UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001
December 18, 1998

72-1015
72-1025
71-9235

MEMORANDUM TO: Susan F. Shankman, Deputy Director
Licensing and Inspection Directorate
Spent Fuel Project Office
Office of Nuclear Material Safety
and Safeguards

FROM: Timothy J. McGinty, Project Manager
Spent Fuel Licensing Section
Spent Fuel Project Office
Office of Nuclear Material Safety
and Safeguards

SUBJECT: SUMMARY OF NOVEMBER 19, 1998, MEETING WITH NAC INTERNATIONAL REGARDING THE NAC-UMS, NAC-MPC AND NAC-STC LICENSING APPLICATIONS (TAC NOS. L22511, L22447 & L22394)

On November 19, 1998, representatives of the Nuclear Regulatory Commission (NRC) and NAC International, Inc. (NAC) met to discuss NAC's Universal Multi-Purpose Canister System (UMS), Multi-Purpose Canister System (MPC), and Storage Transport Cask (STC) applications. The meeting was conducted in two sessions, with the morning portion focusing on the NRC's recent request for additional information (RAI) on the UMS storage design. The afternoon session covered the issues identified during the recent performance of NRC acceptance reviews of the MPC and STC applications. An attendance list for both sessions is included as Attachment 1. Attachment 2 is the overall meeting agenda and a list of the specific technical issues discussed pertaining to the UMS and MPC applications. Attachment 3 includes the handouts provided by NAC at the meeting. The NRC identified issues associated with the MPC and STC applications are included as Attachments 4 and 5, respectively. This meeting was noticed on November 5, 1998.

The UMS portion of the meeting commenced with a presentation by NAC of their planned response to NRC's RAI. NAC is focusing on the Dry Cask Storage System Standard Review Plan (SRP), and the recent issuance of the NRC's interim staff guidance (ISG). NAC committed to providing proposed Technical Specifications in accordance with the staff's improvement initiative. NAC also discussed the planned drop-tests to be performed for the UMS system in early 1999, and its plans to incorporate the results of those tests into the UMS transport application.

Furthermore, NAC discussed the formation of the NAC Nuclear Technology Users Group (NUTUG). NUTUG is chaired by Len Tremblay of Duke Engineering & Services (Yankee Atomic), with the remainder of the executive committee consisting of representatives from Arizona Public Services, Duke Energy, Maine Yankee, New York Power Authority and Virginia Power. The purpose of the NUTUG is to review current industry and regulatory activities in the area of spent nuclear fuel management to promote the collective positions that benefit the NAC system users as a whole.

NF07

NAC discussed the significant changes planned for the UMS application, including:

- designing and testing the canister to leak tight
- a revised analysis method for the vertical concrete cask tipover
- adding a minimum enrichment technical specification supported by analysis
- removing high burnup (> 45 GWd/MTU) fuels from the application
- removing the high seismic analysis from the application

The specific UMS RAI's that were discussed are listed in Attachment 2. The following are highlights of some of the key technical issues that were discussed:

- For RAI 2-16, pertaining to fuel rod pressurization calculations and the gas release fractions, the staff indicated that the current SRP guidance applies to calculations of pressure in the canister. As the RAI indicated, NAC needs to justify the gas release fractions used for the rod pressurization calculation.

- For RAI 4-5, which requested justification of the natural convection correlation utilized at the outer side surfaces of the concrete cask, the two parties discussed the merits of performing a sensitivity analysis to provide additional confidence in the thermal calculations.

- For criticality RAIs in Chapter 6, NAC informed the staff that for the appropriate calculations, the boiling water reactor (BWR) average planar enrichment value was being changed from 3.75 wt. percent to 4.0 wt. percent, per utility requests. Similarly, to address RAI 6-4, NAC intends to use the peak planar average to define the average assembly enrichment in the Technical Specifications.

- In response to RAI questions 6-6 and 6-7, NAC will analyze unchanneled BWR fuel assemblies, and intends to modify the pressurized water reactor (PWR) basket drawings to be consistent with the BWR basket drawings. The modified PWR basket will include closer tolerances and thus minimize the allowance for boral shifting in the basket tubes.

The second portion of the meeting convened in the afternoon, and commenced with a discussion of the results of the NRC acceptance review of NAC's October 1998 MPC RAI response. The issues that the NRC identified during the acceptance review are included as Attachment 4. Attachment 2 contains a listing of specific MPC issues discussed with NAC.

NAC's presentation on the MPC project included several commitments to address technical issues identified by the staff:

- The canister overpack will be removed from the application. Since the non-mechanistic failure of the canister is not required to be analyzed, this beyond design basis accident is unnecessary as a licensing consideration.

- The technical bases for allowable cladding temperatures for stainless steel clad fuel will be added to the MPC Safety Analysis Report (SAR).

- Detailed thermal calculations have already been provided to the staff in response to specific issues, and NAC has committed to a timely response if additional information is requested.

- Critical flaw size and a fracture mechanics evaluation of welds will be added to the MPC SAR to demonstrate reasonable assurance of structural integrity.

- The minimum enrichment and maximum decay heat will be included in the MPC SAR for the proposed contents.

The staff also presented specific technical issues for the NAC STC application, which are included as Attachment 5. Among the more significant issues identified are:

- A structural evaluation which demonstrates that the transportable storage canister can withstand the 200 meter immersion test, as a separate inner container, needs to be provided.

- A bounding thermal conductivity needs to be utilized in the three-dimensional thermal analysis.

- The application needs to fully address the effects of full, zero, and low moderator densities in the criticality evaluations.

During the course of both sessions, the staff re-iterated several aspects of the review process that are cornerstones to meeting the published schedules. These aspects are particularly pertinent to the NAC-MPC and STC applications at this juncture. As a result of the acceptance reviews, NAC needs to be very responsive to the issues identified by the staff. The established review schedule will result in a Safety Evaluation from the staff, however, any outstanding technical issues could, at a minimum, result in the need to establish restrictive conditions. The staff will use conference calls and public meetings to the extent necessary to resolve minor technical and licensing issues.

As identified in Attachments 4 and 5, the staff prioritized the issues identified in the MPC and STC acceptance reviews to aid NAC in responding accordingly. NAC committed, in a subsequent conference call held on November 23, 1998, to the following schedule for completing the responses:

<u>NAC-MPC</u>

- Category A issues will be completed by December 15, 1998.
- Category B issues will be completed by December 22, 1998.
- Category C issues will be completed by December 31, 1998.

<u>NAC-STC</u>

- Category A & B issues will be completed by December 22, 1998.
- Category C issues will be completed by December 31, 1998.

NAC committed to responding to the above issues with weekly submittals, with the first response received on November 30, 1998. NAC also committed to submit the finalized revised SAR pages, for both applications, by January 11, 1999.

No regulatory decisions were requested or made at this meeting.

Docket Nos.: 72-1015, 72-1025, 71-9235

Attachments:
1. Attendance List
2. Meeting Agenda
3. NAC Meeting Handouts
4. NAC-MPC Acceptance Review Issues
5. NAC-STC Acceptance Review Issues

Distribution:

Dockets	NRC File Center	PUBLIC	NMSS R/F	SFPO R/F	WKane
LKokajko	MWHodges	PEng	EEaston	SO'Connor	CFLyon, NRR
ENJensen, OGC	SGagner, OPA	NRC Attendees	VTharpe (3)		

OFC	SFPO	E	SFPO	E				
NAME	TMcGinty:dd		EJLeeds					
DATE	12/18/98		12/18/98					

C = COVER E = COVER & ENCLOSURE N = NO COPY

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November 19, 1998, Meeting
between NAC International
and Nuclear Regulatory Commission

ATTENDANCE LIST

Name	Affiliation
Tim McGinty	NRC/SFPO
William F. Kane	NRC/SFPO
Susan F. Shankman	NRC/SFPO
Fritz Sturz	NRC/SFPO
Elaine Keegan	NRC/SFPO
Kim Gruss	NRC/SFPO
David Tang	NRC/SFPO
Don Carlson	NRC/SFPO
Ron Parkhill	NRC/SFPO
Bill Lee	NAC International
Tom Thompson	NAC International
Mike Yaksh	NAC International
Alan Lin	NAC International
Steve Whitsett	NAC International
Curt Lindner	NAC International
Holger Pfiefer	NAC International
T.A. Bartman	Westinghouse
R.C. Bowser	Westinghouse
J.K. Thayer	Duke Engineering and Services
Len Tremblay	Yankee Atomic
John Rivera	Duke Engineering and Services
David Rivard	Maine Yankee
Joy Russell	Holtec International
Jim Doman	Booz-Allen - DOE
Wil Kenworthy	GSI
Tim Smith	GSI
Morris Schriem	Self
Sidney Crawford	Self
Keith Waldrop	Duke Energy - McGuire

Attachment 1

USNRC SFPO AND NAC INTERNATIONAL MEETING AGENDA

NOVEMBER 19, 1998

NAC-UMS REQUEST FOR ADDITIONAL INFORMATION (A.M.)

- Introductions
- Discussion of Request for Additional Information (RAI)
  - Specific RAIs will be identified for discussion
- Schedule for submittal of RAI responses

NAC-MPC AND NAC-STC ACCEPTANCE REVIEWS (P.M.)

- Introductions
- NAC-MPC RAI Responses and Acceptance Review Results
- NAC-STC RAI Responses and Acceptance Review Results
- Schedule Discussions
- Closing Remarks

ATTACHMENT 2

USNRC SFPO AND NAC INTERNATIONAL MEETING
November 19, 1998

NAC UMS™ Storage System RAIs and Related Concerns to be discussed

1-0 & 2-0 Minimum enrichment specification – NAC approach.

Removal of > 45 GWd/MTU burnup fuel.

2-16 Gas release fractions – NAC approach and references.

4-1 Heat transfer disk - aluminum allowable temperature limits.

4-2 PWR support disk – 17-4 PH stainless steel toughness following elevated temperature exposure.

4-5 Natural convection correlation – VCC outer side surfaces.

4-6 Canister 3-D thermal model gaps – fabrication tolerances/thermal expansion.

4-11 Pressure calculation assumptions – consideration of increased backfill temperature.

5-2 Activity inventory – applicability for "leak-tight" component.

6-0 BWR criticality calculations redone at higher enrichment.

6-5 BWR fuel models – heterogeneous versus homogeneous enrichment studies.

6-6 & 6-7 Effects of mechanical perturbations – shift of poison sheets, unchanneled assemblies, relative thermal expansion of sheets, plates, and tubes.

6-11 BWR fuel design information – Exxon/ANF fuel characteristics.

10-2 SKYSHINE III Code verification – clarify request; NAC QA program invoked.

12-4 Canister surface contamination – ALARA justification based on NAC assumptions.

USNRC SFPO AND NAC INTERNATIONAL MEETING
November 19, 1998

<u>NAC-MPC System Acceptance Review Comments and Related Concerns to be discussed</u>

Nuclear

Similar concerns as NAC UMS™ Storage System RAI comments.
Further discussion as appropriate.

Structural/Thermal

Similar concerns as NAC UMS™ Storage System RAI comments.
Further discussion as appropriate.

General	Removal of Canister Overpack per ISG-3 guidance.
Chapter 2	Stainless steel clad fuel – Justification/technical basis for allowable temperature limits.
Chapter 4	Thermal calculation packages (Potentially Proprietary Discussion)
Chapter 8	Canister closure weld – structural integrity, flaw detection, weld examination.
Chapter 11	Seismic evaluation – margin of 1.1 against overturning and sliding.
	Cask tipover - NAC evaluation results (single basket orientation) and methodology benchmarking.

ATTACHMENT 3

NAC International
UMS/MPC Licensing Status Overview

Presentation to U.S. Nuclear Regulatory Commission

Edward M. Davis
President & CEO
NAC International

November 19, 1998

NAC INTERNATIONAL

UMS/MPC Licensing Status Overview

- On-time NRC delivery (October 30, 1998) of the Universal MPC System (UMS) Request for Additional Information (RAI) welcome
- Rapid acceptance and NRC review of NAC-MPC RAI submittals appreciated
- Our review of both the UMS RAI and NRC-identified MPC open issues finds no insurmountable questions or significant new issues that *can not* be addressed in the next 30 to 60 days
- NAC has requested the active involvement of our Nuclear Technology Users Group executive committee team members and we anticipate their full participation
- In short, we are both prepared and fully committed to meet our schedule commitments for deliverables under the agreed-to "Rules of Engagement." This includes . . .

NAC INTERNATIONAL

UMS/MPC Licensing Status Overview (continued)

NAC-MPC

- Interactive, timely, high-quality and responsive resolution of remaining open NAC-MPC issues
- We understand these to include stainless steel clad fuel allowable temperatures; thermal calculations; canister closure welds; revised specification for fuel to be stored; and technical specification format
- Complete thermal calculation packages were provided to the NRC within less than 48 hours of its initial request on November 11
- Ongoing constructive engagement was also underway with our respective staffs within the same time period
- Both actions were indicative of our commitment to meeting your request for timely, quality responses on an urgent basis
- Goal is a preliminary SER and draft Certificate of Compliance for both storage and transport not later than March 1, 1999

UMS/MPC Licensing Status Overview (continued)

UMS

- Particular focus on NRC's latest guidance from the NUREG-1536 standard review plan and NRC's Interim Staff Guidance
- NAC's response will include submittal of the proposed Technical Specifications in the "improved format and content standard" for tech specs
- Goal is eliminating the need for a second round of RAIs, thereby ensuring that the UMS will receive NRC preliminary SER and draft COC for storage in June 1999 as scheduled

UMS/MPC Licensing Status Overview (continued)

- The challenge of meeting NRC requests for information and addressing open issues related to the licensing of two major spent fuel technologies will be met
- A UMS drop-test is also scheduled early next year and this is proceeding on track to allow us to incorporate the results into our transport RAIs to expedite that review
- All of the above would not be possible without the NRC's new business management paradigm
- We commend the NRC Spent Fuel Project staff for meeting its schedule commitments and for its continued hard work to meet the needs of U.S. utilities and technology companies

NAC Nuclear Technology Users Group (NUTUG)

- Users group for the NAC MPC and UMS systems established
- Chaired by Len Tremblay of Duke Engineering & Services
- Executive Committee membership currently consists of
  - Arizona Public Services
  - Duke Energy
  - Duke Engineering & Services (representing Yankee Atomic Electric Company)
  - Maine Yankee
  - New York Power Authority
  - Virginia Power

NAC INTERNATIONAL

NUTUG Charter

- Charter Mission Statement:

The purpose of the NAC Nuclear Technology Users Group (NUTUG) is to review current industry and regulatory activities in the area of spent nuclear fuel management and to promote a proactive, collective position that benefits NAC Nuclear Technology Users as a whole. The NUTUG will provide a form for sharing operating and licensing experiences among users. The NUTUG may also issue position statements, consult with/inform regulators of issues, and provide similar information to nuclear industry organizations. The NUTUG may sponsor studies or undertake other activities, which are deemed mutually beneficial to the group.

MPC Open Issues

- **Canister Overpack** – Will be removed from the application
- **Stainless Steel Clad Fuel Allowable Temperatures** – Technical bases for allowable temperatures will be added to MPC SAR
- **Thermal Calculations** – Have been sent to the NRC. NAC provided these to the NRC less than 48 hours after the request
- **Canister Closure Welds** – Critical flaw size and fracture mechanics evaluation of welds will be added to the MPC SAR to demonstrate reasonable assurance of structural integrity
- **Minimum Enrichment & Maximum Decay Heat for Fuel** – Will be added to the MPC SAR

NAC UMS, MPC & STC Review Schedules

ACTION	DATE	UMS - Storage	UMS Transport	MPC - Storage	STC - Transport
NAC COMPLETE	10/09/98			Response to 01/27/98 RAI 1	Response to 12/30/97 RAI 1
NRC COMPLETE	10/30/98	Issue RAI 1			
NAC IN PROCESS	01/29/99	Response to RAI 1			
NRC IN PROCESS	03/01/99			Issue Draft SER & CoC Commence Rulemaking	Issue Amendment
NRC	06/01/99	Issue Draft SER & CoC or Issue RAI 2			
NAC	08/13/99	Response to RAI 2 (if applicable)			
NRC	08/30/99		Issue RAI 1		
NRC	11/01/99	Issue Draft SER & CoC (if RAI 2 applicable)			
NAC	12/01/99		Response to RAI 1		

NAC INTERNATIONAL

UMS Issues

- Significant Changes
  - Leak-tight canister (ANSI N14.5-1997)
  - Analysis method of VCC tipover revised (LS-DYNA, NUREG/CR-6608)
  - Standard technical specification format will be adopted
  - Minimum enrichment will be added to fuel specification
  - All fuel >45GWd/MTU burnup will be removed
  - High Seismic Analysis removed
- Areas of discussion
  - Fission gas release fractions
  - Material properties and allowable limits
  - Thermal modeling techniques
  - Factors affecting criticality calculations

MPC Open Issues and Closure Implementation

- NAC, with its utility partners, is anxious to bring all these issues to closure with the NRC as rapidly as possible.
  - To that end, we will provide these responses on a regular basis as soon as they have been prepared and checked, and have received our thorough QA review.
  - We have begun this process already with the submittal of our thermal calculation packages.
  - NAC is committed to provide all responses to the NRC before the end of December 1998 so as to enable it to meet its published schedule.

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Summary of
"Evaluation of Expected Behavior of
LWR Stainless Steel – Clad
In Long Term Dry Storage"
(EPRI TR-106440)

Report Objectives

- Provide a technical basis for continued long term wet and/or dry storage of stainless steel–clad spent nuclear fuel (SS-SNF).
- Compiled/reviewed data and experience relevant to the behavior of SS–SNF in long term storage.
- Assess and prioritize potential failure modes.
- Provide conclusion and recommendations.

Conclusion

- Corrosion of SS-clad is less than 5% in a pool environment and expected to be negligible in an inert dry storage environment.
- Pitting is not a significant source of SS-clad degradation in the pool.
- Stress corrosion crack (SCC) of nonsensitized SS-SNF is not expected in wet or dry storage.
- Stress rupture on SS-SNF should not occur below 430 °C (806 °F).
- Helium embrittlement was factored into the creep rupture modeling.
- Effects of stress rupture for fission products are considered to be negligible.
- Hydrogen induced degradation is expected to be negligible during long-term storage.
- SS-SNF fuel rods with incipient cladding defects would act the same as Zircaloy–clad SNF.
- Low temperature sensitization will not result in cladding degradation in an inert environment.
- Supplemental storage package shielding may be needed due to radiation levels from Cobalt-60.
- SS-SNF has generally cooled longer than Zry-SS and SS-SNF has a higher cladding temperature limit, therefore typical LWR dry storage system should be able to accommodate SS-SNF.

Stainless Steel Fuel Inventory Profile

- Five plants (4 PWR, 1 BWR)
  - Haddem Neck
  - Indian Point - 1
  - La Crosse
  - San Onofre - 1
  - Yankee Rowe
- Stainless Steel Alloys used as clad
  - 304
  - 348H (modified)
  - 348
- Fuel Characteristics (Burnup Ranges, Discharge Dates)
  - Haddem Neck 8.2 to 38.0 MWD/KgM Discharges: 4/70 – 10/91
  - Indian Point - 1 4.7 to 25.2 MWD/KgM Discharges: 4/73 – 1/26
  - La Crosse 5.0 to 21.0 MWD/KgM Discharges: 4/72 – 4/87
  - San Onofre - 1 6.8 to 39.3 MWD/KgM Discharges: 10/70 – 11/92
  - Yankee Rowe 25.9 to 31.6 MWD/KgM Discharges: 2/72 – 10/75

NAC-MPC: RAI Response Acceptance Review Issues

Chapter 1 General Information

1. UMS Question 1-1. B

2. UMS Question 1-2. C

3. UMS Question 1-3. C

   Decay heat values in Table 1.2-6 may not be bounding since it is not apparent how minimum enrichments were utilized in determining fuel assembly heat loads.

4. Figures 1.2-6 & 7 show two pairs of transfer cask lifting trunnions. C

5. UMS Question 1-9. C

6. UMS Question 1-10. C

7. UMS Question 1-12 for Drawing 455-860. B

Chapter 2 Principal Design Criteria

1. UMS Question 2-1. B

2. Confirm number of Westinghouse fuel assemblies to be loaded is 34 and not 35 (see Section 2.1.1). C

3. UMS Question 2-3, applicable to Tables 2.1-1, 2.1-2 and 6.2-2. B

4. Revise Section 2.1.2 to include detailed specifications that clearly define the reconfigured fuel assemblies that may be stored. Specifications should be provided for the fuel rods, encapsulating rods and stainless steel container, as appropriate. Related to UMS Question 2-1. B

5. UMS Question 2-8. C

6. UMS Question 2-10. B

7. Provide the technical basis for the safe storage and retrievability of the Yankee Class stainless steel clad fuel. A

   This can be done by demonstrating that under normal, off-normal, and hypothetical accident conditions either (1) the temperature limits for Zircaloy clad fuel bound the temperature limits for stainless steel clad fuel, or, (2) by other analyses and/or calculation, that the fuel will not degrade significantly under dry cask storage conditions for the license period. To the extent practical, the analysis should consider the effects of manufacturing practices (e.g., welding, heat treating, etc.) and unusual in-reactor service conditions (e.g., power excursions, significant changes in water chemistry, etc.)

ATTACHMENT 4

on the long-term integrity of the stainless steel cladding in a dry storage cask environment. The potential for cladding failure by the following mechanisms should also be addressed as appropriate: creep rupture, sensitization, corrosion, cracking.

8. UMS Question 2-12. C

Chapter 3 Structural

1. UMS Question 3-11. B

2. UMS Question 3-12. Applicable only to Tables 3.4.3.3-1 and -2. B

3. UMS Question 3-13. B

4. UMS Question 3-14. A

5. UMS Question 3-15. A

6. UMS Question 3-28, applicable to Drawing 455-866. B

7. Page 3.4-21: Revise the equation used for calculating the tensile stress area of the bolt. The equation as shown would not render the calculated tensile area of 1.405 in^2. B

8. Figure 3.4.4.1-4: Revise, as appropriate, the listed Y1 coordinate for stress evaluation Location No. 11 to be consistent with that in SAR Figure 11.2.1-1. C

9. UMS Question 3-7. A

Chapter 4 Thermal

1. Provide an explanation as to how radiation was modeled for the canister between adjacent fuel tubes. A response similar in detail to that provided in NAC's 10/8/98 response to MPC RAI # 4-8 would be appropriate. B

2. Justify the short term and long term temperature limits for the stainless steel cladding.

   Table 4.1-4 identifies the cladding temperature limits of 644F and 1058F for long term and short term conditions, respectively. The long term limit coincides with the lower limit for the stainless steel materials shown in Table 4.1-4, but the short term limit exceeds the allowable shown for stainless steel by 250F. Also, the short term limit shown is applicable to zircaloy. A

3. UMS Question 4-3. A

4. UMS Question 4-4, applicable to Table 4.2-4 and for temperatures up to 700 degrees F. B

5. UMS Question 4-5, applicable to Page 4.4-4. A

6. UMS Question 4-7. B

7. UMS Question 4-8, different MPC time limits applicable. A

8. UMS Question 4-9. A

9. Justify the fuel rod backfill pressure of 315 psig for CE fuel used in the determination of maximum internal pressure for normal conditions (SAR Section 4.4.5) and accident pressurization (SAR Section 11.2.1). Also, justify the initial fill gas pressures for the other fuel types shown in Table2.1-1.

   Review of DOE/RW-0184 "Characteristics of Spent Fuel, High-Level Waste, and other Radioactive Wastes Which May Require Long-Term Isolation", Dated 12/87 shows that Combustion Engineering fuels have an initial gas pressure between 300 psig and 450 psig. If no other information is available the higher value of initial gas pressure should be utilized. B

10. UMS Question 4-2. A

Chapter 5 Shielding

1. On page 5.1-3 of the SAR, NAC indicates that a modified version of SAS4 was used to perform a three-dimensional shielding analysis. The SAR does not explain the modifications to the SAS4 code. Any changes to the code need to be documented. The verification and validation process for the code revision needs to be described. B

2. UMS Question 5-1, if applicable. A

3. UMS Question 5-4. MPC RAI Question 5-4 requested the SAS4 shielding calculations that support the dose rates for the canistered fuel. NAC's October 8, 1998, response provided a summary of the shielding analysis. In order to evaluate the summary information provided, submit the SAS1, SAS-2H and SAS4 input decks for the design basis fuel. A

Chapter 6 Criticality

1. UMS Question 6-1(c). A

2. UMS Question 6-2. B

3. UMS Question 6-3. B

4. UMS Question 6-7. B

5. UMS Question 6-8. C

6. UMS Question 6-10. A

7. UMS Question 6-11. A

The applicant does not adequately reference the sources of fuel design information. The staff has design information only on the CE Yankee Class fuel, which is not the bounding fuel type. DOE/RW-0184 lacks information on the Yankee Class fuels of Exxon, Westinghouse, and United Nuclear.

Chapter 7 Confinement

1. UMS Question 7-1. A

2. UMS Question 7-2. B

3. Correct the inconsistency between Section 7.1.3.2, Weld Specifications, and Section 9.1.1.1, where the latter calls for a progressive PT examination and the former just refers to a root and final pass PT examination of the structural lid to shell weld.

   Interim Staff Guidance - 4 states the closure weld may be examined using either volumetric or multiple pass dye penetrant. C

Chapter 8 Operating Procedures

1. UMS Question 8-3. B

2. UMS Question 8-7. B

3. UMS Question 8-8. B

Chapter 9 Acceptance Tests and Maintenance Program

1. UMS Question 9-1. B

2. UMS Question 9-2. B

3. Provide an explanation as to why a progressive PT examination is required every 1/4 inch of weld for the vent/drain port lid welds (refer to Section 9.1.1.1 & Drawing 455-857) and the weld size is only 1/4 inch as shown on Drawing 455-857.

   The aforementioned SAR section and drawing need to delete mention of a progressive PT examination for the port lid welds to reflect the size of the weld specified. C

4. Provide a basis for the spacing between progressive PT examinations, referred to in Section 9.1.1.1, to demonstrate that the critical crack size could be detected. Also, submit the associated fracture mechanics calculation. The critical fracture mechanics calculation should be consistent with ASME Section XI methodology.

Interim Staff Guidance - 4 states the minimum detectable flaw size must be demonstrated to be less than the critical flaw size. A

Chapter 10 Radiation Protection

1. SAR needs to be revised to demonstrate compliance with 72.104(a). In the SAR, the required minimum distance to the controlled area boundary in specified but no calculations. In response to question 10-5, NAC submitted a proprietary response to support the Skyshine III evaluation for off-site doses. A non-proprietary version describing the analysis should be included in the SAR. B

Chapter 11 Accident Analyses

1. The design basis accident for the MPC is not identified. An accident evaluation needs to be performed with the guidance from Interim Staff Guidance -5 for normal, off-normal and hypothetical accident dose estimate calculations for the whole body, thyroid, and skin. NAC does not demonstrate compliance with 72.106(b). This needs to be another RAI question or the subject of a meeting. B

2. UMS Question 11-6. A

3. UMS Question 11-8. A

4. UMS Question 11-13. A

5. Provide additional details, as appropriate, on the tipover handling accident analysis to aid in Staff reviewing.

   The SAR provides only summary descriptions and results. Additional analysis details such as the analytical modeling of the VCC and assumptions for storage pad back fills (SAR Section 12.2.2.10) may be needed for Staff review. B

6. Provide justification for analyzing only the 45° support disk orientation, as bounding, for the VCC subject to the side impact in an tipover accident.

   The SAR discussion of the subject is incomplete. The stress results are drop orientation dependent. Other support disk loading orientations should also be examined to ensure that the bounding structural performance of the ligaments has been evaluated. B

7. Section 11.3.2: The referenced Section 11.2.11 does not provide, as stated in the first paragraph, SAR Page 11.3-17, the analysis results for the basket subject to tipover side impact condition. C

8. UMS Question 11-1, applicable for a 500 degrees F assumption as stated in SAR Section 11.2.1.1.1. A

9. UMS Question 11-5. A

Chapter 12 Conditions for Cask Use

1. UMS Question 12-1. B

2. Provide, as previously requested, the calculation used to determine the 20 hour time limit for removal of the canister from the spent fuel pool and completion of draining of water from the canister. A

3. Provide the calculation which demonstrates that the cladding integrity is maintained during reflooding of the canister using the minimum quench fluid temperature and maximum flowrate. Also, explain why the pressure in the canister during reflooding has been calculated to be below 50 psig (as stated in your response) when then canister is only evaluated for an accident pressure of 35 psig (as stated in SAR Section 11.2.1.2.2, Maximum Canister Stress Due to Internal Pressure). In addition, what controls are in place to ensure that the 50 psig pressure limit (as stated in your response) is not exceeded during reflooding operations.

   Even though the original RAI did not explicitly request the subject calculation to be submitted, the staff was expecting an analytical justification as part of the response since the requested limits were to be supported by analysis. A

4. NAC's response to MPC RAI question 12-1(b) does not appear to justify the storage of damaged fuel rods placed in reconfigured assemblies with respect to ISG-5 dose estimate calculations for normal, off-normal and hypothetical accident conditions. B

5. Surface contamination limit of the canister surface is based upon an assumed limit of 1 mrem per year at 100 meters. Explain how the limits are ALARA. B

6. UMS Question 12-5. B

NAC-STC: RAI Response Acceptance Review Issues

Chapter 1 General Information

1. Specify the minimum initial enrichment of Yankee Class Fuels in Table 1.2-2. C

Chapter 2 Structural

1. Perform a structural evaluation of the MPC-Yankee configuration to demonstrate that the canister, as a separate inner container per 10 CFR 71.63(b), can withstand an external water pressure of 290 psi without canister collapse, buckling, or water inleakage, in accordance with 10 CFR 71.61; clarify the SAR text, as appropriate, to clearly describe the intended use of the canister as a separate inner container.

   Basis. As a separate inner container for Reconfigured Fuel Assemblies, per 71.63(b), the canister must be designed to the requirements of 10 CFR 71.61. The NAC response to Q 1-2 states, "[T]he canister is demonstrated to maintain the containment boundary for all normal conditions of transport and hypothetical conditions..." While SAR Page 1.1-1 states, "[T]he transportable storage canister provides the secondary containment boundary for the transport of Reconfigured Fuel Assembly...," Page 2.1.1-4 provides, however, conflicting description, "[N]o containment credit is taken for the TSC when the NAC-STC is in the transport mode of operation." A

2. Section 2.6.7.5.4: Considering the primary membrane stress intensity limit, $2S_m$, as the basis, reevaluate cask lid bolts and revise stress summaries, in Tables 2.6.7.5-1 thru -4, for normal condition of transport.

   The $3S_m$ stress allowable considered for normal condition of transport is incorrect use of the ASME Section III, Appendix F, Paragraph F-1335 standard which is for the accident, Service Level D, condition. Also, since the bolt stress calculation approaches, as described in Section 2.10.8, are all to result in membrane, in lieu of membrane-plus-bending, stresses in the bolts, the stress intensity limit of $2S_m$, Table 2.1.2-1, should be considered for lid bolt stress evaluation. B

3. Revise the typos of SAR Table 1.2-1 and Section 2.7.9 to assure that the dimensions for end weldment, support disks, and heat transfer disks are consistently and correctly reported.

   As discussed in Q 2-3 of STC RAI #1, the dimension accuracy is a critical safety review consideration for the support disk and aluminum heat transfer disk. Section 2.7.9 describes the support disk to be 69.15" in diameter which is inconsistent with the listed support disk and end weldment dimensions of 0.5 x 68.98 dia. and 0.5 x 69.15 dia, respectively. C

4. UMS Question 1-12 for Drawing 455-860. B

ATTACHMENT 5

Chapter 3 Thermal

1. Provide the basis for the fuel cladding temperature limits identified in Section 3.4 (i.e.716F for Normal Conditions of Transport) and Section 3.5 (i.e. 1200F for Hypothetical Accident Conditions (HAC)) as they apply to stainless steel clad fuel. Also, justify use of PNL-4555 as the basis for the maximum zircaloy fuel cladding temperature limit of 1200F for HAC or use the value of 1058F from PNL-4835.

   PNL-4555 authored by Guenther, demonstrates a correlation between heating rates of unirradiated PWR fuel rods and cladding deformation. However, it does not appear to justify the fuel cladding temperature limit for irradiated zircaloy clad fuel. B

2. Re-analyze the 3-D canister model utilizing a thermal conductivity more representative of the materials in contact and submit the results. Use of an artificially high thermal conductivity (i.e. 100 BTU/hr-in-F) does not conservatively bound the problem being analyzed and may contribute to a heat rejection path that is over estimated. The staff feels that a thermal conductivity consistent with the materials in contact would be sufficient. A

3. Correct the minor inconsistency in the first paragraph of Section 3.1.2, Canistered Fuel, which states that the design basis fuel assembly has a heat load of 345 watts or .347 kw. Also, Table 1.2-2 indicates that 347 watts is the design heat load. C

4. Confirm and correct that the four component maximum temperature values shown onTable3.5-1 are too high by an order of magnitude. C

5. Clarify whether the decay heat value of 12.5 kw for canistered fuel was used in the determination of temperatures shown in Table 3.5-1.

   Section 3.5.1.1 states that the decay heat value of 22.1 kw was used in the accident thermal model. However, the subject table implies by listing the lower decay heat value for canistered fuel that it was used in the accident analysis. C

Chapter 5 Shielding

1. Specify the minimum initial of Yankee Class fuels and determine if the source term and dose rates remain within the bounds of the design basis fuel. B

2. NAC's response to STC RAI Question 5-11(c) does not appear to address the contributions of crud and dross materials to the source specification.

   The extent of the Appendix A RAI analysis should be the subject of meeting discussions. A

Chapter 6 Criticality

1. NAC Response to RAI 6-1: The revised infinite-array calculations do not consider the case with full moderator density inside the packages and zero, or low, moderator density outside the packages. The staff notes that, in general, this neglected case is slightly

more reactive than the reported "optimum" case of full moderator density inside and outside the packages.

This topic should be the subject of meeting discussions. C

2. NAC Response to RAI 6-2: The revised SAR includes engineering drawings and a criticality analysis of the Reconfigured Fuel Assembly. The applicant's analysis assumes that all fuel pellet material remains within the reconfigured fuel rods under normal and accident conditions. Such geometrically intact reconfigured assemblies are shown to be much less reactive than normal fuel assemblies.

   It is not clear to the staff that failed fuel rods will remain geometrically intact under accident conditions. However, in view of the much lower fuel mass in the reconfigured assembly, the staff expects that all credible redistributions of fuel pellet material will still prove significantly less reactive than the limiting cases for normal assemblies.

   The applicant should provide a stronger basis for assuming that reconfigured failed rods remain geometrically intact under accident conditions, or else include a scoping criticality analysis of credible fuel redistributions within the reconfigured assembly. B

3. NAC Response to 6-11: First-round UMS RAI 6-10 is closely related. The trending analysis rejects trends that would be considered under the approved methodology in NUREG/CR-6361.

   This topic should be the subject of meeting discussions. B

4. UMS Question 6-1(c), applicable to the Yankee Class contents. A

5. UMS Question 6-2, with respect to Yankee Class contents. B

6. UMS Question 6-3: This topic should be the subject of meeting discussions. B

7. UMS Question 6-7: This topic should be the subject of meeting discussions. B

8. UMS Question 6-8: This topic should be the subject of meeting discussions. B

9. UMS Question 6-10: This topic should be the subject of meeting discussions. B

10. UMS Question 6-11: This topic should be the subject of meeting discussions. B

11. In describing the criticality analysis models, the SAR repeatedly states that the modeled B-10 content of the boral plates is 75% of the "nominal" content. To avoid confusion, this should be changed to "specified minimum" content.

    The staff notes that, in order to ensure a specified minimum B-10 content, the boral may be fabricated to a somewhat higher so-called "nominal" content specification. C

Chapter 7 Operating Procedures

1. NAC does not adequately describe the loading procedure for canistered fuel configuration. In Section 1.1, NAC indicated that, for the canistered configuration, two spacers of different sizes are used to locate the canister in the cask cavity to maintain the same center of gravity as that for the uncanistered fuel cask. However, in Section 7.1.3.2, there are no steps in the procedure to ensure the proper STC loading sequence of spacers and the canister. Additional steps in the loading procedure of Section 7.1.3.2 are needed to ensure that the cask is used within the design specifications. C