shield height (what is correct value?; what was used in MORSE input?). - Active fuel length for 7x7 BWR fuel (value stated is for 8x8 BWR fuel). - Correct cask and fuel dimensional discrepancies. - Radiation streaming from penetrations?
7.3.2.2 Shielding Analysis	2	<ul style="list-style-type: none"> - Correspondence of dose levels and computer codes used? - Benchmark validations and code biases (need to discuss)?
7.3.2.2.1 Shielding Methodology		
7.3.2.2.2 Shielding Analytical Models	2	<ul style="list-style-type: none"> - Basis for axial burnup distribution in Table 7.3-6? - Atomic density units in Table 7.3-2 (atoms/barn-cm, not atoms/barn-cm²).
7.3.2.2.3 Shielding Results	4	<ul style="list-style-type: none"> - Uncertainties in code results? - Neutron flux distribution in Figure 7.3-10 (how obtained?). - Surface dose rate in Figure 7.3-8 (what are the uncertainties?). - Substantiate decreases in 10-yr and 50-yr dose rates.
7.3.2.3 Dose Assessment	12	<ul style="list-style-type: none"> - External dose rate variation with time in Figure 7.3-10 (how determined?). - External dose rates versus burnup and time in Figures 7.3-11 and 7.3-13 (how determined?). - Doses at various cask dose points in Figure 7.3-14 (how determined; only vertical orientation?; explain). - Dose at given point (all casks included?); cask self-shielding effects (were they considered?). - O-ring seal dose versus time (how calculated?). - Shielding effect of trunnion plates (method used to determine surface dose rate reductions?). - Cobalt-60 activation of stainless steel cask shells (how calculated?). - Maximum dose rate from cobalt-60 activation? - Is 75 rem/hr for 1 or 52 BWR fuel assemblies? Does it apply at 5-year postirradiation time? - Secondary cover outside surface dose (method used to calculate reduction?). - Basis for deeming cobalt-60 activation of fuel cladding insignificant. - Multiplying factors in Figure 7.3-14 (how determined?).
7.3.2.4 Accident Event	2	<ul style="list-style-type: none"> - Loss-of-neutron shield dose rates (how determined?). - Doses at P7 and P12 less than those at P8 and P13 (why?; does air/ground scattering affect doses?).

TABLE 3.5. (contd)

Section of Topical Report	No. of Issues	Items Involved
7.4 Estimated Onsite Collective Dose Assessment	3	<ul style="list-style-type: none"> - Air/ground scattering (how calculated?) - Decay of radiation sources (how determined?). - Cask orientation (explain).
7.4.1 General Description	6	<ul style="list-style-type: none"> - Results showing distance effects (how were effects calculated?). - Dose and dose rate at site boundary (substantiate why conservative). - Neutron and gamma source decay in Figure 7.4-13 (how calculated?). - Annual cask surface dose rate in Figure 7.4-8 (is it correct?; appears off by factor of 1000). - Dose rates in Table 7.4-3 (how determined?; based on single dose point, sum of several dose points, or an average?). - Cask array spacing (what is it?; provide figures for horizontally and vertically oriented cask arrays).
8.0 Accident Analyses		
8.1 Normal and Off-Normal Operations		
8.1.1 Normal Operations	1	<ul style="list-style-type: none"> - Neutron shield outer surface thermal stresses during precipitation (why not considered?).
8.1.1.2 Material Properties	3	<ul style="list-style-type: none"> - Properties and design calculations for basket (need for both normal and accident conditions). - Neutron shield (is effect of borated water included?). - Effect on spent fuel with horizontal storage (need to address).
8.1.1.3 Containment Vessel	1	<ul style="list-style-type: none"> - Design pressure of 50 psig (what is basis?); hydrostatic test data (where is it?).
8.1.1.4 Lifting Devices	1	<ul style="list-style-type: none"> - Need justification for trunnion bolt strength statement.
8.1.1.6 Cask Temperatures	6	<ul style="list-style-type: none"> - Ranges of conditions and parameters for heat transfer calculations (what are ranges?; need model description and basis for models and/or parameters). - Fuel rod temperature versus design criteria (likelihood of exceeding design criteria during normal operations?; sensitivity to models and parameters?). - Neutron shield temperature versus recommended operating limit (is 5°F sufficient margin of safety?). - Derating of maximum decay heat of stored fuel in cask array (why not discussed in Chapter 3?). - Maximum fuel cladding temperature if 20.8 kW per cask allowed in array (what is it?; will it exceed design limit?). - Neutron shield temperature of cask in array (vendor operating limits exceeded if 20.8 kW per cask allowed?).
8.1.1.7 Thermal Expansion	2	<ul style="list-style-type: none"> - See Section 4.2.3.2.2.
8.1.1.8.1 Containment Vessel Analysis	3	<ul style="list-style-type: none"> - Containment vessel flange and penetration analysis (why ASME BPV Code, Subsection NB, not used?). - Provide thermal cycling analysis. - Differential expansion stresses on bolts?

TABLE 3.5. (contd)

Section of Topical Report	No. of Issues	Items Involved
8.1.1.8.3 Skid Stress Analysis		
8.1.1.8.3.1 Lift Lug	1	- Shear stress equation (lug area incorrect).
8.1.1.8.4 Thermal Analysis	5	<ul style="list-style-type: none"> - Sensitivity of temperature profile calculations to interface conductances? - Interface conductances (expected variation?; changes due to thermal cycling over 20-year license period?). - Outer surface heat transfer coefficient (basis for selection?; other correlations considered; if so, why rejected?). - Correlation used (confirmation as conservative estimate?). - Thermal analysis (was it performed for 7x7 BWR fuel?).
8.1.1.8.4.1 Thermal Model		
8.1.1.8.4.1.1 Fuel Region	3	<ul style="list-style-type: none"> - Emissivity for Zircaloy-clad fuel rods (value not appropriate for temperatures of $\leq 800^{\circ}\text{F}$). - Justify use of parameters for thermal conduction and conductive heat transfer. - Maximum fuel rod during vacuum drying?
8.1.1.8.4.1.2 Copper/Stainless Steel Shells	1	- Estimated basket/vessel gap (justify value).
8.1.1.8.4.1.3 Lead/Stainless Steel Interfaces	1	- Effect of thermal cycling on load contact pressure and heat transfer coefficient (provide information).
8.1.1.8.4.2 Experimental Temperatures	1	- Design calculations (support from thermal test?).
8.1.1.8.4.2.1 Results of Experimental Measurements of Centerline-to-Surface Temperature Differences		
8.1.1.8.4.2.1.2 Measurement Procedures	6	<ul style="list-style-type: none"> - Measurement of aluminum pin emissivity in test? - Actual heater pin output/unit length? - Measurement of heater current? - Heater resistance uncertainty? - Measurement of heater resistance for wires and pins? - Provide two dimensions in figure.
8.1.1.8.4.4 Neutron Shield Expansion Tank		
8.1.1.8.4.4.1 Pressure	1	- Discrepancy in "L" value in equation and table.
8.1.1.8.4.4.2 Pressure Stress	2	<ul style="list-style-type: none"> - Neutron shield cavity (bending stress analysis?; fatigue failure possibility?). - Ends and joints of neutron shields (ASME stress intensities?).
8.1.1.8.4.6 Thermal Expansion Analysis	1	- Lead and fuel basket radial expansion (provide information).
8.1.1.9 Off-Normal Operations		
8.1.1.9.1 Leak	1	- Sealing effectiveness of O-rings and secondary cover weld (how assured?).
8.1.1.9.1.1 Cause of Event	1	- Mechanical damage to cask (implied impairments to sealing system?).
8.1.1.9.1.2 Detection of Event	1	- Timely detection of cask pressure change (how assured?).

TABLE 3.5. (contd)

Section of Topical Report	No. of Issues	Items Involved
8.1.1.9.1.3 Analysis of Effects and Consequences	4	<ul style="list-style-type: none"> - Fuel cladding leak may result from a number of causes (include analysis and effects of events). - Cladding degradation (substantiate that activity level is within allowable limits). - Fuel assembly internal gas inventory estimate (explain in detail). - Fuel rod internal gas pressure (provide data).
8.1.1.9.1.4 Corrective Action	1	<ul style="list-style-type: none"> - Vacuum bonnet in place (plug accessibility?; ability to conduct operations?).
8.1.1.9.2 Fission Product Gas Pressure		
8.1.1.9.2.1 Cause of Event	3	<ul style="list-style-type: none"> - Cask internal environment (can monitoring system fail?). - Masking of small pressure changes (diurnal pressure variations?). - Gases inside cask (sampling procedure?).
8.1.1.9.2.3 Analysis of Effects and Consequences	1	<ul style="list-style-type: none"> - Include applicable parts of comments from third and fourth issues in Section 8.1.1.9.1.3.
8.1.1.9.3 Failure of Instrumentation	4	<ul style="list-style-type: none"> - Implication that REA cask be licensed without instrumentation (intent of this section?). - "...no need for instrumentation which penetrates cask containment" (provide basis for this assertion). - Test calibration charts and their accuracy (explain development). - Thermal and pressure instrumentation (provide information; discuss maintenance and replacement).
8.1.1.9.3.1 Cause of Event	1	<ul style="list-style-type: none"> - Cask instrumentation (vulnerability to possible failure mechanisms?).
8.1.1.9.3.2 Detection of Event	1	<ul style="list-style-type: none"> - Excursion from expected readings (failure of instrumentation or actual change in conditions?; describe procedure for identifying failure).
8.2.1 Loss of Neutron Shield ^(a)		
8.2.1.1 Cause of Accident		
8.2.1.2 Accident Analysis	1	<ul style="list-style-type: none"> - Apparent error in Figure 8.2-1 (explain).
8.2.1.3 Accident Dose Calculations	7	<ul style="list-style-type: none"> - Population cumulative and annual doses in mrem (should be in person-rem). - Dose variation with time (clarify statement). - Distance from accident (single person and 100,000 persons at same point?). - Distance from accident (at cask surface or other?). - Neutron shield loss and fire accidents (cask orientation?; how were doses calculated?; provide data; dose response factors used?). - Neutron shield repair (dose consequences to individuals?). - Neutron shield loss and leaking fission product gases (cumulative effect?).

(a) Preceding page in document was missing in the NRC Public Document Room copy. The missing page probably covers "Analysis of Effects and Consequences," "Radiological Impact from Off-Normal Operations," and "Accidents."

TABLE 3.5. (contd)

Section of Topical Report	No. of Issues	Items Involved
8.2.2 Fire		
8.2.2.1 Cause of Accident		
8.2.2.2 Accident Analysis	1	- Failure of O-ring seal (potential problem regarding gaseous release?).
8.2.3 Tip Over	2	- Tipping of one cask into another (impact time appears incorrectly calculated). - Dynamic analysis of casks hitting each other needed.
8.2.4 Drop	3	- Cask drop cases (perform dynamic impact analyses). - Damage to fuel basket (include analyses). - Detailed ASME Code stress quantities (include detailed calculations).
8.2.5 Flood	3	- Cask tipping during flooding (provide calculations showing cask will not tip). - ASME Code limits and design properties used (should be design by analysis rather than formula techniques). - Thermal and hydraulic stresses for flooded condition (calculate and combine).
8.2.4.6 Cask Seal Leakage		
8.2.4.6.2 Accident Analysis	3	- Accidents resulting in breaching of more than one cask (basis for excluding?). - Specific isotopes released during accident (exact quantities assumed?). - Gaseous fission product contents (100% release assumed?).
8.2.4.6.3 Accident Dose Calculations	2	- Dose at site boundary (assumptions used?). - Fifty-mile population dose (assumptions used?).
8.2.4.7 Tornado Missiles	2	- Massive missile impact (impact loads and times incorrectly calculated; need dynamic impact analyses). - Steel sphere impact calculation is in error.
8.2.4.9 Earthquake	3	- Analysis is nonconservative (include vertical acceleration or perform dynamic analysis). - Point of rotation in seismic analyses (demonstrate point does not change significantly during rotation). - Tipping of cask storage array following an earthquake (describe effects and consequences).
8.2.4.12.4.2 Containment Vessel Shell Deflection	3	- Lead pouring and cooldown processes (provide description). - Lead annulus region (provide diagram; tabulate dimensions, temperatures, and ASME stress intensities for critical points). - Critical points during and/or following fire event (provide temperature, thermal expansion, and stress analysis for worst conditions; use BPVC Section NB3133 for collapse evaluation).
8.3 Site Characteristics Affecting Safety Analysis	2	- Credible conditions under which ISFSI ^(a) will operate (establish ranges of site-specific parameters). - Operating limits (summarize in "10.1.2.2 Site Parameter Limits"; provide site parameter limits used in cask design analysis).

(a) Independent Spent Fuel Storage Installation (ISFSI).

TABLE 3.5. (contd)

<u>Section of Topical Report</u>	<u>No. of Issues</u>	<u>Items Involved</u>
9.0 Conduct of Operations		
9.2.2 Test Program Description	2	<ul style="list-style-type: none"> - Final inspection of construction and contact resistance (conduct test to verify). - Final inspection of basket (verify presence of boron poison).
10.0 Operating Controls and Limits		
10.1 Proposed Operating Controls and Limits		
10.1.1 Contents of Operating Controls and Limits	9	<ul style="list-style-type: none"> - Helium pressure in cask environment (acceptable limits?). - Air within cask environment (limiting partial pressure?). - Fission product gases released from fuel to cask internal environment (maximum allowable quantity?). - Required periodic inspections (frequency?). - Surface dose rate (define "somewhat arbitrary limits"; what is operational dose limit?). - Temperature limit of 271°F (does it apply to single cask or array of casks?). - Unprotected cask surface temperature of 271°F (consistent with minimum safety standards?). - Cask spacing to maintain temperature limit (minimum limits?). - Cask lifting height (limit?).
10.1.2 Bases for Operating Controls and Limits	1	<ul style="list-style-type: none"> - Operating controls and limits stated in Section 10.1.1 (need to provide bases).
10.1.2.1 Fuel Characteristic Limits	1	<ul style="list-style-type: none"> - Neutron and gamma sources (insufficient detail provided).
10.1.2.4 Cask Surface/Off Surface Dose Rate Limit	1	<ul style="list-style-type: none"> - Cask surface dose rates during normal and accidental events (reiterate and summarize here).
10.1.2.5 Cask Tightness Control	2	<ul style="list-style-type: none"> - Cask pressure monitoring system (limit on precision?). - Vent and drain part plug leaks (actual limits?).
10.2 Development of Operating Controls and Limits		
10.2.2 Limiting Conditions for Operation		
10.2.2.2.1 Fuel Characteristics	1	<ul style="list-style-type: none"> - Fuel characteristics (state in form of operating limits).
10.2.3 Surveillance Requirements	2	<ul style="list-style-type: none"> - Temperature and pressure sensing instruments (calibration?). - Cask internal environment (provisions for sampling?).
10.2.6 Suggested Format for Operating Controls and Limits	1	<ul style="list-style-type: none"> - Format (follow Regulatory Guide 3.48).

TABLE 3.6. Examples of Some Issues^(a) Involved in Project M-36 (MHI/REA)

Section of Topical Report	No. of Issues	Items Involved
1.0 Introduction and General Description of Installation		
1.2 General Description of Installation	5	<ul style="list-style-type: none"> - Stated weight of cask (loaded or unloaded?). - Dose rate at six feet from cask surface (instead, need cask surface dose rate). - Figure 1.2-2 on page 1.3-1 (should be Figure 1.2-1). - Overall dimensions of basket (why not shown in Figure 1.2-2?). - Basket configuration (inconsistency between Figure 1.2-2 and Drawing 1114-P-1300).
1.3 General Systems Description	1	<ul style="list-style-type: none"> - What ASME BPV^(b) Code used?
1.3.2 Storage Arrays	2	<ul style="list-style-type: none"> - Effect of factors on cask array spacing? - Spacing optimization procedures?
1.3.3 Internal Environment	1	<ul style="list-style-type: none"> - Helium design pressure?
1.4 Identification of Agents and Contractors	1	<ul style="list-style-type: none"> - Discussion (incomplete; include 2 more contractors?).
2.0 Site Characteristics	1	<ul style="list-style-type: none"> - Ranges of site-specific parameters that envelope credible conditions (need to establish; see comments in Section 10).
3.0 Principal Design Criteria		
3.1 Purposes of Installation		
3.1.1 Material to be Stored		
3.1.1.1 General Description	7	<ul style="list-style-type: none"> - Some fuel assemblies may exceed characteristics for acceptable storage. - Table 3.1-1 excludes 14x14 and 17x17 PWR fuel (why?; implication is that 15x15 is only array for which license is sought). - Maximum decay heat rate for casks in an array (why is this not reflected in principal design criteria?). - Subcritical multiplication (excluded from Table 3.1-2; if considered, include here). - Subcritical multiplication in Figure 3.1-1 (how calculated?). - Source term bases (what are they?; why not explicitly stated on Figures 3.1-1 to 3.1-5?). - Figure 3.1-5 (meaning of units kW/BWR?).
3.1.1.2 End-of-Burnup Enrichment	1	<ul style="list-style-type: none"> - Exponential equations (1) and (2) [explain empirical development].
3.1.2 General Operating Functions	2	<ul style="list-style-type: none"> - Fuel claddings (how are they protected against degradation and gross rupture; what are design criteria to assure compliance?). - Vertical cask storage orientation (explain the corrosion-resistant plate).
3.2 Structural and Mechanical Safety Criteria		
3.2.1 Tornado and Wind Loadings		

(a) From Reference 60.

(b) Boiler and Pressure Vessel (BPV).

TABLE 3.6. (contd)

Section of Topical Report	No. of Issues	Items Involved
3.2.1.2 Determination of Forces on Structures	2	- Vulnerability of free-standing cask to tipping; response of cask to tornado versus seismic loadings?
3.2.1.3 Ability of Structures to Perform	1	- "Acceptable limits" for fuel rod temperature and surface dose rate (need to define); loss of neutron shield (can it be repaired as quickly as implied?).
3.2.2 Water Level Flood Design	1	- Cask response to a range of water heights (would help in estimating safety margins).
3.2.3 Seismic Design		
3.2.3.1 Input Criteria	1	- Minimum natural frequency (which stated value is correct, 37.7 or 32 hertz?; Regulatory Guide 1.60 says 33 hertz; if 32 hertz used, any change in seismic considerations?).
3.2.3.2 Seismic-Systems Analysis	1	- Seismic system analysis (discussion missing; if required information elsewhere, so state).
3.2.5 Combined Load Criteria	4	- Method used to combine design loads, accident loads, and loads from natural phenomena? - Referenced section does not exist. - Plastic yielding of stainless steel (what section of ASME Code limits this to 40%?). - Design criterion for fatigue endurance limit (limit in Table 3.2-1 only applicable to high cycle fatigue).
3.3 Safety Protection Systems		
3.3.2 Protection By Multiple Confinement Barriers and Systems		
3.3.2.1 Confinement Barriers and Systems	2	- Define "unacceptable release of radioactive products." - Accident scenarios (should refer to entire Section 8.2, not just 8.2.1).
3.3.2.2 Ventilation-Offgas	3	- Define "no credible" conditions for radioactive release. - Define "very small magnitude" gas release. - Subjective conclusions (would be more relevant to provide design criteria).
3.3 Protection By Equipment and Instrumentation Selection		
3.3.3.2 Instrumentation	3	- What "features to provide testability" are provided? - How is pressure transducer system tested? - Describe the remote monitoring system?
3.3.4 Nuclear Criticality Safety	5	- Why analysis only done for 15x15 PWR fuel? - Criticality analysis (should be for final design). - Analysis (performed for 14x14, 16x16, and 17x17 PWR fuel assemblies?). - Significant difference between calculated and measured K effective: why, and is model validity reduced? - Discrete fuel rods used in criticality analysis benchmark model (shouldn't cask fuel's criticality analysis also use discrete fuel rods?).

TABLE 3.6. (contd)

Section of Topical Report	No. of Issues	Items Involved
3.3.5 Radiological Protection		
3.3.5.2 Shielding	6	<ul style="list-style-type: none"> - Determination of distances and occupancy times in Table 3.3-12 (how done?). - Are data in Table 3.3-11 conservative? - Dose rate in Table 3.3-12 (uncertainties?). - Collective dose consequences associated with instrumentation attachment, cask repair, maintenance, and decommissioning? - Distribution of occupancy times, in particular for 5% and 95% confidence bounds? - Personnel exposure units (rem as stated, or person-rem?).
3.3.5.3 Radiological Alarm Systems	2	<ul style="list-style-type: none"> - Cask malfunction alarms? - Internal cask pressure alarm (is there one?; if not, how are personnel alerted?).
3.3.6 Fire and Explosion Protection	2	<ul style="list-style-type: none"> - States "there are no explosive gases in the fission product gas" in this section but Section 8.2.8.2 says "only negligible quantities of explosive gases will be present in fission product gases" (which statement is correct?; if latter meant, what is meant by "negligible?"). - Was ethylene glycol in neutron shield considered?
3.4 Classification of Structures, Components, and Systems	1	<ul style="list-style-type: none"> - Provide safety and quality assurance classifications.
3.5 Decommissioning Considerations	3	<ul style="list-style-type: none"> - Consideration of neutron activation of cask and fuel basket materials? - Activation products and activities? - Dose-commitment for removal of spent fuel?
4.0 Installation Design		
4.2 Storage Structures		
4.2.3 Individual Unit Description	1	<ul style="list-style-type: none"> - Safety functions of each component (provide description for each component noted in Section 3.4).
4.2.3.1 Materials for Construction	2	<ul style="list-style-type: none"> - Boron-10 content of 0.03 grams per square centimeter (does this refer to 1100 aluminum alloy or to matrix and cladding?). - Carbon steel skid (is there potential for corrosion?).
4.2.3.2 Cask Body and Containment Vessel	1	<ul style="list-style-type: none"> - Statement that "neutron shield will normally be filled at fabrication site" (not always filled at fabrication site?; is reactor only other alternative for filling location?).
4.2.3.3 Basket	1	<ul style="list-style-type: none"> - Thermal effects if no basket/inner wall clearance exists?
4.2.3.5 Internal Environment	1	<ul style="list-style-type: none"> - Basket employs dissimilar metals (is there significant galvanic corrosion potential?).

TABLE 3.6. (contd)

Section of Topical Report	No. of Issues	Items Involved
4.5 Shipping Cask Repair and Maintenance	5	<ul style="list-style-type: none"> - Cask repair and maintenance operations (provide discussion, if only from generic viewpoint; considerations to control contamination and reduce occupational exposure?); loss of neutron shield and leakage at penetrations (what do they constitute?; how repaired?). - Neutron shield rupture disk (how maintained?; is disk replaced annually?). - Rupture disk design pressure (what is tolerance?). - Pressure transducer (what is qualified life?; maintenance or replacement required?). - Silicon rubber O-rings (describe function and lifetime in thermal and radiation environment).
4.9 Appendix	1	<ul style="list-style-type: none"> - Secondary cover, cask body weldment, and removable trunnion (provide additional detail drawings).
5.0 Operation Systems		
5.1 Operation Description		
5.1.1 Narrative Description	2	<ul style="list-style-type: none"> - Cask condition (how is it routinely evaluated?). - Drying cask after it is filled with fuel (vacuum process used?; if so, provide thermal analysis to determine maximum fuel cladding temperature).
5.1.1.2 Cask Removal from Skid	2	<ul style="list-style-type: none"> - Lifting yoke description and drawing? - Trunnion bolt torque?
5.1.1.4 Primary Cover Removal	1	<ul style="list-style-type: none"> - Damage to O-ring grooves?
5.1.1.5 Cask Preparation for Loading Operation	2	<ul style="list-style-type: none"> - Expansion tank side plug not on drawing in Section 4.9; appears to be 6.0 in. above cask bottom (is this correct?; why so difficult to locate?). - Figure 5.1-1 (inaccurately drawn?; inconsistency in neutron shell-to-liquid level distance).
5.1.1.6 Cask Lowering Into Pool	1	<ul style="list-style-type: none"> - Fuel inspection techniques referenced in Items 38 and 39 (what are the techniques?; provide discussion).
5.1.1.8 Cask Lifting From Pool	2	<ul style="list-style-type: none"> - Item 59 and Item 61 (one implies secondary cover in place and other requires that secondary cover be positioned on cask; which is correct?). - Helium in cavity (optional or required?); "if cask is to be stored with helium coolant in cavity" (isn't this statement in error?).
5.1.1.9 Cask Transport to On-Site Storage Area	1	<ul style="list-style-type: none"> - Pressure transducer installation (dose consequences not described in Section 3.3.5.2 or Section 7.4.1; why?).
5.1.2 Flowsheets	1	<ul style="list-style-type: none"> - Cask handling flowsheet (need to provide; consolidate major functions described in Section 5.1.1 and Table 3.3-11).
5.1.3 Identification of Subjects for Safety Analysis	2	<ul style="list-style-type: none"> - Cask criticality prevention (provide summary description; refer to and use material in Section 3.3.4). - Instrumentation and maintenance (same comment above applies).
5.2 Fuel Handling Systems		
5.2.2 Spent Fuel Storage		
5.2.2.1 Safety Features	1	<ul style="list-style-type: none"> - Safety-related features for normal and abnormal conditions (what are they?; action commitment limits?).

TABLE 3.6. (contd)

Section of Topical Report	No. of Issues	Items Involved
5.3 Other Operational Systems	1	- Unloading fuel if cask is to be repaired or decommissioned (provide detailed description of procedure; in particular address gas sampling, cask cool-down, cask flushing and analysis of cask cavity effluent to determine if fuel damage occurred, weld removal, and personnel protection).
5.3.2 Component/Equipment Spares	1	- Spare or alternative equipment installation provisions?
5.4 Operation Support Systems		
5.4.1 Instrumentation and Control Systems	1	- Pressure transducer system instrumentation (provide discussion; address monitoring of safety features to ensure adequate safety).
7.0 Radiation Protection		
7.1 Ensuring That Occupational Radiation Exposures Are As Low As Reasonably Achievable (ALARA)	2	- Substantiate compliance with 10 CFR Part 72.3. - Radiological protection (what makes cask design ALARA?).
7.2 Radiation Sources		
7.2.1 Characterization of Sources		
7.2.1.1 General Description	4	- ORIGEN output results for 5-year postirradiation time (why not presented?); need to list gamma and neutron strengths and energies by isotope). - References for flux-to-dose response functions? - Radiation sources from cladding and structural materials (included in Table 7.2-1?; if not, so state and justify). - Gamma spectrum in Table 7.2-3 (is it indicated that it is for specific time after irradiation?; what are boundaries for gamma energy groups?).
7.2.2 Airborne Radioactive Material Sources	1	- Expected gaseous effluents in accidental release?
7.3.2 Shielding		
7.3.2.1 Radiation Shielding Design Features	3	- Time-dependent items in Table 7.3-1 such as dose rate, heat generation rate, and temperature (identify when reported value applies). - Configuration (address only the one for which the license is being sought; reference to earlier designs is confusing and irrelevant; dimensions in Figure 7.3-1 and Table 7.3-1 do not agree). - Distance from cask top to neutron shield shell top (values in two figures do not agree; what is correct value?).
7.3.2.2 Shielding Analysis	2	- Correspondence of dose levels and computer codes used? - Benchmark validations and code biases (need to discuss).
7.3.2.2.1 Shielding Methodology		
7.3.2.2.2 Shielding Analytical Models	2	- Basis for axial burnup distribution in Table 7.3-7? - Atomic density units in Table 7.3-2 (atoms/barn-cm, not atoms/barn-cm ²).

TABLE 3.6. (contd)

Section of Topical Report	No. of Issues	Items Involved
7.3.2.2.3 Shielding Results	7	<ul style="list-style-type: none"> - Uncertainties in code results? - Neutron flux distribution in Figure 7.3-10 (how obtained?). - Surface dose rate in Figure 7.3-3 (what are the uncertainties?). - Substantiate decreases in 10-yr and 50-yr dose rates. - Components of surface doses in Table 7.3-3 (specific codes used?). - Doses in Figure 7.3-9 (provide neutron and gamma components). - Cask penetrations (dose rates at those locations?).
7.3.2.3 Dose Assessment	12	<ul style="list-style-type: none"> - External dose rate variation with time in Figure 7.3-11 (how determined?). - External dose rates versus burnup and time in Figures 7.3-12, 7.3-13, and 7.3-14 (how determined?). - Doses at various cask dose points in square planar array of casks in Figure 7.3-15 (how determined?). - Dose at given point (all contributing casks included?); cask self-shielding effects (were they considered?). - O-ring seal dose rate versus time (how calculated?). - Shielding effect of trunnion plates (method used to determine surface dose rate reductions?). - Cobalt-60 activation of stainless steel cask shells (how calculated?). - Maximum dose rate from cobalt-60 activation? - Is 75 rem/hr for 1 or 24 PWR fuel assemblies? Does it apply at 5-year postirradiation time? - Secondary cover outside surface dose (method used to calculate reduction?). - Basis for deeming cobalt-60 activation of fuel cladding insignificant? - Multiplying factors (how determined?; address array of horizontally oriented casks).
7.3.2.4 Accident Event	5	<ul style="list-style-type: none"> - Loss-of-neutron shield dose rates (how determined?). - Doses at P7 and P12 are less than those at P8 and P13 (why?; does air/ground scattering affect doses?). - Doses in Figure 7.3-17 (provide neutron and gamma components). - Maximum dose in text is 370 mrem but Figure 7.3-7 shows 382 mrem (which is correct?). - Dose at side of cask for same accident is 600 mrem for BWR cask and 382 mrem for PWR cask (why the large difference?).
7.3.2.5.3 Geometry Verification	1	<ul style="list-style-type: none"> - Picture computer program results for PWR cask (provide).
7.4 Estimated Onsite Collective Dose Assessment		
7.4.1 General Description	7	<ul style="list-style-type: none"> - Results showing distance effects (how were effects calculated?). - Dose and dose rate at site boundary (substantiate why conservative; address horizontally and vertically oriented cask arrays). - Neutron and gamma source decay in Figure 7.4-13 (how calculated?).

TABLE 3.6. (contd)

Section of Topical Report	No. of Issues	Items Involved
		<ul style="list-style-type: none"> - Annual cask surface dose rate in Figure 7.4-8 (is it correct?; appears to be off by factor of 1000). - Dose rates in Table 7.4-3 (how determine?; based on single dose point, sum of several dose points, or average?). - Attachment of instrumentation, cask repair, maintenance, and decommissioning (dose consequences?). - Sequence of operations in Tables 7.4-2 and 7.4-3 (does not agree with those in Section 5.1.8).
8.0 Accident Analyses		
8.1 Normal and Off-Normal Operations		
8.1.1 Normal Operations	2	<ul style="list-style-type: none"> - Neutron shield outer surface thermal stresses during precipitation (why not considered?). - Normal loads imposed on fuel basket during lifting and horizontal storage (provide stress analysis).
8.1.1.2 Material Properties	2	<ul style="list-style-type: none"> - Effect on spent fuel with horizontal storage (need to address). - Neutron shield material in Table 8.1-4 (identify ethylene glycol/water ratio).
8.1.1.3 Containment Vessel	1	<ul style="list-style-type: none"> - Design pressure of 75 psig (what is basis?).
8.1.1.6 Cask Temperatures	6	<ul style="list-style-type: none"> - Ranges of conditions and parameters for heat transfer calculations (what are ranges?; need model description and basis for models and/or parameters). - Fuel rod temperature versus design criteria (likelihood of exceeding design criteria during normal operations?; sensitivity to models and parameters?). - Neutron shield temperature versus recommended operating limit (is 5°F sufficient margin of safety?). - Derating of maximum decay heat of stored fuel in cask array (why not discussed in Chapter 3?). - Maximum fuel cladding temperature if 24 kW allowed per cask in array (what is it?; will it exceed design limit?). - Neutron shield temperature of cask in array (vendor operating limits exceeded if 24 kW per cask allowed?).
8.1.1.8.1 Containment Vessel Analysis	3	<ul style="list-style-type: none"> - Containment vessel flange and penetration analysis (why ASME BPV Code, Subsection NB, not used?). - Provide thermal cycling analysis. - Differential expansion stresses on bolts?
8.1.1.8.4 Thermal Analysis	5	<ul style="list-style-type: none"> - Sensitivity of temperature profile calculations to interface conductances? - Interface conductances (expected variation?; changes due to thermal cycling over 20-year license period?). - Outer surface heat transfer coefficient (basis for selection?; other correlations considered?; if so, why rejected?). - Correlation used (confirmation as conservative estimate?). - Thermal analysis (was it performed for 17x17 PWR fuel?).

TABLE 3.6. (contd)

Section of Topical Report	No. of Issues	Items Involved
8.1.1.8.4.1 Thermal Model		
8.1.1.8.4.1.1 Fuel Region	2	<ul style="list-style-type: none"> - Emissivity for Zircaloy-clad fuel rods (value not appropriate for temperatures $\leq 800^{\circ}\text{F}$). - Justify use of parameters for thermal conduction and conductive heat transfer.
8.1.1.8.4.1.2 Copper/Stainless Steel Shells	1	<ul style="list-style-type: none"> - Basket installed by wedging it against containment shell, hence no gap (revise section accordingly).
8.1.1.8.4.1.3 Lead/Stainless Steel Interfaces	2	<ul style="list-style-type: none"> - Effect of thermal cycling on load contact pressure and heat transfer coefficient (provide information). - The topical report under Project M-33 includes a Section 8.1.1.8.4.2, "Experimental Temperatures." The same questions apply here. Also, deletion of a similar section in this report leaves no connection between the proprietary section and rest of report.
8.1.1.8.4.4 Neutron Shield Expansion Tank		
8.1.1.8.4.4.2 Pressure Stress	2	<ul style="list-style-type: none"> - Neutron shield cavity (bending stress analysis?; fatigue failure possibility?). - Ends and joints of neutron shields (ASME stress intensities?).
8.1.1.8.4.6 Thermal Expansion Analysis	1	<ul style="list-style-type: none"> - Lead and fuel basket radial expansion (provide information).
8.1.2 Off-Normal Operations		
8.1.2.1 Leak	1	<ul style="list-style-type: none"> - Sealing effectiveness of O-rings and secondary cover weld (how assured?).
8.1.2.1.1 Cause of Event	1	<ul style="list-style-type: none"> - Mechanical damage to cask (implied impairments to sealing system?).
8.1.2.1.2 Detection of Event	1	<ul style="list-style-type: none"> - Timely detection of cask pressure change (how assured?).
8.1.2.1.3 Analysis of Effects and Consequences	4	<ul style="list-style-type: none"> - Fuel cladding leak may result from a number of causes (include analysis and effects of events). - Cladding degradation (substantiate that activity level is within allowable limits). - Fuel assembly internal gas inventory estimate (explain in detail). - Fuel rod internal gas pressure (provide data).
8.1.2.1.4 Corrective Action	2	<ul style="list-style-type: none"> - Vacuum bonnet in place (plug accessibility?; ability to conduct operations?). - Absence of radiation exposure to personnel (does this imply operation done remotely?).
8.1.2.2 Fission Product Gas Pressure		
8.1.2.2.1 Cause of Event	3	<ul style="list-style-type: none"> - Cask internal environment (can monitoring system fail?). - Masking of small pressure changes (diurnal pressure variations?). - Gases inside cask (sampling procedure?).
8.1.2.2.3 Analysis of Effects and Consequences	1	<ul style="list-style-type: none"> - Include applicable parts of comments from third and fourth issues in Section 8.1.2.1.3.

TABLE 3.6. (contd)

Section of Topical Report	No. of Issues	Items Involved
8.1.2.3 Failure of Instrumentation	1	- Implication that REA cask be licensed without instrumentation (intent of this section?).
8.1.2.3.1 Cause of Event	1	- Cask instrumentation (vulnerability to possible failure mechanisms?).
8.1.2.3.2 Detection of Event	1	- Excursion from expected readings (failure of instrumentation or actual change in conditions?; describe procedure for identifying failure).
8.1.2.3.3 Analysis of Effects and Consequences	1	- Replacing pressure transducers (dose consequences?).
8.1.2.4 Radiological Impact from Off-Normal Operations	1	- Off-normal operations (what two lead to gas leakage?).
8.2 Accidents	4	- Accident conditions involving impact (perform dynamic analyses for critical points of containment vessel and basket). - Pin emissivity and thermal conductivity (comments in Subsection 8.1.1.8.4 apply). - ASME code stress quantities and limits for each set of analyses (should be summarized). - Information on load combinations (include for each of the accident conditions).
8.2.1 Loss of Neutron Shield		
8.2.1.1 Cause of Accident		
8.2.1.2 Accident Analysis	3	- Ambient temperature assumption (why less than maximum design value of 125°F?). - Ambient temperature (what was assumed?). - Apparent error in Figure 8.2-1 (explain).
8.2.1.3 Accident Dose Calculations	6	- Population cumulative and annual doses in mrem (should be in person-rem). - Dose variation with time (clarify statement). - Distance from accident (single person and 100,000 persons at same point?). - Distance from accident (at cask surface or other?). - Population exposures in Table 7.4-4 (dose equation used for calculations?; if so, which one(s) in list used?). - Neutron shield loss and fire accidents (cask orientation?; how were doses calculated?; provide input/output data; dose response factors used?).
8.2.2 Fire		
8.2.2.1 Cause of Accident	1	- Ethylene glycol (what about its presence?).
8.2.2.2 Accident Analysis	1	- Failure of O-ring seal (potential problem regarding gaseous release?).
8.2.4 Drop	4	- Cask drop cases (perform dynamic impact analyses). - Damage to fuel basket (include analyses). - Corner drop (perform dynamic analysis). - Detailed ASME Code stress quantities (include detailed calculations).

TABLE 3.6. (contd)

<u>Section of Topical Report</u>	<u>No. of Issues</u>	<u>Items Involved</u>
8.2.4 Tip Over	2	<ul style="list-style-type: none"> - The point of rotation, namely the neutron shield (demonstrate it does crush). - Moving or sliding of cask in certain events (consequences?).
8.2.5 Flood	3	<ul style="list-style-type: none"> - ASME Code limits and design properties used (should be design by analysis rather than formula techniques). - Thermal and hydraulic stresses for flooded condition (calculate and combine). - Reference in line 1, paragraph 3, page 8.2-80 ("Section 8.2.5.4.1," not "Section 8.2.4.5.4.1").
8.2.6 Cask Seal Leakage		
8.2.6.2 Accident Analysis	3	<ul style="list-style-type: none"> - Accidents resulting in breaching of more than one cask (basis for excluding?). - Specific isotopes released during accident (exact quantities assumed?). - Gaseous fission product contents (100% release assumed?).
8.2.6.3 Accident Dose Calculations	3	<ul style="list-style-type: none"> - Site boundary (distance assumed?; provide dose calculations for a range of distances). - Dose at site boundary (assumptions used?). - Fifty-mile population dose (assumptions used?).
8.2.7 Tornado Missiles		
8.2.7.2 Accident Analysis	1	<ul style="list-style-type: none"> - Loss of neutron shield (how is "repair" accomplished?).
8.2.7.3 Accident Dose Calculations	2	<ul style="list-style-type: none"> - Neutron shield repair (dose consequences to individuals involved?). - Effect of both neutron shield loss and leaking fission product gases (need to discuss; cumulative effect?).
8.2.7.4 Appendix		
8.2.7.4.1 Massive Missile Analysis	4	<ul style="list-style-type: none"> - Impact loads and times (incorrectly calculated; perform dynamic impact analysis). - Calculated impact force (greater than required tip-over force but text says cask does not tip over; why?). - Massive missile impact (will horizontal cask and skid tip over or just slide?). - Tornado wind loads (will they tip cask over?).
8.2.7.4.2 Penetration Resistance Missile	1	<ul style="list-style-type: none"> - If missile strikes near weld region (damage to cask cover seal?).
8.2.8 Explosive Over-Pressurization		
8.2.8.2 Accident Analysis	3	<ul style="list-style-type: none"> - Fission-product gases (which ones are explosive?). - "Negligible quantities" (meaning?). - Ethylene glycol (what about its presence?).
8.2.9 Earthquake		
8.2.9.4 Appendix		

TABLE 3.6. (contd)

Section of Topical Report	No. of Issues	Items Involved
8.2.9.4.2 Earthquake Tipping Analysis	3	<ul style="list-style-type: none"> - Analysis is nonconservative (include vertical acceleration or perform dynamic analysis). - Point of rotation in seismic analyses (demonstrate point does not change significantly during rotation). - Tipping of cask storage array following an earthquake (describe effects and consequences).
8.2.12 Burial Under Debris		
8.2.12.1 Cause of Accident		
8.2.12.2 Accident Analysis	2	<ul style="list-style-type: none"> - Figure 8.2-42 (appears incorrect; at 100 hours, maximum lead temperature should be at lead melting point; isn't this figure for the REA BWR cask design?); estimated maximum pressure versus test value (too close for adequate safety margin?). - Last sentence on page 8.2-124 (needs interpretation).
8.2.4.12.4.2 Containment Vessel Shell Deflection	3	<ul style="list-style-type: none"> - Lead pouring and cooldown processes (provide description). - Lead annulus region (provide diagram; tabulate dimensions, temperatures, and ASME stress intensities for critical points). - Critical points during and/or following fire event (provide temperature, thermal expansion, and stress analysis for worst conditions; use BPVC Section NB3133 for collapse evaluation).
8.3 Site Characteristics Affecting Safety Analysis	2	<ul style="list-style-type: none"> - Credible conditions under which ISFSI^(a) will operate (establish ranges of site-specific parameters). - Operating limits (summarize in "10.1.2.2 Site Parameter Limits"; provide site parameter limits used in cask design analysis).
9.0 Conduct of Operations		
9.2 Preoperational Testing	1	<ul style="list-style-type: none"> - Compliance with Section 9.2 of Regulatory Guide 3.48 (topical report should provide more specific guidance to utility).
9.2.2 Test Program Description	2	<ul style="list-style-type: none"> - Required contact resistances (conduct test of each cask to verify). - Boron poison plates in basket (verify presence of boron poison by testing).
10.0 Operating Controls and Limits		
10.1 proposed Operating Controls and Limits		
10.1.1 Contents of Operating Controls and Limits	5	<ul style="list-style-type: none"> - Helium pressure in cask environment (acceptable limits?). - Air within cask environment (limiting partial pressure?). - Fission product gases released from fuel to cask internal environment (maximum allowable quantity?). - Required periodic inspections (frequency?). - "Loss of neutron shield does not compromise safety" (statement not consistent with Section 8 accident analysis).

(a) Independent Spent Fuel Storage Installation (ISFSI).

TABLE 3.6. (contd)

Section of Topical Report	No. of Issues	Items Involved
10.1.2 Bases for Operating Controls and Limits	6	<ul style="list-style-type: none"> - Operating controls and limits stated in Section 10.1.1 (need to provide basis). - Surface dose rate (define "somewhat arbitrary limits"; what is operational dose limit?). - Temperature limit of 271°F (does it apply to single cask or array of casks?). - Unprotected cask surface temperature of 271°F (consistent with minimum safety standards?). - Cask spacing to maintain temperature limit (minimum limits?). - Cask lifting height (limit?).
10.1.2.1 Fuel Characteristic Limits	1	<ul style="list-style-type: none"> - Neutron and gamma sources (insufficient detail provided).
10.1.2.2 Site Parameter Limits	1	<ul style="list-style-type: none"> - Operating limits (summarize here; provide site parameter limits used in cask design analysis).
10.1.2.4 Cask Surface/Off Surface Dose Rate Limit	1	<ul style="list-style-type: none"> - Cask surface dose rates during normal and accidental events (reiterate and summarize here).
10.1.2.5 Cask Tightness Control	3	<ul style="list-style-type: none"> - The 0.1 atm/cm³-sec (STP) leak rate (basis?). - Cask pressure monitoring system (limit on precision?). - Vent and drain part plug leaks (actual limits?).
10.2 Development of Operating Controls and Limits		
10.2.2 Limiting Conditions for Operation		
10.2.2.2.1 Fuel Characteristics	1	<ul style="list-style-type: none"> - Fuel characteristic (state in form of operating limits).
10.2.3 Surveillance Requirements	2	<ul style="list-style-type: none"> - Pressure sensing instruments (calibration?). - Cask internal environment (provisions for sampling?).
10.2.6 Suggested Format for Operating Controls and Limits	1	<ul style="list-style-type: none"> - Format (follow Regulatory Guide 3.48).
10.2.6.2 Maximum Internal Pressure	1	<ul style="list-style-type: none"> - Leakage of fission product gases (meaning of term "tolerable"?).

**TABLE 3.7. Examples of Some Issues^(a) Involved in Project M-37
(GNSI, CASTOR V/21)**

Section of Topical Report	No. of Issues	Items Involved
1.0 Introduction and General Description of Installation	2	<ul style="list-style-type: none"> - The descriptions of Castor V on page 1.1-5 and Figure 1.1-2 do not agree. Moreover, the secondary lid representation in Figure 1.1-2 is not consistent with the detailed drawing in Annex 1. The maximum leak rate on page 1.2-1 and in Table 1.2-1 is 10^{-7} millibar litre per second while in Section 3 it is 10^{-6} millibar litre per second. Components described on page 1.2-4 do not correspond to numbers on referenced Figure 1.1-2. Table 1.2-2 is not present. - The overall dimension of 4976 mm described in Section 1.2.4 is inconsistent with the length of 4916 mm shown on drawing A50111 in the Annex. Section 1.2.4 refers to plastic (i.e., polyethylene) in the bottom and secondary cover but the drawings of Annex 1 show no polyethylene. Annex 1 should contain drawings of the primary cover, protection cover, and should show the bolt locations for the protection cover on the cask body. Also on page 1.2-4 the cooling fins of the cask are incorrectly referred to as cast-on. A need for better editing is evident in numerous instances (e.g., BWR is used instead of PWR, Castor 1c is referenced in Table 3.1-11). In drawing A50111, the cask has impact limiters which are not discussed in the report. The thermal design load for the cask is stated to be 21 kW (Section 1.2.2) and 29 kW (Table 1.2-1) and in Section 3 to be 26 kW (Section 3.1.1.2) and 18.1 kW (Table 3.1-11) and in Section 5 to be 22.4 kW (Section 5.1.3.6). GNSI, which submitted the topical report, is not included as a corporate entity, an agent or a contractor.
3.0 Principal Design Criteria	7	<ul style="list-style-type: none"> - All tables and figures in this and all other sections should stand alone, in so far as possible, without need for reference to the text to be understood. The analysis of Section 3.2.3, Seismic Design, is incorrect (see NUREG-800, page 3.7.1-4). The combined loading criteria in Table 3.2-1 are typically for transport casks, not for storage, and were not used in any of the analysis in later sections. Criteria contained in American Society of Mechanical Engineers Boiler and Pressure Vessel Code (ASME BPVC) Section III cannot be applied directly to nodular cast iron but must be appropriately modified to account for the low ductility ($\leq 8\%$). - In general, the stress intensity limits for the materials described in Appendix I of the ASME Code should be used. The German specification for a material should be converted to an equivalent American specification that can be matched with those in Appendix I and the appropriate stress intensity limits defined. In the case of ductile cast iron, a safety factor of 4 on the ultimate tensile strength is recommended based upon a design by analysis approach.

(a) From References 75, 172, and 173. Additional issues are in References 235, 239, and 241. Responses are in References 44, 69-72, and 174-177. Other items are in References 73 and 178; the NRC's response is in Reference 74.

TABLE 3.7. (contd)

Section of Topical Report	No. of Issues	Items Involved
		<ul style="list-style-type: none"> - The stress intensity limits provided by the Code (or Regulatory Guide 7.6) for combinations of primary and secondary stresses are valid only if the material is sufficiently ductile to allow for local plastic deformation. Steels for nuclear application are in the range of 20% elongation and easily meet the criteria for the stress intensity limit associated with secondary stresses. In NRC's judgment an elongation over 12% would still be sufficient. However, if the elongation is below 5% the uncertainties associated with margins of safety on strain limits would be too great to allow higher stress intensity limits for either primary membrane plus primary bending or the combination of primary and secondary stresses. For elongations below 5% only a primary membrane stress intensity limit should be used for design under normal conditions (see also our comments on Section 4). Under accident conditions the stress intensity limit may increase in accordance with code allowables (see also NRC's comments on Section 8). - The criticality calculations should use current computer codes and include benchmark calculations. - No information is provided relating to the quantities of steel, zirconium, and UO_2 in the assembly; the irradiation time; the neutron flux; the materials of the head and foot pieces, etc. Furthermore, the total assembly weight and uranium weight, per cask, cannot be reproduced using the data in Table 3.1-1. If the 15x15 assembly is the reference fuel, the total assembly weight and uranium weight should be 13,650 and 9,345 kg per cask, respectively. - The description of the gamma sources is inadequate. For each source there should be a strength and a list of the associated gamma rays, with branching intensities, that were used in the shielding calculation. The discussion of the activity release remains unclear. Results in Annex 1 do not appear to be for the burnup (35 GWD/MTU) and power 935 MW/MTU) stated. Table A3.1/1-5 shows the burnup and power as 6.362 GWD/MTU and 27.66 MW/MTU, respectively. - Is it a fact that for the Castor V the seal leak rate is greater than for the Castor Ic (10^{-6} versus 10^{-7} mbar 1 sec⁻¹ for each seal)? Figure 3.3-6 is missing. In the area of shielding it is of interest to note that the doses for the "middle of bottom" and "middle of cover" are exactly the same as those for the Castor Ic even though the material thicknesses and source spectrums are different and the ages of the fuel are different (1 versus 5 year old fuel). It appears that the "annual dose rate versus distance from a single cask" in Figure 3.3-14 is for the Castor Ic and not the Castor V. Figure 3.3-15 is missing. Activation calculations for the decommissioning considerations are also unclear. How can the neutron flux and the activation product activity for the Castor V be the same as for the Castor Ic?

TABLE 3.7. (contd)

Section of Topical Report	No. of Issues	Items Involved
4.0 Installation Design	3	<ul style="list-style-type: none"> - This section is incomplete and inconsistent with the design criteria. Reference in Section 4.2, Storage Structures, to Section 1.2 is inappropriate. Section 3 is concerned with design criteria. None of the calculations presented conform to the ASME BPVC Section III design criteria. The cask and basket materials should be described in detail or related to an applicable American Society for Testing and Materials (ASTM) specification. A stress analysis of the basket for normal operations should be presented in this chapter and for accident conditions in Chapter 8. A fatigue analysis of the cask due to severe weather changes should be presented. Annex 1 of Section 4.2 should be properly incorporated into this chapter. - Material covered in Section 4.2.1.7 (Material Properties) through Section 4.2.3.3 (Design Bases and Safety Assurance) of the Castor Ic report has apparently been omitted here. - The reference to GGG-40.3 as the specification describing the ductile cast iron for the Castor V storage cask is inconsistent with the materials properties specified in Section 4.2.1. If the material properties specified for GGG-40.3 in German DIN-Standard 1693 cannot be guaranteed, then reference to this specification should be deleted and the material treated as one uniquely defined by GNS. Material acceptance for nuclear application in the United States is specified by the provisions of Section III of the ASME Boiler and Pressure Vessel Code and should be adhered to by the applicant. Appendix I of the ASME code does not provide stress intensity values for ductile cast iron. Neither is there an ASTM specification for GNS ductile cast iron that can be invoked for establishing stress intensity limits. Consequently, the applicant should follow the procedures outlined in Appendix IV of the Code. While the data called for in Appendix IV applies to nuclear reactor applications and may not in its entirety be applicable to ductile cast iron storage casks, sufficient data relevant to establishing the safety of the storage cask must be provided. Particular attention should be paid to Paragraph IV-1400(d) which requires that the request for approval of the material be described in ASTM specification form. Also, the design data supplied in accordance with Paragraph IV-1400(e) should be substantiated by reference to applicable test data.
5.0 Operational Systems	3	<ul style="list-style-type: none"> - The applicant has not provided adequate thermal analyses that demonstrate with reasonable assurance that gross rupture of the fuel cladding will not occur during normal storage conditions. Benchmark calculations relating testing to the thermal codes used to predict fuel cladding temperatures should be provided. The detailed calculations in Appendix 8 predict maximum rod temperatures that are 75°C higher than those presented in this chapter. The 400°C limit on maximum fuel clad temperature must be substantiated with appropriate safety margins. The maximum and average rod cladding temperature should be clearly related to cladding pressure. Fuel rod temperatures and pressure, cask surface temperature, and cavity pressure and temperatures should be calculated for the cask and discussed with respect to rod failure and radioactive material release.

TABLE 3.7. (contd)

Section of Topical Report	No. of Issues	Items Involved
		<ul style="list-style-type: none"> - The case for establishing maximum allowable fuel rod temperatures should be presented in the body of the report. Your evaluation may take into account published information, such as the recent Department of Energy report, "Technical Basis for Storage of Zircaloy-Clad Spent Fuel in Inert Gases" (PNL-4835), which concludes, as a conservative position, that ample evidence justifies "...extended dry storage in inert cover gases at cladding temperatures up to 380°C." However, merely citing the opinion of others with regard to acceptable fuel rod temperatures is insufficient. The margin of safety against fuel cladding rupture over the design life of the storage installation should be estimated based upon the design basis fuel, fuel rod pressures, cladding creep rupture properties and the uncertainties associated with the parameters. Since cladding deterioration rate is expected to decrease as the temperature of the fuel rod decreases, a life fraction rule may be applied to determine a more realistic margin of safety. - Figure 3.3-11 cannot be used to derive radiation exposure of personnel during handling procedures since it is for the Castor 1c. On page 5.1-11, GNS states "the cask has 74 fins," while in Figure 6.1-2 there are 75 fins. The design value for hat dissipation rate is 29 kW (Table 1.2-1) but the maximum allowed is apparently 26 kW (Section 3.1.1.2) and the calculated rate is 22.4 kW. Table 5.1-1 appears to imply that PWR and BWR cladding temperatures are the same during reactor operation. Figure 5.1-8 is for the Castor 1c according to the text, yet it is identified in caption as representative of the Castor V. Is not Figure 5.1-10 for the Castor 1c?
7.0 Radiation Protection	1	<ul style="list-style-type: none"> - Table 7.3-1 gives the wall thickness as 370 mm while Figure 7.3-1 gives the thickness as 380 mm. (In this case, one surmises that the values in Table 7.3-1 represent effective shielding thicknesses rather than dimensions. This point is not made clearly in the table or the text.) Figure 5.1-2 gives the fin thickness as 50 mm while Figure 7.3-1 gives the thickness as 57 mm. R_2 in Figure 7.3-2 and one half the diameter in Figure 7.3-3 do not agree. In Figure 7.3-4, the total cavity length shown is 4120 mm ($200 + 3660 + 260$) while the value tabulated is 4200 mm. In addition, the maximum assembly length is tabulated as 4065 mm while in Table 3.1-1 the length varies between 4100 and 4150 mm. With regard to the experimental verification of shielding, where is a comparison of the calculated and measured results for the various casks?

TABLE 3.7. (contd)

Section of Topical Report	No. of Issues	Items Involved
8.0 Accident Analysis	2	<ul style="list-style-type: none"> - Comments made on Revision 2 of Castor Ic Topical Report for material covered through Section 8.2.1.2.2 (Tornado Missiles) remain relevant here. The reference to Table 7.2-6 in Table 8.1-2 is incorrect. Table 7.2-6 says nothing about release fractions since it deals with total cask inventory only. In Table 8.1-3 the activity release for the 134 Cs is the same as for the Castor Ic; all other releases are exactly a factor of 10 greater than the Castor Ic. Is this simply fortuitous? Even though the leak rate is a factor of 10 greater for the Castor V, the source terms are certainly different and case B does not involve seal failure. Why should the results be similar? The third lid reference on page 8.1-9 should be removed. With regard to the radiological impact from off-normal operations, the total dose presented is precisely the same as for the Castor Ic while the releases are a factor of 10 greater for the Castor V. How can the dose results be the same? - The Castor V Topical Report did not provide a cask tipping or drop analyses. When these are submitted, the NRC staff suggests that both the models and the values of the parameters be demonstrably conservative. Stress analysis and their results should be based on ASME BVPC criteria and presented for the cask and basket for all potential accident conditions. The analyses should be based on justified conservative models, analytical techniques, and safety margins to reasonably preclude radioactive material release under accident conditions with 10 CFR Part 72 limits.
9.0 Conduct of Operations	1	<ul style="list-style-type: none"> - Acceptance testing performed on production casks should be described in this Section (see Section 9.2.2).
Sections 3, 5, 6, 7, and 8 and Appendix 3	several	<ul style="list-style-type: none"> - Lawrence Livermore National Laboratory (LLNL) raised several questions and comments. The response by GNSI involved changes on 26 pages of the topical report and clarifications on two items. One of two items involved the cask rear breech plate (the material is AISI 321); the other item concerned the unidentified boron content of the fuel basket (it was not identified because no credit is taken in the shielding calculation for the presence of boron).
Fuel Basket	8	<ul style="list-style-type: none"> - NRC was informed that on October 3, 1985, GNSI was notified by EG&G Idaho, Inc., that basket of the CASTOR V/21 cast at the Idaho National Engineering Laboratory (INEL) had been found on September 30, 1985, to have sustained possible damage. Cracks in, or in the immediate vicinity of, weld positions had been observed at up to eight locations at the top of the basket. NRC indicated that information as to what occurred with respect to the cask needs to be obtained and analyzed.

TABLE 3.7. (contd)

Section of Topical Report	No. of Issues	Items Involved
		<p>- In response to NRC staff comments on the preliminary report on the cracking of the basket of the CASTOR V/21 cask at INEL, GNSI is to provide additional structural analysis, material property data, and basket drawings and to identify any remedial action that may be taken to preclude such an event occurring under design basis conditions. Specifically, GNSI is to provide the following items:</p> <ol style="list-style-type: none"> 1. With respect to resolving causes of the weld area failures, these should be related to strain mechanisms. Thus potential mechanisms for these failures should be explained and assessed with respect to INEL demonstration conditions and the CASTOR V/21 cask used. This assessment should be made with reference to the thermal analysis provided in the TSAR. Furthermore, the thermal stress analysis should be revised to include the effect of gusset plates and correct material properties and also should be revised to include reporting the stresses at all points of the structure. 2. Measures with respect to basket fabrication and/or design, which will be taken to assure that such weld failures could not occur both under normal and accident design conditions, should be provided. Tolerances should be shown that control the maximum and minimum gap dimensions. 3. Additional detailed basket drawings at different axial positions to include and exclude gusset plates should be provided with the basket welding plan. Nonstructural welds which are not related to basket structural integrity, i.e., which are without a safety function, should be identified and distinguished from welds which are relied upon to maintain basket structural integrity. 4. A detailed structural analysis of the cask basket under both normal and accident design conditions alluded to in Section 4.0 of the GNSI preliminary report should be provided to demonstrate the structural integrity of the basket. It is assumed by the NRC staff that this analysis will not take any credit for nonstructural welds, i.e., joint welds, including, but not limited to, those welds which show evidence of failure in the INEL demonstration. If such should not be assumed in this analysis, this should be made clear. If measures referred to in item 2 above are taken, it should also be made clear whether this analysis assumes these are implemented. If such measures are taken, they will be reflected in the supplied drawings referred to in item 3 and in cask fabrication procedures and documentation.

TABLE 3.7. (contd)

Section of Topical Report	No. of Issues	Items Involved
		<p>5. Material properties which reflect actual testing done on these materials should be included and their use carefully explained in terms of analyses supplied to the NRC and design commitments for materials with respect to the topical report for the CASTOR V/21 cask, for example, values for Radionox (boronated stainless steel). Particular attention should be directed toward demonstrating that either brittle fracture is not a failure mode or it is, then safety margins against brittle fracture are adequate under dynamic loading conditions.</p> <p>6. Data on weld tests performed should be supplied distinguishing between welds which have a safety function, that is, which are relied upon to maintain the structural integrity of the basket and nonstructural welds.</p> <p>7. As discussed in the meeting of NRC and GNSI on December 13, 1985, based on INEL measurements and the GNSI analysis, explain why there is no effect observed due to sub-critical multiplication with respect to dose calculations (Reference: Correction 7.3.2 in NRC's letter of approval for the CASTOR V/21 topical report). In relation to this explanation, provide drawings with changes related to cask borehole length shortening and the addition of steel plugs at the bottom of boreholes to provide additional reduction of gamma dose rates at positions near the cask wall top and bottom. Also provide an explanation of this improvement to cask shielding design. These changes are to be reflected in cask fabrication procedures and documentation.</p>

TABLE 3.8. Examples of Some Issues^(a) Involved in Project M-39 (NUTECH)

Section of Topical Report	No. of Issues	Items Involved
General Comments from Initial Examination of Report	24	- Design details, descriptions of computer models, and computer calculations (need to provide additional information). NRC's request for additional information involves three design categories: 1) thermal-hydraulics (6 issues), 2) criticality and shielding (5 issues), and 3) structural design (13 issues).
1.0 Introduction and General Description of Installation		
1.1 Introduction	1	- Leakage of helium from DSC ^(b) (assurance that it will not occur?; maximum leakage rate?; measure to monitor DSC atmosphere composition?).
1.2.1 General Description of Installation	1	- Single stand-alone HSM ^(c) or physical arrangements in other than 4x2 array (clearly state which arrangement(s) are being proposed for review).
1.2.2 Principal Design Criteria	2	- Limits on external atmosphere (control of corrosion or algae growth?; discuss). - Duty cycle allowed for materials, particularly concrete (provide discussion).
1.2.4 Safety Features	1	- Cask movement, cask head and truck transport values excluded from table (what are values?).
3.0 Principal Design Criteria		
3.1.1.1 Physical Characteristics	1	- Fuel assembly types considered (text and tables should be consistent); assembly types not enveloped by design criteria (delete or modify design to accommodate); design parameters of reference fuel assembly (any that would lead to nonconservative analysis?).
3.1.2.2 Handling and Transfer Equipment	3	- Handling of transfer cask by single failure-proof crane (if assumed, so state). - Validity of five-foot horizontal drop criterion (specify dimensional criteria to show maximum conceivable horizontal drop height is five feet or less). - Design criteria for transporting DSC in GE IF-300 cask (what vertical acceleration factors measured or assumed?; show that accelerations enveloped by five-foot drop analysis).
3.2 Structural and Mechanical Safety Criteria	1	- Maximum internal pressurization levels (inconsistency in stated values; inconsistency in associated temperature levels; state actual design parameters and be consistent).
3.2.3.1 Seismic Design	2	- Damping value of 2% of critical damping under safe shutdown earthquake (shouldn't it be 3%?). - Maximum vertical ground acceleration listed as 0.1 g and references 10 CFR 72.66 (no such value in 10 CFR 72.66; shouldn't value be 0.17 g?).
3.2.3.2 Seismic System Analysis	1	- Concrete coefficient of friction of 1.0 from ACI 349-80 code (only valid under certain conditions; NUTECH should use 0.6 as stated in ACI 318-83, Section 11.7.4.3).

(a) From References 75, 77, and 160-162. Additional issues (over 20) are in References 78, 79, and 179. Responses are in References 163, 164, and 180-182.

(b) Dry Shielded Canister (DSC).

(c) Horizontal Storage Module (HSM).

TABLE 3.8. (contd)

Section of Topical Report	No. of Issues	Items Involved
3.2.5.2 Dry Shielded Canister (DSC)	2	<ul style="list-style-type: none"> - ASME Level D service limits for accident or faulted conditions (if limits retained, state in structural design criteria that DSC must be disassembled and inspected for damage following an accident). - Retrieval of all fuel assemblies from DSC following an accident (if ASME Level D service limits used, what is assurance that possible deformation will not inhibit easy removal?; issue arises again in last issue under Section 3.3.4).
3.3.3.2 Instrumentation	1	<ul style="list-style-type: none"> - States no need to monitor DSC internal pressure and temperature; limiting pressures and temperatures based on calculations (assurance that calculational techniques are qualified for design use?).
3.3.4 Nuclear Criticality Safety	8	<ul style="list-style-type: none"> - Reactivity as a function of number of fuel assemblies in canister (analyses performed?; were reflector effects considered?). - Design clearances and fabrication tolerances (considered in criticality analysis?); ensuring insertion and removal of fuel assemblies in guide sleeves (design clearances?); worst case reactivity for water-flooded canister (design clearances and fabrication tolerances taken into account?). - Temperature associated with water density assumption (compared to fuel pool temperature limits?; if different, are KENO results conservative?). - Cell homogenization procedure (errors or biases?; biases quantitatively evaluated?; are biases conservative?). - Some DSC modules may use boron sleeves, while others may not (controls to ensure low-burnup fuel assemblies not inadvertently loaded into unborated canisters?). - Boron content of Boral guide sleeves (manufacturing tolerances on boron?; nominal or minimum boron concentration used in criticality analyses?; if a difference, what is magnitude of effect on reactivity?). - Dimensions in Figure 3.3-2 (inconsistent with text and KENO output; is figure in error?). - See last issue under Section 3.2.5.2 as it applies here also (effect on multiplication factor if deformation of DSC internals occurs?).
3.3.4.3 Verification Analysis	4	<ul style="list-style-type: none"> - Criticality analysis results from KENO-IV computer program (verification with calculations?). - Verification analysis involves only three benchmark calculations (statistically significant?). - Calculations of benchmarks as mean value plus two standard deviations (need to consider calculational bias; compare mean value with experimental results; mean value and standard deviation of benchmark calculations?). - Difference between measured and calculated values (see second question above; is computational error consistent with experience of others?; reasons for reported errors in present calculations?).
3.3.7.1 Irradiated Fuel Handling and Storage	1	<ul style="list-style-type: none"> - Storage atmosphere inside DSC to prevent degradation and gross rupture (what design criteria applied to comply with 10 CFR 72.72 (h)?).
3.4 Classification of Structures, Components, and Systems	1	<ul style="list-style-type: none"> - NUHOMS 0708 components important to safety (provide safety and qualification assurance classifications).

TABLE 3.8. (contd)

Section of Topical Report	No. of Issues	Items Involved
4.0 Installation Design		
4.2.3.1 Dry Shielded Canister	2	<ul style="list-style-type: none"> - Canister body is rolled and welded stainless steel plate (maximum leakage rate for longitudinal weld?; inspections and/or tests to assure leakage is below that rate?). - Canister dry film lubricant (will film attract insulating dust or dirt layer and interfere with heat transfer?; how much of canister to be coated?).
5.0 Operation System		
5.1.1.3 Cask Drying Process	2	<ul style="list-style-type: none"> - Backfilling of canister with helium (temperature of helium when valves closed?; temperature measured or inferred?). - Performing helium leak test (differential pressure across canister?).
5.2.2.1 Safety Features	1	<ul style="list-style-type: none"> - Safe operation of spent fuel storage system under both normal and abnormal conditions (safety features provided?; limits selected for commitment to action?).
5.3.2 Component/Equipment Spares	1	<ul style="list-style-type: none"> - Damage to components (provision for installation of spare or alternative equipment?).
7.2.1 Characterization of Sources	2	<ul style="list-style-type: none"> - Design basis neutron and gamma ray source strength based on average burnup values (some combinations of seven fuel assemblies may have higher source strength); use of mean values (implication for selection of fuel assemblies?); conformance to canister design basis source strength limit (how accomplished?). - Flux-to-dose conversion factors used (reference?).
7.3.2 Shielding	12	<ul style="list-style-type: none"> - Computer codes used in shielding analysis (verification analyses performed?; discuss shielding benchmark validations). - Shielding analysis (concrete composition used?; variation in water content over service life considered?). - HSM wall and roof gamma ray surface dose rate (relative contribution of primary and secondary gamma rays?). - HSM penetration calculation (discussion missing; quadrature order used in DOT-IV calculation?; justify adequacy of quadrature). - Neutron attenuation through HSM air exhaust penetration greater than that for gamma rays (explain). - Ray effects in DOT-IV calculations (measures to minimize?; ray effects evident in results?). - Application of DOT-IV code to shield penetration problem (verification analysis performed?). - "Cask-DSC Annular Gap" dose rates (averaged over cask surface or gap dose rates?). - Cask-DSC annular gap calculation (what was angular quadrature?; justify that angular quadrature adequate for modeling); calculated fluxes (ray effects observed?); streaming through gap (observed in results?). - QADMOD/albedo procedure (analysis or data to benchmark validity?).

TABLE 3.8. (contd)

Section of Topical Report	No. of Issues	Items Involved
		<ul style="list-style-type: none"> - Air exhaust penetration vicinity (secondary gamma rays considered?). - Proper axial positioning of DSC in HSM during normal operation and in seismic event (shielding implications?).
7.4 Estimated On-Site Dose Assessment	2	<ul style="list-style-type: none"> - In figure, total dose slightly less than scattered dose at 1000 feet (plotting error or computer code problem?). - Basis for dose rates in table (not evident in text; how were dose rates obtained?).
8.0 Analysis of Design Events		
8.1.1 Normal Operation Structural Analysis	2	<ul style="list-style-type: none"> - Table showing materials properties (excludes materials including concrete, steel reinforcing, Boral, etc.; provide materials properties for them). - Long-term storage of spent fuel horizontally (lack of experience exists; address effects on fuel and its retrievability).
8.1.1.1 Normal Operation Loads	8	<ul style="list-style-type: none"> - Fission gas release in design internal pressure calculation (basis for assuming 25%?). - Normal average canister gas temperature (450°F or 429°F?). - Spacer disc temperature distribution (is Figure 8.1-2 correct?; explain for closest canister/guide tube point). - Thermal cycling effects (explain use of temperature distributions); cyclic fatigue analysis (heat-up cycles for insulated loaded DSC considered?; if not, would their inclusion modify results?). - Density of normal concrete (inconsistency in values used; clarify). - Equations 8.1-9 and 8.1-10 (terms missing; if equations used, results may be incorrect; clarify). - Modulus of elasticity (incorrect value used; need to correct). - Azimuthal orientation of DSC in HSM (how assured?).
8.1.1.2 Dry Shielded Canister Loads Analysis	2	<ul style="list-style-type: none"> - Effect of thermal variations on canister shell (show what overall deflected shapes are; what are ASME stress intensities at six locations on the DSC and include discontinuity effects between shell and top cover plate). - Drawings (insufficient information on Boral basket/top lead plug gap; provide actual shop drawings instead of conceptual drawings).
8.1.1.3 Dry Shielded Canister Internal Basket Loads Analysis	1	<ul style="list-style-type: none"> - Fuel assembly irradiation growth (source of 0.75 in. value?); calculations (insufficient information on drawings to permit verification).
8.1.1.4 DSC Support Assembly Loads Analysis	1	<ul style="list-style-type: none"> - Thermal analysis (provide assembly procedure specifying torque of nuts).

TABLE 3.8. (contd)

Section of Topical Report	No. of Issues	Items Involved
8.1.1.5 Horizontal Storage Module Loads Analysis	6	<ul style="list-style-type: none"> - Reference (NUREG-0800, not NUREG-0880). - Ultimate shear value (provide calculational details). - Stresses in corbel and ultimate strength capacity of corbel (provide calculations). - Acceptance criteria for concrete temperature (state interpretation of ACI 349-80; relate to meteorological information). - Analysis of local concrete temperatures (excludes conduction through support bolts and support rail; effect of this on local concrete temperatures and compliance with ACI 349-80 acceptance criteria?); concrete thermal conductivity (effect of thermal cycling?; effect of rebar?). - DSC end in contact with concrete wall (concrete temperature of same magnitude as calculated DSC temperatures would result; clearance between DSC end and concrete wall?; concrete temperature adjacent to DSC end?).
8.1.2 Off-Normal Events	1	<ul style="list-style-type: none"> - Thermal radiation shield (failure of supports not addressed; justify that event is not feasible or provide analysis of consequences; how would such failure be detected?).
8.1.2.2 Blockage of the Horizontal Storage Module Inlet-Cause of Event and Detection	2	<ul style="list-style-type: none"> - HSM air flow with inlets blocked (appears to be random process with zero mean value; provide experimental confirmation of assumed flow magnitude). - Pressure (inconsistent values/conversions).
8.1.3.1 Thermal-Hydraulics of the Horizontal Storage Module Principles of HSM Cooling System	5	<ul style="list-style-type: none"> - Reference 8.44 (excluded from reference list). - Axial peaking factor used in thermal analysis (this additional limiting design parameter not cited in Table 3.1-1; justify this limit if it is imposed on stored fuel). - Restrictions on application of Equations 8.1-37 through 8.1-40 (compliance with restrictions?). - Solar radiation (levels considered in heat transfer analysis?; levels associated with hottest days?). - Maximum concrete ceiling temperature (inconsistent values stated; analysis excludes direct conduction through supports); severe summer condition considered (off-normal or transient condition?; if so, what is limiting ambient condition?; stated value of 70°F is quite low for a limiting ambient condition).
8.1.3.3 Thermal-Hydraulic Analysis of the Canister Inside the Transfer Cask Model	2	<ul style="list-style-type: none"> - Emissivity of stainless steel transfer cask exterior (inconsistency in stated values; justify use of different values). - Variation of input parameters such as emissivity, axial peaking factor, etc. (sensitivity of calculated results to the variations?); margin of safety to limiting conditions (how established?).
8.1.5 Design Basis Internal Pressure Loads	1	<ul style="list-style-type: none"> - DSC integrity (describe any hydrostatic testing; was pressurizing loaded DSC with helium considered as a test?; in such a test, how would test pressure relate to design pressure and accident pressure?).

TABLE 3.8. (contd)

Section of Topical Report	No. of Issues	Items Involved
8.2.2.2 Accident Analysis	2	<ul style="list-style-type: none"> - Coefficient of friction between NUHOMS^(a) system and concrete pad (≤ 0.6 instead of 1.0?). - NUHOMS system (consider requiring tie downs to permanently anchor NUHOMS system to concrete pad or set HMS in key-way imbedded in pad?).
8.2.3.2 Accident Analysis	6	<ul style="list-style-type: none"> - Values for E (use values at maximum calculated temperature, not room temperature). - Maximum vertical earthquake acceleration (use at least 0.17 g, not 0.1 g; redo calculations). - V_w, V_r, and I terms (assumptions on wall volume, roof weight, etc.); calculations (effect of DSC and its support?). - Parameter in Table 8.2-6 (M_u instead of V_u?). - Combining horizontal seismic loads from two directions (excluded from text; include unless considered to be negligible). - Longitudinal sliding of seismically loaded cask along rails (not included in text; if possibility exists, any damage to HSM or DSC grapple?; consequences if DSC cannot slide?).
8.2.4.2 Accident Analysis	2	<ul style="list-style-type: none"> - References to two tables (incorrect). - Flood level (could cause thermal stress in canister; were these loadings included in pressure loadings?).
8.2.5.2 Accident Analysis	5	<ul style="list-style-type: none"> - Decelerations assumed for five-foot drop height (revise discussion to reflect actual process for obtaining values; reference data by appendix). - Deceleration for top end vertical drop (why not use the more conservative data in Appendix V2 of NEDO-10084-3?). - Possibility of cask dropping vertically through five feet, then tipping over and slapping down (no discussion provided; slap-down case more severe than five-foot vertical drop; discuss reason for excluding slap-down case). - Attenuation or amplification characteristics of liner (for evaluation, NRC needs information on liner material and dimensions). - Horizontal drop case (effect of two-inch diameter rods not considered; discuss omission or recalculate spacer disc stresses).
8.2.8 Dry Storage Canister Leakage	1	<ul style="list-style-type: none"> - DSC designed for no leakage under any conditions (statement cannot be substantiated as very small leakage rates cannot be detected or measured; provide estimate of leakage rate and base on experimental data or analysis).
8.2.9.1 Accident Analysis	4	<ul style="list-style-type: none"> - Fission gas release (inconsistency in values assumed; provide basis). - Helium leakage (would it increase substantially during pressurization transient?; address expected rate at higher temperature and pressure). - DSC fatigue (also consider effect of diurnal temperature variation). - Maximum inventory of fill and fission gases in one fuel assembly (what is it?; how was value established?; criteria needed?; discuss).

(a) NUTECH Horizontal Modular Systems (NUHOMS).

TABLE 3.8. (contd)

Section of Topical Report	No. of Issues	Items Involved
8.2.10 Load Combinations	4	<ul style="list-style-type: none"> - Load combinations (excludes effects of wind, tornado, or flood; compliance with ACI 349-80?); materials properties (need evaluation at worst thermal case). - HSM corbels, bearing plates, bolts and reinforcing steel (provide analysis for two load combination cases; specify corbel temperature and use associated material properties). - Material properties for thermal accident load (inconsistency in temperature values assumed; clarify). - DSC support assembly stresses (provide temperature of DSC/tee support contact point).
10.0 Operating Controls and Limits		
10.1 Proposed Operating Controls and Limits	1	<ul style="list-style-type: none"> - Decay heat of stored fuel (axial power peaking should be equal to or less than 1.08).
10.2.2 Technical Conditions and Characteristics	2	<ul style="list-style-type: none"> - Condition 3 concerning DSC helium leak rate of primary weld (should apply to all canister welds, including axial seam weld; address leak rate as function of temperature). - Limiting heat load of 1-kW assembly (why excluded?).
10.2.3 Surveillance Requirements	3	<ul style="list-style-type: none"> - Inspection of internal air passages for blockage (indicate provisions). - Water entering HSM (provisions for draining?; inspections to assure water does not accumulate?). - Air flow through module (how is acceptability of flow rate determined?; how will flow rate be measured?; compliance with flow criteria?).
10.3.1.1 Fuel Specifications	1	<ul style="list-style-type: none"> - Parameters (clarify; which are for fuel assembly and which for canister?).
10.3.2.1 DSC Vacuum Pressuring During Drying	1	<ul style="list-style-type: none"> - Criteria for drying operation (basis for one hour at pressure?; was possibility of a waterlogged fuel rod considered?).
10.3.2.2 DSC Helium Backfill Pressure	1	<ul style="list-style-type: none"> - Assuring that thermal equilibrium has been reached (need to specify hold time at this pressure).
10.3.2.3 DSC Helium Leakage Rate of Primary Weld	2	<ul style="list-style-type: none"> - Helium calculation (appears to be at 2, rather than 1.5, atmospheres). - Single leak assumed (leakage may occur at several locations; was this type of leakage considered?; discuss).
10.3.2.7 Maximum Air Exit Temperature	2	<ul style="list-style-type: none"> - Design based on passive cooling (if temperature greater than 100°F and fans provided, passive cooling mechanism lost; include fans in design or eliminate as a corrective action). - Cooling air temperature rise (design features to preclude certain measurements and yet comply with temperature rise criterion?; more frequent measurements may be needed if assurance cannot be provided?).
10.3.3.1 Surveillance of the HSM Air Inlets	2	<ul style="list-style-type: none"> - No blockage of outlets and no internal blockage (how assured?; does surveillance include flow or temperature rise measurement?). - Air blockage (if HSMs inspected once per week, how can it be assumed air blockage would be no longer than 48 hours?; this question also relates to Section 8.2.7.2).

TABLE 3.8. (contd)

Section of Topical Report	No. of Issues	Items Involved
Appendix B - Details of Heat Transfer Analysis of the NUHOMS System		
B.1 Apparent Thermal Conductivity of the Gas Between the Inner Surface of the Canister Shell and the Outer Surface of the Guide Sleeve	6	<ul style="list-style-type: none"> - Thermal conductivity of gas in a horizontal or vertical annulus (discrepancy in assumptions; explain). - Fuel assembly length (listed but not used in equations; is this correct?; why is value cited?). - Thermal conductivity of helium (why was 620°F value included?; why was 80°F used in determining average value?; justify use of 1.6 as representative apparent thermal conductivity value). - Temperature difference of 300°C assumed in h_{eff} calculation (assumption does not appear needed). - Statement on page B.16 (two-step approach to using Wooten-Epstein Formula appears closer to actual application). - K_{fuel} values (provide corrected values).
Appendix C - Sample Computer Input and Output for the NUHOMS Structural Analysis	1	<ul style="list-style-type: none"> - STARDYNE finite element analysis data (insufficient data provided to permit checking of stress data discussed in Section 8; provide sufficient quantity of STARDYNE and post-processor computer output data to trace maximum stress in each NUHOMS component evaluated).
Quality Assurance (QA) Program ⁽¹⁶⁰⁾	1	<ul style="list-style-type: none"> - Section 11 of topical report (Section 11 does not address 10 CFR 50, Appendix B criteria IV and VII-XIV; if NUTECH to subcontract construction of Robinson-2 ISFSI^(a) or any other ISFSI, Section 11 needs to be expanded to address each of the criteria in 10 CFR 50 Appendix B as referenced in 10 CFR Part 72, Subpart 3; NUTECH should revise its QA program in Section 11).
Additional Information Requested ⁽¹⁶¹⁾	5	<ul style="list-style-type: none"> - Ambient air temperature (need better definitions for average lifetime, upper bound for long term, and maximum short term). - In-service visual inspection program (define and recommend to users). - Leakage of helium from DSC (provide reasons why it is of no concern). - Criticality analysis methods (need more information). - Shielding considerations (need more information).
Additional Information Requested on QA Program ⁽¹⁶²⁾	22	<ul style="list-style-type: none"> - Additional information requested by NRC on 22 items (involve QA requirements for managers/administrators/supervisors, "stop work" authority, commitment on QA controls, resolution of disputes, indoctrination/training/qualification programs, commitment regarding designs, procurement documents, review of concurrence with project plans etc., surveillance on suppliers, records provided by suppliers, certificates of conformance from suppliers, hardware identification requirements, calibrating standards, special handling/shipping/etc. requirements, procedural control on bypassing required inspections/tests/etc., nonconforming items, nonconformance reports, QA records commitment, QA records transmittal/retention commitment, audit deficiency data, procurement/inspection/etc. that are important to safety, and deleting last sentence of Section 11.2 or justifying not doing so).

(a) Independent Spent Fuel Storage Installation (ISFSI).

TABLE 3.8. (contd)

Section of Topical Report	No. of Issues	Items Involved
Request for Additional Information (163,164)	1	<ul style="list-style-type: none"> - NRC requested information on items (included thermally induced bending stresses in DSC cover plates, neutron dose at HSM air inlets, source references for dose conversion factors in canister leakage analysis, cladding and HSM concrete temperature decay curves, shielding information, DOT-IV I/O, g loadings for casks, MORSE code input and output, and review of HSM skyshine calculations.

**TABLE 3.9. Examples of Some Issues^(a) Involved in Project M-40
(NAC S/T cask)**

Section of Topical Report	No. of Issues	Items Involved
Quality Assurance (QA) Program	12	- The NAC ^(b) QA program (included NRC's requests such as identify items of the NAC S/T ^(c) cask that are important to safety; describe how management, above and outside the QA organization, will regularly assess the scope, status, adequacy, and compliance of the QA program to 10 CFR 72; discuss NAC's supplier surveillance; etc.).
QA Program	4	- The NAC QA program (provide clarification and/or additional information on four open items involving six of the preceding issues.)
1.2 General Description of the Installation		
1.2.1 Principal Design Criteria	1	- Design surface dose rate (use mrem/hr consistently, not mr/hr).
2.0 Site Characteristics		
2.1 Geography and Demography of Site Selected		
2.1.2 Site Description	1	- Reference to 10 CFR 71.68(b) [should be 10 CFR 72.68(b)]; NAC statement on page 2.1-2 (effluent release component not mentioned).
2.3 Meteorology		
2.3.4 Diffusion Estimates	1	- Atmospheric diffusion factor (how calculated?; what were assumptions?; references?).
2.7 Summary of Site Conditions Affecting Construction and Operating Requirements	1	- Minimum distance from cask to ISFSI ^(d) site boundary (different values in Table 2.7-1 and Section 7.4.3).
3.0 Principal Design Criteria		
3.1 Purpose of Installation		
3.1.1 Materials to be Stored	6	- Gamma and neutron source strengths and fission product inventory (was ORIGEN used?; identify code; append output results). - Gamma results in Table 3.1-3 (why not per assembly rather than per cm of active length?; total source strength per assembly?). - Neutron spectrum results (need for design basis fuel). - Gamma source in Table 3.1-4 (why MeV/sec instead of photons/sec?; provide summary of totals per cask). - Fission product gas inventory (if xenon-131 negligible, why include?; no summary of totals per cask in Table 3.1-5; are iodine-129, cesium-134, and cesium-137 sources?). - Materials in head and foot pieces and structural materials (activation product activities not addressed; provide spectral and source information).

(a) From References 83-85. Six additional questions (and responses) may be found in References 183 and 184.

(b) Nuclear Assurance Corporation (NAC).

(c) Storage/transport (S/T).

(d) Independent Spent Fuel Storage Installation (ISFSI).

TABLE 3.9. (contd)

Section of Topical Report	No. of Issues	Items Involved
3.2 Structural and Mechanical Safety Criteria		
3.2.6	2	<ul style="list-style-type: none"> - Selection of 56 points for stress output (are highest stress points guaranteed to be included?). - Table 3.2-2 says location M_0 is closure bolt threads (in finite element analysis, how is this level of resolution achieved?).
3.3 Safety Protection Systems		
3.3.2 Protection by Multiple Confinement Barriers and Systems		
3.3.2.1 Acceptance Test Criteria	1	<ul style="list-style-type: none"> - Secondary barrier for upper closure lid (if no test can be done, what is basis for two-barrier system claim?; same applies to three-barrier claim for penetrations).
3.3.2.2 Fuel Loading Test Criteria	1	<ul style="list-style-type: none"> - Secondary barriers to radionuclide migration (none of those barriers can be pressure tested independently of primary barrier; defective secondary seal cannot be detected; confinement maintained if primary seal malfunctions?).
3.3.2.3 Fundamental Leakage Criteria	1	<ul style="list-style-type: none"> - Leak requirement of 10^{-6} atm cm^3/sec (how established?; A_2 value used?; derivation of A_2 value?).
3.3.3 Protection by Equipment and Instrumentation Selection		
3.3.3.2 Instrumentation	1	<ul style="list-style-type: none"> - Discussion of "features to provide testability" (requirement not met); cavity pressure transducer system (how tested for proper operational function?).
3.3.4 Nuclear Criticality System		
3.3.4.1 Control Methods for Prevention of Criticality	1	<ul style="list-style-type: none"> - Taking credit for PWR fuel burnup (for each initial enrichment, specify expected nominal burnup, minimum burnup limit, and expected fissile materials; discuss benefits, limitations, and potential hazards in using burnup credit; justify using neutron multiplication factor limit of 0.95; required administrative controls?; burnup meter used?; assurance that specific fuel operating histories and minimum burnups are achieved?; discuss transport of fuel on public high-ways and disposal of fuel; if fuel has to be transferred to another cask for transport, need to address this issue).
3.3.4.3 Verification Analysis	1	<ul style="list-style-type: none"> - Verification model (homogenized fuel or discrete pins assumed?; need to use 123-group neutron cross section.)
3.3.5 Radiological Protection	1	<ul style="list-style-type: none"> - Information in this section (Regulatory Guide 3.48 requirements not met; need annual collective dose estimates for various ISFSI operations).
3.3.5.3 Radiological Alarm Systems	1	<ul style="list-style-type: none"> - Cavity pressure transducer system (is there an alarm?; if not, how are personnel alerted if seal fails or neutron shield lost?).

TABLE 3.9. (contd)

Section of Topical Report	No. of Issues	Items Involved
3.3.6 Fire and Explosion		
3.3.6.1 Fire Protection	3	<ul style="list-style-type: none"> - Maximum 10°F temperature increase estimated from any site (basis?). - Analysis for fire (how are criteria taken into account?). - An 800°C fire for 30 minutes (response of cask?).
3.3.6.2 Explosion Protection	1	<ul style="list-style-type: none"> - Assumption of overpressure of 1.0 psi to qualify cask for resisting explosion (does not seem to be an adequate level).
3.3.7 Materials Handling and Storage		
3.3.7.1 Spent Fuel Handling and Storage	1	<ul style="list-style-type: none"> - Handling requirements (address maximum temperature casks attain during cask drying process after loading).
3.5 Decommissioning Considerations		
3.5.1 Storage Casks	1	<ul style="list-style-type: none"> - ORIGEN calculation (input for irradiation conditions and materials?; constant neutron flux through wall thickness?; covers and bottom excluded; consequences of cask wall activation?; compare activation product activities with limits; why is activity at one year shown?; is one year maximum time cask would remain empty before final disposition?).
4.0 Installation Design		
4.2 Storage Structures		
4.2.1.3 Properties of Materials		
4.2.1.3.2 Thermal Properties of Materials	2	<ul style="list-style-type: none"> - Density of solid neutron shield (inconsistency in values used; what is correct?). - Thermal conductivity (use of pure aluminum not conservative; use A356 and 5052 aluminum alloys).
4.2.3 Individual Unit Description		
4.2.3.2 Cask Components		
4.2.3.2.3 Radiation Protection Components	1	<ul style="list-style-type: none"> - Statement that "the neutron shields also suppress secondary gamma generation" (explain).
4.8 Normal Operations		
4.8.1 Normal Operation Conditions- Structural Analysis		
4.8.1.1 Discussion	1	<ul style="list-style-type: none"> - Fuel basket (not mentioned; include structural analysis; demonstrate that it can sustain normal operating loads without significant deformations; assess long-term behavior for its combined thermal and loading environment).
4.8.1.2 Weights and Centers of Gravity	1	<ul style="list-style-type: none"> - Use of rounded values (use actual weights; in some cases, higher weights unconservative).
4.8.1.3 Cask Body Analysis		

TABLE 3.9. (contd)

Section of Topical Report	No. of Issues	Items Involved
4.8.1.3.1 Discussion-Loads-Methods of Analysis	1	- Bounding thermal cases (only two listed, minimum ambient environment coupled with maximum heat decay load may not be covered by two listed cases).
4.8.1.3.2 Finite Element Model Description	3	<ul style="list-style-type: none"> - Model weight (inconsistency in value used; same computer model used for all axisymmetric geometries?; was wrong density or geometry input used?). - Finite element model of separate components (provide more detail in description; identify bolt beam elements; describe element/closure lid and element/cask body interfaces). - Interface between lead and inner and outer shells (explain how "gap elements" describe it).
4.8.1.3.3 Detailed Analysis	2	<ul style="list-style-type: none"> - P_{tr} and P_{cr} equations (appear incorrect; distance between supports is 155.30, not 159.8). - Title of third paragraph on page 4.8-32 (typographical error).
4.8.1.3.4 Inner Shell Stability Analysis	1	- Page 4.8-61 (L/R, not T/R).
4.8.1.5 Closure Valve and Instrumentation Feedthrough Cover Analysis		
4.8.1.5.1 Internal Pressure	1	- Internal helium gas pressure (32.4 psig or 32.4 psia?).
4.8.1.6.1 Internal Pressure Analysis	4	<ul style="list-style-type: none"> - Shield tank wall in horizontal position (include stress analysis). - Factor of 1.5 in last equation on page 4.8-88 (significance?). - Equation for S_e (appears incorrect). - Page 4.8-91 (S_e, not S_s).
4.8.1.6.2 Internal Vacuum Analysis	2	<ul style="list-style-type: none"> - Margin of safety equation (not defined). - Units on page 4.8-93 (lb, not in./lb).
4.8.1.7 Storage Support Skid Analysis	2	<ul style="list-style-type: none"> - Symbols A_t, A_s, A_{by}, etc. (define; do this throughout report). - Figure on page 4.8-111 (unclear; all figures should have captions).
4.8.2.2 Thermal Properties of Materials	1	- Coefficient equation for vertical storage mode (different than one in reference; discuss validity of equations and uncertainties).
4.8.2.3 Thermal Analysis Models		
4.8.2.3.1 Cask Body	1	- Thermal analysis (show all dimensions and materials in Figure 4.8-15).
4.8.2.3.2 Fuel Storage Basket	2	<ul style="list-style-type: none"> - Aluminum storage basket model (show all dimensions, particularly gaps, and equivalent thickness of homogeneous material). - Fuel bundles (how modeled as heat sources?).
4.8.2.3.3 Maximum Fuel Rod and Temperature	4	<ul style="list-style-type: none"> - Fuel rod analysis (parameters used?; parameter values used in Wooten-Epstein correlation?). - Fuel rod (maximum internal pressure?; include data in Table 3.1-1). - Temperature versus time in storage (provide plot for hottest fuel rod). - Heating 5 analysis and thermal model (show input and output).

TABLE 3.9. (contd)

Section of Topical Report	No. of Issues	Items Involved
4.8.2.6 Normal Operations Conditions- Multicask Arrays	1	- MULTICASK program (describe; what are modeling assumptions?; was it benchmarked?).
4.8.3 Normal Operations Conditions- Criticality Analysis		
4.8.3.2 Keno Model	1	- Use of 0.95 neutron multiplication factor limit (requires an improved model; square cask array may not be conservative; use discrete pin model and 123-group neutron cross-section; provide information on aluminum and boron sheet thicknesses).
4.8.3.3 Criticality Analysis Results	1	- Calculations of burnup, uranium-235 depletion, and fissile isotope fractions (computer code used?; show input and output; number of calculations performed for enrichment and burnup limits in Table 4.8-37?; percentages of fissile isotopes present?; criticality sensitivity to burnup?).
5.0 Operation Systems		
5.1 Operation Description		
5.1.1 Narrative Description		
5.1.1.1 Initial Receipt	1	- Breaking of pressure transducer wires (consequences of break with respect to repair and acceptance?).
5.1.1.2 Fuel Loading	2	- Burnup of each fuel bundle (how verified?; burnup meter used?; 100% administrative control difficult to achieve). - Drying process (how is fuel rod temperature controlled?).
5.1.1.4 ISFSI Storage	1	- Regulatory Guide 3.48 requirement to "describe means that will be routinely used during storage to evaluate the condition of the casks" (no discussion provided, although annual inspection appears to be considered; provide discussion of means and of capability for remote monitoring of internal cask pressure).
5.1.2 Flowsheets	3	- Simplified flow sheets in Table 5.1-1, Table 5.1-2, and Table 5.1-3 (need to be annotated to show specific operations from Section 5.1.1.1, Section 5.1.1.2, and Section 5.1.1.3 associated with each box). - Times and distances in Table 5.1-4 (how estimated?; are they conservative?; need time, personnel, and distance estimates for all operations with more than one task to compute occupational dose).
5.1.3 Identification of Subjects for Safety Analysis		
5.1.3.2 Chemical Safety	1	- Neutron shield mixture "requires only normal care in its use" (meaning?; routine checks or replacement of mixture required?).
5.1.3.4 Instrumentation	1	- Pressure transducer (failure impairs ability of utility to detect cask with leaking seal or cask that has lost neutron shield).

TABLE 3.9. (contd)

<u>Section of Topical Report</u>	<u>No. of Issues</u>	<u>Items Involved</u>
5.2 Fuel Handling Systems	2	<ul style="list-style-type: none"> - Spent fuel storage operations involving placement, storage surveillance, and removal (summarize; refer to and use Section 5.1.1 material). - Safety-related features of spent fuel storage system (summarize; state limits for a commitment to action; refer to alarm levels).
5.3 Other Operating Systems		
5.3.2 Component/Equipment Spares	1	<ul style="list-style-type: none"> - This section (no NAC^(a) response provided; address associated ISFSI repair or maintenance design features; discuss pressure transducer system reliability; address design provisions to minimize radiation exposure during repair operations).
5.4 Operation Support Systems	1	<ul style="list-style-type: none"> - Pressure transducer system instrumentation (discuss; address monitoring).
6.0 Waste Confinement and Management	1	<ul style="list-style-type: none"> - Radiological impact of gaseous radioactive wastes, i.e., effluents, in normal operations (need summary).
7.0 Radiation Protection		
7.1 Ensuring that Occupational Radiation Exposures Are As Low As Is Reasonably Achievable (ALARA)		
7.1.2 Design Considerations	1	<ul style="list-style-type: none"> - Statements that "liquid neutron shield tank leakage will be visually obvious" and that "no close-proximity inspections of the cask are required" (how will leak be obvious from a distance?; how long will leak be "visually obvious"?).
7.1.3 Operational Considerations	1	<ul style="list-style-type: none"> - Statement that "...worst effect of a man-induced accident would be puncture and loss of the liquid neutron shield" (operational procedures in this event?).
7.2 Radiation Sources		
7.2.1 Characterization of Sources	3	<ul style="list-style-type: none"> - Gamma source strength (inconsistency in values used); neutron source spectrum (where is it?); head and foot piece regions (gamma source strengths?); active fuel (why total energy flux/cm instead of total photon flux/cm?). - Subcritical multiplication and associated fission gammas (any consideration of these?). - Requirement that "sources of radiation... be described in the manner needed as input to the shielding design calculations" (inadequate information in this section and in Section 3.1.1).
7.2.2 Airborne Radioactive Material Sources	2	<ul style="list-style-type: none"> - Various release fractions (references?; why are iodine-129, cesium-134, and cesium-137 not considered?). - Requirement to describe "provisions made for personnel protective measures" (any provisions made?; if not, why not?).
7.3 Radiation Protection Design Features		

(a) Nuclear Assurance Corporation (NAC).

TABLE 3.9. (contd)

Section of Topical Report	No. of Issues	Items Involved
7.3.1 Installation Design Features	1	- Specific activity, physical and chemical characteristics, and expected concentrations of all sources in Section 7.2 (need to provide here; discuss design radiation dose rate); requirement of regulatory position 2 of Regulatory Guide 8.8 (not fulfilled).
7.3.2 Shielding	1	- Shielding calculations (flux-to-dose conversion factors used?); QAD-CG 3D calculations (rectangular parallelepiped source regions homogenized?); gamma dose buildup factors (provide more information); material buildup (why product of factors used?); all assumptions (substantiate).
7.3.2.1 Analysis Source Description	5	- Reaction of americium-241 [(a,n), not (q,n)]. - Percent of neutron source from spontaneous fissions of curium-242 and -244 (inconsistency in values used; why the difference?; append ORIGEN results). - Statement that "curium spontaneous fission neutron spectrum is quite similar to that of californium-252" (basis?; if californium-252 spontaneous fission spectrum used, why not presented?). - QAD-CG three dimensional source region (are there differences in sources 1, 2, and 3 in Figure 7.3-2?; if not, why numbered as shown?). - Source region materials in Table 7.3-2 (where are atom densities?; where is total density for active fuel source region?).
7.3.2.2 Shielding Analysis Dose Points	1	- Dose point locations in Figure 7.3-3 (present actual locations by coordinate position; dose points 1, 4, and 8 incorrectly shown); locations for maximum dose (how determined?; other dose points involved?).
7.3.2.3 Shielding Analysis Models	2	- QAD-CG three dimensional model in Figure 7.3-6 (coordinates for fuel region and voids?; axial coordinate references?; where are head piece, foot piece, gas plenum, and active fuel regions of source region?). - Shield material densities and compositions in Table 7.3-3 (atom densities in atoms/barn-cm for shield materials?; C for ethylene glycol?; K in potassium tetraborate?; Si in silicon value for cumulative density of silicon rubber compared to Table 4.2-15; why is cumulative density of stainless steel 7.30 g/cc instead of 7.9 g/cc?).
7.3.2.3 Shielding Analysis Results-Surface Dose Rates	1	- Dose rate at indicated location (at a point or average about the location?; uncertainties with dose rate results?; shouldn't this be Section 7.3.2.4?).
7.4 Estimated On-Site Collective Dose Assessment		
7.4.1 Analysis Methodology	1	- Twenty-meter cutoff between $1/r$ and $1/r^2$ (how determined?; approach for calculating off-surface dose (needs to be validated)).
7.4.2 Analysis Results	1	- Use of term "ISFSI boundary" for 100-meter location (controlled area boundary is more accurate term); Table 7.4-2, Table 7.4-3, Table 7.4-4, and Table 7.4-5 (confusion because term "ISFSI boundary" used); multiple cask arrays (for 25 mrem/yr requirement, show how location computed).

TABLE 3.9. (contd)

Section of Topical Report	No. of Issues	Items Involved
7.4.3 Occupational Dose	1	- Personnel exposure time history (Table 5.4-4 should be Table 5.1-4); dose rates in Table 7.4-6 (how computed?; was DDSE used?; dose point location (what used for each task?); multitask operations (exposure duration and distance?); occupational doses in Table 5.1-4 and Table 7.4-6 (insufficient information to verify doses in latter table); optional annual inspection (dose?); dose rates in Table 7.4-6 (really as low as about 10^{-5} ?).
7.4 Airborne Radiation		
7.4.4.2 Analysis-Normal Operations Conditions Basis	5	- Section title (inconsistent with assumption of 10% failed fuel rods). - Values for A_2 for krypton-85 and xenon-131 (why are uncompressed gas values used?). - Reference to Section 3.2.2.3 on page 7.4-21 (isn't it Section 3.3.23?). - Downstream pressure of 2 atmospheres (how set?; Section 4.8.1.5.1 says 1 atmosphere). - A_2 values (how related to 10 CFR 72 requirements?; calculational method for A_2 for mixture of isotopes is incorrect).
7.4.4.3 Analysis-Off-Normal Operations and Accident Event Conditions	1	- Section title (inconsistent with assumption of 100% failed fuel rods; 100% represents accident conditions only; comments on A_2 values and downstream pressure apply here also).
7.4.4 Boundary Dose	1	- "Normal" and "Off-Normal/Accident" (titles should be consistent with earlier definitions); whole body dose rate from hydrogen-3 inhalation (basis for computing as if it were for krypton-85?); diffusion factor (is it conservative?; why different than values in Safety Guide 1.25?).
7.6 Estimated Off-Site Collective Dose Assessment	1	- The 100-meter boundary (inconsistent); off-site dose (need some generic estimate).
7.7 NAC S/T Cask Performance	1	- Section 7.4.4.2 "analyzes leakage for normal and off-normal conditions" and Section 7.4.4.3 "for accident conditions" (analyses consistent with statements but not consistent with titles of sections themselves).
8.0 Accident Analysis		
8.1 Off-Normal Operation	1	- Scenario (is this for 10% failed fuel rod case?; according to Section 7.2.2 it should be).
8.1.1 Event-Leakage Through a Cask Closure		
8.1.1.3 Analysis of Effects and Consequences-Leakage Through a Cask Closure	1	- Contained barrier seal develops leak (specify magnitude; amount of helium lost?; support by facts that radiological impact is "minimal").
8.1.2 Event-Fission Product Gas Release	1	- Considered as off-normal event (cannot be according to Section 7.2.2.2; 100% failed fuel rods is an accident event).

TABLE 3.9. (contd)

Section of Topical Report	No. of Issues	Items Involved
8.1.2.1.3 Shielding Analysis-Loss of Neutron Shield	2	<ul style="list-style-type: none"> - Combined gamma and neutron dose rates in Table 8.2-14 (comments in Section 7.3.2.3 apply here also). - Meeting requirements of 10 CFR 72.68(b) for various cask array scenarios (NRC questions analysis procedure; why were not doses computed for minimum distances shown in Section 7.4.2?).
8.1.2.3 Analysis of Effects and Consequences-Fission Product Gas Release	2	<ul style="list-style-type: none"> - Statement "that leakage and airborne radiation are far below established criteria" (substantiate; summarize results from Section 7.4.4). - Total free gas per rod value on page 8.1-8 (how determined?).
8.1.2.4 Corrective Action-Fission Product Gas Release	1	<ul style="list-style-type: none"> - Statement that "cask safely contains all fission products released from the fuel rods" (incorrect, some loss with leakage).
8.1.3 Event-Failure of Instrumentation		
8.1.3.4 Corrective Actions-Failure of Instrumentation	2	<ul style="list-style-type: none"> - Statement that no corrective action required if pressure transducer system fails (NRC does not agree). - Rupture of neutron shield (will storage conditions be outside of operating limits?; if so, a failed pressure transducer system should be repaired or replaced).
8.1.4 Radiological Impact from an Off-Normal Event	2	<ul style="list-style-type: none"> - Statement that "fission product gas release" will have no radiological impact" because it does not "involve leakage from the cask" (not correct as there is an established leak rate). - The use of the 100-meter "ISFSI boundary" (inconsistent; substantiate statement that radiological impact at 100 meters is "less than 5 rem criterion").
8.2 Accidents		
8.2.1 Accident-Loss of Neutron Shield		
8.2.1.1 Cause of Accident-Loss of Neutron Shield	1	<ul style="list-style-type: none"> - Remote monitoring of casks (how will leak be detected?; seems to require close-proximity visual inspection).
8.2.1.2 Accident Analysis-Loss of Neutron Shield	1	<ul style="list-style-type: none"> - Neutron shield loss (will there be pressure increase?; if so, would be direct structural consequence of accident).
8.2.1.2.1 Thermal Analysis-Loss of Neutron Shield	2	<ul style="list-style-type: none"> - Reference for thermal analysis (Table 4.8-17, not Table 4.8-14). - Heat transfer across neutron shield annulus (how calculated?).
8.2.1.2.3 Shielding Analysis-Loss of Neutron Shield	3	<ul style="list-style-type: none"> - Neutron shield loss (repair procedures?). - Neutron shield loss (occupational dose consequences?). - Neutron shield loss (provide thermal analysis).
8.2.1.3 Accident-Cask Burial Under Debris		
8.2.1.3.2 Accident Analysis-Cask Burial Under Debris	1	<ul style="list-style-type: none"> - Cask burial analysis (provide thermal model).
8.2.2 Accident-Fire	1	<ul style="list-style-type: none"> - Rupture of neutron shield or slumping of lead (could fire cause these?; if so, address radiological consequences).

TABLE 3.9. (contd)

Section of Topical Report	No. of Issues	Items Involved
8.2.3 Accident-Cask Tip Over	1	- Statement that "there are no credible means by which the... cask could experience a tip-over" (not true, as discussed in Section 8.2.10 below).
8.2.3.2 Accident Analysis-Cask Tip Over	2	- Rupture of fuel rods or slumping of lead (could these occur if cask tips over?; if so, address radiological consequences). - Statement that center of gravity moves through 58.71 vertical inches, which is not as severe as a 55 g oblique drop accident (need supporting discussion as it is not obvious).
8.2.3.2.1 Structural Analysis-Tipping Against Another Cask	4	- Most severe angle is 45° if cask tips and strikes a horizontal cask (corresponding figure, Figure 8.2-2, is unclear). - Notation on page 8.2-55 (confusing; difference between m and M and between r and R?). - Calculation of moment of inertia (incorrect). - Rotational kinematics discussion (incorrect).
8.2.3.3 NAC S/T Cask Performance-Cask Tip Over Accident	1	- Statement that 1SFSI site boundary dose is less than 10 CFR 72 criteria if neutron shield tank lost and fuel ruptures (need to be more specific; dose at certain distance for single tipped cask and for an array of casks?).
8.2.4 Accident Cask Drop		
8.2.4.1 Discussion	1	- "...design impact force of 55 g on the cask is established" (define meaning of 55 g; discuss impact limiter and maximum drop height).
8.2.4.2 Detailed Analysis	3	- Cask drop (does rupture of some fuel rods or slumping of lead occur?; if so, address radiological consequences). - Loading (is this a static calculation?; inertially applied impact loadings may be nonconservative). - Cask contents excluded from analysis because they produce negligible stress of 37 psi (NRC's rough calculation shows 745 psi).
8.2.4.2.2 Side Drop	2	- Load distribution in Figure 8.2-4 (any supporting experimental or theoretical evidence?; were sufficient terms used in Fourier expression?). - Plots, described on page 8.2-146, to adjust radial stresses (include sample plots).
8.2.4.2.3 Analysis-Oblique Drop	1	- Statement that "slapdown" is less severe than initial impact due to larger impact area (large impact areas generally imply higher stress; address "slapdown" as a secondary impact that may increase stresses).
8.2.5 Accident-Flood		
8.2.5.2 Accident Analysis-Flood		

TABLE 3.9. (contd)

Section of Topical Report	No. of Issues	Items Involved
8.2.5.2.2 Structural Analysis	3	<ul style="list-style-type: none"> - Tip-over of vertically stored cask or roll-over of horizontally stored cask (will neutron shield tank collapse?). - P_v, P_v, and $*P_v$ on page 8.2-266 (confusing; define); reference to page 3-55 of Mark's Mechanical Engineering Handbook (delete; include reference to Bernoulli's equation and list assumptions used). - Overturning moment (lower if water buoyancy taken into account and, as consequence, lower water velocity for overturning).
8.2.6 Accident-Cask Seal Leakage		
8.2.6.2 NAC S/T Cask Performance-Cask Seal Leakage	2	<ul style="list-style-type: none"> - Whole body dose from hydrogen-3 inhalation (as noted above in Section 7.4.4.4, why computed as krypton-85?). - Diffusion factor (comments above also relevant here; dose from single cask higher if more conservative value of 5.0×10^{-2} used).
8.2.7 Accident-Tornado Winds	1	<ul style="list-style-type: none"> - Bernoulli's equation (see comments in Section 8.2.5.2.2).
8.2.8 Accident-Tornado Missiles		
8.2.8.1 Cause of Accident-Tornado Missiles	1	<ul style="list-style-type: none"> - Impact of massive missile on cylindrical cask (seems inconceivable it can result in uniform external pressure).
8.2.8.2 Accident Analysis-Tornado Missiles		
8.2.8.2.1 Analysis-Massive Missile Vertical Cask-Tip Over	5	<ul style="list-style-type: none"> - Reference to Figure 8.2-1 on page 8.2-297 (seems to be wrong). - On same page, W_h called final potential energy (it is initial potential energy). - On page 8.2-294, angular velocity symbol (missing). - Angular velocity of cask after impact (isn't this W_i?). - Impact force on cask (function of missile deceleration rather than force necessary to maintain equilibrium).
8.2.8.2.3 Analysis-Protective Barrier Missile Vertical Cask Tip-Over	1	<ul style="list-style-type: none"> - Paragraph on page 8.2-361 identical to one on page 8.2-357 (accuracy of former?).
8.2.10 Accident-Earthquakes		
8.2.10.2 Accident Analysis-Earthquakes Natural Frequency Analysis	3	<ul style="list-style-type: none"> - This section (unnecessary). - Tip-over analysis for vertical cask (only 1 of 5 cases considered is relevant here; when corrected, conclusion is revised--the cask <u>will</u> tip in the event of the cited earthquake). - Roll-over analysis for horizontal cask (again, only 1 of the 5 cases considered is relevant here; that case also incorrectly calculated; when corrected, conclusion is reversed--cask will roll off of its skid during Design Basis Earthquake east of Rockies under worst conditions).
8.4 References-Section 8	1	<ul style="list-style-type: none"> - References (some are missing; they are probably on page 8.4-1, which is also missing).

TABLE 3.9. (contd)

Section of Topical Report	No. of Issues	Items Involved
10.0 Operating Controls and Limits		
10.1 Proposed Operating Controls and Limits		
10.1.2 Bases for Operating Controls and Limits	1	- Assumption that cask dynamic loading will result in a "g" load not exceeding 55 g (establish operation limits so cask will not experience deceleration greater than 55 g).
10.1.2.1 Fuel Characteristic Limit	2	- Gamma flux in Table 10.1-1 (why MeV/sec instead of photons/sec?); distance from cask to boundary (for single cask, minimum distance is 187.2 meters, not 100 meters; why not used?); atmospheric diffusion factor (is it conservative?). - Page 10.1-4 (1×10^{-6} , not 1×10^6).

TABLE 3.10. Examples of Some Issues^(a) Involved in Project M-41 (Westinghouse, MC-10)

Section of Topical Report	No. of Issues	Items Involved
1.2 General Description of Installation		
1.2.5 Structural Features	1	- Containment of neutron absorber (description is not complete; function of 0.25-in. thick steel plates?).
1.3 General Systems Description		
1.3.1 Cask Configuration	1	- Neutron absorber in third cover in Figure 1.2-1 (identify for clarity).
1.3.2 Dry Storage Cask Design Basis	5	<ul style="list-style-type: none"> - Maximum neutron and gamma surface dose rates versus principal design criteria for neutron plus gamma (explain; is maximum surface dose rate for gamma correct?). - Metal ring closure design (description is confusing; welded seal ring or metallic seal or both?). - Outer cask wall (indicate that some fraction has no neutron absorber and has increased local neutron dose); neutron absorber in seal cover and bottom (description of encapsulation not complete). - Neutron absorber and shield ring (point out location in Figure 1.3-5; describe shield ring attachment). - Prevention of criticality function (effect of spacer pad deformation?).
1.3.2.2 Fuel Parameters	1	- Westinghouse 17x17 fuel assembly in Figure 1.3-6 (why used if 15x15 fuel assembly is the reference?).
3.0 Principal Design Criteria		
3.1 Purpose of Installation		
3.1.1 Materials to be Stored	3	<ul style="list-style-type: none"> - Radiological characteristics of spent fuel to be stored (requirement to describe not met; refer to material in Section 7.2.1; summarize spectral and source information). - Gaseous activity sources (information in this section and Section 7.2.1 is incomplete; address fission product gas inventory). - Activation product activities in head and foot piece materials and structural materials (need to address; provide spectral and source information).
3.2 Structural and Mechanical Safety Criteria		
3.2.2 Water Level (Flood) Design	1	- Neutron shielding panel's ability to withstand >100-ft static water head (define more precisely this statement: "without being grossly deformed").
3.2.5 Combined Load Criteria	1	- Allowable for accident conditions $P_m + P_b$ (should be $3.6 S_m$ or S_u , not $3.6 S_m$ or $1.05 S_u$).
3.3 Safety Protection Systems		
3.3.1 General	1	- "Shielding protection... with welded encapsulating structure..." (needs clarification; implies cask surrounded by welded steel structure during handling).

(a) From References 92-94 and 185. Additional issues are in Reference 186. Responses are in References 93, 94, and 186-201.

TABLE 3.10. (contd)

Section of Topical Report	No. of Issues	Items Involved
3.3.2 Protection by Multiple Confinement Barriers and Systems		
3.3.2.2 Drydown Procedure	2	<ul style="list-style-type: none"> - Shield plate (where is it in Figure 3.3-17; is shield plate same as shield cover?). - System moisture content (references?).
3.3.2.3 Multiple Confinement Barriers	4	<ul style="list-style-type: none"> - Metallic seal (inconsistency in three figures). - Leak rate criteria for each cover and combined cover system (what is it?). - "g" in equation for F_g (significance?). - Static g load (how computed?).
3.3.3 Protection by Equipment and Instrumentation Selection		
3.3.3.2 Instrumentation	2	<ul style="list-style-type: none"> - Instrumentation calibration port (inconsistency in location as shown on two figures). - Description of pressure monitoring operation ("continuous" not stated here but is in Section 10.2.3).
3.3.4 Nuclear Criticality System		
3.3.4.1 Control Methods for Prevention of Criticality	1	<ul style="list-style-type: none"> - Borated water environment for underwater loading and unloading (why used?; explain why use of nonfixed poison is considered acceptable).
3.3.4.1.4 KENO Geometry Models	2	<ul style="list-style-type: none"> - Fuel cell retainer clips (run full length of cell?). - Homogeneous and discrete fuel models in benchmark calculations (provide more detailed description; neutron cross-section data used?); discrete fuel model and assumptions (similar to those used in benchmarking KENO-IV code against criticality tests?); evaluation of hard neutron spectrum effects (use KENO V to model discrete fuel rods and use 123 neutron cross-section library).
3.3.4.1.5 Results of Criticality Analysis	2	<ul style="list-style-type: none"> - Evaluation of increased reactivity with off-center fuel and minimum water gap (clarify description; reevaluation needed?). - Fuel enrichment (restrict to 3.6 or less until reevaluation ensures that K_{eff} is <0.95 or this type of fuel assembly excluded for use).
3.3.4.1.6 Accident Analysis	1	<ul style="list-style-type: none"> - Cask drop event (indicates cask basket would not maintain its structural integrity); unborated water in cask cavity (criticality could occur); basket design (explain why considered acceptable); cask tip over on storage pad (could fuel be readily removed?).
3.3.5 Radiological Protection	2	<ul style="list-style-type: none"> - Gaseous radioactivity sources (describe also in Section 3.1.1). - Maximum surface dose rate (list as design criteria; also cite criteria in 10 CFR 72.67 and 10 CFR 72.68).
3.3.5.2 Shielding	2	<ul style="list-style-type: none"> - Estimate of collective doses (in man-rem) per year for various operations (Regulatory Guide 3.48 requirement not met; provide summary of collective dose for major operations; summary of dose to individual is of interest but not required). - Dose at controlled area boundary and minimum site boundary distance (why no summary presented to meet 10 CFR 72.67 requirements for single and multiple cask arrays?).

TABLE 3.10. (contd)

Section of Topical Report	No. of Issues	Items Involved
3.5.3.5 Radiological Alarm System	2	<ul style="list-style-type: none"> - "Unacceptable" in statement "unacceptable releases..." (meaning?; refers to 10 CFR Part 20 and 10 CFR Part 72?); release of 1 rem (why considered "small magnitude?"); alarms (why considered unnecessary?). - Sections 4.3.7 and 5.4.1 (referenced here but do not exist).
3.5 Decommissioning Considerations	9	<ul style="list-style-type: none"> - Expression "cask itself" (includes the covers?; if so, which ones?; if not, why excluded?). - "in-house computer code" for activation product activity calculations (provide more detailed description). - $^{58}\text{Ni}(n,\gamma)^{59}\text{Ni}$, $^{62}\text{Ni}(n,\gamma)^{63}\text{Ni}$, and $^{54}\text{Fe}(n,\gamma)^{55}\text{Fe}$ activation reactions (why not considered?). - Individual isotopic results (why not included in Table 3.5-1?). - Three cask components (total weights used for each?). - Reference time for activities (what was it?; was it one day after unloading?). - Cask considered as low-level waste (justify; relate to 10 CFR Part 61). - Flux-to-dose conversion factors used with KAP-VI code (what were they?). - Statement that cask components can be readily shipped (justify; relate to 49 CFR Part 173); possible recycle for commercial use (discuss; relate to 10 CFR Part 30).
4.2 Storage Structures	1	<ul style="list-style-type: none"> - Lifting yoke, lifting sling, and shield lifting cables (include stress analysis here if they are to be included in this topical report).
4.2.3 Individual Unit Description		
4.2.3.1 Cask Body and Containment Vessel	1	<ul style="list-style-type: none"> - Statement "prevents fission products streaming" (meaning?; refers to gases and particulates?).
4.2.3.2 Basket Assembly	2	<ul style="list-style-type: none"> - Method of basket assembly (description unclear; provide more detailed fabrication drawings). - Page 4.2-4 (upside down).
4.2.3.3 Lid Closure and Seal System	4	<ul style="list-style-type: none"> - Metal closure ring (same as G-ring?; if so, NRC questions reuse as a safe practice). - Filling seal cover cavity with neutron absorber (how is void-free fill guaranteed?; cavity filled after seal cover is in place on loaded cask?). - Twelve bolts in seal cover (describe purpose). - Seal ring "placed over seal weld" (any means of securing it to cask?).
4.2.4 Thermal and Hydraulic Design Basis		
4.2.4.6 Fuel Rod Cladding Maximum Temperatures	3	<ul style="list-style-type: none"> - Experience data base for recommending 380°F as maximum (data base, as yet, insufficient); assessing acceptability of stated temperature limit (provide temperature decay curve for at least 20-yr storage period and maximum anticipated fuel rod internal pressure). - Maximum fuel rod temperature calculation (provide copy of Reference 2); test fuel assembly versus storage fuel assembly (same configurations?); boundary conditions or basket configurations (how taken into account in referenced equation?). - Basket, stainless steel sleeves and cask walls (clearances between them at normal operating conditions?).

TABLE 3.10. (contd)

Section of Topical Report	No. of Issues	Items involved
4.2.5 Structural Analysis	4	<ul style="list-style-type: none"> - All structural analyses (include references to detailed drawings and/or sketches; free body diagrams would be useful). - All equations used (list assumptions and cite references; use most recent edition of cited references, if possible). - Particular "load amplification factors" (provide justification for selections). - Each component analyzed (specify design temperature used).
4.2.5.3 Materials	2	<ul style="list-style-type: none"> - Materials properties required for thermal stress evaluations (data missing; provide properties). - All materials in Table 4.2-13 (cite ASME reference).
4.2.5.4 Vessel Analysis	2	<ul style="list-style-type: none"> - Radius-to-thickness ratio of cask is about 5:1 (justify use of thin shell theory for calculation of stress due to pressure). - Shell wall stresses due to lifting and handling loads (not included in this section; refer to discussion in Section 4.2.5.13); bending and membrane stresses (combine in Table 4.2-15).
4.2.5.9 Shield Cover Stud Analysis	1	<ul style="list-style-type: none"> - Section b ($P+Q = 34,735$ psi, not $P+Q+34,735$ psi).
4.2.5.10 Primary Cover Cap Screw Analysis	1	<ul style="list-style-type: none"> - Section b ($P+Q = 36,945$ psi, not $P+Q+36,945$ psi).
4.2.13 Trunnion and Attachment Analysis	2	<ul style="list-style-type: none"> - Analysis (sketch is needed); lifting hook moment arm is 6.5 in. (why is 6.5-3.5 used in equation?). - Statement that safety factor of 5:1 used in trunnion design (not clear from stress summary that this factor applied).
4.2.5.14 Internal Basket Analysis	5	<ul style="list-style-type: none"> - Equation for σ_b (source?; does not appear to be in units of stress). - Basket thermal stress, especially regarding possible constraints against free expansion (demonstrate negligible character in more detail). - Basket stress levels versus stress intensity limits (need to compare). - Fatigue failure criteria due to thermal stresses (within ASME code requirements?). - Failure mode for aluminum basket (is creep a significant mode?).
4.2.5.15 Shipping Skid Analysis	1	<ul style="list-style-type: none"> - Trunnion diameter (inconsistency in value in this and two other sections; clearly indicate correct diameter).
4.2.5.16 Structural Analysis Summary	8	<ul style="list-style-type: none"> - BORAL and NS-3 (provide information demonstrating properties are adequate). - Accident allowables for primary plus bending stresses (values in Table 4.2-16 are incorrect; should be lesser of $3.6 S_m$ or S_u). - Accident stud and screw allowables (values in Table 4.2-16 are incorrect; should be $2.4 S_m$). - Stress intensity limit for pure shear stresses ($0.6 S_m$, not S_m). - P_m for seal cover cap screws (text and table values inconsistent). - P_m for trunnion (text and table values inconsistent; text says this is shear stress, not primary membrane stress).

TABLE 3.10. (contd)

Section of Topical Report	No. of Issues	Items Involved
		<ul style="list-style-type: none"> - Seal cover weld, basket, and shipping skid (allowables missing in table). - Trunnion and trunnion bolts (they are in category of special lifting devices; must meet requirements of ANSI N14.6, Sections 3.2.1 and 6.2.1; allowances in Table 4.2-15 are inappropriate).
4.2.6 Material Evaluation		
4.2.6.2 Discussion of Major Materials	2	<ul style="list-style-type: none"> - Behavior of aluminum in dry storage (include some computations). - Aluminum oxide (amount of moisture absorbed?; significance of it compared to moisture left after cask drained?).
4.5 Shipping Cask Repair and Maintenance	1	<ul style="list-style-type: none"> - Cask repair and maintenance operations (present generic discussion).
5.0 Operation Systems		
5.1 Operation Description		
5.1.1 Narrative Description		
5.1.1.7 Removing the Cask from the Pool	1	<ul style="list-style-type: none"> - Operations 4, 5, and 6 (inconsistency in statements regarding raising of cask; explain).
5.1.1.10 Cask Maintenance and Inspection	2	<ul style="list-style-type: none"> - "High pressure" and low pressure" alarms (truly indicative of optical and audible alarms?; is "high pressure alarm" level 2.88 psia?; is "low pressure alarm" level 1.0 psia?; why is "low pressure alarm" setting so much below design steady state internal pressure of 2.5 psia?). - Operations 1 and 2 (performed remotely?); Operations 3 and 4 (indicate that they are performed in close proximity to cask surface).
5.1.1.12 Cask Unsealing	1	<ul style="list-style-type: none"> - Operation 1, opening of cask primary lid/seal lid vent (any radiation hazard via inhalation to individuals?).
5.1.1.13 Cask Preparation for Unloading Operations	1	<ul style="list-style-type: none"> - Operation 1, removal of cask cavity vent closure and attachment of high-temperature, high-pressure rated hose (any radiation hazard via inhalation to individuals?).
5.1.2 Flowsheets	2	<ul style="list-style-type: none"> - Simplified sequence of operations in Table 5.1-1 (break down into arrival, loading, storage, unloading, etc.). - Detail on source terms for radiation exposure determinations (Regulatory Guide 3.48 requirements not met; provide information, some of which is in Section 7.4.4.1; discuss time and distance estimates and whether or not they are conservative).
5.1.3.4 Instrumentation	1	<ul style="list-style-type: none"> - Monitoring performance of some or all casks (inconsistency between statements in this section and Sections 5.1.1.10 and 10.2.3; clarify).
5.1.3.5 Maintenance Techniques	1	<ul style="list-style-type: none"> - Replacement and recalibration of monitoring instrumentation (present generic discussion).

TABLE 3.10. (contd)

Section of Topical Report	No. of Issues	Items Involved
5.2 Fuel Handling Systems	2	<ul style="list-style-type: none"> - Spent fuel storage operations (include placement, storage, surveillance, and removal; summarize; cite Section 5.1.1 for materials). - Safety-related features for safe operation under both normal and off-normal conditions (summarize; state limits for commitment to action; cite alarm levels presented in Section 5.1.1.10.).
5.3 Other Operating Systems	1	<ul style="list-style-type: none"> - Component/equipment spares (where is response required by Section 5.3.2 of Regulatory Guide 3.48?); continuity of safety under normal and off-normal conditions (address design features associated with repair/maintenance operations); pressure monitoring system (reliability?); exposure to radiation during repairs (design provisions to minimize exposure?).
5.4 Operation Support Systems	1	<ul style="list-style-type: none"> - Pressure transducer system instrumentation (discuss); monitoring and redundancy of safety features to ensure adequate safety (need to address).
6.0 Waste Confinement	1	<ul style="list-style-type: none"> - Radiological impact of normal operations associated with gaseous radioactive waste (summary required); gaseous radioactive sources discussed in Section 3.3.5 but not here (provide discussion).
7.0 Radiation Protection		
7.1 Ensuring that Occupational Radiation Exposures Are As Low As Is Reasonably Achievable (ALARA)		
7.1.2 Design Considerations	2	<ul style="list-style-type: none"> - Cask repair and maintenance procedures and required times (need discussion); occupational exposure during maintenance (indicate design directed for minimizing exposure); regulatory position 2 of Regulatory Guide 3.48 (requirements not fulfilled; need to address each section separately and sequentially); sections that are site specific or not relevant (identify; provide discussion where relevant). - Dose rate and individual dose for a single operation (why not cite "highest dose rate" and "highest individual dose for single operation"?).
7.2 Radiation Sources		
7.2.1 Characterization of Sources	3	<ul style="list-style-type: none"> - "Storage canisters" on page 7.2-1 (what are they?). - ORIGEN calculation (basic input parameters?; can output be appended?). - "Fuel assembly structure combination" (include gas plenum and head and foot piece regions?; from Section 7.2.1.2, appears that only grid structure included).
7.2.1.1 Neutron Source Terms	1	<ul style="list-style-type: none"> - Source of radiation needed as input for shielding design (discussion here and in Section 3.1.1 is incomplete; where is neutron source spectrum?).
7.2.1.2 Gamma Ray Source Terms	4	<ul style="list-style-type: none"> - Gamma ray source terms (requirements of Regulatory Guide 3.48 not satisfied); 20-group versus 7-group gamma ray structure (inconsistency as to what used here and in Section 7.3.2.2). - Head and foot piece regions (source terms?; results included in Table 7.2-4?). - Gamma source strengths (in Table 7.2-4, why are units MeV/sec instead of photons/sec?).

TABLE 3.10. (contd)

Section of Topical Report	No. of Issues	Items Involved
		<ul style="list-style-type: none"> - Says "...relative importance of various groups of gamma rays" discussed in Section 7.3.2 (neither 7- nor 20-group structure discussed in that section).
7.2.2 Airborne Radioactive Material Sources	1	<ul style="list-style-type: none"> - Response to this section (incomplete; information is in Sections 3.3.5, 8.2.1.2, and 8.2.1.3; need to address fission product gas inventory).
7.3 Radiation Protection Design Features		
7.3.1 Installation Design Features	3	<ul style="list-style-type: none"> - Specify activity, physical and chemical characteristics, and expected concentrations of all sources in Section 7.2 (provide information); design radiation dose rate for storage area and maintenance and repair activity and estimates of radioactive materials that might be discharged during storage (provide information; refer to other sections of report); regulatory position 2 of Regulatory Guide 3.48 (requirements not fulfilled). - Dose rate reduction by shield cover (what is "tolerable level consistent with ALARA principles"?; shouldn't this level be quantified?); meeting radiation protection objectives during cask cleanup and sealing (quantify?). - Reference to Section 7.3.2.1 regarding radiation levels when closing of cask (incorrect; no discussion in that section).
7.3.2.1 Radiation Shielding Design Features	1	<ul style="list-style-type: none"> - Figure 7.3-1 (is active fuel region zone A?; is gas plenum region zone B?; are zone C, the top end plugs, and zone D, the fuel assembly outlet, the head piece region?; are zone E, bottom end plug, and zone F, fuel assembly inlet, the foot piece region?).
7.3.2.2 Shielding Analysis	4	<ul style="list-style-type: none"> - Materials specifications and properties in Table 7.3-3 (list B and C compositions separately instead of B₂C; in BISCO NS-3 components, is "d" supposed to be 0?; data on helium in cask cavity?; why is water not in Table 7.3-4 but is in Table 7.3-3?). - Atom densities in Table 7.3-4 (atom density units?; 0 in fuel pellet?; C in BORAL core?; bottom nozzles, bottom end plugs, springs, top end plugs, and top nozzles?; helium in cask cavity?; primary and seal covers?; where does water come from?). - Atom densities in Table 7.3-4 (NRC concerned about stated values; regarding verification, some values are correct, some vary in second and third decimal and others by factors of 2 to 10; uranium-238 value low by about factor of 2; all elements, except Fe, in cell closure are low by about factor of 10; boron-10 and boron-11 in BORAL core could only be verified after computing B content of B₂C). - Units in Table 7.3-6 (mrem/hr per photon/cm², not mrem/hr per γ/cm²-second).
7.3.2.2.1 Shielding Analysis Models	6	<ul style="list-style-type: none"> - Volume fractions and atom densities (in Table 7.3-4 but not Table 7.3-3?). - Volume fractions (how determined?). - Spent fuel array, bottom nozzles, bottom end plugs, springs, top end plugs, top nozzles, and neutron shield shell (what comprises balance?).

TABLE 3.10. (contd)

Section of Topical Report	No. of Issues	Items Involved
		<ul style="list-style-type: none"> - Correcting elemental atom densities for materials mix of various components for homogenized spent fuel region (achieved via volume fractions?; same question applies to fuel assembly structures above and below fuel zone as well). - Source term associated with structure activation (included in analysis?). - Homogenized cylindrical spent fuel region (if used, what was radius and how determined?).
7.3.2.2.2 Shielding Results	5	<ul style="list-style-type: none"> - Figures 7.3-15, 7.3-24, and 7.3-34 (why are fuel assembly outlet, top end plugs, and fuel rod springs not shown?). - Primary and secondary gamma dose rates (statement regarding "isodose contour data..." for neutron dose rates same as for gamma except "top closure" replaced by "overall cask"; what is difference?; why is "top closure" significant?). - Multiplication factor footnote in Table 7.3-8 (why is "cooling fin effect" not indicated?). - Seal cover secondary gamma dose rate (why higher than for shield cover?). - Normalized burnup effects (result of numerous shielding calculations?; result of ratio of numerous source terms?; if based on source terms, were spectral results identical at each burnup increment?).
7.4 Estimated On-Site Collective Dose Assessment		
7.4.1 Analytical Models	2	<ul style="list-style-type: none"> - Type of buildup factors (exposure dose or absorbed dose?); water dose buildup factors (used because alternatives were no buildup or buildup from higher Z materials?). - Neutrons in uncollided dose calculations (how does MAP code treat?; can figure from Reference 4 be presented here?).
7.4.2 Single Cask Dose Rates		
7.4.2.1 Cask Surface Dose Rates and Energy Spectra	1	<ul style="list-style-type: none"> - Peaks in primary gamma dose rate through top and bottom surfaces (why peaks at 80-cm radius?).
7.4.2.2 Dose Rate Versus Distance from a Single Cask	1	<ul style="list-style-type: none"> - Gamma ray dose rates in Figures 7.4-7 through 7.3-9 (represent primary and secondary contributions?).
7.4.3 Multi-Cask Dose Rates		
7.4.3.2 Dose Rates from Casks Stored On Storage Pads	1	<ul style="list-style-type: none"> - Distance D, in Figure 7.4-11 (measured from?; unclear whether it originates from location 1 or location 4; should it not be measured from location 6?).
7.4.4 Radiological Dose Assessment		
7.4.4.1 Occupational Exposures	2	<ul style="list-style-type: none"> - Table 7.4-9 (comments in Section 5.1.2 apply here also; important parameter, individual's location with respect to cask surface, missing; how was dose at each location computed?; was MAP used?).

TABLE 3.10. (contd)

Section of Topical Report	No. of Issues	Items Involved
		<ul style="list-style-type: none"> - Table 7.4-9 (break down into operations associated with loading, transporting, storing, maintaining, and unloading; if individual receives dose of 690 mrem from single cask then four casks could deliver nearly 2.8 rem annually; from quarterly standpoint, no one individual could be involved with no more than two Westinghouse casks; why were such dose issues not addressed?; is assumed background dose rate of 2.5 mrem/hr included in Table 7.4-9?).
8.0 Accident Analysis		
8.1 Off-Normal Operation	1	<ul style="list-style-type: none"> - Leakage of cavity contents (implies possible release of radioactive gases; section contains no assessment of radiological impact as required by Regulatory Guide 3.48).
8.1.1 Component or System Malfunction		
8.1.1.2 Detection of Event	1	<ul style="list-style-type: none"> - Reference to Figure 5.1-2 (does not exist; Figure 5.1-1 is meant).
8.1.1.3 Analysis of Effects and Consequences	1	<ul style="list-style-type: none"> - Leakage of radioactive gases (significant for this event?).
8.1.1.4 Corrective Actions	1	<ul style="list-style-type: none"> - Repair of leaking cask (how long will it take?; provide estimate).
8.1.2.3 Analysis of Effects and Consequences	1	<ul style="list-style-type: none"> - Reference (should be Section 8.1.1.3).
8.1.2.4 Corrective Actions	1	<ul style="list-style-type: none"> - Reference (should be Section 8.1.1.4).
8.2 Accidents		
8.2.1 Loss of Confinement Barrier		
8.2.1.2 Accident Analysis	1	<ul style="list-style-type: none"> - Sources (why only krypton-85 considered?; what about helium-3, cesium-129, and cesium-137?).
8.2.1.3 Accident Dose Calculations	2	<ul style="list-style-type: none"> - Source term data (incomplete; provide more information on krypton-85; substantiate statement that krypton-85 is "the only isotope that needs to be considered"). - Krypton-85 dose factors (refer to skin or whole body?; provide references and sample calculation).
8.2.2 Loss of Neutron Shield		
8.2.2.1 Cause of Accident	1	<ul style="list-style-type: none"> - Loss of neutron shield (implication is that fire is only cause; what about cask drop, missile impact, etc.?; this paragraph is in contradiction of statement in Section 8.2.4.2).
8.2.2.3 Accident Dose Calculations	3	<ul style="list-style-type: none"> - Neutron and gamma dose rates at side of cask (how were increases determined?; dose increases at top and bottom?). - Repairs subsequent to loss of neutron shield (last sentence raises doubt as to whether any repairs occur; only action indicated is movement to "more advantageous location" or placing "supplemental shielding... exterior to cask"). - The 25 mrem limit of 10 CFR 72.67(a) is cited for "normal operations and anticipated occurrences" (for accidents, limit from 10 CFR 68(a) is 5 mrem).

TABLE 3.10. (contd)

Section of Topical Report	No. of Issues	Items Involved
8.2.3 Cask Tip Over		
8.2.3.1 Cause of Accident	1	- Cask tip-over due to tornado wind velocity (section suggests this is unlikely; however, Section 8.2.7.4 shows wind pressure sufficient to topple cask; reconcile difference; clearly define "not likely").
8.2.3.3 Accident Dose Calculations	1	- Seals (loads on them?; any additional leakage?); neutron shield (affected by cask tip over?; if so, what is dose consequence?).
8.2.4 Fire	1	- Curves in Figure 8.2.4-1 (illegible captions).
8.2.4.3 Accident Dose Calculations	1	- Reference to Section 8.2.1 (connection is not clear; dose due to fire?).
8.2.5 Flood		
8.2.5.3 Accident Dose Calculations	1	- Flood circumstances (can entire array of casks lose their shielding?; reference to Section 8.2.2 for dose calculation but this is for single cask only).
8.2.5.4 Limiting External Pressure	1	- Equations on pages 8.2-16 and 8.2-17 (provide references).
8.2.5.5 Toppling Velocity	1	- Statement that "effect of toppling is conservatively bounded by five-foot drop accident" (see comments in Section 8.2.3.2).
8.2.6 Cask Drop		
8.2.6.2 Accident Analysis		
8.2.6.2.1 Horizontal Drop	4	- Localized loss or compression of neutron shield (is there any?). - WE CAN code (what is it?; has it been benchmarked?). - All materials (provide specifics of constitutive material models; list primary references or refer to other sections of report where information presented; for purely elastic models such as steel, list only elastic components). - Basket structural integrity (maintained for a horizontal drop?; provide analysis and results).
8.2.6.2.2 Horizontal Drop on Top Corner	3	- Rotation drop (ability of fuel assemblies to withstand it?; possibility of radioactive gas release to cask cavity if fuel assemblies damaged?). - Localized loss or compression of neutron shield (is there any?). - Free body diagrams (provide; which way is cask moving?).
8.2.6.3 Accident Dose Calculations	2	- Neutron absorber [is there an additional local loss (crush) of it in Section 8.2.6.2.4?; if so, dose consequence calculation required]. - Figures 8.2.6-2 through 8.2.6-7 (need more explanation; analysis belonging to each figure?; how was impact load determined?); stress output (coordinate system used?; how were g's calculated?).

TABLE 3.10. (contd)

Section of Topical Report	No. of Issues	Items Involved
8.2.7 Tornado Wind and Missile		
8.2.7.3 Accident Dose Calculations	1	- Analysis showing cask tips over under worst tornado wind storm conditions (provide dose analysis for entire cask array tipping simultaneously; see comments in Section 8.2.5.3).
8.2.7.5 Tornado Missiles	1	- Equation Plin (reference?).
8.2.8 Explosion		
8.2.8.2 Accident Analysis	1	- Gas pipe explosion (possible also to ignite gas, resulting in pressures over 1 psi?).
8.2.9 Earthquake		
8.2.9.2 Accident Analysis	1	- Statement that 0.5 g horizontal acceleration is at least as severe as any maximum base acceleration in the contiguous United States (not possible to generically state that 0.5 g base acceleration is governing West of the Rockies; see requirement in 10 CFR 72, paragraph 66).
8.2.10 Lightning	1	- Current surge effect on cask sealing integrity and instrumentation (no evaluation included).
8.2.11 Roll Over	1	- Accident dose calculations (need to address; include in a Section 8.2.11.3).
8.2.12 Burial Under Debris		
8.2.12.3 Accident Analysis	1	- Figure 8.2.12-1 (captions illegible).
Appendix 12.1-2	1	- Containment related welds for Category I cask (should conform to requirements of Section III of ASME B&PV ^(a) code as described in NUREG/CR-3019).
Quality Assurance (QA) Program	4	<ul style="list-style-type: none"> - Involvement of QA program with every facet of manufacturing cycle (clarify this to a commitment or justify not doing so). - Term "safety related" in QA program (appears to meet definition of "important to safety" in 10 CFR Part 72; if Westinghouse agrees, so document in Section II of topical safety analysis report or in QA program described in WCAP-8370/7800). - MC-10 cask items that are important to safety (identify). - Section 1.5 of topical safety analysis report says no other topical reports are referenced (Section II references topical report WCAP-8370/7800; statements appear contradictory; clarify).
Some Additional Information Needed	2	<ul style="list-style-type: none"> - Free body diagrams for cask designs (need to provide). - Existing drawings (too reduced in size to be legible in some cases; replace).

(a) Boiler and Pressure Vessel (B&PV).

TABLE 3.10. (contd)

<u>Section of Topical Report</u>	<u>No. of Issues</u>	<u>Items Involved</u>
Fuel Basket Material	1	- Design of cask fuel basket and materials used (unresolved issue and of major concern to NRC; specifically NRC concerned that aluminum basket is not an ASME Code Class 1 material; NRC/LLNL ^(a) recommended that basket material be changed to ASME Code Class 1 material or further evaluation be performed on acceptability of aluminum basket).

^(a) Lawrence Livermore National Laboratory (LLNL).

TABLE 3.11. Examples of Some Issues^(a) Involved in Project M-42 (TN-24P)

<u>Section of Topical Report</u>	<u>No. of Issues</u>	<u>Items Involved</u>
1.0 Introduction and General Description		
1.1 Introduction		
1.1.2 Principal Design Features of Installation	1	- Drawings (poor quality; makes it difficult to understand principal design features; refer to full-sized drawings of components important to safety).
1.2 General Description of Installation		
1.2.2 Principal Design Criteria	2	- Design criteria (why was Subsection NC, article NC-3200 of Section III of ASME B&PV ^(b) Code chosen instead of Subsection NB?). - Nitrogen as cavity's primary fill gas (potential for corrosion if water present?; how do you assure that essentially all water removed from cask cavity?).
1.2.5 Structural Features	6	- Five reference drawings in Appendix 1A (do not provide enough dimensional detail). - Table 1.2-3 (basket plate material excluded; identify material in table). - Sprayed metallic coating on cavity surfaces and outer shell (what is the coating(s); briefly describe coating process). - Poison rods in fuel assembly (how installed?; how held in place during handling, storage, and accidents?; provide poison rod assembly drawing). - Copper on fuel basket (cladding or coating?; how applied?; what process used?; provide assurance that copper remains in place at $\geq 600^{\circ}\text{F}$ and for up to 20 years). - Copper (thickness used?; inconsistent thickness values used in Drawing 1A-9, Table 3.3-1, and Section 5.1.3.6 in comparison with Figure 5.1-7; validate that copper will conduct heat at 425 Btu/hr-ft- $^{\circ}\text{F}$ at temperatures associated with normal and accident conditions).
1.3 General Systems Description	1	- Cask is passive system (be more specific regarding minimal monitoring and maintenance activities over 20-year cask design lifetime).
2.0 Site Characteristics		
2.1 Geography and Demography of Site Selected	1	- "the minimum dose rates specified in 10 CFR Part 72 can be met with a cask-to-boundary distance well below 328 ft" (applies to how many casks?; from Table 7.3-3, appears it would be 21 or less).
2.3 Meteorology	1	- Tornado missile loading (needs to be addressed per NUREG-0800, Section 3.5.1.4).
3.0 Principal Design Criteria		
3.1 Purposes of Installation		

(a) Over 160 additional issues are in References 95 and 96. Responses are in References 98 and 202-204.

(b) Boiler and Pressure Vessel (B&PV).

TABLE 3.11. (contd)

Section of Topical Report	No. of Issues	Items Involved
3.1.1 Materials to be Stored	6	<ul style="list-style-type: none"> - ORIGEN calculations (irradiation conditions used?); radiological characteristics of spent fuel (demonstrate that 3.2% enrichment more conservative than 3.7% enrichment); 15x15 assembly (demonstrate it is conservative for analysis). - Thermal, gamma, and neutron sources for design basis 17x17 PWR fuel assembly (important that spectral conditions be same or distribution be shifted to lower energies; gamma and neutron sources affect shielding analysis; isotope content must also be considered in safety evaluation). - Gamma and neutron sources (why no spectral information provided?; include or refer to Section 7.2.1). - Fission product gas inventory (why no mention in this section?; include or refer to Section 7.2.2). - Materials in head and foot pieces and structural materials (what are activation product activities?; why not addressed here?; provide spectral and source information; refer to Section 7.2.1). - Criticality design for 17x17 assembly (depends on use of poison rods; results do not directly apply to 14x14 and 15x15 assemblies to be included in supporting criticality analyses; include associated poison rod information).
3.0 Structural and Mechanical Safety Criteria		
3.2.1 Tornado and Wind Loadings	3	<ul style="list-style-type: none"> - Effect of tornado missiles (need to provide analysis as specified in NUREG-0808, Section 3.5.1.4). - Drag coefficient of 1 in tornado loads analysis (provide supporting reference and/or equations). - Tornado wind load (how is calculated 332 psf converted to linear load of 204 lb/in.?).

**TABLE 3.12. Examples of Some Issues^(a) Involved in Project M-43
(C-E Dry-Cap)**

Section of Topical Report	No. of Issues	Items Involved
General	1	- References to 10 CFR Part 71 requirements (NRC is not reviewing the topical report under Part 71, but under Part 72).
1.0 Introduction and General Description of Installation		
1.1.2 General Description of the Installation	1	- Ventilated roofed enclosure for multiple cask storage (no supporting calculations to justify adequacy of this arrangement).
1.1.5 Description of the Spent Fuel to be Stored	2	- Minimum burnup for fuel (specify; also specify operating history and cooling times). - Thermal outputs for BWR and PWR fuel (appear to low for 40,000 Mwd/MTU and 5-year cooling time; provide ORIGEN outputs for decay heat, gamma and neutron spectrum, and rapid nuclide activities; use those outputs for thermal, shielding, and containment analyses).
1.2 General Description of Installation	2	- Criteria for subcriticality of K_{eff} <0.95 (should also reflect use of fresh fuel and all biases). - Cask penetration (describe).
1.2.2 Principal Design Criteria		
1.2.2.2 Design Criteria for the Cask	12	- Maximum design ambient temperature of 38°C (100°F) for cask (too low for credible upper bound applicable to entire U.S.A.). - Cask seals maximum leak rate of 10^{-8} cc/sec (for each of two seals or two seals in combination?; if latter, what are individual seal leak rates?; if for each seal, what is leak rate for combination?). - Verifying maximum leak rate of 10^{-8} cc/sec under field conditions (explain how difficulties can be overcome). - Radiation dose criteria (10 CFR Part 71.47 cited; why no references to 10 CFR Parts 72.67(a) and 72.68(b) for limits for normal and accident conditions?). - Radiation dose levels for nuclear operations (levels appear high, considering ALARA). - Limits of 380°C and 570°C for fuel cladding for normal and accident conditions, respectively (basis?; experience data bases to support 380°C are, as yet, insufficient); acceptability of stated temperature limit (provide temperature decay curve for both BWR and PWR fuel for at least 20-year storage period and maximum anticipated fuel rod internal pressure). - Cladding subjected to 570°C (how long before it ruptures?). - Expression "shall maintain integrity" on page 1.8 (clarify; does this refer to specific limits on stress and deformation, to some level of physical damage, or to some level of radioactive release?). - Cask and its internal structures (Subsection NB of Section III of ASME B&PV ^(b) Code universally applied; why not Subsection NG for the fuel basket?).

(a) These issues and 230 additional issues are in References 99 and 100. Responses are in Reference 205.

(b) Boiler and Pressure Vessel (B&PV).

TABLE 3.12. (contd)

<u>Section of Topical Report</u>	<u>No. of Issues</u>	<u>Items Involved</u>
		<ul style="list-style-type: none"> - Section 1 (discuss approach for establishing stress limits for normal and accident conditions; e.g., what ASME Code Service Levels used?). - Qualifying storage cask for 30-ft drop (is intent to use the impact limiters in Figure 1.2-5?; if so, do impact limiters remain in place?). - Limiting storage cask to height of one foot above pad (what provisions made if impact limiters not installed?).

TABLE 3.13. Examples of Some Issues^(a) Involved in Project M-46 (FWEA)

Section of Topical Report	No. of Issues	Items Involved
General	1	- All drawings in topical report (provide copies that are large enough to read).
1.0	1	- Design limits and their bases, e.g., maximum bulk and local concrete temperatures, peak fuel cladding temperature (need to state).
1.0	1	- Interface between material for which final one-time approval is sought and site-specific elements (include MVDS ^(b) drawings).
1.1.1	1	- General description (state whether MVDS design can support an unlimited number of vault module extensions, or state maximum number of extensions that can be supported by the MVDS design in this submittal).
1.1.1	1	- General description (identify, as by inclusion of a table, those documents that provide further detail, analysis, or otherwise support the design and that are considered as part of the report for purposes of the review).
Figures 1.1, 1.2, 1.3.1-1 through -5, and 4.2.2-1 through -8	1	- Access and egress provisions (NRC questions whether design as documented in these figures provides adequate access and egress; there are apparent accessibility deficiencies in personnel access to the charge face level of the charge hall, in the "emergency exit from the charge hall," in the limited access for equipment to the charge hall, and in provisions for personnel or automated equipment access to the intake or out-flow plenums for the vault module).
Figure 1.1-1	1	- Concrete design (three drawings here and in A8 do not show shear reinforcement for concrete in several potentially high shear areas).
Figure 1.1-2	1	- Concrete slab bridge between vault module at charge hall floor (purpose?; appears to unnecessarily load concrete vault shielding wall).
Figure 1.1-3 and Figure A.8.1	1	- Roof truss (two different designs shown; which is correct?).
Figure 1.3.1-3	1	- Foundation design under fan room (are provisions for shear and bending forces adequate?; design appears to unnecessarily increase probability of failure of slab below fan room).
1.3.3.6 and Figure 1.3.3-1	1	- Gas service system design, specifically design of service point (should there be valve setting for "off"; appears it would be desirable).

(a) The issues shown are from Reference 104; there are approximately 104 more issues listed in that reference.

(b) Modular Vault Dry Store (MVDS).

TABLE 3.13. (contd)

<u>Section of Topical Report</u>	<u>No. of Issues</u>	<u>Items Involved</u>
1.3.3.3 and Figure 1.3.3-1	4	- Shielded storage tube (SST) design (if necessary, can entire SST be readily removed without human access to vault area?; what must be done if an SST is to be removed intact?; what is necessary to seal SST with shield plug in place against gas leakage from within SST if SST is removed intact?; can the fuel handling machine, FHM, remove the SST?).
Page 1.3-7 and Figure 1.3.3-4	1	- SST material (provide specification; stating "carbon steel" is insufficient).

**TABLE 3.14. Examples of Some Issues^(a) Involved in Project M-51
(NAC cask for consolidated fuel)**

Section of Topical Report	No. of Issues	Items Involved
General	1	- Information on blue sheets (will all of it be entered into topical report?; some sheet entries appear to be comments, others revisions).
1.0 Introduction and General Description of Installation		
1.1.3 Type, Form, Quantity, and Potential Sources of Spent Fuel: Design Basis Fuel	2	- Design basis on page C1.1-3 (confusing; basis for derating cask load from 28.3 kW to 20 kW?). - Design heat load (clarify on page C1.1-4; for NAC-C28 S/T, is it 20 kW or 28.3 kW?; relationship to a possible tipover condition?).
1.2.1 Principal Design Criteria	1	- Fuel rod cladding stress versus maximum cladding temperature (to perform a more detailed confirmatory analysis, the topical report should include a temperature decay curve for the hottest fuel rod over at least a 20-year storage period).
1.2.5 Passive Heat Dissipation	1	- Heat transfer from fuel rods to cask body (description excludes effect of canister on heat transfer capabilities; why?).
3.0 Principal Design Criteria		
3.1 Purpose of Installation		
3.1.1 Materials to be Stored	1	- Stainless steel gas plenum springs (why was cobalt activation not considered?).
3.3.4.1 Control Methods for Prevention of Criticality	1	- Primary control method involves "administrative controls prescribing a minimum number of consolidated fuel rods" (actual number not listed here but is in Section 10; state here as it is important).
3.5 Decommissioning Considerations		
3.5.1 Storage Casks	1	- Cask and basket activation product calculations (materials and weights used?).
4.0 Installation Design		
4.2 Storage Structures		
4.2.1 Structural Specifications		
4.2.1.1 Cask Description	2	- Cask body detail in Figure 4.2-2 (describe cellular silicone and silicone foam; need more detail on materials and clearer understanding of dimensions). - Neutron shield lid for consolidated fuel cask in Figure 4.2-9 (why doesn't it show increased 304 stainless steel plate thickness?).
4.2.1.3 Properties of Materials	2	- Mean value for stainless steel's coefficient of thermal expansion (how was "mean value" determined?). - Impact limiter in Figure 4.2-7 (material for honeycomb illegible; crush strength 6200 or 2200?).

(a) These issues and over 80 additional issues are in References 109, 111, and 112. Responses are in References 111-113 and 206-208.

**TABLE 3.15. Examples of Some Issues^(a) Involved in Project M-49
(NUTECH NUHOMS-24P)**

Section of Topical Report	No. of Issues	Items Involved
1. (Question 1, page 1.2-1)	1	- DSC ^(b) and HSM ^(c) sizes shown in Table 1.2-1 (are all intended for approval by the NRC or is this topical report restricted to 24 PWR spent fuel assemblies?).
1. (Question 2, page 1.2-1)	1	- Statement that "any number of HSMs can be interconnected" (what is NUTECH implying?; NUTECH seeking approval for how many modules?).
1. (Question 3, page 1.2-9)	1	- Lowering of water in DSC prior to welding (level of annulus water also lowered?; is there a detailed procedure for the drying operation?).
1. (Question 4, pages 1.2-9 and 1.3-5)	1	- System steps (are steps 7 and 8 reversed?; seems difficult to perform seal weld with water up to top of DSC; procedure appears inconsistent with one outlined on page 4.7-2 and -3).
1. (Question 5, page 1.2-9)	1	- Table 1.2-3 and Figure 5.1-3 [there are significant variations between them (e.g., before draining DSC, weld after draining, sealing annulus, etc.); which is correct?].
1. (Question 6, page 1.3-3)	1	- Two sentences referring to research (can sources be referenced?; if work of Guenther of PNL cited, state that it involved tests of unirradiated fuel).
1. (Question 7, page 1.3-4)	1	- "as low as possible" (how low?; specifically, what is trailer bed height?; how high will cask need to be lifted to place it on bed?; dimensions of cask support skid?).
1. (Question 8, page 1.3-6)	1	- "Inner DSC sealing" (description inconsistent with Figure 5.1-3; 6 in. here varies from 4 in. in paragraph 5.1.1.3).
1. (Question 9, page 1.3-13)	1	- Figure on this page and other figures in report (two 90-degree bends in outlet air passage shown but drawings show no bends; which are correct?).
1. (Question 10, page 1.3-14)	1	- Bottom cover plate (appears to have square shield; does this provide a gamma or neutron leakage source?).
1. (Question 11, page 1.3-14 (Figure 1.3-2a) and Appendix E (Drawing DUK-03-2002))	1	- Expansion tank (shown in figure but not in Transfer Cask drawings; no calculations provided to demonstrate tank volume is adequate).
3. (Question 1, page 3.1-1)	1	- Continued reference to storage of 32 PWR fuel assemblies (confusing; does NUTECH intend to supply all design information for storing 32 as well as 24 PWR fuel assemblies?).
3. (Question 2, page 3.1-2)	1	- Administrative procedure to ensure that neutron and gamma ray source terms will be less than design basis (should description be included in this topical report or is this to be deferred for review in the specific license application?).

(a) The issues shown are from Reference 209; there are approximately 166 more issues listed in that reference.

(b) Dry Shielded Canister (DSC).

(c) Horizontal Storage Module (HSM).

TABLE 3.15. (contd)

Section of Topical Report	No. of Issues	Items Involved
3. (Question 3, page 3.1-2)	1	- Neutron source (burnup is substantially higher than 7-assembly design (40 versus 33 Gwd/tonne), so higher and hotter neutron source would be anticipated, also cooling period extended to a minimum of 10 years; are neutron and gamma ray source spectra and strength consistent?).

TABLE 3.16. Examples of Some Issues^(a) Involved in Project M-44 (NuPac)

Section of Topical Report	No. of Issues	Items Involved
1.0 Introduction and General Description of Storage System		
1.2.2	2	<ul style="list-style-type: none"> - Pad on which casks are to be placed (discuss; if site specific, so state but at least describe minimal design envelope for pad). - Dry load procedures and handling equipment (not addressed in Revision 0 of topical report).
1.4	1	<ul style="list-style-type: none"> - Figure 1.4-1 (said to show 50-cask array but 68 casks depicted; explain).
2.0 Principal Design Criteria		
2.1	9	<ul style="list-style-type: none"> - Ambient temperature in Table 2.2-1 (incorporate an accident design criterion with maximum ambient temperature of 125°F; discuss consequences of ambient temperature on fuel cladding and on long-term durability of concrete; discuss consequences in Section 11.
2.2 (page 2-3)	1	<ul style="list-style-type: none"> - Rainfall rate (not included; discuss potential for concrete spalling due thermal gradients that might be produced).
2.2.2 (page 2-5)	1	<ul style="list-style-type: none"> - Thermal gradients, a potential source of concrete spalling, produced by flooding (not discussed).
2.2.5	1	<ul style="list-style-type: none"> - Combined load criteria (this section does not correspond with ANSI/ANS-57.9-1984, Sections 6.17.3.1 and 6.17.3.2.1 for concrete and steel structures; NRC's basic concern is that thermal loads are not adequately considered in off-normal and accident conditions; thermal loads should be combined with all off-normal and accident cases, as shown in ANSI/ANS-57.9-1984, which will not require two accident cases to be postulated to occur simultaneously since thermal condition is not an accident condition, but rather is a constant condition).
2.2.5 (page 2-9)	1	<ul style="list-style-type: none"> - Accident loads (discussion excludes a drop with rotation resulting in impact on cylinder upper rim; discuss rationale for excluding this case; aircraft crash impact or fire has been left to a site-specific application).
2.3.3.1	1	<ul style="list-style-type: none"> - Discussion of equipment (not satisfactory); crane, transport vehicle or trailer, lift fixture and support pad (may be site specific but need to describe in enough detail to demonstrate feasibility; define minimum design envelope).
2.3.6 (page 2-13)	1	<ul style="list-style-type: none"> - Events [fire or explosive events are credible per Regulatory Guide 3.48, paragraph 2.2 entry 7 (aircraft impact); can be addressed in site-specific license application; NuPac should modify their statement accordingly].

(a) These issues and over 100 additional issues are in References 119 and 210. Responses are in Reference 120.

TABLE 3.16. (contd)

Section of Topical Report	No. of Issues	Items Involved
2.4 (page 2-14)	1	- Decontaminable coatings used so cask exterior surface can be cleaned "to meet site-specific standards" [define bounds of "site-specific standards" that proposed coatings can meet; what standards (e.g., for alpha and beta contamination limits for offsite shipping containers) are used for design basis for acceptable level of decontamination?].
2.4	1	- Decontaminable coating (identify; provide evidence of suitability of material for environment in question).
3.0 Structural Evaluation		
3.1.1 (page 3-1)	1	- Inner concrete surfaces that are against liner (are these considered "exposed" due to high temperatures to which they are subjected?; should they receive minimum reinforcement in accordance with Section 7.12, ACI 349-80 or 349-85?).
3.1.2 (page 3-2)	1	- Live loads due to extreme thermal gradient produced by heavy, cold rain on a high temperature day (e.g., summer thunderstorm) [not estimated; significant potential for stresses for concrete spalling may result; this would be a "live load" in combination of load expressions, per ANSI/ANS-57.9-1984 (neither an "other-than-normal" nor an "accident situation")].

TABLE 3.21. Examples of Some Issues Involved^(a) in the Environmental Report for the H. B. Robinson-2 Independent Spent Fuel Storage Installation (Docket No. 72-3).

1. Current vegetation cover and use(s) of areas to be disturbed for construction of the ISFSI^(b) (what are they?; was this land disturbed during initial station construction?).
2. Spoil material created during construction (identify quantity); excavated or other spoil material (identify locations/areas required for stockpiling).
3. Erosion of soils during construction (quantities and effluent concentrations expected?; will runoff drain into Lake Robinson?).
4. Areas disturbed during construction (identify reclamation plans; provide information on revegetation species).
5. New access roads required for construction (identify).
6. Species inhabiting site or immediate site vicinity (i.e., within five miles) (identify state or federally threatened and endangered species).
7. Erosion control during construction (what measures will be implemented?).
8. Construction work force (size?); duration of construction (anticipated length?); work force size (contrast work force size with that of activities during typical maintenance outages).
9. Noise levels at nearest site boundary (provide estimate of anticipated levels during construction of the spent fuel storage facility).
10. Impacts of construction work force on local housing and school systems (if appropriate, provide an evaluation of impacts).
11. Proposed action and associated impacts (identify realistic alternatives to the proposed action; describe potential impacts associated with each).
12. Permits and approvals (what other permits and approvals will be needed for the proposed project?).
13. Location of additional modules (where will the five additional modules be located in relation to the initial three modules?).
14. ISFSI controlled area (in accordance with the definition in 10 CFR Part 72, what is the H. B. Robinson-2 ISFSI controlled area?).
15. Occupational dose commitment (how does the estimated annual collective dose commitment from ISFSI operations compare to that from present operations of H. B. Robinson-2?).
16. Offsite dose commitment (how does the estimated offsite collective dose commitment from ISFSI operations compare to that from reactor operations?).
17. Radiation monitoring (in addition to the existing H.B. Robinson-2 radiation monitoring program, what radiation monitoring will be conducted for the ISFSI?).
18. Assumptions (what are the assumptions used in defining the maximum exposed member of the public and what is the expected annual dose to that individual from the multiple contributions of the ISFSI and H. B. Robinson-2?).

(a) The NRC indicated that these questions⁽³⁰⁾ were intended only as a guide to help the utility in preparing the Environmental Report.

(b) Independent Spent Fuel Storage Installation (ISFSI).

TABLE 3.22. Examples of Some Issues^(a) Involved in the Safety Analysis Report of the H. B. Robinson-2 Independent Spent Fuel Storage Installation (Docket No. 72-3)

Section of SAR ^(a)	Question Number	Issue(s) Involved
1.2.1 General Description	1	- Drawings, other than conceptual drawings, for items associated with the ISFSI ^(b) [no drawings provided for any items; provide engineering (shop) drawings for DSC, ^(c) HSM, ^(d) support pad for HSM, modifications to the IF-300 shipping cask, etc.; include all drawings for all components as they will be used at the H. B. Robinson-2 site].
1.2.3 Principal Design Criteria	2	- Statement that "for the <u>majority</u> of the fuel to be stored the radiation sources will be less than or equal to the source described in the NUKOMS Topical Report" (implies that some fuel may exceed the design criteria; maximum radiation that would be encountered?; how does it modify calculated doses?).
1.2.3.1 Structural Features	3	- SAR ^(e) states that "the structural features of the DSC ^(c) are defined, to a large extent, by the cask drop accident" and that the 5-foot drop postulated by NUTECH "envelopes any postulated drop accident that could occur at the Robinson site" [several situations (e.g., cask loading, cask/canister/assembly loading and skid loading) are shown in which cask is more than five feet from ground; why are drops in these situations not credible?; consider vertical and horizontal cask orientations].
1.2.4 Operating and Fuel Handling Systems	4	- Cask movement, cask head and truck transport (why were values for these excluded from Table 1.2-3?).
1.2.5 Safety Features	5	- HSM ^(d) designed so <u>average</u> surface dose rate is less than 20 mrem/h (is average surface dose rate an appropriate design parameter?; what is maximum surface dose rate?).
1.2.5 Safety Features	6	- DSC and cask are filled with demineralized water to reduce contamination of cask exterior (effectiveness of this process in reducing DSC contamination?; what are expected surface dose levels?).
1.2.5 Safety Features	7	- Contamination of cask from fuel pool water (surface dose rate on cask?).
1.3.1.1 Canister Design	8	- Length of DSC for specific type of irradiated fuel to be stored (specify; "approximately 180 inches" inadequate for site-specific application).

(a) The issues shown are from References 211-214; there are approximately 91 more issues listed in References 8, 10, 137, 142, and 211-230.

(b) Independent Spent Fuel Storage Installation (ISFSI).

(c) Dry Shielded Canister (DSC).

(d) Horizontal Storage Module (HSM).

(e) Safety Analysis Report (SAR).

TABLE 3.23. Distribution of Questions/Requests Associated with the Safety Analysis Report for the Independent Spent Fuel Storage Installation at Robinson-2

Safety Analysis Report		Title ^(a)	Questions/Requests		
Section	Subsection		No.	Subtotal	Total
1.0		INTRODUCTION AND GENERAL DESCRIPTION OF INSTALLATION			20
	1.2	General Description of Installation		7	
	1.2.1	General Description	1		
	1.2.3	Principal Design Criteria:	1		
		--Structural Features	1		
	1.2.4	Operating and Fuel Handling Systems	1		
	1.2.5	Safety Features	3		
	1.3	General Systems Descriptions		12	
	1.3.1	Systems Descriptions:			
		--Canister Design	1		
		--Horizontal Storage Module	3		
		--Transfer Cask	2		
		--System Operation	6		
	1.5	Material Incorporated By Reference	1	1	
2.0		SITE CHARACTERISTICS			9
	2.1	Geography and Demography:		4	
	2.1.1	Site Location	1		
	2.1.2	Site Description:			
		--Other Activities Within the Site Boundary	2		
	2.1.3	Population Description and Trends	1		
	2.2	Nearby Industrial, Transportation, and Military Facilities		2	
	2.2.1	Locations and Routes	1		
	2.2.2	Descriptions	1		
	2.3	Meteorology		1	
	2.3.2	Local Meteorology	1		
	2.6	Geology and Seismology		2	
	2.6.2	Vibratory Ground Motion:			
		--Earthquake History	1		
	2.6.4	Stability of Subsurface Materials:			
		--ISFSI Foundation	1		
3.0		PRINCIPAL DESIGN CRITERIA	3	3	14
	3.1	Purpose of the Installation		6	

TABLE 3.23. (contd)

Safety Analysis Report		Title(a)	Questions/Requests		
Section	Subsection		No.	Subtotal	Total
3.132	3.1.1	Material To Be Stored:			
		--Radiological Characteristics	1		
	3.1.2	General Operating Functions:			
		--Overall Functions of the Facility	2		
		--Handling and Transfer Equipment	3		
	3.2	Structural and Mechanical Safety Criteria		2	
	3.2.5	Seismic Design:			
		--Input Criteria	1		
		--Seismic-System Analysis	1		
	3.3	Safety Protection System		3	
	3.3.1	General	1		
	3.3.3	Protection By Equipment and Instrumentation Selection:			
		--Instrumentation	1		
	3.3.5	Radiological Protection:			
		--Radiological Alarm System	1		
	4.0	INSTALLATION DESIGN	2	2	8
	4.1	Summary Description		2	
	4.1.2	Principle Features:	1		
		--Controlled Area	1		
	4.2	Storage Structures		2	
	4.2.1	Structural Specifications:			
		--Construction, Fabrication, and Inspection	1		
	4.2.3	Individual Unit Description	1		
	4.6	Cathodic Protection	1	1	
	4.7	Fuel Handling Operation System		1	
	4.7.1	Structural Specifications	1		
	5.0	OPERATIONS SYSTEMS	1	1	18
	5.1	Operation Description		12	
	5.1.1	Narrative Description:			
		--Preparation of the Transfer Cask and Canister	3		
		--Cask Drying Process	2		
		--DSC Sealing Operation	1		
		--Loading Canister into the HSM	2		

TABLE 3.23. (contd)

Safety Analysis Report		Title (a)	Questions/Requests		
Section	Subsection		No.	Subtotal	Total
3.133		--Monitoring Operations	1		
		--Unloading the DSC from the HSM	1		
	5.1.3	Identification of Subjects for Safety Analysis:			
		--Criticality Prevention	1		
		--Instrumentation	1		
	5.2	Fuel Handling Systems		3	
	5.2.1	Spent Fuel Handling and Transfer:			
		--Functional Design	2		
		--Safety Features	1		
	5.3	Other Operating Systems		1	
	5.3.2	Components/Equipment Spares	1		
	5.4	Operation Support System	1	1	
	6.0	WASTE CONFINEMENT AND MANAGEMENT			1
	6.5	Radiological Impact of Normal Operations - Summary	1	1	
	7.0	RADIATION PROTECTION			10
	7.1	Ensuring That Occupational Radiation Exposures Are ALARA		2	
	7.1.1	Policy Considerations	1		
	7.1.2	Design Considerations	1		
	7.3	Radiation Protection Design Features		2	
	7.3.2	Shielding:			
		--Shielding Analysis	2		
	7.4	Estimated Onsite Collective Dose Assessment		5	
	7.4.1	Operational Dose Assessment	3		
	7.4.2	Storage Term Dose Assessment	2		
	7.6	Estimated Offsite Collective Dose Assessment		1	
	7.6.2	Analysis of Multiple Contribution	1		
	8.0	ANALYSIS OF DESIGN EVENTS			12
	8.1	Normal and Off-Normal Operations		4	
	8.1.1	Normal Operation Analysis	1		
	8.1.2	Off-Normal Operation Analysis:			
		--Transport	3		
	8.2	Accident Analysis		6	

TABLE 3.23. (contd)

Safety Analysis Report		Title (a)	Questions/Requests		
Section	Subsection		No.	Subtotal	Total
	8.2.2	Tornado/Tornado Generated Missile	1		
	8.2.3	Earthquake:			
		--Accident Analysis	2		
	8.2.4	Drop Accident	2		
	8.2.5	Lightning:			
		--Analysis of Effects and Consequences	1		
	8.3	Foundation Design	1	1	
10.0		OPERATING CONTROLS AND LIMITS	1	1	8
	10.1	Fuel Specifications	4	4	
	10.3	Limits for the Maximum Air Temperature Rise After Storage	1	1	
	10.4	Surveillance of the HSM Air Inlets			
11.0		QUALITY ASSURANCE	1	1	1

(a) Abbreviations: Independent Spent Fuel Storage Installation (ISFSI), Dry Shielded Canister (DSC), and Horizontal Storage Module (HSM).

TABLE 3.24. Examples of Some Issues^(a) Involved in the Environmental Report for the Oconee Independent Spent Fuel Storage Installation (Docket No. 72-4)

1. Census data for 1970 (Table 2.2-1 in ER^(b) and Table II-1 in the FEIS^(c)) are both based on these data, but a comparison of the tables reveals marked discrepancies in number of persons residing in sectors around site; clarify).
2. Projected number of residents within 50-mile radius of Oconee (Table 2.1-1 in ER indicates no residences within one mile of site but USGS data show some 17 structures, at least some of which could be residences; resolve apparent discrepancies; provide up-to-date information for ER Tables 2.2-1 and 2.1-2).
3. Population statistics (include transient population associated with nearby universities; confirm that these people are represented or correct data).
4. Location of nearest residence (identify, based on more recent population data; nearest residence in the future?; distance from HSM^(d) center to nearest residence?).
5. ISFSI^(e) location (ER Figure 2.1-3 lacks scale; ISFSI occupies a little over one-half acre, so why are 304 acres required for construction of the ISFSI?); ISFSI length (will 88 modules fit?); short and long routes from cask handling area to ISFSI (why does it take an hour to traverse the longer route, which is less than one-half mile?).
6. Distance from ISFSI to boundary of uncontrolled area (to evaluate compliance with 10 CFR 72.68, identify direction and minimum distance).
7. Species on site (any on State or Federal lists of threatened or endangered species?).
8. Part of proposed storage area currently used for parking (location of new parking area?; assess impacts associated with its construction).

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- (a) The issues shown are from Reference 231; there are approximately 26 more issues listed in that reference.
 (b) Environmental Report (ER).
 (c) Facility Environmental Impact Statement (FEIS).
 (d) Horizontal Storage Module.
 (e) Independent Spent Fuel Storage Installation (ISFSI).

TABLE 3.25. Examples of Some Issues^(a) Involved in the Safety Analysis Report for the Oconee Independent Spent Fuel Storage Installation (Docket No. 72-4)

Section of SAR ^(b)	Issue(s) Involved
1. (Question 1, page 1.1-2, para. 1.1)	- Reuse of the DSC ^(c) for second charge of fuel rods (qualify last sentence; such reuse is not addressed in NUTECH's topical report or in this SAR).
1. (Question 2, page 1.2-2, para. 1.2.3)	- Radiation sources to be stored that will exceed those in reference topical report NUHOMS-24P (not described; not mentioned in criticality discussion or shielding analysis; clarify).
1. (Question 3, Table 1.2-2)	- Performance of step 7 before step 8 (contrary to NUTECH's topical report; invalidates use by reference of corresponding radiation shielding operations; also contrary to para. 3.1.2.1 and para. 7.1.2; need to address this inconsistency).
1. (Question 4, Table 1.2-2)	- Step 2 omits sealing the DSC/Transfer Casks annulus (contrary to SAR paras. 3.3.2.1 and 7.1.2 and to NUTECH's topical report Table 1.2-3; need to address this inconsistency).
1. [Question 5, para 1.2-5(b)]	- Internal shield blocks in the HSM ^(d) to reduce scatter dose out of air inlet (are these the shield blocks that comprise the shielded ventilation plenum?).
1. [Question 6, para. 1.3.1.7(b)]	- Transfer cask and DSC inside the transfer cask are filled with water (topical report says "filled with demineralized water"; other information, including Figure 5.5-1, indicates the DSC will be filled with borated water; be specific; if borated water used, what is boron concentration?).

- (a) The issues shown are from Reference 145; there are approximately 143 more issues listed in that reference.
 (b) Safety Analysis Report (SAR).
 (c) Dry Storage Canister (DSC) in the SAR same as Dry Shielded Canister (DSC) in the NUTECH topical report.
 (d) Horizontal Storage Module (HSM).

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334. Stewart, W. L., (Virginia Electric and Power Company). October 17, 1986. Letter to J. N. Grace (NRC), "Virginia Electric and Power Company, Surry Independent Spent Fuel Storage Installation, Confirmation of Temporary Technical Specification Relief." Docket Nos. 50-280 and 72-2. (Accession No. 8611130025-861017-Y)
335. Rouse, L. C., (NRC). November 10, 1986. Letter to W. L. Stewart (Virginia Electric and Power Company), "Virginia Electric and Power Company-Amendment of Materials License No. SNM-2501, Amendment No. 1." Docket Nos. 50-280 and 72-2. (Accession No. 861140159-861110-Y)
336. Roberts, J. P., (NRC). March 6, 1987. Memorandum for L. C. Rouse (NRC), "Meeting with Virginia Electric and Power Company (VEPCO)." Docket Nos. 50-280 and 72-2. (Accession No. 8703130165-870306-Y)
337. Cunningham, R. E., (NRC). May 1, 1987. Letter to V. J. Barnhart (General Nuclear Systems, Inc.). "Request for Withholding Information from Public Disclosure." Docket Nos. 50-280 and 72-2 and Project No. M-37. (Accession No. 8705070053-870501-P,Y)
338. Cunningham, R. E., (NRC). October 29, 1987. Letter to V. J. Barnhart (General Nuclear Systems, Inc.), "Request for Withholding Information from Public Disclosure." Docket Nos. 50-280 and 72-2 and Project No. M-37. (Accession No. 8711030499-871029-Y)
339. Stewart, W. L., (Virginia Electric and Power Company). November 16 1987. Letter to Director, Division of Fuel Cycle, Medical, Academic and Commercial Safety, Office of Nuclear Material Safety and Safeguards (NRC), "Virginia Electric and Power Company, Surry Power Station, Independent Spent Fuel Storage Installation, Proposed Technical Specification Change." Docket Nos. 50-280 and 72-2. (Accession No. 8712110037-871116-Y)

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341. Stewart, W. L., (Virginia Electric and Power Company). December 31, 1987. Letter to U.S. Nuclear Regulatory Commission, "Virginia Electric and Power Company, Surry Power Station Independent Spent Fuel Storage Installation, Technical Specification Change for Use of Burnup Credit." Docket Nos. 50-280 and 72-2. (Accession No. 8801110327-871231-C)
342. Rouse, L. C., (NRC). January 27, 1988. Letter to Virginia Electric and Power Company, "Categorical Exclusion Under the Provision of 10 CFR 51.22(c)11." Docket Nos. 50-280 and 72-2. (Accession No. 8802030263-880127-Y)
343. Rouse, L. C., (NRC). April 15, 1985. Letter to E. E. Utley (Carolina Power and Light Company), "Notice of Consideration of a Materials License for the Storage of Spent Fuel at the Carolina Power and Light Company H. B. Robinson Steam Electric Plant Unit 2 Site." Docket Nos. 50-261 and 72-3. (Accession No. 8504220284-850415-Y)
344. Roberts, J. P., (NRC). June 7, 1985. Memorandum for L. C. Rouse (NRC), "Meeting with Carolina Power and Light (CP&L) Company." Docket Nos. 50-261 and 72-3. (Accession No. 8506180302-850607-Y)
345. Roberts, J. P., (NRC). July 26, 1985. Memorandum for L. C. Rouse (NRC), "Meeting Between Carolina Power and Light Company and Nuclear Regulatory Commission Staff." Docket Nos. 50-261 and 72-3. (Accession No. 8508070024-850726-Y)
346. Sturz, F., (NRC). September 10, 1985. Memorandum for L. C. Rouse (NRC), "H. B. Robinson Unit 2 Site Visit." Docket Nos. 50-261 and 72-3. (Accession No. 8509130419-850910-Y)
347. Zimmerman, S. R., (Carolina Power and Light Company). October 14, 1985. Letter to L. C. Rouse (NRC), "H. B. Robinson Steam Electric Plant, Unit No. 2, Docket No. 72-3, Response to Comments on the Independent Spent Fuel Storage Installation License Application." Docket Nos. 50-261 and 72-3. (Accession No. 8511130338-851014-Y)
348. Zimmerman, S. R., (Carolina Power and Light Company). October 25, 1985. Letter to J. P. Roberts (NRC), "H. B. Robinson Independent Spent Fuel Storage Installation, Docket No. 72-3, Site Plan." Docket Nos. 50-261 and 72-3. (Accession No. 8512100663-851025-Y)

349. Zimmerman, S. R., (Carolina Power and Light Company). January 13, 1986. Letter to J. P. Roberts (NRC), "H. B. Robinson Independent Spent Fuel Storage Installation, Docket No. 72-3, Submittal of Additional Information." Docket Nos. 50-261 and 72-3. (Accession No. 8602190023-860113-Y)
350. Zimmerman, S. R., (Carolina Power and Light Company). January 30, 1986. Letter to J. P. Roberts (NRC), "H. B. Robinson Independent Spent Fuel Storage Installation, Docket No. 72-3, Submittal of Additional Information." Docket Nos. 50-261 and 72-3. (Accession No. 8602210470-860130-Y)
351. Zimmerman, S. R., (Carolina Power and Light Company). February 4, 1986. Letter to J. P. Roberts (NRC), "H. B. Robinson Independent Spent Fuel Storage Installation, Docket No. 72-3, Submittal of Additional Information." Docket Nos. 50-261 and 72-3. (Accession No. 8602110154-860204-Y)
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353. Zimmerman, S. R., (Carolina Power and Light Company). March 11, 1986. Letter to L. C. Rouse (NRC), "H. B. Robinson Independent Spent Fuel Storage Installation, Docket No. 72-3, Schedule for Construction of Independent Spent Fuel Storage Installation (ISFSI)." Docket Nos. 50-261 and 72-3. (Accession No. 8603260147-860311-Y)
354. Roberts, J. P., (NRC). March 24, 1986. Memorandum for L. C. Rouse (NRC), "Meeting with Carolina Power and Light Company (CP&L)." Docket Nos. 50-261 and 72-3. (Accession No. 8603280270-860324-Y)
355. Rouse, L. C., (NRC). March 31, 1986. Letter to E. E. Utley (Carolina Power and Light Company), "Notice of Issuance and Finding of No Significant Impact and Issuance of Environmental Assessment." Docket Nos. 50-261 and 72-3. (Accession No. 8604080024-860331-Y; the enclosures are 8604080025-860331-Y and 8604080027-860331-Y, respectively)
356. Roberts, J. P., (NRC). April 3, 1986. Memorandum for L. C. Rouse (NRC), "Proposed Technical Specifications for the Carolina Power and Light Company (CP&L) Independent Spent Fuel Storage Installation (ISFSI)." Docket Nos. 50-261 and 72-3. (Accession No. 8604240393-860403-Y)
357. Zimmerman, S. R., (Carolina Power and Light Company). July 10, 1986. Letter to J. P. Roberts (NRC), "H. B. Robinson Independent Spent Fuel Storage Installation, Docket No. 72-3, Submittal of Current Drawings." Docket Nos. 50-261 and 72-3.

358. Zimmerman, S. R., (Carolina Power and Light Company). July 22, 1986. Letter to J. P. Roberts (NRC), "H. B. Robinson Independent Spent Fuel Storage Installation, Docket No. 72-3, Submittal of Current Drawings." Docket Nos. 50-261 and 72-3. (Accession No. 8608050415-860722-Y)
359. Cutter, A. B., (Carolina Power and Light Company). October 31, 1986. Letter to L. C. Rouse (NRC), "H. B. Robinson Independent Spent Fuel Storage Installation, Docket No. 72-3, Request for License Amendment." Docket Nos. 50-261 and 72-3. (Accession No. 8611240068-861031-Y)
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361. Rouse, L. C., (NRC). April 16, 1987. Letter to E. E. Utley (Carolina Power and Light Company), "Carolina Power and Light Company-Amendment of Materials License No. SNM 2502, Amendment No. 1." Docket Nos. 50-261 and 72-3. (Accession No. 8704270018-870416-Y)
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365. Eury, L. W., (Carolina Power and Light Company). November 25, 1987. Letter to Office of Nuclear Material Safety and Safeguards (NRC), "H. B. Robinson Independent Spent Fuel Storage Installation, Docket No. 72-3, Request for License Amendment." Docket Nos. 50-261 and 72-3. (Accession No. 8712310136-871125-Y)
366. Rouse, L. C., (NRC). February 11, 1988. Letter to E. E. Utley (Carolina Power and Light Company), "Carolina Power and Light Company-Amendment of Materials License No. SNM 2502, Amendment No. 2." Docket Nos. 50-261 and 72-3. (Accession No. 8802170204-880211-Y)

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368. Tucker, H. B., (Duke Power Company). October 11, 1988. Letter to H. Thompson (NRC), "Oconee Nuclear Station, Docket Nos. 72-4 and 50-269/270/287, Independent Spent Fuel Storage Installation (ISFSI), Request to Initiate ISFSI Site Preconstruction Activities." Docket Nos. 50-269 and 72-4. (Accession No. 8810190448-881011-Y).
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370. Tucker, H. B., (Duke Power Company). August 16, 1988. Letter to Document Control Desk (NRC), "Oconee Nuclear Station, Docket No. 72-4, Independent Spent Fuel Storage Installation (ISFSI), Response to NRC Questions - Environmental Report." Docket Nos. 50-269 and 72-4. (Accession No. 880B290098-880816-Y).
371. Rouse, L. C., (NRC). March 7, 1985. Letter to W. L. Stewart (Virginia Electric and Power Company), "Review of Submittal of February 8, 1985." Docket Nos. 50-280 and 72-2. (Accession No. 8503210021-850307-C).
372. Stewart, W. L., (Virginia Electric and Power Company). September 9, 1985. Letter to R. E. Cunningham (NRC), "Virginia Electric and Power Company, Surry Power Station Unit Nos. 1 and 2, Additional Information - ISFSI." Docket Nos. 50-280 and 72-2. (Accession No. 8510150041-850909-V).
373. Stewart, W. L., (NRC). December 27, 1985. Letter to R. E. Cunningham (NRC), "Virginia Electric and Power Company, Surry Power Station Unit Nos. 1 and 2, Additional Information - ISFSI - CASTOR V/21 Basket Indications." Docket Nos. 50-280 and 72-2. (Accession No. 8601290309-851227-Y)
374. Barnhart, V. J., (General Nuclear System, Inc.). November 22, 1985. Letter to L. C. Rouse (NRC).
375. Reid, R. W., (NRC). October 16, 1979. Letter to R. H. Groce (Maine Yankee Atomic Power Company), "Docket No. 50-309, Maine Yankee Atomic Power Company, Notice of Proposed Issuance of Amendment to Facility Operating License". Docket No. 50-309. (Accession No. 7911120494).
376. Garrity, J. H., (Maine Yankee Atomic Power Company). 1981. Letter to R. A. Clark (NRC), "Maine Yankee Spent Fuel Storage Modification - Complete Report," Docket No. 50-309, dated October 5, 1981.

377. Sears, P. M., (NRC). June 24, 1986. Letter to J. B. Randazza (Maine Yankee Atomic Power Company), "Maine Yankee Atomic Power Station, Spent Fuel Pool Masonry Wall-Request for Additional Information." Docket No. 50-309. (Accession No. 8606260438-860624-Q)
378. Whittier, G. D., (Maine Yankee Atomic Power Company). September 4, 1986. Letter to A. C. Thadani (NRC), "Response to Request for Information Concerning Spent Fuel Pool Masonry Wall." Docket No. 50-309. (Accession No. 8609120017-860904-P).
379. Kober, R. W., (Rochester Gas and Electric Corporation). June 12, 1984. Letter to D. M. Crutchfield (NRC), "Responses to Staff Questions, R. E. Ginna Nuclear Power Plant, Docket No. 50-244." (Accession No. 8406180221-840612-P).
380. Kober, R. W., (Rochester Gas and Electric Corporation). April 2, 1984. Letter to H. R. Denton (NRC), "Application for Amendment to Operating License." Docket NO. 50-244. (Accession No. 8404100119-840402-P).
381. Kober, R. W., (Rochester Gas and Electric Corporation). August 13, 1984. Letter to W. A. Paulson (NRC), "Responses to Questions." Docket No. 50-244. (Accession No. 8410150273-840813-P).
382. Zwolinski, J. A., (NRC). November 14, 1984. Letter to R. W. Kober (Rochester Gas and Electric Corporation), "Increase of the Spent Fuel Pool Storage Capacity."
383. Crutchfield, D. M., (NRC). June 12, 1985. Letter to R. W. Kober (Rochester Gas and Electric Corporation), "Storage of Consolidated Fuel." Docket No. 50-244. (Accession No. 8506140402-850612-P).
384. Miller, C. L., (NRC). August 6, 1985. Letter to Rochester Gas and Electric Corporation, "Summary of June 4, 1985, Meeting on Storage of Consolidated Fuel." Docket No. 50-244. (Accession No. 8508080410-850806-P)
385. Zwolinski, J. A., (NRC). November 8, 1985. Letter to R. W. Kober (Rochester Gas and Electric Corporation), "Notice of Consideration of Issuance of Amendment to License and Opportunity for Prior Hearing." Docket No. 50-244. (Accession No. 8511150031-851108-P).
386. Lear, G. E., (NRC). December 12, 1985. Letter to R. W. Kober (Rochester Gas and Electric Corporation), "Environmental Assessment and Finding of No Significant Impact-Consolidated Spent Fuel Storage." Docket No. 50-244. (Accession No. 8512190011-851212-P).
387. Council, W. G., (Northeast Nuclear Energy Company). March 30, 1984. Letter to J. R. Miller (NRC), "Millstone Nuclear Power Station, Unit 2, Spent Fuel Disposition Plans." Docket No. 50-336. (Accession No. 8404180358-840330-P).

388. Osborne, D. B., (NRC). June 4, 1984. Letter to Northeast Nuclear Power Station, Unit 2, "Summary of Meeting with Northeast Nuclear Energy Company on Spent Fuel Disposition Plans for Millstone Unit 2." Docket No. 50-336. (Accession No. 8406150141-840604-P)
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390. Jaffe, D. H., (NRC). August 21, 1986. Letter to J. F. Opeka (Northeast Nuclear Energy Company), "Request for Additional Information." Docket No. 50-336. (Accession No. 8608250420-860821-P.)
391. Jaffe, D. H., (NRC). August 27, 1986. Letter to J. F. Opeka (Northeast Nuclear Energy Company), "Request for Additional Information." Docket No. 50-336. (Accession No. 8609040362-860827-P.)
392. Jaffe, D. H., (NRC). September 22, 1986. Letter to J. F. Opeka (Northeast Nuclear Energy Company), "Request for Additional Information." Docket No. 50-336. (Accession No. 8609240055-860922-P.)
393. Opeka, J. F., (Northeast Nuclear Energy Company). October 22, 1986. Letter to A. C. Thadani (NRC), "Millstone Nuclear Power Station, Unit No. 2, Storage of Consolidated Spent Fuel." Docket No. 50-336. (Accession No. 8610290076-861022-P.)
394. Mroczka, E. J., (Northeast Nuclear Energy Company). May 27, 1987. Letter to U.S. Nuclear Regulatory Commission, "Millstone Nuclear Power Station, Unit No. 2, Storage of Consolidated Spent Fuel." Docket No. 50-336. (Accession No. 8706100396-870527-P.)
395. Jaffe, D. H., (NRC). June 2, 1987. Letter to E. J. Mroczka (Northeast Nuclear Energy Company), "Amendment No. 117 to Facility Operating License No. DPR-65 for Millstone Nuclear Power Station, Unit No. 2." Docket No. 50-336. (Accession No. 8706150005-870602-P.)
396. Mroczka, E. J., (Northeast Nuclear Energy Company). August 4, 1987. Letter to U.S. Nuclear Regulatory Commission, "Millstone Nuclear Power Station, Unit No. 2, Additional Information for Forty-Year License Extension." Docket No. 50-336. (Accession No. 8708110297-870804-P.)
397. Jaffe, D. H., (NRC). June 30, 1987. Letter to E. J. Mroczka (Northeast Nuclear Energy Company), "Expiration Date for the Millstone Unit 2 Operating License (TAC No. 64245)." Docket No. 50-336.
398. Mroczka, E. J., (Northeast Nuclear Energy Company). August 11, 1987. Letter to U.S. Nuclear Regulatory Commission, "Millstone Nuclear Power Station, Unit No. 2, Fuel Consolidation Program, Section 50.59 Evaluation of the Consolidation Process." Docket No. 50-336. (Accession No. 8708170457-870811-P.)

399. Mroczka, E. J., (Northeast Nuclear Energy Company). January 4, 1987. Letter to U.S. Nuclear Regulatory Commission, "Millstone Nuclear Power Station, Unit No. 2, Spent Fuel Consolidation (TAC No. 65274)." Docket No. 50-336. (Accession No. 8801130113-880104-P.)
400. Jaffe, D. H., (NRC). October 30, 1987. Letter to E. J. Mroczka (Northeast Nuclear Energy Company), "Millstone Nuclear Power Station, Unit No. 2 Spent Fuel Consolidation Process (TAC No. 65274)." Docket No. 50-336.
401. Rouse, L. C., (NRC). December 2, 1987. Letter to A. E. Scherer (Combustion Engineering, Inc.), Project No. M-43.

APPENDIX A

CHRONOLOGICAL LISTING OF LICENSING ACTIVITIES FOR INDIVIDUAL
TOPICAL REPORTS ON DRY SPENT FUEL STORAGE CASK AND MODULE DESIGNS

APPENDIX A

CHRONOLOGICAL LISTING OF LICENSING ACTIVITIES FOR INDIVIDUAL TOPICAL REPORTS ON DRY SPENT FUEL STORAGE CASK AND MODULE DESIGNS

3.1.1(a) General Nuclear Systems, Inc. (GNSI): CASTOR Ic, Project M-34

GNSI submitted their topical report on the CASTOR Ic dry spent fuel storage cask in January 1982.⁽³⁵⁾ The submittal was docketed under Project M-34 in June 1982.⁽³⁶⁾ GNSI met with the NRC several times^(35,36,58,166,232, and 233) and provided additional information.⁽¹⁶⁷⁻¹⁷⁰⁾ The NRC evaluated the original topical report⁽³⁶⁾ and Revisions 1-3^(59,165, and 234) to the report. The topical report for that cask was approved by the NRC; however, use of the cask is no longer anticipated because of its small capacity.⁽³⁴⁾

3.1.2 Combustion Engineering, Inc. (C-E): Project M-35

C-E submitted a topical safety analysis report entitled "Dry Storage Cask for an Independent Spent Fuel Storage Installation," CEND-270, that was docketed by NRC on November 24, 1982, under Project M-35.⁽³⁸⁾ NRC's comments on the topical report were provided to C-E on February 23, 1983.⁽³⁸⁾ C-E and NRC discussed the submittal and revising of CEND-270 in May 1983.⁽³⁹⁾ At the May 1983 meeting, C-E informed the NRC that its plans for proceeding with development of the dry storage cask design were uncertain and that no schedule had been developed for revising the topical report in response to NRC's comments. In June 1984, NRC informed C-E that in view of the elapsed time with no further action, that NRC had terminated this project review from its record system.⁽⁴⁰⁾

3.1.3 Mitsubishi Heavy Industries Limited (MHI): MSF-IV [formerly REA 2023 (PWR)], Project M-33

Ridihalgh, Eggers and Associates (REA) submitted a topical safety analysis report entitled "Topical Report for the REA 2023 Dry Storage Cask

(a) Refers to corresponding subsection of Section 3.1, "Topical Reports on Dry Spent Fuel Storage Cask and Module Designs."

for BWR Spent Fuel" (Revision 0) to the NRC in April 1983.⁽⁶¹⁾ The topical report was docketed by the NRC under Project M-33 on May 5, 1983. PNL provided comments on a January 1983 draft of the report to DOE's Commercial Spent Fuel Management Program Office; the comments were forwarded DOE (Richland Operations Office) to the NRC.⁽⁶¹⁾ The NRC evaluated the report and transmitted their comments to REA on July 13, 1983.⁽⁶²⁾ REA's response to NRC's comments was submitted to the NRC on October 11, 1983.⁽¹⁷¹⁾ The NRC performed a safety review of the design and operation of the ISFSI described in the topical report and transmitted the detailed comments to REA on October 25, 1983.⁽⁶³⁾ The rights to the REA cask designs for BWR and PWR fuel and the associated REA topical report submittals were purchased (effective date was April 5, 1985) by MHI.^(34,64) The NRC staff met with MHI on May 5, 1985, to discuss the acquisition of the REA dry storage technology by MHI.⁽²³⁵⁾ On May 22, 1985, MHI formally requested of the NRC that the applicant of record for Projects M-33 and M-36 be shifted from REA to MHI and that those project numbers be held open beyond May 28, 1985.⁽²³⁶⁾ MHI indicated to the NRC that it was initially seeking approval for the PWR cask design (Project M-36).⁽³⁴⁾

3.1.4 Mitsubishi Heavy Industries Limited (MHI): MSF-IV [formerly REA 2023 (PWR)], Project M-36

REA submitted a topical safety analysis report entitled "Topical Report for the REA 2023 Dry Storage Cask for PWR Spent Fuel," which was docketed by the NRC under Project M-36 on June 10, 1983.⁽⁶⁰⁾ Transferring of the rights to the REA cask designs for BWR and PWR fuel from REA to MHI is described above in Section 3.1.3. MHI indicated to the NRC that it was initially seeking approval for the PWR (REA 2023) cask design; however, MHI renamed it MSF IV and substantially changed the cask basket and the neutron shield part of the cask.⁽³⁴⁾ On May 21, 1986, MHI informed the NRC that because of budgetary considerations (MHI is faced with significantly higher estimated costs for this effort than were originally projected) they are forced to temporarily suspend work on Project M-36.⁽⁴¹⁾

3.1.5 General Nuclear Systems, Inc. (GNSI): CASTOR V/21, Project M-37

The "Topical Safety Analysis Report for the CASTOR V Cask Independent Spent Fuel Storage Installation (Dry Storage)" submitted to the NRC on December 16, 1983, by GNSI was docketed by the NRC on January 23, 1984, under Project M-37.⁽⁶⁵⁾ On April 2, 1984, the NRC transmitted⁽⁶⁵⁾ the comments from their review of the topical report to GNSI. On February 12, 1985, the NRC transmitted⁽²³⁷⁾ a letter to GNSI indicating several problems with Revision 1 of the topical report that was submitted on January 22, 1985: 1) this topical report (i.e., Revision 0) was originally submitted by GNSI but Revision 1 was submitted with a cover letter from Gesellschaft für Nuklear Service mbH (GNS)--the NRC requested that GNSI clarify this matter by letter indicating that GNSI is submitting the topical report and that GNS is acting for it; also in GNSI's letter of January 30, 1985, this revision was incorrectly identified at Revision 3, 2) the submittal is incomplete (several items were missing), and 3) Appendix 7, "Quality Assurance Handbook for Dry Spent Fuel Storage Cask" was not present in Revision 1 (however, NRC assumed it will be submitted separately). On February 15, 1985, GNSI indicated that the designation for the CASTOR V (Vb) Cask has been changed to CASTOR V/21.⁽⁶⁶⁾ On February 27, 1985, the NRC noted that they had received a response from GNSI to one of the problem areas in NRC's letter⁽²³⁷⁾ of February 12, 1985. On March 22, 1985, GNSI provided⁽¹⁷⁶⁾ responses to several questions and comments from Lawrence Livermore National Laboratory (LLNL); the responses include revised versions of 26 pages, information on the cask rear breach plate material, and information on the shielding calculation (involved the fuel basket boron content assumption). On April 15, 1985, the NRC noted⁽²³⁸⁾ that the German report entitled (English translation) "Assessment of Specific Questions of Dry Interim Storage of Spent Fuel Assemblies in a Shipping Cask Storage Site at AHAUS" had been declassified; this report is also in Appendix 12 of Revision 0 of the topical report under Project M-37. On October 10, 1985, GNSI notified⁽²³⁹⁾ the NRC that on October 3 EG&G Idaho, Inc. notified GNSI about several indications observed on the top end of the CASTOR V/21, borated stainless steel fuel basket; GNSI indicated that, based on their preliminary evaluation, they determined that this condition did not constitute a defect as defined by

10 CFR 21. On October 3 and 4, 1985, the NRC met with Bundesanalt fur Material prufung (BAM) to discuss 1) final arrangements for NRC to rely on BAM as an agent for inspection for dry storage casks produced in the Federal Republic of Germany (FRG) and 2) problems and actions to be taken with regard to damage (cracks in, or in the immediate vicinity of, weld positions had been observed at up to eight locations at the top of the fuel basket) to a CASTOR V/21 dry storage cask.⁽¹⁷²⁾ On October 14, 1985, the NRC requested from GNSI an analysis and evaluation of the cause(s) of the basket cracks and an assessment of the need for design changes. On November 27, 1985, GNSI transmitted⁽⁶⁹⁾ to the NRC the information requested by NRC in the letter of October 14, 1985; the response addressed the observations made at the Idaho National Engineering Laboratory (INEL) as well as analyses of a fuel basket under design thermal conditions described in the topical safety analysis report. On December 5, 1985, the NRC met with GNSI to discuss GNSI progress regarding investigation of the implications of the nonstructural weld cracks found in October 1985 in the basket of the CASTOR V/21 cask during the unlicensed DOE demonstration at INEL. On December 13, 1985, the NRC met with GNSI to discuss GNSI's preliminary report on the cracks in the fuel basket noted at INEL;⁽¹⁷⁴⁾ in response to NRC staff comments, GNSI was to provide additional information. On December 20, 1985, the NRC transmitted a letter⁽¹⁷³⁾ to GNSI outlining the seven main items covered at the meeting⁽¹⁷⁴⁾ on December 13 and to which GNSI was to respond. On February 3, 1986, GNSI furnished⁽⁷⁰⁾ the NRC with 1) GNSI's evaluation of the fuel basket used in the unlicensed demonstration program at INEL and an assessment of the impact of the observations at INEL on the licensed use of the CASTOR V/21 cask at an ISFSI as requested in NRC's letter of October 14, 1985; 2) the items discussed at the December 13, 1985, meeting;⁽⁶⁹⁾ and 3) a response to the items listed in NRC's letter⁽¹⁷³⁾ of December 20, 1985. On February 28, 1986, GNSI submitted an alternative design for a cask fuel basket that does not employ borated stainless steel as a basket material.⁽⁴⁴⁾ On April 29, 1986, the NRC noted⁽¹⁷⁷⁾ the changes proposed by GNSI in the CASTOR V/21 fuel basket design (the design refers to an all stainless steel basket). On April 30, 1986, the NRC approved use of the CASTOR V/21 cask with a stainless steel (not borated type) fuel basket.⁽⁴⁴⁾ On August 15, 1986, GNSI transmitted⁽⁷¹⁾

to the NRC the results of analyses and tests performed to validate the use of borated stainless steel (Radionox A-18) for the fabrication of the welded fuel basket of the CASTOR V/21 storage cask. The transmittal included 1) basket analyses and comments, 2) the evaluation (Revision 1, dated August 1986) of the demonstration test at INEL in which it was concluded that the observed indications (i.e., slight indications of separation at certain joints in the basket structure) were caused by constrained thermal expansion; and 3) revisions to the topical safety analysis report. The NRC met with GNSI on October 7, 1985, to discuss the safety review of the CASTOR V/21 borated stainless steel fuel basket design; the NRC concluded that GNSI had adequately explained and analyzed the cause of the basket weld cracks at INEL but that GNSI would need to 1) analyze the effects of dynamic impact on the borated stainless steel base material and weld area material of the redesigned basket, and 2) present the GNSI test data needed to acceptably complete the material characterization.⁽¹⁷⁵⁾ On February 5, 1987, GNSI submitted⁽⁷²⁾ to the NRC the results of the engineering evaluation and testing program that culminated GNSI's efforts to validate the use of borated stainless steel (in particular, Radionox A-18 with a nominal 1.03 wt% boron content) in the fabrication of the welded basket for the CASTOR V/21 cask. On April 3, 1987, the NRC concluded⁽⁴⁴⁾ that the borated stainless steel basket design meets the requirements of 10 CFR Part 72; the NRC also indicated that several items in Revision 1 of the topical report need to be clarified, corrected, and/or added. On June 16, 1987, GNSI informed⁽⁷³⁾ the NRC that it was discovered in the week of June 1, 1987, that some material used in the fabrication of a cask basket contained a boron content that was not within the tolerance of the material specification approved by the NRC. On October 2, 1987, the NRC sent a letter⁽⁷⁴⁾ to GNSI regarding the NRC staff's review of the two fuel baskets (for CASTOR V/21 cask serial numbers 008 and 009) with material deviations noted in GNSI's letters June 16⁽⁷³⁾ and August 3, 1987. The NRC reviewed the issue and concluded that use of the material does not compromise safety. On September 27, 1988, GNSI transmitted⁽¹⁷⁸⁾ to the NRC a letter describing the GNS review/audit of the material deviation problem and the GNSI/GNS corrective actions to be implemented. GNS found the cause of the problem to be the control of

documents relevant to manufacture in compliance with the NRC license and dissemination of the latest approved document to the appropriate organization.

3.1.6 Nutech Engineers, Inc. (NUTECH): NUHOMS-07P, Project M-39

The dry storage concrete module independent spent fuel storage installation submittal, "Topical Report for the NUTECH Horizontal Modular Storage System for Irradiated Nuclear Fuel" (NUH-001), was docketed by the NRC on December 19, 1984, under Project M-39.⁽⁷⁵⁾ In February 1985, the NRC requested⁽⁷⁷⁾ additional information in three design categories (see Table 3.8). NUTECH subsequently provided two responses.^(180,240) NUTECH and the NRC discussed the topical report at a meeting in April 1987.⁽¹⁷⁹⁾ NRC transmitted⁽¹⁶⁰⁾ the results of their review of the quality assurance program in June 1985. NUTECH met with the NRC in August 1985 to discuss NUTECH's approach to responding to NRC's comments;⁽²⁴¹⁾ issues involving criticality, thermal hydraulics, and shield penetrations remained open at that time and were to be discussed at a NUTECH/NRC meeting in September 1985. NUTECH provided responses to NRC's detailed comments on the topical report on September 6, 1985.⁽¹⁸¹⁾ NUTECH transmitted a letter⁽¹⁶¹⁾ as a follow-up to the NUTECH/NRC meeting in September 1985; five main areas were identified (See Table 3.8) where NRC wants more information and NUTECH provided an action plan for revising the topical report accordingly. An NRC memorandum⁽⁷⁸⁾ regarding the NUTECH/NRC meeting in September 1985 indicates the need for NUTECH to provide additional information in the areas of shielding penetration estimates and criticality calculations. On September 25, 1985, NRC transmitted⁽¹⁶²⁾ a request to NUTECH for additional information concerning the quality assurance program. In an October 1985 letter,⁽⁷⁹⁾ NRC suggests that each new concrete module and canister be examined after its first full year of service. On November 4, 1985, NUTECH provided⁽¹⁸²⁾ responses to NRC's request for more information on the quality assurance program. On November 11, 1985, NUTECH submitted Revision 1⁽²⁴²⁾ to the topical report. In January 1986, NUTECH transmitted^(163,164) the additional information requested by the NRC (see Table 3.8 in Section 3.0). Revision 1 of the topical report was accepted as a reference by NRC on March 28, 1986.⁽⁴⁶⁾ An NRC memorandum⁽⁸⁰⁾ indicates that Duke Power Company (with

NUTECH Engineers, Inc.) intends to apply by April 1, 1988, for a license under 10 CFR Part 72 for dry storage of spent fuel at the Oconee Nuclear Station site. To facilitate this application, NUTECH will submit by January 4, 1988, an amendment to its currently approved topical report for its NUHOMS design. NUTECH provided⁽²⁴³⁾ detailed calculations for the horizontal storage module load combinations; also included was a reference showing how the cracked section analysis was performed.

3.1.7 Nuclear Assurance Corporation (NAC): S/T (26 PWR), Project M-40

NAC submitted its "Topical Safety Analysis Report for the NAC Storage/Transport Cask for Use at an Independent Spent Fuel Storage Installation" to the NRC on December 13, 1984.⁽⁸²⁾ The topical report was docketed by the NRC under Project M-40 on December 19, 1984.⁽⁸³⁾ On October 1, 1985, the NRC requested additional information (involved 12 questions) on the NAC quality assurance (QA) program.⁽⁸⁴⁾ The NRC transmitted their comments on the topical safety analysis report for the NAC storage/transport (S/T) cask on October 25, 1985.⁽⁸³⁾ The NRC and NAC met on March 27, 1986, to discuss the comments.⁽⁸⁶⁾ On April 25, 1987, NAC submitted the response⁽²⁴⁴⁾ to the October 1, 1985, questions.⁽⁸⁴⁾ On June 16, 1986, the NRC requested⁽⁸⁵⁾ clarification and/or additional information (involved four questions on the NAC quality assurance program) from NAC. On July 1, 1986, NAC provided⁽²⁴⁵⁾ responses to NRC's comments on the topical safety analysis report. On July 22 through 23, 1986, NRC met with NAC at Lawrence Livermore National Laboratory (LLNL) to discuss NAC's responses to the NRC staff's initial safety review comments on the topical safety analysis report.⁽⁹¹⁾ At that time, it was indicated that NAC was to submit a different cask basket design in the revised topical report: the design will accommodate 26 PWR spent fuel assemblies without reliance on burnup credit with respect to fuel basket criticality design. NAC provided⁽²⁴⁶⁾ responses to NRC's June 16, 1986, questions. On January 30, 1987, NAC provided⁽²⁴⁷⁾ responses, proposed specific NAC actions, and actual specific NAC actions to NRC's detailed comments on the topical safety analysis report. On February 24, 1987, NRC and NAC met to discuss the safety review of the topical safety analysis report.⁽⁸⁸⁾ On March 3, 1987, the NRC transmitted⁽⁸⁹⁾ to NAC their detailed comments on Revision 1 (submitted⁽²⁴⁸⁾ by NAC on December 18, 1986) to the

topical safety analysis report. On April 29, 1987, NRC and NAC met to discuss a new dry storage cask design for storing consolidated PWR fuel in canisters (see Project M-51, cask capacity: 28 canisters) and the topical report (in particular the quality assurance program) for NAC's S/T dry cask design for intact spent fuel assemblies.⁽⁸⁷⁾ On September 11, 1987, NAC furnished⁽²⁴⁹⁾ responses to seven questions raised by LLNL on September 8, 1987. LLNL contacted NAC on September 30, 1987, regarding another question.⁽²⁵⁰⁾ On October 21, 1987, NAC submitted responses to 23 questions raised by LLNL on October 7, 1987, and responses to 10 questions raised by LLNL on October 13, 1987.⁽²⁵¹⁾ The NRC indicated on June 16, 1987, that it had reviewed the NAC submittals and determined that NAC described a QA program for its S/T casks that meets the requirements of Subpart G of 10 CFR 72.⁽²⁵²⁾ On November 5, 1987, NAC provided clarification requested by LLNL on the primary penetration cover.⁽²⁵³⁾ On November 13, 1987, NAC furnished responses⁽²⁵⁴⁾ to three questions raised by LLNL on October 26, 1987. On November 19, 1987, NAC transmitted⁽²⁵⁵⁾ responses to five questions raised by LLNL on November 17, 1987. On November 20, 1987, NRC and NAC met to discuss new NAC dry storage cask designs:⁽²⁵⁶⁾ these cask designs will be similar to the submitted NAC S/T cask design, but one cask (Project M-51) will store rods from up to 56 consolidated PWR spent fuel assemblies in 28 cans and the other cask (Project M-52) will store 31 intact PWR assemblies with allowance for burnup credit taken in cask criticality design (both will be associated with the licensing activities for an ISFSI at Surry). On November 25, 1987, NAC provided a response⁽²⁵⁷⁾ to a question raised by LLNL on November 23, 1987. On December 2, 1987, NAC transmitted⁽¹⁸³⁾ responses to three questions raised by LLNL. On December 17, 1987, NAC provided responses⁽¹⁸⁴⁾ to questions raised by LLNL on December 8, 1987. On January 27, 1988, NRC met with NAC to discuss submittal of a new topical report for a dry spent fuel storage cask design for storage of consolidated spent fuel rods; technical issues involving criticality analysis, thermal analysis, shielding, and dose rates were discussed.⁽¹¹⁰⁾

3.1.8 Westinghouse Electric Corporation: MC-10, Project M-41

The topical report entitled "Topical Safety Analysis Report for the Westinghouse MC-10 Cask for an Independent Spent Fuel Storage Installation

(Dry Storage)" was docketed by the NRC on February 13, 1985, under Project M-41.(92) In September 1985, NRC requested additional information;(185) in October 1985, Westinghouse provided a response to that request.(187) In December 1985, the NRC transmitted the comments from their review of the report.(92) In August 1986, Westinghouse provided(188) responses to the NRC's comments. In reviewing the cask materials, the NRC and Lawrence Livermore National Laboratory (LLNL) raised major concern relative to the basket material. Specifically, the NRC was concerned that the basket material was not an ASME Code Class 1 material.(93) It was recommended that Westinghouse change the basket material (aluminum) to an ASME Code Class 1 material or provide further evaluation to demonstrate the acceptability of the aluminum basket. Westinghouse resolved the issue by changing the basket material to stainless steel and indicating that several engineering reanalyses would be performed to reflect the new material.(93,94) Westinghouse provided responses to NRC and LLNL questions/requests in December 1986,(189) February 1987,(190) and May 1987.(191) In July 1987, Westinghouse provided responses to LLNL questions.(186) In July 1987, Westinghouse submitted to NRC the responses to LLNL questions.(192) In July 1987, Westinghouse provided responses to LLNL's questions concerning MC-10 radiation protection.(193) Westinghouse's responses to LLNL questions regarding governing differential equations and derivations was provided in July 1987.(194) Responses by Westinghouse to LLNL's questions on the thermal analysis review of MC-10 were furnished in July 1987.(195) Westinghouse provided additional supporting information on responses to LLNL questions in July 1987.(196) Westinghouse's response to an LLNL question on fuel rod pressure was provided in August 1987.(197) The Westinghouse response to an LLNL question about the MC-10 thermal analysis review was provided in September 1987.(198) The thermal analysis response by Westinghouse to an LLNL question on fuel cladding integrity was transmitted in September 1987.(199) The Westinghouse response to an LLNL question regarding the design basis heat load was furnished in September 1987.(200) Westinghouse provided an updated response to an LLNL question concerning the fuel rod cladding temperature limit in September 1987.(201) The NRC has completed a safety review of the Westinghouse topical report(258) for the MC-10 metal cask and approved the topical report for use

as a reference in 10 CFR Part 72 site-specific license applications.⁽³⁷⁾ In September 1987, the NRC issued⁽⁴⁷⁾ the letter of approval and the safety evaluation report.

3.1.9 Transnuclear, Inc. (TN): TN-24P, Project M-42

On August 30, 1985, TN submitted their TN-24 Dry Storage Cask Topical Report.⁽⁹⁵⁾ The report was docketed by the NRC under Project M-42. On July 18, 1986, the NRC transmitted to Transnuclear the comments from its review of the topical report.⁽⁹⁵⁾ On December 15, 1985, Transnuclear provided⁽²⁰²⁾ NRC with a description of the approach Transnuclear expects to follow in furnishing the additional information requested by NRC.⁽⁹⁵⁾ On February 27, 1987, Transnuclear transmitted⁽²⁰³⁾ responses (involved pages of TN document E-9213, Quality Assurance Program for Design, Fabrication, Inspection, and Testing of Storage Systems for Spent Fuel and Associated Radioactive Materials) to an NRC verbal request. On July 17, 1987, NRC met with TN at Lawrence Livermore National Laboratory (LLNL) to discuss the submittal of a revised topical report.⁽⁹⁶⁾ Nine key items and 14 other important items were discussed at the meeting on July 17, 1987. The nine key items were design drawings, missiles, seismic loading, cask drops and tipping, basket loading criteria, containment boundary definition, poison spider assembly, basket material characterization, and cobalt-60 source characterization. The 14 other items were principal design criteria, structural features, general systems description, geography and demography of site selected, materials to be stored, structural and mechanical safety criteria, safety protection systems, shipping cask repair and maintenance, cathodic protection, structural analysis of containment vessel, operations systems, radiation protection, accidents, and operating controls and limits. A major unresolved issue remained at the end of the meeting on July 17, 1987. Criticality design for the fuel basket relies on insertion of a "poison spider assembly" of rods into a fuel assembly. The NRC questioned whether this approach can satisfy design criteria of 10 CFR Part 72 for at-reactor site storage, at least in the nonsite-specific context of a topical report. On July 27, 1988, TN provided⁽²⁰⁴⁾ NRC with copies of the TN-24 Topical Report E-1707, Revision 1, and a response (E-10463) to NRC's questions;⁽⁹⁵⁾ TN addressed NRC questions⁽⁹⁵⁾ about quality assurance in an earlier letter [K. Goldmann (TN) to

L. C. Rouse (NRC), January 30, 1987]. NRC met with TN on July 29, 1988, to discuss the submittal of a revised topical report; changes made in response to NRC comments were discussed (a significant change in the fuel basket design has been made).⁽⁹⁸⁾ The NRC is currently reviewing the revised topical report that was submitted.⁽²⁾

3.1.10 Combustion Engineering Inc. (C-E): Dry Cap -P24 and -B60,
Project M-43

The NRC's letter of March 16, 1983⁽²⁵⁹⁾ included the initial set of comments from the NRC's review of the Topical Safety Analysis Report for a Combustion Engineering Dry Storage Cask for an ISFSI (in this case, it was associated with the ISFSI at Surry). The letter included 34 pages of specific comments and several general comments. Briefly, the NRC's general comments included the following:

1. The topical report is not sufficiently substantive in many areas to facilitate an orderly review. It is particularly lacking in an adequate description of the cask and its safety related appurtenances, details relating to failure criteria and analysis, description of sequential procedures for filling, sealing, lifting and transporting the cask, and illustration of models.
2. The report needs to be expanded to include sufficient information for reviewers to get a clearer picture of the cask and the analyses that were performed.
3. While the report addresses a single cask some consideration should be given to a generic treatment of casks in typical arrays such as in-line, square, etc. Sufficient information should be provided to enable the site-specific license applicant to evaluate a particular array with regard to thermal and radiological conditions within the array and at the boundaries, and to use reviewed methodologies for array calculations in subsequent license applications.
4. It would facilitate matters if copies of the references (excluding voluminous program descriptions) were submitted with the report.

On December 19, 1985, Combustion Engineering, Inc. (C-E) submitted its "Topical Safety Analysis Report for the Combustion Engineering Dry-Cap Cask for an Independent Spent Fuel Storage Installation (Dry Storage)" to the NRC; the NRC docketed the report under Project M-43.⁽⁹⁹⁾ On February 18, 1986, the NRC transmitted⁽⁹⁹⁾ a request (involves three items) to C-E for additional information on the quality assurance program. On March 14, 1986, NRC

met with C-E to discuss C-E's spent fuel dry storage cask topical report; C-E was concerned that the review of their topical report had not yet begun.⁽²⁶⁰⁾ On March 26, 1986, C-E furnished⁽²⁰⁵⁾ responses to NRC's comments⁽⁹⁹⁾ dated February 18, 1986. On May 21, 1986, C-E transmitted⁽²⁶¹⁾ a letter to the NRC requesting that NRC attempt to meet the schedule for approving the topical report. On August 7, 1986, the NRC transmitted to C-E the NRC's comments from the review of the topical report.⁽¹⁰⁰⁾ The NRC also had one general comment at that time: there are numerous references in the topical report to 10 CFR Part 71 requirements, but the NRC is not reviewing this report under Part 71, but under Part 72. On October 17, 1986, C-E provided⁽²⁶²⁾ proprietary responses to some of NRC's comments.⁽¹⁰⁰⁾ On June 4, 1987, C-E submitted⁽²⁶³⁾ Revision 2 of the topical report, CEND-273-NP/P, which is dated April 1987. On June 25, 1987, NRC transmitted⁽¹⁾ comments on Revision 2 of the topical report.⁽²⁶³⁾ The NRC observed a number of discrepancies and omissions and requested that C-E re-edit and resubmit the topical report. The NRC plans to complete the review of the topical report, except for the criticality portion (the NRC is examining the issue of burnup credit allowance in criticality analysis on a generic basis with a view toward establishing a branch technical position related to cask storage design). The NRC indicated that the portion can be left as either a site-specific issue in a license application or for future review upon development of a staff technical position. On August 14, 1987, C-E submitted⁽²⁶⁴⁾ Revision 3 of the topical report; Revision 3 reflects the latest information on the cask design. C-E also indicated that considerable resources have been expended in preparing and defending the topical report, CENPD-273, with credit-for-burnup application to dry storage. On November 23, 1987, C-E furnished⁽²⁶⁵⁾ the additional information requested⁽¹⁰⁰⁾ by the NRC; the questions generally relate to the magnitude of fuel rod internal pressure and cask cavity pressure and to the potential for adverse effects resulting from the expected pressures. On December 2, 1987, the NRC indicated⁽¹⁰¹⁾ that the NRC staff's technical concerns relative to credit-for-burnup application to dry spent fuel storage are expressed in a recent paper.⁽³⁾ On July 25, 1988, C-E transmitted⁽²⁶⁶⁾ copies of proprietary drawings in Appendix A of Revision 3 (August 1987) of the topical report to Lawrence Livermore Laboratory (LLNL)

as requested by NRC. On August 15, 1988, C-E transmitted⁽²⁶⁷⁾ a response to a request from LLNL; the request included providing a detailed description of the impact limiter, describing the load limiters on the trunnions, and providing a sketch of the lift rig attached with the load limiters in place. C-E's revised topical report is currently being reviewed by the NRC.⁽²⁾

3.1.11 FW Energy Applications, Inc. (FWEA): MVDS, Project M-46

FWEA's "Topical Report for the Foster Wheeler Modular Vault Dry Store (M.V.D.S.) for Irradiated Nuclear Fuel" (FWEA-08-27164 Revision 0) was submitted to the NRC on September 15, 1986.^(102,103) The topical report was docketed by the NRC under Project M-46 on September 30, 1986.⁽¹⁰⁴⁾ On October 14, 1986, FWEA provided a response⁽²⁶⁸⁾ to an NRC question. The NRC was having difficulty determining the capacity of the modular vault dry store (MVDS) on which the FWEA topical report is based. FWEA stated that the smallest MVDS will have two vault modules: the PWR vault modules are designed for 83 storage tubes, and the BWR vault modules for 150 storage tubes. On February 27, 1987, the NRC transmitted⁽¹⁰⁴⁾ questions/comments from their review of the report to FWEA. The questions/comments covered the proposed MVDS design, the proposed use of air as a cover gas, and the quality assurance program. The letter⁽¹⁰⁴⁾ also indicated that NRC needs the design calculation and other information that was discussed at the meeting of NRC and FWEA on February 5 and 6, 1987. On March 24, 1987, FWEA provided⁽²⁶⁹⁾ information requested⁽¹⁰⁴⁾ by the NRC. On May 4, 1987, the NRC transmitted⁽¹⁰⁵⁾ comments on the material submitted⁽²⁶⁹⁾ by FWEA on March 24, 1987; the NRC indicated that progress of the staff's review is blocked in some areas because of a lack of information. On May 13, 1987, FWEA provided⁽²⁷⁰⁾ additional information that NRC requested⁽¹⁰⁴⁾ on the quality assurance program. On May 22, 1987, FWEA furnished^(271,272) additional responses to NRC's request.⁽¹⁰⁴⁾ On June 8, 1987, the NRC sent a letter⁽²⁷³⁾ to FWEA indicating it agreed that FWEA should visit PNL to discuss experimental and theoretical matters regarding fuel oxidation. On June 10, 1987, FWEA transmitted⁽²⁷⁴⁾ responses to some more of NRC's questions/comments.⁽¹⁰⁴⁾ On June 12, 1987, FWEA furnished⁽²⁷⁵⁾ additional information, which in combination with three previous transmittals, completes the FWEA responses to the NRC's questions/comments.⁽¹⁰⁴⁾ The NRC and FWEA met on June 18, 1987, to

discuss the status of the FWEA topical report review⁽²⁷⁶⁾; the NRC staff concluded that the FWEA design for the MVDS was sufficiently complete and final so a safety review could be completed (tentative target date: end of July 1987). The NRC also requested clarifying information on a General Electric Company of the United Kingdom (GEC) version of a thermal code used in the MVDS design. On August 7, 1987, the NRC transmitted⁽¹⁰⁶⁾ comments to FWEA from the NRC's review of the topical report (FWEA-08-27164, Revision 0) and the additional information and proposed responses submitted by FWEA in response to the NRC's questions/comments.⁽¹⁰⁴⁾ The NRC indicated it was still evaluating the proposed use of air as a cover gas in conjunction with or in place of nitrogen. On September 2, 1987, NRC met with FWEA and GEC to discuss FWEA's proposed responses to NRC staff comments on the topical report for the MVDS design.⁽¹⁰⁷⁾ The FWEA responses appeared satisfactory, with a few exceptions. FWEA was to continue examining Comment 28 (involves use of Viton O-rings) and Comments 60 and 61 (they address a fuel handling drop accident). One other issue (involves storage of spent fuel in air) remains unresolved. The NRC also requested responses to two other comments. On September 11, 1987, FWEA transmitted⁽²⁷⁷⁾ responses to most of the NRC's comments.⁽¹⁰⁶⁾ On February 1, 1988, NRC met with FWEA and GEC to discuss the status of the topical report review.⁽²⁷⁸⁾ On March 23, 1988, the NRC issued⁽⁵¹⁾ its letter of approval for Revision 1 of FWEA's topical report; however, the NRC staff did not accept the use of air as a storage cover gas for the MVDS at that time, but did not reject the contention that continued research in this area may subsequently result in allowance of such use. The NRC also indicated that Section 10, "Development and Operating Controls and Limits," of the topical report is not to be cited as a reference. On April 14, 1988, NRC met with FWEA and GEC to discuss site-specific implications of FWEA's MVDS design topical report.⁽²⁷⁹⁾

3.1.12 Nuclear Assurance Corporation (NAC): Modified S/T (for 28 canisters of consolidated fuel rods), Project M-51

The NRC and NAC met in January 1988 to discuss NAC's submittal of a new topical report for a stainless steel and lead dry storage cask for storage of consolidated spent fuel rods at an ISFSI.⁽¹¹⁰⁾ The proposed cask design was

a modification of the NAC S/T cask with a basket designed to hold 28 canisters containing spent fuel rods from 56 PWR fuel assemblies.⁽¹²⁴⁾ Technical issues involving criticality analysis, thermal analysis, shielding and dose rates were discussed.⁽¹¹⁰⁾ This design was associated with a 10 CFR Part 72 amendment application planned by Virginia Power for its Surry Power Station ISFSI. NAC submitted its "Topical Safety Analysis Report for the NAC Storage/Transport Cask Containing Consolidated Fuel for Use at an Independent Spent Fuel Storage Installation"⁽¹⁰⁸⁾ to the NRC; the topical report was docketed by the NRC on February 24, 1988, under Project M-51.⁽¹⁰⁹⁾ NRC transmitted comments from their review of the topical report on June 9, 1988.⁽¹⁰⁹⁾ NAC provided responses to the June 9, 1988, comments on July 20, 1988.⁽²⁰⁶⁾ NRC and NAC met on July 21, 1988, to discuss NAC's responses to the NRC staff's comments. The NAC responses are to be incorporated into Revision 1 of the topical report.^(206,280) On August 23, 1988, NAC provided a response to an LLNL request for clarification of the fuel basket dimensions.⁽²⁰⁷⁾ On August 24, 1988, NAC transmitted responses to eight questions (pertained to the neutron shield and the fuel basket) raised by LLNL on August 23 and 24.⁽¹¹¹⁾ On September 1, 1988, NAC furnished the additional information requested by LLNL.⁽²⁰⁸⁾ On September 7, 1988, the NRC indicated acceptance of Revision 1 of the topical report for the cask design as a reference.⁽⁵²⁾

3.1.13 Nutech Engineering Inc. (NUTECH): NUHOMS-24P, Project M-49

On February 26, 1988, NUTECH submitted the "Topical Report for the NUTECH Horizontal Modular Storage System for Irradiated Nuclear Fuel" (NUH-001, Revision 2, February 1988).⁽¹¹⁴⁾ The topical report was docketed by the NRC under Project M-49. This report describes the design features and safety analyses of the NUHOMS-24P design, which is similar to the earlier system (NUHOMS-07P under Project M-39) that has been reviewed and approved by the NRC. On March 8, 1988, the NRC indicated that a serious problem of potential confusion was raised by NUTECH's use of the same report number, NUH-001, for the topical report under Project M-39 (NUHOMS-07P design) and under Project M-49 (NUHOMS-24P design).⁽¹¹⁵⁾ The NRC requested that NUTECH resolve the problem by re-examining the report number used and revise it.

NUTECH responded⁽¹¹⁶⁾ on March 18, 1988, to the NRC's request and redesignated the topical report for the NUHOMS-24P design as NUH-002; NUTECH planned to reprint the entire document with the NUH-002 designation subsequent to completion of the NRC review. On May 24, 1988, the NRC transmitted⁽²⁸¹⁾ questions to NUTECH from the review of the topical report. On July 7, 1988, NUTECH provided responses⁽²⁰⁹⁾ to the NRC's questions.⁽²⁸¹⁾ In August 1988, NUTECH furnished proprietary additional responses^(282,283) to the NRC's question.⁽²⁸¹⁾ The responses concerned the DSC^(a) structural analysis, DSC support structural analysis, transfer cask structural analysis, DSC and HSM^(b) thermal analysis, HSM structural analysis, transfer cask and HSM shielding analysis, and the NUHOM-24P criticality analysis. On August 10, 1988, NUTECH transmitted⁽²⁸⁴⁾ the NUHOMS-24P Topical Report NUH-002, Revision 1. On September 13, 1988, NUTECH provided⁽²⁸⁵⁾ a partial response (proprietary) to the NRC's questions⁽²⁸⁶⁾ (Questions 1, 2, 5, 9, and 10) of August 31, 1988. On September 26, 1988, NUTECH transmitted proprietary responses⁽²⁸⁷⁾ to the NRC's questions⁽²⁸⁶⁾ (Questions 3, 4, 7, and 8). On October 12, 1988, the NRC requested information from NUTECH on 12 items.⁽²⁸⁸⁾ On October 19, 1988, NUTECH announced that its NUHOMS technology (specifically the NUHOMS-24P design) was selected by Baltimore Gas and Electric (BG&E) for its Calvert Cliffs plant. On November 1, 1988, the NRC requested additional information.⁽²⁸⁹⁾ On November 9, 1988, NUTECH transmitted proprietary responses⁽²⁹⁰⁾ to NRC's questions of October 12⁽²⁸⁸⁾ and November 1, 1988.⁽²⁸⁹⁾ NUTECH also provided⁽²⁹¹⁾ supplementary proprietary information on one response⁽²⁹⁰⁾ sent earlier. On December 7, 1988, NUTECH transmitted proprietary information concerning "ALGOCJG Cask Top Model-Top Drop."⁽²⁹²⁾ The NRC is currently reviewing the NUHOMS-24P design in association with Duke Power Company's license application for an ISFSI at its Oconee Station site.⁽²⁾

3.1.14 Nuclear Packaging, Inc. (NuPac): CP-9, Project M-44

On October 9, 1986, NRC met with Nuclear Packaging, Inc. (NuPac) to discuss NuPac's submittal of a topical for a dry concrete spent fuel storage

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- (a) Dry Shielded Canister (DSC).
 - (b) Horizontal Storage Module (HSM).

cask design.⁽²⁹³⁾ NuPac indicated it was preparing a drop testing program in concert with analysis to address safety issues regarding dynamic impact effects on a concrete cask. NuPac sought to perform such drop testing on scale models of its CP-9 cask design. Use of scaling, as NuPac noted, introduces variables of its own. The NRC indicated that further consideration was needed to adequately scope and plan a testing program. As a result of the meeting on October 9, 1986, NuPac was to examine previous scale model drop testing efforts and analyses. On March 3, 1987, the NRC transmitted⁽¹¹⁹⁾ six comments to NuPac regarding their internal report, "Impact Performance and Testing of the NuPac CP-9 Scale Model Test Program," that was submitted to NRC on December 17, 1985. NRC concluded at that time that they saw nothing incorrect in NuPac's understanding of scale model testing, but NRC indicated they did not have sufficient information to judge whether the NuPac test program and topical report would be adequate. On March 23, 1988, the NRC docketed the "Topical Safety Analysis Report for the NuPac CP-9 Concrete Storage Cask (TP-08)" under Project M-44; the NRC reviewed Revision 0 and Revision 1 and transmitted comments to NuPac.⁽²¹⁰⁾ On October 28, 1988, Pacific Sierra Nuclear Associates provided responses⁽¹²⁰⁾ to NRC's questions concerning the concrete temperatures of the CP-9 storage cask. Pacific Sierra Nuclear Associates indicated that it believes the answers will clarify their use of structural concrete in the CP-9 cask design and show that this concrete meets the current ACI-349 code. It was reported⁽²⁾ in January 1989 that the NRC is continuing its review of NuPac's concrete cask design model CP-9.

3.1.15 General Nuclear Systems, Inc. (GNSI): CASTOR X, Project M-50

The NRC staff met twice^(122,294) with Virginia Power (VP) and GNSI to discuss the submittal of the CASTOR X topical report. On January 13, 1988, GNSI indicated to the NRC that it was preparing a topical safety analysis report for a new design spent fuel storage cask for use at an ISFSI.⁽²⁹⁵⁾ The cask, denoted as CASTOR X/28-33, is an enhanced capacity storage cask designed for spent fuel cooled for a minimum of ten years. The cask design is based on ductile cast iron construction and the safety evaluation methods employed in the CASTOR Ic (Project M-34) and CASTOR V/21 (Project M-37) topical safety analysis reports previously approved by the NRC. The CASTOR

X/28-33 design has an accommodation for the use of burnup credit in the criticality design of the basket for the 33-fuel assembly cask. The topical report is to be submitted for initial use at the ISFSI at VP's Surry Nuclear Station. VP intends to obtain an amendment to their license (SNM-2501) for burnup credit. On July 1, 1988, GNSI submitted its "Topical Safety Analysis Report for the CASTOR X Cask for an Independent Spent Fuel Storage Installation (Dry Storage)" to the NRC⁽¹²¹⁾ The topical report was docketed by the NRC under Project M-50. On December 29, 1988, GNSI provided⁽²⁹⁶⁾ a) the CASTOR X design drawings specified in their initial application,⁽¹²¹⁾ b) revised information pertaining to the structural analyses performed in support of the CASTOR X basket options, and c) a revision to the Cask Body Material Specification (BS-04).

3.1.16 Nuclear Assurance Corporation (NAC): NAC-31 S/T, Project M-52

On May 18, 1988, NRC met with NAC to discuss submittal of a new NAC dry spent fuel storage cask design topical report;⁽¹²⁵⁾ the cask is to have a capacity of 31 PWR fuel assemblies and be designed to take allowance for burnup credit. At the meeting, structural aspects of the 31-assembly fuel basket design were discussed; those aspects may affect cask impact limiter design. On July 20, 1988, NAC submitted its "Topical Safety Analysis Report for the NAC S/T Cask Containing 31 Intact Fuel Assemblies."⁽¹²³⁾ The NRC docketed the topical report under Project M-52. NRC and NAC met on July 21, 1988, to discuss this submittal of a topical report for a new dry spent fuel storage cask design for use at an ISFSI.⁽¹²⁴⁾ The stainless steel and lead cask design is for storage of up to 31 Westinghouse PWR spent fuel assemblies. Maximum enrichment is limited to 1.9 wt% uranium-235 for storage of startup core fuel. NAC has a contract with Virginia Power for delivery of this type cask in August 1989. A license applicant could apply seeking burnup credit allowance for this cask design for fuel with a higher initial enrichment at some future date in a site-specific license application.⁽¹²⁴⁾ On December 14, 1988, NAC indicated they were most interested in obtaining NRC's review comments regarding taking credit in the safety analysis for the structural strength of the borated aluminum in the fuel basket of this NAC-31 S/T cask.⁽⁵³⁾ The borated aluminum uses enriched boron-10 to minimize the

structural effects of the 1.0 wt% boron-10 concentration. In January 1989, the NRC indicated⁽²⁾ that it is continuing its review of the topical report.

3.1.17 Nuclear Assurance Corporation (NAC): S/T (for 28 intact PWR fuel assemblies), Project M-54

NRC and NAC met on December 12, 1988, to discuss NAC's submittal of the "Topical Safety Analysis Report for the NAC Storage/Transport Cask Containing 28 Intact Fuel Assemblies for Use at an Independent Spent Fuel Storage Installation."⁽²⁹⁷⁾ NAC submitted the topical report on December 14, 1988.⁽⁵³⁾ The topical was docketed by the NRC under Project M-54. Virginia Power owns a cask of this type. On December 14, 1988, NAC requested that NRC review the Project M-54 application prior to the Project M-51 (NAC-31 S/T cask) application since the NAC-28 S/T cask is already at a customer's site.⁽⁵³⁾

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APPENDIX B

TABLE B.1 - QUESTIONS, REQUESTS, AND RESPONSES ASSOCIATED WITH THE
GENERAL NUCLEAR SYSTEMS, INC., CASTOR V/21 CASK INVOLVED IN THE
LICENSING OF THE INDEPENDENT SPENT FUEL STORAGE INSTALLATION AT SURRY

TABLE B.1. Questions, Requests, and Responses Associated with the General Nuclear Systems, Inc., CASTOR V/21 Cask Involved in the Licensing of the Independent Spent Fuel Storage Installation at Surry

Questions/Requests ^(a)	Responses ^(a)	Comments ^(a)
<p>QUESTION (September 30, 1985):⁽¹⁴⁾ The calculated cask dose rates reported in the TSAR should be increased to account for the effects of neutron subcritical multiplication.</p>	<p>RESPONSE: This transmittal⁽¹⁵⁾ provides you with our response to item 7.3.2, "Shielding", of the TSAR corrections listing provided in the NRC letter of September 30, 1985.⁽¹⁴⁾ The following discussion provides our assessment of this effect. In addition, a modification to the CASTOR V/21 gamma shielding is proposed based on dose rate measurements obtained during the demonstration test at INEL.</p> <p><u>Subcritical Neutron Multiplication</u></p> <p>The effect of neutron subcritical multiplication is not considered in our shielding calculations. This effect is significant only if fresh undepleted fuel is loaded into the cask together with spent fuel. For storage of typical 15 x 15 PWR fuel assemblies, the k_{eff} for fresh fuel at 3.7% U-235 in a helium environment will be about 0.4. In spent fuel the U-235 content is depleted to less than 1.0%. The k_{eff} will also be lower (in the range of 0.1), and the overall effect of the subcritical neutron multiplication in the case of spent fuel (about 10%) is covered by the conservatism of the calculation method used in our TSAR. This conclusion is further supported by the dose rate measurements at INEL. Prior to performance of the test, GNS/GNSI conducted a dose rate study that accounted for the source strength and characteristics (including power history and cooling time) of the actual fuel selected for the test. The shielding analysis was performed using the same methods described in Section 7 of the TSAR. The enclosed three figures⁽¹⁵⁾ demonstrate the good agreement between measured and calculated results. Therefore, it is our conclusion that the dose rates presently reported in the TSAR are acceptable.</p> <p><u>Gamma Shielding Design</u></p> <p>The dose rate measurements at INEL demonstrated that the CASTOR V/21 performed well within the design criteria of 200 mrem/h at the surface and 10 mrem/h at 2 meters. Figure 1 for the side wall</p>	<p>See submittal⁽¹⁵⁾ for figures.</p>

TABLE B.1. (contd)

Questions/Requests ^(a)	Responses ^(a)	Comments ^(a)
<u>QUESTION (September 30, 1985) (contd):</u>	<u>RESPONSE (contd):</u>	
	<p>indicates that a gamma peak was observed at the elevations corresponding to the fuel upper and lower end fittings. Even though the design requirements were satisfied, the following modification is proposed to reduce this local peak:</p> <ol style="list-style-type: none"> 1. The depth of the moderator bore holes will be reduced by 250 mm. 2. Steel plugs, 600-mm long, will be inserted at the cask bottom end of the borings instead of the polyethylene material. <p>These changes will result in a reduction by a factor of eight of the gamma peak in those regions. Since the neutron dose in these regions is very low, the effectiveness of the neutron shielding is not impaired. This shielding improvement affects only the local gamma peak and does not change average dose rate values presently reported in the TSAR (Table 7.3-4).</p> <p>This change will be appropriately documented in the forthcoming revision of the TSAR.</p>	
<p><u>REQUEST (October 14, 1985):</u>⁽⁶⁷⁾ The recent discovery of cracks in the fuel basket of the CASTOR V/21 cask used in the Virginia Electric and Power Company/Department of Energy (VEPCO/DOE) dry spent fuel storage cooperative demonstration program is of great interest to us. We recognize that these cracks were discovered after the cask had been involved in a demonstration under conditions exceeding the design basis set forth in your "Topical Safety Analysis Report For the CASTOR V/21 Cask Independent Spent Fuel Storage Installation (Dry Storage)," Revision 1. However, in view of these developments since our approving the topical report and considering potential impacts on the licensed use of the CASTOR V/21 cask, we request you provide us with an analysis and evaluation of the cause(s) of the cracking found in the demonstration cask. Also, please assess the need, if any, to revise your Topical Safety Analysis Report and provide any necessary changes along with the appropriate analysis. Because the use of a CASTOR V/21 cask at an</p>	<p><u>RESPONSE:</u> GNSI and VEPCO met with the NRC Staff on November 1, 1985,⁽²⁹⁸⁾ to discuss the status of damage analysis of cracks sustained in nonstructural welds in the CASTOR V/21 basket at an unlicensed demonstration at Idaho National Engineering Laboratory (INEL) under conditions exceeding the cask design basis.</p> <p>GNSI is conducting thermal stress analysis and other activities to resolve the cause of the cracks observed. At this time one explanation, yet to be confirmed, is that the combination of: 1) the basket fabrication for the INEL demonstration, which was deliberately made with less margin for expansion, i.e., a tighter fit for thermal efficiency; and 2) a demonstration heat load of about 28.4 kW, which exceeded the design basis of 21 kW, resulted in sufficient stress to cause cracks in</p>	<p>See NRC memorandum⁽²⁹⁸⁾ and the subsequent activities that are described below in the middle column.</p>

TABLE B.1. (contd)

Questions/Requests ^(a)	Responses ^(a)	Comments ^(a)
<p><u>REQUEST (October 14, 1985) (contd):</u></p> <p>independent spent fuel storage installation requires a site-specific safety evaluation and a license, we intend to hold in abeyance any supplemental action on our approval of your Topical Safety Analysis Report of the CASTOR V/21 cask pending receipt and review of your analysis.</p>	<p><u>RESPONSE (contd):</u></p> <p>nonstructural welds. It was stated that these cracks should not occur for a basket constructed to tolerances shown in the Topical Report for the CASTOR V/21 under the design heat load.</p> <p>In order to resolve questions raised by this occurrence, GNSI will submit a preliminary report in November 1985 detailing information on the cask baskets and analyses performed. Also information on similar CASTOR baskets routinely used in the FRG in Type B casks with respect to weld durability is expected to be included.</p> <p>At this time it appears that weld cracks were observed in only 4 locations adjacent to spent fuel assemblies generating about 1.8 kW per assembly heat load (design basis was 1.0 kW). Also, since these are nonstructural welds, it would appear basket safety is not compromised.</p> <p>After submittal of the preliminary GNSI report, another meeting of the NRC Staff with GNSI will be scheduled. Assuming progress is made, GNSI would subsequently submit a second and probably final report on this occurrence.</p>	
<p><u>REQUESTS/QUESTIONS (December 20, 1985):</u>⁽²⁹⁹⁾ The NRC met with GNSI on December 13, 1985, to resolve uncertainties, which were raised with respect to the CASTOR V/21 cask basket by the basket weld crack incident during the unlicensed cask demonstration at Idaho National Engineering Laboratory (INEL) and which were subsequently addressed in the GNSI preliminary report of November 1985 to the NRC. However, to be clear with respect to the determinations reached in the meeting, the NRC outlined the main points which were covered therein and to which GNSI is responding in its next communication to the NRC. Those points follow:</p> <ol style="list-style-type: none"> 1. With respect to resolving causes of the weld area failures, these should be related to strain mechanisms. Thus potential mechanisms for these failures should be explained and assessed with respect to INEL demonstration conditions and the 	<p><u>RESPONSE:</u> GNSI and VEPCO met with the NRC staff on February 24, 1985,⁽³⁰⁰⁾ to discuss resolution of questions (the points in the NRC letter of December 20, 1985)⁽²⁹⁹⁾ concerning the CASTOR V/21 cask basket originally raised in October 1985 during a Department of Energy spent fuel storage demonstration using a CASTOR V/21 cask at the Idaho National Engineering Laboratory. After discussion of these points and of new data that GNSI is obtaining by tests and analyses (see Enclosure 3 of the NRC memorandum⁽³⁰⁰⁾, a figure illustrating one weld test), a set of comments was drawn up to document how the points might be addressed to aid in their resolution. This set of comments was discussed, and a final version of it agreed on before the meeting concluded. The set of comments is listed below:</p>	<p>See NRC memorandum⁽³⁰⁰⁾ and the subsequent activities that are described below.</p>

TABLE B.1. (contd)

Questions/Requests ^(a)	Responses ^(a)	Comments ^(a)
<u>REQUESTS/QUESTIONS (December 20, 1985) (contd):</u>	<u>RESPONSE (contd):</u>	
<p>CASTOR V/21 cask used. This assessment should be made with reference to the thermal analysis provided in the TSAR. Furthermore, the thermal stress analysis should be revised to include the effect of gusset plates and correct material properties and also should be revised to include reporting the stresses at all points of the structure.</p>	<p><u>Point 1</u></p>	
<p>2. Measures with respect to basket fabrication and/or design, which will be taken to assure that such weld failures could not occur both under normal and accident design conditions, should be provided. Tolerances should be shown that control the maximum and minimum gap dimensions.</p>	<p>(a) <u>TSAR Thermal Analysis</u></p>	
	<p>The documents should show the details of the ANSYS finite element model both with and without gusset plates.</p>	
	<p>(b) <u>TSAR Thermal Analysis</u></p>	
	<p>For the joints, a description/drawing of the ANSYS finite element model should be included. If no detailed (2D) analysis was made, discuss how the stresses at the joints were calculated.</p>	
	<p>(c) <u>TSAR Thermal Analysis</u></p>	
	<p>Discuss for the joints how the beam model-to-plate model interface was made.</p>	
	<p>(d) <u>TSAR Thermal Analysis</u></p>	
	<p>Tabulate and locate stresses for all joints, highest stresses in plates, and highest stresses in ring.</p>	
	<p>(e) <u>TSAR Thermal Analysis</u></p>	
	<p>Include gusset plate temperature distribution.</p>	
	<p>(f) <u>TSAR Thermal Analysis</u></p>	
	<p>Present and define the acceptance criteria based on strain. Show that all calculated stresses meet the acceptance criteria. Will probably be in the table noted in item (d)</p>	
	<p>(g) <u>INEL Report</u></p>	
	<p>Develop failure analyses and criteria based on strain or displacement for J1, J2 and show how failure is related to acceptance criteria. Acceptance criteria includes ASME code case for low ductility austenitic steels.</p>	

TABLE B.1. (contd)

Questions/Requests ^(a)	Responses ^(a)	Comments ^(a)
<u>REQUESTS/QUESTIONS (December 20, 1985) (contd):</u>	<u>RESPONSE (contd):</u>	
	<u>Point 2</u>	
5. Material properties which reflect actual testing done on these materials should be included and their use carefully explained in terms of analyses supplied to us and design commitments for materials with respect to the Topical Report for the CASTOR V/21 cask, for example, values for Radionox (boronated stainless steel). Particular attention should be directed toward demonstrating that either brittle fracture is not a failure mode or if it is, then safety margins against brittle fracture are adequate under dynamic loading conditions.	(a) <u>TSAR Thermal Analysis</u> Present discussion on how the G1, G2 gaps are assured by measurement and note tolerances.	
6. Data on weld tests performed should be supplied distinguishing between welds which have a safety function, that is, which are relied upon to maintain the structural integrity of the basket and nonstructural welds.	(b) <u>TSAR Thermal Analysis</u> Present data on basket ring and cask cavity diametrical measurements.	
7. As discussed in our meeting, based on INEL measurements and your analysis, explain why there is no effect observed due to subcritical multiplication with respect to dose calculations (Reference: Correction 7.3.2 in our letter of approval for the CASTOR V/21 Topical Report). In relation to this explanation, provide drawings with changes related to cask borehole length shortening and the addition of steel plugs at the bottom of boreholes to provide additional reduction of gamma dose rates at positions near the cask wall top and bottom. Also provide an explanation of this improvement to cask shielding design. These changes are to be reflected in cask fabrication procedures and documentation.	(c) <u>TSAR Thermal Analysis</u> Present analysis to show the effect of noncentered basket. <u>Point 3</u> (a) <u>TSAR General</u> (Submit to NRC; send as soon as possible) The final design drawing detailing how the basket will be built. Note the weld lengths and location for SN-B, and all welds, gussets, spacers. (b) <u>TSAR General</u> Correct Figure 1 in Appendix 1.	
	<u>Point 4</u>	
	(a) <u>TSAR General</u> Include the statement that no welds fail during any TSAR conditions. (b) <u>TSAR Dynamic Drop</u> Correct the space between welds in TSAR due to error (see SER comments).	

TABLE B.1. (contd)

Questions/Requests ^(a)	Responses ^(a)	Comments ^(a)
<u>REQUESTS/QUESTIONS (December 20, 1985) (contd):</u>	<p><u>RESPONSE (contd):</u></p> <p>(c) <u>TSAR Dynamic Drop</u></p> <p>Detailed structural analysis of the cask basket under both normal and accident design conditions alluded to in Section 4.0 of your preliminary report should be provided to demonstrate the structural integrity of the basket. If measures referred to in point 2 above are taken, it should also be made clear whether this analysis assumes they are implemented.</p> <p><u>Point 5</u></p> <p>(a) <u>TSAR Material Report Addendum</u></p> <p>GNS will report on results of compact tension fracture toughness tests (per ASTM E-399). Tests will be performed with 20-mm plate at 350°C looking at base material, weld material, and heat-affected zone. Number of tests is 3.</p> <p>(b) <u>TSAR Material Specification</u></p> <p>Radionox material specification will define Charpy-V notch values at 350°C. The lowest of 3 samples will equal or exceed 15 ft-lb. Reference will be made to the DIN standards 50115 and 50122.</p> <p>(c) <u>TSAR Material Specification</u></p> <p>Material specification will delineate minimum elongations at 350°C as follows:</p> <p>Base Material - 15% Weld Material - 10%</p> <p>Reference will be made to the DIN standards 50145 and 50120</p>	

TABLE B.1. (contd)

Questions/Requests ^(a)	Responses ^(a)	Comments ^(a)
<u>REQUESTS/QUESTIONS (December 20, 1985) (contd):</u>	<u>RESPONSE (contd):</u>	
	(d) <u>ISAR Dynamic Drop</u>	
	A discussion of the applicability of the basket material should be made, with particular attention directed toward demonstrating that either brittle fracture is not a failure mode or if it is, that safety margins against brittle fracture are adequate under dynamic loading conditions.	
	(e) <u>ISAR Material Report Addendum</u>	
	A report will be prepared defining the results of tests performed to date including C_y , yield tests, and elongations. The report will include relevant details on such factors as how specimens were prepared, extensometer readings, etc. Information will be on both base metal and weld metal.	
	(f) <u>ISAR Material Specifications</u>	
	The following material properties will be defined as a function of the temperature range (20-400°C):	
	<ul style="list-style-type: none"> • Young's Modulus • Expansion Coefficient • Yield Strength • Ultimate Strength • Poisson Ratio (only at room temperature) 	
	All references will be provided.	
	<u>Point 6</u>	
	Questions in this area were covered in Points 1-5, above.	
	It was proposed by VEPCO that, as an alternative to assure that VEPCO's license review would not be held up by resolution of the borated stainless steel basket matter, VEPCO could use for near-term storage, if needed, a basket made of unborated stainless steel. This would eliminate basket structural concerns. A criticality analysis could	

TABLE B.1. (contd)

Questions/Requests ^(a)	Responses ^(a)	Comments ^(a)
<u>REQUESTS/QUESTIONS (December 20, 1985) (contd):</u>	<u>RESPONSE (contd):</u>	
<p>REQUEST (April 30, 1986):⁽⁴³⁾ Questions regarding the basket design and materials remain to be fully resolved. While continuing its tests and analyses of the original basket design, GNSI submitted by letter dated February 28, 1986, an alternative design for a cask fuel basket which does not employ borated stainless steel as a basket material.</p>	<p>be submitted by GNSI for its topical report for a lower initial fuel enrichment, and this could be reviewed by NRC. There was no objection to this approach by NRC staff. Resolution of the borated stainless steel basket matter would continue to proceed concurrently with submittals by GNSI.</p>	
<p>NRC staff has reviewed this alternative basket design, as subsequently revised in GNSI letters dated March 25, 1986, and April 23, 1986. The NRC analyses of the CASTOR V/21 all-stainless-steel basket indicate the following:</p>	<p>RESPONSE: Since October 1985, the NRC has received information and analyses from GNSI regarding the reported basket cracks and data and information characterizing the borated stainless steel (Radionox A 18) with nominal 1.03 weight percent boron to be used in the revised basket design submitted in GNSI letter of September 30, 1986. GNSI has carried out an engineering evaluation and testing program to validate the use of borated stainless steel in its basket design for the CASTOR V/21. By letter dated February 5, 1987, GNSI submitted its results to the NRC in a report entitled, "CASTOR V/21 Borated Stainless Steel Basket Evaluation, January 1987," which included a material evaluation report for the material Radionox A 18, a revision for the CASTOR V/21 topical report, and an evaluation of the basket weld area crack incident at Idaho National Engineering Laboratory (INEL). The NRC concluded⁽³⁰¹⁾ that the INEL incident has been satisfactorily explained and that the steps taken in the design and fabrication of the revised fuel basket using Radionox A 18 to prevent a recurrence are adequate.</p>	See NRC letters. (34,301)
<ol style="list-style-type: none"> 1. For spent fuel of an initial enrichment of ≤ 2.2 wt% U-235, k_{eff} is found to be ≤ 0.95, which is an acceptable value for meeting the subcriticality requirements of 10 CFR Section 72.73. Our conclusion is based on independent staff analysis using the KENO V program. The applicant's analysis also results in a $k_{eff} \leq 0.95$. 2. The staff has also performed independent structural analyses for the fuel basket design, which, as noted above, has been revised from that originally presented and analyzed in the TSAR (i.e., there have been changes in addition to excluding the use of borated stainless steel as a basket material). Our analyses indicate the following: <ol style="list-style-type: none"> a. An elastic analysis of the basket for thermal stresses shows all stresses to be within acceptable limits for unborated stainless steel. b. A dynamic elastic analysis based upon a 150-g load shows that the basket will maintain fuel assembly criticality safety provided the 20-mm plates at the corner junctions are connected by 		

TABLE B.1. (contd)

Questions/Requests (a)	Responses (a)	Comments (a)
<u>REQUEST (April 30, 1986) (contd):</u>		See NRC Letters. (34,301)
<p>full length, full penetration welds. This method of construction is preferred over the alternative featuring the bolted construction. Furthermore, the analysis shows that the 5-mm gusset plates will buckle at about 110 g's. Consequently, gusset thicknesses should be increased to 10-mm thick.</p>		
<p>The changes in item 2.b above and other changes to the drawing relating to the deletion of items 37, 38, and 39, the flux trap spacer dimensions, and deletion of the reference to RADIONOX material, as indicated in the GNSI letter of April 23, 1986, should be reflected in the corrected TSAR. With the changes discussed above, the all-stainless-steel fuel basket design, when used with the CASTOR V/21 cask design, meets the structural requirements for normal, off-normal and accident conditions as required under 10 CFR Section 72.72.</p>		
<p><u>REQUEST (April 3, 1987):</u>(301) Based on the NRC staff's previous review and analyses, as documented in the September 30, 1985, letter of approval with enclosed Safety Evaluation Report, and the staff's supplemental review and analyses as documented in NRC's April 30, 1986, letter of approval for an all-stainless-steel basket, the staff concludes that, with the revisions stated in the GNSI letter dated February 7, 1987, the borated stainless steel basket design submitted by GNSI in its September 30, 1986, letter meets the requirements of 10 CFR Part 72. Further, with this basket design the CASTOR V/21 nodular cast iron cask design, as documented in the TSAR and approved in NRC's letter of September 30, 1985, acceptably meets the requirements of 10 CFR Part 72, provided that initial fuel enrichment is restricted to ≤ 3.5 wt% U-235 for spent fuel to be stored in the cask.</p>		
<p>It is requested that General Nuclear Systems, Inc. publish an approved version of this report, with proprietary information in a separate binder, as per Item 3, "Proprietary Information," of the Introduction to Regulatory Guide 3.48, and submit 10 copies for docketing within 90 days of the receipt of this letter.</p>		

TABLE B.1. (contd)

Questions/Requests ^(a)	Responses ^(a)	Comments ^(a)
<u>REQUEST (April 3, 1987) (contd):</u>		
<p>In the approved version, there are items relating to the CASTOR V/21 Topical Safety Analysis Report, Revision 1, which need to be clarified or corrected (see our September 30, 1985, letter to you, enclosure 3, TSAR Corrections, our April 30, 1986, letter to you and the enclosure of this letter). Also, an analysis for cask end on drop with impact limiter was performed by the staff and the cask design was found acceptable (see Enclosure 2, Safety Evaluation Report, page 21, of our September 30, 1985, letter to GNSI), but no such analysis has been included by GNSI in the TSAR. For completeness, it should be provided. While these items do not affect our assessment of safety and compliance with the requirements of 10 CFR Part 72 as reflected by our safety evaluations, they would be incorporated in the TSAR.</p>		
<p>This revision is also to incorporate this letter and the April 30, 1986, and September 30, 1985, letters of approval, following the title page. It should also include a listing identifying with submittal dates supporting supplemental information submitted after the TSAR, i.e., Revision 1 and docketed under Project M-37. The report identification of the approved report is to have an "A" suffix.</p>		
<p>(a) Abbreviations: Independent Spent Fuel Storage Installation (ISFSI); Topical Safety Analysis Report (TSAR); General Nuclear Systems, Inc. (GNSI); Virginia Power (VP) or Virginia Electric and Power Company (VEPCO); and Federal Republic of Germany (FRG).</p>		

B.10

APPENDIX C

CHRONOLOGICAL LISTING OF LICENSING ACTIVITIES FOR
LICENSE APPLICATIONS FOR DRY STORAGE OF SPENT FUEL
AT INDEPENDENT SPENT FUEL STORAGE INSTALLATIONS

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CHRONOLOGICAL LISTING OF LICENSING ACTIVITIES FOR LICENSE APPLICATIONS FOR DRY STORAGE OF SPENT FUEL AT INDEPENDENT SPENT FUEL STORAGE INSTALLATIONS

3.2.1.1(a) Virginia Power Company (Surry)

1982: VEPCO submitted the LA, SAR, and ER for the dry cask ISFI at the Surry Power Station in October 1982.⁽¹³²⁾ The NRC received the information on October 13, 1982; the application was assigned Docket No. 72-2 on October 15, 1982.⁽⁹⁰⁾ The documents supporting the LA⁽¹³²⁾ indicate that reports were to be submitted to the NRC by the cask manufacturers that describe the casks' design and safety features. These reports were to be prepared by Ridihaigh, Eggers and Associates (REA), Gesellschaft Fuer Nuklear-Service mbH (GNS), and Combustion Engineering (C-E) and are referenced in the documents. The NRC stated in its letter⁽⁹⁰⁾ of December 3, 1982, that details of the cask(s) must be provided before licensing action can be completed.

1983: In VEPCO's letter of January 3, 1983,⁽¹¹⁾ it is noted that VEPCO inadvertently included safeguards information in the license application. VEPCO's letter of January 25, 1983,⁽³⁰²⁾ enclosed a supplement to the SAR, which reflected the actual submittal of a C-E report, CENPD-270, that describes their design for dry storage casks for use at the Surry ISFSI. The NRC's letter of March 16, 1983,⁽²⁵⁹⁾ indicates that they were unable to locate one page of the supplement⁽³⁰²⁾ to the SAR and requested that page. The NRC also stated in the letter⁽²⁵⁹⁾ that it would be helpful if each supplement were 1) dated and identified by a supplement number; 2) accompanied by a dated and number supplement index listing the pages to be removed and the pages to be inserted; and 3) identified by number and/or date on each new page containing new material and the newest material on a page be identified by a vertical margin bar. Enclosed with the NRC's letter⁽²⁵⁹⁾ was a copy of the initial set of topical report review comments from NRC to C-E (there were 34 pages of comments; the general comment from that letter is

(a) Part numbers correspond to the subsections in Section 3.2.

included in Section 3.2.7 of this report). In the letter of April 8, 1983,⁽¹²⁾ the NRC states that they completed their initial review of the licensing application but, because of the lack of specifics in VEPCO's initial application, the NRC requested that VEPCO completely resubmit the required safeguards plan. VEPCO resubmitted the required safeguards plan: the security plan was submitted in their letter of July 29, 1983,⁽³⁰³⁾ and the nuclear security personnel training and qualification program sent in their letter of August 19, 1983.⁽³⁰⁴⁾ The NRC indicated⁽¹⁶⁾ that it was not necessary to wait to respond to all the questions in one supplement to the application, but rather, to make as many individual supplements as is convenient; also, the information in the ER and SAR text should be updated to match the responses whenever a specific topic is fully developed, or sooner if the applicant desires. It is noted in VEPCO's letter of October 25, 1983,⁽³⁰⁵⁾ that an incomplete set of revised pages was forwarded to the NRC in an earlier submittal⁽³⁰²⁾ (letter dated January 25, 1983, concerning the addition of the C-E Topical Report CENPD-270 to the ISFSI SAR). The VEPCO letter⁽³⁰⁵⁾ also added the REA Topical Report No. 2023/PI to the ISFSI SAR.

1984: VEPCO's letter of March 2, 1984,⁽³⁰⁶⁾ provided responses to 14 of NRC's questions. The NRC's letter of March 30, 1984,⁽³⁰⁷⁾ confirms discussions with VEPCO regarding physical protection requirements for dry cask storage of spent fuel, which was conducted at Surry on February 10, 1984. The letter⁽³⁰⁷⁾ identifies two issues that were beyond the scope of commitments contained in the proposed security plan: the two issues (withheld from public disclosure because they involve "Safeguards Information") were essential elements of an effective physical protection program and must be incorporated into the security system for the ISFSI and appropriately committed to in the security plan for the facility. VEPCO's letter of June 20, 1984⁽³⁰⁸⁾ included responses to five of the NRC's questions. VEPCO's letter of June 25, 1984,⁽¹⁷⁾ provided responses to three of the NRC's questions,⁽¹⁶⁾ which concerned seismological, geological, and structural bases for the Surry dry cask ISFSI. Responses to 17 of NRC's questions⁽¹⁶⁾ were provided in VEPCO's letter of September 21, 1984.⁽³⁰⁹⁾ The NRC requested more detailed information regarding the soil liquefaction analysis for the Surry ISFSI in their letter of October 1, 1984.⁽¹⁸⁾ VEPCO's letter

of October 24, 1984,⁽³¹⁰⁾ provided responses to five of the NRC's questions.⁽¹⁶⁾ Additional information (involving seven questions/requests) was requested by the NRC in their letter of November 14, 1984.⁽³¹¹⁾ VEPCO's letter of November 20, 1984,⁽³¹²⁾ provided a response to an NRC question.⁽¹⁸⁾ VEPCO's letter of December 4, 1984,⁽³¹³⁾ provided responses to two of NRC's questions.⁽¹⁶⁾ The response to one of NRC's questions⁽¹⁶⁾ was provided in VEPCO's letter of December 10, 1984.⁽³¹⁴⁾

1985: The Topical Safety Analysis Report for the CASTOR V/21 Cask Independent Spent Fuel Storage Installation (Dry Storage) was received by the NRC on January 22, 1985.⁽³¹⁵⁾ VEPCO's letter of February 8, 1985,⁽³¹⁶⁾ provided responses to seven of the NRC's questions.⁽³¹¹⁾ An NRC memorandum dated March 7, 1985,⁽¹⁹⁾ indicates that Surry ISFSI SAR Table 7.3-2, "Neutron Response Functions and Cask Surface Fluxes," references cask neutron surface flux leakage spectra in the CASTOR V/21 Topical Report⁽³¹⁵⁾ that do not exist. Apparently GNSI supplied this information to VEPCO before the revised Topical Report was submitted to the NRC and then inadvertently omitted it from the Topical Report. Also it is noted in the NRC memorandum⁽¹⁹⁾ (concerning clarification of dose calculations submitted in VEPCO's letter⁽³¹⁶⁾ that Surry ISFSI ER Figure 3.5-3, "Normalized Surface Dose Rate On Cask Versus Age of Spent Fuel," does not reflect the GNSI CASTOR V/21 Topical Report data and that the NRC requested that this figure be revised and incorporated in the Surry ISFSI Final Safety Analysis Report (FSAR), when submitted. In its letter of March 15, 1985,⁽³¹⁷⁾ Virginia Electric and Power Company indicated a change in its business name: the utility requested that the company be listed as "Virginia Power." The NRC memorandum of March 27, 1985,⁽³¹⁸⁾ provided a summary of the meeting with Virginia Power on March 20, 1985, regarding the Surry ISFSI. VP's letter of April 10, 1985,⁽³¹⁹⁾ involved a request to initiate ISFSI construction. On April 12, 1985, the NRC issued⁽³²⁰⁾ an environmental assessment (with a finding of no significant impact) in connection with the ISFSI application. The NRC memorandum of April 15, 1985,⁽³²¹⁾ enclosed copies of the Federal Register Notice and the Environmental Assessment concerning the Surry ISFSI to the Office of Congressional Affairs. The NRC letter of April 23, 1985,⁽³²²⁾ requested drawings and a description of the upgraded concrete pad specifications. VP's letter

of May 9, 1985,⁽³²³⁾ provided a response to the NRC's request.⁽³²²⁾ The NRC letter of June 10, 1985,⁽³²⁴⁾ indicates that proceeding with pad construction would be at the risk of Virginia Electric and Power Company, and that construction of the pad and installation of the security equipment should be appropriately documented to confirm that they are constructed as described in the application. VP's letter of June 10, 1985,⁽³²⁵⁾ enclosed a revised Surry ISFSI license application and supporting documents. The NRC memorandum of July 18, 1985,⁽³²⁶⁾ provides information from the meeting between VP and NRC staff on July 10, 1985, to discuss proposed license conditions and preoperational testing for the Surry ISFSI (one of the enclosures was the technical and operations manual for the CASTOR V/21 cask). The NRC memorandum of July 22, 1985,⁽³²⁷⁾ contains information about a meeting between VP and the NRC staff on July 18, 1985, to discuss proposed license conditions and technical specifications for VP's proposed storage of spent fuel in an ISFSI at its Surry Power Station site. The NRC indicated, that with some modifications and additions, the set of conditions and specifications presented provided a generally acceptable basis for development of final license conditions and technical specifications. The NRC memorandum of August 5, 1985,⁽¹³⁾ provides information about a meeting of NRC staff with VEPCO and a visit to the Surry Power Station site on July 30, 1985. The purpose of the meeting was to discuss proposed license conditions and technical conditions and to discuss the following NRC question areas: station health physics procedures, station administrative procedures, station training program, cask handling procedures to prevent a cask from tipping over, and the cask pressure switch monitor. The purpose of the visit was to examine the path a spent fuel storage cask would travel to emplacement at the proposed Surry ISFSI. The NRC letter of September 30, 1985,⁽¹⁴⁾ concerns NRC's acceptance as a reference of "Topical Safety Analysis Report for the CASTOR V/21 Cask Independent Spent Fuel Storage Installation (Dry Storage)," Revision 1. The NRC letter of October 14, 1985,⁽⁶⁷⁾ to GNSI and the NRC letter of October 15, 1985,⁽⁶⁸⁾ to VEPCO concern the recent discovery of cracks in the fuel basket of the CASTOR V/21 cask used in the VEPCO/DOE dry storage cooperative demonstration program and the requests from NRC for information on that matter. The NRC memorandum of November 6, 1985,⁽²⁹⁸⁾ provides information about the

meeting of NRC staff with GNSI and VEPCO on November 1, 1985, to discuss status of damage analysis of cracks sustained in nonstructural welds in the CASTOR V/21 basket at an unlicensed demonstration at Idaho National Engineering Laboratory (INEL) under conditions exceeding the cask design basis. The GNSI letter of January 2, 1985,⁽¹⁵⁾ provides a response to the NRC's request⁽¹⁴⁾ that calculated cask dose rates reported in the TSAR should be increased to account for the effects of neutron subcritical multiplication.

1986: The NRC note of February 6, 1986,⁽³²⁸⁾ to file pertains to a statement that VEPCO proposed to add to Section 10 of the SAR; the NRC indicates that the statement appears adequate. The VEPCO letter of February 19, 1986,⁽³²⁹⁾ was the submittal to NRC of Amendment 3 to the SAR for the Surry ISFSI. The NRC memorandum of February 27, 1986,⁽³⁰⁰⁾ and the NRC letters of February 28, 1986,^(330,331) concerned the meeting of NRC staff with VEPCO and GNSI on February 24, 1985, to discuss resolution of questions concerning the CASTOR V/21 cask basket originally raised in October 1985 during a Department of Energy spent fuel storage demonstration using a CASTOR V/21 cask at the Idaho National Engineering Laboratory. Included in the NRC's letter of February 27, 1986,⁽³²⁹⁾ was a statement that it was proposed by VEPCO that, as an alternative to assure that VEPCO's license review would not be held up by resolution of the borated stainless steel basket matter, VEPCO could use for near-term storage, if needed, a basket made of unborated stainless steel. This would eliminate basket structural concerns. A criticality analysis could be submitted by GNSI for its topical report for a lower initial fuel enrichment, and this could be reviewed by NRC. There was no objection to this approach by NRC staff. Resolution of the borated stainless steel basket matter would continue to proceed concurrently with submittals by GNSI. The VEPCO letter of March 25, 1986,⁽³³²⁾ to the NRC indicates that VEPCO concurs with GNSI's request to include the optional all stainless steel basket design in the CASTOR V/21 TSAR. The NRC letter of April 30, 1986,⁽⁴³⁾ pertains to the NRC's acceptance as a reference of "Topical Safety Analysis Report for the CASTOR V/21 Cask Independent Spent Fuel Storage Installation (Dry Storage)," Revision 1, with alternative fuel basket design; it also includes some NRC requests. The NRC letter of July 2, 1986,⁽¹³³⁾ enclosed Materials License No. SNM-2501, which authorizes VEPCO for a 20-year term to

receive, possess, store, and transfer spent fuel from Surry-1 and -2 in an ISFSI located at VEPCO's Surry Power Station site. Enclosed with the Materials License was the NRC's Environmental Assessment and Finding of No Significant Impact and the NRC's Safety Evaluation Report for the Surry ISFSI. The VEPCO letter of October 10, 1986,⁽³³³⁾ proposed a change in the Technical Specification for the Surry ISFSI. The proposed change corrects the description of the operating characteristics of the storage cask interlid pressure switch and specifies that the switch is calibrated at standard conditions. The NRC letter of October 17, 1986,⁽³³⁴⁾ granted VEPCO temporary technical specification relief; the temporary waiver pertains to the proposed change⁽³³³⁾ clarifying the setpoint and calibration conditions for the storage cask interlid pressure switch and permits use of the ISFSI facility in support of the current Surry-2 refueling outage. The NRC letter of November 10, 1986,⁽³³⁵⁾ to VEPCO concerns the NRC's issuance of Amendment No. 1 to Materials License No. SNM-2501 for the Surry ISFSI.

1987: The NRC memorandum of March 6, 1987,⁽³³⁶⁾ provides information on a meeting of NRC staff with VEPCO on March 5, 1987, to discuss VEPCO's dry storage activities. VEPCO's plans include the use of one or two other cask types than the CASTOR V/21. The NRC letter of April 3, 1987,⁽⁴²⁾ informed VEPCO that the NRC had completed its safety review of the GNSI Topical Report for the CASTOR V/21 nodular cast iron cask design incorporating a borated stainless steel basket for storage of spent fuel with an enrichment of ≤ 3.5 wt% uranium-235 and determined that this design meets the requirements of 10 CFR Part 72 and is acceptable. The enclosure with that letter⁽⁴²⁾ is the NRC letter of April 3, 1987,⁽³⁰¹⁾ to GNSI on the same subject; however, the latter letter also contains a few requests from the NRC. The NRC letter of May 1, 1987,⁽³³⁷⁾ to GNSI approves GNSI's request to withhold some specific information (i.e., certain appendices and cask basket design information) from public disclosure. The NRC memorandum of October 17, 1987,⁽¹²²⁾ provides information about a meeting of the NRC staff with GNSI and VEPCO on October 7, 1987, to discuss the submittal of a topical report by GNSI for the design of the CASTOR X dry spent fuel storage cask, and an associated submittal by VEPCO of a license amendment for its Surry ISFSI. The NRC letter of October 29, 1987,⁽³³⁸⁾ to GNSI approves GNSI's request to withhold some

specific information (i.e., certain cask basket information, TSAR pages, and TSAR Appendices. The VEPCO letter of November 16, 1987,⁽³³⁹⁾ to the NRC proposed Surry ISFSI Technical Specification changes involving the environmental monitoring program and the semi-annual report. The VEPCO letter of December 30, 1987,⁽³⁴⁰⁾ transmitted the first annual update to the SAR for the Surry ISFSI. The VEPCO letter of December 31, 1987,⁽³⁴¹⁾ to the NRC confirms the conversations between the NRC staff and VEPCO at the meeting on October 7, 1987, concerning VEPCO's intent to submit a license amendment and proposed technical specification change under 10 CFR 72 that would allow the use of storage casks at the Surry ISFSI that utilize burnup credit for criticality analyses. In the letter,⁽³⁴¹⁾ VEPCO also indicates that GNSI intends to submit a TSAR for an enhanced capacity storage cask denoted as the CASTOR X/28-33 (it will employ burnup credit in the criticality design of its basket for the 33-element cask). VEPCO hoped that the NRC could complete its review of the TSAR and license amendment by January 1989 to allow use of a CASTOR X/28-33 in FY 1989 under the DOE/VP/EPRI Dry Storage Cask Demonstration Program.

1988: The NRC letter of January 27, 1988⁽³⁴²⁾ to VEPCO approves the proposed change involving the semi-annual report in the Surry ISFSI Technical Specifications. The NRC memorandum of January 27, 1988,⁽¹¹⁰⁾ provides information about a meeting of the NRC staff with Nuclear Assurance Corporation (NAC) to discuss the submittal of a new topical report for a dry spent fuel storage cask design for storage of consolidated spent fuel rods. The memorandum⁽¹¹⁰⁾ indicates that technical issues involving criticality analysis, thermal analysis, shielding and dose rates were discussed. The design is associated with a 10 CFR Part 72 license amendment application planned by VP for its Surry ISFSI. The NRC memorandum of March 1, 1988,⁽²⁹⁴⁾ provides information about a meeting on February 25, 1988, of the NRC staff, VP, and GNSI to discuss the submittal of the GNSI CASTOR X/28-33 topical report and of the VP amendment application for use of the CASTOR X/28-33 at its Surry dry storage site. The 33-assembly basket is actually designed to allow for burnup credit. The memorandum indicated that the NRC anticipated

that GNSI would submit the topical report in April 1988. VP was to do the burnup credit analysis separately for its license amendment application submittal in May 1988.

3.2.1.2 Carolina Power and Light Company (Robinson-2)

1985: CP&L submitted the LA (included the ER as Chapter 12) and SAR in February 1985.⁽¹³⁵⁾ NRC's letter of April 15, 1985,⁽³⁴³⁾ indicates that a Notice of Consideration of a Materials License for the Storage of Spent Fuel in a dry storage concrete module system at Robinson-2 was transmitted for publication in the Federal Register. The NRC memorandum of May 2, 1985,⁽²¹³⁾ provides information on a meeting of the NRC staff with NUTECH on April 17-18, 1985, to discuss NRC staff comments on the "Topical Report for the NUTECH Horizontal Modular System for Irradiated Fuel" (NUH-001, Revision 0). The NRC letter of May 3, 1985,⁽²⁹⁾ informed CP&L that CP&L's submittal⁽¹³⁵⁾ of a License Application for an ISFSI does not comply with 10 CFR 72.20 (the ER does not fully meet some of the requirements). The letter⁽²⁹⁾ indicates that the ER should particularly include a statement of the purpose, or need, for the proposed action and a discussion of alternatives; also, the ER should address the siting evaluation factors contained in 10 CFR 72 (specifically 72.61 through 72.70). The NRC memorandum of June 7, 1985,⁽³⁴⁴⁾ provides information about a meeting of NRC staff with CP&L on May 30, 1985, to discuss CP&L's application for dry storage of spent fuel in an ISFSI at its H. B. Robinson Unit 2, (Robinson-2) nuclear power plant site; CP&L revealed that it intends to make major changes in the design of the dry shielded canister (DSC) to be emplaced in the concrete horizontal storage module (HSM). The NRC letter of June 28, 1985,⁽³⁰⁾ to CP&L provided a list of environmental questions (intended only as a guide) that might be helpful in preparing the ER. The NRC memorandum of July 26, 1985,⁽³⁴⁵⁾ provides information on a meeting of NRC staff with CP&L on July 17, 1985, to discuss CP&L's application for storage of spent fuel in an ISFSI at Robinson-2. The NRC memorandum of September 10, 1985,⁽³⁴⁶⁾ provides information about an environmental site visit by the NRC to the proposed site of the ISFSI at Robinson-2. The CP&L letter of October 14, 1985,⁽³⁴⁷⁾ provides a response to Question 1 in the NRC comments of July 12, 1985, on the Robinson-2 ISFSI license application. The NRC letter of September 20, 1985,⁽¹⁰⁾ to CP&L contained several

requests. The NRC requested information on the onsite collective dose commitment to workers constructing the five additional modules adjacent to the original three filled modules and to personnel gathering data during the first year demonstration period. The NRC requested that CP&L provide this information in the same format as Tables 7.4-1 and 7.4-2 of Robinson-2 ISFSI Safety Analysis Report. Also, the NRC indicated that a detailed ISFSI site plan drawing and aerial photographs would be most helpful to its review. The NRC letter of October 21, 1985,⁽²¹⁵⁾ to CP&L pertained to a missing drawing and an extra copy of another drawing in CP&L's submittal of October 14, 1985.⁽³⁴⁷⁾ The CP&L letter of October 22, 1985,⁽²¹⁶⁾ to NRC enclosed a drawing (it was incorrectly labeled in an earlier transmittal) that NRC requested. The CP&L letter of October 25, 1985,⁽³⁴⁸⁾ to NRC enclosed the site plans requested by NRC;⁽¹⁰⁾ the letter also indicated that the aerial photographs have been submitted separately. The CP&L letter of October 30, 1985,⁽²¹¹⁾ transmitted to the NRC the responses to 93 of NRC's questions in their letter of July 12, 1985. The CP&L letter of November 4, 1985,⁽²¹⁷⁾ to NRC provided a response to NRC's request of September 20, 1985.⁽¹⁰⁾ The CP&L letter of December 9, 1985,⁽²¹⁸⁾ to NRC enclosed drawings requested by the NRC at a meeting on November 18, 1985. The NRC memorandum of November 20, 1985,⁽²¹⁹⁾ provides information about a meeting of NRC staff with CP&L to discuss CP&L responses (docketed November 5, 1985, under Docket No. 72-3) to NRC comments; the memorandum indicated that, as of that time, the NRC staff could not determine if their comments had been resolved, though progress had been made.

1986: The CP&L letter of January 13, 1986,⁽³⁴⁹⁾ to NRC enclosed additional information (i.e., specification packages) in support of the ISFSI license application. The CP&L letter of January 30, 1986,⁽³⁵⁰⁾ to NRC enclosed additional information (drawings) in support of the ISFSI. The CP&L letter of February 4, 1986,⁽³⁵¹⁾ to NRC provided additional information (site plan and fabrication specification for the Horizontal Storage Module) in support of the ISFSI license application. The CP&L letter of February 13, 1986,⁽³⁵²⁾ to NRC enclosed additional information (computer input and output for 30-foot drop analysis and for foundation analysis) in support of the ISFSI license request. The CP&L letter of February 17, 1986,⁽²¹⁴⁾ to NRC

provides additional information on certain enhancements to CP&L's earlier responses⁽²¹¹⁾ (specifically involved were Questions 3, 12, 14, 26, 34, 37, 61, 62, 70, 72, 74, 76, 78, 82, and 84). The CP&L letter of February 25, 1986,⁽²²⁰⁾ to the NRC provides additional information and clarification of CP&L's response to Question 39. The CP&L letter of March 11, 1986,⁽³⁵³⁾ to NRC concerns the schedule for construction of the ISFSI. The CP&L telecopy of March 14, 1986,⁽¹³⁶⁾ to NRC contains information on the dry shielded canister (DSC) enveloping load combination and the maximum DSC stresses for an 8-ft bottom end drop accident. The NRC memorandum of March 24, 1986,⁽³⁵⁴⁾ provides information on a meeting of NRC staff with CP&L on March 19, 1986, to discuss proposed license technical specifications. The NRC letter of March 31, 1986,⁽³⁵⁵⁾ to CP&L indicates that the NRC has issued an environmental assessment in connection with CP&L's application⁽¹³⁵⁾ for authority to construct and operate a dry storage ISFSI at Robinson-2. The NRC memorandum of April 3, 1986,⁽³⁵⁶⁾ provides information on the proposed Technical Specifications for the CP&L ISFSI. The CP&L telecopy of April 3, 1986,⁽¹³⁷⁾ to NRC provides responses to two NRC questions: one question involved assumptions on fuel enrichment and burnup for ORIGEN-2 analyses and the other involved the lubricant to be used on the DSC. The CP&L letter of April 3, 1986,⁽²²¹⁾ to NRC encloses revised security documentation in response to the NRC letter of May 28, 1985. The NUTECH letter of April 11, 1986,⁽¹⁴²⁾ to NRC encloses a description of the NUTECH analysis of the thermal load on the concrete foundation, as requested by the NRC to expedite the review. The NRC memorandum of April 21, 1986,⁽⁸⁾ provides information on an NRC site visit to CP&L's Robinson-2 to examine the site proposed for the CP&L concrete module and stainless steel canister storage system ISFSI. The memorandum⁽⁸⁾ indicates that the outstanding issues remaining to be closed by submittals from CP&L include final quality assurance commitments and completion of CP&L's internal plant review to confirm operations of the ISFSI and Robinson-2 have no significant safety impact on one another, that no changes to the Robinson-2 operating license are needed for the ISFSI, and that the ISFSI operates independently. The CP&L letter of April 22, 1986,⁽²²²⁾ to NRC provides additional information (five specific items) in support of the ISFSI

license application. The CP&L letter of May 21, 1986,⁽⁹⁾ to the NRC transmits the results of an internal CP&L review of the ISFSI: the conclusion was that the ISFSI is "independent" as defined in 10 CFR Part 72. The CP&L letter of July 10, 1986,⁽³⁵⁷⁾ to NRC transmits current copies of 12 drawings and 7 specifications, as released for procurement and construction, for the Robinson-2 ISFSI. The CP&L letter of July 22, 1986,⁽³⁵⁸⁾ to NRC transmits current copies of three drawings and one specification, as released for procurement and construction, for the Robinson-2 ISFSI. The NRC letter of August 13, 1986,⁽¹³⁴⁾ to CP&L encloses Materials License No. SNM-2502, which authorizes CP&L for a 20-year term to receive, possess, store, and transfer spent fuel from Robinson-2 in an ISFSI at CP&L's Robinson-2 site. The NRC staff's Safety Evaluation Report for the Robinson-2 ISFSI was also enclosed in the NRC letter.⁽¹³⁴⁾ The CP&L letter of October 31, 1986,⁽³⁵⁹⁾ to the NRC requests an amendment to the license issued August 13, 1986, for the Robinson-2 ISFSI. The proposed amendment changes the minimum effective boron-10 loading in Technical Specification 5.3 from 0.02 g/cm² to 0.004 g/cm² (the change is being made to permit the use of a more readily available poison material). The NRC letter of November 10, 1986,⁽²²³⁾ to CP&L indicates that the NRC concluded from their review of CP&L's license amendment request⁽³⁵⁹⁾ that it contains insufficient information to support the proposed action. The letter⁽²²³⁾ indicated that CP&L did not adequately describe or characterize the material proposed for use and did not provide sufficient data to enable the NRC to confirm the CP&L criticality analysis. The NRC memorandum of December 4, 1986,⁽²²⁴⁾ provides information about a meeting of NRC staff with CP&L to discuss CP&L's request⁽³⁵⁹⁾ for an amendment to its license (SNM-2502) for storage of spent fuel in its Robinson-2 ISFSI.

1987: The CP&L letter of January 2, 1987,⁽²²⁵⁾ to the NRC provides responses to NRC's requests that were made during a meeting of NRC staff with CP&L on December 2, 1986. The CP&L letter of January 8, 1987,⁽³⁶⁰⁾ to the NRC concerns the submittal of a proprietary drawing. The NRC letter of January 29, 1987,⁽²²⁶⁾ to CP&L includes several requests for information and several questions. The CP&L letter of February 17, 1987,⁽²²⁷⁾ to the NRC transmits responses to the requests and questions in the NRC letter.⁽²²⁶⁾

The CP&L letter of March 4, 1987,⁽²¹²⁾ to the NRC transmits Amendment No. 1 to the SAR for the Robinson-2 ISFSI. The NRC letter of March 30, 1987,⁽²²⁸⁾ to CP&L concerns two missing pages in CP&L's recent submittal⁽²¹²⁾ and includes a request for copies of the missing pages. The NRC letter of April 7, 1987,⁽²²⁹⁾ to CP&L includes those comments about CP&L's updated ISFSI SAR.⁽²¹²⁾ The NRC note of April 15, 1987,⁽²³⁰⁾ to file concerns NRC comments on procedural errors with respect to canister weld check tests. The NRC letter of April 16, 1987,⁽³⁶¹⁾ to the NRC transmits Amendment No. 1 to the Robinson-2 ISFSI Materials License No. SNM-2502. The CP&L letter of June 5, 1987,⁽³⁶²⁾ to the NRC encloses copies of two pages that were inadvertently omitted from the submittal⁽²¹²⁾ of Amendment No. 1 to the Robinson-2 ISFSI SAR. The CP&L letter of August 13, 1987,⁽³⁶³⁾ to the NRC transmits Amendment No. 2 to the Robinson-2 ISFSI SAR. The CP&L letter of September 29, 1987,⁽³⁶⁴⁾ to the NRC transmits the status of six drawings in response to an NRC request. The CP&L letter of November 25, 1987,⁽³⁶⁵⁾ to the NRC requests an amendment (proposed changes are administrative and editorial in nature) to the license of the Robinson-2 ISFSI.

1988: The NRC letter of February 11, 1988,⁽³⁶⁶⁾ to CP&L issued Amendment No. 2 to Materials License No. SNM-2502 for the Robinson-2 ISFSI. The NRC letter of February 19, 1988,⁽³⁶⁷⁾ to the NRC enclosed a copy of Amendment No. 2 to Robinson-2 ISFSI Materials License No. SNM-2502.

3.2.1.3 Duke Power Company (Oconee)

1987: The NRC received a letter of intent⁽¹⁴⁴⁾ from Duke Power Company (Duke) in September 1987 and NUTECH Engineers, Inc., regarding Duke's intention to apply in April 1988⁽¹⁴⁶⁾ for a license for an Independent Spent Fuel Storage Installation (ISFSI) under 10 CFR 72 to store Oconee spent fuel in the modified NUHOMS-24P (holds 24 PWR fuel assemblies) module.^(37,45,128) The format for and submittal of the NUTECH topical report for the modified NUHOMS-24P were discussed at a meeting with the NRC on October 15, 1987.⁽⁴⁵⁾ Design differences between the NUHOMS-24P and the NUHOMS-07P (the topical report is docketed under Project M-39) and the design criteria for the NUHOMS-24P design proposed for the Oconee site were also described.⁽⁴⁵⁾

1988: On March 31, 1988, Duke transmitted⁽¹⁴³⁾ a letter to the NRC with proposed amendments to the Technical Specifications for Facility Operating Licenses DPR-38, -47, and -55 (Oconee-1, -2, and -3, respectively) to permit Duke to operate an ISFSI it plans to construct at the Oconee site. On May 24, 1988, the NRC requested information from NUTECH, Inc., on NUTECH's topical report (NUH-002 Revision 1) to complete its review of that report. On June 24, 1988, the NRC transmitted⁽²³¹⁾ to Duke the NRC's⁽³²¹⁾ comments on the Environmental Report (ER) for the Oconee ISFSI that was submitted by Duke on March 31, 1988.⁽³⁶⁸⁾ On July 27, 1988, the NRC transmitted⁽³⁶⁹⁾ to Duke the NRC's comments on the ISFSI Safety Analysis Report (SAR) submitted⁽¹⁴³⁾ by Duke on March 31, 1988. On August 16, 1988, Duke provided⁽³⁷⁰⁾ responses to NRC's questions⁽²³¹⁾ that were raised on the ER on June 24, 1988. On August 31, 1988, the NRC requested⁽¹⁴⁹⁾ additional information from NUTECH (some of the information requested by NRC on May 24, 1988, was not supplied or was not supplied in sufficient detail and some calculations that were provided did not appear to correspond to the NUTECH design). On September 19, 1988, Duke furnished⁽⁷⁾ a response to NRC that addressed an issue that arose during NRC's review of the License Application. The issue centered on the question of whether, and to what extent, the spent fuel storage canisters to be used in the Oconee ISFSI will be compatible with the transportation cask ultimately chosen by DOE to transport spent nuclear fuel temporarily stored at reactor sites to permanent repositories. On October 11, 1988, Duke requested⁽³⁶⁸⁾ approval to initiate preconstruction activities at the Oconee ISFSI site. On October 12, 1988, the NRC requested additional information from NUTECH. On November 1, 1988, the NRC requested⁽¹⁵⁰⁾ additional information to confirm the design basis deceleration levels for the cask drop case (the NRC indicates that it is possible that the deceleration levels presented by NUTECH could be higher than those stated in the topical report, NUH-002, Revision 1). On November 23, 1988, Duke transmitted⁽¹⁴⁵⁾ responses to NRC's questions⁽³⁶⁹⁾ on the safety analysis report for the Oconee ISFSI.

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APPENDIX D

TABLE D.1: QUESTIONS, REQUESTS, AND RESPONSES ASSOCIATED WITH
LICENSING OF THE INDEPENDENT SPENT FUEL STORAGE INSTALLATION
AT SURRY

TABLE D.1. Questions, Requests, and Responses Associated with Licensing of the Independent Spent Fuel Storage Installation at Surry

Questions/Requests (a,b)	Responses (a,b)	Comments (a,b)
<p><u>QUESTION 1.1.1</u>⁽¹⁶⁾: It is stated in SAR Section 2.1.3.1, "Population Distribution and Trends," that the population projections (0-10 mile radii) are the same as those presented in the environmental report on the Surry Power Station Units 3 and 4, and they are used in this SAR because there was little change in this population between 1970 and 1980. This seems inconsistent with the projections of population for the 1980-1990 period for the 0-5 mile radii showing a decrease of over 60 percent and for the 5-10 mile radii which more than doubled. Explain the difference in the population trends between these two decades.</p>	<p><u>RESPONSE</u>: The problems which arise when comparing the 1980 0-10 mile population distribution with the 1990 0-10 mile projection can be linked to the different methodologies used to derive these estimates. The two distributions do not compare well because they were not derived in the same manner and were not based on the same data set. The 1980 0-10 mile population distribution was based on rural house counts taken from the USGS topographic maps. The number of houses was then converted to a population figure through application of a multiplier based on the average number of persons to housing units. This figure was taken from the 1977 City and County Data Book.^(k) This technique attempts to estimate the permanent resident population and therefore does not include transient and institutionalized populations. This estimate was first presented in the 1980 Surry Power Station Transport Evacuation Study.^(l) The use of USGS topographic maps to estimate area population produce incomplete and outdated estimates of population.</p> <p>The 1990 population projection which includes transient, institutionalized and permanent populations is based on 1970 census data, adjusted for growth. Developed with a different methodology using a different data base, the 1990 population consequently did not appear consistent with the 1980 estimate.</p> <p>In response to Question 1.1.5, the 0-10 mile and 10-50 mile total population and population projections have been revised and are shown in Figures 1-10. The figures are indicative of reasonable, consistent population growth trends for the 1980-2020 period.</p> <p>Surry ISFSI ER Section 2.1 (specifically Subsection 2.1.2), "Geography and Demography", and SAR Section 2.1 (specifically Subsection 2.1.3), "Geography and Demography of Site Selected" have been updated to reflect revised 0- to 10-mile and 10- to 50-mile estimates and projections of population.</p>	<p>See submittals (306,325) Q and R pertain to SAR Section, 2.0, "The Site Characteristics" and ER Section 2.0, "The Site and Environmental Interfaces."</p>

TABLE D.1. (contd)

Questions/Requests (a,b)	Responses (a,b)	Comments (a,b)
<p>QUESTION 1.1.2E⁽¹⁶⁾: The large increase in population within 10 miles between 1980 (61,711), as shown in ER and SAR Figure 2.1-3, "Population Distribution: 0-10 and 10-50 Miles for 1980," and 1990 (161,454), as shown in ER and SAR Figure 2.1-4, "Population Distribution: 0-10 and 10-50 Miles for 1990," seems unreasonable. Subsequent rates of growth to 2020 are much smaller. Please justify or correct these numbers.</p>	<p>RESPONSE: The apparent inconsistency found when comparing the 1980 10-mile radius population figure with its supposed 1990 counterpart as is the answer to Question 1.1.1., is one of differing methodologies. These two figures cannot be compared since they were not derived in the same manner and were not based on the same raw data. The 1980 population, Figure 2.1-3, referenced in the Surry ISFSI Environmental Report was based on rural house counts taken from USGS topographic maps. The number of houses was then converted to a population figure through the application of a multiplier based on the average number of persons to housing unit. This figure was taken from the 1977 City and County Data Book.^(k) The 1980 figure referenced in the question does not include transient and institutionalized populations. The technique used was an attempt at estimating the total permanent resident population at the time. The estimate referenced above was first presented in the 1980 Surry Power Station Transport Evacuation Study.^(l) More recent information, however, has shown that the housing count technique used significantly underestimated the permanent population within 10 miles of the site.</p> <p>The 1990 population projection of 161,454 includes the local transient and institutionalized populations, i.e., it represents the total expected population for 1990. As the population projections for the years 2000, 2010, and 2020 also represent total populations, respectively, the apparent explosive growth between 1980 and 1990 appears as a distinct discontinuity in an otherwise consistent upward trend in population.</p> <p>In response to Question 1.1.5, the 0-10 mile and 10-50 mile 1980 total population and population projections have been revised. As these revised population distributions described total population and they have a common raw data base (1980 U.S. Census Data), no discontinuities appear in the population trends between 1980 and the year 2020. These distributions are shown in Figures 1-10.</p> <p>Surry ISFSI ER Section 2.1.2, "Population Distribution," and SAR Section 2.1.3, "Population Distribution and Trends," have been updated to</p>	<p>See submittals.^(306,325) Q and R pertain to SAR Section, 2.0, "The Site Characteristics" and ER Section 2.0, "The Site and Environmental Interfaces."</p>

TABLE D.1. (contd)

Questions/Requests ^(a,b)	Responses ^(a,b)	Comments ^(a,b)
<u>QUESTION 1.1.2E: (contd)</u>	<u>RESPONSE: (contd)</u> reflect revised 0- to 10-mile and 10- to 50-mile estimates and projections of population.	
<u>QUESTION 1.1.3E⁽¹⁶⁾</u> : Attachment 1 to Figure 2, page 4-C-13 of the Virginia Radiological Emergency Plan, revised August 1981, indicates a 1980 population within the 10 mile EPZ of 79,991. This is a considerably higher than the 61,711 reported in ER Section 2.1.2.1, "Population within Ten Miles." Please explain the basis of the difference.	<u>RESPONSE:</u> The inconsistency between the Virginia Radiological Emergency Plan (1981) estimate of the population within the 10-mile EPZ (79,991) and the corresponding estimate in the Surry ISFSI ER and SAR of 61,711, as the answer to Question 1.1.2, can be explained by the differences in methodology used to develop the two estimates. The original estimate of the 1980 resident population within 10 miles of the site was determined from rural house counts taken from USGS topographic maps. These house counts were converted to population by applying a multiplier average of persons to housing units obtained from 1977 City and County Data Book. ^(k) The derived population estimate, 61,711 was presented initially in the 1980 Surry Power Station Transport Evacuation Study. ^(l) It was later determined that the house counting technique was insufficient in determining a reasonable projected figure. The Virginia Radiological Emergency Response Plan, ^(m) June 1983 indicates a 1980 resident EPZ population of 84,574. ⁽³⁰⁶⁾ Surry ISFSI ER Section 2.1.2.1 and SAR Section 2.1.3.1 were revised to indicate a total population estimate shown on Figure 1 of the response to Question 1.1.5. ⁽³⁰⁶⁾ The population estimate in ISFSI ER Section 2.1, "Geography and Demography," and SAR Section 2.1, "Geography and Demography of Site Selected," have been revised. ⁽³²⁵⁾	See submittals. (306,325) Q and R pertain to SAR Section, 2.0, "The Site Characteristics" and ER Section 2.0, "The Site and Environmental Interfaces."
<u>QUESTION 1.1.4⁽¹⁶⁾</u> : The projected population distributions in Figure 2.1-3 through -7 ("Population Distribution: 0-10 and 10-50 Miles for 1980, 1990, 2000, 2010, and 2020," respectively) should be checked for errors. For example, the NE sector in Figure 2.1-3, "Population Distribution: 0-10 and 10-50 Miles for 1980," for 40 to 50 miles has an indicated population of 4,000; yet, this area is totally within the Chesapeake Bay. How were the population distributions for 10-50 mile area estimated?	<u>RESPONSE:</u> The 1980 10-50 mile population estimate indicating a population of 4000 in the 40-50 mile NE sector was taken directly from the Surry Power Station Transportation Evacuation Study. ^(l) Though some grids show this sector to be entirely within the Chesapeake Bay, it is possible that the grid used in the reference was centered at location which though near the Surry Station, was displaced enough to push the sector into populated land areas. Note that the reference ^(l) described the	See submittals. (306,325) Q and R pertain to SAR Section, 2.0, "The Site Characteristics" and ER Section 2.0, "The Site and Environmental Interfaces."

TABLE D.1. (contd)

Questions/Requests ^(a,b)	Responses ^(a,b)	Comments ^(a,b)
<u>QUESTION 1.1.4: (contd)</u>	<p data-bbox="953 310 1163 337"><u>RESPONSE: (contd)</u></p> <p data-bbox="953 358 1535 407">population distribution in tabular form and did not include a sectorized grid representation.</p> <p data-bbox="953 428 1535 532">The 1990-2020 population projections provided in the Surry ISFSI ER and SAR were taken from the 1973 Surry Power Station Units 3 & 4 Preliminary Safety Analysis Report.^(N)</p> <p data-bbox="953 553 1535 695">In response to Question 1.1.5, the 0-10 mile and 10-50 mile 1980 total population and population projections have been revised and are shown in Figures 1-10. Note that the revised 1980 10-50 mile population estimate shows no population within the NE 40-50 mile sector (Chesapeake Bay).</p> <p data-bbox="953 716 1535 889">Surry ISFSI ER Section 2.1 (specifically Subsection 2.1.2), "Geography and Demography," and SAR Section 2.1 (specifically Subsection 2.1.3), "Geography and Demography of Site Selected," have been updated to reflect revised 0- to 10-mile and 10- to 50-mile estimates and projections of population.</p>	
<p data-bbox="256 906 905 1133"><u>QUESTION 1.1.5E⁽¹⁶⁾</u>: Why are 10-year old transient population estimates incorporated by reference in ER Section 2.1.2.3, "Transient and Institutionalized Population," when more current estimates are available in the Virginia Radiological Emergency Plan, referred to above? Does VEPCO still consider the population estimates and projections from the Surry-3 and -4 Environmental Report to be valid? If not, furnish the most and valid current population data.</p>	<p data-bbox="953 906 1535 1198"><u>RESPONSE:</u> The Surry Power Station Units 3 and 4 ER transient population estimates,^(O) incorporated by reference in the Surry ISFSI ER Section 2.1.2.3, were considered the best available estimates of transient population at the time of the ISFSI ER's development. Since then, a more current source of transient population information (Virginia Radiological Emergency Response Plan - June 1983)^(P) has become available. Consequently, a revised description of the nearby transient and institutionalized population based on this new source follows:</p> <ol data-bbox="953 1219 1535 1463" style="list-style-type: none"> <li data-bbox="953 1219 1535 1463">1. Transient Population <ul data-bbox="995 1268 1535 1463" style="list-style-type: none"> <li data-bbox="995 1268 1535 1317">a. Jamestown Colonial National Historical Park 2,500 <li data-bbox="995 1338 1535 1365">c. College of William and Mary 8,300 <li data-bbox="995 1386 1535 1414">e. Colonial Williamsburg Foundation 15,400 <li data-bbox="995 1435 1535 1463">i. Busch Gardens 35,000 	<p data-bbox="1570 906 1932 1036">See submittals.^(306,325) Q and R pertain to SAR Section, 2.0, "The Site Characteristics" and ER Section 2.0, "The Site and Environmental Interfaces."</p>

TABLE D.1. (contd)

Questions/Requests ^(a,b)	Responses ^(a,b)	Comments ^(a,b)
QUESTION 1.1.5E: (contd)	RESPONSE: (contd)	
	n. Yorktown Colonial National Historical Park	2,500
	g. Chippokes State Park	55
	Total	63,755
	2. Institutionalized Population	600
	b. Walsingham Academy	600
	d. Eastern State Hospital	1,500
	f. Williamsburg Community Hospital	400
	g. Pines Convalescent Center	290
	h. Camp Peary	300
	j. Naval Supply Depot - Cheatham Annex	200
	k. Ball Corporation	400
	l. Yorktown Naval Mine Depot	2,400
	m. Badische Corporation	550
	o. Fort Eustis	8,500
	p. City Farm Penal Facility	150
	Total	15,290

In response to the part of Question 1.1.5 concerning the validity of the Surry 3 and 4 Environmental Report population estimates and projections, these items have been revised.

The revised 1980 0-10 mile total population estimate shown in Figure 1 was taken directly from the June 1983 Virginia Radiological Emergency Response Plan.^(p) The 1980 10-50 mile total population estimate shown in Figure 6 was based on 1980 census data.^(q) Specifically, the population by sector for the 10-50 mile grid was calculated by summing the area weighted fraction of the population contributed by counties which comprise the sector. Large cities were added as appropriate and so were

TABLE D.1. (contd)

Questions/Requests ^(a,b)	Responses ^(a,b)	Comments ^(a,b)
<u>QUESTION 1.1.5E: (contd)</u>	<p><u>RESPONSE: (contd)</u></p> <p>not included in the area averaging procedure. Note that county population was assumed to be uniformly distributed.</p> <p>The 1990-2020 population projections for 0-10 miles and 10-50 miles shown in Figures 2 to 5 and 7 to 10 were based on Virginia county projections found in the Virginia Population Projections 2000^(r) and supplemental information provided by the Virginia Department of Planning and Budget Research Section.^(s) Again the population by sector was calculated by summing the area weighted fraction of the projected population contributed by the counties which comprise the sector, and adding the population of large cities which fall in the sector.</p> <p>Surry ISFSI ER Section 2.1.2 and SAR Section 2.1.3 were updated to reflect the revised 0-10 mile and 10-50 mile estimates and projections of population. Figures 1-10 replace SAR and ER Figures 2.1-3 through 2.1-7.⁽³⁰⁶⁾</p> <p>Surry ISFSI ER Section 2.1, "Geography and Demography," and SAR Section 2.1, "Geography and Demography of Site Selected," have been updated to reflect revised 0- to 10-mile and 10- to 50-mile estimates and projections of population.⁽³²⁵⁾</p>	
<p><u>QUESTION 1.2.1⁽¹⁶⁾</u>: It is stated in SAR Sections 2.2.1, "Location and Routes," 2.2.3, "Effects of Potential Accidents," and 2.2.3.1.4, "Pipelines," that the Commonwealth Natural Gas Corporation and the Colonial Pipeline Company own pipelines which cross the southeast corner of the Surry property. How many pipelines are there? SAR Table 2.2-6, "Pipeline Data," indicates six and notes that the two Commonwealth National Gas Corporation lines lie four feet beneath the river bed. How about the remaining four? Once these pipelines emerge from the river are they buried or above ground?</p>	<p><u>RESPONSE:</u> There are six pipelines which run near or cross the Surry Power Station property. As reported in Table 2.2-6 of the Surry ISFSI SAR. A detailed description of each pipeline follows below:</p> <p>A. Commonwealth Natural Gas Corporation Pipelines</p> <ol style="list-style-type: none"> 1. Two 8-in. diameter natural gas lines 2. Single 10-in. diameter natural gas line 3. Single 12-in. diameter natural gas line which serves the Surry Power Station Gas Turbine Generator. This line branches from the 10-inch line. 	<p>See submittals.^(306,325) Q and R pertain to SAR Section, 2.0, "The Site Characteristics" and ER Section 2.0, "The Site and Environmental Interfaces."</p>

TABLE D.1. (contd)

Questions/Requests ^(a,b)	Responses ^(a,b)	Comments ^(a,b)
<u>QUESTION 1.1.5E: (contd)</u>	<u>RESPONSE: (contd)</u> 4. Single 12 3/4-in. diameter natural gas line which was recently completed. This line does not cross Surry Power Station. These natural gas pipelines have maximum and normal operating pressures of 600 and 150 psi, respectively. These lines run under the James River at a depth of 3 to 4 feet. The pipelines remain buried 30-36 inches below grade after emerging from the river. Table 2.2-6, "Pipeline Data," of the Surry ISFSI SAR has been revised to include the information requested.	
<u>QUESTION 1.2.2⁽¹⁶⁾</u> : Section 2.2.3.1.2, "Waterborne Traffic," of the SAR states that dredged channel in the river is 2.5 miles from the ISFSI at the closest point (this distance is also used in the accompanying accident analysis), but in SAR Table 2.2-2, "Chemical Compounds Shipped on the James River," the distance is given as 1.5 miles. Which is correct?	<u>RESPONSE:</u> Section 2.2.3.1.2, "Waterborne Traffic," of the Surry ISFSI SAR states that the dredged channel in the James River is 2.5 miles from the ISFSI site at its closest point. Inspection of Figure 2.3-25, "General Topography: 5-Mile Radius," of the Surry ISFSI ER confirms this statement. The 2.5-mile distance refers to the separation between the site and the mid-river channel. The 1.5-mile distance given in the SAR Table 2.2-2, "Chemical Compounds Shipped on the James River," (referenced from the Surry Onsite Toxic Chemical Analysis, Vol. II, NUS, June 1981) refers to the minimum separation of the Surry Power Station control room and the James River. Note that the NUS analysis does not specify what part of the James River is used to calculate the minimum separation. The Surry ISFSI site is farther from the James River than the control room.	See submittals. ^(306,325) Q and R pertain to SAR Section, 2.0, "The Site Characteristics" and ER Section 2.0, "The Site and Environmental Interfaces."
<u>QUESTION 1.3.1⁽¹⁶⁾</u> : SAR Section 3.2.1.1, "Applicable Design Parameters," identifies the design basis extreme ambient temperatures of -20°F and 115°F. These temperatures were selected because they exceed the extreme temperatures recorded at the Norfolk and Richmond National Weather Service Stations as reported in SAR Section 2.3.2.1.1, "Local Climatological Data." The conservatism of the upper temperature extreme should be clearly established in SAR Section 2.3.2, "Local Meteorology," because, for a truly passive spent fuel storage system, the unaided atmosphere will serve as the principal heat	<u>RESPONSE:</u> The response to this question has been incorporated into the ISFSI SAR Section 2.3, "Meteorology," and ER Section 2.3, "Meteorology."	See submittal. ⁽³²⁵⁾ Q and R pertain to SAR Section, 2.0, "The Site Characteristics" and ER Section 2.0, "The Site and Environmental Interfaces."

TABLE D.1. (contd)

Questions/Requests (a,b)	Responses (a,b)	Comments (a,b)
<u>QUESTION 1.3.1: (contd)</u>		
sink. Provide a rigorous basis for establishing these temperatures. The discussion could answer the following questions:		
1. Are these temperatures reported from nearby weather service stations representative of the region?		
2. Did the onsite meteorological station temperature records correlate with the nearby water service stations records?		
3. Does the site have peculiar micrometeorological conditions that could cause a difference between its readings and the nearby weather stations readings?		
4. What is the probable length and frequency of occurrence of excessively hot periods?		
5. What is the worst combination of climatology conditions which would adversely affect the ability of the ambient air to remove heat from the cask surface?		
6. What are the recurrence intervals and duration for the selected extreme temperatures?		
The following references may be of help in developing the statistical basis for the discussion: "Extreme Meteorological Events in Nuclear Power Plants, Excluding Tropical Cyclones," IAEA Safety Guide No. 50-SG-S11A, and "Probability Estimates of Temperature Extremes for the Contiguous United States," NUREG/CR-1390, May 1980.		
<u>QUESTION 1.3.2</u> ⁽¹⁶⁾ : In SAR Section 2.3.2, "Local Meteorology," the basis for insolation design parameters should be provided. The conservatism of the solar heat load burden at the ISFSI site should be substantiated in a discussion that justifies the selection of the 90 percent transmissivity factor and the 100-hour exposure period, or these should be changed to the more severe conditions.	<u>RESPONSE:</u> Section 3.2.1 of the Surry ISFSI SAR describes the design solar heat burden at the site as 800 g-cal/cm ² per day (10-hour exposure). This maximum insolation value represents about 90 percent of the solar radiation incident at the top of the atmosphere at the ISFSI latitude. ^(t) The 90-percent assumption refers to a 0.9 transmissivity factor which is representative of a conservative clear sky transmissivity ^(u) , i.e., the percent of solar radiation at the top of the atmosphere which makes it to the ground after absorption or scattering by atmospheric constituents other than	See submittals. (306,325) Q and R pertain to SAR Section 2.0, "The Site Characteristics."

TABLE D.1. (contd)

Questions/Requests ^(a,b)	Responses ^(a,b)	Comments ^(a,b)
QUESTION 1.3.2: (contd)	<p>RESPONSE: (contd)</p> <p>clouds. The 10-hour exposure period is based on the normal daily hours of sunshine during summer at the ISFSI site.^(v) Note the length of the exposure period is included for information only and consequently it does not enter into the calculation of the daily maximum insolation.</p> <p>Comparing the 800 g-cal/cm² per day, insolation value with the normal daily total solar radiation (June 21) at the ISFSI site of 500 g-cal/cm² gives more evidence of the conservatism of this insolation design criteria.</p> <p>Note Regulatory Guide 7.8^(w) describes maximum insolation data as part of the general and initial conditions to be used for both normal and hypothetical accident conditions for shipping casks. The Regulatory Guide defines the maximum insolation for horizontally transported casks as 800 g-cal/cm² per day. For stationary and vertically stored casks, which are more representative of the ISFSI casks, the Regulatory Guide defines the design insolation as 400 g-cal/cm². Consequently, the ISFSI cask design insolation of 800 g-cal/cm² appears to be ultraconservative as it is double the appropriate Regulatory Guide 7.8 maximum insolation for stationary casks.</p> <p>The response to this question has been incorporated into ISFSI SAR Section 2.3, "Meteorology."</p>	
<p>QUESTION 1.3.3⁽¹⁶⁾: Provide a discussion of the potential for lightning strikes at the ISFSI. This discussion could include the following topics:</p> <ol style="list-style-type: none"> 1. Onsite experience with lightning strikes on Surry Power Station structures and switchyard facilities. 2. A correlation of the frequency and intensity of both single and multiple lightning strikes associated with regional thunderstorms. 3. The expected frequency of thunderstorms at the ISFSI site. 4. The limiting case for energy release associated with a lightning strike. 	<p>RESPONSE: ISFSI ER Section 2.3, "Meteorology," and SAR Section 2.3, "Meteorology" have been revised to reflect the response to this question.</p>	<p>See submittal.⁽³²⁵⁾ Q and R pertain to SAR Section, 2.0, "The Site Characteristics" and ER Section 2.0, "The Site and Environmental Interfaces."</p>

TABLE D.1. (contd)

Questions/Requests (a,b)	Responses (a,b)	Comments (a,b)
<p><u>QUESTION 1.3.4</u>⁽¹⁶⁾: Provide a correlation between the Surry site specific data, developed over the years the onsite meteorological program has been established at the Surry site, and the Richmond and Norfolk data provided in SAR/ER Tables 2.3-1 through 2.3-4 ("Selected National Weather Service Stations for Meteorological Extremes in the Surry Site Region," "Richmond-Meteorological Normals, Means, and Extremes," and "Monthly Meteorological Means for Temperature and Precipitation for Stations in the Surry Site Region," respectively).</p>	<p><u>RESPONSE</u>: Surry ISFSI ER Section 2.3, "Meteorology," and SAR Section 2.3, "Meteorology," have been revised to reflect the response to this question.</p>	<p>See submittal (325) Q and R pertain to SAR Section, 2.0, "The Site Characteristics" and ER Section 2.0, "The Site and Environmental Interfaces."</p>
<p><u>QUESTION 1.3.5</u>⁽¹⁶⁾: Describe the methodology for obtaining the X/Q values in ER Table 7.1-3, "Assumptions Used for Loss of Confinement Barrier Analysis," and ER Figure 7.1-1, "Surry Accident Meteorology." Include the models and input data used.</p>	<p><u>RESPONSE</u>: New information is now available concerning the dispersion factors (X/Q) for a loss of Confinement Barrier Accident at the proposed dry cask Independent Spent Fuel Storage Facility (ISFSI) at Surry Power Station. Dispersion factors were calculated for thirteen downwind locations from 0.313 to 50 miles. The Surry Dry Cask ISFSI Environmental Report will be revised to incorporate this current information. The methodology used to determine the atmospheric dispersion factors (X/Q) for the ISFSI is as follows:</p> <p>Accidents were assumed to result in ground-level effluent releases from the Dry Cask Independent Spent Fuel Storage Facility (ISFSI). For various time periods after an accident, (0-2 hr, 0-8 hr, 8-24 hr, 1-4 day, and 40-30 day), atmospheric dispersion factors (X/Q) were calculated at the Exclusion Area Boundary (EAB) and population receptor midpoints for emissions from the ISFSI.</p> <p>X/Q dilution factors were calculated using the bivariate normal, or Gaussian diffusion model (Regulatory Guide 1.145 Rev. 1), modified for source configuration and lateral meander under neutral and stable conditions. Meteorological input parameters were determined from onsite meteorological data acquired during the January 1, 1976 through December 31, 1982 (7-year) time period. The parameters included hourly average values (based on 15-minute averages) of wind speed and wind direction at the 33-ft level, and atmospheric stability determined from the temperature differences measured between the 33- and 15-ft levels. Atmospheric stability was classified according to the temperature gradient values for the various Pasquill stability categories (A through G).</p>	<p>See submittals (313,325) for Figure 1 and Tables 1-13. Q and R pertain to SAR Section 2.0, "The Site Characteristics."</p>

TABLE D.1. (contd)

Questions/Requests (a,b)	Responses (a,b)	Comments (a,b)
QUESTION 1.3.5E: (contd)	<p data-bbox="932 318 1136 342"><u>RESPONSE: (contd)</u></p> <p data-bbox="932 367 1503 464">Hourly average X/Q values for the 1-hour (representative of the 0- to 2-hour period) accident period were calculated using Equations 1 through 3 listed below.</p> <p data-bbox="932 488 1503 537">For D, E, F, and G stability conditions, when the wind speed is less than 6 meters per second:</p> $X/Q = (\bar{V}_{33} \pi \Sigma_y \sigma_z)^{-1} \quad (1)$ <p data-bbox="932 602 1503 854"> where X/Q = Relative Concentration (sec/m³) $\bar{V}_{33}$ = Wind Speed at 33 ft above plant grade (m/sec) π = 3.14159 . . . Σ_y = Lateral plume spread with meander and building effects $\Sigma_y = M \sigma_y$ for distances up to 800 m $\Sigma_y = (M-1) \sigma_y 800 \text{ m} + \sigma_y$ for distances beyond 800 m $\sigma_y \sigma_z$ = Lateral and vertical plume spread (m) </p> <p data-bbox="932 870 1503 967">Figure 1, "Meander Factor Versus Wind Speed," depicts the functional relationship of M (meander factor) with respect to wind speed and atmospheric stability.</p> <p data-bbox="932 984 1503 1130">If the X/Q value calculated in Equation (1) is less than the greater X/Q value of either of the following equations, it is retained; otherwise, the applicable X/Q value which is the greater of those calculated by Equations (2) and (3) becomes limiting</p> $X/Q = [\bar{V}_{33} (\pi \sigma_y \sigma_z + A/2)]^{-1} \quad (2)$ $X/Q = [\bar{V}_{33} (3\pi \sigma_y \sigma_z)]^{-1} \quad (3)$ <p data-bbox="932 1243 1503 1341">where A = The smallest vertical plan cross-sectional building area (m²). For both the containment and ISFSF releases, the minimum containment building area of 15,667 ft² is used.</p> <p data-bbox="932 1365 1503 1487">For all A, B, and C stability conditions, and for D, E, F, and G stability conditions when the wind speed is greater than or equal to 6 meters per second, the greater of the two X/Q values calculated from Equations (2) and (3) become limiting.</p>	

TABLE D.1. (contd)

Questions/Requests (a,b)	Responses (a,b)	Comments (a,b)
<p>QUESTION 1.3.5E: (contd)</p>	<p><u>RESPONSE:</u> (contd)</p> <p>Each valid hour of the January 1, 1976 through December 31, 1982 onsite meteorological data was used for the calculation. An hour of data was considered valid if recovery of the wind speed, wind direction, and temperature difference (ΔT) was simultaneously accomplished. For each valid hour of meteorological data, a X/Q value was calculated as described above, where the wind direction determined the applicable downwind sector. The EAB and population midpoint distances were used (along with the stability class), to determine magnitudes of σ_y and σ_z.</p> <p>For the hours with calm wind speeds, a wind speed of 0.75 miles per hour was assigned. The wind directions during these calm conditions were assigned in proportion to the directional distribution of the non-calm through 3.4 mile per hour wind speed conditions. Regulatory Guide 1.1.45 states non-calm wind speeds below 3.4 miles per hour provides a reasonable method for defining the distribution of wind directions, the last valid wind direction and the actual recorded wind speed were coupled.</p> <p>For each downwind sector, all non-zero X/Q values were stored and arranged in descending order and 0.5 percent X/Q values were chosen. These values were compared and the sector with the largest X/Q value determined the ultimate design basis 0.5 percent X/Q.</p> <p>For time periods greater than 2 hours, X/Q values were determined graphically by techniques outlined in Regulatory guide 1.145. The 0.5 percent value for the 0- to 2-hour period was plotted at the two hour time period on logarithmic vs. time coordinates, while the annual average X/Q value for the same sector was plotted at 8,760 hours. Logarithmic interpolation was applied to locate values for the time periods corresponding to 0- to 8-hours, 8- to 24-hours, 1-4 days, and 4-30 days following an accidental release. The calculation of the annual average X/Q value for each sector was determined using Equation (4), which is based on Regulatory Guide 1.111, Revision 1.</p>	

TABLE D.1. (contd)

Questions/Requests (a,b)	Responses (a,b)	Comments (a,b)
QUESTION 1.3.5E: (contd)	<p>RESPONSE: (contd)</p> $\langle X/Q \rangle_{K,1} = \sum_{j=1}^{n_1} \frac{2.032 \Omega}{N \bar{V}_{33} \left(\sigma_{z_k}^2 + \frac{ch_b^2}{\pi} \right)^{1/2}} \quad (4)$ <p>where n = Number of hours of wind in a particular 22.5 degree sector l = Index for a particular 22.5 degree sector k = Index for a particular receptor distance j = Index for a number of hours $2.032 = (2/\pi)^{1/2} (2/16)^{-1}$ $\pi = 3.14159 \dots$ Ω = Terrain recirculation factor = $\exp [-he^2 (2\sigma_{z_j}(x)^{-1})]$ N = Total number of hours of wind in all sectors for applicable averaging period $\bar{V}_{33}$ = Average wind speed at 33-ft above plant grade (m/sec) σ_z = Vertical dispersion coefficient (m) h_b = building height (m) c = Building shape factor = 0.5 he = Effective stack height (m)</p> <p>Additional dispersion including the building wake effect $(\sigma_{z_k}^2 + ch_b^2/\pi)^{1/2}$ was limited to less than or equal to 1.732.</p> <p>Tables 1 through 13 present resultant X/Q values at the EAB and various population receptors for emissions originating from the dry cask spent fuel storage facility.</p> <p>Surry ISFSI SAR Section 2.3, "Meteorology," has been revised to reflect the response to this question.</p>	<p>See submittals (313,325) for Tables 1-13. Q and R pertain to SAR Section 2.0, "The Site Characteristics."</p>
QUESTION 1.3.6 ⁽¹⁶⁾ : Provide an analysis of the X/Q values based on onsite meteorological data and appropriate atmospheric diffusion models.	<p>RESPONSE: The calculated X/Q values based on the methodology described in the response to Question 1.3.6 are provided in the attached Tables 1 through 13.</p>	

TABLE D.1. (contd)

Questions/Requests ^(a,b)	Responses ^(a,b)	Comments ^(a,b)
<u>QUESTION 1.3.6:</u> (contd)	<u>RESPONSE:</u> (contd)	
	[The tables provide 0.5-percent ground level X/Q values at the 0.313-, 0.5-, 1.5-, 2.0-, 2.5-, 3.5-, 4.5-, 7.5-, 15.0-, 25.0-, 35.0-, 45.0-, and 50.0-mile population receptors for the 0- to 720-hour period following an accident at the Independent Spent Fuel Storage Installation]	
	Surry ISFSI SAR Section 2.3, "Meteorology," has been revised to reflect the response to this question.	
<u>QUESTION 1.3.7</u> ⁽¹⁶⁾ : Since the ISFSI will be located close to the primary and back-up meteorological towers, and the casks provide a continuing heat source; provide an analysis of the impact of the meteorological measurements at these towers.	<u>RESPONSE:</u> Surry ISFSI SAR Section 2.3, "Meteorology," has been revised to reflect the response to this question.	See submittal. ⁽³²⁵⁾ Q and R pertain to SAR Section 2.0, "The Site Characteristics."
<u>QUESTION 1.3.8E</u> ⁽¹⁶⁾ : In ER Section 5.6.2, "Climatological Impact," it is stated that the cask surface temperature may reach 260°F, and concluded that the affected area for atmospheric heating and fogging during precipitation would be small and any enhancement of fog beyond the site boundary would be negligible. Provide the calculations and bases for these conclusions. Include input parameters and equations used in the calculation.	<u>RESPONSE:</u> Surry ISFSI ER Section 5.6.2, "Climatological Impact," has been revised to reflect the response to this question.	See submittal. ⁽³²⁵⁾ Q and R pertain to ER Section 5.0, "Environmental Effects of Facility Operation."
<u>QUESTION 1.4.1</u> ⁽¹⁶⁾ : The historical earthquake data presented and the development of the ISFSI design earthquake (0.07g) utilizing Trifunac and Brady's 1975 study "The Correlation of Seismic Intensity Scales with Peaks of Recorded Strong Ground Motion" does not correspond to the information presented for the selection of 0.15g safe shutdown earthquake for the Surry Power Station. For example, the 1927 Coastal Plain earthquake occurring near the central New Jersey coast is not identified in the historical earthquake data for the ISFSI. Adopt the corresponding g value developed for the nuclear power plant, or justify this new analysis in terms of the criteria of Appendix A of 10 CFR Part 100. In any case the g value should not be less than 0.10 g per paragraph 72.66 (a)(b)(iii) or 10 CFR Part 72.	<u>RESPONSE:</u> There are three parts to the response and they are as follows:	See response submittal; ⁽³²⁵⁾ it applies to NRC Questions 1.4.1, 3.2.1, and 3.2.2. Q and R pertain to SAR Section 2.0, "The Site Characteristics."
<u>NOTE:</u> SAR Section 2.6.2.5, "Procedures to Determine the Design Earthquake," uses Applied Technology Council's seismic zonation map in their 1978 publication,	1. <u>Comments Regarding Questions 1.4.1, 3.2.1, and 3.2.2.</u> NRC Questions 1.4.1, 3.2.1, and 3.2.2 presented NRC concerns with respect to the seismological, geological, and structural design bases for the Surry Dry Cask ISFSI. A meeting was held between Virginia Power and the NRC on March 8, 1984 to discuss the issues raised in the NRC questions. Due to the nature of the questions, and the fact all three deal with the determination of the design seismic event and its effect on the structural slab, cask, and stored spent fuel, no individual responses are made to these questions. Instead, a single response that answers the concerns raised by the NRC, as related to all three questions, was prepared	

TABLE D.1. (contd)

Questions/Requests ^(a,b)	Responses ^(a,b)	Comments ^(a,b)
QUESTION 1.4.1: (contd)	RESPONSE: (contd)	
<p>"Tentative Provisions for the Development of Seismic Regulations for Buildings" NBS Special Publications 510, for additional justification. This reference is based on USGS Open File Report published in 1978, and it is not suitable for determining design earthquakes for structures. This was discussed in supplementary information published in the <u>Federal Register</u>, Volume 45, No. 220 on November 12, 1980 accompanying the promulgation of 10 CFR Part 72. In addition, there is an update on the USGS report. It is "Probabilistic Estimates of Maximum Acceleration and Velocity in Rock in the Contiguous United States," by S. T. Algermissen, D. M. Perkins, P. C. Thenhaus, S. L. Hauson and B. L. Bender, U.S.G.S., Open-File Report 82-1033, 1982. This report includes preliminary maps of horizontal acceleration (expressed as percent of gravity) with a 90 percent probability of not being exceeded in 10, 50, and 250 years. On the 10-year map, the maximum g value for site area is 0.04, on the 50-year map, the maximum g value for the site area is 0.10 and on the 250-year map, the maximum g value for the site area is 0.20. The differences between the results of this study and those presented in Surry ISFSI SAR are significant relative to both g values and recurrence interval. This report is more recent than any of the references cited in connection with Section 2.6, "Geology and Seismology," of the SAR.</p>	<p>and is presented below. This approach was discussed upon at the March 8, 1984 meeting.</p> <p>The Surry Dry Cask Independent Spent Fuel Storage Installation (ISFSI) is designed to store spent fuel resulting from the operation of Surry Power Station Units 1 and 2. The spent fuel will be stored in dry sealed surface storage casks (SSSCs), which provide shielding and confinement of the radioactive fission products. The ISFSI facility will consist, in its final stage, of three separate reinforced concrete slabs. A general site layout for the ISFSI is shown in Figure 4.1-1, "General Site Layout," of the Safety Analysis Report. Each concrete slab, which will have overall dimensions of 32 feet in width, 230 feet in length, and 3 feet in thickness, is designed to support 28 SSSCs. The slab will be supported on a 7-foot-thick bed of compacted backfill material, which is then underlain by the naturally occurring site soils.</p> <p>Separate investigations and analyses, outside those previously performed for Surry Units 1 and 2 and the once proposed Surry Units 3 and 4, were conducted for the Surry ISFSI. These analyses were performed as part of the response to the NRC questions and later discussions with the NRC regarding these questions. At the March 8, 1984 meeting between Virginia Power and the NRC, specific criteria for the resolution of NRC concerns were discussed. These criteria are as follows:</p> <p>a. A Design Earthquake (DE) that is developed based on criteria of 10 CFR 72.66(b).</p>	

TABLE D.1. (contd)

Questions/Requests (a,b)	Responses (a,b)	Comments (a,b)
<u>QUESTION 1.4.1: (contd)</u>	<u>RESPONSE: (contd)</u>	
	<p>b. A cask tip-over must be assumed regardless of analyses that demonstrate that the integrity of the structural pad is maintained and that the cask will not tip over. This criteria is independent of a specific seismic acceleration. Assuming a cask tip-over, the analysis must demonstrate that: 1) criticality is within acceptable limits, 2) there is no loss of confinement, and 3) fuel is removable after a tip-over.</p> <p>c. Analyses of the slab under DE must demonstrate that the design function of the ISFSI is not adversely affected and that there is no impact on the public health and safety.</p>	
	<p>In order to meet these criteria, the additional analyses and investigations which were performed included the determination of a Design Earthquake (DE) based on 10 CFR 72.66(b), a site specific investigation, a soil stability analysis (static and dynamic), and design analyses of the structural slab. Additional analyses in support of their licensing efforts are being performed by General Nuclear Systems, Inc. (GNSI), the cask vendor, to determine the criticality, cask integrity, and basket integrity based on a hypothetical cask overturning event.</p>	
	<p>The extent of the information required to be submitted to address the NRC's concerns with respect to seismicity and stability of sub-surface materials has been included in revised SAR Section 2.6, "Geology and Seismology," and new SAR Appendix 3A, "Structural Considerations for the ISFSI Concrete Slab."</p>	
	<p>A summary of revised SAR Section 2.6, "Geology and Seismology," and new Appendix 3A, "Structural Considerations for the ISFSI Concrete Slab," is contained in Parts 2 and 3 of this</p>	

TABLE D.1. (contd)

Questions/Requests (a,b)	Responses (a,b)	Comments (a,b)
<u>QUESTION 1.4.1: (contd)</u>	<p data-bbox="940 329 1144 354"><u>RESPONSE: (contd)</u></p> <p data-bbox="997 378 1518 521">response. A discussion of the structural analysis as requested by the NRC, the response of the cask due to an overturning event, and the conclusions of these additional analyses are provided in new Appendix 3A, "Structural Considerations for the ISFSI Concrete Slab."</p> <p data-bbox="940 545 1518 1138">2. <u>Seismology.</u> The requirements of 10 CFR 72.66(b) stipulate that for determining the seismic design level of a dry cask facility, a site-specific investigation must be performed to establish site suitability commensurate with the specific requirements of the ISFSI. Due to the inherent safety of the SSCs and the fact that the structural slabs are not important to safety, the approach that was taken to determine the proper seismic design level was based on the use of a building code type seismic design level. Determination of this type of seismic design level depends principally on historic site intensity or probabilistic site acceleration at approximately the 500-year return period and not on a maximum credible site intensity. Applicable studies, which are referenced in revised Section 2.6, "Geology and Seismology," of the Safety Analysis Report, indicate that the appropriate probabilistic acceleration is 5 percent of gravity or less and that the historic intensity is VI Modified Mercalli (MM) or less for the Surry ISFSI site. This intensity can be related to a peak horizontal acceleration of 6.6 percent of gravity.</p> <p data-bbox="982 1162 1518 1375">Based on the results of this site-specific investigation, a conservative value for the seismic design level or Design Earthquake (DE) of 7 percent of gravity at the foundation level was adopted for the Surry ISFSI. The details of the site-specific investigation and the development of the Design Earthquake are contained in revised Section 2.6, "Geology and Seismology," of the Safety Analysis Report.</p>	

TABLE D.1. (contd)

Questions/Requests (a,b)	Responses (a,b)	Comments (a,b)
<u>QUESTION 1.4.1: (contd)</u>	<u>RESPONSE: (contd)</u>	
	<p>3. <u>Stability of Subsurface Materials.</u> A site-specific subsurface investigation and laboratory testing program was conducted for the JSFSI in April and May 1982. The investigation included the drilling of nine test borings, the installation of an observation well, and taking of both undisturbed and disturbed soil samples. The boring logs resulting from the investigation are shown in Figures 2.6-32 through 2.6-42, "Boring Logs," of the Safety Analysis Report. The boring location plan is shown in Figure 2.6-43, "Boring Location Maps." All field work was monitored by a geotechnical engineer. Select recovered samples obtained during the investigation were sent to a testing laboratory for determination of the engineering properties of the site soils. The results of both the field and laboratory investigations were then used to determine the static and dynamic stability of the site soils.</p>	
	<p>The static analyses that were performed included a bearing capacity and settlement analysis. Based on the results of these analyses, it was determined that it will be necessary to excavate and replace the upper soil with a compacted backfill material. In order to obtain the required minimum factor of safety of 3.0 for bearing capacity, 7 feet of soil below the bottom of the slab will be excavated and replaced with fill. The fill will be placed to a minimum density of 95 percent of optimum modified proctor density (ASTM D 1557). The bearing capacity factor of safety with the structural backfill in place is greater than 3.0. The calculated settlement due to static loading is less than 2.0 inches.</p>	

TABLE D.1. (contd)

Questions/Requests (a,b)	Responses (a,b)	Comments (a,b)
<u>QUESTION 1.4.1: (contd)</u>	<u>RESPONSE: (contd)</u>	
	<p>The dynamic analysis of the site soils was performed to determine the soil response to dynamic loading. The dynamic loading considered in the analysis was the Design Earthquake (DE) with a maximum ground acceleration of 7 percent of gravity at the foundation level. The analysis indicated that the stability of the site soils would not be adversely affected by the level of dynamic loading. The magnitude of subsidence under dynamic loading can be considered insignificant and will have no adverse effect on the structural slab. A liquefaction analysis was also performed on all soil layers below the maximum ground water level. The analysis, which was based on the "Simplified Procedure" developed by Seed and Idriss, (c,d) indicated that the minimum factor of safety against liquefaction occurring is 1.5. The calculations using the simplified procedure do not incorporate any adjustment factors for the silt content of any specific soil layer. In addition, the dynamic stresses induced by the DE in the cohesive soil layer are considerably less than the shear strength of these layers. Therefore, no reduction of shear strength will result.</p> <p>In summary, it can be concluded that the site soils at the ISFSI site will provide a safe and stable foundation under both static and dynamic loading conditions. The details of the soil stability analysis are contained in revised Section 2.6, "Geology and Seismology," of the SAR.</p>	

TABLE D.1. (contd)

Questions/Requests (a,b)	Responses (a,b)	Comments (a,b)
<p><u>QUESTION 1.5.1</u>⁽¹⁶⁾: Provide a summary of all factors developed in the site characteristics chapter that are deemed significant to the selection of design bases for the ISFSI. For each factor, identify if it was newly developed in the ISFSI SAR, or if it was referenced from a document previously submitted to the NRC. If it is referenced, provide the specific reference, its date and revision, and the applicable page numbers.</p>	<p><u>RESPONSE</u>: Table 1.5.1, "Site Characteristics Summary," in the response⁽³⁰⁶⁾ provides a summary of the major site characteristics established for the ISFSI. As indicated in the table, the majority for the values that are cited in the ISFSI SAR are identical to those developed and documented for the Surry Power Station. Differences are explained as appropriate.</p> <p>The information provided in Table 1.5.1 is included in Section 2.7 of the ISFSI SAR.</p> <p>Table 2.7-1, "Site Characteristics Summary," of the ISFSI SAR has been added to include the response to this question.⁽³²⁵⁾</p>	<p>See submittals.^(306,325) Q and R pertain to SAR Section 2.0, "The Site Characteristics."</p>
<p><u>QUESTION 2.1.1</u>⁽¹⁶⁾: SAR Section 3.1.1, "Materials to be Stored," should be expanded to include the allowable limits on all pertinent characteristics of the spent fuel that can affect the design and operation of any portion of the ISFSI system. An allowable limit should be in terms of maximum, minimum, or a range of values, as appropriate and not an average or typical value. These limits provide the basis for assessing the compatibility of the ISFSI system with the spent fuel to be stored. Confirmatory analyses and performance requirements of the design and operation of the ISFSI system and its components will be required to envelope these limits. It is expected that most of these limits will be included in the limiting conditions for operation of the ISFSI and as such should be readily verifiable.</p> <p><u>NOTE</u>: If no limit is identified for a pertinent characteristic, then it will be assumed that the design and operation of the ISFSI will accommodate all possible values of that particular characteristic. For example, Section 10.1.1.1, "Fuel to be Stored at ISFSI," states "The fuel shall be stored unconsolidated and shall not be extensively damaged." It further identifies damage</p>	<p><u>RESPONSE</u>: VEPCO has identified the following pertinent characteristics that can affect the design and operation of the ISFSI:</p> <ol style="list-style-type: none"> Initial Fuel Enrichment Fuel Burnup Heat Generation Spent Fuel Physical Configuration/Condition <p>The allowable limits for each of these characteristics are discussed below.</p> <p><u>Initial Fuel Enrichment</u></p> <p>The initial fuel enrichment of any fuel that is stored in the ISFSI will be limited to a maximum of 3.5 weight percent U-235. VEPCO has conservatively specified that the cask must be analyzed for 3.7 weight percent U-235.</p>	<p>See submittals.^(310,325) Q and R pertain to SAR Sections 3.0, "Principal Design Criteria," and 10.0, "Operating Controls and Limits."</p>

TABLE D.1. (contd)

Questions/Requests (a,b)	Responses (a,b)	Comments (a,b)
<u>QUESTION 2.1.1: (contd)</u>	<u>RESPONSE: (contd)</u>	
<p>relative to maintaining cooling geometry and the ability to insert and remove the fuel from the storage cask. There is no mention of the initial fuel pin cladding integrity relative to its function as the primary confinement barrier to the release of radioactive material. Therefore, if that is the intention, a substitute barrier for the fuel pin cladding would have to be provided in the ISFSI system.</p>	<p><u>Fuel Burnup</u></p> <p>The burnup of any fuel that is stored in the ISFSI will be limited to a maximum of 35,000 MWD/MTU.</p> <p><u>Heat Generation</u></p> <p>The heat generation rate by an individual fuel assembly is dependent on three factors: The initial fuel enrichment, the fuel burnup, and the amount of decay time after discharge. VEPCO has used 1 kW (maximum) per fuel assembly as the design criteria for the cask analyses and has required that the cask be analyzed to show that the fuel temperature will be less than 380°C. For the first casks placed in the ISFSI, VEPCO will limit the heat generation rate to 0.7 kW per fuel assembly.</p> <p><u>Spent Fuel Physical Configuration/Condition</u></p> <p>Only spent fuel irradiated at the Surry Power Station Unit Nos. 1 and 2 with the physical configuration as listed in items 1, 2, and 3 of SAR Table 3.1-1 will be stored in the ISFSI. The fuel stored shall be intact (unconsolidated), shall be unfailed (no known cladding defects) and shall not have visible physical damage which would inhibit insertion or removal from the cask fuel basket.</p> <p>Surry ISFSI SAR Section 3.1, "Purpose of Installation," and 10.1, "Technical Specifications," have been revised to reflect the response to this question.</p>	
<p><u>QUESTION 2.1.2(16):</u> For each pertinent characteristic identified in SAR Section 2.1.1, "Site Location," provide the method by which the characteristic will be verified.</p>	<p><u>RESPONSE:</u> In response to Question 2.1.1 the following pertinent fuel characteristics that can affect the design and operation of the ISFSI were discussed: a.) initial fuel enrichment; b.) fuel burnup; c.) heat generation, and d.) spent fuel physical configuration/condition. The method of verification for each of these characteristics is discussed below.</p>	<p>See submittals (310,325) Q and R pertain to SAR Sections 9.0, "Conduct of Operations," and 10.0, "Operating Controls and Limits."</p>
<p><u>NOTE:</u> A verification method need not directly measure a pertinent characteristic. Another characteristic, more amenable to verification, could be used to assure the existence of the pertinent characteristic. Also a verification method can accommodate more than one pertinent characteristic.</p>		

TABLE D.1. (contd)

Questions/Requests ^(a,b)	Responses ^(a,b)	Comments ^(a,b)
<u>Question 2.1.2: (contd)</u>	<p data-bbox="945 324 1522 349"><u>Initial Fuel Enrichment and Fuel Burnup</u></p> <p data-bbox="945 373 1522 609">Fuel management records shall be utilized to verify that the initial fuel enrichment and fuel burnup are within the limits discussed in the response to Question 2.1.1. Each fuel assembly is engraved with a unique identification number (based on ANSI/ANS 57.8) and a vendor identification, which is unique to the site for which the fuel assemblies were fabricated. This will allow visual confirmation of the identity of the fuel assemblies placed in the cask.</p> <p data-bbox="945 641 1522 657"><u>Heat Generation</u></p> <p data-bbox="945 690 1522 852">The heat generation rate of a fuel assembly is based on three factors: initial fuel enrichment, burnup, and cooling time after discharge. Fuel management records will be used to obtain these three factors and an NRC approved code such as ORIGEN will be utilized to ensure that the heat generation is less than 0.75 kW per assembly.</p> <p data-bbox="945 885 1522 901"><u>Spent Fuel Physical Configuration/Condition</u></p> <p data-bbox="945 933 1522 1112">Fuel management records will be reviewed to ensure that the assemblies to be put in the cask have not been previously identified as being failed fuel (leakers). The fuel assemblies shall also be visually inspected (e.g., using television cameras) for physical damage which could potentially cause problems during insertion and/or removal from the storage cask.</p> <p data-bbox="945 1144 1522 1226">Surry ISFSI SAR Sections 9.1, "Organizational Structure," and 10.1, "Technical Specifications," have been revised to reflect the response to this question.</p>	

TABLE D.1. (contd)

Questions/Requests ^(a,b)	Responses ^(a,b)	Comments ^(a,b)
<p>QUESTION 3.1.1⁽¹⁶⁾: Elaborate on SAR Table 3.3-1, "Design Criteria for Dry Sealed Surface Storage Casks," by providing the design criteria and performance specifications to be imposed on cask designers and suppliers. The following paragraphs present examples of topics that should be addressed as an indication of the breadth of coverage needed. They should not be considered as a comprehensive listing of cask design requirements, nor should any example requirements be considered as mandatory for the Surry ISFSI system. The discussion of each topic should include the basis for any requirement.</p> <p><u>Spent Fuel Storage Environment:</u> Identify the required storage environment for the spent fuel inside the cask and the range of external conditions under which the environment must be maintained. This could include the following characteristics: maximum cladding temperature; cask internal pressure; storage atmosphere and its allowable impurities (e.g., moisture); corrosion protection; and fuel element spacing, support and protection.</p> <p><u>Physical Constraints:</u> These are the limits placed on the physical characteristics of the cask due to pre-established ISFSI interfaces: size limits due to at-reactor handling and decontamination facilities; weight limits due to at-reactor crane constraints and design parameters of onsite transporter, roadways, storage pad and placement crane; and cask appendages required to match any existing cask handling; monitoring, and servicing hardware and subsystems.</p> <p><u>Material Considerations:</u> In order to assure material compatibility with its use and environment, consider the following topics: limits on degradation due to radiation damage, weathering, and temperature extremes; corrosion resistance and compatibility with fuel element materials; and qualification of uncoded materials.</p> <p><u>Mechanical Requirements:</u> These following features are required for the proper operation of the casks: support structure design constraints to preclude spent fuel loading damage, to control criticality, and to assure pest-storage and recovery operation fuel element removal; cask appendages required to interface with cask handling, monitoring, servicing, closure, testing, onsite transport, and placement hardware and subsystem design; and cask lid closure and sealing requirements to control releases.</p>	<p>RESPONSE: The information requested by this question is contained in the GNSI Topical Report.^(e)</p>	<p>See submittal.⁽³²⁵⁾ Q and R pertain to SAR Section 3.0, "Principal Design Criteria."</p>

TABLE D.1. (contd)

Questions/Requests (a,b)	Responses (a,b)	Comments (a,b)
<u>QUESTION 3.1.1: (contd)</u>		
<u>Structural Requirements:</u> The following requirements identify the forces the cask must be able to resist:		
<ul style="list-style-type: none"> a) Identify all individual site and system related environmental and operational loads to be considered in the cask's structural analysis. (These loads should be quantified, or identified in such a manner that they can be readily quantified by the cask designer.) b) Specify the required variance for each load to be considered in the structural analysis. c) For each different operational or environmental condition to be analyzed, provide the applicable combined loading equations with the site and system defined loads inserted. d) List operational and environmental conditions not considered in the loading equations and the ISFSI design feature that precludes their consideration. (For example: "Impact loads due to casks overturning during the design earthquake are not considered. The casks are placed on a Seismic Category 1 concrete pad, and the following analysis shows that casks of a L/D ratio of less than 4 to 1 will not tip over due to the motion of the concrete pad during by the design earthquake. All cask having a L/D ratio in excess of 4 to 1 will be anchored to the concrete pad to prevent overturning due to the design earthquake." The mentioned analysis is then presented.) e) Specify and special structural code requirements. 		
<u>Thermal Characteristics:</u> Thermal performance specification for individual casks may include: amount of heat to be used dissipated to the atmosphere under limiting operational and environmental conditions; the required heat capacity available to accommodate thermal fluctuations beyond the limiting conditions, during transient operational modes, and for accident recovery operations; and allowable peak temperatures within the cask.		
<u>Nuclear and Radiological Characteristics:</u> Provide the limiting specification for: neutron and gamma shield requirements based on ALARA studies for all operational modes; allowable leak rates for gaseous and particulate material; and criticality considerations.		
<u>Special Features:</u> Provide the requirements for special features associated with the cask. These could include:		

TABLE D.1. (contd)

Questions/Requests ^(a,b)	Responses ^(a,b)	Comments ^(a,b)
<u>QUESTION 3.1.1: (contd)</u>		
cask atmosphere purging appendages; monitoring instrumentation for stored spent fuel condition, cask internal atmosphere, and cask condition; material accountability seals; decontamination and recovery features; and lightning protection.		
<u>QUESTION 3.1.2⁽¹⁶⁾</u> : The various design criteria and performance specification identified in response to Question 3.1.1 will be reflected in actual design and manufacture of the storage casks. For each requirement identified in the response to Question 3.1.1, specify the actions that VEPCO will take to assure that the requirement is properly executed by the cask supplier. These actions could include: specific quality control requirements in a quality assurance program, code usage and stamping, analytical verifications, acceptance tests, and prototype testing. Keep in mind that a performance characteristic or a design requirement need not be verified directly, it can be verified by qualifying a related characteristic or feature that is a good indicator for the basic requirement.	<u>RESPONSE</u> : The Dry Spent Fuel Storage Casks are categorized as Category I equipment. As such the VEPCO Nuclear Power Station Quality Assurance Manual (NASQAM) delineates the requirements for the engineering, procurement, fabrication, and inspection of this equipment. The procurement documents (specifications, requisitions, etc.) are reviewed technically prior to use to ensure that the proper criteria have been specified. During the cask design phase, vendor information (drawings, specifications, procedures, etc.) are reviewed to ensure compliance with VEPCO's technical requirements. During the fabrication, VEPCO's Shop Inspectors (Quality Assurance) will visit the vendor's shop to ensure compliance with VEPCO's requirements and to witness parts of the cask fabrication and testing. Until VEPCO is satisfied that the cask meets our technical requirements the vendor may not ship the cask to VEPCO.	See submittals. ^(17,325) Q and R pertain to SAR Section 11.0, "Quality Assurance."
	This same system has been used successfully by VEPCO since Surry Units Nos. 1 and 2 were licensed to purchase nuclear grade components and is described in both VEPCO's Quality Assurance Program Topical Report VEP-1-4A and the VEPCO Nuclear Power Station Quality Assurance Manual.	
	Surry ISFSI SAR Section 11.1, "Quality Assurance Program Description-Virginia Power," has been revised to reflect the response to this question.	
<u>QUESTION 3.2.1⁽¹⁶⁾</u> : Provide a detailed analysis of the stresses in stored spent fuel elements due to the forces caused by a cask tip-over during the design earthquake. This analysis should include the basis for all assumptions and the factor of safety between the resultant stresses and the threshold stress for fuel damage. The verifiable post-reactor condition of the fuel should be quantified. That is, what allowance is made for the difference in structural integrity between irradiated and	<u>RESPONSE</u> : See response to Question 1.4.1, above.	See submittals. ^(17,325) Q and R pertain to SAR Section 2.6, "Geology and Seismology."

TABLE D.1. (contd)

Questions/Requests ^(a,b)	Responses ^(a,b)	Comments ^(a,b)
<u>QUESTION 3.2.1: (contd)</u>		
unirradiated fuel. How is this difference verified in terms of accepting individual spent fuel elements for storage in a cask.		
<u>QUESTION 3.2.2(16)</u> : In lieu of responding to Question 3.2.1, or if the resultant factor of safety identified from the analysis is too low; the concrete storage pad can be upgraded to a Seismic Category 1 item. If this is done, provide an appropriate structural analysis of the concrete slab and its supporting soil.	<u>RESPONSE</u> : See response to Question 1.4.1, above.	See submittals.(17,325) Q and R pertain to SAR Section 2.6, "Geology and Seismology."
<u>QUESTION 3.3.1(16)</u> : What are the physical devices used on the handling equipment to limit impact loads during normal and off-normal operating conditions?	<p><u>RESPONSE</u>: The casks are designed to withstand potential conditions experienced during normal or off-normal handling as described in SAR Section 3.3, "Safety Protection Systems," and 5.2, "Fuel Handling Systems." Operating procedures, where necessary, will ensure that the casks are handled within these limits. The transporter equipment utilized in the handling of casks will be selected based on adequacy for the operations to be performed, and will be verified in writing to be in compliance with all applicable codes, and standards, prior to cask handling operations. This will be established by prior Engineering review of the entire movement and documented in formal procedures. Adequate supervision, engineering and health physics coverage will be provided to ensure that the equipment is used properly and that pre-written operating procedures are followed. These procedures will include:</p> <ol style="list-style-type: none"> 1. Location and stable position on the transporter (number, type, location and strength requirements of attachments). 2. Maximum speed of the transporter. 3. Required plant support groups to be present during the move. 4. Organization chart of responsible parties. 5. Defined haul path. 6. Allowable environment limits (high wind etc.) 7. Maximum height above surface(s) which may be employed during all cask motions. 8. Reference to proper positioning, lowering, and leveling procedures through load release. 9. Check lists required for all milestone points throughout move. 	See submittals.(306,325) Q and R pertain to SAR Section 5.0, "Operations Systems."

TABLE D.1. (contd)

Questions/Requests ^(a,b)	Responses ^(a,b)	Comments ^(a,b)
<u>QUESTION 3.3.1: (contd)</u>	<p><u>RESPONSE: (contd)</u></p> <p>10. An emergency list of requirements for a dropped cask will be developed.</p> <p>These would be applicable for movement from the plant to the ISFSI, for movements of cask(s) within the ISFSI, or return trips from the ISFSI to the plant. During transport, it is not envisioned necessary but would be acceptable to stop the transporter and/or rest the cask on the ground for a short time (e.g., a day). These contingencies and associated actions such as temporary security, health physic coverage, cleaning etc., will be included in the procedures.</p> <p>Therefore, no physical devices are required for the handling equipment to limit impact loads or lifts.</p> <p>The response to this question has been incorporated into Surry ISFSI SAR Section 5.2, "Fuel Handling System."⁽³²⁵⁾</p>	
<u>QUESTION 3.3.2⁽¹⁶⁾</u> : What are the physical devices used to prevent lifts in excess of those specified for handling equipment.	<p><u>RESPONSE:</u> The response to Question 3.3.2 is included in the response to Question 3.3.1, above.</p> <p>The response to this question has been incorporated into Surry ISFSI SAR Section 5.2, "Fuel Handling Systems."</p>	See submittals. ^(306,325) Q and R pertain to SAR Section 5.0, "Operations System."
<u>QUESTION 3.3.3⁽¹⁶⁾</u> : Provide the details of requalification activity for the cask and spent fuel following an off-normal handling accident.	<p><u>RESPONSE:</u> The off-normal handling accident has been evaluated and is described in the GNSI Topical Report. The analysis shows that there will be no damage to the cask body and basket integrity.</p> <p>If an off-normal handling accident were to occur, the following steps will be taken:</p> <ol style="list-style-type: none"> 1. Health Physics personnel will perform radiation surveys of the cask. 2. A visual inspection of the cask body will be performed with particular attention to the area of the lifting trunion. The trunion will be removed and replaced if required. 3. The cask will be moved to the Surry Power Station decontamination building where the secondary and primary lids will be leak tested. 	See submittal. ⁽³²⁵⁾ Q and R pertain to SAR Section 8.2, "Accidents."

TABLE D.1. (contd)

Questions/Requests ^(a,b)	Responses ^(a,b)	Comments ^(a,b)
<u>QUESTION 3.3.3: (contd)</u>	<u>RESPONSE: (contd)</u>	
	4. After the secondary lid is removed, a gas sample will be obtained from the interior of the cask body to check for an unusual amount of Kr-85.	
	5. If there is no primary lid seal damage and no Kr-85 present at a level indicating fuel failure, the cask will be resealed using normal procedures and moved back to the ISFSI and placed on the storage pad.	
	6. If it is determined that the fuel must be removed from the cask, the interior of the cask will be flooded and then the water in the interior will be sampled prior to placing in the pool. If the water sample shows unacceptable levels the water will be drained and processed as radwaste. The cask will then be reflooded and resampled prior to removing the lid to prevent uncontrolled releases of contamination to the fuel pool water.	
	7. The cask will then be moved into the fuel pool; the primary lid and the fuel will be removed.	
	8. If the fuel was removed due to the detection of potential fuel damage, the fuel will be inspected and any fuel assemblies containing rods with cladding damage will be identified as being damaged and these assemblies will be stored in the fuel pool.	
	9. If the fuel was removed due to seal damage, the cask will be removed from the pool and repaired prior to further use.	
	10. The cask will then be reloaded with fuel using normal procedures and will be moved back to the ISFSI and placed on the storage pad.	
	ISFSI SAR Section 8.2.10, "Cask Drops," has been revised to reflect the response to this question.	

TABLE D.1. (contd)

Questions/Requests ^(a,b)	Responses ^(a,b)	Comments ^(a,b)
<p>QUESTION 4.1.1E⁽¹⁶⁾: What was the neutron spectrum and flux-to-dose response used to estimate the neutron surface dose rate and dose versus distance (ER Figure 3.5-2, "Dose Rate from One Cask of 5-Year-Old Spent Fuel Versus Distance," and ER Figure 3.5-3, "Normalized Surface Dose Rate on Cask Versus Age of Spent Fuel") for a cask?</p>	<p>RESPONSE: Design criteria for the Sealed Surface Storage Casks (SSSCs) limit the surface doses to 100 mrem per hour neutron and 20 mrem per hour gamma. Neutron and gamma dose rates, normalized to these design surface dose rates, versus distance from the cask were provided in ER Figures 3.5-1 and 3.5-2 (SAR Figures 7.3-3 and 7.3-4), "Dose Rate from One Cask of Five-Year-Old Spent Fuel Versus Distance [0 to 140 ft]" and "Dose Rate from One Cask of Five-Year-Old Spent Fuel Versus Distance [0 to 700 ft]", respectively, for 5-year old fuel. The neutron source assumed within the SSSCs was provided in ER Table 3.5-2 (SAR Table 7.2-1), "Average Neutron Source for Westinghouse 15x15 Fuel."</p> <p>The above referenced figures have been revised (see attachment) to reflect dose rate analyses for a GNSI CASTOR V cask. The dose rate versus distance curves for the neutron transport were generated using a series of adjoint ANISN^(x) calculations in an infinite air medium. The adjoint fluxes from these calculations were folded with the flux leakage spectra which were calculated based on data supplied by GNSI.^(y) The neutron leakage spectra, which is based on a burnup of 35,000 MWd/MTU with an initial enrichment of 3.7 wt% uranium-235, is representative of fuel discharged from the Surry Power Station and was used to calculate the dose rates shown in ER Figures 3.5-1 and 3.5-2.</p> <p>For the gamma-ray transport, simple point-kernel calculations using infinite-medium dose rate build-up factors in dry air were performed. The gamma photon point source was also constructed from the photon leakage spectra calculated based on data provided by the GNSI. The flux-to-dose response functions used in the analyses were taken from ANSI/ANS-6.1.1-1977.^(z)</p> <p>The results for the total dose rates (neutron plus gamma) are presented in the attached new ER Figure 3.5-4 (SAR Figure 7.3-5), "Total Dose Rate from 84 GNS Casks Versus Distance [0 to 9000 ft]." These dose rates assume a full 84-cask configuration, adjusted for decay.</p>	<p>See submittals. (309,325) Q and R pertain to SAR Section 7.0, "Radiation Protection," and ER Section 3.0, "The Facility."</p>

TABLE D.1. (contd)

Questions/Requests (a,b)	Responses (a,b)	Comments (a,b)
<u>QUESTION 4.1.1E: (contd)</u>	<u>RESPONSE: (contd)</u>	
	Applicable portions of the above information will be included in SAR Sections 7.2, "Radiation Sources," and 7.3, "Radiation Protection Design Features," and ER Sections 3.5, "Radwaste Systems and Source Terms" and 4.4, "Radioactivity." ⁽³⁰⁹⁾	
	Surry ISFSI SAR Section 7.3, "Radiation Protection Design Features," and Surry ISFSI ER Section 3.5, "Radwaste Systems and Source Terms," have been revised to reflect the response to this question. ⁽³²⁵⁾	See submittals. ^(309,325) Q and R pertain to SAR Section 7.0, "Radiation Protection," and ER Section 3.0, "The Facility."
<u>QUESTION 4.1.2E⁽¹⁶⁾</u> : What were the assumptions used in postulating the steel-lead-water wall used to shield the fuel for calculating the gamma surface dose rate and energy spectrum?	<u>RESPONSE:</u> The GNSI cask utilizes steel and polyethylene instead of steel-lead-water to shield the neutron and gamma radiation. The GNSI Topical Report ^(e,y) provides detailed discussion of the design features as well as the structures of the cask.	
	The assumptions used in calculating the cask surface dose rates and energy spectrum are discussed in Section 3.3.5, "Radiological Protection," and Section 7, "Radiation Protection," of the GNSI Topical Report. ^(e) A discussion of the calculation of neutron and gamma dose rates has been provided in the response to Question 4.1.1E (above) and in revised Surry ISFSI SAR Section 7.3, "Radiation Protection Design Features," and revised Surry ISFSI ER Section 3.5, "Radwaste Systems and Source Terms."	See submittals. ^(306,309,325) Q and R pertain to SAR Section 7.0, "Radiation Protection," and ER Section 3.0, "The Facility."
<u>QUESTION 4.1.3E⁽¹⁶⁾</u> : What computer code was used to generate the data in Table 3.5-1, "Photon Spectrum as a Function of Time for Fission Products," and Table 3.5-2, "Average Neutron Source for Westinghouse 15x15 Fuel," of the ER and Table 7.2-1, "Average Neutron Source for Westinghouse 15x15 Fuel," Table 7.2-2, "Average Photon Sources 150 Days After Discharge for Westinghouse 15x15 Fuel," Table 7.2-3, "Photon Spectrum as a Function of Time for Fission Products," and Table 7.2-4, "Photon Spectrum as a Function of Time for Fission Products," of the SAR?	<u>RESPONSE:</u> The data for Table 3.5-1 and 3.5-2 of the ER and Tables 7.2-1, 7.2-2, 7.2-3 and 7.2-4 of the SAR were calculated by Westinghouse using ORIGEN II computer code. The neutron and gamma sources used in the GNSI shielding analysis and the revised dose rate versus distance curves [ER Figures 3.5-1 and 3.5-2 (SAR Figures 7.3-3 and 7.3-4)] in the response to Question 4.1.1E (above) were also generated with the ORIGEN code.	
	The response to this question has been incorporated in Surry ISFSI SAR Section 7.2, "Radiation Sources," and Surry ISFSI ER Section 3.5, "Radwaste Systems and Source Terms."	

TABLE D.1. (contd)

Questions/Requests (a,b)	Responses (a,b)	Comments (a,b)
<p>QUESTION 4.1.4⁽¹⁶⁾: How will vendors demonstrate compliance with the cask surface dose rate criteria? What other computer codes or calculational techniques will be acceptable to VEPCO to demonstrate compliance? What are the key input parameters needed to demonstrate compliance?</p>	<p>RESPONSE: Section 3.3.5, "Radiological Protection," and Section 7, "Radiation Protection," of the GNSI Topical Report^(e,y) provide a detailed discussion of the standard computer codes and calculational techniques used in determining the GNSI cask surface dose rates. The calculated cask surface dose rates of 13.9 mrem per hour (neutron) and 16.5 mrem per hour (gamma) comply with the surface dose rate criteria established in ER Section 3.5, "Radwaste Systems and Source Terms," and SAR Section 3.3.5, "Radiological Protection." The use of standard, industry-accepted computer codes is acceptable to Virginia Power for demonstrating compliance of the SSSCs with the design criteria for surface dose rate provided in ER Section 3.5, "Radwaste Systems and Source Terms." The SSSC vendors have been provided with Westinghouse source data for use in their computer analyses. The key input data to be used in the calculations are the spent fuel design information (source term data) and the cask design information. Westinghouse source data, representative of the fuel to be stored in the SSSCs, have been utilized in the shielding calculations.</p>	<p>See submittals. (306,309,325) Q and R pertain to SAR Section 3.3, "Safety Protection Systems," and ER Section 3.0, "The Facility."</p>
<p>QUESTION 4.2.1E⁽¹⁶⁾: Section 7.3.2, "Shielding," of the SAR says, "Except during cask placement and scheduled surveillance the ISFSI will not be normally occupied." What about operations at the LLWSF collocated with the ISFSI? No occupational exposure from the ISFSI to workers involved in operations at the LLWSF have been provided. Provide estimated occupational exposures to workers at the LLWSF from the ISFSI and the basis for your calculations.</p>	<p>RESPONSE: The dose rates at the LLWSF from the ISFSI are time-dependent since an average of four casks per year are expected to be placed on the storage pads and the stored fuel decays as a function of the storage period. The highest dose rate from the ISFSI will occur after all the casks have been placed.</p> <p>To estimate the additional occupational exposure which may be received by workers at the LLWSF, several assumptions were made, following:</p> <ul style="list-style-type: none"> All 84 casks are located 145 feet from the LLWSF. (The average distance between the casks and the LLWSF is approximately 145 feet). Credit was taken for the decay of the cask surface dose rates which, for 5-year-old fuel, have been calculated by GNSI to be 13.9 mrem per hour (neutron) and 16.5 mrem per hour (gamma) at the time of cask placement. Refer to Table 3.3-9, "CASTOR V/21 Dose Rates Versus Distance [0 to 2.0 meters]" of the GNSI topical report.^(y) 	<p>See submittals. (309,325) Q and R pertain to SAR Section 7.0, "Radiation Protection," and ER Section 4.0, "Environmental Effects of Site Preparation and Facility Construction."</p>

TABLE D.1. (contd)

Questions/Requests ^(a,b)	Responses ^(a,b)	Comments ^(a,b)
<u>QUESTION 4.2.1E: (contd)</u>	<p><u>RESPONSE: (contd)</u></p> <ul style="list-style-type: none"> • Credit was taken for air attenuation of the neutron and gamma dose rates as discussed in the response to Question 4.1.1E (above) • No credit was taken for the shielding effect of one cask behind another. • The manhours normally spent at the LLWSF were estimated, based on historical data, to be approximately 2160 manhours per year. This estimate includes the total amount of time spent by workers at the LLWSF delivering and preparing waste for shipment. • No credit was taken for the shielding afforded by the LLWSF building. <p>Based on the above, the total annual occupational exposure received at the LLWSF due to the ISFSI was calculated to be 8.9 man-rem (neutron) and 0.9 man-rem (gamma) for a total of 9.8 man-rem.</p> <p>For additional information on occupational exposure resulting from the ISFSI, see the response to Question 4.4.2E (below).</p> <p>The response to this question has been incorporated into Surry ISFSI SAR Section 7.4, "Estimated Onsite Collective Dose Assessment," and ER Section 4.4, "Radioactivity."</p>	See submittals. (309,325) Q and R pertain to SAR Section 7.0, "Radiation Protection."
<u>QUESTION 4.2.2⁽¹⁶⁾: Provide an ALARA justification for collocating the LLWSF and the ISFSI.</u>	<p><u>RESPONSE:</u> The ALARA justification for collocating the LLWSF and the ISFSI is based on the following considerations:</p> <ul style="list-style-type: none"> • The ISFSI and LLWSF are centrally located within the Surry site boundary, thus minimizing offsite exposures. • The centralized location of the ISFSI and the LLWSF is of sufficient distance from the Surry Power Station such that the increased dose to Surry Station personnel is not significant. The annual occupational exposure to Surry site personnel from the completely filled ISFSI is only 1.5 man-rem compared with 1490 man-rem from normal Surry Plant operation.^(aa) This increase 	

TABLE D.1. (contd)

Questions/Requests (a,b)	Responses (a,b)	Comments (a,b)
<p><u>QUESTION 4.2.2: (contd)</u></p>	<p><u>RESPONSE: (contd)</u></p> <p>exposure to Surry Station personnel is less than one-tenth of one percent.</p> <ul style="list-style-type: none"> The LLWSF is a facility that has limited occupancy, and, as such, represents a low exposure potential for personnel. In addition, the dose rates to workers from sources within the LLWSF are much greater than those that will result from ISFSI operations. A proven heavy load route has been built past the LLWSF and a perimeter fence has already been built. Both of these are also needed for the ISFSI. <p>The response to this question has been incorporated into Surry ISFSI SAR Section 7.1, "Ensuring that Occupational Radiation Exposures Are As Low As Is Reasonably Achievable (ALARA)."(325)</p>	
<p><u>QUESTION 4.2.3⁽¹⁶⁾</u>: Section 7.3.2.2, "Dose Rate Versus Distance," of the SAR indicates that the dose rate analysis at the restricted area fence does not include the contribution from the LLWSF. Revise Table 4.4-1, "Dose Rates Along Fence During ISFSI Operation," of the ER and Table 7.3-1, "Dose Rates Along Fence During ISFSI Operation," of the SAR to include the contribution from the LLWSF.</p>	<p><u>RESPONSE:</u> ER Table 4.4-1, "Dose Rates Along Fence During ISFSI Operation," and SAR Table 7.3-1, "Dose Rates Along Fence During ISFSI Operation," attached, have been revised to include the maximum dose rate contribution from the LLWSF (0.6 mrem per hour at the ISFSI restricted area fence). This is conservative since the contribution to the dose rate from the LLWSF is dependent on the extent to which the LLWSF is used and will be generally less than this design maximum value. The revised tables also report the dose rates from the ISFSI based on the cask surface dose rates in Section 3.3 of the GNSI topical report.(Y)</p> <p>As shown in the revised tables, even with the maximum dose rate contribution from the LLWSF, the restricted area limit of 2 mrem per hour (10 CFR 20.105) will not be exceeded at any time during ISFSI operation. With the GNSI casks, the proposed extension of the fence as shown in ER Figure 4.4-1 and SAR Figure 7.3-1 would not be necessary.</p> <p>Applicable portions of the above information will be included in SAR Section 7.3.2, "Shielding," and ER Section 4.4, "Radioactivity."(309) SAR Section 7.3.2.2, "Dose Rate Versus Distance," will also be</p>	<p>See submittals.(309,325) Q and R pertain to SAR Section 7.0, "Radiation Protection," and ER Section 4.0, "Environmental Effects of Site Preparation and Facility Construction."</p>

TABLE D.1. (contd)

Questions/Requests ^(a,b)	Responses ^(a,b)	Comments ^(a,b)
QUESTION 4.2.3E: (contd)	RESPONSE: (contd)	
	corrected to reference the 0.6 mrem per hour design maximum LLWSF dose rate contribution. ⁽³⁰⁹⁾	
	Surry ISFSI SAR Section 7.3, "Radiation Protection Design Features," and ER Section 4.4, "Radioactivity," have been revised to include the contribution from the LLWSF. ⁽³²⁵⁾	
QUESTION 4.2.4E ⁽¹⁶⁾ : In Section 7.4, "Estimated Onsite Collective Dose Assessment," of the SAR, no total collective occupational dose is provided. How many workers at the Surry Power Station will receive the additional 56 mrem yr from the ISFSI? What is the additional occupational dose to workers at the LLWSF from the ISFSI: What are the bases for the dose rate estimate used in SAR Tables 7.4-1, 7.4-2 and 7.4-3 (ER Tables 4.4-2, 4.4-3 and 4.4-4), which are "Occupational Exposures for Cask Loading, Transport, and Emplacement (One Time Exposure)," "Surry ISFSI Maintenance Operations," and "Occupational Exposures from Construction," respectively.	RESPONSE: The additional exposure to workers at the Surry Power Station (SPS) from ISFSI operations has been estimated as described below. All workers at the Surry Power Station in offices, non-shielded buildings, or in the plant yard were considered in the occupational dose calculation. This population includes a normal work force of utility and contractor personnel as well as the increased staffing required during outages. As a bounding estimate, the total number of workers assumed was 600, spending a total of 1,248,000 manhours per year in the SPS yard area and offices.	See submittals. ^(309,325) Q and R pertain to SAR Section 7.0, "Radiation Protection," and ER Section 4.0, "Environmental Effects of Site Preparation and Facility Construction."
In SAR Table 7.4-3 (ER Table 4.4-4), what is the occupational dose due to excavation and construction? Provide a total collective occupational dose for ISFSI operations and assess how it affects the collective occupational dose at the Surry Power Station.	Based on calculations performed in response to Question 4.1.1E (above), the dose rates from the ISFSI to the yard location 2100 feet away are 1.1E-3 mrem per hour (neutron) and 1.1E-4 mrem per hour (gamma). The collective annual exposure for station workers due to the ISFSI is therefore calculated to be 1.37 man-rem (neutron) and 0.137 man-rem (gamma). This results in an average annual exposure per worker of 2.3E-3 man-rem (neutron) and 0.23E-4 man-rem (gamma).	
	SAR Table 7.4-1, "Occupational Exposures for Cask Loading, Transport, and Emplacement (One Time Exposure)," and ER Table 4.4-2, "Occupational Exposures for Cask Loading, Transport, and Emplacement (One Time Exposure)," attached, which list occupational exposures for cask loading, transport and emplacement, have been revised ⁽³⁰⁹⁾ to reflect the cask surface dose rates in Section 3.3, "Protection By Equipment and Instrumentation Selection," of the GNSI cask topical report. ⁽⁷⁾ The design surface dose rates of 13.9 mrem per hour (neutron) and 16.5 mrem per hour (gamma) were utilized even though during cask transfer individuals will be at least 10 feet from the cask. During preparation for emplacement, individuals are close to the cask and	

TABLE D.1. (contd)

Questions/Requests ^(a,b)	Responses ^(a,b)	Comments ^(a,b)
QUESTION 4.2.4E: (contd)	<p data-bbox="940 318 1146 342">RESPONSE: (contd)</p> <p data-bbox="940 367 1520 529">are assumed to be exposed to the design cask surface dose rates and the dose rate contributions from two full storage pads and from the LLWSF. During final positioning, the individual is also assumed to receive dose rate contributions from the four newly placed casks. For these cases, neutron and gamma rates are combined.</p> <p data-bbox="940 553 1520 1081">SAR Table 7.4.2, "Surry ISFSI Maintenance Operations (Annual Exposures) and ER Table 4.4-3, "Surry ISFSI Maintenance Operations (Annual Exposures)," have also been revised⁽³⁰⁹⁾ to reflect the cask surface dose rates tabulated in the GNSI cask topical report.^(y) The dose rates in these tables were calculated assuming all storage pods were filled with casks. Neutron and gamma dose rates were combined and the contribution from the LLWSF was also considered. To estimate the dose rates for operability tests and calibration, the individual location was assumed to be at the control panel location at the perimeter fence entrance. Visual surveillance was based on a walkdown of each of the three pads at a distance no closer than 2 meters to the casks. During instrument repairs, the worker was assumed to be positioned between two rows of casks. The six surrounding casks (all within 16 feet of the worker) were the predominant dose contributors during repair work, but dose rates from the other pads and the LLWSF were also considered.</p> <p data-bbox="940 1105 1520 1435">SAR Table 7.3-4, "Occupational Exposures From Construction," and ER Table 4.4-4, "Occupational Exposures From Construction," attached, have been revised⁽³⁰⁹⁾ to include man-hours for excavation of each slab to reflect the new GNSI cask surface dose rates. These tables are based on constructing each slab in sequence as the previous slab becomes filled. Slab 1 considers the design maximum dose rate from the LLWSF. Slab 2 considers the dose rate from the LLWSF and the 28 casks on Slab 1. Slab 3 considers the dose rate from the LLWSF plus the dose rate from all of the casks on Slabs 1 and 2. The manhours for excavation for each slab are 2060.</p>	

TABLE D.1.. (contd)

Questions/Requests ^(a,b)	Responses ^(a,b)	Comments ^(a,b)
<u>QUESTION 4.2.4E: (contd)</u>	<u>RESPONSE: (contd)</u>	
	<p>The collective occupational dose for ISFSI operations is provided in new Table 7.4-4, "Annual Doses from ISFSI Operations," of the SAR and new Table 4.4-5, "Annual Doses from ISFSI Operations," of the ER (attached).⁽³⁰⁹⁾ The man-rem listed in these tables are the annual collective doses which represent worst-case neutron and gamma dose rate contributions from the total facility capacity of 84 casks. As evidenced in these tables, the man-rem resulting from ISFSI operations would only be approximately 1.6 percent of the annual man-rem for operation of the Surry Power Station Units 1 and 2 without the ISFSI.^(aa)</p> <p>The response to this question has been incorporated into Surry ISFSI SAR Section 7.4, "Estimated Onsite Collective Dose Assessment," and ER Section 4.4, "Radioactivity."</p>	
<p><u>QUESTION 4.3.1E⁽¹⁶⁾</u>: SAR Section 7.6 is titled "Estimated Offsite Collective Dose Assessment," yet no collective offsite dose is evaluated. Only a maximum dose to an individual located at 1.53 miles is given. Figure 2.1-3, "Population Distribution: 0-10 and 10-50 Miles-1980," in the SAR indicates that 3 people reside within the 0-1 mile annulus and 49 people in the 1-2 mile annulus of the Surry Power Station. Explain the discrepancy about the location of the nearest individual and provide an offsite collective dose assessment.</p>	<p><u>RESPONSE:</u> As shown in Figure 1 of the response to Question 1.1.5E (above), 48 permanent residents are located in the 0-2 mile annulus based on latest available data. The nearest permanent resident is located at 1.53 miles from the site. Figure 2.1-3 of the SAR will be replaced as described in the response to Question 1.1.5E.</p> <p>Based on the dose rate versus distance calculations performed for the response to Question 4.1.1E and the assumption that all of the residents in the 0-2 mile annulus are located at the same distance from the ISFSI as the nearest resident at 1.53 miles, the collective annual dose from ISFSI operations would be 9.6E-8 man-rem (neutron) and 2.5E-7 man-rem (gamma). The total collective offsite dose would be 3.5E-7 man-rem per year. These doses assume a total of 84 casks and are adjusted for decay.</p> <p>The rapid attenuation of neutron and gamma dose rates beyond a 2-mile radius makes the contribution to the total collective dose from more distant populations negligible. For further information on offsite doses from the ISFSI, see the response to Question 4.3.3E (below).</p>	<p>See submittals.^(309,325) Q and R pertain to SAR Section 7.0, "Radiation Protection," and ER Section 5.0, "Environmental Effects of Facility Operation."</p>

TABLE D.1. (contd)

Questions/Requests (a,b)	Responses (a,b)	Comments (a,b)
<u>QUESTION 4.3.1E: (contd)</u>	<u>RESPONSE: (contd)</u> Surry ISFSI SAR Figure 2.1-3, "Population Distribution: 0-10 and 10-50 Miles-1980," has been revised and a collective offsite dose has been calculated accordingly and is reported in revised SAR Section 7.6, "Estimated Offsite Collective Dose Assessment," and ER Section 5.2, "Radiological Impact from Routine Operation."	
<u>QUESTION 4.3.2E</u> ⁽¹⁶⁾ : Relative to SAR Section 7.6.1, "Effluent and Environmental Monitoring Program," the principle contributor to dose from the ISFSI will be neutrons, and the information about environmental monitoring does not provide enough information about capabilities for measuring doses from neutrons around the restricted area fence. Describe the type, number and locations of the TLD's to be placed around the ISFSI restricted area fence.	<u>RESPONSE:</u> As indicated in SAR Section 7.0, "Radiation Protection," portable neutron and gamma survey meters will normally be used for cask surveillance operations. In addition, to provide a continual monitoring capability, at least two gamma-sensitive TLDs will be placed on each side of the ISFSI controlled area fence. In order to assess the neutron dose from the ISFSI, a correlation between measurements from the portable survey meters and the TLDs will be used. The neutron dose rate at the perimeter fence around the ISFSI will be measured using the neutron survey meters and will be compared to the gamma dose rate measured using the gamma survey meters. The integrated gamma dose recorded by the TLDs mounted on the controlled area fence can be used to estimate the integrated neutron dose by using the ratio of the gamma and neutron dose rates measured in the same location by the portable survey meters. By using this basic procedure, the neutron dose to the environment from the ISFSI can be reliably determined. The response to this question has been incorporated into Surry ISFSI SAR Section 7.6, "Estimated Offsite Collective Dose Assessment," and ER Section 6.2, "Proposed Operational Monitoring Programs."	See submittals. (309,325) Q and R pertain to SAR Section 7.0, "Radiation Protection," and ER Section 6.0, "Effluent and Environmental Measurements and Monitoring Programs."
<u>QUESTION 4.3.3E</u> ⁽¹⁶⁾ : Relative to SAR Section 7.6.2, "Analysis of Multiple Contribution," if the maximum dose to an individual from ISFSI operations were added to the design objective doses specified in 10 CFR Part 50, Appendix I, for releases of radioactive material from reactor operation, the limits of 10 CFR 72.67 and 40 CFR Part 190 could be exceeded. Considering the uncertainty in calculating the neutron dose rate at distances of	<u>RESPONSE:</u> For the purpose of offsite exposure analyses, dose rate versus distance calculations have been performed for the response to Question 4.1.1E (above). These curves utilize the cask surface dose rates listed in Table 3.3-6 of the GNSI cask topical report ^(y) and also account for air attenuation of the neutron and gamma dose rates. Using these curves and assuming a total	See submittals. (309,325) Q and R pertain to SAR Section 7.0, "Radiation Protection," and ER Section 5.0, "Environmental Effects of Facility Operation."

TABLE D.1. (contd)

Questions/Requests ^(a,b)	Responses ^(a,b)	Comments ^(a,b)
<u>QUESTION 4.3.3E: (contd)</u>	<u>RESPONSE: (contd)</u>	
<p>1.5 miles, and the difficulty of measuring such low doses from neutrons; how will you ensure that the dose to an individual from ISFSI operations when combined with doses from reactor operations does not exceed these limits? Provide your method of demonstrating, by calculational procedures based on models and data, that the actual dose of an individual from ISFSI operations when combined with the doses from Surry Power Station Units 1 and 2 and other Uranium Fuel Cycle facilities does not exceed the 25 mrem/yr limit specified in 10 CFR 72.67 and 40 CFR Part 190.</p>	<p>of 84 casks adjusted for decay as shown in Table 3.3-9 of the GNSI cask topical report,^(Y) the annual dose to the nearest permanent resident (1.53 miles away) due to ISFSI operations would be $2.0\text{E-}6$ mrem from neutron and $5.3\text{E-}6$ mrem from gamma.</p> <p>To determine the radiological impact of ISFSI operations under the requirements of 10 CFR 72.67 and 40 CFR 190, the additional dose rate contributions from the LLWSF and the Surry Power Station (SPS) Units 1 and 2 have been considered. (Other radioactive site sources have a negligible impact on the offsite dose.) The annual dose to the nearest permanent resident from the LLWSF would be $4.4\text{E-}2$ mrem based on the design maximum LLWSF dose rate of 0.6 mrem per hour at the ISFSI controlled area fence. Using the whole-body dose guidelines from 10 CFR 50 Appendix 1, the maximum annual dose to the nearest permanent resident from the Surry Power Station would be 3 mrem due to liquid effluents and 5 mrem due to gaseous effluents for each unit. Therefore, the maximum total annual dose to the nearest permanent resident would be $7.3\text{E-}6$ mrem (ISFSI) + $4.4\text{E-}2$ mrem (LLWSF) + 16 mrem (SPS Units 1 & 2) = 16 mrem.</p> <p>It is concluded that the actual dose to the nearest permanent resident from ISFSI operations in combination with the maximum permissible dose from the Surry Power Station and other radioactive site sources does not exceed the 25 mrem per year limit specified in 10 CFR 72.67 and 40 CFR 190. The above calculation is conservative since the actual Surry Power Station effluent doses are below the 10 CFR 50 Appendix 1 limits. This is shown in Appendix 11A to the Updated FSAR for SPS Units 1 and 2.</p> <p>The response to this question has been incorporated into Surry ISFSI SAR Section 7.6, "Estimated Off-site Collective Dose Assessment," and ER Section 5.2, "Radiological Impact from Routine Operation."</p>	

TABLE D.1. (contd)

Questions/Requests (a,b)	Responses (a,b)	Comments (a,b)
<p><u>QUESTION 4.4.1</u>⁽¹⁶⁾: The procedures for the decommissioning of the ISFSI should be addressed conceptually for the purposes of demonstrating that it is a manageable task. To the extent possible, VEPCO should identify: specific levels of contamination and activation products expected at the end of useful cask life, the specific procedures anticipated for clean-up of the cask, and the expected disposition of the cask.</p>	<p><u>RESPONSE</u>: As discussed in ER Section 5.8, "Decommissioning and Dismantling," and SAR Section 9.6, "Decommissioning," the dry cask design concept utilized at the Surry ISFSI features inherent ease and simplicity of decommissioning. At the end of the service lifetime for the ISFSI, cask decommissioning could be accomplished by one of the following options:</p> <ol style="list-style-type: none"> 1. The ISFSI cask, including the spent fuel stored inside, could be shipped to a suitable fuel repository for permanent storage. Depending on licensing requirements existing at the time of shipment offsite, placement of the entire ISFSI cask inside a supplemental shipping container or overpack would be considered. 2. The spent fuel could be removed from the ISFSI cask and shipped in a licensed shipping container to a suitable fuel repository. If desirable, cask decontamination could be accomplished through the use of conventional high pressure water sprays to further reduce contamination on the cask interior. The sources of contamination on the interior of the cask would be crud from the outside of the fuel rods and the crud left by the spent fuel pool water. The expected low levels of contamination from these sources could be easily removed with a high pressure water spray. After decontamination, the ISFSI cask could either be cut up for scrap or partially scrapped and any remaining contaminated portions shipped as radioactive waste to a disposal facility. 3. For surface decontamination of the ISFSI cask, chemical etching using hydrochloric acid or nitric acid can be applied to remove the contaminated surface of the cask. Alternatively, electro-polishing can also be used to achieve the same result. <p>As discussed further in the response to Question 4.4.2E (below) and Section 3.5, "Decommissioning Considerations," of the GNSI cask topical report, the cask materials will be only slightly activated by the neutron flux emanating from the stored spent fuel. Consequently, it is expected that after application of the decontamination process as</p>	<p>See submittals. (309,325) Q and R pertain to SAR Section 9.0, "Conduct of Operations," and ER Section 5.0, "Environmental Effects of Facility Operation."</p>

TABLE D.1. (contd)

Questions/Requests ^(a,b)	Responses ^(a,b)	Comments ^(a,b)
<u>QUESTION 4.4.1: (contd)</u>	<p data-bbox="955 316 1165 341"><u>RESPONSE: (contd)</u></p> <p data-bbox="955 365 1533 479">described above, the radiation level due to activation products will be negligible and the cask could be scrapped. A detailed evaluation will be performed at the time of decommissioning to determine the appropriate mode of disposal.</p> <p data-bbox="955 511 1533 625">Since the remaining ISFSI site structures and components, including the concrete base pads, are expected to receive no activation or contamination, additional ISFSI decommissioning procedures are not required.</p> <p data-bbox="955 657 1533 795">The spent fuel pool at Surry Power Station will remain functional until the ISFSI is decommissioned. This will allow the pool to be utilized to transfer fuel from the storage casks to licensed shipping containers for shipment offsite if this decommissioning options is chosen.</p> <p data-bbox="955 820 1533 917">Surry ISFSI SAR Section 9.6, "Decommissioning," and ER Section 5.8, "Decommissioning and Dismantling," have been revised to reflect the response to this question.</p>	
<p data-bbox="262 933 882 1079"><u>QUESTION 4.4.2E⁽¹⁶⁾</u>: Provide the basis and supporting analysis for your conclusion in ER Section 5.8, "Decommissioning and Dismantling," that "the cask materials will be only very slightly activated as a result of their long-term exposures to the relatively small neutron flux..."</p>	<p data-bbox="955 933 1533 1079"><u>RESPONSE:</u> A cask activation analysis has been performed by GNSI to quantify specific activity levels of cask materials. These activation calculations, and the assumptions under which they were performed, are described in Section 3.5 of the GNSI cask topical report.^(Y)</p> <p data-bbox="955 1112 1533 1299">Based on the results of the GNSI analysis, it is concluded that the cask materials will be only very slightly activated as a result of long-term exposure to the relatively small neutron flux. Hence, as further discussed in the response to Question 4.4.1E (above), surface decontamination is expected to be adequate to decommission the casks from nuclear service.</p> <p data-bbox="955 1323 1533 1416">Surry ISFSI SAR Section 9.6, "Decommissioning," and ER Section 5.8, "Decommissioning and Dismantling," have been revised to reflect the response to this question.</p>	<p data-bbox="1533 933 1938 1063">See submittals (309,325) Q and R pertain to SAR Section 9.0, "Conduct of Operations," and ER Section 5.0, "Environmental Effects of Facility Operation."</p>

TABLE D.1. (contd)

Questions/Requests ^(a,b)	Responses ^(a,b)	Comments ^(a,b)
<u>QUESTION 4.5.1E</u> ⁽¹⁶⁾ : Other radiation from the uranium fuel cycle is included in 10 CFR 72.67(a)(3). Provide an assessment of the combined effects of the ISFSI and the Surry Power Station.	<u>RESPONSE</u> : The response to this question is provided in the response to Question 4.3.3.E (above) and has been incorporated into Surry ISFSI SAR Section 7.6, "Estimated Offsite Collective Dose Assessment," and ER Section 5.2, "Radiological Impact from Routine Operation."	See submittals. (309,325) Q and R pertain to SAR Section 7.0, "Radiation Protection," and ER Section 5.0, "Environmental Effects of Facility Operation."
<u>QUESTION 4.6.1E</u> ⁽¹⁶⁾ : More information is needed about your capabilities for assessing neutron doses in the environment.	<u>RESPONSE</u> : The response to this question is provided in the response to Question 4.3.2E (above) and has been incorporated into Surry ISFSI SAR Section 7.6, "Estimated Offsite Collective Dose Assessment," and ER Section 6.2, "Proposed Operational Monitoring Programs."	See submittals. (309,325) Q and R pertain to SAR Section 7.0, "Radiation Protection," and ER Section 6.0, "Effluent and Environmental Measurements and Monitoring Programs."
<u>QUESTION 4.6.2</u> ⁽¹⁶⁾ : Provide a complete description of any radiation monitoring program to be established for the ISFSI related activities. This discussion should address the following issues as appropriate: a) location and type of instrumentation for ISFSI storage area radiation monitoring; b) type and location of instrumentation for individual cask radiation monitoring; c) monitoring schedule; and d) high radiation alarm system.	<p><u>RESPONSE</u>: As discussed in SAR Section 3.3.5, "Radiological Protection," the totally passive and inherently safe nature of the SSCs eliminates any need for instrumentation to perform a safety function; there are no credible events which could result in unacceptable releases of radioactive materials. However, to monitor cask performance and to record dose rates from the SSCs, a radiation monitoring program for the ISFSI will be established.</p> <p>Area radiation monitoring will be performed routinely by dose integrating devices (TLDs) located on the ISFSI controlled area fence. At least two TLDs will be located on each side of the ISFSI. These are in addition to the TLDs already maintained in areas around the ISFSI as part of the environmental and radiation monitoring program for the Surry Power Station as described in SAR Section 7.6, "Estimated Offsite Collective Dose Assessment," and ER Section 6.0, "Effluent and Environmental Measurements and Monitoring Programs." No individual cask radiation monitoring is necessary.</p> <p>Occupational exposure within the ISFSI will be controlled by health physics operational procedures. All entry to the facility will be controlled by the radiation monitoring equipment and personnel dosimetry required for entry to the facility. As a minimum, gamma and neutron dose rate measuring instruments, personnel TLDs and self-reading dosimeters will be required after the placement of the first cask. In addition, routine radiation surveys of the ISFSI, as shown in SAR</p>	See submittals. (309,325) Q and R pertain to SAR Section 7.0, "Radiation Protection," and ER Section 6.0, "Effluent and Environmental Measurements and Monitoring Programs."

TABLE D.1. (contd)

Questions/Requests ^(a,b)	Responses ^(a,b)	Comments ^(a,b)
<u>QUESTION 4.6.2: (contd)</u>	<p><u>RESPONSE: (contd)</u></p> <p>Table 7.4-2, "Surry ISFSI Maintenance Operations (Annual Exposures), will be performed by health physics personnel on a monthly basis.</p> <p>Applicable portions of the above information will be included in SAR Sections 5.1 ("Operations Description"), 7.1 ("Ensuring that Occupational Radiation Exposures Are As Low As Is Reasonably Achievable (ALARA)") and 7.6 ("Estimated Offsite Collective Dose Assessment") and ER Section 6.2, "Proposed Operational Monitoring Programs").</p> <p>The ISFSI is a passive installation which requires little maintenance and contains a radioactive source of predictable source strength (as discussed in response to Question 4.1.1E, above). Calculation results presented in responses to NRC Questions 4.2.1 and 4.2.4, above, have demonstrated that the radiation levels are too low to actuate a high radiation alarm system. Thus installation of such system would be of no benefit in controlling personnel radiation exposure.</p> <p>The response to this question has been incorporated into Surry ISFSI SAR Section 7.6, "Estimated Offsite Collective Dose Assessment," and ER Section 6.2, "Proposed Operational Monitoring Programs."</p>	
<p><u>QUESTION 4.7.1⁽¹⁶⁾</u>: The radiation protection procedures related to a high radiation situation should discuss the following topics: a) safety precautions, b) personnel required (by skilled/specialty level), c) repair equipment/material required, d) provisions for workers protection, and e) provisions for protection of other personnel.</p>	<p><u>RESPONSE:</u> The SAR and the GNSI cask topical report⁽⁷⁾ describe the totally passive design and inherently safe nature of the storage casks. Since the cask is a passive system, little maintenance work is expected to be performed in the storage area other than periodic repair to the lighting, security systems, and non-safety related cask instrumentation. As discussed in Section 4.5 of the GNSI cask topical report,⁽⁷⁾ only minor maintenance of the cask is expected. These maintenance activities do not require any special personnel skills, equipment, or materials and can be performed by the plant maintenance crew with readily available materials. Personnel training will be performed as discussed in Section 9.3, "Training Program," of the SAR. Regularly scheduled surveillance is required, but calculations have shown</p>	<p>See submittals. (309,325) Q and R pertain to SAR Section 9.0, "Conduct of Operations."</p>

TABLE D.1. (contd)

Questions/Requests ^(a,b)	Responses ^(a,b)	Comments ^(a,b)
<u>QUESTION 4.7.1: (contd)</u>	<p><u>RESPONSE: (contd)</u></p> <p>that doses received by personnel from all anticipated activities are within the guidelines set forth in 10 CFR 20 (see responses to Questions 4.2.1 and 4.2.4, above).</p> <p>There are no credible events during normal storage that could lead to a high radiation condition at the ISFSI. Accident analyses in Section 8 of the GNSI cask topical report^(y) also show no credible event leading to the high radiation release during transport, placement, or storage. In the unlikely event that maintenance on the cask involving cask integrity is to be performed, the cask would be returned to the Surry Power Station and the work performed inside the station.</p> <p>Entrance to the ISFSI and all work performed inside will require a radiation work permit and will be controlled by health physics personnel. Table 7.5-5, "Virginia Electric and Power Company Health Physics (Radiation Protection) Manual for Nuclear Power Stations," of the SAR lists the Surry Power Station health physics procedures. These procedures will be revised, where applicable, prior to operation of the ISFSI. The revised procedures will ensure safety of personnel performing surveillance and maintenance at the ISFSI.</p> <p>Surry ISFSI SAR Section 9.4, "Normal Operations," has been revised to reflect the response to this question.</p>	
<p><u>QUESTION 5.1.1E⁽¹⁶⁾</u>: Report separately the capital cost, and operation maintenance cost components of the total lifetime cost reported in ER Section 9.1.3, "Store Spent Fuel at the Surry Site in Dry Storage Casks," and in ER Section 9.1.4, "Build a New Independent Spent Fuel Storage Pool at Surry."</p>	<p><u>RESPONSE:</u> The response is as follows:</p> <p>a. <u>Reference: Surry ISFSI ER Section 9.1.3, "Store Spent Fuel at the Surry Site in Dry Storage Casks."</u></p> <p>In the design proposed to store spent fuel at the Surry site in dry storage casks, the expected estimated cost over the lifetime of the project in 1982 base dollars would be a low for capital expenditures of 57.8 million and a high of 90.6 million with an operation and maintenance cost of 1.9 million.</p>	<p>See submittals. (306,325) Q and R pertain to LA Section 1.0, "General and Financial Information," and to ER Section 9.0, "Alternative Energy Sources and Sites."</p>

TABLE D.1. (contd)

Questions/Requests (a,b)	Responses (a,b)	Comments (a,b)
<p><u>QUESTION 5.1.1E: (contd)</u></p>	<p><u>RESPONSE: (contd)</u></p> <p>b. <u>Reference: Surry ISFSI ER Section 9.1.4, "Build a New Independent Spent Fuel Storage Pool at Surry"</u></p> <p>The estimated cost in 1982 base dollars to utilize the Independent Spent Fuel Storage Pool method to store the spent fuel at the Surry site would be 58.2 million for capital cost with an operation and maintenance expenditure of 53.0 million dollars.</p> <p>The projected total combined expected cost of the two components would be 111.2 million dollars. The accuracy of the estimate could vary the cost by plus or minus ten percent.</p> <p>c. <u>VEPCO-ISFSI Assumptions</u></p> <ul style="list-style-type: none"> The engineering cost for the permanent plant and equipment is included in the estimated cost for the facility. VEPCO management costs are included for the permanent plant and equipment; and the operations and maintenance cost. <p>Surry ISFSI LA Section 1.4, "Estimated Project Costs," and ER Section 9.1 "Alternatives to the Proposed Action," have been revised to reflect the response to this question.</p>	
<p><u>QUESTION 5.2.1⁽¹⁶⁾</u>: What special provisions will be added to your Quality Assurance Program to accommodate the ISFSI activity.</p> <p><u>NOTE</u>: During the development of the specific information for ISFSI Quality Assurance Program, it is suggested that you review your existing program for attributes described in the attachment "QA Checklist for Dry Storage Casks."</p>	<p><u>RESPONSE</u>: 10CFR72.80 requires that a quality assurance program be based on the criteria of 10CFR50 Appendix B. VEPCO presently has a program which meets these criteria and is described in the VEPCO Topical Report Quality Assurance Program Operations Phase VEP-1-4A which has been approved by the NRC. This program is implemented at VEPCO's nuclear power stations by the VEPCO Nuclear Power Station Quality Assurance Manual (NPSQAM). As the Dry Casks ISFSI will be located at the Surry Power Station, the ISFSI activities will fall under the NPSQAM. No changes to this program are required for the ISFSI activities.</p> <p>Surry ISFSI SAR Section 11.1, "Quality Assurance Program Description-Virginia Power," has been revised to reflect the response to this question.</p>	<p>See submittal (325) Q and R pertain to SAR Section 11.0, "Quality Assurance."</p>

TABLE D.1. (contd)

Questions/Requests (a,b)	Responses (a,b)	Comments (a,b)
<p><u>QUESTION 5.2.2</u>⁽¹⁶⁾: What special provisions will be added to your Emergency Plan to accommodate the ISFSI activity?</p> <p><u>NOTE</u>: It is suggested that an adjunct to the existing Emergency Plan add the following events: a) for the Notification of Unusual Events Category: loss of cask neutron shield, cask seal leakage, cask drop or other handling mishap, and b) for the Alert Category: loss of all fuel confinement barriers from some undefined cause.</p>	<p><u>RESPONSE</u>: The only changes that will be made to the Surry Power Station Emergency Plan will be to add the following events:</p> <ol style="list-style-type: none"> 1. For the Notification of Unusual Events <ol style="list-style-type: none"> a. Cask Seal Leakage b. Cask Drop 2. For the Alert Category <ol style="list-style-type: none"> a. Loss of Cask/Fuel Containment Barriers <p>The loss of cask neutron shield will not be addressed as it is not a credible event for the GNS cask as it is an integral part of the cask body.</p> <p>Surry ISFSI SAR Section 9.5, "Emergency Planning," have been revised to reflect the response to this question.</p>	<p>See submittal.⁽³²⁵⁾ Q and R pertain to SAR Section 9.0, "Conduct of Operations."</p>
<p><u>QUESTION (October 1, 1984)</u>⁽¹⁸⁾: The proposed revision to Section 2.6.4.8, "Liquefaction Potential," of the Surry ISFSI SAR presents the procedure, including mathematical relations, used to predict soil liquefaction potential at various depths below the proposed ISFSI. This procedure consists of evaluating the shear stresses expected during the design earthquake loading. It is stated in Subsections 2.6.4.8.2 and 2.6.4.8.3 that the calculated factors of safety are 2.5 and 1.5, respectively, for the Pleistocene Sand and Miocene Silty Sand layers.</p> <p>We understand that these liquefaction potential calculations were performed at a variety of depths below the water table. Please provide the calculational results for the range of depths examined in the two sand layers mentioned above. Please specify what specific parameters (e.g., effective overburden pressures, standard penetration field values, etc.) were used in making the above calculations. A tabular presentation form, which clearly identifies the values of the various parameters used in the calculation, as well as the calculational result, would be very useful.</p>	<p><u>RESPONSE</u>: The response to this questions have been incorporated into SAR Section 2.6, "Geology and Seismology."</p>	<p>See submittal.⁽³²⁵⁾ Q and R pertain to SAR Section 2.0, "The Site Characteristics."</p>

TABLE D.1. (contd)

Questions/Requests ^(a,b)	Responses ^(a,b)	Comments ^(a,b)
<p>QUESTION 1 (November 14, 1984)⁽³¹¹⁾: Regarding your response to Question 4.1.1E, what are the neutron energy flux response functions that were used to calculate the neutron dose rate versus distance from the cask flux leakage spectra supplied by GNSI.</p>	<p>RESPONSE: The following figures and tables have been revised and are attached hereto:</p> <p>ER Figure 3.5-1 (SAR Figure 7.3-3), "Dose Rate from One Cask of Five-Year-Old Spent Fuel Versus Distance" (0 to 140 ft)</p> <p>ER Figure 3.5-2 (SAR Figure 7.3-4), "Dose Rate from One Cask of Five-Year Old Spent Fuel Versus Distance" (0 to 700 ft)</p> <p>ER Figure 3.5-4 (SAR Figure 7.3-5), "Normalized Surface Dose Rate Versus Age of Spent Fuel"</p> <p>ER Figure 3.5-5 (SAR Figure 7.3-6), "Total Dose Rate from 84 Casks of Spent Fuel Versus Distance" (0 to 9,500 ft)</p> <p>SAR Table 7.3-2, "Neutron Response Functions and Cask Surface Fluxes"</p> <p>SAR Table 7.3-3, "Adjoint Fluxes for a 1 n/sec Per Group"</p> <p>SAR Table 7.3-4, "Total Neutron Dose Rates for Four Distances from One Cask" (50, 460, 1500 and 2460 m)</p> <p>The neutron transport results shown on the dose rate versus distance figures above were generated using a series of "adjoint"^(f,g) ANISN^(h) runs. These calculations were performed with a BUGLE-80⁽ⁱ⁾ cross section set for an infinite air medium. As explained in References (f) and (g), the adjoint method is the preferred analytical technique when more than one set of sources must be evaluated at a given detector location for a response of interest. For the adjoint analyses reported here, the "adjoint source" was the flux-to-dose conversion factor reported in SAR Table 7.3-2. Four separate ANISN analyses were performed at distances of 50, 460, 1500 and 2460 meters. The resulting adjoint fluxes were presented in SAR Table 7.3-3. These adjoint fluxes were then folded with the cask surface flux leakage spectra supplied by the cask vendor over the area of the cask's top and side.^(j) See SAR Table 7.3-2 last column. The 84 group structure in the cask vendor's leakage spectrum for the side of the cask was collapsed to the 47 group BUGLE structure prior to folding the</p>	<p>See submittals^(316,325) for figures and tables. Q and R pertain to SAR Section 7.0, "Radiation Protection," ER Section 3.0, "The Facility," and ER Section 4.0, "Environmental Effects of Site Preparation and Facility Construction."</p>

TABLE D.1. (contd)

Questions/Requests ^(a,b)	Responses ^(a,b)	Comments ^(a,b)
<u>QUESTION 1 (November 14, 1984): (contd)</u>	<p data-bbox="942 315 1157 337"><u>RESPONSES: (contd)</u></p> <p data-bbox="942 363 1520 435">data. The 47 group BUGLE structure for the adjoint curves was collapsed to fit the cask vendor's 13 group leakage spectrum for the top of the cask.</p> <p data-bbox="942 461 1520 552">The resultant dose rates are presented in SAR Table 7.3-4. These tabular data were then used to construct the neutron portion of the dose rate versus distance curves referred to above.</p> <p data-bbox="942 578 1520 695">The dose rate for any cask could be obtained at the same distances so long as the neutron spectrum and surface area of the cask is known. Note that the surface area is the cylindrical surface from which neutrons emanate and does not include cooling fins.</p> <p data-bbox="942 721 1520 837">All data are based on a burnup of 35,000 Mwd/MTU with an initial enrichment of 3.5 weight percent uranium-235. This is representative of fuel discharged from Surry Power Station, and is considered the base case for analysis purposes.</p> <p data-bbox="942 863 1520 1097">The current Surry Power Station Units 1 and 2 Technical Specifications authorize the use of 4.1 weight percent uranium-235 enrichment. If all 84 casks were loaded with this fuel, at 45,000 Mwd/MTU burnup, the base case would be increased by a factor of approximately 2.2. To allow for this, and to provide for variations in future cask designs or vendors, a factor of three has been designated as a multiplier. The curves listed above reflect a multiple of three times the base case.</p> <p data-bbox="942 1123 1520 1266">Similarly, ER Table 4.4-1, 4.4-2, 4.4-3, 4.4-4, and 4.4-5 (SAR Tables 7.3-1, 7.4-1, 7.4-2, 7.4-3, 7.4-4) and SAR Table 7.3-4 have been revised and attached hereto to reflect the enveloping factor of three. (Note that SAR Tables 7.3-2 and 7.3-3, do not include the enveloping factor.)</p> <p data-bbox="942 1292 1520 1433">The response to this question has been incorporated into Surry ISFSI SAR Sections 7.3, "Radiation Protection Design Features," and 7.4, "Estimated On-site Collective Dose Assessment," and Surry ISFSI ER Sections 3.5, "Radwaste Systems and Source Terms," and 4.4, "Radioactivity."</p>	

TABLE D.1. (contd)

Questions/Requests (a,b)	Responses (a,b)	Comments (a,b)
<p>QUESTION 2 (November 14, 1984)⁽³¹¹⁾: In your response to Question 4.2.4E and 4.3.1E, neutron and gamma dose rates were cited for distances greater than shown in ER Figure 3.5-2, "Dose Rate from One Cask of Five-Year-Old Spent Fuel Versus Distance," and SAR Figure 7.3-4, "Dose Rate from One Cask of Five-Year-Old Spent Fuel Versus Distance." Please provide additional figures showing neutron and gamma dose rates, from one cask of five-year-old spent fuel, versus distance out to the nearest permanent resident (1.53 mile). ER Figure 3.5-4, "Total Dose Rate from 84 Casks of Spent Fuel Versus Distance," and SAR Figure 7.3-5, "Total Dose Rate from 84 Casks of Spent Fuel Versus Distance," should also show neutron and gamma components of the total dose rate.</p>	<p>RESPONSE: New ER Figure 3.5-5 (SAR Figure 7.3-6), "Dose Rate from One Cask of Five-Year-Old Fuel Versus Distance" (0 to 9,500 ft), attached hereto, shows the dose rate versus distance out to the nearest permanent resident for one cask with five years decay of the fuel multiplied by the enveloping factor of three. These figures indicate the neutron and gamma components as well as the total.</p> <p>Revised ER Figure 3.5-4 (SAR Figure 7.3-5), "Total Dose Rate from 84 Casks of Spent Fuel Versus Distance" (0 to 9,500 ft) is provided herein to show the dose rate versus distance out to the nearest permanent resident for 84 casks (ISFSI with all three pads full). Decay factors have been used assuming that four casks are placed in the ISFSI each year for 21 years, and each new group of four casks has a minimum of five years decay of the fuel.</p> <p>The enveloping factor of three was then applied. The resulting annual dose from ISFSI operations to the nearest permanent resident is 2.49E-5 mrem per year. This supersedes the value previously provided in response to Question 4.3.1E. The resulting dose at the nearest site boundary is 136 mrem per year, and the collective annual population dose is 1.2E-6 manrem per year.</p> <p>Surry ISFSI SAR Figures 7.3-5, "Total Dose Rate from 84 Casks of Spent Fuel Versus Distance," and 7.3-6, "Dose Rate from One Cask of Five-Year Old Spent Fuel Versus Distance," and ER Figures 3.5-4, "Total Dose Rate from 84 Casks of Spent Fuel Versus Distance," and 3.5-5, "Dose Rate from One Cask of Five-Year-Old Spent Fuel Versus Distance," have been added to reflect the requested distance.</p>	<p>See submittals. (316,325) Q and R pertain to SAR Section 7.3, "Radiation Protection Design Features," and ER Section 3.5, "Radwaste Systems and Source Terms."</p>
<p>QUESTION 3 (November 14, 1984)⁽³¹¹⁾: Please provide revised figure showing the neutron and gamma dose rate at the cask surface versus time that was used to calculate dose rates from a full 84-cask configuration, adjusted for decay.</p>	<p>RESPONSE: ER Figure 3.5-3 (SAR Figure 7.3-2), "Normalized Surface Dose Rate One Cask Versus Age of Spent Fuel," has been revised, attached hereto, to show the normalized GNSI cask surface dose rates versus time for both gamma and neutron radiation. The GNS cask side dose rates are 7.9 mrem/h neutron and 22.3 mrem/h gamma. With the enveloping factor of three, these dose rates become 23.7 and 66.9 mrem/h, respectively.</p>	<p>See submittals. (316,325) Q and R pertain to ER Section 3.5, "Radwaste Systems and Source Terms," and SAR Section 7.3, "Radiation Protection Design Features."</p>

TABLE D.1. (contd)

Questions/Requests (a,b)	Responses (a,b)	Comments (a,b)
<p>QUESTION 3 (November 14, 1984): (contd)</p>	<p>RESPONSE: (contd)</p> <p>The GNS cask top dose rates are 24.7 mrem/h neutron and 0.3 mrem/h gamma. With the enveloping factor of three, these dose rates become 74.1 mrem/h and 0.9 mrem/h, respectively.</p> <p>The above numbers supersede those previously provided in response to Question 4.1.4.</p> <p>Surry ISFSI ER Figure 3.5-3, "Normalized Surface Dose Rate On Cask Versus Age of Spent Fuel," and Surry ISFSI SAR Figure 7.3-2, "Normalized Surface Dose Rate On Cask Versus Age of Spent Fuel," have been revised to show the normalized GNSI cask surface dose rates versus time.</p>	
<p>QUESTION 4 (November 14, 1984)⁽³¹¹⁾: Your response to Questions 4.2.1E and 4.2.2 discusses occupational exposures received at the LLWSF due to the ISFSI. In order to better assess the impact the ISFSI has on the LLWSF occupational exposures, please provide an estimate of the occupational exposure at the LLWSF without the collocated ISFSI.</p>	<p>RESPONSE: The design dose to workers at the LLWSF due solely to LLWSF operations is calculated to be in the range of 10.8 to 21.6 man-rem per year. This is based on a typical, historical, occupancy time of 2160 man-hours per year, as assumed in the previous response to Question 4.2.1E and design basis radiation levels. Depending on exactly what operations are taking place (drum handling, movement, monitoring, swiping, waiting, etc.), these 2160 manhours are assumed to be spent in radiation areas corresponding to the LLWSF average design basis values of 5 to 10 mrem/h.</p> <p>The average personnel exposure over the last three years has been much less, approximately 2.5 man-rem/year, for the LLWSF.</p> <p>During the projected 21 years that it takes to load the ISFSI to its design capacity of 84 casks, the dose of workers at the LLWSF due to the ISFSI will range from 0.20 man-rem per year for the first four casks to up to 2.7 (or 8.1 when multiplied by the enveloping factor of three) man-rem per year for all 84. These doses are based on assumptions contained in the previous response to Question 4.2.1E (Reference: VEPCO's September 21, 1984, Letter No. 392 to Mr. R. E. Cunningham of NRC) except that the exact location of each group of four casks has been used, instead of an average distance of 145 feet from the LLWSF. After reaching the peak, the dose will decline continuously thereafter.</p>	<p>See submittals. (316,325) Q and R pertain to SAR Section 7.0, "Radiation Protection," and ER Section 4.0, "Environmental Effects of Site Preparation and Facility Construction."</p>

TABLE D.1. (contd)

Questions/Requests ^(a,b)	Responses ^(a,b)	Comments ^(a,b)
<u>QUESTION 4 (November 14, 1984): (contd)</u>	<p><u>RESPONSE: (contd)</u></p> <p>ER Table 4.4-5 (SAR Table 7.4-4), "Annual Doses from ISFSI Operations," has been revised (attached hereto) to reflect the re-calculated dose to LLWSF workers due solely to the ISFSI.</p> <p>The response to this question has been incorporated into Surry ISFSI SAR Section 7.4, "Estimated Onsite Collective Dose Assessment," and ER Section 4.4, "Radioactivity."</p>	
<u>QUESTION 5 (November 14, 1984)⁽³¹¹⁾</u> : In your response to Question 4.2.3E, ER Table 4.4-1, "Dose Rates Along Fence During ISFSI Operation," does not include the dose rate contribution from the LLWSF as indicated in the accompanying narrative. Please revise this table to include the dose rate contribution from the LLWSF assuming that it was filled to design capacity.	<u>RESPONSE:</u> Surry ISFSI ER Table 4.4-1, "Dose Rates Along Fence During ISFSI Operation," and SAR Table 7.3-1, "Dose Rates Along Fence During ISFSI Operation," have been revised to include the enveloping factor of three, and the dose rate contribution from the LLWSF assuming that it was filled to design capacity.	See submittals. ^(316,325) Q and R pertain to ER Section 4.4, "Radioactivity," and SAR Section 7.3, "Radiation Protection Design Features."
<u>QUESTION 6 (November 14, 1984)⁽³¹¹⁾</u> : In reviewing the ER, SAR and your response to Question 4.2.3E, some confusion has arisen about your use of the term "restricted area." Please review the use of this term as applied to the ISFSI to ensure that it is consistent with health physics practices at the Surry Power Station.	<u>RESPONSE:</u> The Surry Updated FSAR Figure 2.1-12 illustrates the existing Surry Power Station restricted area boundary. The restricted area boundary for the ISFSI will be identical with the one shown on the figure. The Surry ISFSI SAR and ER have been revised accordingly. Designation of this boundary for the ISFSI is not in conflict with health physics practices at the Surry Power Station.	See submittals. ^(316,325) Q and R pertain to the SAR (e.g., Section 7.0, "Radiation Protection") and the ER (e.g., Section 4.0, "Environmental Effects of Site Preparation and Facility Construction").
<u>QUESTION 7 (November 14, 1984)⁽³¹¹⁾</u> : The LA (Section 1.0, "Introduction and General Description of Installation," and Section 3.0, "Principal Design Criteria") indicates 82 casks at the ISFSI. Yet, the ER (Section 3.0, "The Facility") and the SAR (Section 4.0, "Installation Design," and Section 7.0, "Radiation Protection") imply 84 casks at the site. Please clarify.	<p><u>RESPONSE:</u> The Surry Dry Cask ISFSI is presently designed to accommodate a total of 84 dry spent fuel storage casks. Each of the GNSI Castor V/21 casks is capable of accommodating 21 fuel assemblies. Each fuel assembly has 0.46 MTU. The total design capacity of the facility is 811.44 MTU.</p> <p>The response to this question has been incorporated into Section 1.2, "License Sought" and Section 3.0, "Technical Information-Safety Analysis Report" of the LA; Section 3.1, "Purpose of Installation," of the ISFSI SAR; and Section 3.5, "Radwaste Systems and Source Term," of the ISFSI ER.</p>	See submittals. ^(316,325) Q and R pertain to Section 1.0, "General and Financial Information" and Section 3.0, "Technical Information-Safety Analysis Report," of the LA; Section 3.0, "Principal Design Criteria," of the SAR; and Section 3.0, "The Facility," of the ER.

TABLE D.1. (contd)

Questions/Requests (a,b)	Responses (a,b)	Comments (a,b)
<p>QUESTION (March 7, 1985)^(19,371): In reviewing your submittal of February 8, 1985,⁽³¹⁶⁾ we find⁽³⁷¹⁾ that SAR Table 7.3-2, "Neutron Response Functions and Cask Surface Fluxes," incorrectly cites as a reference, for the cask neutron surface flux leakage spectra, the Topical Safety Analysis Report for the CASTOR V/21 Cask Independent Spent Fuel Storage Installation (Dry Storage) by General Nuclear Systems, Inc. (GNSI), February 1985. These data are not included in the topical report. For completeness, we request that you contact GNSI and have them submit the information cited to the Project M-37 docket file for you to appropriately use as a reference in your Surry ISFSI Final Safety Analysis Report. We also request that the input and output to the computer codes used in the cask shielding analysis that generated the neutron and gamma surface flux leakage spectra be placed in the docket file.</p>	<p>RESPONSE: VP requested the CASTOR V vendor, General Nuclear Systems, Inc., (GNSI) to submit to the NRC information not found in the text of the CASTOR V Topical Report but which was supplied to VP and cited by it.⁽³¹⁸⁾ GNSI completed answering the NRC's questions and submitted its final topical report.⁽³¹⁹⁾</p>	<p>See memorandum⁽³¹⁸⁾ and submittal.⁽³¹⁹⁾ Q and R pertain to SAR Section 7.3, "Radiation Protection Design Features," and ER Section 3.5, "Radwaste Systems and Source Terms."</p>
<p>Additionally, Surry ISFSI ER Figure 3.5-3, "Normalized Surface Dose Rate On Cask Versus Age of Spent Fuel," does not reflect the GNSI CASTOR V/21 Topical Report data. We request that this figure be revised and incorporated into the Surry ISFSI Final Safety Analysis Report, when submitted.⁽¹⁹⁾</p>		
<p>QUESTION (April 23, 1985)⁽³²²⁾: To better document the design change involving the upgraded specifications for the concrete pad, the NRC requests that VP submit the drawings shown to the NRC staff (at April 10-15, 1985, meeting) and a description of the upgraded concrete pad specifications.</p>	<p>RESPONSE: Enclosed in response⁽³²³⁾ to your letter dated April 23, 1985⁽³²²⁾ are five (5) copies of the design drawings of the Surry Dry Cask Independent Spent Fuel Storage Installation layout and storage pad details. In addition to the drawings we are also forwarding the description of the facility which will be included in the revised ISFSI Safety Analysis Report and additional details on the characteristics of the reinforcing steel to be used and soil spring and damping factors used in the design analyses. A description of the analysis of the storage pads is included in our letter of June 25, 1984 (Serial No. 303).⁽¹⁷⁾</p>	<p>See submittals^(17,323) for the drawings and the analysis description. Q and R pertain to SAR Section 4.0, "Installation Design."</p>
<p>QUESTION (August 5, 1985)⁽¹³⁾: At the meeting on July 30, 1985, the NRC requested that VEPCO submit additional information on the training of the personnel who would perform fuel handling and cask handling for the Dry Cask ISFSI.</p>	<p>RESPONSE: VEPCO maintains training programs at both of its nuclear power stations. The Surry Power Station training program is described in Section 12.3 of the Surry Power Station Updated Final Safety Analysis Report.</p>	<p>See submittal.⁽³⁷²⁾ Q and R pertain to SAR Section 9.3, "Training Programs."</p>
	<p>The operator training program for Surry Power Station is currently being reviewed for INPO accreditation to ensure it meets the requirements of INPO 85-002, "The Accreditation of Training in</p>	

TABLE D.1. (contd)

Questions/Requests ^(a,b)	Responses ^(a,b)	Comments ^(a,b)
<u>QUESTION (August 5, 1985): (contd)</u>	<u>RESPONSE: (contd)</u>	
	<p>the Nuclear Industry." It is presently expected that the accreditation process will be completed by Spring 1986.</p> <p>Cask handling and fuel loading operations will be performed by Surry Power Station licensed operators or under the direct supervision of senior licensed personnel. Appropriate training and evaluation instruments will be provided for the operators responsible for those tasks identified as relating to the Dry Storage Casks and the ISFSI. Documentation of all training and evaluations will be maintained in the individuals training file and evaluations will be maintained in the individuals training file at the Surry Training Center.</p> <p>Upon the satisfactory completion of this training, the operators will be certified by VEPCO as being qualified for operation of the Dry Cask ISFSI.</p>	
<p><u>QUESTION (October 15, 1985)^(67,68)</u>: The recent discovery of cracks in the fuel basket of the CASTOR V/21 cask used in the Virginia Electric and Power Company/Department of Energy (VEPCO/DOE) dry cask storage demonstration at Idaho National Engineering Laboratory is of great interest to us with respect to your use of CASTOR V/21 casks at the Surry Independent Spent Fuel Storage Installation (ISFSI). We recognize that these cracks were discovered after the cask had been involved in a demonstration under conditions exceeding the design basis set forth in the General Nuclear Systems, Inc., Topical Safety Analysis Report. However, we have asked General Nuclear Systems, Inc., to provide their analysis and evaluation of the cause of the cracking in that cask, assess the need, if any, to revise their Topical Safety Analysis Report and submit necessary changes with the appropriate analysis. In view of these recent developments, we request that you also reassess the use of the CASTOR V/21 cask at the Surry ISFSI and provide us with a report of your finding along with any necessary license application changes.</p>	<p><u>RESPONSE:</u> In response to your letter of October 15, 1985, we have reassessed the use of the CASTOR V/21 cask for the Surry Power Station Unit Nos. 1 and 2 ISFSI. As part of that reassessment, we have reviewed the report which General Nuclear Systems, Inc., (GNSI) forwarded to you in their letter dated November 22, 1985,⁽³⁷⁴⁾ and we have attended the meetings between GNSI and NRC on this subject. Pending the final resolution of this matter, which will be completed once GNSI submits the additional information requested by NRC, we see no effect on any of the documentation in our ISFSI license application, and we still intend to proceed with the use of the GNSI CASTOR V/21 dry storage cask.</p>	<p>See submittals.^(373,374) Q and R pertain to SAR Section 4.2.3, "Individual Unit Description," and to Section 4.2.1 of the GNSI Topical Report.^(e)</p>
<p>Although we are coordinating with DOE on this matter, we request that you keep us informed about any new developments and data. As more information becomes available we may contact you with more specific requests.</p>		

TABLE D.1. (contd)

Questions/Requests (a,b)	Responses (a,b)	Comments (a,b)
<p>(a) Abbreviations: Independent Spent Fuel Storage Installation (ISFSI); License Application (LA); Safety Analysis Report (SAR); Environmental Report (ER); Question (Q); Response (R); General Nuclear Systems, Inc. (GNSI); Virginia Electric and Power Company (VEPCO), and more recently known as Virginia Power (VP); Sealed Surface Storage Cask (SSSC); Low Level Waste Storage Facility (LLWSF); Final Safety Analysis Report (FSAR); Exclusion Area Boundary (EAB); Emergency Planning Zone (EPZ); Cast Iron Cask Storage and Transport of Radioactive Materials (CASTOR); NUS Corporation (NUS); and dose integrating devices (TLDs).</p> <p>(b) The "Table of Contents" for the LA, SAR, and ER may be found in References 325, 329, and 329, respectively.</p> <p>(c) Seed, H. B., I. M. Idriss, "Simplified Procedure for Evaluating Soil Liquefaction Potential," <u>Journal of the Soil Mechanics and Foundation Division ASCE</u>, Vol. 97, No. SM9, September 1971.</p> <p>(d) Seed, H. B., I. M. Idriss, I. Arrango, "Evaluation of Liquefaction Potential Using Field Performance Data," <u>Journal of Geotechnical Engineering ASCE</u>, Vol. 109, No. 3, March 1983.</p> <p>(e) "Topical Safety Analysis Report for the CASTOR V/21 Cask Independent Spent Fuel Storage Installation (Dry Storage)," General Nuclear Systems, Inc., January 1985.</p> <p>(f) Cain, V. R., "The Use of Discrete Ordinates Adjoint Calculations", <u>A Review of the Discrete Ordinates S Method for Radiation Transport Calculations</u>, ORNL-RSIC-19, March 1968, pp. 85-94.</p> <p>(g) Bell, G. I., and S. Glasstone, <u>Nuclear Reactor Theory</u>, Chapter 6.1 - "The Adjoint Function and Its Applications," Van Nostrand Reinhold Company, New York, 1970.</p> <p>(h) Engle, W. W., Jr., "A User's Manual for ANISN, A One-Dimensional Discrete Ordinates Transport Code with Anisotropic Scattering," K-1643, Union Carbide Corporation, Nuclear Division, June 1973.</p> <p>(i) "BUGLE-80, Coupled 47-Neutron, 20-Gamma-Ray, P3, Cross-Section Library for LWR Shielding Calculations," DLC-75, R. W. Roussin, ORNL, June 1980.</p> <p>(j) "Topical Safety Analysis Report for the CASTOR V/21 Cask Independent Spent Fuel, Storage Installation (Dry Storage)" by General Nuclear Systems, Inc., January 1985.</p> <p>(k) U.S. Department of Commerce, <u>1977 City and County Data Book</u>.</p> <p>(l) Voorhess, Alan M. and Associates, "Surry Power Station Transportation Evacuation Study," May 1980.</p> <p>(m) Virginia Radiological Response Plan (COVRERP) Annex I-V to Volume II. The Commonwealth of Virginia, Emergency Operations Plan Peacetime Disasters, Revised 1983.</p> <p>(n) Surry Power Station Units 3 and 4 Preliminary Safety Analysis Report (PSAR), VEPCO, 1973.</p> <p>(o) Surry Power Station Units 3 and 4 Environmental Report, Vol. 1, VEPCO, April 1973.</p> <p>(p) Virginia Radiological Emergency Response Plan (COVRERP) Annex I-V to Volume II, The Commonwealth of Virginia, Emergency Operation Plan Peacetime Disasters. June 1983.</p> <p>(q) 1980 Census of Population Vol. 1. Characteristics of Population, Chapter A - Number of Inhabitants, Part 48, VA., U.S. Department of Commerce, Bureau of Census, February 1982.</p> <p>(r) Virginia Population Projections 2000, Virginia Department of Planning and Budget, Research Section, January 1983.</p> <p>(s) Supplemental Population Projections 2010 and 2020, Virginia Department of Planning and Budget, Research Section, January 1983.</p> <p>(t) List, R. J. (ed). "Meteorological Tables," 6th ed., Smithsonian Institute, 1951, p. 417.</p> <p>(u) Wallace, John M. and Peter V. Hobbs, <u>Atmospheric Science</u>, Academic Press, 1977.</p> <p>(v) Visher, Stephen S., <u>Climatic Atlas of the United States</u>, Harvard University Press, 1966.</p> <p>(w) U.S. NRC Regulatory Guide 7.8, "Load Combinations for Structural Analysis of Shipping Casks," May 1977.</p> <p>(x) Engle, W. W., <u>A User's Manual for ANISN - A One-Dimensional Discrete Ordinates Transport Code with Anisotropic Scattering</u>, ORNL Report K-1693, June 1973.</p> <p>(y) "Topical Safety Analysis Report for the CASTOR V Cask Independent Spent Fuel Storage Installation (Dry Storage)," General Nuclear Systems, Incorporated, December 1983.</p> <p>(z) American National Standard for Neutron and Gamma-Ray Flux-to-Dose Factors, ANSI/ANS-6.1.1 - 1977 (N666).</p> <p>(aa) <u>Occupational Radiation Exposure at Commercial Nuclear Power Reactors 1982</u>. NUREG-0713, Volume 4, December 1983.</p>		

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APPENDIX E

TABLE E.1: QUESTIONS, REQUESTS, AND RESPONSES ASSOCIATED WITH
LICENSING OF THE INDEPENDENT SPENT FUEL STORAGE INSTALLATION
AT ROBINSON-2

TABLE E.1. Questions, Requests, and Responses Associated with Licensing of the Independent Spent Fuel Storage Installation at Robinson-2

Questions/Requests (a,b)	Responses (a,b)			Comments (a,b)																															
<p><u>QUESTION 1</u> (pertains to SAR Section 1.2.1, "General Description"): No drawings, other than conceptual drawing, have been provided for any of the items associated with the ISFSI. Please provide engineering (shop) drawings from which CP&L/NUTECH expects to manufacture the following items: DSC, HSM, support pad for HSM, modification to the IF-300 shipping cask such as the docking collar, cask liner, cask extension and lid, grapple assembly, the hydraulic ram, the trailer used to tow the IF-300 cask, the cask skid, the alignment system, cask position control system, cask clamping system, and the instrumentation systems for the DSC. Include all drawings for all components as they will be used for the H. B. Robinson site; for example, if in a 1x3 array, an HSM outside module differs from an inside module, provide drawings for both modules and drawings for the pad(s) if two arrays of 1x3 and 1x5 are intended. In general, if CP&L/NUTECH intend to manufacture/fabricate/install equipment for the H.B. Robinson ISFSI, engineering drawings should be submitted for each part of each piece of equipment.</p>	<p><u>RESPONSE:</u> In addition to the conceptual drawing that was included in the license application, the engineering drawings listed below are also provided for the H. B. Robinson ISFSI.</p> <table><tr><th>Package</th><th>Drawing No.</th><th>Title</th></tr><tr><td rowspan="5">Dry Shielded Canister</td><td>RNT162-M-2500</td><td>ISFSI DSC Canister Assembly</td></tr><tr><td>RNT162-M-2501</td><td>ISFSI DSC Basket Assembly</td></tr><tr><td>RNT162-M-2503</td><td>ISFSI DSC Spacer Disc</td></tr><tr><td>RNT162-M-2509</td><td>ISFSI DSC Shell Assembly</td></tr><tr><td>RNT162-M-2511</td><td>ISFSI DSC Plug Assembly</td></tr><tr><td></td><td>RNT162-M-2513</td><td>ISFSI DSC Top Cover Plate</td></tr><tr><td rowspan="5">Horizontal Storage Module</td><td>RNT162-C-1101</td><td>ISFSI HSM General Layout and Details</td></tr><tr><td>RNT162-C-1102</td><td>ISFSI HSM Foundation Plan, Sections and Details</td></tr><tr><td>RNT162-C-1103</td><td>ISFSI HSM Roof Plan and Wall Sections</td></tr><tr><td>RNT162-C-1104</td><td>ISFSI HSM Embedment Sections and Details</td></tr><tr><td>RNT162-C-1107</td><td>ISFSI HSM Air Outlet Precast Shielding Block Plan and Sections</td></tr><tr><td></td><td>RNT162-C-1109</td><td>ISFSI HSM Heat Shield Assembly Sections and Details</td></tr></table>			Package	Drawing No.	Title	Dry Shielded Canister	RNT162-M-2500	ISFSI DSC Canister Assembly	RNT162-M-2501	ISFSI DSC Basket Assembly	RNT162-M-2503	ISFSI DSC Spacer Disc	RNT162-M-2509	ISFSI DSC Shell Assembly	RNT162-M-2511	ISFSI DSC Plug Assembly		RNT162-M-2513	ISFSI DSC Top Cover Plate	Horizontal Storage Module	RNT162-C-1101	ISFSI HSM General Layout and Details	RNT162-C-1102	ISFSI HSM Foundation Plan, Sections and Details	RNT162-C-1103	ISFSI HSM Roof Plan and Wall Sections	RNT162-C-1104	ISFSI HSM Embedment Sections and Details	RNT162-C-1107	ISFSI HSM Air Outlet Precast Shielding Block Plan and Sections		RNT162-C-1109	ISFSI HSM Heat Shield Assembly Sections and Details	<p>See submittals. (212,213) Q and R pertain to SAR Section 1.2, "General Description of Installation." The drawings are proprietary. (215,216)</p>
Package	Drawing No.	Title																																	
Dry Shielded Canister	RNT162-M-2500	ISFSI DSC Canister Assembly																																	
	RNT162-M-2501	ISFSI DSC Basket Assembly																																	
	RNT162-M-2503	ISFSI DSC Spacer Disc																																	
	RNT162-M-2509	ISFSI DSC Shell Assembly																																	
	RNT162-M-2511	ISFSI DSC Plug Assembly																																	
	RNT162-M-2513	ISFSI DSC Top Cover Plate																																	
Horizontal Storage Module	RNT162-C-1101	ISFSI HSM General Layout and Details																																	
	RNT162-C-1102	ISFSI HSM Foundation Plan, Sections and Details																																	
	RNT162-C-1103	ISFSI HSM Roof Plan and Wall Sections																																	
	RNT162-C-1104	ISFSI HSM Embedment Sections and Details																																	
	RNT162-C-1107	ISFSI HSM Air Outlet Precast Shielding Block Plan and Sections																																	
	RNT162-C-1109	ISFSI HSM Heat Shield Assembly Sections and Details																																	

TABLE E.1. (contd)

Questions/Requests ^(a,b)	Responses ^(a,b)			Comments ^(a,b)
<u>QUESTION 1: (contd)</u>	<u>RESPONSE: (contd)</u>			
	<u>Package</u>	<u>Drawing No.</u>	<u>Title</u>	
		RNT162-C-1110	ISFSI DSC Support Assembly Plan Sections and Details	
		RNT162-C-1112	ISFSI DSC Seismic Retainer Assembly Sections and Details	
		RNT162-C-1113	ISFSI HSM Thermocouple Location Assembly Sections and Details	
		RNT162-C-1114	ISFSI Heat Shield Thermocouple Location and Details	
		RNT162-C-1116	ISFSI HSM Lightning Protection, Sections and Details	
	Instru- mented Canister	RNT162-M-2600	ISFSI Instrumented DSC Assembly	
		RNT162-M-2609	ISFSI Instrumented DSC Penetration Plug Assembly	
		RNT162-M-2621	ISFSI Instrumented DSC Basket-Sensor Assembly	
		RNT162-M-2622	ISFSI Instrumented DSC Basket-Canister	
	Transfer Cask Skid and Align- ment System	RNT162-M-2300	ISFSI Trailer/Skid Assembly	
	Hydraulic RAM System	RNT162-M-2200	ISFSI Ram Assembly	

TABLE E.1. (contd)

Questions/Requests ^(a,b)	Responses ^(a,b)	Comments ^(a,b)
QUESTION 1: (contd)	RESPONSE: (contd)	
	<p>These drawings represent the general arrangement information for each of the major equipment assemblies and the module structure. Additional information responding to the remaining comments on the license application will refer to these drawings for specific details.</p>	
<p>QUESTION 2 (pertains to SAR Section 1.2.3, "Principal Design Criteria"): This section and others in the SAR make a statement to the effect "For the majority of the fuel to be stored the radiation sources will be less than or equal to the source described in the NUHOMS Topical Report." This statement would imply that there may be some fuel which exceeds the design criteria. What is the maximum radiation source that would be encountered? How does it modify the calculated doses?</p>	<p>RESPONSE: The quoted sentence is inappropriate and shall be changed to read: "For the fuel to be stored, the radiation sources shall be less than or equal to the sources described in Table 1.2-1." Under no circumstances will fuel of higher radiation source strength be loaded into the HBR ISFSI.</p>	<p>See submittals. (211,212) Q and R pertain to SAR Section 1.2, "General Description of Installation."</p>
<p>QUESTION 3 (pertains to SAR Section 1.2.3.1, "Structural Features"): It is stated in the SAR that "the structural features of the DSC are defined, to a large extent, by the cask drop accident" and that the 5-ft drop postulated by NUTECH "envelopes any postulated drop accident that could occur at the Robinson site." The handling operations illustrated in Section 1.3.1.7 and Figure 1.3-5 show several situations in which the cask is more than five feet off of the ground (e.g., cask loading, cask/canister/assembly loading and skid loading). Please explain why drops in these situations are not credible even though they result in the cask being more than five feet off of the ground. Consider both a vertical as well as a horizontal orientation of the cask.</p>	<p>RESPONSE: Section 8.0 (especially Section 8.2.4) of the SAR has been revised to include an 8-ft drop accident analysis for both vertical and horizontal cask orientation. A single failure-proof crane, as described in Section 15.7.5 of the updated FSAR, will be used to remove the spent fuel cask from the fuel pool to the decontamination area. After all operations in the cask handling area are completed, the crane will be used to move the cask to the concrete floor next to the skid/trailer. A redundant yoke will be used during these operations so there will be no possibility of a drop accident. Once on the concrete floor, the redundant yoke will be removed from the cask. The cask will then be raised just above the cask support cradle which will be held in the horizontal position on the skid. The cask will then be lowered into the cradle and rotated 90° into the final horizontal position. These handling operations have been performed at H. B. Robinson in the past prior to shipment of fuel offsite. At no time during this operation will the cask be more than 8 feet off the ground.</p>	<p>See submittals. (211,212,214) Q and R pertain to SAR Section 1.2.3, "Principal Design Criteria," and SAR Section 8.0, "Analysis of Design Events."</p>

TABLE E.1. (contd)

Questions/Requests (a,b)	Responses (a,b)	Comments (a,b)
<p><u>QUESTION 3: (contd)</u></p>	<p><u>RESPONSE: (contd)</u></p> <p>Since the cask will be lifted from the top end, no possibility of a top end drop exists. If the yoke failed during the tilting operation, the cask would land on its steel ring fins located near the top of the cask's outer shell. The maximum height of a drop during this tilting operation is enveloped by an 8-ft horizontal drop accident. In conclusion, the 8-ft drop in either a vertical or horizontal orientation will envelope all drop conditions present during the transfer operation.</p> <p>After the cask is lowered into the saddle, the trailer will be moved to the HSM site. During this operation there is no possibility of dropping the cask since the design of the cradle and saddle are such that they will hold the cask firmly in place. Additionally, the entire cask/skid assembly will be anchored to the trailer bed with tie-down brackets. The possibility of the cask/skid/trailer tipping or rolling over has also been considered. Since the center of gravity of the cask will be less than 8 feet from ground level during this operation, such an unlikely postulated event is enveloped by the 8-ft drop criteria.</p>	
<p><u>QUESTION 4 (pertains to SAR Section 1.2.4, "Operating and Fuel Handling Systems"):</u> What are the values for cask movement, cask head and truck transport which were not included in Table 1.2-3?</p>	<p><u>RESPONSE:</u> No quantitative values are associated with cask movement, cask head, and truck transport in Table 1.2-3. In moving the cask, several criteria must be met. First, the spent fuel handling crane must be capable of lifting the cask and rotating it from a vertical to a horizontal position. Second, the cask lid or head must be removable while the cask is in a horizontal position. Finally, the capabilities of the trailer and skid arrangement must not be exceeded by the size of the cask and the loaded DSC, since it must be transported to the HSM. Consequently, no values exist for the various parameters and therefore "N/A" will be added to the revised SAR Table 1.2-3.</p>	<p>See submittals. (211,212) Q and R pertain to SAR Section 1.2, "General Distribution of Installation."</p>

TABLE E.1. (contd)

Questions/Requests (a,b)	Responses (a,b)	Comments (a,b)
<p>QUESTION 5 (pertains to SAR Section 1.2.5, "Safety Features"): The HSM is designed so that the average surface dose rate is less than 20 mrem/h. Is the average surface dose rate an appropriate design parameter? What is the maximum surface dose rate?</p>	<p>RESPONSE: The average dose rate over the entire DSC surface is used for the source definition in SKYSHINE dose calculations. Also, the average surface dose rate is used to determine the dose rate at points away from the HSM. For particular locations on the HSM, local dose rates are reported.</p> <p>The local dose rate limits for specific points on the HSM surface are specified in Chapter 10 of the NUHOMS Topical Report.</p> <p>The expected, calculated, surface dose rates at various locations are provided in Section 7.0, "Radiation Protection," of the SAR. (212)</p>	<p>See submittals. (211,212) Q and R pertain to SAR Section 1.2, "General Description of Installation," and SAR Section 7.0, "Radiation Protection."</p>
<p>QUESTION 6 (pertains to SAR Section, "1.2.5 Safety Features"): The DSC and cask are filled with demineralized water to reduce contamination of the exterior of the DSC. How effective is this process in reducing contamination of the DSC? What are the expected surface dose levels?</p>	<p>RESPONSE: Four steps are taken to assure that the DSC exterior contamination is held to a minimum: 1) the transfer cask interior is cleaned or decontaminated as necessary if the contamination is greater than 2×10^{-4} $\mu\text{Ci}/\text{cm}^2$, 2) the DSC-cask annulus is filled with demineralized water, 3) the annulus is sealed with waterproof tape to eliminate the intrusion of pool water, and 4) the DSC exterior is swiped for contamination after the annular region is drained.</p> <p>It is anticipated that the DSC external surface contamination will be less than one half the internal surface contamination of the transfer cask if the procedures described in the NUHOMS Topical Report are followed.</p> <p>The DSC surface dose levels resulting from contamination will be insignificant compared to the dose rate due to the irradiated fuel. Should there arise a need to remove the DSC without fuel assemblies, the surface dose levels should be easily dealt with using existing health physics practice.</p>	<p>See submittals. (211,212) Q and R pertain to SAR Section 1.2, "General Description of Installation."</p>

TABLE E.1. (contd)

Questions/Requests (a,b)	Responses (a,b)	Comments (a,b)
<p>QUESTION 7 (pertains to SAR Section 1.2.5, "Safety Features"): What is the surface dose on the cask from fuel pool water contamination?</p>	<p>RESPONSE: The IF-300 cask is licensed for use at the H. B. Robinson site. The fuel handling operation with the DSC inside the cask has not introduced changes to the existing approved procedures. Hence, the surface dose will be within the limits reported in the cask licensing documents.</p> <p>As it exits the pool, the cask passes through a spray ring, which washes it with demineralized water. This prevents pool water contaminants from drying out on the cask body and simplifies further decontamination to the limits reported in the cask licensing document.</p>	<p>See submittals. (211,212) Q and R pertain to SAR Section 1.2, "General Description of Installation."</p>
<p>QUESTION 8 (pertains to SAR Section 1.3.1.1, "Canister Design"): Specify the length of the DSC for the specific type of irradiated fuel to be stored. "Approximately 180 inches" is not adequate to a site specific application.</p>	<p>RESPONSE: The overall length of the DSC canister, including the grapple plate assembly, is 179.5 in. as shown in Drawing RNT162-M-2500.</p>	<p>See submittals. (211,212) Q and R pertain to SAR Section 1.3, "General Systems Description."</p>
<p>QUESTION 9 (pertains to SAR Section 1.3.1.2, "Horizontal Storage Module"): The Code of Federal Regulations 10 CFR 72.66 is not specifically applicable to the ISFSI design proposed by CP&L/NUTECH. See Question 2 in the NRC response to the NUTECH submittal (Section 3.2.3.1).</p>	<p>RESPONSE: The Code of Federal Regulations 10 CFR 77.66 is not applicable and reference to it will be removed from page 1.3-2 of the SAR. Site specific seismic acceleration coefficients as specified in the plant FSAR were applied for the H. B. Robinson ISFSI design. An acceleration of 0.2 g was applied in both horizontal directions. Vertically, an acceleration of 2/3 the horizontal acceleration was applied. To account for possible multiple mode excitations the above accelerations were multiplied by a factor of two. The resulting accelerations of 0.4 g and 0.267 g, horizontal and vertical respectively, were utilized in the design.</p>	<p>See submittals. (211,212) Q and R pertain to SAR Section 1.3.1, "Systems Descriptions."</p>
<p>QUESTION 10 (pertains to SAR Section 1.3.1.2, "Horizontal Storage Modules"): "Certain civil work" is not specific. Please be specific.</p>	<p>RESPONSE: The HSMs will be placed in service on the load bearing foundation shown in Drawing RNT162-C-1102. Drawing RNT162-C-1101 shows the general layout of the foundation work which includes a 4-in. mud slab under the mat foundation to allow for a smooth working surface.</p>	<p>See submittals. (211,212) Q and R pertain to SAR Section 1.3.1, "Systems Descriptions."</p>

TABLE E.1. (contd)

Questions/Requests ^(a,b)	Responses ^(a,b)	Comments ^(a,b)
<u>QUESTION 10: (contd)</u>	<p><u>RESPONSE: (contd)</u></p> <p>Before placing the mud slab, relocation of underground utilities, excavation, backfill and leveling will be performed as necessary.</p> <p>After this initial civil work is performed, the 4-in. mud slab will be placed on the subgrade to provide a working surface for the foundation shown on Drawing RNT162-C-1102.</p>	
<p><u>QUESTION 11 (pertains to SAR Section 1.3.1.2, "Horizontal Storage Module"):</u> Nominal surface dose rates at the front of the HSM are quoted as less than 20 mrem/h. This implies that there is a range. Is the maximum dose rate greater than 20 mrem/h?</p>	<p><u>RESPONSE:</u> Refer to the response to Question 5 regarding SAR Section 1.2.5, "Safety Features," for an explanation of <u>nominal</u> surface dose rates.</p>	<p>See submittals.(211,212) Q and R pertain to SAR Section 1.3.1, "Systems Descriptions."</p>
<p><u>QUESTION 12 (pertains to SAR Section 1.3.1.3, "Transfer Cask"):</u> If CP&L has modified or intends to modify the GE IF-300 shipping cask in any way in order to accomplish the transfer of the DSC from the spent fuel pool to the HSM, it should provide details of these modifications, and show that the IF-300 cask as it was originally licensed has not been nor will be degraded in any safety-significant way.</p>	<p><u>RESPONSE:</u> The modifications will not affect the integrity of the IF-300 cask as it was originally licensed. Use of the cask with the modifications is site-specific and necessary for the transfer of spent fuel from the spent fuel pool to the HSM. No modifications will be made to the cask which would interfere with the cask license for off-site shipment of spent fuel. Additional information about IF-300 cask modifications is provided in SAR Sections 1.3.1.3 and 8.2.4. Cask lid drawings were submitted separately.(214)</p>	<p>See submittals.(211,212,214) Q and R pertain to SAR Section 1.3.1, "Systems Descriptions."</p>
<p><u>QUESTION 13 (pertains to SAR Section 1.3.1.3, "Transfer Cask"):</u> The transfer cask to be used is the IF-300 cask. Has an analysis been performed to show that the resulting dose rates during transfer are less than 200 mrem/h design limits?</p>	<p><u>RESPONSE:</u> Yes, the results of this analysis are provided in Table 7.3-2, "Shielding Analysis Results," in Section 7.0, "Radiation Protection."</p>	<p>See submittals.(211,212) Q and R pertain to SAR Section 1.3.1, "Systems Descriptions."</p>
<p><u>QUESTION 14 (pertains to SAR Section 1.3.1.7, "Systems Operation"):</u> If CP&L intends to use a single failure-proof crane for system operations, it should so state either in SAR Section 1.0, "Introduction and General Description of Installation," or SAR Section 3.0, "Principal Design Criteria." Also CP&L should include the topic(s), Cask Tilting from Vertical to Horizontal Position on Transporter/Skid, in their discussion of system operation.</p>	<p><u>RESPONSE:</u> As stated in SAR Section 1.0, "Introduction and General Description of Installation," the cask lifting operation will be identical to the existing HBR2 lifting procedure. Additionally, the cask tilting operation from vertical to horizontal position on the Transport Skid will be identical to the procedure described for the IF-300 since the</p>	<p>See submittals.(211,212,214) Q and R pertain to SAR Section 1.3.1, "Systems Descriptions," and SAR Section 1.0, "Introduction and General Description of Installation."</p>

TABLE E.1. (contd)

Questions/Requests (a,b)	Responses (a,b)	Comments (a,b)
<u>QUESTION 14: (contd)</u>	<u>RESPONSE: (contd)</u>	
	<p>same tilting cradle will be used. Consequently, additional description will be added to SAR Section 1.0 that references the HBR2 FSAR, HBR2 operating procedures, and the IF-300 Topical Report. Additional description has been provided in SA Section 1.3 for the cask tilting operation, referencing appropriate documents. (214)</p>	
<p><u>QUESTION 15 (pertains to SAR Section 1.3.1.7, "System Operation"):</u> Paragraphs (n), (p), and (q) discuss the use of the rear concrete shield plug in the HSM. This component was not used in the design described in the NUTECH Topical Report. Since no design data is provided in the SAR for this plug (e.g., dimensions, installation procedures, locking procedure, etc.), please provide such data in order that the use of this plug can be reviewed. Has a shielding analysis of the plug been performed? If so, what are the results?</p>	<p><u>RESPONSE:</u> The rear access opening, as shown in Drawing RNT162-C-1104, consists of a 14-inch diameter pipe embedded in the module wall. A removable two-inch steel cover plate is bolted to the HSM to cover the opening. The Hydraulic Ram Assembly, as shown in Drawing RNT162-M-2200, enters the HSM through the rear access opening. Once the DSC is pulled into the HSM cavity, the DSC Seismic Retainer, as shown in Drawing RNT162-C-1112, is inserted into the rear access. This assembly couples with the grapple assembly on the end of the DSC to prevent axial movement during a seismic event. Once the Retainer is locked in place the two-inch plate is bolted down. Paragraphs (n), (p), and (q) will be revised to incorporate this information.</p> <p>A shielding analysis of the radiation streaming through the rear access opening (during unloading or loading) has been performed. Without the plug in place, the dose rate at the outside of the penetration will be less than 1 mrem/h.</p>	<p>See submittals. (211,212) Q and R pertain to SAR Section 1.3.1, "Systems Description."</p>
<p><u>QUESTION 16 (pertains to SAR Section 1.3.1.7, "System Operation"):</u> The SAR refers to a design criteria for water content of the DSC. What is the criteria for water content of the DSC?</p>	<p><u>RESPONSE:</u> The criteria for water content of the DSC is provided in Section 10.3.2.1 of the Topical Report. Holding the DSC at 1 torr for one hour will result in a moisture content of approximately 1.18 grams.</p>	<p>See submittals. (211,212) Q and R pertain to SAR Section 1.3.1, "Systems Descriptions."</p>
<p><u>QUESTION 17 (pertains to SAR Section 1.3.1.7, "System Operation"):</u> The placement of the DSC in the HSM involves operations through a penetration in the rear of the HSM. The Topical Report does not contain an analysis of the design of the penetration through the rear of the HSM nor the resulting dose rates. What are the predicted dose rates through the rear penetration?</p>	<p><u>RESPONSE:</u> As detailed in the response to Question 15, the predicted dose rate through rear penetration is less than 1 mrem/h.</p>	<p>See submittals. (211,212) Q and R pertain to SAR Section 1.3.1, "Systems Descriptions."</p>

TABLE E.1. (contd)

Questions/Requests ^(a,b)	Responses ^(a,b)	Comments ^(a,b)
<p><u>QUESTION 18</u> (pertains to SAR Section 1.3.1.7, "System Operation"): The Topical Report does not provide any analysis of the effectiveness of the rear shield on the DSC. What is the dose rate on the rear (bottom) surface of the DSC?</p>	<p><u>RESPONSE:</u> The contact dose rate on the bottom end of the DSC is 1.5 mrem/h at the DSC centerline. However, this end of the DSC will be positioned in the front of the HSM (i.e., the "top" of the DSC is the end drawn into the HSM by the grapple; it is secured to the "rear" of the HSM by the seismic retainer).</p> <p>The contact dose rate of the top end of the DSC is 32 mrem/h, and, as stated in the response to Question 15 (SAR Section 1.3.1.7), the dose rate outside the rear access of the HSM without its shield plate is 1 mrem/h.</p>	<p>See submittals. (211,212) Q and R pertain to SAR Section 1.3.1, "Systems Descriptions."</p>
<p><u>QUESTION 19</u> (pertains to SAR Section 1.3.1.7, "System Operation"): Please note that SAR Figure 1.1-1 is not consistent with the Topical Report. In addition, SAR Figure 1.3-2 is not consistent with SAR Figure 1.1-1.</p>	<p><u>RESPONSE:</u> The Topical Report is based on a design where the ram operates from the front of the HSM. This operation facilitates back to back modules in a 4x2 array. The HSM design for H. B. Robinson is a 3x1 array which is loaded and unloaded through a rear access opening with the hydraulic ram. In SAR Figure 1.3-2, the HSM air flow diagram, the module does not show the rear access opening. This figure will be changed to agree with SAR Figure 1.1-1.</p>	<p>See submittals. (211,212) Q and R pertain to SAR Section 1.3, "General Systems Descriptions," and SAR Section 1.1, "Introduction."</p>
<p><u>QUESTION 20</u> (pertains to SAR Section 1.5, "Material Incorporated by Reference"): The NUTECH Topical Report has been incorporated by reference, as it should be. However, it should be noted that there are several significant differences between the design described by NUTECH in their Topical Report and that proposed in the SAR. For example, the design described in the SAR is for a three-module, back-loaded system whereas the design described in the Topical Report is for an eight-module, front-loaded system. Care should be taken throughout this SAR to fully document (i.e., provide analysis, design drawings, etc.) any portion of the design that is different from that described in the SAR.</p>	<p><u>RESPONSE:</u> The SAR will be revised to reflect all specific design differences. Design drawings listed in the response to Question 1 show the site specific design. Appropriate analyses will be added to the SAR to reflect any significant differences between the Topical Report and the SAR.</p>	<p>See submittals. (211,212) Q and R pertain to SAR Section 1.0, "Introduction and General Description of Installation."</p>
<p><u>QUESTION 21</u> (pertains to SAR Section 2.1.1, "Site Location"): Is the latitude of 34° 12'12" north correct for the site?</p>	<p><u>RESPONSE:</u> The correct latitude is 34° 24'12" north.</p>	<p>See submittals. (211,212) Q and R pertain to SAR Section 2.1, "Geography and Demography."</p>

TABLE E.1. (contd)

Questions/Requests (a,b)	Responses (a,b)	Comments (a,b)																																				
<p><u>QUESTION 22</u> (pertains to SAR Section 2.1.2.1, "Other Activities Within the Site Boundary"): There is an extension of the Seaboard Coastline Railroad spur which enters the plant area north of the plant and provides for delivery of coal to HBR1. These deliveries occur about three times per week. SAR Figure 1.1-2 indicates this spur extension passes close by the proposed ISFSI location but as there is no scale provided on the figure, the actual distance between the ISFSI and the rail spur cannot be determined. The reasons for interest in this information are the possible effects that could conceivably result from a derailment, or some other railroad related accident, at that point. Such a postulated accident is not mentioned in SAR Section 8.0. Whether such an accident is credible with significant effects, or otherwise, should be addressed.</p> <p>In connection with SAR Figure 1.1-2, comparison of that figure with Figure 2.1.1-3 (an aerial photograph) in the HBR2 Updated FSAR creates the impression that the two figures do not agree spatially or with various rail spurs' orientation/configuration.</p>	<p><u>RESPONSE:</u> The location chosen at H. B. Robinson for the ISFSI will be maintained with a distance of 20 feet from the protected area boundary fence to the HSM structure. This fence is located north of the proposed ISFSI location. Four feet farther north is a nuisance fence which is part of the protected area fence system. The first track of the rail spur is approximately ten feet north of the second fence. Thus, there will be approximately 34 feet between the HSM and the rail spur.</p> <p>This track is primarily used to store empty cars. There does exist the possibility of having loaded cars on this track, but this is generally not the case.</p> <p>Figure 2.1.1-3 in the H. B. Robinson updated FSAR is not a current aerial photograph. Figure 1.1-2 of the SAR provides more up-to-date information regarding the present site plan.</p>	<p>See submittals. (211,212) Q and R pertain to SAR Section 2.1.2, "Site Description."</p>																																				
<p><u>QUESTION 23</u> (pertains to SAR Section 2.1.3, "Population Distribution and Trends"): Phone conversations with State Statistical Office Research personnel indicated that population projections provided by the Federal Government have been found to be consistently low. It is therefore suggested that a request for, "Total Population Projections by County" be addressed to Mr. Mike MacFarlane, Rember C. Dennis Building, Room 337, 1000 Assembly Street, Columbia, South Carolina, 29201. This information is up to date and will be sent free of charge. It should provide more accurate data regarding population distribution and trends than is contained in the HBR2 Updated FSAR.</p>	<p><u>RESPONSE:</u> The Research and Statistical Office of South Carolina provided a county by county listing of population projections for five-year intervals from the year 1980 to 2000. Listed below are the projections and estimate for the four counties within the ten-mile radius of the site.</p> <table><tr><th>County</th><th>1980</th><th>1985</th><th>1990</th><th>1995</th><th>2000</th></tr><tr><td>Chesterfield</td><td>38,200</td><td>40,400</td><td>43,700</td><td>45,300</td><td>46,900</td></tr><tr><td>Darlington</td><td>62,800</td><td>67,700</td><td>74,300</td><td>78,500</td><td>82,800</td></tr><tr><td>Kershaw</td><td>39,100</td><td>41,100</td><td>44,200</td><td>45,500</td><td>47,100</td></tr><tr><td>Lee</td><td>18,900</td><td>19,300</td><td>19,700</td><td>19,400</td><td>20,000</td></tr><tr><td>The Area</td><td>159,000</td><td>168,500</td><td>181,900</td><td>188,700</td><td>196,800</td></tr></table> <p>The 1980 populations are estimates and the others are projections.</p>	County	1980	1985	1990	1995	2000	Chesterfield	38,200	40,400	43,700	45,300	46,900	Darlington	62,800	67,700	74,300	78,500	82,800	Kershaw	39,100	41,100	44,200	45,500	47,100	Lee	18,900	19,300	19,700	19,400	20,000	The Area	159,000	168,500	181,900	188,700	196,800	<p>See submittals. (211,212) Q and R pertain to SAR Section 2.1, "Geography and Demography."</p>
County	1980	1985	1990	1995	2000																																	
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TABLE E.1. (contd)

Questions/Requests ^(a,b)	Responses ^(a,b)	Comments ^(a,b)
<p><u>QUESTION 24</u> (pertains to SAR Section 2.2.1, "Locations and Routes"): Principal transportation routes or facilities are said to include a railroad line (1600 feet east).</p> <p>There is no evidence of any railroad line to the east of the site on any figures/maps provided.</p> <p><u>QUESTION 25</u> (pertains to SAR Section 2.2.2, "Descriptions"): Principal transportation routes are said to include SC 151 (1/2 mile east). Highway 151 passes the site some 3,000 feet to the west.</p> <p>The locations of the railroad line and the highway referred to should be corrected.</p> <p>More detailed description of the small airport 2 1/2 miles east needs to be provided and should include the following: a) definition of "small planes" (i.e., single engine only or single and twin, and maximum weight); b) number of runways and dimensions; c) runway orientation; d) departure and approach patterns; e) operational capacity [i.e., air-ground-air-communication availability, flight operations restricted to Visual Flight Rules (VFR) operations or otherwise]; and f) traffic loading (i.e., average number of landings and takeoffs per day).</p> <p>These kinds of information need to be available in order to estimate the probability of a plane crash at the ISFSI and the consequences thereof. Such a possibility should be addressed in SAR Section 8.0.</p> <p><u>QUESTION 26</u> (pertains to SAR Section 2.3.2, "Local Meteorology"): The ISFSI SAR reports the maximum site area 24-hour precipitation as having been 4.76 inches in September of 1979, with the passage of the remains of Hurricane David. The HBR Updated FSAR reports the highest recorded 24-hour precipitation in the site area was 7.66 inches in August of 1949. Assuming a repetition of the latter rainfall, can it be shown that the water would not accumulate in the bottom of the HSM?</p>	<p><u>RESPONSE:</u> The railroad is 1600 feet west of the site.</p> <p><u>RESPONSE:</u> The location of the highway (approximately 1/2 mile west) will be correct in the SAR.</p> <p>Both twin and single engine planes, as well as an occasional Lear jet, use the runway. The maximum weight of these planes is approximately 4000 pounds. The airport has one runway, 3300 feet long, with a left downwind approach. The runway is oriented in a north-northeast to south-southwest direction and has a heading of 20 degrees or 200 degrees depending upon the direction of approach. Standard departure and approach patterns are not generally used. However, when approaching the airport, using the Chesterfield, South Carolina VOR, a radial approach reading of 154 degrees is used. The Hartsville Airport is 16.6 nautical miles from Chesterfield. A Unicom system is used for communications. There are approximately 30 landings and takeoffs per day.</p> <p><u>RESPONSE:</u> The air inlet and outlet openings are the only source of possible rain water intrusion. If rain penetrates the air inlet openings, it falls into the enclosed chamber underneath the air inlet shielding slab. This area is sloped toward the module front which contains two 1-inch diameter drain pipes. Consequently, all rainfall entering the air inlet penetration will drain through these embedded pipes. The drain pipes will be inspected periodically to ensure that they remain unclogged.</p>	<p>See submittals. (211,212) Q and R pertain to SAR Section 2.2, "Nearby Industrial, Transportation, and Military Facilities."</p> <p>See submittals. (211,212) Q and R pertain to SAR Section 2.2, "Nearby Industrial, Transportation, and Military Facilities."</p> <p>See submittals. (211,212,214) Q and R pertain to SAR Section 2.3, "Meteorology."</p>

TABLE E.1. (contd)

Questions/Requests ^(a,b)	Responses ^(a,b)	Comments ^(a,b)
<u>QUESTION 26: (contd)</u>	<u>RESPONSE: (contd)</u>	
<p>If water did accumulate in the bottom of the HSM, can it be ascertained how long it would take to evaporate to dryness given the fairly high average relative humidity (77%) typical of the months of August and September?</p>	<p>The air outlet openings are covered by air outlet shielding blocks which contain a metal flashing as shown on Drawing RNT162-C-1107. This flashing branches out over 4 inches from the shield opening at 45° to provide rain protection. Additionally, the air outlet penetration is completely surrounded by a 4 in. x 4 in. curb which eliminates any seepage down into the penetration. Therefore, if rain enters through the shielding block it is retained by the curb. Also the module roof is sloped to prevent rainwater buildup or ponding. Estimated annual dose data for module surveillance has been added to SAR Table 7.4-2.</p>	
<p><u>QUESTION 27 (pertains to SAR Section 2.6.2.1, "Earthquake History"):</u> The fourth paragraph of this part of the ISFSI SAR states: "The site is located in Zone 1 of the Uniform Building Codes' Map of Equal Seismic Probability. Zone 1 is characterized as a zone of light earthquake activity which would result in minor damage." The characterization of Zone 1 agrees with that presented by the U.S. Coast and Geodetic Survey, "Seismic Probability Map of the United States, 1949" (Appendix 2.5E of the HBR2 Updated FSAR) and the "Seismic Risk Map of the United States" by S. T. Algermisson, U.S. Geological Survey (U.S.G.S.) 1969. Algermisson's map, however, shows the site as being located in Zone 2, characterized by expected moderate damage. This map depicts the probability of the risk of damage but not the probability of future earthquakes occurrences. A 1982 study by S. T. Algermisson et al., U.S.G.S., presents seismic hazard in terms of percent of earth's gravity that have only a 10% probability of being exceeded in 10, 50, and 250 years. The "Miscellaneous Field Studies Map MF 620," prepared by the U.S.G.S., in cooperation with the U.S. Atomic Energy Commission, 1974, presents predictions of both earthquake intensity and frequency.</p>	<p><u>RESPONSE:</u> Reference to the Uniform Building Code is incorrect and will be revised as suggested to the U.S. Coast and Geodetic Survey Map.</p> <p>The Uniform Building Code locates the ISFSI in Zone 2, characterized by moderate damage, which corresponds to Intensity VII on the Modified Mercalli Scale. As noted, the largest regional earthquake occurred at Charleston, South Carolina in 1886. This earthquake's projected effect on the Hartsville area suggests a high VI to low VII Modified Mercalli intensity, which is reflected in the Uniform Building Code Seismic Zone Map. The design maximum ground acceleration of 0.2 g provides an ample factor of safety with no loss of function based on an earthquake of this potential.</p>	<p>See submittals. (211,212) Q and R pertain to SAR Section 2.6.2, "Vibratory Ground Motion."</p>
<p>It is suggested that additional information be provided to either verify that the site is located in an area of light seismic activity with only minor damage expected or to establish the true seismic characteristics of the site area and provide a more definitive estimate of the probability of seismic activity therein.</p>		

TABLE E.1. (contd)

Questions/Requests (a,b)	Responses (a,b)	Comments (a,b)
<p><u>QUESTION 28</u> (pertains to SAR Section 2.6.4.3, "ISFSI Foundation"): Please provide data on the analysis performed to evaluate potential soil liquefaction at the ISFSI site. This data should be in sufficient detail so as to permit an appropriate NRC review of the analysis and, as such, should include data on soil characteristics, analysis methodology, etc.</p>	<p><u>RESPONSE:</u> A copy of the "Report of Geotechnical Exploration, Dry Storage Module Project"^(c) is provided.</p>	<p>See submittals. (211,212) Q and R pertain to SAR Section 2.6.4, "Stability of Subsurface Materials."</p>
<p><u>QUESTION 29</u> (pertains to SAR Section 3.0, "Principal Design Criteria"): Somewhere in Section 3.0 (either 3.2 or 3.3.7), should be the specific statement that 125°F is the design basis maximum inlet air temperature. This fact is mentioned in the introduction section (Section 1.2.3.2) but not in the design criteria section.</p>	<p><u>RESPONSE:</u> SAR Section 3.2 will be revised to include a section on thermal loading. The section will include a specific statement stating that 125°F is the design basis maximum inlet air temperature. This temperature envelopes the maximum ambient temperature in the site area of 107°F which is stated in Section 2.3.2.1.2 of the updated FSAR.</p>	<p>See submittals. (211,212)</p>
<p><u>QUESTION 30</u> (pertains to SAR Section 3.0, "Principal Design Criteria"): SAR Section 3.0 (probably 3.2) should also define the design basis solar radiation for the storage system. This design basis criteria should be based on information presented in SAR Section 2.0.</p>	<p><u>RESPONSE:</u> The value for the design basis solar radiation for the storage system is 188 Btu/hr-ft². This will be included in the revised SAR. The design basis solar radiation, based on information contained in Section 2.0 of the SAR, is bound by the NUTECH Topical Report.</p>	<p>See submittals. (211,212)</p>
<p><u>QUESTION 31</u> (pertains to SAR Section 3.1.1.3, "Radiological Characteristics"): The radiation source of the majority of the fuel stored is stated to be less than or equal to the design criteria in the Topical Report; however, it is suggested that other combinations of decay time and burnup may be acceptable if additional calculations are performed. Operationally, what procedures exist to insure that these additional calculations are performed? What are the computational procedures?</p>	<p><u>RESPONSE:</u> If a fuel assembly is intended to be stored and does not meet the maximum burnup/minimum cooling time specified in Section 3.1.1.3, additional calculations must be performed to assure that the design parameters given in SAR Tables 1.2-1 and 3.1-2 are not exceeded. The computational procedures are similar to those used to establish burnup and cooling time guidelines. ORIGEN2, or a similar code, will be used to determine the isotopic composition and resulting generation rates, given a particular fuel enrichment and irradiation history.</p> <p>Operational procedures for ensuring the performance of the calculations are not included in the SAR. Such activities shall be addressed in CP&L detailed site operational procedures or QA procedures for fuel handling.</p>	<p>See submittals. (211,212) Q and R pertain to SAR Section 3.1.1, "Material to be Stored."</p>

TABLE E.1. (contd)

Questions/Requests (a,b)	Responses (a,b)	Comments (a,b)
<p><u>QUESTION 32 (pertains to SAR Section 3.1.2.1, "Overall Functions of the Facility"):</u> The statement that the ISFSI is designed to be "totally passive" may be correct for some of the units, but other units apparently will be instrumented. CP&L is asked to provide not only a description of this instrumentation, but also to show that the confinement integrity of the DSC is not compromised by instrument penetrations.</p>	<p><u>RESPONSE:</u> The function of the ISFSI will not be changed by the instrumentation, since instrumentation will be added for the sole purpose of collecting data. Thus, the ISFSI will still be "totally passive." The instruments will not compromise the confinement integrity of the DSC as shown in the Penetration Plug Assembly Drawing RNT162-M-2609. Other Instrumented DSC drawings are RNT162-M-2600, RNT162-M-2621, and RNT162-M-2622. See the response to Question 64 regarding Section 5.4, "Operation Support System" for a detailed description of the instrumentation confinement for the DSC penetration.</p>	<p>See submittals. (211,212) Q and R pertain to SAR Section 3.1.2, "General Operating Functions."</p>
<p><u>QUESTION 33 (pertains to SAR Section 3.1.2.1, "Overall Functions of the Facility"):</u> The statement, "Once the DSC is properly positioned within the HSM,..." leads to the question, what is properly positioned, and how is it determined that the DSC is properly positioned? CP&L should delineate the criteria for proper axial as well as radial positioning of the DSC in the HSM. Question 8 in Section 8.1.1.1 and Question 6 in Section 8.1.1.5 of the NRC letter of May 29, 1985, to NUTECH provide some reasons for the concern. See also Question 54 in Section 5.1.1.6, below.</p>	<p><u>RESPONSE:</u> Two-inch thick steel stopping blocks will be bolted to the end of the WT4 x 33.5 DSC Support Assembly rails. These stop plates establish the final position of the DSC, 3.5 inches from the HSM concrete wall in the axial position.</p> <p>To assure radial positioning identical to that of the thermal model, however, markings will be placed on the top cover plate which will indicate proper azimuthal orientation of the canister.</p>	<p>See submittals. (211,212) Q and R pertain to SAR Section 3.1.2, "General Operating Functions."</p>
<p><u>QUESTION 34 (pertains to SAR Section 3.1.2.2, "Handling and Transfer Equipment"):</u> In this section and elsewhere (e.g., SAR Section 5.2.1.1, "Functional Description"), the SAR asserts specifically that the IF-300 provides shielding along the axial length of the IFA's. Does the amount of shielding give the 200 mrem/h design criteria? Is adequate shielding provided on the longitudinal axis?</p>	<p><u>RESPONSE:</u> The IF-300 cask is designed to transport irradiated PWR fuel after a 120-day cooling period while maintaining a maximum surface dose rate of less than 200 mrem/h. The H. B. Robinson fuel will have been cooled for 5 years, or 1825 days. As a result, the dose rate along the axial length of the transfer cask will be significantly lower than 200 mrem/h when loaded with H. B. Robinson fuel. SAR Table 7.3-2 presents the calculated transfer cask surface dose rate as 5.6 mrem/h.</p> <p>On the longitudinal axis, the two ends must be considered. The bottom end of the transfer cask contains sufficient shielding to assure a surface dose rate of less than 200 mrem/h when loaded with 120-day cooled fuel. The use of 1825-day cooled fuel,</p>	<p>See submittals. (211,212,214) Q and R pertain to SAR Section 3.1.2, "General Operating Functions."</p>

TABLE E.1. (contd)

Questions/Requests (a,b)	Responses (a,b)	Comments (a,b)
<u>QUESTION 34: (contd)</u>	<u>RESPONSE: (contd)</u> and the thick lead DSC end plug will reduce the dose rate to far below the 200 mrem/h. At the top end of the fuel, note that the IF-300 shielded cask head will not be used during operations. A steel plate will be used in its place. Substantial shielding will be provided by the top DSC lead shield plug. SAR Table 7.3-2 shows that the surface dose at the DSC top cover plate is 31.7 mrem/h. The cask-DSC annular gap dose rate is 23.1 mrem/h. Hence, adequate shielding will be provided on both ends of the fuel. Additional detail was added to SAR Section 7.3.2.2 and SAR Table 7.3-2 with respect to the shielding analysis and its results. (214)	
<u>QUESTION 35 (pertains to SAR Section 3.1.2.2, "Handling and Transfer Equipment"):</u> SAR Table 3.1-2 gives acceptable radiological criteria for storage of material in the ISFSI as 1.48×10^{16} MeV/sec for gamma and 1.17×10^9 n/sec for neutron sources. The value given in the Topical Report are 1.67×10^{16} MeV/sec and 9.98×10^8 n/sec respectively. What is the reason for the difference? Has the shielding analysis been performed for the larger neutron source strengths? If so, what are the results?	<u>RESPONSE:</u> The difference, as described in SAR Section 7.3.2.2, is due to the increased burnup and decreased enrichment of the H. B. Robinson site-specific fuel. The shielding analyses have been updated and results are shown in SAR Table 7.3-2.	See submittals. (211,212) Q and R pertain to SAR Section 3.1.2, "General Operating Functions."
<u>QUESTION 36 (pertains to SAR Section 3.1.2.2, "Handling and Transfer Equipment"):</u> CP&L is requested to respond to the Question 3.1.2.2 (number 3) of the NRC response to the NUTECH Topical Report.	<u>RESPONSE:</u> A vertical acceleration factor of 2 g has been assumed per the criteria of 10 CFR Part 71.45(b)(1). This acceleration is lower than the deceleration values associated with the drop accident as reported in SAR Section 8.2.4.	See submittals. (211,212) Q and R pertain to SAR Section 3.1.2, "General Operating Functions."
<u>QUESTION 37 (pertains to SAR Section 3.2.3.1, "Input Criteria"):</u> It is stated in this section that the analysis that supports the conclusion that the stresses resulting from the HBR2 site-specific earthquake ground accelerations are enveloped by those presented in the Topical Report is presented in SAR Section 8.2. However, no analysis is shown in SAR Section 8.2--only the results from an analysis. Please either revise the statements in this section or revise SAR Section 8.2.	<u>RESPONSE:</u> A detailed seismic accident analysis has been added to SAR Section 8.2.3 to show the site-specific seismic analysis for the DSC, DSC Support Assembly, and the HSM.	See submittals. (211,212,214) Q and R pertain to SAR Section 3.2.3, "Seismic Design."

TABLE E.1. (contd)

Questions/Requests ^(a,b)	Responses ^(a,b)	Comments ^(a,b)
<p><u>QUESTION 38</u> (pertains to SAR Section 3.2.3.2, "Seismic-System Analysis"): The comment given above under Question 37 (Section 3.2.3.1) is equally applicable to the statement given in this section. Also it is stated in this section that details of the foundation design are provided in SAR Section 8.3. SAR Section 8.3 only gives a general description of the foundation design and does not include either details of the design nor analysis data justifying the design. Please either revise the statements in this section or revise SAR Sections 8.2 and 8.3.</p>	<p><u>RESPONSE:</u> SAR Section 8.3 will be revised to include specific details for the foundation design. SAR Section 8.2 will be revised for site specific analyses that differ from the NUHOMS Topical Report. Refer to answer to Question 86 for more information on foundation analysis.</p>	<p>See submittals. (211,212,214) q and R pertain to SAR Section 3.2.3, "Seismic Design."</p>
<p><u>QUESTION 39</u> (pertains to SAR Section 3.3.1, "General"): It is noted in SAR Table 3.3-1 that only the Transfer Cask and the DSC are "Safety Related." Please explain why the other items are not considered to be important to safety even though they provide such functions as shielding and protection of the DSC.</p>	<p><u>RESPONSE:</u> In establishing the quality classifications for the various component parts of the ISFSI, only "Safety Related" or "Non-Safety Related" categories were used. (211) Although this designation is based on the specific need for component performance during accident conditions, it is not the only basis for establishing whether any part is important to the safe and proper operation of the ISFSI system.</p> <p>The HSM is not considered "Safety Related" since it does not provide the necessary storage environment for the spent fuel. The modules do, however, provide shielding and structural protection for the DSC. For these reasons, the concrete structure has been designed in accordance with ACI 349-80. The drawings and specifications for concrete construction establish the level of testing and inspection required in accordance with CP&L's "Radwaste Related" Q/A program for H. B. Robinson. This feature is important to the safe operation of the ISFSI and appropriate levels of documentation and control are being applied.</p> <p>The remaining primary components (ram, skid, trailer) are important to the successful loading of the DSC into the HSM, however, they are not "Safety Related." Failure of these items would not endanger the health and safety of plant employees or the general public. The importance of the safe successful transfer and loading of the canisters is upheld in the design drawings and specifications through material quality, welding, tolerances, and performance testing.</p>	<p>See submittals. (211,212,220) q and R pertain to SAR Section 3.3, "Safety Protection System."</p>

TABLE E.1. (contd)

Questions/Requests (a,b)	Responses (a,b)	Comments (a,b)
<p>QUESTION 39: (contd)</p>	<p>RESPONSE: (contd)</p> <p>The letter of February 25, 1986, (220) provides the following additional information and clarification to Question 39. The CP&L Quality Assurance (QA) Program recognizes only the terms "Safety Related" and "Non-Safety Related." The Dry Storage Canister and Transfer Cask are considered Safety Related and are subject to a QA Program in conformance to the requirements of 10 CFR 50 Appendix B.</p> <p>The Horizontal Storage Module, Foundation, Transfer Components, and Instrumentation are Listed as "Non-Safety Related." Footnote 3 notes that CP&L will apply the "Radwaste-Related" QA Program to these items. The QA Program for Radioactive Waste Management Systems has been in existence for several years, has applied to certain activities at CP&L's nuclear plants, and, as such, is auditable by the NRC's Region II staff. The QA Program for Radioactive Waste Management Systems is consistent with the requirements of 10 CFR 72.80. The Program is Chapter 19 of the overall corporate QA manual as shown in the H. B. Robinson Updated Final Safety Analysis Report, Section 17.2.2.1. Attachment 1 to this letter (220) is a brief summary of the Program, which will be incorporated into Chapter 11 of the ISFSI SAR, if appropriate, upon a satisfactory resolution to this question.</p> <p><u>Attachment 1</u></p> <p>The Corporate QA Program also provides a specific program for Radioactive Waste Management Systems. This program parallels the appropriate requirements of the 18 criteria of 10 CFR 50, Appendix B and has been developed to match the level of control necessary for equipment, materials, and processes which are important to maintain the integrity of the Radioactive Waste Management System.</p> <p>This program requires that design, procurement, installation, and testing be accomplished in a planned and controlled manner in accordance with</p>	

TABLE E.1. (contd)

Questions/Requests (a,b)	Responses (a,b)	Comments (a,b)
<u>QUESTION 39: (contd)</u>	<u>RESPONSE: (contd)</u>	
	approved procedures. These procedures include provisions, as necessary, to ensure that:	
	1. Design and procurement documents include applicable design requirements, and are reviewed for adequacy. Deviations are controlled.	
	2. Purchased material, equipment, and services conform to the requirements of drawings, specifications, and purchase order documents.	
	3. Material is inspected to the extent necessary to assure conformance with technical and QA requirements of the purchase order documents.	
	4. Material and equipment are handled and stored to prevent damage and deterioration.	
	5. Items which have passed the required inspections and tests established by the specification are identified.	
	6. Conditions adverse to quality are properly identified and corrected.	
	7. Instructions, procedures, and drawings include appropriate qualitative and quantitative acceptance criteria for determining that important activities have been satisfactorily accomplished.	
	8. Tools, gages, instruments, and other measuring and testing devices used in activities affecting quality are properly calibrated and controlled.	
	9. Sufficient records including results of reviews and inspections are maintained.	
	10. The issuance of documents such as instructions, procedures, and drawings, including changes thereto, which prescribe all activities affecting quality are controlled.	

TABLE E.1. (contd)

Questions/Requests ^(a,b)	Responses ^(a,b)	Comments ^(a,b)
<p><u>QUESTION 40</u> (pertains to SAR Section 3.3.3.2, "Instrumentation"): CP&L should give complete design details for all instrumentation and show that instrumentation penetrations and systems do not cause any reduction in the ability of the DSC or HSM to meet the design criteria. See Question 1 (pertains to SAR Section 1.2.1) above.</p>	<p><u>RESPONSE:</u> Drawings RNT162-M-2600, RNT162-M-2621, and RNT162-M-2609 show design details for the DSC instrumentation. The only purpose of the instrumentation will be to collect data. It will cause no reduction in the operational ability of the DSC. Instrumentation in the HSM will be limited to thermocouple placement inside the HSM and at the heat shields. These thermocouples will not cause any loss of ability for the HSM to meet the design criteria. Drawings RNT162-C-1113 and RNT162-C-1114 show HSM and Heat Shield thermocouple configurations. For a detailed description of the instrumentation penetration, see response to Question 64, regarding SAR Section 5.4, "Operation Support System."</p>	<p>See submittals. (211,212) Q and R pertain to SAR Section 3.3.3, "Protection by Equipment and Instrumentation Selection."</p>
<p><u>QUESTION 41</u> (pertains to SAR Section 3.3.5.3, "Radiological Alarm System"): Does CP&L intend to include a radiation monitoring system as a part of the instrumentation package for the first three DSCs and HSMs? If CP&L does not intend to do so, why?</p>	<p><u>RESPONSE:</u> A radiation monitoring system will not be included in the instrumentation package for the first three DSCs and the 3x1 HSM array. Instead, radiation measurements will be taken after fuel loading as stated in the Topical Report. Quarterly surveys by Health Physics personnel will be conducted at the ISFSI site as performed in other areas of the plant.</p>	<p>See submittals. (211,212) Q and R pertain to SAR Section 3.3.5, "Radiological Protection."</p>
<p><u>QUESTION 42</u> (pertains to SAR Section 4.1.2.2, "Construction, Fabrication and Inspection"): Regarding HSM, CP&L should be specific about which American Concrete Institute codes and which American Society of Testing and Material (ASTM) Standards are used for this activity. Please indicate the specific code and describe how it was used, and what the result was for every case.</p>	<p><u>RESPONSE:</u> Part (a) of Section 4.2.1.2, "Construction, Fabrication and Inspection," of the ISFSI SAR will be modified to state that the HSM is designed in accordance with the requirements of ACI 349-80. All other standard codes and ASTM standards not listed in this section of the report are part of the testing, fabrication and construction requirements listed in Chapter 3 of ACI 349-80. Construction of the HSM will be in accordance with ACI 301-84, "Specifications for Structural Concrete for Buildings."</p>	<p>See submittals. (211,212) Q and R pertain to SAR Section 4.2.1, "Structural Specifications."</p>
<p>CP&L has not provided sufficient detail to be able to evaluate the ability of the reinforcing dowels to prevent possible overturning or sliding during "any accident condition." Also, this information is not included in SAR Section 8.2 ("Accident Analysis") of the submittal.</p>	<p>The design details of the HSM tie-down system are shown in Drawing RNT162-C-1102. As detailed in the response to Question 81, the dowels prevent overturning or sliding during accident conditions. Analysis of the reinforcing dowels and shear keys will be incorporated in Section 8.2, "Accident Analysis," of the SAR.</p>	

TABLE E.1. (contd)

Questions/Requests ^(a,b)	Responses ^(a,b)	Comments ^(a,b)
<p>QUESTION 43 (pertains to SAR Section 4.2.1.2, "Construction, Fabrication and Inspection"): Regarding the DSC, CP&L should be specific about which ASME and which ASTM codes and standards and which American Welding Society codes and standards it intends to use for this activity. Please indicate the specific code and describe how it was used, and what the result was for every case.</p>	<p>RESPONSE: The Dry Shielding Canister will be designed in accordance with the Documents, Codes, and Standards section of the fabrication specifications. A portion of this specification is listed below showing the appropriate ASME, CFR, AWS, ANSI, ASTM and other documents used for this activity.</p> <p><u>1.6.2</u> - Applicable portions of the latest revisions (unless otherwise stated) of the following documents in effect on the date of the purchase order shall apply where referenced in this specification.</p> <p><u>1.6.2.1</u> - American Society of Mechanical Engineers Boiler and Pressure Vessel Code (ASME Code).</p> <p><u>1.6.2.1.1</u> - ASME Code, Section II, Part C, 1983 Edition, with all addenda up to and including Winter 1983 Addenda.</p> <p><u>1.6.2.1.2</u> - ASME Code, Section III, Division 1, Subsection NCA, 1983 Edition, with all addenda up to and including Winter 1983 Addenda.</p> <p><u>1.6.2.1.3</u> - ASME Code, Section III, Division 1, Subsection NB, 1983 Edition, with all addenda up to and including Winter 1983 Addenda.</p> <p><u>1.6.2.1.4</u> - ASME Code, Section V, 1983 Edition, with all addenda up to and including Winter 1983 Addenda.</p> <p><u>1.6.2.1.5</u> - ASME Code, Section IX, 1983 Edition, with all addenda up to and including Winter 1983 Addenda.</p> <p><u>1.6.2.2</u> - Code of Federal Regulation (CFR).</p> <p><u>1.6.2.2.1</u> - CFR, Title 10, Part 21, "Reporting of Defects and Noncompliance."</p> <p><u>1.6.2.2.2</u> - CFR, Title 10, Part 50, Appendix B, "Domestic Licensing of Production and Utilization Facilities."</p>	<p>See submittals. (211,212) Q and R pertain to SAR Section 4.2.1, "Structural Specifications."</p>

TABLE E.1. (contd)

Questions/Requests ^(a,b)	Responses ^(a,b)	Comments ^(a,b)
QUESTION 43: (contd)	<p><u>RESPONSE: (contd)</u></p> <p><u>1.6.2.3</u> - American Welding Society (AWS) D.1.1-83, "Structural Welding Code."</p> <p><u>1.6.2.4</u> - American National Standards Institute (ANSI).</p> <p><u>1.6.2.4.1</u> - ANSI N45.2, "Quality Assurance Program Requirements for Nuclear Facilities."</p> <p><u>1.6.2.4.2</u> - ANSI N45.2.1, "Cleaning of Fluid Systems and Associated Components During Construction Phase of Nuclear Power Plants."</p> <p><u>1.6.2.4.3</u> - ANSI N45.2.2, "Packaging, Shipping, Receiving, Storage, and Handling of Nuclear Power Plants."</p> <p><u>1.6.2.5</u> - American Society for Testing and Materials (ASTM).</p> <p><u>1.6.2.5.1</u> - ASTM B29, "Standard Specification for Pig Lead."</p> <p><u>1.6.2.5.2</u> - ASTM A380, "Standard Recommended Practice for Cleaning and Descaling Stainless Steel Parts, Equipment, and Systems."</p> <p><u>1.6.2.6</u> - American Society of Nondestructive Testing, SNT-TC-1A, "Recommended Practice for Nondestructive Testing Personnel Qualification and Certification."</p>	
QUESTION 44 (pertains to SAR Section 4.2.3, "Individual Unit Description"): CP&L has stated that SAR Figures 4.2-1 and 4.2-2 ("Dry Shielded Canister" and "Horizontal Storage Module," respectively) are "from the Topical Report and are included in this (CP&L) report." Comparison of these drawings in both submittals reveals that neither drawing in the CP&L submittal correspond to any drawing in the NUTECH Topical Report. Outstanding	<p><u>RESPONSE:</u> Design drawings as listed in the response to Question 1 have been provided to show actual site-specific designs. Figures in the SAR will be revised and/or omitted to reflect the H. B. Robinson design. Additionally, references to the Topical Report will be eliminated in cases where site-specific designs deviate from Topical Report designs. SAR Section 4.2.3, "Individual Unit Description," will be revised accordingly.</p>	See submittals. (211,212) Q and R pertain to SAR Section 4.2, "Storage Structures."

TABLE E.1. (contd)

Questions/Requests (a,b)	Responses (a,b)	Comments (a,b)
<p><u>QUESTION 44: (contd)</u></p> <p>discrepancies in CP&L Figure 4.2-1 are the left end cap of the DSC, Sections B-B, C-C, and lack of dimensions. An outstanding discrepancy in CP&L Figure 4.2-2 is the storage array. In addition, the rear plug shown in Figure 4.2-2 has not been discussed or analyzed in the NUTECH Topical Report. NUTECH does not have a 3x1 array in their Topical Report. It is important to note that the conceptual layout for the 3x1 array implies that there is a right, a left, and a center module. However, no drawings have been supplied which show any of the required designs. CP&L should provide actual engineering (shop) drawings. CP&L should also make the correct references to Topical Report drawings, or in the case where CP&L site specific drawings deviate from the Topical Report drawings, CP&L should delete reference to the Topical Report.</p>		
<p><u>QUESTION 45 (pertains to SAR Section 4.3.2, "Electrical System"):</u> It is noted in this section that the first three DSC's may contain instruments for measuring temperature and radiation. Are any of these instruments to be located within the DSC confinement boundary, and if so, what provision will be made to insure that no leakage occurs through such penetrations during the life of the system?</p>	<p><u>RESPONSE:</u> Two of the first three DSCs will contain thermocouples to record temperature information. No instruments measuring radiation will be used. For a detailed explanation of the instrumentation and the related penetration integrity, refer to the response to Question 64.</p>	<p>See submittals. (211,212) Q and R pertain to SAR Section 4.3, "Auxiliary Systems."</p>
<p><u>QUESTION 46 (pertains to SAR Section 4.6, "Cathodic Protection"):</u> Even though the canister support assembly may be painted with "Carbo-Zinc 11," there is a high probability that some of the paint will be scraped off as the DSC slides into the HSM. Given the fact that the Type A36 structural steel has a higher electromotive (galvanic) potential than the Type 304 stainless steel, the corrosion which results will occur on the support rails over the 50-year design life will not cause failure of the support rails. CP&L should also draw reference to the expected rainfall and show how much water is expected to collect inside the HSM. It should then discuss the expected rate of evaporation of any water that collects in the corrosion problem.</p>	<p><u>RESPONSE:</u> The top surface of the DSC rail is to be machined to produce a maximum surface roughness of 63 microinches. The machined surface is to be plated with a high-phosphorus electroless nickel finish having a minimum thickness of 0.0005 inch. The remaining surfaces of the rail shall be painted with Carbo-Zinc 11. Consequently, the A36 structural steel will not have any exposed surfaces during its 50-year design life. Therefore, corrosion of the A36 steel will be controlled. This design change will insure the integrity of the support structure for its design life.</p>	<p>See submittals. (211,212) Q and R pertain to SAR Section 4.7, "Fuel Handling Operation System."</p>
	<p>As stated previously in the response to Question 26, no rainwater buildup will occur in the HSM due to the drain pipes and flashing protection. Hence, excessive moisture which could accelerate corrosion will not collect inside the HSM.</p>	

TABLE E.1. (contd)

Questions/Requests (a,b)	Responses (a,b)	Comments (a,b)
<p><u>QUESTION 47 (pertains to SAR Section 4.7.1, "Structural Specification"):</u> The CP&L report (SAR) states that: "The engineering and design bases for other components (transport skid and hydraulic ram) are discussed in Chapter 3 of the NUHOMS Topical Report." It is interesting to note that the NUTECH Topical Report states that "The handling equipment required to implement the NUHOMS storage system will be site-specific. This equipment includes a cask handling crane at the reactor fuel pool, a transfer cask, a cask skid, heavy-haul vehicle and a hydraulic ram" (page 3.1-8 NUTECH Topical Report). Although NUTECH has provided some of the criteria for the above equipment neither NUTECH nor CP&L has supplied any design drawings, design calculations, fabrication procedures, or in any way presented a credible case that these items can meet the design criteria established by NUTECH. Because these items are site-specific, CP&L is requested to do so.</p>	<p><u>RESPONSE:</u> The design basis for the handling and transfer equipment is discussed in the NUTECH Topical Report Section 3.1.2.2 and Table 3.1-5. The detailed design of such a system is site-specific, but not safety related. The transfer crane is already licensed at the H. B. Robinson site. The other equipment is not safety related. Drawings have been provided, however, to show the configuration of the other equipment. The Ram Assembly is shown in Drawing RNT162-M-2200 while the Trailer/Skid Assembly is shown on Drawing RNT162-M-2300. Fabrication procedures will be developed by the fabricator(s) based on requirements set forth in the procurement specifications.</p>	<p>See submittals. (211,212) Q and R pertain to SAR Section 4.7, "Fuel Handling Operation System."</p>
<p><u>QUESTION 48 (pertains to SAR Section 5.1.1.1, "Preparation of the Transfer Cask and Canister"):</u> CP&L states that: "Measures will be taken to ensure that no known failed fuel will be placed in dry storage." They also reference fuel specifications in SAR Section 10.0 ("Operating Controls and Limits"). However, neither SAR Section 5.0 ("Operations Systems") nor SAR Section 10.0 ("Operating Controls and Limits") define any procedure which will ensure that defective, i.e., ruptured, cracked, pin-holed, etc., fuel rods will not inadvertently be loaded into the DSC. CP&L is requested to define this procedure and specify the type of measuring instruments they intend to use to verify that no leaky fuel rods will be stored in the DSC. CP&L is also requested to specify the criteria that they intend to use, i.e., permissible level of radioactive isotope escaping from fuel rods, etc.</p>	<p><u>RESPONSE:</u> Visual inspection is to be performed through closed circuit television cameras. "A fuel assembly will be acceptable for dry storage if no fuel rod defects (i.e., ruptures, cracks, etc.) are visible. The coolant activity while the fuel to be stored was in the core indicated no failed fuel."</p>	<p>See submittals. (211,212) Q and R pertain to SAR Section 5.1.1, "Narrative Description."</p>
<p><u>QUESTION 49 (pertains to SAR Section 5.1.1.1E, "Preparation of the Transfer Cask and Canister"):</u> CP&L is requested to provide details of the "spanner" referred to. What is it? How is it installed?</p>	<p><u>RESPONSE:</u> The collar has been redesigned to eliminate the need for a spanner, which was to be used to temporarily secure the cask liner in place. References to a spanner will be omitted from the next revision of the SAR.</p>	<p>See submittals. (211,212) Q and R pertain to SAR Section 5.1.1, "Narrative Description."</p>

TABLE E.1. (contd)

Questions/Requests (a,b)	Responses (a,b)	Comments (a,b)
QUESTION 50 (pertains to SAR Section 5.1.1.1, "Preparation of the Transfer Cask and Canister"): CP&L is requested to address the following questions: Number 1 of Section 10.2.2.2; 10.3.2.1; 10.3.2.2; 10.3.2.3 (1 and 2) which may be found in the NRC response to the NUTech Topical Report.	<p>RESPONSE: The five parts to the response are as follows,</p> <p>a) <u>Technical Conditions and Characteristics</u> - As stated in response to Question 1, Section 1.1, the <u>Metals Handbook</u>, R. M. Parke, takes the position that "helium does not diffuse through solid iron." Additionally, the results of discussion with scientists in the helium industry (at Union Carbide and the U.S. Bureau of Mine's Helium Operation) indicate that as a practical matter, helium does not diffuse through steel or stainless steel. Based on the above discussion, it can be assumed that the only means for helium to leak from the canister is through defects in the DSC welds. However, all welds will be nondestructively examined. If these examinations show indications of a defect, the defect will be repaired in accordance with ASME, Section III, Subsection NB requirements and reexamined. Those fabrication requirements will provide a DSC that has defect free welds. Therefore leak rate does not apply to these welds.</p> <p>b) <u>10.3.2.1 DSC Vacuum Pressure During Drying</u> - See response to Question 94 of this submittal.</p> <p>c) <u>10.3.2.2 DSC Helium Backfill Pressure</u> - The backfilling procedures have been changed to include pressure monitoring until an equilibrium has been reached.</p> <p>d) <u>10.3.2.3 DSC Helium Leakage Rate of Primary Weld, Question 1</u> - The 631 cm³ of He does correspond to 2 atmospheres and 473 cm³ corresponds to 1.5 atmospheres. The value of 631 cm³ will be changed to 473 cm³. This revision has no impact on the leak rate specification of the primary weld.</p> <p>e) <u>10.3.2.3 DSC Helium Leakage Rate of Primary Weld, Question 2</u> - See the above response to Question 10.2.2.2 of the NUTech Topical Report.</p>	See submittals. (211,212) Q and R pertain to SAR Section 5.1.1, "Narrative Description."

TABLE E.1. (contd)

Questions/Requests ^(a,b)	Responses ^(a,b)	Comments ^(a,b)
<p>QUESTION 51 (pertains to SAR Section 5.1.1.3, "Cask Drying Process"): See Question 1 and 2 in Section 5.1.1.3 of the NRC response to the NURECH Topical Report.</p>	<p>RESPONSE: The two parts of the response are as follows:</p> <p>a) <u>5.1.1.3 Cask Drying Process, Question 1</u> - The procedures have been revised to include hydrostatic testing of the DSC and monitoring the pressure.</p> <p>The following procedures will be incorporated in the Topical Report:</p> <ul style="list-style-type: none"> • Attach a self priming pump to valve #2 of the siphon line and drain more than 15 gallons from the DSC. • Remove the self priming pump. • Dry any water from top lead plug and DSC interface and then seal weld the upper stainless steel cladding plate of the top lead plug to the canister body. • Connect 0-75 psig pressure gauge to valve #2 on vent tube, and open valves #1 and #2. • Connect demineralized water supply to intake side of hydro-pump and connect hose from discharge of hydro-pump to valve #2 of siphon tube. • Open valves #1 and #2 on siphon tube. • Activate hydro-pump and pressurize the DSC to 50 psig as read on pressure gauge. • Once internal pressure of the DSC has reached 50 psig, close valves #1 and #2 on siphon tube and disengage hydro-pump. • Monitor pressure for 10 minutes. 	<p>See submittals, (211,212) Q and R pertain to SAR Section 5.1.1, "Narrative Description."</p>

TABLE E.1. (contd)

Questions/Requests (a,b)	Responses (a,b)	Comments (a,b)
<u>QUESTION 51: (contd)</u>	<p data-bbox="961 272 1167 297"><u>RESPONSE: (contd)</u></p> <ul style="list-style-type: none"> <li data-bbox="978 321 1541 467">• After 10 minutes, examine the primary closure weld (weld between top lead plug and DSC shell) for leaks. Continue monitoring the pressure throughout the examination. If a leak is detected or an internal pressure drop is detected, release pressure, and repair weld. <li data-bbox="978 492 1541 613">• If no leaks are detected, disconnect siphon hose from hydro-pump discharge and connect to plant's low-level radioactive waste system. Open valves on siphon tubes, allowing pressure to drop to atmospheric pressure. <li data-bbox="978 638 1255 662">• Connect helium to DSC. <li data-bbox="978 686 1436 711">• Pressurize DSC with 1.5 atm of helium. <li data-bbox="978 735 1423 760">• Leak-sniff the weld for helium leaks. <li data-bbox="978 784 1367 808">• If a leak is found, repair weld. <li data-bbox="978 833 1377 857">• If no leak, release the pressure. <li data-bbox="978 881 1541 954">• Remove the pressure gauge from valve #2 on the vent tube and connect in accordance with the existing procedure steps #53 and #58. <li data-bbox="978 979 1541 1027">• Open valve #2 and allow the premeasured quantity of helium to flow into the DSC cavity. <li data-bbox="978 1052 1541 1101">• Close valve #1 and #2 on helium. Monitor pressure until pressure equilibrium is reached. <p data-bbox="961 1117 1541 1206">b) <u>5.1.1.3 Cask Drying Process, Question 2</u> - The differential pressure across the canister wall when the helium sniffer leak test is performed is 1.5 atm.</p>	

TABLE E.1. (contd)

Questions/Requests (a,b)	Responses (a,b)	Comments (a,b)
<p><u>QUESTION 52</u> (pertains to SAR Section 5.1.1.3, "Cask Drying Process"): See Question 1 in Section 4.2.3.1 of the NRC response to the NUTech Topical Report.</p>	<p><u>RESPONSE:</u> The longitudinal weld will be radiographed to assure that the integrity of the weld is equal to that of the parent metal. See Question 50, regarding SAR Section 5.1.1.1 ("Preparation of the Transfer Cask and Canister"), Subquestion 10.2.2.2 for additional details concerning the leak rate.</p>	<p>See submittals. (211,212) Q and R pertain to SAR Section 5.1.1, "Narrative Description."</p>
<p><u>QUESTION 53</u> (pertains to SAR Section 5.1.1.4, "DSC Sealing Operation"): In this operation CP&L states, "place the cask lid onto the cask and bolt the lid into place." Reference to SAR Figure 5.1-1, "Cask Extension for GE IF-300 Cask," shows a cask extension and a lid as substitute parts for the existing IF-300 cask deceleration limiter. Also, SAR Figure 5.1-3, "Handling Operations Flow Sheet," does not refer to the deceleration limiter. However, reference to the NUTech Figure 5.1-2 shows a cask deceleration limiter. CP&L and NUTech should be aware that the deceleration limiter is a structural energy absorbing device designed to crush in the event of a cask drop. If this device is not used by CP&L, then the reconfigured IF-300 shipping cask must be evaluated by CP&L for site-specific use. For example, refer to Questions 12 and 13 (pertain to SAR Section 1.3.1.3) above.</p>	<p><u>RESPONSE:</u> Use of the deceleration limiter, a structural energy absorbing device designed to crush in the event of a cask drop, is not required for the H. B. Robinson site-specific ISFSI. The response to Question 3 (pertains to SAR Section 1.2.3.1, "Structural Features") above describes the handling of the cask. This response also describes the handling of the cask. This response also describes the envelope for a horizontal drop accident. Prior to placing the IF-300 into the fuel pool, the docking collar (cask extension) will have been installed on the cask. When the loaded cask is being handled in the tilting operation and then transported to the HSM, both the cask collar and cask load will be in place. Therefore, the deceleration limiter will not be required.</p>	<p>See submittals. (211,212) Q and R pertain to SAR Section 5.1.1, "Narrative Description."</p>
<p><u>QUESTION 54</u> (pertains to SAR Section 5.1.1.6, "Loading of Canister into the HSM"): With reference to the proper axial positioning of the DSC within the HSM both during normal operation and in case of seismic event, what provision is CP&L taking to achieve correct axial positioning? See Question 12 in Section 7.3.2 and Question 6 in Section 8.1.1.5 of the NRC response to the NUTech Topical Report.</p>	<p><u>RESPONSE:</u> As described in response to Question 33, a two-inch stop plate will be bolted to the DSC Support Assembly rails. These plates will assure correct axial positioning of the DSC 3.5 inches from the HSM end wall. Once the canister is in place the seismic retainer shown in drawing RNT162-C-1112 will be inserted through the rear access opening to couple with the grapple assembly. This anchoring operation is described in the response to Question 15. Consequently, due to the seismic retainer and stop plates, axial positioning during normal operation and possible seismic events is assured.</p>	<p>See submittals. (211,212) Q and R pertain to SAR Section 5.1.1, "Narrative Description."</p>

TABLE E.1. (contd)

Questions/Requests ^(a,b)	Responses ^(a,b)	Comments ^(a,b)
<p><u>QUESTION 55 (pertains to SAR Section 5.1.1.6E, "Loading of Canister in to the HSM"):</u> CP&L is requested to present details for the "optical alignment system." What does it consist of? How will CP&L use it? What are the specifications? What is the measurement resolution of the system? Are alignment marks to be affixed to the IF-300 cask or the HSM? Does CP&L intend to perform a dry run of the alignment system before attempting to load a DSC filled with spent fuel into the HSM?</p>	<p><u>RESPONSE:</u> The services of an outside specialty contractor will be secured to provide equipment, fixtures, and labor required to determine the centerlines of the transfer cask and DSC support assembly, determine alignment of the RAM mounting plate with the DSC support assembly, and services required to bring these components into alignment.</p> <p>The following drawings will be used by the specialty contractor as references for development of procedures and design alignment tooling; RNT162-M-2400, RNT162-M-2401, RNT162-M-2402, and RNT162-M-2222.</p> <p>Typical optical instruments which might be used are: an alignment telescope, bore targets, and optical transit squares.</p> <p>These instruments are accurate to the tenths of mils. Alignment marks will be affixed to the HSM and cask (or skid) and designed for a 20-year life. Handling tests will be performed on the modules.</p>	<p>See submittals. (211,212) Q and R pertain to SAR Section 5.1.1, "Narrative Description."</p>
<p><u>QUESTION 56 (pertains to SAR Section 5.1.1.7, "Monitoring Operations"):</u> SAR Section 5.1.1.7 reports that personnel will visually monitor HSM air inlets and perform maintenance work as required. This work is likely to be performed in an area where radiation streaming and secondary gamma photon production is present. Have estimated dose rates been calculated for work around the air inlets?</p>	<p><u>RESPONSE:</u> Estimated dose rates have been calculated for air openings. The bounding condition exists at the HSM air outlets. The calculated dose rate there is 38 mrem/h <u>inside</u> the shield block opening, flush with the top of the HSM.</p>	<p>See submittals. (211,212) Q and R pertain to SAR Section 5.1.1, "Narrative Description."</p>
<p><u>QUESTION 57 (pertains to SAR Section 5.1.1.8, "Unloading of the DSC from the HSM"):</u> What does the last sentence in this section mean? Does it mean that CP&L will make provisions to return the DSC to the spent fuel pool only during the demonstration year? Under what conditions would CP&L return the DSC to the fuel pool, "if necessary." Please explain.</p>	<p><u>RESPONSE:</u> CP&L can return the DSC, inside the IF-300, to the spent fuel pool during the life of the facility. Possible conditions upon which the DSC would be returned include exceeding the design limits shown in SAR Section 10.2, "Limits for the Surface Dose Rate of the HSM While the DSC is in Storage," and Section 10.3, "Limits for the Maximum Air Temperature Rise After Storage." Additionally, if shipping were required, the DSC would be returned to the decontamination area or to the fuel pool for removal of the cask collar and placement of the BWR head on the IF-300. The last sentence in SAR Section 5.1.1.8, "Unloading of the DSC from the ISFSI," will be revised to reflect this information.</p>	<p>See submittals. (211,212) Q and R pertain to SAR Section 5.1.1, "Narrative Description."</p>

TABLE E.1. (contd)

Questions/Requests (a,b)	Responses (a,b)	Comments (a,b)
<p><u>QUESTION 58</u> (pertains to SAR Section 5.1.3.1, "Criticality Prevention"): Refer to Question 5 in Section 3.4.4, of the NRC response to NUTECH.</p>	<p><u>RESPONSE:</u> BORAL tubes will be used in all seven sleeves of the DSCs at the Robinson ISFSI. Consequently, no controls are needed to ensure that low burnup assemblies are not inadvertently loaded into unborated canisters.</p>	<p>See submittals. (211,212) Q and R pertain to SAR Section 5.1.3, "Identification of Subjects for Safety Analysis."</p>
<p><u>QUESTION 59</u> (pertains to SAR Section 5.1.3.4, "Instrumentation"): See Question 40 (pertains to SAR Section 3.3.3.1, "Instrumentation") and Question 32 (pertains to SAR Section 3.1.2.1, "Overall Functions of the Facility") of this submittal, and modify this section of the licensing application to reflect the actual instrumentation for the first three units.</p>	<p><u>RESPONSE:</u> SAR Section 5.1.3.4, "Instrumentation," will be revised to include details of the instrumented DSC and HSM as shown on design drawings RNT162-M-2600, RNT162-M-2609, RNT162-M-2621, RNT162-M-2622, RNT162-C-1113, and RNT162-C-1114. The instrumentation description is contained in the response to Question 64.</p>	<p>See submittals. (211,212) Q and R pertain to SAR Section 5.1.3, "Identification of Subjects of Safety Analysis."</p>
<p><u>QUESTION 60</u> (pertains to SAR Section 5.2.1.1, "Irradiated Fuel Cask"): SAR Table 5.1-2 refers to a "Transport Lid." Is it possible that CP&L intended to refer to a "Transport System?" Also, as mentioned in Question 1 (pertains to SAR Section 1.2.1, "General Description") CP&L has not provided any engineering drawings, specifications, procedures, or calculations for any of the items mentioned in SAR Figure 5.1-2, "Hydraulic Ram System." These include systems and subsystems of the modifications to the IF-300 shipping cask; the transport system; the optical alignment system; and the hydraulic ram. CP&L needs to show that these items are designed to proper specifications, and in the case of the shipping cask, that the modifications do result in reduced safety.</p>	<p><u>RESPONSE:</u> SAR Table 5.2-1 ("Transfer System Component Description"), not Table 5.1-2, refers to a "Transport Lid." This typographical error will be corrected to read "Transport Skid." The ISFSI Ram Assembly drawing RNT162-M-2200 has been provided to show details of the items mentioned in SAR Figure 5.2-1, "Hydraulic Ram System." Additionally, drawing RNT162-M-2300 has been provided to show details of the Transfer Cask Skid and Alignment System. For an example of a typical alignment system see the NUTECH Topical Report, Section 10.3.2.8. The modifications to the shipping cask are shown in drawings RNT162-M-2701, RNT162-M-2702 and RNT162-M-2703. For a detailed description of the modifications refer to the response to Question 53.</p>	<p>See submittals. (211,212) Q and R pertain to SAR Section 5.2.1, "Spent Fuel Handling and Transfer."</p>
<p><u>QUESTION 61</u> (pertains to SAR Section 5.2.1.1C, "Irradiated Fuel Cask"): The SAR indicates that a new cask lid has been designed for the ISFSI operation (see also SAR Figure 5.1-1, "Cask Extension for GE IF-300 Cask"). Has a shielding analysis been performed to determine if the radiological criteria are satisfied? If so, what are the results of the analysis?</p>	<p><u>RESPONSE:</u> Radiological criteria are satisfied by the shielded end plug in the DSC. The end plug reduces the dose criteria and therefore further analysis to determine the additional shielding capability of the new cask lid was deemed unnecessary. Shielding analysis information for the DSC end plug has been incorporated into SAR Section 7.3.2.2.</p>	<p>See submittals. (211,212,214) Q and R pertain to SAR Section 5.2.1.1, "Functional Description."</p>

TABLE E.1. (contd)

Questions/Requests (a,b)	Responses (a,b)	Comments (a,b)
<p><u>QUESTION 62</u> (pertains to SAR Section 5.2.1.2, "Safety Features"): It is stated in this section that "the amount of force which the ram may exert is limited to 6,200 pounds unless overridden by the systems operator." Either the systems operator should not be permitted to override this amount of force or the design criteria should be revised to reflect an appropriate higher permissible force. Please make an appropriate correction to the SAR.</p>	<p><u>RESPONSE:</u> The nominal ram load is expected to be 6,200 pounds. However, the ram and DSC and DSC supports were designed to withstand 20,000 pounds. Therefore, the operator can override the controls and apply 20,000 pounds of force, the maximum force obtainable from the ram.</p> <p>SAR Section 5.2.1.2 has been revised to reflect a maximum ram capacity of 20,000 pounds.⁽²¹⁴⁾ All analyses were performed for this maximum load capacity. SAR will be revised at appropriate locations to reflect this change.</p>	<p>See submittals.^(211,212,214) Q and R pertain to SAR Section 5.2.1, "Spent Fuel Handling and Transfer."</p>
<p><u>QUESTION 63</u> (pertains to SAR Section 5.3.2, "Components/Equipment Spares"): In the event that components could be damaged during the life of the installation, what provision is made for installation of spare or alternative equipment to provide for continuity of safety under normal and off-normal conditions?</p>	<p><u>RESPONSE:</u> The only component postulated to be damaged during the life of the installation is the air outlet shielding block. As described in SAR Section 8.2.1, "Loss of Air Outlet Shielding," a tornado induced missile could damage or knock off the shielding blocks. Consequently, two additional shielding blocks will be precast during construction and maintained as spares by CP&L. SAR Section 5.3.2, "Components/Equipment Spares," will be revised to reflect this information. SAR Section 8.2.1, "Loss of Air Outlet Shielding," describes the process to replace a damaged or lost shielding block.</p>	<p>See submittals.^(211,212) Q and R pertain to SAR Section 5.3, "Other Operating Systems."</p>
<p><u>QUESTION 64</u> (pertains to SAR Section 5.4, "Operating Support System"): Describe the instrumentation systems for the first three instrumented DSC/HSM systems. See these previous questions: Question 59 (pertains to SAR Section 5.1.3.4, "Instrumentation"), and Question 40 (pertains to SAR Section 3.3.3.1, "Instrumentation"), and Question 32 (pertains to SAR Section 3.1.2.1, "Overall Functions of the Facility").</p>	<p><u>RESPONSE:</u> Instrumentation is not required to support the operation of the ISFSI. For research purpose, however, two of the DSCs to be installed at the H. B. Robinson facility have been designed to accept instrumentation. Instrumentation was included as part of an agreement between CP&L, EPRI and the DOE to augment the U.S. database on LWR fuel rods in dry storage.</p> <p>The DSC thermocouples will be connected to an external cable by means of a specially designed feed-through. This feed-through incorporates the same backup weld-seal philosophy used in the DSC</p>	<p>See submittals.^(211,212) Q and R pertain to SAR Section 5.0, "Operation Systems."</p>

TABLE E.1. (contd)

Questions/Requests ^(a,b)	Responses ^(a,b)	Comments ^(a,b)
QUESTION 64: (contd)	<p><u>RESPONSE: (contd)</u></p> <p>containment design. Details are shown in Drawings RNT162-M-2600 and RNT162-M-2609. After the penetration plug assembly has been welded to the bottom of the OSC cover plate, a sleeve will be welded over the plug, forming a redundant seal. Thermocouple sheaths will likewise be brazed to the plug assembly at inner and outer penetrations. Leakage through the individual sheaths will be prevented by metalizing the aluminum oxide insulating crystals at the inner and outer penetrations.</p> <p>HSM instrumentation will consist of thermocouples cast in place in the concrete and others attached to the surface and at various locations on the heat shield. Details of the HSM instrumentation are shown on drawings RNT162-C-1113 and RNT162-C-1114.</p>	
<p>QUESTION 65 (pertaining to SAR Section 6.5, "Radiological Impact of Normal Operations - Summary"): The ALARA considerations presented in the SAR seem to be a statement that radiological impact was considered and that several design criteria were established. Another interpretation of ALARA would suggest that the cost impact of radiological impact reductions be considered and that the impact be as low as reasonably achievable where "reasonably" is taken in the sense of cost benefit. For example, what would the cost impact be of reducing the radiological impact from the ISFSI by a factor of 2? Has it been shown that further reductions in radiological impact are impractical or unreasonable? Have reductions in the radiological impact been considered at all?</p>	<p><u>RESPONSE:</u> Numerous radiological impact reductions have been evaluated and some have been incorporated in the design (this was how the various shield thicknesses were derived). Detailed ALARA considerations are presented in the SAR in Section 7.1, "Ensuring that Occupational Radiation Exposures are ALARA." However, as a summary, the design of the H. B. Robinson ISFSI was evaluated against existing, established ALARA guidelines. For instance, the system was designed so that the offsite does will be substantially below the one mrem/year guideline established by the NRC for additional waste storage facilities (see NRC generic letter 81-38). Also, the ISFSI does not violate the existing licensed H. B. Robinson on-site health physics and ALARA procedures.</p>	<p>See submittals. (211,212) Q and R pertain to SAR Section 6.0, "Waste Confinement and Management."</p>
<p>QUESTION 66 (pertains to SAR Section 7.1.1, "Policy Consideration"): The northern side of the ISFSI borders on a railroad track. What type of access control is provided in this area? Can unbadged personnel work in this area?</p>	<p><u>RESPONSE:</u> See the response to Question 22, which specifies the dimensions near the railroad track.</p>	<p>See submittals. (211,212) Q and R pertain to SAR Section 7.1, "Ensuring that Occupational Radiation Exposures are ALARA."</p>

TABLE E.1. (contd)

Questions/Requests (a,b)	Responses (a,b)	Comments (a,b)
<p><u>QUESTION 67</u> (pertains to SAR Section 7.1.2, "Design Considerations"): Are the docking collar and the HSM recess design identical to that described in the NUTECH Topical Report? Has a shielding analysis of the docking arrangement been performed? If so, what are the results.</p>	<p><u>RESPONSE:</u> The docking collar and the HSM recess are identical to that described in the Topical Report. A shielding analysis has been performed and will be presented in the revised SAR.</p>	<p>See submittals. (211,212) Q and R pertain to SAR Section 7.1, "Ensuring that Occupational Radiation Exposures are ALARA."</p>
<p><u>QUESTION 68</u> (pertains to SAR Section 7.3.2.2, "Shielding Analysis"): The shielding analysis results of SAR Table 7.3-2 (based on table of same number in NUTECH Topical Report) use units of mR/h for both neutron and gamma dose rates. Since the Roentgen (R) is defined only for photon radiation, are we to assume that the neutron dose rates are in terms of mrad/h or mrem/h?</p>	<p><u>RESPONSE:</u> Units on SAR Table 7.3-2, "Shielding Analysis Results," will be changed from mR/h to mrem/h.</p>	<p>See submittals. (211,212) Q and R pertain to SAR Section 7.3.2, "Shielding."</p>
<p><u>QUESTION 69</u> (pertains to SAR Section 7.3.2.2, "Shielding Analysis"): SAR Table 7.3-2, "Shielding Analysis Results," is identical to that presented in the Topical Report. Considering the facts that the source strengths differ, the cask docking arrangement is different, and there is a rear plug, please explain why the shielding analysis results are identical. Also the table shows no results for the dose rates for the rear penetration.</p>	<p><u>RESPONSE:</u> A revised SAR Table 7.3-2, "Shielding Analysis Results," will be presented which will include values for dose rates at the rear penetration.</p>	<p>See submittals. (211,212) Q and R pertain to SAR Section 7.3.2, "Shielding."</p>
<p><u>QUESTION 70</u> (pertains to SAR Section 7.4.1, "Operational Dose Assessment"): Table 7.4-1, "Summary of Estimated Onsite Doses During Fuel Handling Operations," of the CP&L ISFSI SAR was constructed from NUTECH Topical Report. What was the basis used for estimating the dose rates in this table? Are these dose rates estimated by extrapolating the results of SAR Table 7.3-2, "Shielding Analysis Results?" Were conversion factors used to convert mR to mrem (for gamma), or were they simply considered to be equivalent?</p>	<p><u>RESPONSE:</u> Dose rates in SAR Table 7.4-1, "Summary of Estimated Onsite Doses During Fuel Handling Operations," were extrapolated from surface dose rates obtained in shielding calculations, including the analysis which produced SAR Table 7.3-2, "Shielding Analysis Results." Dose rates are presented in mrem/h and were converted from fluxes as described in Section 7.2.1 of the Topical Report. More detail was added to the shielding analysis (SAR Section 7.3.2) and SAR Section 7.4.1 about dose rate estimation. (214)</p>	<p>See submittals. (211,212,214) Q and R pertain to SAR Section 7.4, "Estimated Onsite Collective Dose Assessment."</p>
<p><u>QUESTION 71</u> (pertains to SAR Section 7.4.1, "Operational Dose Assessment"): With regard to SAR Table 7.4-1, "Summary of Estimated Onsite Doses During Fuel Handling Operations," are the estimated times required to complete a task "best estimates" to complete the task as defined, or do they have some allowance for cases in which the task work does not proceed as planned (e.g., equipment failure).</p>	<p><u>RESPONSE:</u> The estimated times shown in SAR Table 7.4-1, "Summary of Estimated Onsite Doses During Fuel Handling Operations," are conservative estimates. They are based on engineering judgment and experience with identical and/or similar operations.</p>	<p>See submittals. (211,212) Q and R pertain to SAR Section 7.4, "Estimated Onsite Collective Dose Assessment."</p>

TABLE E.1. (contd)

Questions/Requests (a,b)	Responses (a,b)	Comments (a,b)												
<p><u>QUESTION 72 (pertains to SAR Section 7.4.1, "Operational Dose Assessment"):</u> Please provide missing data on SAR Table 7.4-1, "Summary of Estimated Onsite Doses During Fuel Handling Operations," on line entitled "Transport Cask to HSM."</p>	<p><u>RESPONSE:</u> The following data will be added to SAR Table 7.4-1, "Summary of Estimated Onsite Doses During Fuel Handling Operations," on the line entitled "Transport Cask to HSM":</p> <table><tr><td>Number of Personnel</td><td>5</td></tr><tr><td>Time (hours)</td><td>0.5</td></tr><tr><td>Average Distance from Cask Surface (feet)</td><td>5.0</td></tr><tr><td>Dose Rate (mrem/h)</td><td>2.0</td></tr><tr><td>Dose Rate per Person (mrem)</td><td>1.0</td></tr><tr><td>Total Personnel Dose (mrem)</td><td>5.0</td></tr></table> <p>The missing data has been added to SAR Table 7.4-1 and procedures used in obtaining the results are described in SAR Section 7.4.1. (214)</p>	Number of Personnel	5	Time (hours)	0.5	Average Distance from Cask Surface (feet)	5.0	Dose Rate (mrem/h)	2.0	Dose Rate per Person (mrem)	1.0	Total Personnel Dose (mrem)	5.0	<p>See submittals. (211,212,214) Q and R pertain to SAR Section 7.4, "Estimated Onsite Collective Dose Assessment."</p>
Number of Personnel	5													
Time (hours)	0.5													
Average Distance from Cask Surface (feet)	5.0													
Dose Rate (mrem/h)	2.0													
Dose Rate per Person (mrem)	1.0													
Total Personnel Dose (mrem)	5.0													
<p><u>QUESTION 73 (pertains to SAR Section 7.4.2, "Storage Term Dose Assessment"):</u> SAR Table 7.4-2, "Estimated Annual Onsite Doses During Storage Phase," contains estimate of doses to personnel assigned to various buildings and work areas around the plant. However, only some of the buildings and work areas are included. Why are not all plant facility occupants covered by this table?</p>	<p><u>RESPONSE:</u> SAR Table 7.4-2, "Estimated Annual Onsite Doses During Storage Phase," shows typical buildings and areas at various locations surrounding the HSM. Dose rates in other locations not listed in SAR Table 7.4-2, "Estimated Annual Onsite Doses During Storage Phase," would be the same as those listed which are located at similar distances away from the HSM.</p>	<p>See submittals. (211,212) Q and R pertain to SAR Section 7.4, "Estimated Onsite Collective Dose Assessment."</p>												
<p><u>QUESTION 74 (pertains to SAR Section 7.4.2, "Storage Term Dose Assessment"):</u> In SAR Figure 7.4-2, "Dose Rate Versus Distance from Surface of HSM (Assuming 8 Modules)," dose rate is specified as a function of distance from an eight-module facility. How are these results relatable to those reported in the NUTECH Topical Report. Were these results generated by an independent SKYSHINE-II run, or were they obtained by scaling the NUTECH results to site-specific conditions? If an independent SKYSHINE-II run was performed, are the input and output data available? If scaling was performed, what was the scaling factor used and what was the basis for this factor?</p>	<p><u>RESPONSE:</u> Scaling was used to obtain SAR Figure 7.4-2, "Dose Rate Versus Distance from Surface of HSM (Assuming 8 Modules)." SKYSHINE-II results were scaled by a factor of 8/184. Direct doses were scaled by a factor which decreases linearly from 1.0 (on contact) to 8/92 at a distance away from the array. Figure 7.4-2 is graphically incorrect in the existing SAR Figure 7.4-2; it has been completely revised to represent a three-module facility. (214)</p>	<p>See submittals. (211,212,214) Q and R pertain to SAR Section 7.4, "Estimated Onsite Collective Dose Assessment."</p>												
<p><u>QUESTION 75 (pertains to SAR Section 7.6.2, "Analysis of Multiple Contribution"):</u> In SAR Section 7.6.2, "Analysis of Multiple Contribution," it is estimated that the dose to the maximally exposed member of the public from ISFSI operations is 0.3 mrem/yr. What is the estimated dose to this individual from combined site operations? Such results are necessary to demonstrate compliance with 40CFR190. Also, the dose is compared to the "average</p>	<p><u>RESULTS:</u> The estimated additional dose contribution of the ISFSI to the maximally exposed member of the public is 0.3 mrem/yr, which is insignificant compared to the 25 mrem/yr allowed by 40CFR190 for combined operation.</p> <p>Guidance may be taken from NUREG-0800 (Appendix 11.4-A, Section B, "Storage Facility Requirements")</p>	<p>See submittals. (211,212) Q and R pertain to SAR Section 7.6, "Estimated Offsite Collective Dose Assessment."</p>												

TABLE E.1. (contd)

Questions/Requests ^(a,b)	Responses ^(a,b)	Comments ^(a,b)
<p><u>QUESTION 75: (contd)</u></p> <p>natural background dose of 200 mrem per year." This value is very high for "natural background dose" as the term is conventionally used (i.e., the sum of radiation from cosmic and terrestrial sources). Wouldn't 100 mrem/yr be a more appropriate value? Also, while a comparison to background is a good means of placing dose in perspective, a more appropriate comparison would be with applicable standards.</p>	<p><u>RESPONSE: (contd)</u></p> <p>and USNRC letter SECY-81-383 (Section III, paragraph "a") which describe the application of 40CFR90 to the storage of low level wastes at reactor sites.</p>	<p></p>
<p><u>QUESTION 76 (pertains to SAR Section 8.0, "Analysis of Design Events"):</u> A variety of analysis results are presented to substantiate these results. Please provide details of any analysis for which results are presented unless the source of the analysis is available from one of the documents identified in SAR Section 1.5, "Material Incorporated by Reference," as being incorporated by reference.</p>	<p><u>RESPONSE:</u> SAR Section 8.0, "Analysis of Design Events," will be revised to present analyses for certain design events which are significantly different from the Topical. If the analyses are similar the appropriate section of the Topical will be referenced. For analyses in which the technique and methodology are identical but the loading differs, the results in the Topical will be factored accordingly and presented in the SAR. For new analyses not included in the Topical, such as the foundation and instrument penetration analyses, complete detailed analyses will be added to the revised Section 8.0. The foundation analyses can be found in the response to Question 86. Therefore, all new or significantly revised analyses will be added to Section 8.0 of the SAR. SAR Section 8.0 has been revised⁽²¹⁴⁾ to present design analyses which are significantly different from those presented in the NUTECH Topical Report.</p>	<p>See submittals. (211,212,214)</p>
<p><u>QUESTION 77 (pertains to SAR Section 8.1.1, "Normal Operation Analysis Design Basis Internal Pressure Loads"):</u> CP&L states that the DSC is designed for "internal pressures up to 50 psig under accident conditions." The NUTECH Topical Report indicates that the design accident results in an internal pressure of 39.7 psig. CP&L should verify which is correct and change the text accordingly.</p>	<p><u>RESPONSE:</u> SAR Section 8.1.1 b), "Design Basis Internal Pressure Loads," inadvertently states the accident pressure as 50 psig inside the DSC. The correct value of 39.7 psig will be included in Section 8.1.1, "Normal Operation Analysis Design Basis Internal Pressure Loads," of the revised SAR.</p>	<p>See submittals. (211,212) Q and R pertain to SAR Section 8.1, "Normal and Off-Normal Operations."</p>
<p><u>QUESTION 78 (pertains to SAR Section 8.1.2.1, "Transport"):</u> CP&L states that, "Malfunctioning of the auxiliary components (i.e., crane, transporter, ram, etc.) does not relate to the safe functioning of the DSC." The meaning of this statement is unclear if, for example, the malfunctioning of the crane included a drop</p>	<p><u>RESPONSE:</u> Malfunctioning equipment is that which fails to operate in the normal or usual manner. The DSC canister is designed to withstand a load greater than any malfunctioning equipment could apply. Since a single failure proof crane will be used, a postulated malfunction does not include a</p>	<p>See submittals. (211,212,214) Q and R pertain to SAR Section 8.1.2, "Off-Normal Operation Analysis."</p>

TABLE E.1. (contd)

Questions/Requests ^(a,b)	Responses ^(a,b)	Comments ^(a,b)
<p><u>QUESTION 78: (contd)</u></p> <p>of more than 5 feet, or if the ram were used in excess of 6200 pounds of thrust in the extraction mode, etc. CP&L is asked to define precisely what is meant by "malfunctioning."</p>	<p><u>RESPONSE: (contd)</u></p> <p>drop of over 8 feet. The ram capacity is limited to a 20,000 pounds thrust. All components are designed for this loading. Therefore, a ram malfunction does not include a loading exceeding 20,000 pounds. Consequently, no auxiliary component could apply any load greater than that for which the DSC assembly was designed. SAR Section 5.2.1.2 describes the ram operation in more detail.⁽²¹⁴⁾</p>	
<p><u>QUESTION 79 (pertains to SAR Section 8.1.2.1, "Transport"):</u> CP&L is referred to Question 3, in Section 3.1.2.2 of the NRC response to NUTECH.</p>	<p><u>RESPONSE:</u> See response to Question 36 as the questions are identical.</p>	<p>See submittals.^(211,212) Q and R pertain to SAR Section 8.1.2, "Off-Normal Operation Analysis."</p>
<p><u>QUESTION 80 (pertains to SAR Section 8.2.1, "Loss of Air Outlet Shielding-Recovery"):</u> In view of CP&L's statement in SAR Section 5.3.2, "Components/Equipment Spares," not to stock any spares it seems that the replacement of an outlet could pose a problem. How does CP&L intend to resolve this?</p>	<p><u>RESPONSE:</u> Two extra shielding blocks shall be maintained in case of tornado caused projectiles damaging the existing shielding blocks. SAR Section 5.3.2, "Components/Equipment Spares," will be revised to reflect this change. SAR Section 8.2.1, "Loss of Air Outlet Shielding," describes the replacement process if a shielding block is lost in a tornado.</p>	<p>See submittals.^(211,212) Q and R pertain to SAR Section 8.2, "Accident Analysis."</p>
<p><u>QUESTION 81 (pertains to SAR Section 8.2.2, "Tornado/Tornado Generated Missiles"):</u> CP&L states that the walls of the HSM are "anchored into the concrete," however, there are no design drawings and no analysis provided to substantiate the claim that "there is no possibility of overturning or sliding." CP&L should provide design and analysis for their site-specific application. See Questions 1 and 2 in Section 8.2.2.2 of the NRC response to NUTECH.</p>	<p><u>RESPONSE:</u> The HSM foundation drawing, RNT-162-C-1102 shows the steel reinforcing dowels that anchor the HSM walls into the foundation. Using the maximum tornado wind design loading on the HSM, the maximum uplift pressure along the outside wall was determined assuming a rigid body motion. This uplift pressure force of 1.56 k/ft was applied to all module wall/foundation joints. Using the maximum load factor from ACI 349-80 of 1.7, the minimum area required to resist overturning or sliding was calculated to be 0.03 in.²/ft. Conservatively, two #6 reinforcing bars with an area of 0.44 in.² were used per each foot of wall. Thus, no possibility of overturning or sliding exists for the CP&L HSM.</p>	<p>See submittals.^(211,212) Q and R pertain to SAR Section 8.2, "Accident Analysis."</p>
<p><u>QUESTION 82 (pertains to SAR Section 8.2.3.1, "Accident Analysis"):</u> As previously discussed, the analysis results presented in this section should be backed up by appropriate analysis data. Please provide such data in sufficient detail so as to permit the NRC to review/substantiate the results as presented.</p>	<p><u>RESPONSE:</u> See the response to Question 76. As stated in response to Question 76, Section 8.0 of the SAR has been revised to reflect differences between NUTECH and HBR designs.⁽²¹⁴⁾</p>	<p>See submittals.^(211,212,214) Q and R pertain to SAR Section 8.2.3, "Earthquake."</p>

TABLE E.1. (contd)

Questions/Requests (a,b)	Responses (a,b)	Comments (a,b)
<p><u>QUESTION 83</u> (pertains to SAR Section 8.2.3.1, "Accident Analysis"): CP&L states that the HSMs are anchored to the concrete foundation. CP&L should either provide the analysis and designs of components to prove that no overturning or sliding occurs or else delete this discussion.</p>	<p>RESPONSE: As detailed in the response to Question 81, dowels are provided to prevent overturning or sliding due to tornado winds or projectiles. Previous calculations showed that the maximum uplift or sliding force is resisted by dowels extending from the foundation through the HSM walls.</p>	<p>See submittals. (211,212) Q and R pertain to SAR Section 8.2.3, "Earthquake."</p>
<p><u>QUESTION 84</u> (pertains to SAR Section 8.2.4, "Drop Accident"): CP&L is encouraged to examine the NRC response to NUTECH. Specifically, CP&L should consider Questions 3 and 4 in Section 8.2.5.2 and 1 and 2 in Section 3.2.5.2.</p>	<p>RESPONSE: The response consists of the four parts,</p> <p>a) <u>8.2.5.2 Accident Analysis, Question 3</u> The limiting drop decelerations for the horizontal and vertical orientations are 54.4g and 76.8g, respectively. These decelerations are based on an 8-foot drop of the IF-300 cask. As shown in Figure 1 of the response, the maximum height the cask center of gravity drops during slap down is approximately 4.75 feet. Therefore, the resulting decelerations will be less than the 8-foot values and do not need to be considered in the analysis.</p> <p>b) <u>8.2.5.2 Accident Analysis, Question 4</u> The ISFSI Cask Liner Assembly is shown in drawing RNT162-M-2703. As called out in the drawing, the material for the liner is Type 304 stainless steel.</p> <p>c) <u>3.2.5.2 Dry Shielded Canister, Question 1</u> The structural design criteria has been changed to state: "Following a drop accident, the DSC must be disassembled and fuel assemblies removed and inspected for damage."</p> <p>d) <u>3.2.5.2 Dry Shielded Canister, Question 2</u> For the horizontal drop accident only two components in the DSC basket, the spacer disk and the BORAL tube, could deform and cause interference with the fuel retrieval operation. To verify that no interference will occur, the elastic deformation of the spacer disk and BORAL tubes will be added to the maximum plastic deformation and then compared to the minimum gap between the fuel assembly and the inside of the BORAL tube. The maximum elastic deformation of the spacer disk is 0.0301 inch at the fuel cell centerline. The maximum BORAL</p>	<p>See submittals. (211,212,214) Q and R pertain to SAR Section 8.2, "Accident Analysis."</p>

TABLE E.1. (contd)

Questions/Requests (a,b)	Responses (a,b)	Comments (a,b)
QUESTION 84: (contd)	<p data-bbox="945 272 1150 297"><u>RESPONSE: (contd)</u></p> <p data-bbox="991 321 1522 630">tube deflection is 0.0072 inch. Since the stress in the spacer disk has exceeded yield for the drop accident, additional plastic deformation will occur. An upper limit plastic analysis was performed assuming a three hinged, uniformly loaded beam. The maximum deflection calculated was conservatively enveloped by this value. Since the BORAL tube stress is far less than yield stress, no permanent deformation occurs. Summing the total calculated deflections yields 0.0533 inch. The minimum gap is 0.324 inch. Consequently, no interference for the horizontal drop accident is present.</p> <p data-bbox="991 654 1522 914">For the vertical orientation drop, the deflection in the lead casing is the only situation which could cause interference with the fuel. The maximum elastic deflection for the 5-foot drop case is 0.32 inch at the lead casing centerline. For a built-in beam subject to a central load (similar to the resultant of the lead plug reaction), Hodge ("Plastic Analysis of Structure," 1959) shows that the maximum plastic displacement is only 11% greater than the maximum elastic displacement.</p> <p data-bbox="991 938 1522 1109">Conservatively, the vertical drop deflection will be increased by 25% and compared to the minimum gap of 0.5 inch between the fuel and the lead casing. The maximum deflection is 0.40 inch. Clearly no interference will occur which could inhibit the easy retrieval of all fuel assemblies.</p> <p data-bbox="945 1133 1522 1201">SAR Section 8.2.4 has been revised to present a more detailed description of the drop analysis. (214)</p>	
QUESTION 85 (pertains to SAR Section 8.2.5.2, "Analysis of Effects and Consequences"): Even though CP&L has provided reference for the lightning protection system, they have not provided any conceptual drawings, engineering drawings, or any indication of materials or dimensions of the system. Please provide.	RESPONSE: The lightning protection system drawing for the 3x1 HSM array has been provided. See Drawing RNT162-C-1116 for details of the system.	See submittals. (211,212) Q and R pertain to SAR Section 8.2.5, "Lightning."

TABLE E.1. (contd)

Questions/Requests (a,b)	Responses (a,b)	Comments (a,b)
<p><u>QUESTION 86</u> (pertains to SAR Section 8.3, "Foundation Design"): A general description of the foundation design is presented in this section. However, the data presented in this section should be supplemented by detailed design drawings, analysis details, finite element model (FEM) descriptions, and any other data that is required to support a review of this design.</p> <p>In addition to the above, CP&L should specifically provide:</p> <ul style="list-style-type: none"> • The drawings used to model the FEM analysis. • The input data for the FEM analysis. • The output from the FEM analysis for all load combinations. • The multiplication factors for the load combinations (See ACI-349-80, ANSI/ANS 57.7-1981). • Comparison table of calculated stresses with allowable stresses for critical sections for both steel and concrete. • Development and specification of elastic coil spring model plus size factors for foundation and references. • Reference the "presumptive allowable soil bearing capacity of 3000 psf." 	<p><u>RESPONSE:</u> Drawing RNT162-C-1102 has been provided to show the final foundation configuration. Section 8.3, "Foundation Design," of the H. B. Robinson SAR will be revised as follows to provide additional analysis and description of the foundation design.</p> <p><u>8.3 Foundation Design</u></p> <p>To provide a means of transmitting the reaction loads of the ISFSI modules to the ground, a rectangular, flat plate type, mat foundation was selected. The mat foundation is ideally suited for the ISFSI since it spreads out the loadings and, consequently, reduces the soil bearing pressure while at the same time minimizing the differential settlements.</p> <p>To accommodate the ISFSI modules, the front cask unloading area, and the hydraulic ram area behind the modules, an overall foundation size of 28 ft - 9 in. by 60 ft - 0 in. was selected. The HSM Foundation Slab will be 3-feet thick. A construction joint will connect the Foundation Slab to the Cask Unloading Slab which will be 2-feet thick starting from a point 5 feet from the module front. The Ram Mounting Slab at the rear of the modulus is to be 8-inches thick and will connect to the 3-foot foundation by an expansion joint. All foundation concrete will be 4000 psi normal weight concrete with the HSM foundation poured on a 4-inch mud slab. The HSM Foundation and the Cask Unloading Slab are to be interlaced with continuous two-way reinforcing top and bottom. Number 9 bars will be used for tensile reinforcement and as dowels to anchor the HSM walls to the foundation. The Ram Mounting Slab will have a number 5 bar continuous two-way reinforcing on the bottom only. Welded wire fabric will be placed on the top of the 8-inch slab.</p> <p>For analysis purposes, a STARDYNE rectangular plate finite element model was constructed. The model consists of 255 nodes and 224 plate elements. At each node, a ground support spring was added to</p>	<p>See submittals. (211,212) Q and R pertain to SAR Section 8.0, "Analysis of Design Events."</p>

TABLE E.1. (contd)

Questions/Requests (a,b)	Responses (a,b)	Comments (a,b)
QUESTION 86: (contd)	<p data-bbox="934 267 1144 292"><u>RESPONSE: (contd)</u></p> <p data-bbox="934 316 1522 511">simulate the soil elastic properties. The elastic soil spring is obtained by modifying the experimental modulus of subgrade reaction and then multiplying by the tributary area associated with each node. The resulting values of the spring stiffnesses were input to the finite element model as a ground stiffness matrix. The method for finding the stiffness K is shown below:</p> $K = K_v \frac{B + 1^2}{28} A$ <p data-bbox="934 625 1459 657">(Reference: Teng, "Foundation Design," 1962.)</p> <p data-bbox="934 673 1470 795">Where: K_v = experimental modulus of subgrade reaction = 100 lb/in.³ (Reference: Law Eng. Test. Company) B = foundation width = 28.75 feet A = nodal tributary area (varies)</p> <p data-bbox="934 820 1470 868">Five separate load cases were considered in the foundation design:</p> <ol data-bbox="934 885 1501 1015" style="list-style-type: none"> 1) Center module loading 2) Outside module loading 3) Dead Weight + Live Load 4) Dead Weight + Tornado Wind/Impact (lengthwise) 5) Dead Weight + Tornado Wind/Impact (widthwise) <p data-bbox="934 1031 1512 1242">Since Cask Unloading and Ram Mounting slabs are to be cast in place after HSM construction, the differential settlement due to HSM dead weight will not be experienced by the Cask Unloading Slab. Consequently, for load cases 1) and 2) the dead weight was not included. For load cases 1) and 2) the appropriate trailer loads are applied as concentrated loads at nodal locations in the unloading areas. Since only one loading or unloading</p>	

TABLE E.1. (contd)

Questions/Requests (a,b)	Responses (a,b)	Comments (a,b)
<u>QUESTION 86: (contd)</u>	<u>RESPONSE: (contd)</u>	
	<p>operation will occur at a time, the two load cases were evaluated independently. Load case 3) consists of the dead weight of the three modules containing the DSCs. Additionally, a live load of 200 psf is postulated for the HSM room. These loads are applied as pressure loads on the appropriate plate elements of the STARDYNE model. Load cases 4) and 5) are the maximum uplift load combinations caused by tornado loadings in the two horizontal directions. Using a conservative wind pressure of 400 psf applied on the module walls and roof, plus the reaction loads caused by a 3967 lb automobile traveling at 184.8 ft/s applied to the top of the module in the same direction combined with the module dead weight, the maximum uplift forces were calculated. Live loads have been excluded since they would cancel out uplift loads. Comparing the tornado loads with the HSM seismic loadings shows that tornado loads are much more severe. Consequently, seismic loads will not be included in the foundation analysis. Once the uplift forces have been calculated they are applied as negative pressures on the appropriate plate elements corresponding to the HSM/foundation connection surface. Tornado wind and impact loads are evaluated for both directions.</p> <p>Since an uplift force is created by the tornado loads, the foundation itself will have a negative bearing or uplift along the edge of the module. The soil itself does not resist uplift. Results from load cases 4) and 5) were reviewed for the effects of the uplift along the foundation edge. Minimal uplift was experienced in low stressed areas. Therefore, results from load cases 4) and 5) are not significantly effected by the uplift since the high stressed areas are not in that vicinity.</p>	

TABLE E.1. (contd)

Questions/Requests (a,b)	Responses (a,b)	Comments (a,b)												
<u>QUESTION 86: (contd)</u>	<u>RESPONSE: (contd)</u> The maximum calculated bearing stresses are listed below for the five load cases: <table><tr><th><u>Load Case</u></th><th><u>Maximum Bearing Pressure (psf)</u></th></tr><tr><td>1)</td><td>247</td></tr><tr><td>2)</td><td>463</td></tr><tr><td>3)</td><td>1605</td></tr><tr><td>4)</td><td>2210</td></tr><tr><td>5)</td><td>671</td></tr></table> <p>For sandy soils present at the bearing level of the mat foundation, allowable soil bearing pressures in the range of 3000 to 4000 pounds per square foot are recommended by the Southern Standard Building Code per the geotechnical exploration performed at the H. B. Robinson site by Law Engineering Testing Company. Since the maximum soil contact pressure produced by the HSM Foundation analysis is 2210 psf the bearing strength is sufficient. For normal dead weight and live loads the bearing pressure is only 1605 psf.</p> <p>The reinforcement design was based on the element bending moment results from the finite element analysis. Using the ultimate design method, the reinforcement was designed to withstand all postulated load combination bending moments with a conservative load factor of 1.7 applied to envelop all load combination factors in Section 9.2 of ACI 349-80. A tabulation of the results for the HSM Foundation and the Cask Unloading Slab is presented below.</p>	<u>Load Case</u>	<u>Maximum Bearing Pressure (psf)</u>	1)	247	2)	463	3)	1605	4)	2210	5)	671	
<u>Load Case</u>	<u>Maximum Bearing Pressure (psf)</u>													
1)	247													
2)	463													
3)	1605													
4)	2210													
5)	671													

TABLE E.1. (contd)

Questions/Requests (a,b)	Responses (a,b)			Comments (a,b)
QUESTION 86: (contd)	RESPONSE: (contd)			
	Load Case	Slab Thickness	Calculated M (K.in./in.)	M _u (K.in./in.)
	1)	2'-0"	13.78	87
	1)	3'-0"	28.87	186
	2)	2'-0"	13.07	87
	2)	3'-0"	24.55	186
	3)	2'-0"	N/A	87
	3)	3'-0"	46.39	186
	4)	2'-0"	N/A	87
	4)	3'-0"	80.12	186
	5)	2'-0"	57.58	87
	5)	3'-0"	155.75	186

The moment capacity of the sections are calculated per methods identical to the NUTECH topical equation 8-1-32. The 3'-0" slab with number 9 bars at 9 inches yields an ultimate strength of 186 k.inch per inch section. The 2'-0" slab with number 9 bars at 12 inches yields an ultimate strength of 87 k.inch per inch section. Therefore, all bending moments experienced by the foundation are below ultimate capacity.

The maximum uplift pressure exerted by the module wall is 3.1 psi. For a 1'-0" section of module the resulting uplift is 1.56 k/ft section. The dowel area required can be calculated by:

$$A = \frac{(UPLIFT)(1.7)}{(0)(f_y)}$$

Where: UPLIFT = 1.56 k
 0 = 0.9 = Factor for Tension
 f_y = 60 ksi

Therefore, Area A = 0.05 in². Conservatively two number 6 bars with an area of 0.88 in.² will be used every 12 inches to prevent uplift. Keyways will be used between the module foundation interface to prevent sliding. Assuming the maximum horizontal tornado loads are shared by the walls

TABLE E.1. (contd)

Questions/Requests (a,b)	Responses (a,b)	Comments (a,b)
QUESTION 86: (contd)	<p data-bbox="947 282 1150 305">RESPONSE: (contd)</p> <p data-bbox="947 331 1520 422">perpendicular to the load yields a maximum shear force of 24 k/ft, including the load factor of 1.7. The nominal shear strength of the keyway and dowels can be found from:</p> $V_n = (V_s + V_c) \phi$ <p data-bbox="947 496 1520 734">Where: $V_c = 2\sqrt{f_c'} bwd$ V_c = concrete shear strength (k) $f_c' = 4000$ psi $bw = 9$ inches $d = 12$ inches $V_s = (A_v)(f_y) =$ steel reinforcement shear strength (k) $A_v = 0.88$ in.² $f_y = 60$ ksi $\phi = 0.85$ = shear factor</p> <p data-bbox="947 756 1520 828">Consequently, $V_n = 56.49^k$ which exceeds the maximum factored shear force of 24 k. Thus, the module will not slide or overturn.</p>	<p data-bbox="947 854 1520 1209">The 8-inch Ram Mounting Slab was designed by hand calculation suggested by Teng ("Foundation Design," 1962) and Bowles, ("Foundation Analysis and Design," 1977). By applying the maximum factored spider leg loadings from the hydraulic Ram a simple span is approximated by treating the soil as a uniform load and the spider leg as reaction points. A maximum factored moment of 32.3 k.in./ft is calculated. Using the ultimate strength method with number 5 bars at 12 inches yields an ultimate strength of 64.2 k.in./ft for the 8-inch slab. Welded wire fabric was placed in the top of the slab for shrinkage and temperature reinforcement. Additionally, all punching shear from the ram supports was shown to be negligible.</p>

TABLE E.1. (contd)

Questions/Requests (a,b)	Responses (a,b)	Comments (a,b)
<p><u>QUESTION 86: (contd)</u></p>	<p><u>RESPONSE: (contd)</u></p> <p>Furthermore, the Cask Unloading Slab was analyzed for bearing and punching shear due to the hydraulic cylinder. A maximum bearing stress of 2.34 ksi was calculated, which is less than the allowable of 4.76 ksi calculated from ACI 349-80 Section 10.16. The maximum punching shear was also found to be under code allowables.</p>	
<p><u>QUESTION 87 (pertains to SAR Section 10.0, "Operating Controls and Limits"):</u> The NUTECH Topical Report identifies several limiting conditions for operation (see Section 10.3.2 of the Topical Report) along with the rationale of their inclusion. The CP&L submittal identifies fewer controls and limits. Please justify this difference.</p>	<p><u>RESPONSE:</u> The development of the H. B. Robinson ISFSI SAR was based on evaluation of the criteria contained in Sections 10.3.1 and 10.3.2 of the Topical Report. Results of this evaluation by CP&L indicate that the following controls and limits are required for the safe operation of the H. B. Robinson ISFSI.</p> <p>The following limits and controls were presented in Section 10.0, "Operating Controls and Limits," of the H. B. Robinson ISFSI SAR:</p> <ul style="list-style-type: none"> 10.1 Fuel Specifications 10.2 Limits for the Surface Dose Rate of the HSM While the OSC is in Storage 10.3 Limits for the Maximum Air Temperature Rise After Storage 10.4 Surveillance of the HSM Air Inlets <p>The remaining limiting conditions presented in the Topical Report have been incorporated at H. B. Robinson as follows:</p> <p><u>DSC Vacuum Pressure During Drying</u> - This requirement will be included in the operating procedures for the handling and loading of spent fuel into the canisters.</p> <p><u>OSC Helium Backfill Pressure</u> - This requirement will be included in the operating procedures for the handling and loading of spent fuel into the canisters.</p>	<p>See submittals. (211,212)</p>

TABLE E.1. (contd)

Questions/Requests (a,b)	Responses (a,b)	Comments (a,b)
QUESTION 87: (contd)	<p><u>RESPONSE: (contd)</u></p> <p><u>DSC Helium Leakage Rate of Primary Weld</u> - This requirement will be included in the construction specification for welding the end closure assembly on the canister during the loading and transfer of spent fuel.</p> <p><u>DSC Dye Penetrant Test of Secondary Weld</u> - This requirement will be included in the construction specifications for welding the end closure assembly on the canister during the loading and transfer of spent fuel.</p> <p><u>Dose Rate at End of DSC Lead Shield Plug</u> - This is specified as a design condition for the selection of the lead shield plug and is the basis for the assembly shown in RNT 162-M-2600.</p> <p><u>Alignment of Cask and RSM for the DSC Transfer Operation</u> - This is provided in a specification to supply equipment, procedures, and services for the alignment activities.</p>	
<p>QUESTION 88 (pertains to SAR Section 10.1, "Fuel Specifications"): The specifications on burnup and enrichment may violate criticality limits without use of BORAL sheets. Please clarify.</p>	<p><u>RESPONSE:</u> BORAL tubes, not sheets, will be specified for use in the H. B. Robinson DSCs. Therefore, no criticality limits will be violated.</p>	<p>See submittals (211,212) Q and R pertain to SAR Section 10.0, "Operating Controls and Limits."</p>
<p>QUESTION 89 (pertains to SAR Section 10.1, "Fuel Specification"): Why is a minimum cooling period not shown?</p>	<p><u>RESPONSE:</u> The minimum cooling period is five years and will be shown in the next revision to the SAR.</p> <p>Heat output and burnup are the most important criteria for fuel specification. However, the minimum cooling period is five years and will be shown in the SAR.</p>	<p>See submittals (211,212) Q and R pertain to SAR Section 10.0, "Operating Controls and Limits."</p>
<p>QUESTION 90 (pertains to SAR Section 10.1, "Fuel Specifications"): Refer to question under Section 10.1 of the NRC response to NUTECH.</p>	<p><u>RESPONSE:</u> The axial peaking power factor of 1.08 was conservatively applied to the uniform heat generation rate in order to determine the worst</p>	<p>See submittals (211,212) Q and R pertain to SAR Section 10.0, "Operating Controls and Limits."</p>

TABLE E.1. (contd)

Questions/Requests ^(a,b)	Responses ^(a,b)	Comments ^(a,b)
<p>QUESTION 90: (contd)</p>	<p>RESPONSE: (contd)</p> <p>case temperature distribution in a two dimensional cross section of the DSC and HSM. Even with this conservatism, the maximum fuel temperature was calculated to be 338°C under the worst ambient temperature condition of 52°C. This maximum fuel temperature is 62°C less than the accepted limiting fuel rod temperature of 400°C in the GNS Topical SAR. Thus, the ΔT from ambient to the DSC internals could increase by 22% before the maximum fuel temperature would be reached. If the relationship were linear (i.e., $Q = HA\Delta T$, where Q = heat generation, A = area, H = over heat transfer coefficient, and ΔT = temperature difference), then a 22% increase in Q (resulting in a peaking factor of 1.32) would be required before the temperature limit would be violated. However, since both conduction ($Q = f(\Delta T^{1.33})$) and radiation ($Q = f(\Delta T^4)$) are significant, a much larger increase in Q can be tolerated before any cladding temperature limits are reached.</p> <p>Therefore, the axial peaking factor was not a limiting parameter but a conservatism in the analysis. The only thermal parameter for fuel is the heat output per fuel assembly and, as shown in SAR Section 10.1 ("Fuel Specifications"), the decay power per assembly must be less than or equal to 1 kW.</p>	
<p>QUESTION 91 (pertains to SAR Section 10.1, "Fuel Specifications"): CP&L should state what the fuel weight and distance between spacers is limited to. These details were specified by NUTECH, and are, in fact, a basis of their design. If CP&L intends to deviate from the NUTECH design envelope, it should provide the necessary analysis.</p>	<p>RESPONSE: Fuel for the H. B. Robinson ISFSI will be similar to the Westinghouse 15x15/204 (e.g., EXXOM 15x15/204). The Westinghouse 15x15/204 was used as the enveloping case in the NUTECH Topical Report. Therefore, the fuel assembly weight of 650 kg and the maximum length between spacers of 26.19 inches are the limiting design parameters for the CP&L design. SAR Section 10.1, "Fuel Specifications," will be revised to show this information.</p>	<p>See submittals (211,212) Q and R pertain to SAR Section 10.0, "Operating Controls and Limits."</p>

TABLE E.1. (contd)

Questions/Requests (a,b)	Responses (a,b)	Comments (a,b)
<p><u>QUESTION 92 (pertains to SAR Section 10.3.5, "Maximum Air Temperature Rise After Storage"):</u> See Question 1 in Section 10.3.2.7 of the NRC response to NUTECH.</p>	<p><u>RESPONSE:</u> The addition of fans will be eliminated as a corrective action for a temperature rise greater than 100°F. If the temperature still exceeds the 100°F limit after all blockage is checked for and removed, the OSC will be removed from the HSM.</p>	<p>See submittals. (211,212) Q and R pertain to SAR Section 10.3, "Limits for Maximum Air Temperature Rise After Storage."</p>
<p><u>QUESTION 93 (pertains to SAR Section 10.4, "Surveillance of the HSM Air Inlets"):</u> See Questions 1 and 2 in Section 10.3.3.1 of the NRC response to NUTECH.</p>	<p><u>RESPONSE:</u> Section 10.4, "Surveillance of the HSM Air Inlets," will be revised to read, "Surveillance of the HSM Air Inlets and Outlets." Appropriate portions of the SAR will be revised to include the surveillance of the air outlets along with the inlets. There will be no need to inspect internal pathways since the bird screens on the inlets and outlets will stop debris which may significantly alter air flow through the HSM. Internally, there will be no loose or moving parts which could fail and block the air flow. Also, nonvisible portions of the air passages will not be conducive to organic growth. Therefore, inspection of the air inlets and outlets will assure there is no blockage of air passages in the HSM.</p> <p>Surveillance will not include measurement of air flow or temperature rise of cooling air. Analysis in Chapter 8 of the NUTECH Topical demonstrated that no temperature limits will be exceeded even if the air inlet and outlet passages of the HSM are completely blocked for 48 hours. Complete blockage of the inlets and outlets, however, is not expected except for in an accident situation. As stated in SAR Section 10.4, the air inlets and outlets will be inspected within 24 hours after an accident or abnormal event. This will assure that if a module is completely buried, flow can be restored within another 24 hours.</p> <p>For normal operations, inspection will be performed once every seven days. Since complete blockage is not postulated as a normal event, this inspection interval assures a maximum of two open air outlets. Since Chapter 8 of the Topical showed that partial</p>	<p>See submittals. (211,212) Q and R pertain to SAR Section 10.0, "Operating Controls and Limits."</p>

TABLE E.1. (contd)

Questions/Requests (a,b)	Responses (a,b)	Comments (a,b)
<u>QUESTION 93: (contd)</u>	<u>RESPONSE: (contd)</u> blockage will not result in unacceptable temperatures, two open outlets will be adequate for seven days. At that time, local obstructions can be removed.	
<u>QUESTION 94 (pertains to SAR Section 10.4, "Surveillance of the HSM Air Inlets"):</u> CP&L is requested to answer the Questions 1, 2, and 3 of Section 10.2.3, 10.3.2.1, 10.3.2.3 (1 and 2), and 10.3.2.7 (1 and 2) of the NRC response to the NUTECH Topical Report.	<u>RESPONSE:</u> The response consists of the eight following parts, a) <u>10.2.3 Surveillance Requirements, Question 1</u> Visual inspection of the internal air passages will be conducted from the outside of the HSM. Internal regions not visible from outside will be located in an area which is nonconductive to organic growth due to radiation and high temperatures. Additionally, the air inlets and outlets are covered by bird screens which prevent penetration inside the HSM. b) <u>10.2.3 Surveillance Requirements, Question 2</u> Water build-up will not occur inside the HSM. See the response to Question 26 for details explaining the drainage system. A visual inspection of the drain pipes will be performed to assure they are not clogged. See Drawing RNT162-C-01101 for the location of the drain pipes. c) <u>10.2.3 Surveillance Requirement, Question 3</u> As stated in the response to Question 93, surveillance does not include measurement of air flow or temperature rise of cooling air. As long as the air flow is not totally blocked for more than 48 hours, no temperature limits will be exceeded. For normal operations, seven day surveillance is sufficient since complete	See submittals. (211,212) Q and R pertain to SAR Section 10.0, "Operating Controls and Limits."

TABLE E.1. (contd)

Questions/Requests (a,b)	Responses (a,b)	Comments (a,b)
<u>QUESTION 94: (contd)</u>	<u>RESPONSE: (contd)</u>	
	<p>blockage is not feasible for any normal event. See response to Question 93 for additional details.</p>	
	d) <u>10.3.2.1 DSC Vacuum Pressure During Drying</u>	
	<p>Based on engineering judgment, one hour at the specified pressure should be sufficient.</p>	
	<p>A water-logged fuel assembly has not been considered. Prior to inserting the fuel assemblies into the DSC, a fuel assembly will be visually examined for its structural and mechanical integrity. Any results of non-destructive examination which the fuel assembly may have previously undergone will also be reviewed to insure that fuel assemblies do not contain any nonvisible defects.</p>	
	<p>Additionally, at the temperature and pressure to which the fuel rods will be exposed during the drying operations, any water in the fuel rods should evaporate and be suctioned from the DSC cavity.</p>	
	e) <u>10.3.2.3 DSC Helium Leakage Rate of Primary Weld, Question 1</u>	
	<p>The value of 631 cm³ of He corresponds to 2 atmosphere of pressure. The 473 cm³ corresponds to 1.5 atmosphere. This value was revised in the NUTECH Topical Report. The change has no impact on the leak rate specification of the primary weld.</p>	
	f) <u>10.3.2.3 DSC Helium Leakage Rate of Primary Weld, Question 2</u>	
	<p>See response to Question 50, Subquestion 10.3.2.3 (Question 2) as both are identical.</p>	

TABLE E.1. (contd)

Questions/Requests ^(a,b)	Responses ^(a,b)	Comments ^(a,b)
QUESTION 94: (contd)	<p>RESPONSE: (contd)</p> <p>g) <u>10.3.2.7 Maximum Air Exit Temperature, Question 1</u></p> <p>See answer to Question 92 of this document since the questions are identical.</p> <p>h) <u>10.3.2.7 Maximum Air Exit Temperature, Question 2</u></p> <p>There are no moving components within the HSM that could significantly alter the air flow or cause an increase in temperature. The results of the initial measurements will provide adequate assurance that more frequent measurements are not necessary. Surveillance of inlets and outlets as mentioned in the response to Question 93 will assure proper operation.</p>	
REQUEST (May 28, 1985) ⁽²²¹⁾ : The NRC letter of May 28, 1985, requested revised security documentation.	RESPONSE ⁽²²¹⁾ : CP&L submitted revised security documentation in response to NRC's request.	See submittal. ⁽²²¹⁾ Q and R pertain to LA Section 9.0, "Physical Protection."
QUESTION (September 20, 1985) ⁽¹⁰⁾ : After reviewing your Environmental Report for the Independent Spent Fuel Storage Installation (ISFSI) at the H. B. Robinson Steam Electric Plant and conducting a site visit, we need some additional information to complete our environmental review.	<p>RESPONSE⁽²¹⁷⁾: The dose burden to workers constructing the five additional modules at the H. B. Robinson ISFSI was calculated based on the following assumptions:</p> <ul style="list-style-type: none"> • The distance from the three initial modules will vary from contact to approximately 50 feet. • The dose rate will vary from 8.2 mrem/h at contact to 2.73×10^{-3} mrem/h at 50 feet. • For conservatism, the dose rate at 25 feet was assumed. • A construction crew of ten people, on the average, will work one 8-hour shift per day for thirty days. <p>Based on these assumptions, the dose would be 10.8 mrem/person, as shown in Table 1 below.</p>	See submittal. ⁽²¹⁷⁾ Q and R pertain to the ER.
<p>The information we require pertains to the onsite collective dose commitment to workers constructing the five additional modules adjacent to the original three filled modules and to personnel gathering data during the first year demonstration period. Please provide this information in the same format as Tables 7.4-1 and 7.4-2 of your ISFSI Safety Analysis Report. Also, as discussed with John McLean of your staff during our site visit on August 29, 1985, a detailed ISFSI site plan drawing and aerial photographs would be most helpful to our review.</p>		

TABLE E.1. (contd)

Questions/Requests ^(a,b)	Responses ^(a,b)	Comments ^(a,b)
<u>QUESTION (September 20, 1985): (contd)</u>	<u>RESPONSE: (contd)</u>	
	<u>TABLE 1. Estimated Onsite Doses During Construction of Additional Modules</u>	
	Operation: Construct 5 additional modules Location: Area near modules Number of Personnel: 10 Time (h/yr): 240 Average Distance from Facility (ft): 25 Dose Rate (mrem/hr): 0.045 Dose Per Person (mrem/yr): 10.8 Total Personnel Dose (mrem/yr): 108	
<u>QUESTION (November 20, 1985)⁽²¹⁹⁾</u> : At the meeting on November 18, 1985, between the NRC staff and CP&L, the NRC determined that some drawings referenced in CP&L's responses need to be supplied to reviewers to resolve questions.	<u>RESPONSE⁽²¹⁸⁾</u> : CP&L submitted the additional sets of drawings requested by the NRC in the November 18, 1985, meeting. Drawings showing the modifications to the shipping cask were to be submitted at a later date.	See submittal. ⁽²¹⁸⁾
<u>QUESTION (March 19, 1986)⁽¹³⁷⁾</u> : What lubricant is used on the DSC?	<u>RESPONSE⁽¹³⁷⁾</u> : The lubricant used on the DSC is EVERLUBE 6122.	See telecopied submittal. ⁽¹³⁷⁾ Also, see response to QUESTION (April 16, 1986).
<u>QUESTION (April 2, 1986)⁽¹³⁷⁾</u> : What initial fuel enrichment and fuel exposure are assumed in the ORIGEN-2 analysis in Section 3.0, "Principal Design Criteria," of the SAR?	<u>RESPONSE⁽¹³⁷⁾</u> : The ORIGEN-2 analysis referenced in SAR Section 3.0, "Principal Design Criteria," is based on an initial fuel enrichment of 3.5 wt% uranium-235 and a fuel exposure of 35,000 MWD/MTU.	See telecopied submittal. ⁽¹³⁷⁾
<u>REQUEST (April 11, 1986)⁽¹⁴²⁾</u> : The NRC reviewer requested a description of the NUTEC analysis of the thermal load on the concrete foundation.	<u>RESPONSE⁽¹⁴²⁾</u> : NUTEC submitted this description of analysis of the thermal load on the HSM foundation: <u>Thermal Load on HSM Foundation</u> The thermal load on the concrete foundation is very small. This load was not explicitly combined into the load combination in the CP&L NUHOMS SAR because a load factor of 1.7 was conservatively used for all loading conditions. The moment was on the conservative side. However, a more detailed analysis was performed to evaluate the effect of the thermal load on the concrete foundation. The following assumptions were used in this analysis:	See submittal. ⁽¹⁴²⁾

TABLE E.1. (contd)

Questions/Requests (a,b)	Responses (a,b)	Comments (a,b)
<p><u>REQUEST (April 11, 1986): (contd)</u></p>	<p><u>RESPONSE: (contd)</u></p> <ol style="list-style-type: none"> 1. The foundation is restrained from bending deformation by the HSM walls. 2. The effective moment of inertia of the concrete section is 1.5 times the cracked section. 3. The thermal gradients through the thickness of the foundation are given by the HEATING6 analyses. <p>For the 3 ft-0 in. thick foundation, the moment due to the thermal gradient is $0.7178 \times \Delta T$ in.-kip/in. The moment due to normal operation is 13.38 in.-kip/in. and the moment due to the accident temperature is 22.76 in.-kip/in.</p> <p>The moment (conservatively) reported in the SAR was 155.75 in.-kip/in. Since the accident thermal load governs, the resultant moment for the load combination containing the thermal load is 178.51 in.-kip/in. which is less than the 3 ft-0 in. foundation capacity of 186 in.-kip/in. Therefore the HSM foundation is structurally adequate.</p>	
<p><u>QUESTION (April 16, 1986)⁽⁸⁾</u>: Outstanding issues remaining to be closed by submittals from CP&L include final quality assurance commitments and completion of CP&L's internal plant review to confirm operations of the ISFSI and Unit 2 have no significant safety impact on one another, that no changes to the Unit 2 operating license are needed for the ISFSI and that the ISFSI operates independently.</p>	<p><u>RESPONSE</u>: CP&L's letter of April 22, 1985, (222) provides additional information in support of the H. B. Robinson Independent Spent Fuel Storage Installation (ISFSI) license application. Specific items are listed below.</p> <ol style="list-style-type: none"> 1. A discussion of the impact of planned and existing chemical and waste oil storage facilities and further discussion of the railroad track usage is contained in Enclosure 1 in the form of a proposed revision to SAR Section 2.1.2.1. 2. A revised SAR Section 11 (Quality Assurance) is included in Enclosure 2. The contents of this revision have been discussed with the appropriate reviewer and reflect his comments. 	<p>See submittal (222) for Enclosures 1-4.</p>

TABLE E.1. (contd)

Questions/Requests (a,b)	Responses (a,b)	Comments (a,b)
QUESTION (April 16, 1986): (contd)	<p data-bbox="947 272 1150 293">RESPONSE: (contd)</p> <ol style="list-style-type: none"> <li data-bbox="961 321 1524 370">3. Enclosure 3 is a discussion of the lubricant, Everlube 6122, to be used in the ISFSI. <li data-bbox="961 394 1524 508">4. Enclosure 4 is a mark-up of SAR Tables 8.2-1 and 8.2-2 providing revised stress values for the 8-foot drop accident, deadweight, and drop loads. This information has been previously telecopied to you. <li data-bbox="961 532 1524 678">5. A question was raised in our March 19, 1986 meeting regarding the material for the Dry Storage Canister support rod. The material is SA 497 Type XM-19 (Nitronic 50). This information was provided to the reviewer in a telephone conversation. <p data-bbox="947 703 1524 841">The CP&L letter of May 21, 1986,⁽⁹⁾ states that the planned H. B. Robinson Plant, Unit No. 2 (HBR2) Independent Spent Fuel Storage Installation (ISFSI) has undergone internal review by Carolina Power & Light Company. The following conclusions resulted from this review:</p> <ol style="list-style-type: none"> <li data-bbox="961 865 1524 1060">1. The ISFSI can operate independently without affecting the safety and operation of HBR2 since there are no physical connections between the reactor unit and the ISFSI other than connections which serve no safety-related functions (power for ISFSI lighting and security equipment) and the ISFSI security alarm indications. <li data-bbox="961 1084 1524 1174">2. The ISFSI can recover from normal or off-normal incidents or accidents without affecting the safety and operation of the nuclear unit at HBR2. <li data-bbox="961 1198 1524 1247">3. The nuclear reactor unit at HBR2 does not affect the safety and operation of the ISFSI. <li data-bbox="961 1271 1524 1344">4. No changes to the HBR2 10 CFR Part 50 operating license are required as a result of the ISFSI. <p data-bbox="947 1369 1524 1412">In conclusion, the HBR2 ISFSI is "independent" as defined in 10 CFR Part 72.</p>	

TABLE E.1. (contd)

Questions/Requests (a,b)	Responses (a,b)	Comments (a,b)
<p><u>QUESTION (November 10, 1986)</u>(223,224): The NRC staff has examined CP&L's request for a license amendment submitted by letter dated October 31, 1986. We have concluded there is insufficient information in this submittal to support the proposed action. In particular, you have not adequately described or characterized the material proposed for use, and you have not provided sufficient data to enable us to confirm your criticality analysis.</p>	<p><u>RESPONSE</u>(225): Carolina Power & Light Company's letter dated October 31, 1986, requested a license amendment regarding Technical Specification 5.3 for the H. B. Robinson Independent Spent Fuel Storage Installation. During a meeting with our staff on December 2, 1986, certain materials were deemed necessary for completion of the staff review. Accordingly, this letter transmits:</p>	<p>See submittal(225) for the four items that were transmitted. Q and R pertain to the Materials License.</p>
<p>At our meeting on December 2, 1986, (224) to discuss these matters, CP&L indicated that material specifications, a drawing, a comparison to previous design with explanation of changes and further criticality analysis information will be supplied by mid-December 1986.</p>	<ol style="list-style-type: none"> 1. Drawing No. RNT162-M-2504 (a proprietary drawing). 2. Discussion of the KENO criticality analysis. 3. Materials data sheets for the poison material. 4. Summary comparison of proposed change versus original configuration. 	
<p><u>QUESTION (January 29, 1987)</u>(226): The NRC staff has examined CP&L's request for a license amendment submitted by letter dated October 31, 1986, supplemented by additional information submitted with your letter dated January 2, 1987. However, a reference made in your submittal appears to be mislabeled. This is Reference 3, Spacer Disk 30 Foot Drop Analysis Package, File No. CPL018-0508, Revision 0, which is referenced in a December 17, 1986 memorandum (BWF-86-007, an enclosure to your January 2, 1987, submittal). Accordingly, we request that you submit to us the proper analysis package, with correct spacer disk and tube support dimensions.</p>	<p><u>RESPONSE</u>(227): As requested by your letter of January 29, 1987, we have assembled the following information to support our October 31, 1986 request for amendment to the exiting Technical Specifications for Material License SNM-2502:</p>	<p>See submittal(227) for Attachments 1-3. Q and R pertain to the Materials License.</p>
<p>In addition, we request further information regarding the Foundries Montupet boron aluminum alloy proposed for guide sleeve use in your submittal. In particular, what is the variation in boron content normally found in a minimal 5-weight percent alloy? In your submittal of January 2, 1987, you have provided data on the measured boron content of the alloy material proposed for sleeve guide use. Are these representative of this material?</p>	<p><u>Attachment 1</u></p> <p>A specific structural analysis is attached based on the current design of the spacer disk. These calculations essentially replaced the ratioing presented in the original license documents based on the NUHOMS Topical Report. The attached document replaces Reference 3 in our October 31, 1986 submittal.</p> <p><u>Attachment 2</u></p> <p>A detailed technical description of aluminum-boron material has been obtained from the French manufacturer, including examples of other similar uses of this alloy for criticality control. Specific responses to the January 29, 1987, letter are attached along with the vendor's catalog information.</p>	

TABLE E.1. (contd)

Questions/Requests (a,b)	Responses (a,b)	Comments (a,b)
<u>QUESTION (January 29, 1987): (contd)</u>	<u>RESPONSE: (contd)</u>	
Also, provide the isotopic characterization of the boron in terms of B ¹⁰ present for this material. At this level of boron present in aluminum, to what degree is boron in solution uniformly throughout the material? Discuss this matter in terms of quality controls maintained for the production of material. Also, provide the material properties of the boron aluminum alloy.	<p data-bbox="947 326 1081 354"><u>Attachment 3</u></p> <p data-bbox="947 375 1522 440">A copy of the latest, approved revision of Drawing No. RNT162-M-2504 is attached (it is a proprietary drawing).</p> <p data-bbox="947 467 1522 634">As discussed with the members of your staff, we understand that the information provided with this letter would satisfy all of the technical questions regarding the new guide sleeve design. With this level of confidence, we have released the manufacturer to proceed with fabrication of sleeves and final assembly of the canisters.</p> <p data-bbox="947 662 1522 824">In an effort to provide an additional level of conservatism in the criticality calculations, we will commit to load spent fuel assemblies for the eight canisters (56 total assemblies) that do not exceed an initial enrichment of 3.2%. This is within the 3.5% limit shown in Section 2.2.1 of the Technical Specifications.</p> <p data-bbox="947 852 1522 943">We are anticipating delivery of the completed canisters in early April 1987, and will need to have an approved license amendment in place to support our receipt inspection.</p>	Q and R Pertain to SAR Section 8.4, "DSC Instrumentation Penetration Design."
<u>REQUEST (March 30, 1987)⁽²²⁸⁾</u> : Upon reviewing Amendment 1 to the Safety Analysis Report for the Independent Spent Fuel Storage Installation at H. B. Robinson, it was noticed that two pages, which were on the insert list, were missing from your submittal of March 4, 1987. The missing pages are 8.4-1 and 8.4-2. Please send us 20 copies of these missing pages so that our copies will be complete.		

TABLE E.1. (contd)

Questions/Requests ^(a,b)	Responses ^(a,b)	Comments ^(a,b)
<p>QUESTION (April 7, 1987)⁽²²⁹⁾: In examining your update of the H. B. Robinson Independent Spent Fuel Storage Installation (ISFSI) Safety Analysis Report submitted by letter dated March 4, 1987, some procedures proposed appear to differ from technical specifications of your Materials License SNM-2502 for spent fuel storage at the Robinson ISFSI. Specifically, in Section 5.1.1.3 on page 5.1-2 of the SAR with respect to item k), there appears to be no provision for a helium leak test for the final prefabricated plug weld. See Appendix A of SNM-2502, Specification 4.6.1. In this previous version of the SAR, item 1) provided for such a check. Perhaps this item was deleted by mistake.</p> <p>Similarly, in Section 5.1.1.4 on page 5.1-2 of the SAR, there appears to be no provision of dye penetrant testing of the seal weld for the cover plate. See again Appendix A of SNM-2502, Specification 4.6.1.</p>	<p>RESPONSE⁽²³⁰⁾: CP&L telephoned the NRC on April 13, 1987, in response to the NRC's question on procedural errors with respect to canister weld check tests. CP&L stated that actual procedures are correct and that CP&L adheres to the license technical specifications. The SAR will be corrected in the next update in August.</p>	<p>See submittal⁽²³⁰⁾ a and R pertain to SAR Section 5.1.1, "Narrative Description."</p>
<p>(a) Abbreviations: Independent Spent Fuel Storage Installation (ISFSI); License Application (LA); Safety Analysis Report (SAR); Environmental Report (ER); Question (Q); Response (R); Carolina Power and Light Company (CP&L); NUTECH Engineers, Inc. (NUTECH); Topical Safety Analysis Report (TSAR); Dry Shielded Canister (DSC); Horizontal Storage Module (HSM); NUTECH Horizontal Modular Storage (NUHOMS) systems; H. B. Robinson Steam Electric Plant (HBR); H. B. Robinson Steam Electric Plant, Unit 1 (HBR1) and Unit 2 (HBR2); General Electric Company (GE); Quality Assurance (QA); not applicable (N/A).</p> <p>(b) Questions 1 to 94 were in NRC's letter of July 12, 1985. Questions 2 to 94 are also in CP&L's submittal⁽²¹¹⁾ of October 30, 1985. The "Table of Contents" for LA, SAR, and ER are in References 135, 212, and 135 (see Chapter 12), respectively.</p> <p>(c) D. J. Murphy and J. A. Tice (Law Engineering Testing Company), Letter to D. M. Koss (Carolina Power and Light Company), "Report of Geothermal Exploration, Dry Storage Module Project, H. B. Robinson Steam Electric Plant, Hartsville, South Carolina, Law Engineering Testing Company (LETCO) Job No. J47284-0548," February 8, 1985.</p>		

APPENDIX F

CHRONOLOGICAL LISTING OF LICENSING ACTIVITIES FOR LARGE-SCALE
STORAGE OF CONSOLIDATED SPENT FUEL

APPENDIX F

CHRONOLOGICAL LISTING OF LICENSING ACTIVITIES FOR LARGE-SCALE STORAGE OF CONSOLIDATED SPENT FUEL

3.3.1(a) Maine Yankee

1979: On September 18, 1979, Maine Yankee Atomic Power Company submitted⁽¹⁵²⁾ to the NRC an application requesting an increase in the spent fuel storage pool capacity at Maine Yankee from 953 to 1545 fuel assemblies by using a rod consolidation technique. On October 16, 1979, the NRC notified⁽³⁷⁵⁾ Maine Yankee Atomic Power Company that it had requested publication of a notice in the Federal Register concerning this proposed amendment to the Maine Yankee Facility Operating License that involves expansion of the spent fuel pool capacity by a modified spent fuel pin concept rather than through an increase in storage locations. The utility submitted a letter to the NRC on October 18, 1979.⁽²¹⁾

1980: The utility submitted a letter to the NRC on September 29, 1980.⁽²¹⁾

1981: On October 5, 1981, the utility submitted to the NRC the complete report on the Maine Yankee fuel storage modification.⁽³⁷⁶⁾ The utility also submitted letters to the NRC on July 28, September 29, and October 26, 1981.⁽²¹⁾

1982: The NRC completed their review of the application and issued a Safety Evaluation and Environmental Impact Appraisal on June 16, 1982.⁽²⁰⁾ These documents indicated NRC's satisfactory resolution of the open items listed in Section 3.0 of the Safety Evaluation. These open items pertain to procedures and operating limits that must be implemented as part of the proposed change. The utility provided letters to the NRC on February 10, May 7, July 21, and September 7, 1982.⁽²¹⁾

1983: No pertinent items.

(a) Part numbers correspond to the subsections in Section 3.3.

1984: The state of Maine ordered⁽¹⁵³⁾ the utility to withdraw its application to the NRC for licensing of large-scale rod consolidation at Maine Yankee. This state would allow, and the NRC authorized,⁽²¹⁾ the plant to consolidate only up to 20 spent fuel assemblies.

1985: No pertinent items.

1986: On June 24, 1986, the NRC requested⁽³⁷⁷⁾ additional information concerning the spent fuel pool masonry wall. On September 4, 1986, the utility provided a response⁽³⁷⁸⁾ to NRC's June 24, 1986 request; the response included comments regarding consolidated fuel rod bundles.

3.3.2 Ginna

1984: On June 12, 1984, Rochester Gas and Electric Corporation provided responses⁽³⁷⁹⁾ to NRC's questions concerning an application⁽³⁸⁰⁾ requesting an increase in the storage capacity of the spent fuel pool storage racks and in the response to Question 11, the utility indicated it was not seeking approval for storage of consolidated fuel. However, in the responses to NRC questions that were sent⁽³⁸¹⁾ by the utility to the NRC on August 13, 1984, it was indicated in the introduction that analyses of the spent fuel storage racks were made for standard fuel assemblies and for canisters of consolidated fuel rods. On November 14, 1984, the NRC issued⁽³⁸²⁾ Amendment No. 65 to the facility operating license for Ginna to increase the spent fuel pool storage capacity; it is noted on page 9 of the associated Safety Evaluation that the utility had requested the NRC staff to evaluate the adequacy of the storage racks if at sometime in the future the utility decides to implement a rod consolidation program.

1985: On February 27, 1985, the utility submitted⁽¹⁵⁵⁾ an application for an amendment to Ginna's Facility Operating License to allow the storage of consolidated fuel. Personnel from the utility met with the NRC staff on June 4, 1985,⁽¹⁵⁴⁾ to discuss the licensability of storage of consolidated fuel on the Ginna site. On June 10, 1985, the utility provided a response⁽¹⁵⁶⁾ to a request from the meeting⁽¹⁵⁴⁾ to discuss the licensability of storage of consolidated fuel on the Ginna site. On June 10, 1985, the utility provided a response⁽¹⁵⁶⁾ to a request from the meeting⁽¹⁵⁴⁾; the request involved prior changes that had been submitted (and incorporated in

the application) but that had not yet been approved. The response stated that the requirement (i.e., of 3.11.5) insures that should a handling accident occur during the movement of a consolidated fuel canister, the dose rate at the exclusion area boundary would satisfy the requirements of 10CFR100. On June 12, 1984, the NRC indicated⁽³⁸³⁾ to the utility that from the NRC's preliminary evaluation of the application for an amendment, and as a result of the meeting on June 4, 1985, that the NRC staff had determined that the request has merit, and the concept is licensable assuming satisfactory resolution to any technical issues that may arise. On August 6, 1985, the NRC transmitted⁽³⁸⁴⁾ a summary of June 4, 1985, meeting⁽¹⁵⁴⁾ to the utility. On November 8, 1985, the NRC transmitted⁽³⁸⁵⁾ a notice that was to be published in the Federal Register; the notice indicates that the proposed amendment to Ginna's Technical Specifications would 1) increase the spent fuel pool (SFP) storage capacity by authorizing storage of consolidated fuel canisters in the existing SFP and imposing restrictions on the storage of the consolidated fuel canisters in the SFP, and 2) modify the current prohibition on the use of the Auxiliary Building crane over the SFP to authorize use of the crane to move consolidated fuel canisters. The NRC issued⁽³⁸⁶⁾ the Environmental Assessment on the storage of consolidated fuel on December 12, 1985; the NRC's finding was that there would be no significant impact. On December 16, 1985, the NRC issued⁽²⁴⁾ Amendment No. 12 to Ginna's Facility Operating License; the amendment changes the Technical Specifications to authorize storage of consolidated fuel canisters in the existing SFP and to authorize use of the Auxiliary Building crane to move consolidated fuel canisters. The letter⁽²⁴⁾ also included the Safety Evaluation. However, the letter⁽²⁴⁾ indicated that the utility's request to increase the capacity of the SFP will be handled by a separate licensing section.

1986: On December 2, 1986, the utility submitted⁽²⁵⁾ a letter to the NRC concerning the Safety Evaluation.⁽²⁴⁾ The utility requested that NRC review some inconsistencies between the Safety Evaluation and the utility's application and provide clarification of statements in the Safety Evaluation.

3.3.3 Millstone-2

1984: On March 30, 1984, the utility, Northeast Nuclear Energy Company (NNECO), submitted⁽³⁸⁷⁾ its spent fuel disposition plans. The two proposed license amendments that were discussed were reracking of the spent fuel pool and fuel consolidation. On June 4, 1984, the NRC transmitted⁽³⁸⁸⁾ to NNECO a summary of the meeting that was held on May 17, 1984. One of the subjects discussed at the meeting was a forthcoming license amendment request for fuel assembly consolidation.

1985: No pertinent items.

1986: On May 21, 1986, NNECO submitted to the NRC a request for an amendment to its operating license for Millstone-2 to allow storage of consolidated fuel.⁽¹⁵⁸⁾ The NRC letter of July 25, 1986,⁽³⁸⁹⁾ to NNECO included seven questions. The NRC letter of August 21, 1986,⁽³⁹⁰⁾ to NNECO enclosed 10 additional questions. The NRC letter of August 27, 1986,⁽³⁹¹⁾ to NNECO enclosed five additional questions and comments. The NRC letter of September 22, 1986,⁽³⁹²⁾ enclosed five additional questions. The NNECO letter of October 3, 1986,⁽³¹⁾ to the NRC provides responses to NRC's seven questions.⁽³⁸⁹⁾ The NNECO letter of October 22, 1986,⁽³⁹³⁾ to the NRC encloses responses to NRC's 10 questions.⁽³⁹⁰⁾

1987: The NNECO letter of October 28, 1986⁽²⁸⁾ to the NRC provides responses to NRC's five questions.⁽³⁹²⁾ The NNECO letter of October 30, 1986,⁽²⁶⁾ to the NRC encloses responses to NRC's five questions.⁽³⁹¹⁾ The NNECO letter of January 2, 1987,⁽²⁷⁾ to the NRC provides additional clarification that NRC requested on one of NNECO's responses. The NRC letter of March 9, 1987,⁽²³⁾ to NNECO enclosed two additional questions. The NNECO letter of April 30, 1987,⁽²²⁾ to the NRC provided responses to NRC's two questions⁽²³⁾ and enclosed a copy of the Fuel Consolidation Demonstration Program Description. The letter⁽²²⁾ also noted that the NRC staff had indicated that all other technical issues had been adequately addressed and that no further NNECO action was necessary for NRC issuance of the license amendment. The NNECO letter of May 27, 1987,⁽³⁹⁴⁾ to the NRC requests an amendment to the application⁽¹⁵⁸⁾ to allow storage of up to ten consolidated fuel assemblies (i.e., in five storage locations) in the spent fuel pool at

Millstone-2. The NRC letter of June 2, 1987,⁽³⁹⁵⁾ to NNECO includes Amendment No. 117 to the license (DPR-65) for Millstone-2. It limits spent fuel consolidation to the demonstration phase during which ten spent fuel assemblies will be consolidated into five canisters. The letter⁽³⁹⁵⁾ indicates that the NRC staff is continuing to review the issues associated with full scale consolidation of spent fuel. The NNECO letter of August 4, 1987,⁽³⁹⁶⁾ to the NRC provides a response to an NRC question.⁽³⁹⁷⁾ The NNECO letter of August 11, 1987,⁽³⁹⁸⁾ to the NRC encloses the "Section 50.59 Evaluation of the Consolidation Process." On October 30, 1987, the NRC requested⁽¹⁵⁹⁾ responses to 12 additional questions.

1988: The NNECO letter of January 4, 1988,⁽³⁹⁹⁾ to the NRC provides the responses to NRC's questions.⁽⁴⁰⁰⁾ The NRC letter of February 24, 1988,⁽¹⁵⁷⁾ to NNECO encloses a copy of the "Environmental Assessment and Finding of No Significant Impact" that relates to NNECO's application for a license amendment⁽¹⁵⁸⁾ concerning consolidation of spent fuel at Millstone-2. The enclosure addresses the unlimited use of the spent fuel consolidation process at Millstone-2. The amendment was issued by the NRC in March 1988.

APPENDIX G

TABLE G.1: QUESTIONS, REQUESTS, AND RESPONSES ASSOCIATED WITH
LICENSING OF ROD CONSOLIDATION AT MILLSTONE-2

TABLE G.1. Questions, Requests, and Responses Associated with the Licensing of Rod Consolidation at Millstone-2

Questions/Requests ^(a,b)	Responses ^(a,b)	Comments ^(a,b)
<p><u>QUESTION 1 (July 25, 1986):</u>⁽³⁸⁹⁾ Figure 3.9-3 shows the minimum required fuel assembly exposure as a function of initial enrichment for storage in Region 2 as consolidated fuel. If fuel rods from different assemblies and of different enrichments can be consolidated in one canister, what value of initial enrichment is assumed in complying with Figure 3.9-3?</p>	<p><u>RESPONSE:</u>⁽³¹⁾ For the canister under consideration, the pin with the highest enrichment determines the enrichment assumed for compliance with Figure 3.9-3 of the license amendment request.⁽¹⁵⁸⁾</p>	<p>See submittal.⁽³¹⁾</p>
<p><u>QUESTION 2 (July 25, 1986):</u>⁽³⁸⁹⁾ How is the reactivity effect of less than a full consolidated storage box (less than 352 rods) accounted for?</p>	<p><u>RESPONSE:</u>⁽³¹⁾ The consolidation process permits the placement of solid metal rods in positions where fuel rods are missing. For those instances where solid rods are not used, a limited number of fuel rods can be omitted based on the attached Figure 1. Using Figure 1, the reactivity effect of less than a full consolidated storage box can be established by determining the maximum number of fuel rods that can be omitted while maintaining K-eff at 0.95 or less.</p>	<p>See submittal⁽³¹⁾ for Figure 1.</p>
<p><u>QUESTION 3 (July 25, 1986):</u>⁽³⁸⁹⁾ What are the values of the biases and calculated uncertainties referred to for Regions 1 and 2, and how were they derived?</p>	<p><u>RESPONSE:</u>⁽³¹⁾ The value of the bias is 0.00138 and the 95/95 confidence level calculation uncertainty is 0.00714. The validation report is enclosed as Attachment 1.</p>	<p>See submittal⁽³¹⁾ for Attachment 1.</p>
<p><u>QUESTION 4 (July 25, 1986):</u>⁽³⁸⁹⁾ Explain in more detail how the Region 2 allowable burnup for each initial enrichment accounts for the underestimation of K-effective due to the assumption for uniform axial burnup.</p>	<p><u>RESPONSE:</u>⁽³¹⁾ The nonuniform burnup distribution, which produced the highest difference in reactivity in the Region 2 spent fuel rack when compared with the uniform distribution results, is shown on page 3-6 of the license amendment request.⁽¹⁵⁸⁾ The K-eff is 0.0114. This K-eff was used in Figure 3-3⁽¹⁵⁸⁾ to determine the burnup needed to accommodate the increase in reactivity due to non-uniform burnup. The burnup was found for each initial enrichment shown in Figure 3-3⁽¹⁵⁸⁾. The actual maximum burnup was 1,400 MWd/T for a K-eff of 0.0114. For conservatism, the burnup correction for any initial enrichment was assumed to be 1,800 MWd/T. This value was added to the maximum uniform burnup requirement for each initial enrichment.</p>	<p>See submittals.^(31,158)</p>

TABLE G.1. (contd)

Questions/Requests ^(a,b)	Responses ^(a,b)	Comments ^(a,b)
<p><u>QUESTION 5 (July 25, 1986):</u>⁽³⁸⁹⁾ If Figure 3-4 is based on an infinite array of consolidated fuel, justify why Figures 3-4 and 3-5 need not be derived based on the higher reactivity configuration of one storage pattern of consolidated fuel boxes surrounded by an infinite array of regular fuel assemblies.</p>	<p><u>RESPONSE:</u>⁽³¹⁾ Spent Fuel Pool Technical Specification 3/4.920, "SPENT FUEL POOL", will ensure that the K-eff of the spent fuel pool will always be less than 0.95 for any mix of unconsolidated or consolidated fuel. The Technical Specification requires that the blocked cell remain until the Region II STORAGE PATTERN of the spent fuel pool racks has been filled. At this time, consolidated fuel can be placed in a previously cell-blocked location only if it is completely surrounded by consolidated fuel. In this way, the unconsolidated fuel will be next to consolidated fuel which is stored in a 3 out of 4 pattern. The reactivity of consolidated fuel adjacent to the unconsolidated fuel is less than K-eff 0.95 since it is 3 out of 4, and not 4 out of 4.</p>	See submittal. ⁽³¹⁾
<p><u>QUESTION 6 (July 25, 1986):</u>⁽³⁸⁹⁾ Technical Specification 5.6.1.d should include additional wording to clarify that consolidated fuel can be stored in the fourth location of the storage rack only if the surrounding locations are occupied by consolidated fuel storage boxes.</p>	<p><u>RESPONSE:</u>⁽³¹⁾ We propose that Technical Specification 5.6.1.d be modified to read as follows:</p> <p>"Region II of the spent fuel storage pool is designed to permit storage of consolidated fuel in the fourth location of the storage rack and ensure a K-eff less than or equal to 0.95. Placement of consolidated fuel in the fourth location is only permitted if all surrounding cells of the STORAGE PATTERN are occupied by consolidated fuel."</p>	See submittal ⁽³¹⁾ for revised page 5-5.
<p><u>QUESTION 7 (July 25, 1986):</u>⁽³⁸⁹⁾ The NRC Staff recommends that a Technical Specification Surveillance Requirement be incorporated for consolidated fuel to verify the integrity of the fuel and structural elements before movement or placement in the spent fuel pool.⁽³⁸⁹⁾ What method does NNECO propose to verify the integrity of the storage canister after it has been loaded with fuel rods?⁽³¹⁾</p>	<p>The attached revised page 5-5 reflects incorporation of this change.</p> <p><u>RESPONSE:</u>⁽³¹⁾ Section 4.6.2 of the license amendment request⁽¹⁵⁸⁾ describes the Quality Assurance requirements with which the design, procurement and fabrication of the consolidated fuel storage boxes will comply to ensure that all manufacturing and installation activities conform to the acceptable quality requirements throughout all areas of performance.</p>	

TABLE G.1. (contd)

Questions/Requests ^(a,b)	Responses ^(a,b)	Comments ^(a,b)
<u>QUESTION 7:</u> (July 25, 1986) ⁽³⁸⁹⁾ (cont'd)	<p data-bbox="951 305 1213 337"><u>RESPONSE:</u>⁽³⁹³⁾ (cont'd)</p> <p data-bbox="951 362 1533 646">Static and impact analysis were performed to verify the adequacy of the consolidated fuel storage box design for all the service loads associated with both the consolidation operation and storage in the spent fuel racks. The results of the structural analysis demonstrate that the consolidation box and cover can be safely lifted and transported using the cover as the lift point. The consolidation box is designed such that it will not be overstressed when subjected to a tensile load of 6000 pounds. The insert assembly supporting the weight of the fuel rods can withstand an impact of 5 g.</p> <p data-bbox="951 670 1533 841">The cover assembly is a spring-loaded self-locking device which has a visual indicator when the cover has been engaged and locked in place. Finally, the cover is dimensionally similar to the upper end fitting of the fuel assembly, thereby permitting the consolidated fuel storage box to be transported by the fuel handling tool/system.</p> <p data-bbox="951 865 1533 1101">Additionally, it should be noted that, prior to placement of a consolidated storage box, in the spent fuel racks, the consolidation operation will have transported the fully loaded consolidated storage box to the temporary racks within the Cask Laydown Area to permit access by the fuel handling tool/system, demonstrating that the fully-loaded consolidated storage box can be transported and placed in the racks while maintaining its integrity.</p> <p data-bbox="951 1125 1533 1291">These measures were introduced into the design of the consolidated fuel storage box so that there would be no increase in the probability of a fuel handling accident as a result of storing consolidated spent fuel. We consider these measures to be adequate without any augmentation of the previously proposed surveillance requirement.</p>	
<u>QUESTION 1</u> (August 21, 1986) ⁽³⁹⁰⁾ Does the proposed spent fuel storage design include storage of spent fuel assemblies for Millstone Unit 2 only, or will it accommodate fuel from Millstone Units 1 and 3 or other plants?	<u>RESPONSE:</u> ⁽³⁹³⁾ The proposed spent fuel storage design has been analyzed for and intended for the storage of consolidated spent fuel originating from only the reactor of Millstone Unit No. 2.	See submittal. ⁽³⁹³⁾

TABLE G.1. (contd)

Questions/Requests ^(a,b)	Responses ^(a,b)	Comments ^(a,b)
<p>QUESTION 2 (August 21, 1986):⁽³⁹⁰⁾ Describe the method (procedure) utilized for installing the consolidated fuel assemblies into Region II of the spent fuel pool. Show that the heat load generated by stored spent consolidated fuel assemblies does not exceed the values of 15.2 and 37.8×10^6 Btu/h for normal and abnormal maximum heat load cases.</p>	<p>RESPONSE:⁽³⁹³⁾ The method (procedure) which will be utilized for installing the consolidated fuel assemblies into Region II of the spent fuel pool is essentially the same method used for movement of nonconsolidated fuel assemblies. The cover assembly for a consolidated fuel storage box is a spring-loaded self-locking device that is installed on the consolidation box after the fuel rods have been loaded into the box. The cover is dimensionally similar to the upper end fitting of a fuel assembly, thereby permitting the consolidated fuel storage box to be transported as would a standard fuel assembly using the fuel handling tool/system. This feature is described on page 4-4 of the license amendment request. Proposed Technical Specification 3/4 9.20 of the license amendment request requires that the blocked cell of the STORAGE PATTERN remain until the entire Region II STORAGE PATTERN of the spent fuel pool racks has been filled. At this time, consolidated fuel can be placed in a previously blocked cell location only if it is completely surrounded by consolidated fuel. In this way, unconsolidated fuel will only be next to consolidated fuel which is stored in a 3 out of 4 pattern. The reactivity of consolidated fuel adjacent to the unconsolidated fuel is less than K-eff 0.90 since it is 3 out of 4, not 4 out of 4.</p> <p>The decay heat fraction curves for constant power operating times of 1, 2, and 3 years were plotted from values calculated with the ORIGEN point depletion code. The curves were used to calculate the heat generated by the stored spent fuel, taking into account the power operating time experienced by each fuel batch, assuming annual refueling and 150 hours decay time for the most recently unloaded batch (normal operation) or full core offload. For abnormal operation, the full core offload is assumed to occur 36 days after the most recent refueling shutdown. Intact fuel assemblies must have at least five years decay times before they are consolidated.</p>	<p>See submittals^(158,393)</p>

TABLE G.1. (contd)

Questions/Requests ^(a,b)	Responses ^(a,b)	Comments ^(a,b)
<u>QUESTION 2 (August 21, 1986):</u> ⁽³⁹⁰⁾ (contd)	<u>RESPONSE:</u> ⁽³⁹³⁾ (cont'd)	
	<u>Summary of Normal Operation Heat Generation</u>	<u>Btu/h</u>
	1. Unconsolidated fuel assemblies, all 3 years operating time: (5) 1/3-core batches (150 hours, 1,2,3 and 4 years decay time) (Total 362 assemblies)	12.59×10^6
	2. Unconsolidated damaged fuel assemblies, assumed 2 years time: 10 assemblies (1 year decay time)	0.17×10^6
	3. Consolidated fuel, all 3 years operating time: (19) 1/3-core batches, (5 to 23 years decay time) (Total 1376 assemblies)	2.41×10^6
	Maximum Heat Load	15.17×10^6
	<u>Summary of Abnormal Operation Heat Generation</u>	<u>Btu/h</u>
	1. Full core offload, all 150 hours decay time: (3) 1/3-core batches (3,2, and 1 years operating time) (Total 217 assemblies)	27.94×10^6
	2. Unconsolidated fuel assemblies, all 3 years operating time: (5) 1/3-core batches (36,401,766,1131, and 1496 days decay time) (Total 362 assemblies)	7.27×10^6
	3. Unconsolidated damaged fuel assemblies, assumed 2 years operating time: 10 assemblies (1 year plus 36 days decay time)	0.15×10^6
	4. Consolidated fuel, all 3 years operating time: (19) 1/3-core batches (5 to 23 years decay time; effect of additional 36 days is negligible) (Total 1376 assemblies)	2.41×10^6
	Maximum Heat Load	37.77×10^6

TABLE G.1. (contd)

Questions/Requests (a,b)	Responses (a,b)	Comments (a,b)
<u>QUESTION 2 (August 21, 1986):</u> ⁽³⁹⁰⁾ (contd)	The above heat loads are based on the spent fuel stored in the pool at the end of the plant life and therefore will not be exceeded.	
<u>QUESTION 3 (August 21, 1986):</u> ⁽³⁹⁰⁾ Provide the results of CEPOOL calculations including hypotheses (assumptions) used to develop conservative temperature conditions in spent fuel assemblies.	<p data-bbox="905 391 1535 852"><u>RESPONSE:</u>⁽³⁹³⁾ The spent fuel pool holds storage racks comprised of cells for the Millstone Unit No. 2 spent fuel assemblies. The CEPOOL computer code was used in the analysis of the spent fuel pool thermal hydraulics. Using the flow network method, this code predicts temperatures, velocity, and coolant quality within each cell. Each cell has an internal flow resistance and is connected to its neighbors at the cell inlet through cross flow resistance. A constant axial pressure drop is maintained across the flow network by the entire pool. Coolant in the fuel region, upon transferring heat from the fuel, heats up and becomes lighter than the non-fuel region. The difference in coolant densities in these two regions creates a natural circulation loop to direct flow from the non-fuel region (downcomer) to the fuel region (riser) thereby creating a natural circulation flow loop for removal of heat from the fuel.</p> <p data-bbox="905 878 1535 1339">The conservative method for evaluation of the fuel cooling in the Millstone Unit No. 2 analysis focuses on a row of cells containing the center cell farthest from the pool walls, thus maximizing the hydraulic resistance between the center cell and the downcomer (see attached Figure 1). Furthermore, it is conservatively assumed in the analysis that cooling flow is provided only by the section of the downcomer adjacent to the row of cells. Hence, thermal hydraulic conditions derived from CEPOOL's flow network are conservative relative to the actual pool conditions where cross flow from adjacent rows of cells in the lower plenum can occur. The row of cells considered in the design calculations is analyzed assuming that it is loaded to capacity with the hottest fuel (i.e., minimum discharge time from the reactor) and including a 1.55 radial peaking factor with a 10% additional uncertainty allowance in the heat rates.</p> <p data-bbox="905 1365 1535 1455">When a box of consolidated fuel is stored in a fuel cell of a rack, a water gap exists between the cell wall and the consolidation box (see attached Figure 2). Thus, coolant can bypass the fuel rods</p>	See submittal ⁽³⁹³⁾ for Figure 1 and Table 1.

TABLE G.1. (contd)

Questions/Requests (a,b)	Responses (a,b)	Comments (a,b)
<u>QUESTION 3 (August 21, 1986):</u> (390) (cont'd)	<p data-bbox="947 305 1213 332"><u>RESPONSE:</u> (393) (cont'd)</p> <p data-bbox="947 354 1528 592">within the box and flow through the gap. All heat generated by the fuel within the box is assumed to be removed by the coolant flowing through the box. However, the coolant density change associated with this heat removal is also assumed to exist as driving pressure for the bypass flow through the box/cell wall gap. This introduces conservatism into the analysis since no credit is taken for heat removal from the fuel by lateral heat transfer to the coolant in the box/cell wall gap.</p> <p data-bbox="947 618 1528 760">The attached Table 1 shows the results of CEPPOOL calculations for three cases. The cases are: 1) row of intact fuel, 2) row of consolidated fuel, and 3) worst combination of intact and consolidated fuel (intact fuel in all cell locations except the center cell which contains consolidated fuel).</p> <p data-bbox="947 786 1528 1047">These cases were analyzed at normal and accident conditions. Normal operation comprises water temperature of 150°F at the base of the racks and a minimum pool depth of 23 feet of water above the fuel. Accident conditions assume that, as a result of loss of external cooling, coolant is evaporated to a minimum pool depth of 10 feet of water above the racks and that the racks are blocked by a dropped fuel consolidated canister. Water temperature at the base of the racks is assumed to be 212°F under accident conditions.</p> <p data-bbox="947 1073 1528 1291">As can be seen from the results presented in Table 1, the maximum coolant temperature for the cases of all consolidated fuel and mixed storage of consolidated and intact fuel are less than the case of all intact fuel for both normal operation and accident conditions. Therefore, with respect to thermal hydraulic performances, consolidated storage of spent fuel is no worse than the storage of freshly discharged intact fuel assemblies.</p>	See submittal. (393)
<u>QUESTION 4 (August 21, 1986):</u> (390) Indicate the maximum fuel pin cladding temperature you determined together with a sample calculation used to derive this temperature. Include any hypotheses (assumptions) used.	<u>RESPONSE:</u> (393) Assuming the worst possible scenario, where boiling occurs along the whole length of fuel, the maximum fuel pin cladding temperature is calculated using an equation	

TABLE G.1. (contd)

Questions/Requests ^(a,b)	Responses ^(a,b)	Comments ^(a,b)
<u>QUESTION 4 (August 21, 1986):</u> ⁽³⁹⁰⁾ (cont'd)	<p data-bbox="955 292 1228 332"><u>RESPONSE:</u>⁽²⁹⁹⁾ (cont'd)</p> <p data-bbox="955 341 1501 422">provided by McAdams^(a) for low pressure boiling water. This equation is for use in the fully developed nucleate boiling:</p> $q/A = 0.074 (\Delta T)^{3.86} \quad 30 < p < 100 \text{ psia}$ <p data-bbox="955 487 1543 544">The maximum local heat flux for freshly discharged fuel is:</p> $q/A = \left\{ \frac{97.6 \text{ kW/assembly} \times 3412 \text{ Btu/KW-hr}}{176 \text{ rod/assembly} \times \frac{136.7 \times \pi \times 0.44 \text{ in.}^2}{144 \text{ in.}^2/\text{ft}^2}} \right\} =$ $q/A = 1442 \text{ Btu/h-ft}^2$ <p data-bbox="955 706 1501 738">Thus, substituting and solving for the film ΔT,</p> $\Delta T = 12.9^\circ\text{F}$ <p data-bbox="955 803 1543 925">The maximum fuel pin cladding temperature is 253°F at the base of the fuel racks. This cladding temperature is far below temperatures typically encountered by the fuel during residence in the reactor (653°F).</p> <p data-bbox="955 950 1543 1187">The critical heat flux to cause departure from nucleate boiling (DNB) can be estimated using the expression developed by Zuber^(a). This critical heat flux at spent fuel pool thermal hydraulic conditions is calculated to be $433,000 \text{ Btu/h-ft}^2$ which is much larger than the maximum heat flux (1442 Btu/h-ft^2) from even freshly discharged spent fuel. Therefore, DNB will not occur, and the assumption of nucleate boiling provides a conservative estimate of the maximum cladding temperatures.</p>	

TABLE G.1. (contd)

Questions/Requests (a,b)	Responses (a,b)	Comments (a,b)
<p>QUESTION 5 (August 21, 1986).⁽³⁹⁰⁾ Provide the dimensions of the BORAFLEXTM insert shown in Figure 4.4a of your May 21, 1986 submittal⁽¹⁵⁸⁾ which appears to have been omitted. What quality control measures will be used to assure the boron-10 loading of 0.03 gm/cm²?</p>	<p><u>RESPONSE:</u>⁽³⁹³⁾ The dimensions of the BORAFLEX poison (neutron) material are $141 + 1/4$, $-1/8$ inches long by $8-1/8 \pm 1/16$ inches wide, by 0.110 ± 0.007 inches thick. This material is encapsulated within two, 0.029-inch thick sheets of stainless steel, of equivalent length and width, spot-welded together to form a composite (i.e., sandwich). This composite, in turn, becomes an integral component of the spent fuel poison box assembly.</p> <p>The manufacturer of the BOROFLEX material has in place a quality control program which meets the requirements of ANSI N-45.2 and 10 CFR 50 Appendix B. The manufacturing process is controlled by a series of strict internal manufacturing and testing procedures which meet these requirements. In addition, the vendor has been audited by Combustion Engineering and several other utilities and vendors to ensure their compliance with these requirements.</p>	See submittals. ^(158,393)
<p>QUESTION 6 (August 21, 1986).⁽³⁹⁰⁾ Definition 1.39 on page 1.8 of this submittal⁽¹⁵⁸⁾ describes "Storage Pattern" which is used to limit placement of consolidated fuel assemblies in Technical Specification LCO 3.9.20, as follows:</p> <p><u>"STORAGE PATTERN"</u></p> <p>"1.39 The Region II spent fuel racks contain a cell-blocking device in every fourth rack location for criticality control. This fourth location will be referred to as the blocked location. A STORAGE PATTERN refers to a blocked location and adjacent and diagonal Region II cell locations surrounding the blocked location."</p> <p>In order to clarify this definition, it is recommended that the word "all" be added prior to the word "adjacent." Further, it should be noted that the words, "all adjacent and diagonal" include cells in adjacent modules if that is your intention. Confirm that this is the case.</p>	<p><u>RESPONSE:</u>⁽³⁹³⁾ The words "adjacent and diagonal" were intended to include those cells sharing a cell wall with the cell containing the blocking device and those cells which, while not sharing a cell wall, directly touch the corners of the cell containing the blocking device. The total arrangement includes nine cells, the center cell containing the blocking device and the eight cells surrounding the center cell. The cells that constitute a STORAGE PATTERN that do not have a blocking device installed may in fact be included in more than one STORAGE PATTERN.</p> <p>We propose that Technical Specification Definition 1.39 be modified to read as follows:</p> <p>"1.39 The Region II spent fuel racks contain a cell blocking device in every fourth rack location for criticality control. This fourth location will be referred to as the blocked location. A STORAGE PATTERN refers to a blocked location and all adjacent and diagonal Region II cell locations surrounding the blocked location."</p> <p>The attached revised page 1-8 reflects incorporation of this change.</p>	See submittal ⁽³⁹³⁾ for revised page 1-8.

TABLE G.1. (contd)

Questions/Requests ^(a,b)	Responses ^(a,b)	Comments ^(a,b)
<p>QUESTION 9 (August 21, 1986):⁽³⁹⁰⁾ Technical Specification 4.9.18 refers to Figure 3.9-1 for fuel enrichment and burnup in order to permit placement of a spent fuel assembly in Region II of the spent fuel pool yet it is not contained in the submittal. Page 3/4 9.23 of the present Technical Specification contains Figure 3.9-1, which is identical to Figure 3.9-3 of the May 21, 1986 submittal.⁽¹⁵⁸⁾ Please make suitable corrections to correct this apparent discrepancy.</p>	<p>RESPONSE:⁽³⁹³⁾ Page 3/4 9-23 of the existing Technical Specification for Millstone Unit No. 2 contains Figure 3.9-1, which is titled "MINIMUM REQUIRED FUEL ASSEMBLY EXPOSURE AS A FUNCTION OF INITIAL ENRICHMENT TO PERMIT STORAGE IN REGION II." Figure 3.9-3 of the May 21, 1986 submittal⁽¹⁵⁸⁾ is titled, "MINIMUM REQUIRED FUEL ASSEMBLY EXPOSURE AS A FUNCTION OF INITIAL ENRICHMENT TO PERMIT STORAGE IN REGION II AS CONSOLIDATED FUEL." The attached Figure 3 is an "overlay" of Figure 3.9-1 and 3.9-3 which serves to demonstrate the differences in the two figures. The inclusion of two figures rather than one was based on the decision that separate figures dealing with two distinct operations, 1) storage of intact spent fuel assemblies in Region II vs. Region I, and 2) storage in Region II of spent fuel as consolidated fuel, would cause less confusion for personnel involved in these aspects of the process.</p>	<p>See submittal⁽³⁹³⁾ for Figure 3.</p>
<p>QUESTION 10 (August 21, 1986):⁽³⁹⁰⁾ Describe the means utilized to maintain control over the spent fuel subassemblies in order to ensure against premature consolidation of spent fuel assemblies, and maintain a proper storage condition after consolidation has taken place.</p>	<p>RESPONSE:⁽³⁹³⁾ Procedures will be written to address the questions and concerns raised. The current plant procedure for placing a fuel assembly into a Region II rack will be expanded and utilized for identifying the candidate assemblies for the consolidation operation. Since the Engineering Forms associated with establishing the burnup of the fuel assemblies are retained for the life of the plant, they also permit identification of the candidate fuel assemblies in Region II that have achieved 5-year decay. All candidate assemblies will be taken from Region II of the spent fuel pool.</p>	<p>See submittal.⁽³⁹³⁾</p>
<p>QUESTION 11 (August 21, 1986):⁽³⁹⁰⁾ In response to the NRC staff's Safety Evaluation Report dated November 22, 1985, you noted that an amendment request, targeted for July 1986, would be prepared to restrict plant operation in event both spent fuel pool cooling trains are not available. Please provide the latest schedule for receipt of this amendment.</p>	<p>RESPONSE:⁽³⁹³⁾ The amendment request to restrict plant operation in the event both spent fuel pool cooling trains are not available was submitted to the NRC staff September 26, 1986.</p>	<p>See submittal.⁽³⁹³⁾</p>

TABLE G.1. (contd)

Questions/Requests (a,b)	Responses (a,b)	Comments (a,b)																								
<p><u>QUESTION 12 (August 21, 1986):</u> (390) What is the magnitude of the difference in calculated reactivity between the 12-group KENO-IV calculations (used for benchmarking) and the 16-group DOT-2W calculations (used for consolidated fuel calculations) and how was this accounted for in the fuel rack calculations? What organization performed these calculations and what is their previous experience?</p>	<p><u>RESPONSE:</u> (393) Combustion Engineering (C-E) performed the consolidated fuel calculations, using a DOT-II model benchmarked against KENO-IV and KENO-V. The differences in calculated reactivity between a 123-group KENO-IV model and the DOT model were evaluated by C-E, and were considered to be insignificant.</p> <p>The following table for an infinite-array triangular pitch critical experiment (Ref. 3-1 of the license amendment request⁽¹⁵⁸⁾) compares KENO-V, KENO-IV, and DOT results versus number of energy groups:</p> <table><tr><th colspan="4">INFINITE ARRAY MULTIPLICATION FACTORS</th></tr><tr><th>No. of Groups</th><th>KENO-V</th><th>KENO-IV</th><th>DOT-II</th></tr><tr><td>123</td><td>--</td><td>1.114 ± 0.004</td><td>--</td></tr><tr><td>50</td><td>1.115 ± 0.005</td><td>1.111 ± 0.003</td><td>--</td></tr><tr><td>27</td><td>1.128 ± 0.005</td><td>1.112 ± 0.004</td><td>--</td></tr><tr><td>16</td><td>1.117 ± 0.005</td><td>1.110 ± 0.004</td><td>1.1177</td></tr></table> <p>The finite-array KENO-V 16-group model calculated a multiplication factor of 0.996 ± 0.002 for a critical experiment (Ref. 3-1 of the license amendment request⁽¹⁵⁸⁾) with a geometry equivalent to the consolidated fuel rack geometry.</p>	INFINITE ARRAY MULTIPLICATION FACTORS				No. of Groups	KENO-V	KENO-IV	DOT-II	123	--	1.114 ± 0.004	--	50	1.115 ± 0.005	1.111 ± 0.003	--	27	1.128 ± 0.005	1.112 ± 0.004	--	16	1.117 ± 0.005	1.110 ± 0.004	1.1177	<p>See submittals. (158,393)</p>
INFINITE ARRAY MULTIPLICATION FACTORS																										
No. of Groups	KENO-V	KENO-IV	DOT-II																							
123	--	1.114 ± 0.004	--																							
50	1.115 ± 0.005	1.111 ± 0.003	--																							
27	1.128 ± 0.005	1.112 ± 0.004	--																							
16	1.117 ± 0.005	1.110 ± 0.004	1.1177																							
<p><u>QUESTION 3.1.1 (August 27, 1986):</u> (391) Reference 2, page 19, (158) lists ASTM-A240 and ASTM-A479 as materials for the new spent fuel racks while Reference 1, page 4-6, (158) lists ASTM-A240 and ASTM-A276. Please explain the difference. ASTM-A276 is not listed in the reviewer's ASTM specifications nor is it an ASME code material under the normal corresponding SA-276 designation.</p>	<p><u>RESPONSE:</u> (26) The material listing of Section 4.b on page 4-6 of the license amendment request⁽¹⁵⁸⁾ is not complete. Section 4.b should have read, "ASTM-A276 or ASTM-A479," to be consistent with the material shown on page 4-42 of the license amendment request. ASTM-A276 was an option that was considered for use in fabrication of the spent fuel racks, however, ASTM-A276 was not used. The documentation packages and material certification provided with the spent fuel racks show that ASTM-A240 plate and strip and ASTM-A479 shapes and bar were used in fabrication. Both ASTM-A240 and ASTM-A479 are ASME code materials.</p>	<p>See submittals. (26,158)</p>																								

TABLE G.1. (contd)

Questions/Requests (a,b)	Responses (a,b)	Comments (a,b)
<p><u>QUESTION 3.1.2 (August 27, 1986):</u>⁽³⁹¹⁾ Reference 1, page 4-11 calls out 3000 psi concrete for construction material for the spent fuel pool/auxiliary building. Three pages later, 4800 psi concrete is listed as a concrete material property for the mathematical model. Reference 3, page 26, calls out 3000 psi concrete for pool material. Please explain the apparent conflict.</p>	<p><u>RESPONSE:</u>⁽²⁶⁾ The construction material used for the Millstone Unit No. 2 spent fuel pool is called out as 3,000 psi, 28-day strength concrete. The actual test cylinder concrete compressive strengths were obtained for the spent fuel pool structure. The test reports showed that the concrete had an actual minimum 90-day compressive strength of 4,800 psi. The value of 4,800 psi was therefore used in concrete strength calculations. Per ACI 349, the appropriate strength reduction factors were also used in the calculations to account for, among other things, variations in material strengths of the construction material.</p>	<p>See submittals.^(26,158)</p>
<p><u>QUESTION 3.1.3 (August 27, 1986):</u>⁽³⁹¹⁾ Reference 2, Section 4.2, starting on page 20, discusses a rather extensive test program to be used in conjunction with the analysis of the Spent Fuel Rack System but the testing program and the use of resulting data are not addressed in the SAR (Reference 1). Please discuss as the testing portion seemed to be an important step in doing a reliable analysis of the Spent Fuel Rack System.</p>	<p><u>RESPONSE:</u>⁽²⁶⁾ Three different series of structural tests were conducted on a prototype consolidated fuel storage box (CFSB) to obtain static and dynamic properties of the box and fuel rods for incorporation in the computer models. As the name implies, CFSBs are the boxes in which fuel rods are stored in the spent fuel racks after consolidation. The test series were:</p> <ol style="list-style-type: none"> 1. Static load-deflection tests to measure the lateral stiffness of the box with fuel rods. 2. Static compressive tests on a short box section to measure the local wall stiffness of the box. 3. Forced vibration tests on a loaded box in air and in water to measure natural frequencies, mode shapes, critical damping ratios, excitation forces, and magnification ratios. <p>Results of the tests were evaluated and correlated with the analytical models to obtain a computer model of the CFSB loaded with fuel rods.</p>	<p>See submittals.^(26,158) Tables 1 and 2 and Figures 1-5 are in the response submittal.⁽²⁶⁾</p>

TABLE G.1. (contd)

Questions/Requests (a,b)	Responses (a,b)	Comments (a,b)
QUESTION 3.1.3 (August 27, 1986): (391) (cont'd)	<p data-bbox="951 297 1199 329"><u>RESPONSE:</u> (26) (cont'd)</p> <p data-bbox="951 354 1266 378">LATERAL LOAD DEFLECTION TEST</p> <p data-bbox="951 402 1524 686">The objective of this series was to obtain the static deflection characteristics of the CFSB when subjected to push-pull lateral load cycles applied at the center of the span. A full-length prototype was specifically fabricated for the test and was pin-supported at each end. It stood vertically on the test stand and was filled with depleted fuel rods on a triangular pitch. Two fuel rod compaction ratios were tested, 378 rods loaded in rows of 18 (tight compaction) and 352 rods in the same configuration (intermediate compaction, i.e., 2:1 consolidation).</p> <p data-bbox="951 711 1188 735">FORCED VIBRATION TEST</p> <p data-bbox="951 760 1524 1068">The objective of this series was to identify the vibrational characteristics of the consolidated fuel storage box in air and in water when loaded with fuel rods at three different compaction ratios, and also when empty. The test CFSB and the fuel rod configurations were the same as in the lateral load deflection test, and the compaction ratios were: tight (378 rods), intermediate (352 rods) and loose (306 rods). The third compaction ratio was included because the fuel rods were expected to affect vibration damping, whereas they had practically no effect on static deflections of the CFSB.</p> <p data-bbox="951 1092 1524 1279">Parameters of interest were the CFSB's natural frequencies, associated mode shapes and modal critical damping ratios. These parameters were determined over a wide range of sinusoidal and constant displacement excitations induced through the lower support pin. Furthermore, the required forces and the strain distributions along the CFSB were measured for each response.</p>	

TABLE G.1. (contd)

Questions/Requests ^(a,b)	Responses ^(a,b)	Comments ^(a,b)
<u>QUESTION 3.1.3 (August 21, 1986):</u> (391) (cont'd)	<u>RESPONSE:</u> (26) (cont'd) LOCAL STIFFNESS TEST The objective of this series was to obtain the local stiffness property characteristics of the consolidated fuel storage box when loaded with fuel rods. Two compaction ratios were tested, namely tight (378 rods) and intermediate (352 rods). For this purpose, short sections of the CFSB, filled with depleted fuel rodlets, were compression-tested in a horizontal position at several load increments in a tensile test machine. Load versus deflection characteristics were obtained for the upper end, mid-section and lower end of the CFSB. The upper and lower ends included the locking cover and the box floor, respectively. Loading was applied to the upper surface of the box as a transverse line load in the center of the length for the mid-section, and at the upper and lower ends. The box section deformations were measured by the tensile machine extensometer and by dial indicators. CFSBs in the spent fuel racks are not supported at their upper ends. The upper end test section was therefore cantilevered from the tensile machine support plate in order to include a beam mode effect in the measured deformation and stiffness characteristics of the upper end. After completion of all three types of local stiffness tests in the elastic region, the three types were repeated, extending loads beyond the linear range. The plastic range was characterized either by a plateau on the load versus deflection curve or by large formations of the box walls.	

TABLE G.1. (contd)

Questions/Requests ^(a,b)	Responses ^(a,b)	Comments ^(a,b)
<u>QUESTION 3.1.3 (August 27, 1986):</u> ⁽³⁹¹⁾ (cont'd)	<p data-bbox="947 310 1192 337"><u>RESPONSE:</u>⁽²⁶⁾ (cont'd)</p> <p data-bbox="947 362 1339 383">CONSOLIDATED FUEL STORAGE BOX MODEL</p> <p data-bbox="947 407 1522 789">A CESHOCK model was developed to simulate the structural characteristics of the consolidated fuel storage box (CFSB) loaded with fuel rods in water. The CFSB model properties were developed in steps. First, the static lateral load deflection test data were evaluated and analyzed to identify the static stiffness, shape and boundary conditions of the CFSB as tested. A static model was derived with stiffness obtained directly from the CFSB structural dimensions, and the calculated results were compared with the test results. In the next step, the structural weight and hydrodynamic effects were added to obtain a lumped-mass dynamic model of the CFSB and fuel rods. Results calculated with this model were then compared with data from the forced vibration test.</p> <p data-bbox="947 813 1283 834">MODEL-TEST CORRELATION RESULTS</p> <p data-bbox="947 859 1522 1027">The results of the lateral load deflection and forced vibration tests were compared as described in the following paragraphs with calculations using the CFSB model. Data from the CFSB local stiffness test were used to derive elements in the combined storage rack and CFSB model to simulate impacting between the CFSB and the spent fuel storage rack.</p> <p data-bbox="947 1052 1262 1073"><u>Lateral Load Deflection Test</u></p> <p data-bbox="947 1097 1522 1458">Statistical analyses of the test data were performed and average deflection shapes of the deformed consolidated fuel storage box obtained. The data were analyzed in categories related to the compaction ratio and the maximum displacement value (i.e., small or large deformation ranges). It was found that the number of fuel rods and the magnitude of the displacement did not significantly affect the box stiffness. The stiffness of the CFSB and fuel rods appears to be primarily a function of the structural dimensions and material of the box alone. Figure 1 compares static displacements calculated with the computer model and the test data. Two calculated displacement curves are shown, one for a CFSB simply supported</p>	

TABLE G.1. (contd)

Questions/Requests (a,b)	Responses (a,b)	Comments (a,b)
<u>QUESTION 3.1.3 (August 27, 1986):</u> (391) (cont'd)	<p data-bbox="961 305 1220 329"><u>RESPONSE:</u> (26) (cont'd)</p> <p data-bbox="961 354 1545 451">at both ends and one for fixed support at both ends. Agreement between the simply supported curve and the data from the test, in which the CFSB was simply supported, is good.</p> <p data-bbox="961 475 1205 500"><u>Forced Vibration Test</u></p> <p data-bbox="961 524 1545 760">The consolidated fuel storage box was excited at the bottom with a sine-sweep for various levels of input displacements. The double-amplitude displacements ranged from 0.005 to 0.060 inches. The sine-sweep tests yielded the natural frequency of the CFSB and a measure of damping. In addition, the maximum displacements at various elevations, strains and the base reaction force were measured. Selected data are plotted in Figures 2 through 5 inclusive.</p> <p data-bbox="961 784 1545 954">As previously noted, the static lumped-mass model of the CFSB was modified to include the weight of the fuel rods and hydrodynamic effects for tests conducted in water. Each modification was unique for the particular test condition being simulated. Selected test measurements were compared with model properties and simulated dynamic responses.</p> <p data-bbox="961 979 1545 1218">It was found that the frequency of the empty CFSB in air is predicted well by hand calculations based on the structural dimensions of the box. For the empty CFSB in water, the effects of water on frequency can be predicted well through standard theoretical calculation techniques. The water can be represented as additional lumped-weight consisting of terms for the contained and displaced water. When fuel rods are placed in the CFSB, the frequency changes due to the additional weight of</p>	

TABLE G.1. (contd)

Questions/Requests ^(a,b)	Responses ^(a,b)	Comments ^(a,b)
<u>QUESTION 3.1.3 (August 27, 1986):</u> ⁽³⁹¹⁾ (cont'd)	<p data-bbox="940 293 1192 321"><u>RESPONSE:</u>⁽²⁶⁾ (cont'd)</p> <p data-bbox="940 345 1524 391">the rods. The rods do not provide any significant stiffness contribution.</p> <p data-bbox="940 415 1524 678">Table 1 compares first mode test and model frequencies. The agreement is good. The size of the test tank did not have a large effect on the first mode frequency of the CFSB. During the testing, it was noticed that tank vibrations occurred at the second resonance near 30 Hz. Similarly, it was observed that the I-beam strong-back vibrated near the third resonance of about 40 Hz. Since typically seismic motions do not exhibit amplified response greater than 25 Hz, the emphasis in this study was placed on the first mode of vibration.</p> <p data-bbox="940 703 1524 987">Time history model simulations of some test sequences were also conducted to compare responses of other quantities, such as damping, base reaction force and displacement. A comparison of test and model parameters for one such simulation is included in Table 2. It can be seen from Figures 2 and 3 that damping varies more than frequency with the input amplitude. Therefore, in the model simulation of a particular test, the damping value from the individual test should be applied in the model. When this was done as shown in Table 2, the agreement of the model and test data was excellent.</p> <p data-bbox="940 1011 1524 1039"><u>Consolidated Fuel Storage Box Local Stiffness Test</u></p> <p data-bbox="940 1063 1524 1396">Short sections of the consolidated fuel storage box containing fuel rods were each subjected to a transverse line load on the uppermost wall while resting horizontally on a flat surface. Deflections at various locations on the CFSB walls were measured as a function of load. In one test, the load was applied at the end of the section. In the first of these two tests, the box section under the load line included the CFSB floor, and the test measured the local stiffness of the lower end of the CFSB. In the second of the two tests, the box section under the load line included the CFSB cover, and the test measured the local stiffness of the upper end of the CFSB.</p>	

TABLE G.1. (contd)

Questions/Requests (a,b)	Responses (a,b)	Comments (a,b)
<u>QUESTION 3.1.3 (August 27, 1986):</u> (391) (cont'd)	<p data-bbox="953 298 1205 337"><u>RESPONSE:</u> (26) (cont'd)</p> <p data-bbox="953 354 1535 889">The test results showed that the behavior of the CFSB is complicated under this type of load and that it exhibits a great variation in stiffness values. At high loads, local yielding or buckling of the box walls occurs. This is depicted by the drastic changes in the stiffness values. At the bottom and upper sections, an increased load can be attained because of the support provided by the end fixtures. In general, the overall stiffness values are significantly higher than the CFSB stiffness in the beam mode exhibited in the lateral load deflection test. In the beam mode, the CFSB supported its ends and loaded at the center exhibits a stiffness of about 16,600 lb/in. The local CFSB stiffness testing shows values from 100,000 to 300,000 lb/in. CFSB impact stiffnesses for the consolidated fuel storage box were derived from these values. Because the measured stiffness varied with the applied load level, the actual stiffness values to be used in fuel rack seismic analysis for CFSB impacting were based upon the site specific seismic excitation.</p>	
<p data-bbox="254 906 905 1058"><u>QUESTION 3.2.1 (August 21, 1986):</u> (391) The Basis for Limiting Condition for Operation 3.9.6, 3/4.9.7, "Crane Travel - Spent Fuel Storage Building," states that, "specific analysis has been performed for the drop of a consolidated fuel storage box on an intact fuel assembly." Where is the analysis?</p>	<p data-bbox="953 906 1535 1149"><u>RESPONSE:</u> (26) An analysis was performed by Combustion Engineering (C-E) which considered the drop of a fully loaded consolidated fuel storage box onto a stored intact fuel assembly. The total drop height for such an accident is twenty-eight (28) inches to the top of the fuel assembly. A nonlinear multi-spring/mass model (Figure 6) of a fully loaded consolidated fuel storage box dropping twenty-eight (28) inches onto the top of a stored fuel assembly was developed.</p> <p data-bbox="953 1174 1535 1320">The CESHOCK computer code was used to perform a dynamic nonlinear analysis to determine the maximum impact load in the stored fuel assembly due to the drop. The result of that analysis was a maximum impact force in the stored fuel assembly of 100,000 lb.</p> <p data-bbox="953 1344 1535 1435">The calculated peak impact load was then statically applied to the fuel assembly to assess its structural integrity. Assuming no lateral deformation results in the entire impact load axially</p>	

TABLE G.1. (contd)

Questions/Requests ^(a,b)	Responses ^(a,b)	Comments ^(a,b)
<u>QUESTION 3.2.1 (August 27, 1986):</u> ⁽³⁹¹⁾ (cont'd)	<u>RESPONSE:</u> ⁽²⁶⁾ (cont'd)	
	compressing the fuel assembly with the guide tubes compressing first until the ultimate stress of 80,000 psi is reached. This ultimate guide tube stress is reached at a load of 37,840 lb. The remainder of the 100,000 lb load is carried by the fuel rods as the grid cage crushes and the load is applied directly to the top of the fuel rods. A load of 148,750 lb was calculated to be required to produce the yield stress of 25,000 psi in the fuel rods. Since this load is greater than 100,000 lb, the fuel rods will not yield, and the analysis demonstrated that the stored fuel assembly is capable of absorbing the kinetic energy of the drop with no fuel rod failures.	
<u>QUESTION 3.3.1 (August 27, 1986):</u> ⁽³⁹¹⁾ Reference 2, Section 1.4, page 7, discusses a reactivity meter to be used for determination and verification of burnup. This is not mentioned in the SAR's discussion of reactivity determination for storage of spent fuel. Please explain.	<u>RESPONSE:</u> ⁽²⁶⁾ The reactivity meter discussed in Section 1.4, page 7, of the March 30, 1984 ^(b) letter, as stated in the letter, is a state-of-the-art device which is still under development. The engineering design reviews are still ongoing. The determination and verification of burnup is currently controlled administratively and the described reactivity meter is not required for determination of reactivity for the storage of spent fuel.	See submittals ^(26,158)
<u>QUESTION 1 (September 22, 1986):</u> ⁽³⁹¹⁾ Because many fuel rod movements will have been involved in the reconstituted fuel assembly, it is likely activated crud will be released from some fuel rods and adhere to some other rods. Show how an increase in dose rates from such crud will be precluded when consolidated assemblies are returned to storage in the spent fuel pool.	<u>RESPONSE:</u> ⁽²⁸⁾ The potential dose rate consequence from crud redepositing on fuel surfaces is insignificant for three reasons. First, the dose rate from crud is a very small fraction of the overall dose rate relative to the fission products in the fuel. Second, since the object of concern is the overall dose rate approximately 40 feet away, it is irrelevant whether crud is transferred from one rod to another. And third, nearly all of the crud released will be filtered during the consolidation process rather than redeposited on other fuel rods. The fuel consolidation system contains an elaborate filtration system to control and contain any dispersion of crud released from the fuel rods or fuel assembly. The filtration system is designed to capture the crud at the point of release so as not to allow the crud to spread to any area of the fuel pool.	See submittal. ⁽²⁸⁾

TABLE G.1. (contd)

Questions/Requests ^(a,b)	Responses ^(a,b)	Comments ^(a,b)
<u>QUESTION 1 (September 22, 1986):</u> ⁽³⁹²⁾ (cont'd)	<p>The filtration system consists of two individual filtration skids, each with a 250 gpm flow rate, with an in-line strainer at the pump inlet to protect the pump from any large debris. Downstream of the pump are two additional stages of parallel filters to minimize filter usage, thereby holding down the volume of waste products.</p> <p>The filter system is manifolded to all shrouded locations where any fuel rod motion will occur. This manifold and shrouded system provides downward flow in each station. The filter system is sized to ensure sufficient velocity at the tooling-to-workstation interface to entrain and filter any released crud before it can be dispersed within the work area.</p>	
<u>QUESTION 2 (September 22, 1986):</u> ⁽³⁹²⁾ Describe the methodology that will be used to preclude spent fuel pool water from becoming highly contaminated if fuel rod rupture occurs (e.g., close capture cleanup system with filters appropriately shielded), and the method used for identifying rods that are most likely to rupture.	<p><u>RESPONSE:</u>⁽²⁸⁾ The shrouded workstations and positive filtration systems [described in the above response to Question 1 (September 22, 1986)] will preclude the spent fuel pool water from becoming highly contaminated if a fuel rod ruptures. The first stage of the filtration system is a strainer element which will capture large particles. The second and third stages are pleated paper elements which can capture particles in the low micron range. Further, each filter system is also vented</p>	<p>See submittal.⁽²⁸⁾</p>

TABLE G.1. (contd)

Questions/Requests (a,b)	Responses (a,b)	Comments (a,b)
<u>QUESTION 2 (September 22, 1986):</u> (contd)	<p data-bbox="945 300 1155 332"><u>RESPONSE:</u> (contd)</p> <p data-bbox="945 349 1533 787">and connected to the plant off-gas handling system. In the event of a rod rupture, one would only expect the release of insoluble, gaseous nuclides (principally Kr-85). The multiple rod pulling tool is fully shrouded and vented to the plant off-gas system. This feature accounts for the collection and disposal of any released fission gases. At Millstone Unit No. 2, the consolidation system will be operated in the spent fuel pool cask laydown area. The cask laydown area gate will be in place during consolidation operations, such that any uncontrolled contamination will be confined to this work area. The system filters are located on skids adjacent to the bottom of the workstation frame. The filters are therefore covered by the full height of the fuel pool water. Filter change-out will be accomplished underwater, thus providing adequate shielding.</p> <p data-bbox="945 803 1533 933">Prior to consolidation, all candidate fuel assemblies will be pre-inspected with the consolidation system TV cameras to review the condition of the exterior rows of rods. Any fuel assemblies found to have defects will not be consolidated.</p>	
<u>QUESTION 3 (September 22, 1986):</u> (392) Provide all data (mathematical models, parameters, codes and techniques used) that were used in the calculational model devised to determine the increase in dose rates in the spent fuel pool area due to a buildup of radionuclides in the pool water (page 5-4, Attachment 2).	<p data-bbox="945 950 1533 1218"><u>RESPONSE:</u> (28) Calculations have been performed to determine the incremental dose at certain areas around the spent fuel pool due to the proposed increase in spent fuel pool capacity. The dose was first calculated with the pool filled to its present capacity and then predicted for the pool filled to its proposed capacity. A model has been developed to estimate the pool's activity using fuel information available from ORIGEN computer runs generated for the heat load calculations. The model included the following parameters.</p> <ol data-bbox="945 1234 1533 1430" style="list-style-type: none"> 1. Li = Leakage is isotope i into the pool from each cycle, in $\mu\text{Ci/sec}$. 2. f = purification flow constant, in sec^{-1}. The constant can be obtained by dividing the purification flow rate by the pool volume. 3. λ_i = decay constant of isotope i, in sec^{-1}. 	See submittal. (28)

TABLE G.1. (contd)

Questions/Requests (a,b)	Responses (a,b)	Comments (a,b)
QUESTION 3 (September 22, 1986): (contd)	<p data-bbox="961 284 1165 316"><u>RESPONSE:</u> (contd)</p> <p data-bbox="961 332 1533 414">4. γ = fraction of nuclide in the purification system discharge header (If DF=100, then $\gamma = 0.01$. If DF = 10, $\gamma = 0.1$.)</p> <p data-bbox="961 430 1533 462">5. t = time subsequent to pool insertion, in sec.</p> <p data-bbox="961 479 1533 527">6. N_i = activity of each nuclide in the pool, in μCi.</p> <p data-bbox="961 544 1533 625">Using the above parameters, it can be seen that the time rate of change of activity in the pool is given as:</p> $\frac{dN_i}{dt} = L_i - (\lambda_i + f - \gamma f) N_i$ <p data-bbox="961 714 1417 747">The leakage rate, L_i, may be defined as:</p> $L_i = D \alpha_i N_{fi}$ <p data-bbox="961 812 1491 917">where D = Fraction of fuel which leaks (i.e., failed fuel = 1%). α_i = Leak rate coefficient for isotope i. N_{fi} = curies of isotope i in the fuel.</p> <p data-bbox="961 933 1501 1015">After the fuel is placed in the spent fuel pool, the curies of an isotope in the fuel is governed only by the decay rate and leakage rate.</p> $N_{fi} = N_{ofi} e^{-(\lambda_i + D \alpha_i)t}$ <p data-bbox="961 1079 1501 1128">where N_{ofi} is the Curies of isotope i in the fuel at the time the fuel is placed in the pool.</p> <p data-bbox="961 1144 1123 1177">and therefore,</p> $N_i = \frac{D \alpha_i N_{ofi} e^{-(\lambda_i + D \alpha_i)t} [1 - e^{-(\lambda_i + f - \gamma f)t}]}{\lambda_i + f + \gamma f}$ <p data-bbox="961 1339 1501 1388">The values of N_{ofi} are available from the ORIGEN runs. The leak rate coefficient of the isotopes</p>	

TABLE G.1. (contd)

Questions/Requests ^(a,b)	Responses ^(a,b)	Comments ^(a,b)
<u>QUESTION 3 (September 22, 1986):</u> (contd)	<p data-bbox="945 305 1150 329"><u>RESPONSE:</u> (contd)</p> <p data-bbox="945 354 1522 492">were those for full power adjusted for cold fuel conditions. The activities of a number of nuclides were calculated using the equation above, but the only two of significant value were those for cesium-134 and cesium-137. The concentrations of cobalt in the pool water were assumed to be:</p> <p data-bbox="945 516 1522 565">Cobalt-58: 4.16×10^{-3} microcuries per milliliter Cobalt-60: 3.61×10^{-4} microcuries per milliliter</p> <p data-bbox="945 589 1522 800">This data was consistent with samples obtained from the spent fuel of an existing facility and were chosen in lieu of ORIGEN data, since review of the data from several facilities shows that the concentration of cobalt is not directly dependent on the number of assemblies in the pool. Therefore, it was assumed that the concentration of cobalt defined above remains constant for every fuel cycle off-load introduced into the spent fuel pool.</p> <p data-bbox="945 824 1522 946">The pool size used is 25 feet by 42 feet by 38.5 feet. The cleanup flow rate used was 125 gallons per minute, and the decontamination factor of the pool cleanup system was assumed to be 10 ($\gamma = 0.1$) for cobalt and 2 ($\gamma = 0.5$) for cesium.</p>	See submittal. (28)
<u>QUESTION 4 (September 22, 1986):</u> ⁽³⁹²⁾ Provide a table showing expected tritium exposures due to the increased fuel storage capacity as the design maximum spent fuel pool water temperature increases from 120°F to 131°F (page 5-5, Attachment 2).	<p data-bbox="945 971 1522 1190"><u>RESPONSE:</u> (28) The design maximum pool water temperature under normal operation with the increased amount of stored spent fuel increases from 122°F to 131°F. This increase in spent fuel pool temperature will result in a slight increase in the evaporation rate from the spent fuel pool. Because the water evaporating is tritiated, the personnel exposure from airborne tritium in the area of the pool will increase correspondingly.</p> <p data-bbox="945 1214 1522 1399">Several models exist for predicting the rate of evaporation from pools open to the general environment. However, they all tend to predict evaporation rates for spent fuel pools which are much higher than those which actually occur. Therefore, when calculating the evaporation rate for the spent fuel pool, the following assumptions were made:</p>	

TABLE G.1. (contd)

Questions/Requests (a,b)	Responses (a,b)	Comments (a,b)
<u>QUESTION 4 (September 22, 1986):</u> (contd)	<p><u>RESPONSE:</u> (contd)</p> <p>•Because of high pool sweep flow rates directly above the pool, the yearly average air temperatures and relative humidity in the spent fuel pool building will not change as a result of small increases in the spent fuel pool water temperature.</p> <p>•All other factors in the mathematical models used to predict evaporation rates remain unchanged.</p> <p>With these two assumptions, the new evaporation can be predicted as follows:</p> <p>New Evaporation = [Present Evaporation Rate] $\left[\frac{V_n - v}{V_p - v} \right]$</p> <p>$V_n$ =Water vapor pressure at the new pool water temperature in inches of Hg.</p> <p>V_p =Water vapor pressure at the present pool water temperature in inches of Hg.</p> <p>v =Water vapor pressure in the air in inches of Hg.</p> <p>The dose commitment from tritium is currently below detectable levels (i.e., zero), and as such, no table of expected tritium exposure exists. The calculated increase of 28% in tritium exposure is based on evaporation at 122°F to 131°F and can only be given in relative terms because of the lack of baseline tritium concentrations. The small increase of 28% will ensure that the dose commitment from tritium remains below detectable levels.</p>	
<u>QUESTION 5 (September 22, 1986):</u> (401) Your proposal (page 1-5) shows a total of 1277 storage locations including 10 spare cells and 217 cells reserved for a full core offload. What is proposed for the remaining 69 locations?	<p><u>RESPONSE:</u>(28) There are actually 1346 cell locations in the spent fuel pool. The 69 cell locations in question are blocked cells. These locations will remain as blocked cells to accommodate the 3-out-of-4 storage pattern required to support the 5-year decay of intact fuel assemblies. Below is a breakdown of actual cell usage.</p>	See submittals.(28,158)

TABLE G.1. (contd)

Questions/Requests (a,b)	Responses (a,b)	Comments (a,b)
QUESTION 5 (September 22, 1986): (contd)	<p>RESPONSE: (contd)</p> <p>10 spare cells 217 cells reserved for full core offload 688 cells to contain consolidated fuel *362 cells with intact assemblies awaiting 5-year decay 1277 total cells containing fuel</p> <hr/> <p>*The 362 cells are broken down as follows:</p> <ul style="list-style-type: none"> • 157 cells in Region I, containing intact assemblies • 205 cells in Region II, containing intact assemblies, stored in a 3-out-of-4 pattern for an actual total of 274 cells, 69 of which are blocked cells. <p>This provides for:</p> <p>1277 - total cells containing fuel 69 - cells in Region II that are blocked cells 1346 - total cell locations.</p>	
<p>QUESTION (December 8, 1986): (27) In the response to Question 3.2.1 (August 27, 1986), (26,392) Northeast Nuclear Energy Company (NNECO) assumes that no lateral deformation results from 100,000 lb load impacting the fuel assembly. This assumption is not justified and seems invalid since they have determined that the guide tubes are yielded due to the load. A random drop is likely to create an eccentric load and thus increase the likelihood of buckling occurring in the guide tubes and the fuel rods. In addition, crushed guide tubes could conceivably damage and exert lateral loads on adjacent fuel rods. The response to the question is thus inadequate.</p>	<p>RESPONSE: (27) Based on geometric constraints and available maximum clearances, an eccentric drop of the consolidated fuel canister (CFC) is not possible. Due to the length of the CFC (155 1/2 in.), the length (10 in.) the CFC has to travel in the cell prior to impacting the fuel assembly, and the available clearances between the CFC and cell wall (max. clearance = 0.375 in.), a non-axial drop would result in rigid body rotation of the CFC. Rigid body rotation results in the CFC impacting the cell walls. An impact onto the cell walls has been analyzed and found to be acceptable. An axial drop of the CFC onto a stored fuel assembly will result in the floor of the CFC impacting the upper end fitting posts of the fuel assembly. The C-E analysis considered this impact condition.</p>	

TABLE G.1. (contd)

Questions/Requests (a,b)	Responses (a,b)	Comments (a,b)
<u>QUESTION (December 8, 1986):</u> (contd)	<p data-bbox="961 305 1171 337"><u>RESPONSE:</u> (contd)</p> <p data-bbox="961 354 1543 527">The analysis performed by C-E evaluated the maximum dynamic impact load in the stored fuel assembly due to a 28 in. drop of a fully loaded consolidated fuel storage box. The analysis was conservative because it did not take credit for the following factors that reduce the magnitude and minimize the effect of lateral loading on the fuel rods:</p> <ol data-bbox="961 544 1543 1177" style="list-style-type: none"> <li data-bbox="961 544 1543 625">1. Under dynamic loading conditions the effective strength of the guide tube and fuel rod are increased. <li data-bbox="961 641 1543 722">2. At any instant in time, the maximum impact load is not distributed throughout the entire fuel assembly. <li data-bbox="961 738 1543 885">3. Significant fuel rod loading can only occur after the guide tube axial length is reduced by 1.262 in. (i.e., the axial gap between upper end fitting and fuel rods is closed), and all the lateral gaps in a row of fuel rods and the gap between the fuel assembly and CFC are closed. <li data-bbox="961 901 1543 1006">4. Spacer grid cell wall deformation dissipates the lateral load. The spacer grids fail in bending rather than buckling therefore minimizing the transfer of load to the fuel rods. <li data-bbox="961 1023 1543 1079">5. Lateral deformation of the fuel storage cell wall dissipates the lateral load. <li data-bbox="961 1096 1543 1177">6. The drop analysis of the consolidated fuel storage box neglected energy losses occurring in the storage box. 	
<p data-bbox="275 1193 919 1390"><u>INQUIRY 1 (March 8, 1987):</u> (23) Proposed Technical Specification (TS) 3.9.19 would require that candidate fuel assemblies (for consolidation) must have decayed for at least 5 years. Please provide a justification for the proposed decay time in terms of the consolidation process. In this regard, you should provide a complete description of the consolidation process and an associated safety analysis.</p>	<p data-bbox="961 1193 1543 1291"><u>RESPONSE:</u> (22) The proposed TS 3.9.19 requiring that candidate spent fuel assemblies (for consolidation) have a 5-year decay time is not based on consolidation process considerations.</p> <p data-bbox="961 1307 1543 1390">The May 21, 1986 submittal (158) identifies that the thermal hydraulic design of the consolidated fuel storage box, as well as the design basis heat load</p>	<p data-bbox="1585 1193 1948 1226">See submittals. (22,158)</p>

TABLE G.1. (contd)

Questions/Requests (a,b)	Responses (a,b)	Comments (a,b)
<u>INQUIRY 1 (March 8, 1987):</u> (contd)	<p><u>RESPONSE:</u> (contd)</p> <p>for the spent fuel pool (with respect to 688 consolidated boxes), is based upon the 5-year decay criterion. (Refer to submittal pages 1-1, 1-5, 3-11 and Table 3.2-2).</p> <p>This information is consistent with the March 30, 1984 submittal^(b) to the NRC on NNECO's Spent Fuel Disposition Plans for Millstone Unit No. 2 and the May 17, 1984 Summary Meeting with the NRC in Bethesda, Maryland, as documented in the NRC Minutes, dated June 4, 1984.^(c) Furthermore, the NRC Minutes identify that the 5-year decay criterion was specifically discussed as a conservative design basis for the storage of consolidated spent fuel.</p> <p>Attached, for your information, is a copy of a system description on the consolidation process intended for use at Millstone Unit No. 2. Safety evaluations of this consolidation process are currently being conducted. NNECO intends to submit the results of these evaluations to the NRC Staff when complete.</p>	
<p><u>INQUIRY 2 (March 9, 1987):</u>⁽²³⁾ Technical Specification (TS) 5.6.3 provides for a total of 1346 storage locations in the spent fuel pool. The practical limit for fuel storage is 1277 locations due to the need to allow 5 years for decay time of fuel assemblies prior to consolidation. The remaining 69 locations would contain cell blocking devices. You should propose a revised TS 5.6.3, limiting storage to 1277 locations, or justify the need for 1346 locations.</p>	<p><u>RESPONSE:</u>⁽²²⁾ The referenced paragraph 5.6.3 is the Capacity description that appears in the DESIGN FEATURES portion of the Technical Specifications. The purpose of this description is to delineate the actual number of storage cells associated with the spent fuel rack inventory in the pool. This description <u>does not</u> establish the spent fuel capacity of the pool which is determined by:</p> <ol style="list-style-type: none"> 1. A letter to the NRC Staff, dated May 21, 1986, which states the capacity as 1965 fuel assemblies. 2. Attachment 2 to the May 21, 1986 letter, which states on page 1-5 that the storage restrictions and thermal load restrictions imposed by the cooling system establish the maximum spent fuel capacity of 1965 to be distributed as: 	See submittals. ^(22,158)

TABLE G.1. (contd)

Questions/Requests (a,b)	Responses (a,b)	Comments (a,b)
<u>INQUIRY 2 (March 9, 1987):</u> (contd)	<p data-bbox="955 300 1165 324"><u>RESPONSE:</u> (contd)</p> <p data-bbox="955 349 1533 495">10 - spare cells 362 - intact fuel assemblies with less than 5-year decay 217 - reserved for full core offload 688 - cells containing consolidated fuel at 2:1 (equivalent to 1376 intact fuel assemblies)</p> <p data-bbox="955 511 1533 641">3. Attachment 1 to the May 21, 1986 letter, the proposed TS 3.9.20 establishes the storage configuration requirements surrounding the presence or absence of the cell blocking device in Figure 3.9-2 of the submittal.</p> <p data-bbox="955 657 1533 828">If the cell blocking device is surrounded by consolidated fuel, it may be removed and a consolidated storage box placed in the cavity location. However, the cell blocking device's primary function is to prevent inadvertent usage of the blocked location when the area is occupied by intact spent fuel assemblies.</p> <p data-bbox="955 844 1533 1023">Statement (2) above indicated that 362 intact assemblies with less than a 5-year decay comprised part of the total capacity. Specifically, 157 intact assemblies would reside in Region I and 205 intact assemblies would reside in Region II, requiring 69 blocked locations to support the criticality bases for the assemblies in Region II.</p> <p data-bbox="955 1039 1533 1161">Therefore, in order for the total spent fuel pool capacity to support a total spent fuel assembly inventory of 1965, the total spent fuel rack capacity must be 1346 (i.e., 1277 plus 69), as stated in DESIGN FEATURE Section 5.6.3.</p>	

TABLE G.1. (contd)

Questions/Requests ^(a,b)	Responses ^(a,b)	Comments ^(a,b)
<p>QUESTION (June 30, 1987):⁽³⁹⁷⁾ Describe any future planned expansion of on-site spent fuel storage capacity (such as rod consolidation or dry storage) at Millstone Unit No. 2.</p>	<p>RESPONSE:⁽³⁹⁶⁾ In May 1986, a license amendment⁽¹⁵⁸⁾ was submitted to the NRC requesting authorization to store consolidated spent fuel in the Millstone Unit No. 2 spent fuel pool. On June 2, 1987, the NRC issued⁽³⁹⁵⁾ a partial approval of this amendment request which would allow the storage of five (5) consolidated storage canisters to support the fuel consolidation hot demonstration. The NRC is continuing to review the issues associated with the full-scale storage of consolidated fuel. When full-scale storage is approved it will provide the increased storage capacity required to "bridge the time" until the Department of Energy begins to accept fuel under the Nuclear Waste Policy Act.</p> <p>The implementation of fuel consolidation at Millstone Unit No. 2, total utilization of the existing spent fuel racks, together with improvements in fuel design for longer cycle life, can provide for "in pool" storage of all refueling discharges and maintain full core reserve space to the year 2015.</p>	<p>See submittals.^(158,396)</p>

TABLE G.1. (contd)

Questions/Requests (a,b)	Responses (a,b)	Comments (a,b)
<p><u>QUESTION 1 (October 30, 1987):</u> (400) The evaluation of August 11, 1987, indicates that a 3 x 3 temporary storage rack will be used for both intact and consolidated spent fuel. It is the NRC Staff's understanding that this rack was not included in the NRC Staff's review of consolidated spent fuel storage authorized in the recent technical specification change approval. Provide justification for use of this rack to safely store consolidated spent fuel and indicate if such an action will affect the existing technical specification. If so, propose the necessary technical specification change.</p>	<p><u>RESPONSE:</u> (399) The NRC Staff's understanding is correct with respect to the 3 x 3 temporary storage rack not being a part of the recent technical specification change for the storage of consolidated spent fuel at Millstone Unit No. 2.</p> <p>The 3 x 3 temporary storage rack is considered part of the consolidation system and equipment utilized to perform batch processing of spent fuel in the cask laydown area.</p> <p>WNECO does not intend to take credit for the additional storage locations in the 3 x 3 temporary rack as part of the overall authorized limit still under review by the NRC staff.</p> <p>The 3 x 3 temporary storage rack was analyzed by the same methods as the spent fuel racks in the spent fuel pool for seismic/structural, criticality and thermal hydraulic considerations.</p> <p>Additionally, the materials and fabrication of the 3 x 3 temporary storage rack conforms to the requirements of a Quality Assurance Category 1, Seismic Category 1 structure. The 3 x 3 temporary rack is classified as an ANS Safety Class 3 component.</p>	<p>See submittals. (157,398)</p>

TABLE G.1. (contd)

Questions/Requests (a,b)	Responses (a,b)	Comments (a,b)
<p><u>QUESTION 2 (October 30, 1987):</u>⁽⁴⁰⁰⁾ According to the evaluation, the existing spent fuel pool platform crane will be used for moving consolidated fuel storage boxes from the cask laydown area to the spent fuel pool. As the weight of a loaded consolidated fuel box is greater than the design capacity of the crane, explain in detail how the crane will be requalified for this heavy loads handling procedure.</p>	<p><u>RESPONSE:</u>⁽³⁹⁹⁾ An evaluation was performed which addressed the use of the existing spent fuel platform crane for the purposes of moving the consolidated fuel storage boxes from the Millstone Unit No. 2 Cask Laydown Area to the spent fuel pool. The spent fuel pool platform crane procurement, design, and testing documentation was reviewed. It was determined that excess design margins of safety existed on the load bearing components of the crane which enabled the consolidated fuel storage boxes to be safely handled.</p>	See submittals. ^(398,399)
<p><u>QUESTION 3 (October 30, 1987):</u>⁽⁴⁰⁰⁾ Provide more detailed information regarding the criticality sensitivity analysis performed to determine the minimum number of fuel rods that can be placed in the consolidated fuel storage box in order to meet the rack K_{eff} limit of 0.95. Consider both cases with and without the blocking device installed in the rack locations.</p>	<p><u>RESPONSE:</u>⁽³⁹⁹⁾ On October 3, 1986, NNECO submitted⁽³⁹¹⁾ a response to the NRC Staff question dealing with the reactivity effect of less than a full consolidated storage box (less than 352 rods). (Reference, Docket No. 50-336: B12275, A05935.)</p>	See submittal ⁽³⁹⁹⁾ for Figure 1.
<p>If for any reason the storage box cannot be loaded up to the minimum allowed content, indicate what actions you have considered to ensure that K_{eff} limits are maintained.</p>	<p>The response provided at that time stated that the consolidation process permits the placement of solid metal rods in positions where fuel rods are missing. For those instances where solid rods are not used, a limited number of fuel rods can be omitted based on the attached Figure 1. Using Figure 1, the reactivity effect of less than a full consolidated storage box can be established by determining the maximum number of fuel rods that can be omitted while maintaining K_{eff} at 0.95 or less.</p>	

TABLE G.1. (contd)

Questions/Requests (a,b)	Responses (a,b)	Comments (a,b)
<p><u>QUESTION 4 (October 30, 1987):</u>(400) The seismic analyses considered the maximum tipping displacement of the temporary and other storage racks. Verify that the analysis for partially loaded racks has considered the case where all the cells along one edge of the racks have been loaded with consolidated fuel boxes while the other cells are empty. Confirm that unacceptable tipping of the rack will not occur.</p> <p>In addition, provide information regarding the exact location of the feet of the racks with respect to the outermost storage cells.</p>	<p><u>RESPONSE:</u>(399) A seismic analysis has been performed that verifies that the maximum seismic tipping displacement of the 3 x 3 temporary storage rack under nonsymmetric loading conditions is acceptable. Specifically, one of the loading conditions considered was 3 consolidation storage boxes in an outside row while the other cells are empty. For this case, the factor of safety against overturning for the SSE was determined to be 18.</p> <p>The 3 x 3 temporary storage rack has a picture frame base welded to a flat bottom plate. The 3 x 3 temporary storage rack does not have any feet as is the case with the spent fuel racks that occupy the spent fuel pool.</p> <p>In addition, a seismic analysis has been performed that verifies that the maximum seismic tipping displacement of the spent fuel pool storage racks when loaded with consolidated spent fuel in the "worst case" scenario is acceptable. The details and specifics of this analysis was provided to the NRC Staff in Section 4 of our submittal dated May 21, 1986, "Storage of Consolidated Spent Fuel."(158)</p>	<p>See submittals.(158,399)</p>

TABLE G.1. (contd)

Questions/Requests ^(a,b)	Responses ^(a,b)	Comments ^(a,b)
<p><u>QUESTION 5 (October 30, 1987):</u>⁽⁴⁰⁰⁾ Provide more detailed information to ensure that after pulling a row of rods from a spent fuel assembly with the multiple rod pulling tool (MRPT) that no rod will slip or drop from the grip of the tool. If, for some reason (e.g., bent rod, swinging rod caused by forced cooling flow in the work area, etc.), the rods in the MRPT cannot be deposited in the interim transfer canister (ITC) channels, indicate what actions would be taken during the consolidation process.</p> <p>It is the NRC Staff's understanding from your evaluation that rods from one assembly will be deposited into the ITC and then those rods will be moved from the ITC to a consolidated fuel storage box. If so, provide more detailed information about the inner structure of the box which avoids movement or crossing of rods within the box when the rods from the second fuel assembly are inserted in the box. Describe the sequence of actions when loading the rods from the second assembly into the box.</p>	<p><u>RESPONSE:</u>⁽³⁹⁹⁾ Retention of the rods in the grippers is maintained by spring forces that hold the grippers closed in the gripped position without other assistance. These gripping forces are set and tested during final MRPT assembly prior to starting consolidation activities. Combustion Engineering (C-E) provided the results of this testing by letter MP2-87-039, dated February 26, 1987. Further, the rod slip detection system alarms and automatically stops all motion (hoist, X-Y table, etc.), if for some reason a slippage of a rod occurs.</p> <p>The MRPT moves up and down within a shroud which acts to contain and guide the fuel rods at all times. The grippers of the MRPT hold the upper ends of the rods in the original fuel assembly pitch. The fuel rod alignment bars at the bottom of the MRPT shroud maintain alignment of the rods in the original fuel assembly pitch during pulling, transfer by the X-Y table to the Interim Transfer Canister (ITC) and subsequent lowering onto the ITC. Since the alignment bars also act to gauge each rod for abnormalities including oversize diameter, assurance is provided that each rod withdrawn fully into the MRPT shroud will fit into the ITC. If for some other reason the rods cannot be inserted into the ITC, up to one full row of fourteen rods can be moved to the recovery/separation station and deposited there for evaluation and/or recovery action. Unacceptable rods can be separated from good rods there and deposited in the damaged rod station for eventual disposition back to spent fuel pool storage or off-site shipment.</p> <p>The ITC holds a full complement of 352 rods from two 14 x 14 fuel assemblies. All 352 rods are transferred at the same time to the consolidation storage canister. No storage canister inner structure is required with this design. A transferable floor in the bottom of the ITC holds the weight of the rods during the transfer and subsequently becomes the bottom of the consolidation canister.</p>	See submittal. ⁽³⁹⁹⁾

TABLE G.1. (contd)

Questions/Requests (a,b)	Responses (a,b)	Comments (a,b)
<p><u>QUESTION 6 (October 30, 1987):</u> (400) In Section 1.2 of the evaluation, permanent changes to the plant resulting from the fuel consolidation program are identified. Explain in more detail the following.</p> <p>(a) Repositioning of the brackets that support the level switch and temperature elements in the spent fuel pool.</p> <p>(b) The meaning of "the north 'sweep' plates on the spent fuel pool fuel handling machine must be partially coped out."</p>	<p><u>RESPONSE:</u> (399) The responses to (a) and (b) are as follows:</p> <p>a) The strap brackets that support the level switch and temperature elements for the spent fuel pool are located on the southeast intermediate wall elevation 38'6", between the cask laydown area and the spent fuel pool.</p> <p>The fuel consolidation hoist south rail/bearing plate is also located on the intermediate wall elevation 38'6".</p> <p>In order to avoid interference with the rail/bearing plate, the excess material on the strap brackets for the level switch and temperature elements were trimmed back at the ends. The modification did not diminish or alter the load capacity of the bracket supports.</p> <p>b) The "sweep" plates are a nonstructural part of the undercarriage axle assembly on the spent fuel platform crane. The "sweep" plates ensure removal of any inadvertent objects on the rail while the platform crane is in movement.</p> <p>To install the consolidation hoist north rail/bearing plate on the 38' 6" elevation of the cask laydown area without interference, it was necessary to partially cope out the "sweep" plate at the edge of the section.</p> <p>The "sweep" plate will continue to function as designed because the plate section directly above the platform crane rail was not affected by the modification. The modified section was at the edge of the "sweep" plate where the interference existed with the north rail/bearing plate.</p>	<p>See submittal. (399)</p>

TABLE G.1. (contd)

Questions/Requests (a,b)	Responses (a,b)	Comments (a,b)
<p><u>QUESTION 7 (October 30, 1987):</u> (400) In Section 7.2 of the evaluation, it is stated "fuel consolidation may be performed continuously...if the water temperature in the cask laydown area does not exceed 120°F." Provide information on the following:</p> <p>(a) How was this temperature limit established?</p> <p>(b) What is the temperature difference in cooling water between the lower inlet and upper outlet of the consolidated box in the work area?</p>	<p><u>RESPONSE:</u> (399) The responses to (a) and (b) are as follows:</p> <p>a) The temperature limit for the consolidation process in the cask laydown area was established by utilizing the maximum normal operating temperature of 120°F for the spent fuel pool, as defined in the Millstone Unit No. 2 FSAR, Section 9.5.3.1.</p> <p>b) The temperature difference in cooling water between the lower inlet and upper outlet of the consolidation storage canister in the work area is 18°F. The temporary storage rack is a 3 x 3 array of vertical cells capable of holding intact consolidated fuel for batch processing during the consolidation operation. At the completion of a consolidation cycle, the rack can contain up to five consolidation canisters, each full of rods from two fuel assemblies. The coolant temperature rise determination in the canisters is based on a CEPOOL computer case for storage modules with "mouse hole" coolant flow openings. The mouse holes are located at the bottom of each cell side wall to allow for coolant flow. The 18°F temperature rise is calculated for the center consolidation canister since for this case, the coolant must pass below the outermost canister in the rack to reach the center.</p> <p>Additionally, it should be noted that, on October 22, 1986, NNECO submitted a response (393) to the NRC Staff question dealing with the results of the CEPOOL calculations. (Reference, Docket No. 50-336: B12297, A06007.)</p>	<p>See submittals. (393,399)</p>

TABLE G.1. (contd)

Questions/Requests ^(a,b)	Responses ^(a,b)	Comments ^(a,b)
<p><u>QUESTION 8 (October 30, 1987):</u>⁽⁴⁰⁰⁾ Provide information concerning the visual or other inspections of the fuel assembly to be consolidated in order to assure that no damaged rods will be pulled from the assembly.</p>	<p><u>RESPONSE:</u>⁽³⁹⁹⁾ On October 28, 1986, WNECO submitted a response⁽²⁸⁾ to the NRC Staff question dealing with the method used for identifying fuel rods that are most likely to rupture. (Reference, Docket 50-336: B12303, A06101.)</p> <p>The response provided at that time stated that prior to consolidation, all candidate fuel assemblies will be screened and preinspected with the consolidation system TV cameras to review the condition of the exterior rows of fuel rods. Any fuel rods/assemblies previously known to have defects or found to have defects will not be consolidated. Plant fuel history data will be employed to identify known problems that may relate to existing rod defects.</p>	<p>See submittals.^(28,399)</p>
<p><u>QUESTION 9 (October 30, 1987):</u>⁽⁴⁰⁰⁾ Provide information on the plans to test or demonstrate proper operation of the consolidation equipment and procedures in their actual configuration prior to beginning the consolidation of spent fuel.</p>	<p><u>RESPONSE:</u>⁽³⁹⁹⁾ The consolidation system cold demonstration was conducted in December 1986 and the consolidation operating procedure qualification checkout was performed in February 1987, both with the system and equipment in their actual complete configuration. Additionally, prior to the Hot Demonstration, the on-site preoperational checkout of the system was performed with the equipment in its installed configuration in the cask laydown area at Millstone Unit No. 2.</p> <p>It should be noted that during the December 1986 Cold Demonstration, an NRC representative was present and issued an internal report (Project No. M-43) to Mr. L. C. Rouse on what was witnessed during the test.</p> <p>The NRC representative present at the test was Mr. James F. Schneider, Advanced Fuel and Spent Fuel Licensing Branch, Division of Fuel Cycle and Material Safety.</p>	<p>See submittal.⁽³⁹⁹⁾</p>

TABLE G.1. (contd)

Questions/Requests (a,b)	Responses (a,b)	Comments (a,b)
<p><u>QUESTION 10 (October 30, 1987):</u>⁽⁴⁰⁰⁾ After removing the upper end fitting and fuel rods from the spent fuel assembly, provide information on how the end fittings, grid cages, and control rod guide tubes are lifted to their respective work station. Describe in more detail the compacting of the grid cages and control rod guide tubes including related operator exposure radiological considerations and how the compacted radioactive items will be stored.</p>	<p><u>RESPONSE:</u>⁽³⁹⁹⁾ The upper end fitting is cut free of the grid cage/guide tube assembly at the beginning of the process. It is then transported to a storage canister by a pneumatically operated lifting tool which attaches to the center guide tube ID by means of an expanding collet. At the end of the process, the lower end fitting is similarly cut free from the grid cage/guide tube assembly and transported to the storage canister with a pneumatic collet tool. The grid cage/guide tube assembly is also gripped by the center guide tube ID with a pneumatic collet tool and transported to the compactor in-feed chute for compacting or may be returned to the temporary storage rack.</p> <p>A hydraulically powered compactor systematically shears sections of the grid cage/guide tube assembly and compacts each section in three directions to form briquettes. The briquettes are then placed into a storage canister for storage in the 3 x 3 temporary storage rack prior to being moved in the spent fuel pool for storage in the spent fuel racks.</p> <p>During the fuel consolidation operations at Millstone Unit No. 2, no significant change to the dose rates in the spent fuel pool area were experienced. Exposure levels of the consolidation operators and utility personnel were consistent with routine fuel handling, repair, and inspection operations of fuel assemblies.</p>	<p>See submittal.⁽³⁹⁹⁾</p>
<p><u>QUESTION 11 (October 30, 1987):</u>⁽⁴⁰⁰⁾ Concerning the filtration system, provide more detailed information on how the system is designed to avoid the dispersion of fuel pellets and other "heavy" debris in the pool should any rods break during the consolidation process.</p>	<p><u>RESPONSE:</u>⁽³⁹⁹⁾ The present description of the filtration system is in Section 2.1.3 of the 10CFR50.59 Safety Evaluation submitted⁽³⁹⁸⁾ to the NRC dated August 11, 1987 (Docket No. 50-336, B12563).</p> <p>It should be noted that each of the four work stations are fully shrouded such that any releases are contained and directly accessed by the filtration system.</p>	<p>See submittals.^(398,399)</p>

TABLE G.1. (contd)

Questions/Requests (a,b)	Responses (a,b)	Comments (a,b)
<p><u>QUESTION 12 (October 30, 1987).</u>(400) The process and equipment descriptions included in your evaluation appear to indicate that the planned rod consolidation involves new technology. Provide references to earlier consolidation tests or experience which support a conclusion of proven technology.</p>	<p><u>RESPONSE:</u>(399) At the present time, fuel rod consolidation is a proven technology that has been well demonstrated in the nuclear industry as a safe controlled operation.</p> <p>The engineering and tooling designs associated with the systematic handling of fuel rods are deeply rooted in the fuel reconstitution experiences that have been a proven technology for over ten years.</p> <p>Since 1982, the following fuel consolidation demonstrations have all been successful in establishing that spent nuclear fuel can be consolidated safely:</p> <p>Oconee/Westinghouse Ginna/U.S. Tool & Die Surry/EG&G Idaho Millstone/Combustion Engineering Prairie Island/Westinghouse</p> <p>Although the safe operation has been well demonstrated to date, rod consolidation is an evolving technology that will require more focus and attention in the future to improving and enhancing the time motion, efficiency, and reliability of the system.</p>	<p>See submittal.(399)</p>

- (a) Holman, J. P. 1972. Heat Transfer, 3rd Edition, McGraw-Hill Book Company, New York.
- (b) Counsil, W. G. March 30, 1984. Letter to J. R. Miller, "Millstone Nuclear Power Station, Unit No. 2 Spent Fuel Disposition Plans for Millstone Unit No.2."
- (c) Osborne, D. B., (NRC). June 4, 1984. Letter to Northeast Nuclear Energy Company, "Summary of Meeting with NNECO on Spent Fuel Disposition Plans for Millstone Unit No. 2."
- (d) BORAFLEX is a trademark of Bisco Products, Inc.

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