71-9270

NAC INTERNATIONAL

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September 28, 2004

U.S. Nuclear Regulatory Commission
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Attn: Document Control Desk

Subject: Submittal of NAC International Responses to the U.S. Nuclear Regulatory Commission Request for Additional Information for a Requested Amendment of Certificate of Compliance No. 9270 for the UMS® Universal Transport Cask to Incorporate Various Update Information (TAC No. L23726)

Docket No. 71-9270

Reference:

1. Model No. UMS Universal Transport Cask Package, Certificate of Compliance No. 9270, Revision 0, U.S. Nuclear Regulatory Commission, October 31, 2002
2. Safety Analysis Report for the NAC-UMS® Universal Transport Cask, Revision 1, NAC International, December 2002
3. Request for an Amendment of Certificate of Compliance No. 9270 for the NAC-UMS® Universal Transport Cask to Incorporate Various Update Information, NAC International, March 31, 2004
4. Submittal of Supplemental Information to Request for an Amendment of Certificate of Compliance No. 9270 for the NAC-UMS® Universal Transport Cask to Incorporate Various Update Information, NAC International, June 11, 2004
5. Certificate of Compliance No. 9270 for the Model No. NAC-UMS Package, Request for Additional Information, U.S. Nuclear Regulatory Commission, July 8, 2004

NAC International (NAC) herewith submits eight copies of the Responses to Reference 5, the U.S. Nuclear Regulatory Commission (NRC) Request for Additional Information (RAI).

This submittal includes the RAI comments with the NAC responses presented in the standard NAC RAI response format, followed by the associated NAC-UMS® Universal Transport Cask Safety Analysis Report (SAR) changed pages, which are designated as Revision UMST-04B. Four NAC drawings have been updated and are included in Chapter 1 of the SAR. The enclosed SAR changed pages (UMST-04B) and four revised drawings (refer to the list of drawing changes in Attachment 1) are to be inserted as replacement or new additional pages, as applicable, into the Reference 3 amendment request. After replacement/insertion of the changed pages provided in this submittal, the current List of Effective Pages should be used to ensure that the correct page revisions are included in the compiled amendment request package. Note that a List of Effective Pages is provided for each volume of the SAR.

ED20040081

UMSS01

Consistent with NAC administrative practice, all SAR pages changed in this submittal are uniquely identified as Revision UMST-04B, and a revision bar marks each change on a page. Text flow changes are not marked with revision bars. Upon final approval, the changed pages will be reformatted, assigned the next appropriate revision number, and incorporated into the NAC-UMS® Universal Transport Cask SAR.

A significant revision incorporated in this RAI response submittal is NAC's decision to remove the METAMIC neutron absorber material from the requested amendment. Based on current contractual commitments and ongoing fabrication experience, NAC has determined that METAMIC will not be used in the NAC-UMS® fuel baskets in the foreseeable future.

NAC hereby requests approval of this amendment request by December 1, 2004, to support NAC contractual commitments to update the NAC-UMS® Transport Cask Certificate of Compliance (CoC) to incorporate changes made to the system design since Revision 0 of the CoC was issued.

If you have any comments or questions, please contact me on my direct line at (678) 328-1321.

Sincerely,

TC Thompson

Thomas C. Thompson
Director, Licensing
Engineering

Attachment 1, List of Drawing Changes

Enclosure

CC: John Niles (MY) w/o Enclosure
Glenn Michael (APS) w/o Enclosure
Keith Waldrop (Duke) w/o Enclosure

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U.S. Nuclear Regulatory Commission
September 28, 2004

ATTACHMENT 1

LIST OF DRAWING CHANGES

NAC-UMS® TRANSPORT SAR, REV. UMST-04B

ED20040081

LIST OF DRAWING CHANGES

Drawing 790-575, Revision 10 — BWR Fuel Tube, NAC-UMS®

- Change BOM, Items 3 & 4, material callout: IS) Boral; WAS) Boral/Metamic

The proposed change has no effect on the design function or the design of the fuel tube components as described in the SAR. The existing supporting SAR analyses currently reflect the Boral material properties.

Drawing 790-581, Revision 9 – PWR Fuel Tube, NAC-UMS®

- Change BOM, Items 4, 5 & 6, material callout: IS) Boral; WAS) Boral/Metamic

The proposed change has no effect on the design function or the design of the fuel tube components as described in the SAR. The existing supporting SAR analyses currently reflect the Boral material properties.

Drawing 790-605, Revision 11 — BWR Fuel Tube, Over-Sized Fuel, NAC-UMS®

- Change BOM, Items 3 & 4, material callout: IS) Boral; WAS) Boral/Metamic

The proposed change has no effect on the design function or the design of the fuel tube components as described in the SAR. The existing supporting SAR analyses currently reflect the Boral material properties.

Drawing 412-502, Revision 5 – Fuel Can Details Maine Yankee (MY), NAC-UMS®

- Add drawing note: "All welding procedures and qualifications to be in accordance with ASME Section IX."
- Add drawing note: "Visually inspect (VT) all welds in accordance with ASME Section V, Article 9. Acceptance per ASME Section III, NG-5360."

The proposed changes are being added in order to verify that the application design features include sufficient information to demonstrate that the package design satisfies the package standards per 10 CFR 71.31(b), as requested in the NRC's Request for Additional Information regarding the UMS® Transport SAR.

NAC INTERNATIONAL

RESPONSE TO THE

UNITED STATES NUCLEAR REGULATORY COMMISSION

REQUEST FOR ADDITIONAL INFORMATION

(JULY 8, 2004)

NAC UNIVERSAL TRANSPORT CASK (NAC-UMS®)

(TAC NO. L23726, DOCKET NO. 71-9270)

SAR SUBMITTAL – REVISION UMST-04B

SEPTEMBER 2004

NAC INTERNATIONAL RESPONSE
TO
REQUEST FOR ADDITIONAL INFORMATION

TABLE OF CONTENTS

Chapter 1:	General Information	3
Chapter 2:	Structural Evaluation	5
Chapter 3:	Thermal Evaluation	10
Chapter 4:	Containment	13
Chapter 6:	Criticality Evaluation	19
Chapter 8:	Acceptance Tests and Maintenance Program	26

NAC INTERNATIONAL RESPONSE
TO
REQUEST FOR ADDITIONAL INFORMATION

Chapter 1 **General Information**

1-1 Provide a discussion addressing an apparent discrepancy between information provided on page 1.2-9 and Drawing No. 790-573 of the SAR. On page 1.2-9 the following statement is made, "[T]he top nut fabricated from a 3.5 in.-diameter bar, and the spacers, fabricated from 2.5 in. pipe XXS, Type 304 stainless steel." Whereas in Drawing No. 790-573, a diameter of 2.88 in. for the spacers is specified. This value exceeds the nominal diameter of a 2.5 in. pipe.

This information is being requested under the provisions of 10 CFR 71.33 in order to verify that the application accurately describes the package.

NAC Response

Drawing No. 790-573 identifies the spacer (Item 3) as "2 1/2 pipe XXS." This is a callout for "heavy walled" pipe. The nominal outside diameter of this "heavy walled" 2-1/2-inch pipe is actually 2-7/8 inches, which is the spacer outer diameter shown in Zone E-2 of the drawing. Thus, there is no discrepancy between the drawing and the information provided on page 1.2-9.

NAC INTERNATIONAL RESPONSE
TO
REQUEST FOR ADDITIONAL INFORMATION

Chapter 1 General Information

1-2 Provide a revision of Drawing No. 412-502, "Fuel Can Details, Maine Yankee" which adds to the drawing the code and standard requirements for weld inspection and welding procedures and qualifications.

This information is being requested under the provisions of 10 CFR 71.31(b) in order to verify that the application design features include sufficient information to demonstrate that the packaging design satisfies the package standards.

NAC Response

NAC Drawing No. 412-502, Fuel Can Details, Maine Yankee (MY), NAC-UMS®, Revision 5 (see revised UMS® Transport SAR pages) is revised to add the code and standard requirements for weld inspection and welding procedures and qualifications.

NAC INTERNATIONAL RESPONSE
TO
REQUEST FOR ADDITIONAL INFORMATION

Chapter 2 **Structural Evaluation**

2-1 Provide a discussion explaining why the finite element re-analyses of the PWR canister with reduced structural and shield lid weld sizes resulted in changed stresses only in section locations 9, 10, 11, 12, and 13, but not other locations, in Figure 2.6.12.3-1. The staff notes that computed stress components in all other section locations, including Section 8 which is in close proximity to the shield lid weld, remain identical to those analyzed previously for all load cases and load combinations.

This information is being requested under the provisions of 10 CFR 71.35(a) in order to verify that the application has sufficient information to demonstrate that the package satisfies the standards specified in subpart E, Package Approval Standards.

NAC Response

Section 8 in Figure 2.6.12.3-1 is a radial section located 7 inches from the top of the shield lid and is the next closest section to the lid weld region. The change in the stresses for Section 8 due to the change of the weld size (0.125 inch) is considered to be insignificant, as discussed below.

For the end drop orientation, the load through the canister shell does not change as a result of the weld reduction. Regardless of the weld size, the same load from the lid or bottom plate is transmitted to the canister shell. Therefore, the axial stress at Section 8 does not change. For the side drop, the inertia load of the shield lid is transmitted through the shield lid weld to the canister shell, which is supported by the inner shell of the cask body.

The effect of changing the distance from Section 8 to the weld is insignificant for the stress calculations at Section 8. Comparison of the original weld configuration versus the reduced weld configuration shows that the distance from Section 8 to the center of the weld cross-section increases by 1/16 inch, which results in a change in nominal distance from 6.75 inches to 6.8125 inches. Using the methodology presented in Roark's 6^{th} Edition, Table 29, the change in bending stress (dM) at Section 8 is estimated. The expression for the bending moment in the

NAC INTERNATIONAL RESPONSE
TO
REQUEST FOR ADDITIONAL INFORMATION

NAC Response to Question 2-1 (continued)

shell due to an applied moment at the end shows that the bending moment in the shell attenuates as a function $e^{-\lambda x}$.

$$M \approx Ae^{-\lambda x} \text{ and } dM \approx -\lambda(Ae^{-\lambda x})dx$$

Dividing dM by M,

$$dM/M = -\lambda(Ae^{-\lambda x})dx/Ae^{-\lambda x} = -\lambda dx = -(0.281 \times 0.0625) = -0.018 = -1.8\% < 2\%$$

where:

$$\lambda = \left(3 \times \frac{1-\nu^2}{R^2 t^2}\right)^{\frac{1}{4}}$$

R	=	33.5 inch	Canister Radius
t	=	0.625 inch	Canister Shell thickness
ν	=	0.3	Poisson's ratio
dx	=	0.125/2 = 0.0625 inch	Change in distance from the center of the weld to Section 8
x	=	6.8125 inch	Distance from the center of the weld to Section 8

This indicates that, although considered to be small, the stresses at Section 8 are reduced for the smaller weld (less than 2%). By decreasing the size of the weld, the stresses at Section 8 are slightly more attenuated than for the larger weld size. Therefore, the changes in the stresses for Section 8 are considered to be insignificant and, as a result, the stress tables were revised only for Sections 9 through 13. Thus, it is concluded that stress changes at Sections 1 through 7 and Sections 14 through 16 are also insignificant, since they are all located even further from the weld than Section 8.

NAC INTERNATIONAL RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION

Chapter 2 Structural Evaluation

2-2 Provide a discussion describing the stress averaging process as applied to footnotes of Table 2.6.12.6-2 and all other tables alike where stress intensity averages over weld regions are tabulated. This discussion should also include justification, as appropriate, for considering averaged stress intensities over a circular arc region of up to 30° (15° x 2 = 30°).

This information is being requested under the provisions of 10 CFR 71.35(a) in order to verify that the application has sufficient information to demonstrate that the package satisfies the standards specified in Subpart E, Package Approval Standards.

<u>NAC Response</u>

For the 1-foot side drop (Tables 2.6.12.6-2 and 2.6.12.6-3) and the 1-foot bottom corner drop (Table 2.6.12.8-4), bearing stresses are averaged over the weld compression region (contact between the canister shell and the cask inner shell) predicted by the ANSYS finite element model. The region of contact is determined by interpolating the stress component, S_x, obtained from the finite element analysis results at the circumferential locations corresponding to the weld region. When S_x is in compression (or negative), the canister is in contact with the cask inner shell. When S_x is in tension (or positive), the canister is not making contact with the cask. Contact between the canister and cask stops at the location where S_x equals zero. For the following example, S_x equals zero at 15 degrees. The circumferential location of the sections is measured from the plane of symmetry of the finite element model.

Description	Circumferential Location	S_x	S_y	S_z	S_{xy}	S_{yz}	S_{xz}	SI
Section 9	0°	-32.0	5.1	-10.3	-4.0	1.8	0.3	
Section 9	9°	-5.8	4.1	-3.5	-1.1	1.7	-0.5	
Section 9	18°	1.2	0.9	0.6	0.6	0.8	-0.6	
Angle Sx = 0	15.0°	0.00	2.0	-0.6	-0.2	1.1	-0.6	
Section 9 Avg.	**0° – 15.0°**	**-10.9**	**3.8**	**-4.5**	**-1.6**	**1.6**	**-0.3**	**15.4**

NAC INTERNATIONAL RESPONSE
TO
REQUEST FOR ADDITIONAL INFORMATION

NAC Response to Question 2-2 (continued)

For sections where stresses are averaged, the stress intensity is recalculated based on averaged stress components. Stress components are averaged using the trapezoidal averaging method. The following equation is used to average the stress components (conservative).

$$\sigma_{average} = \frac{\sigma_1 + 2\sigma_2 + \sigma_3}{4}$$

where:

$\sigma_{1,2,3}$ = the component stress at three different angular locations (e.g. 0°, 9° and 15° for Section 9).

The ANSYS finite element model of the canister conservatively assumes that the cask inner shell is rigid. Therefore, the analysis underpredicts the actual contact between the canister shell and cask inner shell. With a larger contact area, stresses in the canister shell will be smaller. Therefore, the analysis results are conservative. In accordance with ASME Section III, NB-3227.1(a): *"The average bearing stress for resistance to crushing under the maximum load, experienced as a result of design loads, test loads, or any Service Loadings, except those for which Level D Limits are designated, shall be limited to S_y at temperature..."* Therefore, the bearing stress evaluation is based on the averaged compressive stresses over the contact region (determined by the conservative finite element model) between the canister shell and cask inner shell.

NAC INTERNATIONAL RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION

Chapter 2 **Structural Evaluation**

2-3 Provide a discussion describing the stress averaging process for the maximum bearing stress evaluation, page 2.6-169. This discussion should also include justification, as appropriate, for considering averaged stresses over a circular arc region of up to 30° (15° x 2 = 30°). The staff notes that the previously described stress averaging process, which applied to a relatively smaller stress region over a circular arc of up to 18° (9° x 2 = 18°), was deleted from the present revision, UMST-04A.

This information is being requested under the provisions of 10 CFR 71.35(a) in order to verify that the application has sufficient information to demonstrate that the package satisfies the standards specified in Subpart E, Package Approval Standards.

<u>NAC Response</u>

The description on page 2.6-169 (Section 2.6.12.11) in the UMS® Transport SAR has been revised to include additional discussion for the stress averaging process for the maximum bearing stress evaluation.

NAC INTERNATIONAL RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION

Chapter 3 Thermal Evaluation

3-1 Provide a reference (e.g., the manufacturer's data sheet) for the METAMIC thermal properties and modify Table 3.2-15 so that the thermal properties span the expected range of operation (up to 750°F). Provide the maximum allowable temperature and the safe operating ranges for the METAMIC. Compare the METAMIC properties with the original BORAL properties and justify that both steady-state and transient results presented in Revision 1 are still bounding and applicable.

This information is being requested under the provisions of 10 CFR 71.33 in order to verify that the application has sufficient information to describe in sufficient detail to identify the package accurately and provide sufficient basis for evaluation of the package.

NAC Response

NAC has decided to delete METAMIC from the requested amendment. Accordingly, the reference to METAMIC has been deleted as shown on the following revised License Drawings:

NAC Drawing No. 790-575, Rev. 10	BWR Fuel Tube, NAC-UMS®
NAC Drawing No. 790-581, Rev. 9	PWR Fuel Tube, NAC-UMS®
NAC Drawing No. 790-605, Rev. 11	BWR Fuel Tube, Over-sized Fuel, NAC-UMS®

In addition, all UMS® Transport SAR text specifying METAMIC has been deleted from the requested amendment. Therefore, Table 3.2-15 for Thermal Properties of METAMIC has been deleted.

NAC INTERNATIONAL RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION

Chapter 3 Thermal Evaluation

3-2 Provide a discussion explaining why the results presented in Revision 1 are still applicable for either one of the two proposed configurations for the Maine Yankee fuel can.

This information is being requested under the provisions of 10 CFR 71.33 in order to verify that the application has sufficient information to describe in sufficient detail to identify the package accurately and provide sufficient basis for evaluation of the package.

NAC Response

The difference between the two configurations of the Maine Yankee fuel can is in the width of the tube body: 8.6 inches for the Spent Fuel Can (SFC) Weldment and 8.4 inches for the SFC (Fuel Only) Weldment (Items number 10 and 19 in NAC Drawing No. 412-502). The length and wall thickness of the tube body are identical for both configurations. The thermal analysis for the damaged fuel presented in Revision 1 of the SAR is based on the original configuration (SFC Weldment) with the slightly larger tube width. The debris compaction length of 104 inches used in the thermal analysis model for the original fuel can configuration is slightly conservative for the second fuel can configuration. The slightly smaller tube cross-sectional area of the second fuel can configuration results in a longer length debris region. Since the same debris heat load is distributed over a longer length, the maximum fuel and basket temperature will be slightly reduced. Therefore, the thermal analysis of the original fuel can configuration that is presented in SAR Revision 1 bounds both of the two Maine Yankee fuel can configurations.

NAC INTERNATIONAL RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION

Chapter 3 Thermal Evaluation

3-3 Provide justification for the removal of Maine Yankee site specific definition for damaged high burnup fuel that must be loaded into fuel cans: "These fuel assemblies have more than 1 percent of rods with oxide layers greater than 80 microns or more than 3 percent of rods with oxide layers greater than 70 microns and burnup greater than 45,000 MWD/MTU." from page 3.6-8 of the SAR.

This information is being requested under the provisions of 10 CFR 71.33 in order to verify that the application has sufficient information to describe in sufficient detail to identify the package accurately and provide sufficient basis for evaluation of the package.

NAC Response

The previously removed text is reinstated into the UMS® Transport SAR with the clarification that it is based on NRC guidance provided in ISG-15. It is noted for information that all Maine Yankee site-specific high burnup fuel has been loaded into Maine Yankee damaged fuel cans.

NAC INTERNATIONAL RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION

Chapter 4 Containment

4-1 Provide a discussion that clarifies the definition of damaged fuel, especially for high burn-up assemblies. NAC may want to consider adopting parts (1) through (3) of the definition of damaged fuel in the staff's latest guidance provided in Interim Staff Guidance-1 (ISG-1), Revision 1, or a suitable alternative, to avoid inappropriate classification of fuel assemblies.

ISG-1, Revision 1 states:

> *"Damaged fuel – Spent nuclear fuel is considered damaged for storage or transportation purposes if it manifests any of the following conditions that result in either compromise of cladding confinement integrity or rearrangement (reconfiguration) of fuel bundle geometry:*
>
> *1) The fuel contains known or suspected cladding defects greater than a pinhole leak or hairline crack that have the potential for release of significant amounts of fuel particles into the cask.*
>
> *2) The fuel assembly:*
> *a. Is damaged in such a manner as to impair its structural integrity;*
> *b. Has missing or displaced structural components such as grid spacers;*
> *c. Is missing fuel pins which have not been replaced by dummy rods which displace a volume equal to or greater than the original fuel rod;*
> *d. Cannot be handled using normal (i.e. crane and grapple) handling methods. (Exception: fuel assemblies with repaired lifting bails, support caps, or support tubes, etc., which permit normal handling may be classified as intact. See later discussion.)*
>
> *3) The fuel is no longer in the form of an intact fuel bundle and consists of, or contains, debris such as loose fuel pellets, rod segments, etc."*

Part 4 of the definition in ISG-1, Revision 1, pertaining to the material properties of high burn-up assemblies need not be adopted due to the present lack of cladding material property data.

This information is being requested under the provisions of 10 CFR 71.31 in order to verify compliance with 10 CFR 71.33.

NAC INTERNATIONAL RESPONSE
TO
REQUEST FOR ADDITIONAL INFORMATION

NAC Response to RAI 4-1

The definition of damaged fuel that is appropriate for the approved contents of the UMS® Transport Cask is a modified version of the definition provided in ISG-1, Revision 1. The definition is consistent with the definition of intact fuel. The definition is as follows.

Damaged fuel:

Spent nuclear fuel that manifests any of the following conditions that result in either compromise of cladding confinement integrity or rearrangement (reconfiguration) of fuel bundle geometry:

1) A fuel assembly that contains known or suspected cladding defects greater than pinhole leaks or hairline cracks that have the potential for release of significant amounts of fuel particles into the cask cavity.

2) A fuel assembly that:
   a. is damaged in such a manner as to impair its structural integrity;
   b. has missing or displaced structural components such as grid spacers; or
   c. cannot be handled using normal (i.e., crane and grapple) handling methods.

3) Fuel that is no longer in the form of an intact fuel assembly and consists of, or contains, debris such as loose fuel pellets, rod segments, etc.

Damaged fuel assemblies and high burnup fuel assemblies that are authorized as contents for the NAC-UMS® System are Maine Yankee site-specific fuel assemblies, which have been placed in Maine Yankee Fuel Cans—i.e., damaged fuel cans (DFC) prior to loading into the UMS® canister. All of the Maine Yankee DFCs are placed into basket corner positions. Thus, an intact/damaged fuel classification criterion for high burnup fuel assemblies is not required for the authorized contents for the NAC-UMS® System. The definition of damaged fuel is inserted in SAR Table 1-1.

NAC INTERNATIONAL RESPONSE
TO
REQUEST FOR ADDITIONAL INFORMATION

NAC Response to Question 4-1 (continued)

(Note that PWR intact fuel assemblies with one or more grid spacers missing, or damaged such that the unsupported length of the fuel rods does not exceed 60 inches, or with end fitting damage such as damaged or missing hold-down springs may be loaded as intact fuel as long as they can be handled safely by normal means.)

NAC INTERNATIONAL RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION

Chapter 4 **Containment**

4-2 Provide a discussion clarifying the basis for the assumed failure rates of the high burn-up fuel assemblies, which were used to determine the maximum allowable leak rates in the containment analysis.

This information is being requested under the provisions of 10 CFR 71.31 in order to verify compliance with 10 CFR 71.35.

NAC Response

The assumed failure rate of 20% is based on ISG-15. ISG-15 mirrors guidance provided in ISG-11, Revision 1, in that both documents suggest the use of a 50% failure fraction in the containment analysis of high burnup rods with oxide layers over 70 microns thick. Further, the guidance indicates that a fuel assembly with more than 3% of its fuel rods having greater than 70-micron thick oxide layers be placed into a damaged fuel can (DFC). There are a maximum of four DFCs per basket. The UMS® transport cask payload definition also limits the cask contents to a maximum of 12 high burnup assemblies.

The 20% failure fraction applied in the analysis bounds a standard 3% failure fraction in the 12 standard assemblies, a 4.5% failure fraction in the 8 high burnup assemblies not in cans (3% standard failure fraction and a 50% failure rate of the 3% of high burnup rods over 70 microns), and 50% failure of the 4 canned high burnup assemblies.

The standard or high burnup assemblies loaded in DFCs may be damaged to the extent that a 50% failure rate would not be conservative. Therefore, the 4 assemblies in DFCs are evaluated at a 100% failure fraction, resulting in an average failure rate of 19.7%, which is rounded up to 20% in the analysis.

NAC INTERNATIONAL RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION

NAC Response to Question 4-2 (continued)

Note that applying the 20% failure rate to the radionuclide inventory (gaseous and solid) of 24 high burnup assemblies is clearly conservative compared to the actual source limited to a maximum 12 high burnup assemblies and 12 standard assemblies.

UMS® Transport SAR text in Section 4.2.1.1 is modified to clarify the failure fraction applied in the analysis.

NAC INTERNATIONAL RESPONSE
TO
REQUEST FOR ADDITIONAL INFORMATION

Chapter 4 **Containment**

4-3 Provide a discussion explaining the apparent discrepancy between the assumed PWR high burnup assembly failure percentages in Sections 4.2.1 and 4.5.1 of the SAR. Section 4.2.1, page 4.2-5 states "[T]he 4 high burnup assemblies failing at 50%," whereas Section 4.5.1, page 4.5.1-1 states, "[T]he PWR containment analysis in Section 4.2.1 assumes 4 PWR (high burnup) fuel assemblies failing at 100%."

This information is being requested under the provisions of 10 CFR 71.7 in order to verify that the application has complete and accurate information.

<u>NAC Response</u>

Section 4.2.1 contained text consistent with the ISG-15 guidance on high burnup fuel (50% failure rate of rods with over a 70-micron oxide layer thickness), but did not address the higher potential source associated with damaged fuel assemblies. Normal transport condition analysis in the UMS® Transport SAR Section 4.2.1 is based on a conservative 100% failure rate of the canned assemblies. Text in Section 4.2.1 is revised to correctly reflect the 100% failure rate applied. Note that this is a conservative approach, as rods damaged prior to loading would have, at a minimum, released their gaseous inventory prior to placement into the TSC.

Text in SAR Section 4.5.1 is also revised for clarity.

NAC INTERNATIONAL RESPONSE
TO
REQUEST FOR ADDITIONAL INFORMATION

Chapter 6 **Criticality Evaluation**

6-1 Provide an expanded discussion in Section 6.4.5, page 6.4-22, first paragraph, addressing the methodology used to determine the maximum lengths of exposed PWR fuel for the Westinghouse 15 x 15 PWR fuel assembly.

This information is being requested under the provisions of 10 CFR 71.33 in order to verify that the modeled cases adequately bound all potential reactivity effects.

NAC Response

The detailed discussion for PWR fuel coverage calculations is included on SAR pages 6.4-20 through 6.4-23 of this submittal. The third paragraph on SAR page 6.4-22 includes the result from the "third stage" of the evaluation.

As described in the text at the top of page 6.4-21, fuel coverage is determined by a three-stage process: calculation of the neutron absorber offset, determination of the active fuel offset, and calculation of the fuel exposure.

The first full paragraph on page 6.4-21 describes the components and method considered in the neutron absorber offset calculation. Included in this paragraph is the method change that produced the revised value for the neutron absorber offset that was presented in the Revision UMST-04A SAR changed pages. The information presented in the Revision UMST-04A changed pages used the same method for evaluating the PWR fuel coverage that was used for the calculation of the BWR fuel coverage—i.e., the top end drop evaluation credited the axial shift of the basket and fuel tubes, as well as the fuel assembly shift. Previous top end drop evaluations located the basket at the bottom of the canister cavity for PWR evaluation and at the top of the basket for BWR evaluations. As indicated in Table 6.4-19, shifting the basket reduced the distance from the shield lid to the top of the absorber to 8.89 inches from 10.19 inches for the

NAC INTERNATIONAL RESPONSE
TO
REQUEST FOR ADDITIONAL INFORMATION

NAC Response to Question 6-1 (continued)

Class 1 canister (similar changes occurred for all PWR canister sizes). Footnotes on Table 6.4-19 were modified in the Revision UMST-04A changed pages to be consistent with the text and method changes. Note that the fuel exposure included in the criticality models is conservative for either a shifted or nonshifted basket.

The method employed in the active fuel offset calculation is discussed in the text of the second and third full paragraphs on page 6.4-21. As indicated in this text, the fuel assembly is shifted to contact the shield lid, and fuel rods are shifted into contact with the end-fitting during a top end impact.

Structural evaluations demonstrate that the top end-fitting does not deform during impact and that the minimum plenum spring rebound is 2.39 inches (text changes incorporating the 2.39-inch spring rebound were included in the second bullet, page 6.4-21). PWR evaluations performed prior to those presented in SAR Revision UMST-04A credited no plenum spring rebound, while BWR evaluations took credit for the rebound of the spring. Adding the spring rebound to the end-fitting height, fuel rod end plug length, and compressed spring height resulted in an increase in the minimum distance between the shield lid and the active fuel region. For the W 15×15 fuel assembly, the rebound of 2.39 inches increased the distance to the active fuel region from 5.718 inches to 8.108 inches.

As the fuel exposure is simply the difference between the neutron absorber offset and the active fuel offset (see second paragraph, SAR page 6.4-21 of this submittal), the recalculated exposed fuel length for the W 15×15 assembly decreased from 4.472 inches to 0.782 inch. An exposed fuel height of 4.52 inches was conservatively retained for the criticality analysis, and no revised criticality evaluations were performed in conjunction with the fuel coverage evaluation.

NAC INTERNATIONAL RESPONSE
TO
REQUEST FOR ADDITIONAL INFORMATION

Chapter 6 **Criticality Evaluation**

6-2 Provide a discussion explaining in detail how and why the values for fuel types WE 15x15, WE 17x17 OFA, B&W 15x15, and CE 16x16 (SYS 80) changed for the Neutron Absorber Offset, Fuel Offset, and the Calculated Exposed Fuel Height, Table 6.4-19, page 6.4-39. It is not clear from the changes indicated in the SAR submitted by NAC why these values were altered in this revision.

This information is being requested under the provisions of 10 CFR 71.33 in order to verify that the modeled cases adequately bound all potential reactivity effects.

NAC Response

Details on the changes are provided in NAC's response to RAI 6-1. Changes in the values are the result of revising the PWR analysis method to make it consistent with the BWR analysis method. This revision involved shifting the basket and its components toward the shield lid under top end drop accident conditions and crediting the plenum spring rebound. These changes were marked as revisions in the SAR Revision UMST-04A changed pages on page 6.4-21 for text changes and on page 6.4-39 as footnote changes.

NAC INTERNATIONAL RESPONSE
TO
REQUEST FOR ADDITIONAL INFORMATION

Chapter 6 Criticality Evaluation

6-3 Provide a discussion explaining in detail the changes in the values for the Neutron Absorber Offset, Plenum Spring Rebound Height, and Minimum Intact Hardware Dimension Limit for the five UMS canister classes, Table 6.4-20, page 6.4-39. It is not clear from the changes indicated in the SAR submitted by NAC why these values were altered in this revision.

This information is being requested under the provisions of 10 CFR 71.33 in order to verify that the modeled cases adequately bound all potential reactivity effects.

NAC Response

Changes in the minimum intact hardware definitions were limited to the PWR canister classes (canister Class 1, 2 and 3). As indicated in NAC's responses to RAI 6-1 and RAI 6-2, these changes are the result of making the analysis method for PWR fuel consistent with the BWR evaluation method. In particular, the revised PWR minimum intact hardware dimensions are based on a reduced neutron absorber offset (see Table 6.4-19) and inclusion of plenum spring rebound.

NAC INTERNATIONAL RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION

Chapter 6 Criticality Evaluation

6-4 Provide the reason why the last paragraph of the subsection, "Evaluation of BWR System Top End Impact" in Section 6.4.5, page 6.4-20, was deleted. It is not clear from the changes indicated in the SAR, why the values representing the mechanical perturbations and geometric tolerances were altered. The deleted paragraph provided some insight of how the criticality model was modified to reflect the fuel exposure.

This information is being requested under the provisions of 10 CFR 71.33 in order to verify that the modeled cases adequately bound all potential reactivity effects.

NAC Response

The criticality model for BWR fuel exposure is based on the bulleted description following the first full paragraph on SAR page 6.4-20 of this submittal. The referenced section was deleted to minimize confusion with respect to the connection between the modeled configuration and the maximum fuel exposure calculated for the basket. It did not provide information on how the model was modified. The deleted text was based on previous iterations of the fuel exposure structural and nuclear evaluations; that text did not accurately reflect the connection between the modeled geometry and the actual worst-case configuration. To clarify the differences between the modeled configuration and the applied fuel exposure, the bulleted list on page 6.4-20 of this submittal is augmented with additional text.

NAC INTERNATIONAL RESPONSE
TO
REQUEST FOR ADDITIONAL INFORMATION

Chapter 6 Criticality Evaluation

6-5 Provide the reason why the last paragraph of the subsection, "Evaluation of PWR System Top End Impact" in Section 6.4.5, page 6.4-22, was deleted. It is not clear from the changes indicated in the SAR why the values representing the mechanical perturbations and geometric tolerances were altered. This paragraph provided some insight of how the criticality model was modified to reflect the fuel exposure.

This information is being requested under the provisions of 10 CFR 71.33 in order to verify that the modeled cases adequately bound all potential reactivity effects.

NAC Response

As indicated in NAC's response to RAI 6-4, the deleted text was not meant to provide information on the exposed fuel criticality model. As a result of the changes in the PWR analysis method, the values listed in the deleted section changed. As they provided no distinct analytical benefit, the section was removed. To clarify the differences between the modeled configuration and the applied fuel exposure, the bulleted list on page 6.4-22 is augmented with additional text.

NAC INTERNATIONAL RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION

Chapter 6 Criticality Evaluation

6-6 Provide any additional calculations that were performed to demonstrate the criticality safety of the two Maine Yankee fuel canister configurations, or provide additional justification as to why the fuel can configuration with the larger internal width is bounding for both fuel can configurations.

This information is being requested under the provisions of 10 CFR 71.33 in order to verify that the modeled cases adequately bound all fuel canister configuration.

NAC Response

No additional calculations were performed for the reduced-size fuel can. The reactivity effect of the larger damaged fuel cans included evaluations of 0% to 100% fuel fraction (see second paragraph under "Damaged Fuel Evaluation" on page 6.6.1-7 of SAR Revision 1). This evaluation, therefore, placed varying fuel fractions, including an optimum moderated homogeneous mixture and a solid fuel mass, into the largest cross-sectional area of a damaged fuel can. This analysis by default covered a fissile material content range significantly higher than actually feasible within the damaged fuel can cavity. A smaller damaged fuel can cross-section provides a lower reactivity content by reducing the fissile material quantity at any of the fuel fractions evaluated and is, therefore, clearly a lower reactivity configuration.

NAC INTERNATIONAL RESPONSE
TO
REQUEST FOR ADDITIONAL INFORMATION

Chapter 8 Acceptance Tests and Maintenance Program

8-1 Provide a revision of Chapter 8 that includes Metamic, as one of the neutron absorbers used for criticality control. The revision should include the following:

a. A detailed discussion on how the effective B-10 areal density is measured and determined.

b. The level of B-10 credit requested for the material.

c. A detailed discussion on how the samples are selected from a plate.

d. A detailed discussion of the acceptance criteria based on statical analysis that includes how uncertainties are accounted for in the analysis.

e. A detailed discussion of the rejection criteria in the event a coupon fails to give the minimum effective B-10 areal density.

f. A computation of the areal density to be used in the material along with the specified design minimum.

g. A detailed discussion of qualification tests for any minor or major processing changes to the extent such that they could affect properties of the neutron absorber required to perform its design functions for its design lifetime and environment (e.g., increasing the boron carbide content, changes in mechanical processing techniques and fabrication, etc.). Note that the changes could effect the metallurgical stability of the material.

This information is being requested under the provisions of 10 CFR 71.33 in order to verify that the application has sufficient detail to provide the staff reasonable assurance that material maintains its efficacy and that criticality control is maintained.

NAC INTERNATIONAL RESPONSE
TO
REQUEST FOR ADDITIONAL INFORMATION

NAC Response to RAI 8-1

NAC has decided to delete METAMIC from the requested amendment. Accordingly, the reference to METAMIC has been deleted as shown on the following revised License Drawings:

NAC Drawing No. 790-575, Rev. 10	BWR Fuel Tube, NAC-UMS®
NAC Drawing No. 790-581, Rev. 9	PWR Fuel Tube, NAC-UMS®
NAC Drawing No. 790-605, Rev. 11	BWR Fuel Tube, Over-sized Fuel, NAC-UMS®

In addition, all UMS® Transport SAR text specifying METAMIC has been deleted from the requested amendment.

Therefore, Chapter 8 of the UMS® Transport SAR is not revised and remains unchanged.

September 2004
Revision UMST-04B

NAC-UMS

Universal Multi-Purpose Cask System

SAFETY ANALYSIS REPORT

Volume 1 of 2
Docket No. 71-9270

NAC INTERNATIONAL

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List of Effective Pages

List of Effective Pages

Master Table of Contents

i .... Revision 1
ii .... Revision 1
iii .... Revision 1
iv .... Revision 1
v..................................... Revision 1
vi .... Revision 1
vii .... Revision 1
viii .... Revision 1
ix .... Revision 1
x .... Revision 1
xi .... Revision 1
xii .... Revision 1
xiii .... Revision 1
xiv .... Revision 1
xv .... Revision 1
xvi .... Revision 1
xvii .... Revision 1
xviii .... Revision 1

Chapter 1

1-i .... Revision 1
1-ii .... Revision 1
1-1 .... Revision 1
1-2 .... Revision 1
1-3 .... Revision 1
1-4 .... Revision 1
1-5 .... Revision 1
1-6 .... Revision UMST-04A
1-7 .... Revision UMST-04B
1-8 .... Revision UMST-04B
1.1-1 .... Revision UMST-04A
1.1-2 .... Revision 1
1.1-3 .... Revision UMST-04A
1.1-4 .... Revision UMST-04A
1.1-5 .... Revision 1
1.1-6 .... Revision 1
1.2-1 .... Revision UMST-04A
1.2-2 .... Revision 1
1.2-3 .... Revision 1
1.2-4 .... Revision 1
1.2-5 .... Revision 1
1.2-6 .... Revision 1
1.2-7 .... Revision 1
1.2-8 .... Revision 1
1.2-9 .... Revision UMST-04B
1.2-10 .... Revision UMST-04B
1.2-11 .... Revision UMST-04A
1.2-12 .... Revision 1
1.2-13 .... Revision UMST-04A
1.2-14 .... Revision UMST-04A
1.2-15 .... Revision 1
1.2-16 .... Revision 1
1.2-17 .... Revision 1
1.2-18 .... Revision 1
1.2-19 .... Revision UMST-04A
1.2-20 .... Revision UMST-04B
1.2-21 .... Revision UMST-04A
1.2-22 .... Revision 1
1.2-23 .... Revision 1
1.2-24 .... Revision 1
1.2-25 .... Revision 1
1.3-1 .... Revision 1
1.3.1-1 .... Revision 1

List of Effective Pages (Continued)

1.3.1-2	Revision UMST-04A
1.3.1-3	Revision 1
1.3.1-4	Revision 1
1.3.1-5	Revision 1
1.3.1-6	Revision 1
1.3.1-7	Revision 1
1.3.1-8	Revision 1
1.3.1-9	Revision 1
1.3.1-10	Revision 1
1.3.1-11	Revision 1
1.3.1-12	Revision 1
1.3.2-1	Revision 1
1.3.3-1	Revision 1
1.3.4-1	Revision UMST-04A
1.3.4-2	Revision UMST-04B

License Drawings
35 drawings
Revised as shown in Chapter 1

Chapter 2

2-i	Revision 1
2-ii	Revision 1
2-iii	Revision 1
2-iv	Revision 1
2-v	Revision 1
2-vi	Revision 1
2-vii	Revision 1
2-viii	Revision 1
2-ix	Revision 1
2-x	Revision 1
2-xi	Revision 1
2-xii	Revision 1
2-xiii	Revision 1
2-xiv	Revision 1
2-xv	Revision 1
2-xvi	Revision 1
2-xvii	Revision 1
2-xviii	Revision 1
2-xix	Revision 1
2-xx	Revision 1
2-xxi	Revision 1
2-xxii	Revision 1
2-xxiii	Revision 1
2-xxiv	Revision 1
2-xxv	Revision 1
2-xxvi	Revision 1
2-xxvii	Revision 1
2-xxviii	Revision 1
2-xxix	Revision 1
2-xxx	Revision 1
2-xxxi	Revision 1
2-xxxii	Revision 1
2-xxxiii	Revision 1
2-xxxiv	Revision UMST-04A
2-1	Revision 1
2.1-1	Revision 1
2.1-2	Revision 1
2.1-3	Revision 1
2.1-4	Revision 1
2.1-5	Revision UMST-04A
2.1-6	Revision UMST-04A
2.1-7	Revision 1
2.1-8	Revision 1
2.1-9	Revision 1

List of Effective Pages (Continued)

2.1-10 .... Revision 1
2.1-11 .... Revision 1
2.1-12 .... Revision UMST-04A
2.1-13 .... Revision 1
2.1-14 .... Revision 1
2.1-15 .... Revision 1
2.1-16 .... Revision 1
2.1-17 .... Revision 1
2.1-18 .... Revision 1
2.1-19 .... Revision 1
2.1-20 .... Revision 1
2.1-21 .... Revision 1
2.1-22 .... Revision 1
2.1-23 .... Revision 1
2.1-24 .... Revision 1
2.2-1 .... Revision 1
2.2-2 .... Revision 1
2.2-3 .... Revision 1
2.2-4 .... Revision 1
2.3-1 .... Revision 1
2.3-2 .... Revision UMST-04A
2.3-3 .... Revision 1
2.3-4 .... Revision 1
2.3-5 .... Revision 1
2.3-6 .... Revision 1
2.3-7 .... Revision 1
2.3-8 .... Revision 1
2.3-9 .... Revision 1
2.3-10 .... Revision 1
2.3-11 .... Revision 1
2.3-12 .... Revision 1
2.3-13 .... Revision 1
2.3-14 .... Revision 1
2.3-15 .... Revision 1
2.3-16 .... Revision 1
2.3-17 .... Revision 1
2.3-18 .... Revision 1
2.3-19 .... Revision 1
2.3-20 .... Revision 1
2.3-21 .... Revision 1
2.3-22 .... Revision 1
2.4-1 .... Revision 1
2.4-2 .... Revision 1
2.4-3 .... Revision 1
2.4-4 .... Revision 1
2.4-5 .... Revision 1
2.4-6 .... Revision 1
2.4-7 .... Revision 1
2.4-8 .... Revision 1
2.4-9 .... Revision 1
2.5-1 .... Revision 1
2.5-2 .... Revision 1
2.5-3 .... Revision 1
2.5-4 .... Revision 1
2.5-5 .... Revision 1
2.5-6 .... Revision 1
2.5-7 .... Revision 1
2.5-8 .... Revision 1
2.5-9 .... Revision 1
2.5-10 .... Revision 1
2.5-11 .... Revision 1
2.5-12 .... Revision 1
2.5-13 .... Revision 1
2.5-14 .... Revision 1
2.5-15 .... Revision 1
2.5-16 .... Revision 1

List of Effective Pages (Continued)

2.5-17 .... Revision 1
2.5-18 .... Revision 1
2.5-19 .... Revision 1
2.5-20 .... Revision 1
2.5-21 .... Revision 1
2.5-22 .... Revision 1
2.5-23 .... Revision 1
2.5-24 .... Revision 1
2.5-25 .... Revision 1
2.5-26 .... Revision 1
2.5-27 .... Revision 1
2.5-28 .... Revision 1
2.5-29 .... Revision 1
2.5-30 .... Revision 1
2.5-31 .... Revision 1
2.5-32 .... Revision 1
2.5-33 .... Revision 1
2.5-34 .... Revision 1
2.5-35 .... Revision 1
2.5-36 .... Revision 1
2.5-37 .... Revision 1
2.5-38 .... Revision 1
2.5-39 .... Revision 1
2.5-40 .... Revision 1
2.5-41 .... Revision 1
2.5-42 .... Revision 1
2.5-43 .... Revision 1
2.5-44 .... Revision 1
2.5-45 .... Revision 1
2.5-46 .... Revision 1
2.5-47 .... Revision 1
2.5-48 .... Revision 1
2.5-49 .... Revision 1
2.5-50 .... Revision 1
2.5-51 .... Revision 1
2.5-52 .... Revision 1
2.6-1 .... Revision 1
2.6-2 .... Revision 1
2.6-3 .... Revision 1
2.6-4 .... Revision 1
2.6-5 .... Revision 1
2.6-6 .... Revision 1
2.6-7 .... Revision 1
2.6-8 .... Revision 1
2.6-9 .... Revision 1
2.6-10 .... Revision 1
2.6-11 .... Revision 1
2.6-12 .... Revision 1
2.6-13 .... Revision 1
2.6-14 .... Revision 1
2.6-15 .... Revision 1
2.6-16 .... Revision 1
2.6-17 .... Revision 1
2.6-18 .... Revision 1
2.6-19 .... Revision 1
2.6-20 .... Revision 1
2.6-21 .... Revision 1
2.6-22 .... Revision 1
2.6-23 .... Revision 1
2.6-24 .... Revision 1
2.6-25 .... Revision 1
2.6-26 .... Revision 1
2.6-27 .... Revision 1
2.6-28 .... Revision 1
2.6-29 .... Revision 1
2.6-30 .... Revision 1

List of Effective Pages (Continued)

2.6-31 .... Revision 1
2.6-32 .... Revision 1
2.6-33 .... Revision 1
2.6-34 .... Revision 1
2.6-35 .... Revision 1
2.6-36 .... Revision 1
2.6-37 .... Revision 1
2.6-38 .... Revision 1
2.6-39 .... Revision 1
2.6-40 .... Revision 1
2.6-41 .... Revision 1
2.6-42 .... Revision 1
2.6-43 .... Revision 1
2.6-44 .... Revision 1
2.6-45 .... Revision 1
2.6-46 .... Revision 1
2.6-47 .... Revision 1
2.6-48 .... Revision 1
2.6-49 .... Revision 1
2.6-50 .... Revision 1
2.6-51 .... Revision 1
2.6-52 .... Revision 1
2.6-53 .... Revision 1
2.6-54 .... Revision 1
2.6-55 .... Revision 1
2.6-56 .... Revision 1
2.6-57 .... Revision 1
2.6-58 .... Revision 1
2.6-59 .... Revision 1
2.6-60 .... Revision 1
2.6-61 .... Revision 1
2.6-62 .... Revision 1
2.6-63 .... Revision 1
2.6-64 .... Revision 1
2.6-65 .... Revision 1
2.6-66 .... Revision 1
2.6-67 .... Revision 1
2.6-68 .... Revision 1
2.6-69 .... Revision 1
2.6-70 .... Revision 1
2.6-71 .... Revision 1
2.6-72 .... Revision 1
2.6-73 .... Revision 1
2.6-74 .... Revision 1
2.6-75 .... Revision 1
2.6-76 .... Revision 1
2.6-77 .... Revision 1
2.6-78 .... Revision 1
2.6-79 .... Revision 1
2.6-80 .... Revision 1
2.6-81 .... Revision 1
2.6-82 .... Revision 1
2.6-83 .... Revision 1
2.6-84 .... Revision 1
2.6-85 .... Revision 1
2.6-86 .... Revision 1
2.6-87 .... Revision 1
2.6-88 .... Revision 1
2.6-89 .... Revision 1
2.6-90 .... Revision 1
2.6-91 .... Revision 1
2.6-92 .... Revision 1
2.6-93 .... Revision 1
2.6-94 .... Revision 1
2.6-95 .... Revision 1
2.6-96 .... Revision 1

List of Effective Pages (Continued)

2.6-97 .... Revision 1
2.6-98 .... Revision 1
2.6-99 .... Revision 1
2.6-100 .... Revision 1
2.6-101 .... Revision 1
2.6-102 .... Revision UMST-04A
2.6-103 .... Revision 1
2.6-104 .... Revision 1
2.6-105 .... Revision 1
2.6-106 .... Revision 1
2.6-107 .... Revision 1
2.6-108 .... Revision 1
2.6-109 .... Revision 1
2.6-110 .... Revision 1
2.6-111 .... Revision 1
2.6-112 .... Revision 1
2.6-113 .... Revision 1
2.6-114 .... Revision 1
2.6-115 .... Revision UMST-04A
2.6-116 .... Revision 1
2.6-117 .... Revision 1
2.6-118 .... Revision 1
2.6-119 .... Revision 1
2.6-120 .... Revision 1
2.6-121 .... Revision 1
2.6-122 .... Revision 1
2.6-123 .... Revision 1
2.6-124 .... Revision 1
2.6-125 .... Revision 1
2.6-126 .... Revision 1
2.6-127 .... Revision 1
2.6-128 .... Revision 1
2.6-129 .... Revision 1
2.6-130 .... Revision UMST-04A
2.6-131 .... Revision UMST-04A
2.6-132 .... Revision UMST-04A
2.6-133 .... Revision 1
2.6-134 .... Revision 1
2.6-135 .... Revision UMST-04A
2.6-136 .... Revision UMST-04A
2.6-137 .... Revision UMST-04A
2.6-138 .... Revision UMST-04A
2.6-139 .... Revision UMST-04A
2.6-140 .... Revision UMST-04A
2.6-141 .... Revision 1
2.6-142 .... Revision 1
2.6-143 .... Revision UMST-04A
2.6-144 .... Revision UMST-04A
2.6-145 .... Revision UMST-04A
2.6-146 .... Revision UMST-04A
2.6-147 .... Revision 1
2.6-148 .... Revision UMST-04A
2.6-149 .... Revision UMST-04A
2.6-150 .... Revision UMST-04A
2.6-151 .... Revision UMST-04A
2.6-152 .... Revision UMST-04A
2.6-153 .... Revision UMST-04A
2.6-154 .... Revision UMST-04A
2.6-155 .... Revision UMST-04A
2.6-156 .... Revision 1
2.6-157 .... Revision UMST-04A
2.6-158 .... Revision UMST-04A
2.6-159 .... Revision UMST-04A
2.6-160 .... Revision UMST-04A
2.6-161 .... Revision UMST-04A
2.6-162 .... Revision 1

List of Effective Pages (Continued)

2.6-163 .... Revision UMST-04A
2.6-164 .... Revision UMST-04A
2.6-165 .... Revision UMST-04A
2.6-166 .... Revision UMST-04A
2.6-167 .... Revision UMST-04A
2.6-168 .... Revision UMST-04A
2.6-169 .... Revision UMST-04B
2.6-170 .... Revision UMST-04A
2.6-171 .... Revision 1
2.6-172 .... Revision 1
2.6-173 .... Revision 1
2.6-174 .... Revision UMST-04A
2.6-175 .... Revision 1
2.6-176 .... Revision 1
2.6-177 .... Revision 1
2.6-178 .... Revision 1
2.6-179 .... Revision 1
2.6-180 .... Revision 1
2.6-181 .... Revision 1
2.6-182 .... Revision 1
2.6-183 .... Revision 1
2.6-184 .... Revision 1
2.6-185 .... Revision 1
2.6-186 .... Revision 1
2.6-187 .... Revision 1
2.6-188 .... Revision 1
2.6-189 .... Revision 1
2.6-190 .... Revision 1
2.6-191 .... Revision 1
2.6-192 .... Revision 1
2.6-193 .... Revision 1
2.6-194 .... Revision 1
2.6-195 .... Revision 1
2.6-196 .... Revision 1
2.6-197 .... Revision 1
2.6-198 .... Revision 1
2.6-199 .... Revision 1
2.6-200 .... Revision 1
2.6-201 .... Revision 1
2.6-202 .... Revision 1
2.6-203 .... Revision 1
2.6-204 .... Revision 1
2.6-205 .... Revision UMST-04A
2.6-206 .... Revision 1
2.6-207 .... Revision 1
2.6-208 .... Revision 1
2.6-209 .... Revision 1
2.6-210 .... Revision 1
2.6-211 .... Revision 1
2.6-212 .... Revision 1
2.6-213 .... Revision 1
2.6-214 .... Revision 1
2.6-215 .... Revision 1
2.6-216 .... Revision 1
2.6-217 .... Revision 1
2.6-218 .... Revision 1
2.6-219 .... Revision 1
2.6-220 .... Revision 1
2.6-221 .... Revision 1
2.6-222 .... Revision 1
2.6-223 .... Revision 1
2.6-224 .... Revision 1
2.6-225 .... Revision 1
2.6-226 .... Revision 1
2.6-227 .... Revision 1
2.6-228 .... Revision 1

List of Effective Pages (Continued)

2.6-229 .... Revision 1
2.6-230 .... Revision 1
2.6-231 .... Revision 1
2.6-232 .... Revision 1
2.6-233 .... Revision 1
2.6-234 .... Revision 1
2.6-235 .... Revision 1
2.6-236 .... Revision 1
2.6-237 .... Revision 1
2.6-238 .... Revision 1
2.6-239 .... Revision 1
2.6-240 .... Revision 1
2.6-241 .... Revision UMST-04A
2.6-242 .... Revision 1
2.6-243 .... Revision 1
2.6-244 .... Revision UMST-04A
2.6-245 .... Revision 1
2.6-246 .... Revision 1
2.6-247 .... Revision 1
2.6-248 .... Revision 1
2.6-249 .... Revision 1
2.6-250 .... Revision 1
2.6-251 .... Revision 1
2.6-252 .... Revision 1
2.6-253 .... Revision 1
2.6-254 .... Revision 1
2.6-255 .... Revision 1
2.6-256 .... Revision 1
2.6-257 .... Revision 1
2.6-258 .... Revision 1
2.6-259 .... Revision 1
2.6-260 .... Revision 1
2.6-261 .... Revision 1
2.6-262 .... Revision 1
2.6-263 .... Revision 1
2.5-264 .... Revision 1
2.6-265 .... Revision 1
2.6-266 .... Revision 1
2.6-267 .... Revision 1
2.6-268 .... Revision 1
2.6-269 .... Revision 1
2.6-270 .... Revision 1
2.6-271 .... Revision 1
2.6-272 .... Revision 1
2.6-273 .... Revision 1
2.6-274 .... Revision 1
2.6-275 .... Revision 1
2.6-276 .... Revision 1
2.6-277 .... Revision 1
2.6-278 .... Revision 1
2.6-279 .... Revision 1
2.6-280 .... Revision 1
2.6-281 .... Revision 1
2.6-282 .... Revision 1
2.6-283 .... Revision 1
2.6-284 .... Revision 1
2.6-285 .... Revision 1
2.6-286 .... Revision 1
2.6-287 .... Revision 1
2.6-288 .... Revision 1
2.6-289 .... Revision 1
2.6-290 .... Revision 1
2.6-291 .... Revision 1
2.6-292 .... Revision 1
2.6-293 .... Revision 1
2.6-294 .... Revision 1

List of Effective Pages (Continued)

2.6-295 .... Revision 1
2.6-296 .... Revision 1
2.6-297 .... Revision UMST-04A
2.6-298 .... Revision 1
2.6-299 .... Revision 1
2.6-300 .... Revision 1
2.6-301 .... Revision 1
2.6-302 .... Revision 1
2.6-303 .... Revision 1
2.6-304 .... Revision 1
2.6-305 .... Revision 1
2.6-306 .... Revision 1
2.6-307 .... Revision 1
2.6-308 .... Revision 1
2.6-309 .... Revision 1
2.6-310 .... Revision 1
2.6-311 .... Revision 1
2.6-312 .... Revision 1
2.6-313 .... Revision 1
2.6-314 .... Revision 1
2.6-315 .... Revision 1
2.6-316 .... Revision 1
2.6-317 .... Revision 1
2.6-318 .... Revision 1
2.6-319 .... Revision 1
2.6-320 .... Revision 1
2.6-321 .... Revision 1
2.6-322 .... Revision 1
2.6-323 .... Revision 1
2.6-324 .... Revision 1
2.6-325 .... Revision 1
2.6-326 .... Revision 1
2.6-327 .... Revision 1
2.6-328 .... Revision 1
2.6-329 .... Revision 1
2.6-330 .... Revision 1
2.6-331 .... Revision 1
2.6-332 .... Revision 1
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2.6-357 .... Revision 1
2.6-358 .... Revision 1
2.6-359 .... Revision 1
2.6-360 .... Revision 1

List of Effective Pages (Continued)

2.6-361 .... Revision 1
2.6-362 .... Revision 1
2.6-363 .... Revision 1
2.6-364 .... Revision 1
2.6-365 .... Revision 1
2.6-366 .... Revision 1
2.6-367 .... Revision 1
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2.6-376 .... Revision UMST-04A
2.6-377 .... Revision 1
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2.6-381 .... Revision 1
2.6-382 .... Revision 1
2.7-1 .... Revision 1
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2.7-40 .... Revision 1
2.7-41 .... Revision 1
2.7-42 .... Revision 1
2.7-43 .... Revision 1
2.7-44 .... Revision 1

List of Effective Pages (Continued)

2.7-45 .... Revision 1
2.7-46 .... Revision 1
2.7-47 .... Revision 1
2.7-48 .... Revision 1
2.7-49 .... Revision 1
2.7-50 .... Revision 1
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2.7-64 .... Revision UMST-04A
2.7-65 .... Revision UMST-04A
2.7-66 .... Revision 1
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2.7-75 .... Revision 1
2.7-76 .... Revision 1
2.7-77 .... Revision UMST-04A
2.7-78 .... Revision UMST-04A
2.7-79 .... Revision UMST-04A
2.7-80 .... Revision UMST-04A
2.7-81 .... Revision UMST-04A
2.7-82 .... Revision UMST-04A
2.7-83 .... Revision UMST-04A
2.7-84 .... Revision UMST-04A
2.7-85 .... Revision UMST-04A
2.7-86 .... Revision UMST-04A
2.7-87 .... Revision UMST-04A
2.7-88 .... Revision UMST-04A
2.7-89 .... Revision UMST-04A
2.7-90 .... Revision 1
2.7-91 .... Revision 1
2.7-92 .... Revision 1
2.7-93 .... Revision UMST-04A
2.7-94 .... Revision 1
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2.7-106 .... Revision 1
2.7-107 .... Revision 1
2.7-108 .... Revision 1
2.7-109 .... Revision 1
2.7-110 .... Revision 1

List of Effective Pages (Continued)

2.7-111 .... Revision 1
2.7-112 .... Revision 1
2.7-113 .... Revision 1
2.7-114 .... Revision 1
2.7-115 .... Revision 1
2.7-116 .... Revision 1
2.7-117 .... Revision 1
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2.7-123 .... Revision 1
2.7-124 .... Revision 1
2.7-125 .... Revision 1
2.7-126 .... Revision 1
2.7-127 .... Revision UMST-04A
2.7-128 .... Revision 1
2.7-129 .... Revision 1
2.7-130 .... Revision UMST-04A
2.7-131 .... Revision 1
2.7-132 .... Revision UMST-04A
2.7-133 .... Revision UMST-04A
2.7-134 .... Revision 1
2.7-135 .... Revision 1
2.7-136 .... Revision 1
2.7-137 .... Revision 1
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2.7-151 .... Revision 1
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2.7-155 .... Revision UMST-04A
2.7-156 .... Revision 1
2.7-157 .... Revision 1
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2.7-166 .... Revision 1
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2.7-170 .... Revision 1
2.7-171 .... Revision 1
2.7-172 .... Revision 1
2.7-173 .... Revision 1
2.7-174 .... Revision 1
2.7-175 .... Revision 1
2.7-176 .... Revision 1

List of Effective Pages (Continued)

2.7-177 .... Revision 1
2.7-178 .... Revision 1
2.7-179 .... Revision 1
2.7-180 .... Revision 1
2.7-181 .... Revision 1
2.7-182 .... Revision 1
2.7-183 .... Revision 1
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2.7-190 .... Revision 1
2.7-191 .... Revision 1
2.7-192 .... Revision 1
2.7-193 .... Revision UMST-04A
2.7-194 .... Revision 1
2.7-195 .... Revision UMST-04A
2.7-196 .... Revision UMST-04A
2.7-197 .... Revision 1
2.7-198 .... Revision 1
2.7-199 .... Revision 1
2.7-200 .... Revision 1
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2.7-209 .... Revision 1
2.7-210 .... Revision 1
2.7-211 .... Revision 1
2.7-212 .... Revision 1
2.7-213 .... Revision 1
2.8-1 .... Revision 1
2.9-1 .... Revision 1
2.9-2 .... Revision 1
2.9-3 .... Revision 1
2.9-4 .... Revision 1
2.9-5 .... Revision 1
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2.10.2-5 .... Revision 1
2.10.2-6 .... Revision 1
2.10.2-7 .... Revision 1
2.10.2-8 .... Revision 1
2.10.2-9 .... Revision 1

List of Effective Pages (Continued)

2.10.2-10 .... Revision 1
2.10.2-11 .... Revision 1
2.10.2-12 .... Revision 1
2.10.2-13 .... Revision 1
2.10.2-14 .... Revision 1
2.10.2-15 .... Revision 1
2.10.2-16 .... Revision 1
2.10.2-17 .... Revision 1
2.10.3-1 .... Revision 1
2.10.3-2 .... Revision 1
2.10.3-3 .... Revision 1
2.10.3-4 .... Revision 1
2.10.3-5 .... Revision 1
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2.10.3-11 .... Revision 1
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2.10.3-27 .... Revision 1
2.10.3-28 .... Revision 1
2.10.3-29 .... Revision 1
2.10.3-30 .... Revision 1
2.10.3-31 .... Revision 1
2.10.3-32 .... Revision 1
2.10.3-33 .... Revision 1
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2.10.3-35 .... Revision 1
2.10.3-36 .... Revision 1
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2.10.3-38 .... Revision 1
2.10.3-39 .... Revision 1
2.10.3-40 .... Revision 1
2.10.3-41 .... Revision 1
2.10.3-42 .... Revision 1
2.10.3-43 .... Revision 1
2.10.3-44 .... Revision 1
2.10.3-45 .... Revision 1
2.10.3-46 .... Revision 1
2.10.3-47 .... Revision 1
2.10.3-48 .... Revision 1

7 Model Drawings
Shown at the end of Section 2.10.3.7

2.10.4-1 .... Revision 1
2.10.4-2 .... Revision 1
2.10.4-3 .... Revision 1
2.10.4-4 .... Revision 1
2.10.4-5 .... Revision 1
2.10.4-6 .... Revision 1

List of Effective Pages (Continued)

2.10.4-7 .... Revision 1
2.10.4-8 .... Revision 1
2.10.4-9 .... Revision 1
2.10.4-10 .... Revision 1
2.10.4-11 .... Revision 1
2.10.4-12 .... Revision 1
2.10.4-13 .... Revision 1
2.10.4-14 .... Revision 1
2.10.4-15 .... Revision 1
2.11-1 .... Revision 1
2.11.1-1 .... Revision 1
2.11.1-2 .... Revision 1
2.11.1-3 .... Revision 1
2.11.1-4 .... Revision UMST-04A
2.11.1-5 .... Revision UMST-04A
2.11.1-6 .... Revision UMST-04A
2.11.1-7 .... Revision UMST-04A
2.11.1-8 .... Revision 1
2.11.1-9 .... Revision 1
2.11.1-10 .... Revision UMST-04A
2.11.1-11 .... Revision UMST-04A
2.11.1-12 .... Revision UMST-04A
2.11.1-13 .... Revision UMST-04A
2.11.1-14 .... Revision UMST-04A
2.11.1-15 .... Revision UMST-04A
2.11.1-16 .... Revision 1
2.11.1-17 .... Revision 1
2.11.1-18 .... Revision 1
2.11.1-19 .... Revision 1
2.11.1-20 .... Revision 1
2.11.1-21 .... Revision 1
2.11.1-22 .... Revision UMST-04A
2.11.2-1 .... Revision 1
2.11.2-2 .... Revision 1
2.11.2-3 .... Revision 1
2.11.2-4 .... Revision 1
2.11.2-5 .... Revision 1
2.11.2-6 .... Revision 1
2.11.2-7 .... Revision 1
2.11.2-8 .... Revision 1
2.11.2-9 .... Revision 1
2.11.2-10 .... Revision 1
2.11.2-11 .... Revision 1
2.11.2-12 .... Revision 1
2.11.2-13 .... Revision 1
2.11.2-14 .... Revision 1
2.11.2-15 .... Revision 1
2.11.2-16 .... Revision 1
2.11.2-17 .... Revision 1
2.11.2-18 .... Revision 1
2.11.2-19 .... Revision 1
2.11.2-20 .... Revision 1
2.11.2-21 .... Revision 1
2.11.2-22 .... Revision 1
2.11.2-23 .... Revision 1
2.11.2-24 .... Revision 1
2.11.2-25 .... Revision 1
2.11.2-26 .... Revision 1
2.11.2-27 .... Revision 1
2.11.2-28 .... Revision 1
2.12-1 .... Revision 1
2.12-2 .... Revision 1
2.12-3 .... Revision 1
2.12-4 .... Revision 1
2.12-5 .... Revision 1

List of Effective Pages (Continued)

Chapter 3

3-i .... Revision 1
3-ii .... Revision 1
3-iii .... Revision UMST-04A
3-iv .... Revision 1
3-v .... Revision 1
3-vi .... Revision UMST-04B
3-vii .... Revision 1
3-viii .... Revision 1
3.1-1 .... Revision 1
3.1-2 .... Revision 1
3.1-3 .... Revision 1
3.1-4 .... Revision 1
3.1-5 .... Revision 1
3.1-6 .... Revision 1
3.2-1 .... Revision 1
3.2-2 .... Revision 1
3.2-3 .... Revision 1
3.2-4 .... Revision 1
3.2-5 .... Revision 1
3.2-6 .... Revision 1
3.2-7 .... Revision 1
3.2-8 .... Revision 1
3.2-9 .... Revision 1
3.2-10 .... Revision 1
3.2-11 .... Revision 1
3.2-12 .... Revision 1
3.2-13 .... Revision 1
3.2-14 .... Revision 1
3.2-15 .... Revision 1
3.2-16 .... Revision 1
3.2-17 .... Revision 1
3.2-18 .... Revision 1
3.2-19 .... Revision UMST-04B
3.3-1 .... Revision 1
3.3-2 .... Revision 1
3.3-3 .... Revision 1
3.4-1 .... Revision 1
3.4-2 .... Revision 1
3.4-3 .... Revision UMST-04A
3.4-4 .... Revision UMST-04A
3.4-5 .... Revision UMST-04A
3.4-6 .... Revision 1
3.4-7 .... Revision UMST-04A
3.4-8 .... Revision 1
3.4-9 .... Revision 1
3.4-10 .... Revision 1
3.4-11 .... Revision 1
3.4-12 .... Revision 1
3.4-13 .... Revision UMST-04A
3.4-14 .... Revision UMST-04A
3.4-15 .... Revision 1
3.4-16 .... Revision UMST-04A
3.4-17 .... Revision UMST-04A
3.4-18 .... Revision UMST-04A
3.4-19 .... Revision UMST-04A
3.4-20 .... Revision UMST-04A
3.4-21 .... Revision UMST-04A
3.4-22 .... Revision UMST-04A
3.4-23 .... Revision 1
3.4-24 .... Revision 1
3.4-25 .... Revision 1
3.4-26 .... Revision 1
3.4-27 .... Revision 1

List of Effective Pages (Continued)

3.4-28 .... Revision 1
3.4-29 .... Revision 1
3.4-30 .... Revision 1
3.4-31 .... Revision 1
3.4-32 .... Revision 1
3.4-33 .... Revision 1
3.4-34 .... Revision 1
3.4-35 .... Revision 1
3.4-36 .... Revision 1
3.4-37 .... Revision 1
3.4-38 .... Revision 1
3.4-39 .... Revision 1
3.4-40 .... Revision 1
3.4-41 .... Revision 1
3.4-42 .... Revision 1
3.4-43 .... Revision UMST-04A
3.4-44 .... Revision UMST-04A
3.4-45 .... Revision 1
3.4-46 .... Revision 1
3.4-47 .... Revision 1
3.4-48 .... Revision 1
3.4-49 .... Revision 1
3.4-50 .... Revision 1
3.4-51 .... Revision 1
3.4-52 .... Revision 1
3.4-53 .... Revision 1
3.4-54 .... Revision 1
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3.4-56 .... Revision 1
3.4-57 .... Revision 1
3.4-58 .... Revision 1
3.4-59 .... Revision 1
3.4-60 .... Revision 1
3.5-1 .... Revision 1
3.5-2 .... Revision 1
3.5-3 .... Revision 1
3.5-4 .... Revision 1
3.5-5 .... Revision 1
3.5-6 .... Revision 1
3.5-7 .... Revision 1
3.5-8 .... Revision 1
3.5-9 .... Revision 1
3.5-10 .... Revision 1
3.5-11 .... Revision 1
3.5-12 .... Revision 1
3.5-13 .... Revision 1
3.5-14 .... Revision 1
3.5-15 .... Revision 1
3.5-16 .... Revision 1
3.5-17 .... Revision 1
3.5-18 .... Revision 1
3.5-19 .... Revision 1
3.5-20 .... Revision 1
3.5-21 .... Revision 1
3.5-22 .... Revision 1
3.5-23 .... Revision 1
3.5-24 .... Revision 1
3.5-25 .... Revision 1
3.5-26 .... Revision 1
3.5-27 .... Revision 1
3.5-28 .... Revision 1
3.6-1 .... Revision UMST-04A
3.6-2 .... Revision 1
3.6-3 .... Revision 1
3.6-4 .... Revision 1
3.6-5 .... Revision 1

List of Effective Pages (Continued)

3.6-6	Revision 1
3.6-7	Revision UMST-04A
3.6-8	Revision UMST-04B
3.6-9	Revision 1
3.6-10	Revision 1
3.6-11	Revision 1
3.6-12	Revision 1
3.6-13	Revision 1
3.6-14	Revision 1
3.6-15	Revision 1
3.6-16	Revision 1
3.7-1	Revision 1
3.7-2	Revision 1
3.7-3	Revision 1

Chapter 4

4-i	Revision 1
4-ii	Revision 1
4-iii	Revision 1
4-iv	Revision 1
4-1	Revision 1
4-2	Revision 1
4.1-1	Revision 1
4.1-2	Revision 1
4.1-3	Revision 1
4.1-4	Revision 1
4.1-5	Revision 1
4.1-6	Revision 1
4.1-7	Revision 1
4.2-1	Revision 1
4.2-2	Revision 1
4.2-3	Revision 1
4.2-4	Revision 1
4.2-5	Revision UMST-04B
4.2-6	Revision 1
4.2-7	Revision 1
4.2-8	Revision 1
4.2-9	Revision 1
4.2-10	Revision 1
4.2-11	Revision 1
4.3-1	Revision 1
4.3-2	Revision 1
4.3-3	Revision 1
4.3-4	Revision 1
4.3-5	Revision 1
4.4-1	Revision 1
4.5-1	Revision 1
4.5.1-1	Revision UMST-04B
4.5.2-1	Revision 1
4.5.2-2	Revision 1
4.5.2-3	Revision 1
4.5.2-4	Revision 1
4.5.2-5	Revision 1
4.5.2-6	Revision 1
4.5.2-7	Revision 1
4.5.2-8	Revision 1
4.5.2-9	Revision 1
4.5.2-10	Revision 1
4.5.2-11	Revision 1
4.5.3-1	Revision 1
4.5.3-2	Revision 1
4.5.3-3	Revision 1
4.5.3-4	Revision 1
4.5.3-5	Revision 1
4.5.3-6	Revision 1

List of Effective Pages (Continued)

4.5.3-7 .... Revision 1
4.5.3-8 .... Revision 1
4.5.3-9 .... Revision 1
4.5.3-10 .... Revision 1
4.5.3-11 .... Revision 1
4.5.3-12 .... Revision 1
4.5.3-13 .... Revision 1
4.6-1 .... Revision 1

Chapter 5

5-i .... Revision 1
5-ii .... Revision 1
5-iii .... Revision 1
5-iv .... Revision 1
5-v .... Revision 1
5-vi .... Revision 1
5-vii .... Revision 1
5-viii .... Revision 1
5-ix .... Revision 1
5-1 .... Revision 1
5-2 .... Revision 1
5-3 .... Revision 1
5.1-1 .... Revision UMST-04A
5.1-2 .... Revision 1
5.1-3 .... Revision 1
5.1-4 .... Revision 1
5.1-5 .... Revision 1
5.1-6 .... Revision 1
5.1-7 .... Revision 1
5.1-8 .... Revision 1
5.1-9 .... Revision 1
5.1-10 .... Revision 1
5.2-1 .... Revision 1
5.2-2 .... Revision 1
5.2-3 .... Revision 1
5.2-4 .... Revision 1
5.2-5 .... Revision 1
5.2-6 .... Revision 1
5.2-7 .... Revision 1
5.2-8 .... Revision 1
5.2-9 .... Revision 1
5.2-10 .... Revision 1
5.2-11 .... Revision 1
5.2-12 .... Revision 1
5.2-13 .... Revision 1
5.2-14 .... Revision 1
5.2-15 .... Revision 1
5.2-16 .... Revision 1
5.2-17 .... Revision 1
5.2-18 .... Revision 1
5.2-19 .... Revision 1
5.2-20 .... Revision 1
5.2-21 .... Revision 1
5.2-22 .... Revision 1
5.2-23 .... Revision 1
5.2-24 .... Revision 1
5.2-25 .... Revision 1
5.2-26 .... Revision 1
5.3-1 .... Revision 1
5.3-2 .... Revision 1
5.3-3 .... Revision 1
5.3-4 .... Revision 1
5.3-5 .... Revision 1
5.3-6 .... Revision 1
5.3-7 .... Revision 1

List of Effective Pages (Continued)

5.3-8	Revision 1	5.4-7	Revision 1
5.3-9	Revision 1	5.4-8	Revision 1
5.3-10	Revision 1	5.4-9	Revision 1
5.3-11	Revision 1	5.4-10	Revision 1
5.3-12	Revision 1	5.4-11	Revision 1
5.3-13	Revision 1	5.4-12	Revision 1
5.3-14	Revision 1	5.4-13	Revision 1
5.3-15	Revision 1	5.4-14	Revision 1
5.3-16	Revision 1	5.4-15	Revision 1
5.3-17	Revision 1	5.4-16	Revision 1
5.3-18	Revision 1	5.4-17	Revision 1
5.3-19	Revision 1	5.4-18	Revision 1
5.3-20	Revision 1	5.4-19	Revision 1
5.3-21	Revision 1	5.4-20	Revision 1
5.3-22	Revision 1	5.4-21	Revision 1
5.3-23	Revision 1	5.4-22	Revision 1
5.3-24	Revision 1	5.4-23	Revision 1
5.3-25	Revision 1	5.4-24	Revision 1
5.3-26	Revision 1	5.4-25	Revision 1
5.3-27	Revision 1	5.4-26	Revision 1
5.3-28	Revision 1	5.5-1	Revision 1
5.3-29	Revision 1	5.5.1-1	Revision 1
5.3-30	Revision 1	5.5.1-2	Revision 1
5.3-31	Revision 1	5.5.1-3	Revision 1
5.3-32	Revision 1	5.5.1-4	Revision 1
5.3-33	Revision 1	5.5.1-5	Revision 1
5.3-34	Revision 1	5.5.1-6	Revision 1
5.4-1	Revision 1	5.5.1-7	Revision 1
5.4-2	Revision 1	5.5.1-8	Revision 1
5.4-3	Revision 1	5.5.1-9	Revision 1
5.4-4	Revision 1	5.5.1-10	Revision 1
5.4-5	Revision 1	5.5.1-11	Revision UMST-04A
5.4-6	Revision 1	5.5.1-12	Revision 1

List of Effective Pages (Continued)

5.5.1-13 .... Revision 1
5.5.1-14 .... Revision 1
5.5.1-15 .... Revision 1
5.5.1-16 .... Revision 1
5.5.1-17 .... Revision 1
5.5.1-18 .... Revision 1
5.5.1-19 .... Revision 1
5.5.1-20 .... Revision 1
5.5.1-21 .... Revision 1
5.5.1-22 .... Revision 1
5.5.1-23 .... Revision 1
5.5.1-24 .... Revision 1
5.5.1-25 .... Revision 1
5.5.1-26 .... Revision 1
5.5.1-27 .... Revision 1
5.5.1-28 .... Revision 1
5.5.1-29 .... Revision 1
5.5.1-30 .... Revision 1
5.5.1-31 .... Revision 1
5.5.1-32 .... Revision 1
5.5.1-33 .... Revision 1
5.5.1-34 .... Revision 1
5.5.1-35 .... Revision 1
5.5.1-36 .... Revision 1
5.5.1-37 .... Revision 1
5.5.1-38 .... Revision 1
5.5.1-39 .... Revision 1
5.5.1-40 .... Revision 1
5.5.1-41 .... Revision 1
5.5.2-1 .... Revision 1
5.5.2-2 .... Revision 1
5.5.2-3 .... Revision 1
5.5.3-1 .... Revision 1
5.5.3-2 .... Revision 1
5.5.3-3 .... Revision 1
5.5.3-4 .... Revision 1
5.5.3-5 .... Revision 1
5.5.3-6 .... Revision 1
5.5.3-7 .... Revision 1
5.5.3-8 .... Revision 1
5.5.3-9 .... Revision 1
5.5.3-10 .... Revision 1
5.5.3-11 .... Revision 1
5.5.3-12 .... Revision 1
5.5.3-13 .... Revision 1
5.5.3-14 .... Revision 1
5.5.3-15 .... Revision 1
5.5.3-16 .... Revision 1
5.5.3-17 .... Revision 1
5.5.3-18 .... Revision 1
5.5.3-19 .... Revision 1
5.5.3-20 .... Revision 1
5.5.3-21 .... Revision 1
5.5.3-22 .... Revision 1
5.5.3-23 .... Revision 1
5.5.3-24 .... Revision 1
5.5.3-25 .... Revision 1
5.5.3-26 .... Revision 1
5.5.3-27 .... Revision 1
5.5.3-28 .... Revision 1
5.5.3-29 .... Revision 1
5.5.3-30 .... Revision 1
5.5.3-31 .... Revision 1
5.5.3-32 .... Revision 1
5.5.3-33 .... Revision 1
5.5.3-34 .... Revision 1

List of Effective Pages (Continued)

5.5.3-35 .... Revision 1
5.5.3-36 .... Revision 1
5.5.3-37 .... Revision 1
5.5.3-38 .... Revision 1
5.5.3-39 .... Revision 1
5.5.3-40 .... Revision 1
5.5.3-41 .... Revision 1
5.5.3-42 .... Revision 1
5.5.3-43 .... Revision 1
5.5.3-44 .... Revision 1
5.5.3-45 .... Revision 1
5.5.3-46 .... Revision 1
5.5.3-47 .... Revision 1
5.5.3-48 .... Revision 1
5.5.3-49 .... Revision 1
5.5.3-50 .... Revision 1
5.6-1 .... Revision 1
5.6-2 .... Revision 1

Chapter 6

6-i .... Revision 1
6-ii .... Revision 1
6-iii .... Revision 1
6-iv .... Revision 1
6-v .... Revision 1
6-vi .... Revision 1
6-vii .... Revision 1
6-viii .... Revision UMST-04A
6.1-1 .... Revision UMST-04A
6.1-2 .... Revision UMST-04A
6.1-3 .... Revision UMST-04A
6.2-1 .... Revision UMST-04A
6.2-2 .... Revision 1
6.2-3 .... Revision 1
6.2-4 .... Revision 1
6.3-1 .... Revision 1
6.3-2 .... Revision 1
6.3-3 .... Revision 1
6.3-4 .... Revision UMST-04A
6.3-5 .... Revision 1
6.3-6 .... Revision 1
6.3-7 .... Revision 1
6.3-8 .... Revision 1
6.3-9 .... Revision 1
6.3-10 .... Revision 1
6.3-11 .... Revision 1
6.3-12 .... Revision 1
6.3-13 .... Revision 1
6.3-14 .... Revision 1
6.4-1 .... Revision 1
6.4-2 .... Revision UMST-04A
6.4-3 .... Revision UMST-04A
6.4-4 .... Revision UMST-04A
6.4-5 .... Revision UMST-04A
6.4-6 .... Revision 1
6.4-7 .... Revision 1
6.4-8 .... Revision 1
6.4-9 .... Revision 1
6.4-10 .... Revision 1
6.4-11 .... Revision 1
6.4-12 .... Revision 1
6.4-13 .... Revision 1
6.4-14 .... Revision 1
6.4-15 .... Revision 1
6.4-16 .... Revision 1

List of Effective Pages (Continued)

6.4-17 .... Revision 1
6.4-18 .... Revision 1
6.4-19 .... Revision 1
6.4-20 .... Revision UMST-04B
6.4-21 .... Revision UMST-04B
6.4-22 .... Revision UMST-04B
6.4-23 .... Revision UMST-04B
6.4-24 .... Revision UMST-04B
6.4-25 .... Revision 1
6.4-26 .... Revision 1
6.4-27 .... Revision 1
6.4-28 .... Revision 1
6.4-29 .... Revision 1
6.4-30 .... Revision 1
6.4-31 .... Revision 1
6.4-32 .... Revision 1
6.4-33 .... Revision 1
6.4-34 .... Revision 1
6.4-35 .... Revision 1
6.4-36 .... Revision 1
6.4-37 .... Revision 1
6.4-38 .... Revision 1
6.4-39 .... Revision UMST-04A
6.5-1 .... Revision 1
6.5-2 .... Revision 1
6.5-3 .... Revision 1
6.5-4 .... Revision 1
6.5-5 .... Revision 1
6.5-6 .... Revision 1
6.5-7 .... Revision 1
6.5-8 .... Revision 1
6.5-9 .... Revision 1
6.5-10 .... Revision 1
6.5-11 .... Revision 1
6.5-12 .... Revision 1
6.5-13 .... Revision 1
6.5-14 .... Revision 1
6.5-15 .... Revision 1
6.5-16 .... Revision 1
6.5-17 .... Revision 1
6.5-18 .... Revision 1
6.5-19 .... Revision 1
6.5-20 .... Revision 1
6.5-21 .... Revision 1
6.5-22 .... Revision 1
6.5-23 .... Revision 1
6.5-24 .... Revision 1
6.5-25 .... Revision 1
6.5-26 .... Revision 1
6.5-27 .... Revision 1
6.5-28 .... Revision 1
6.5-29 .... Revision 1
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6.5-31 .... Revision 1
6.5-32 .... Revision 1
6.5-33 .... Revision 1
6.5-34 .... Revision 1
6.5-35 .... Revision 1
6.5-36 .... Revision 1
6.5-37 .... Revision 1
6.5-38 .... Revision 1
6.5-39 .... Revision 1
6.5-40 .... Revision 1
6.5-41 .... Revision 1
6.5-42 .... Revision 1
6.5-43 .... Revision 1

List of Effective Pages (Continued)

6.5-44 .... Revision 1
6.6-1 .... Revision 1
6.6.1-1 .... Revision 1
6.6.1-2 .... Revision 1
6.6.1-3 .... Revision 1
6.6.1-4 .... Revision 1
6.6.1-5 .... Revision 1
6.6.1-6 .... Revision 1
6.6.1-7 .... Revision UMST-04B
6.6.1-8 .... Revision UMST-04B
6.6.1-9 .... Revision UMST-04A
6.6.1-10 .... Revision 1
6.6.1-11 .... Revision 1
6.6.1-12 .... Revision 1
6.6.1-13 .... Revision 1
6.6.1-14 .... Revision 1
6.6.1-15 .... Revision 1
6.6.1-16 .... Revision 1
6.6.1-17 .... Revision 1
6.6.1-18 .... Revision 1
6.6.1-19 .... Revision 1
6.6.1-20 .... Revision 1
6.6.1-21 .... Revision 1
6.6.1-22 .... Revision UMST-04A
6.6.1-23 .... Revision 1
6.6.1-24 .... Revision 1
6.6.2-1 .... Revision 1
6.6.2-2 .... Revision 1
6.6.2-3 .... Revision 1
6.6.2-4 .... Revision 1
6.6.2-5 .... Revision 1
6.6.2-6 .... Revision 1
6.6.2-7 .... Revision 1
6.6.2-8 .... Revision 1
6.6.2-9 .... Revision 1
6.6.2-10 .... Revision 1
6.6.2-11 .... Revision 1
6.6.2-12 .... Revision 1
6.6.2-13 .... Revision 1
6.6.2-14 .... Revision 1
6.6.2-15 .... Revision 1
6.6.2-16 .... Revision 1
6.6.2-17 .... Revision 1
6.6.2-18 .... Revision 1
6.6.2-19 .... Revision 1
6.6.2-20 .... Revision 1
6.6.2-21 .... Revision 1
6.6.2-22 .... Revision 1
6.6.2-23 .... Revision 1
6.6.2-24 .... Revision 1
6.6.2-25 .... Revision 1
6.6.2-26 .... Revision 1
6.6.2-27 .... Revision 1
6.6.2-28 .... Revision 1
6.6.2-29 .... Revision 1
6.6.2-30 .... Revision 1
6.6.2-31 .... Revision 1
6.6.2-32 .... Revision 1
6.6.2-33 .... Revision 1
6.6.2-34 .... Revision 1
6.6.2-35 .... Revision 1
6.6.2-36 .... Revision 1
6.6.2-37 .... Revision 1
6.6.2-38 .... Revision 1
6.6.2-39 .... Revision 1
6.6.2-40 .... Revision 1

List of Effective Pages (Continued)

6.6.2-41 .... Revision 1
6.6.2-42 .... Revision 1
6.6.2-43 .... Revision 1
6.6.2-44 .... Revision 1
6.6.2-45 .... Revision 1
6.6.2-46 .... Revision 1
6.6.2-47 .... Revision 1
6.6.2-48 .... Revision 1
6.6.2-49 .... Revision 1
6.6.2-50 .... Revision 1
6.6.2-51 .... Revision 1
6.6.2-52 .... Revision 1
6.6.2-53 .... Revision 1
6.6.2-54 .... Revision 1
6.6.2-55 .... Revision 1
6.6.2-56 .... Revision 1
6.6.2-57 .... Revision 1
6.6.2-58 .... Revision 1
6.6.2-59 .... Revision 1
6.6.2-60 .... Revision 1
6.6.2-61 .... Revision 1
6.6.2-62 .... Revision 1
6.6.2-63 .... Revision 1
6.6.2-64 .... Revision 1
6.6.2-65 .... Revision 1
6.6.2-66 .... Revision 1
6.6.2-67 .... Revision 1
6.6.2-68 .... Revision 1
6.6.2-69 .... Revision 1
6.6.2-70 .... Revision 1
6.6.2-71 .... Revision 1
6.6.2-72 .... Revision 1
6.6.2-73 .... Revision 1
6.6.2-74 .... Revision 1
6.6.2-75 .... Revision 1
6.6.2-76 .... Revision 1
6.6.2-77 .... Revision 1
6.6.2-78 .... Revision 1
6.6.2-79 .... Revision 1
6.6.2-80 .... Revision 1
6.6.2-81 .... Revision 1
6.6.2-82 .... Revision 1
6.6.2-83 .... Revision 1
6.6.2-84 .... Revision 1
6.6.2-85 .... Revision 1
6.6.2-86 .... Revision 1
6.6.2-87 .... Revision 1
6.6.2-88 .... Revision 1
6.6.2-89 .... Revision 1
6.6.2-90 .... Revision 1
6.6.2-91 .... Revision 1
6.6.2-92 .... Revision 1
6.6.2-93 .... Revision 1
6.6.2-94 .... Revision 1
6.6.3-1 .... Revision 1
6.6.3-2 .... Revision 1
6.6.3-3 .... Revision 1
6.6.3-4 .... Revision 1
6.6.3-5 .... Revision 1
6.6.3-6 .... Revision 1
6.6.3-7 .... Revision 1
6.6.3-8 .... Revision 1
6.6.3-9 .... Revision 1
6.6.3-10 .... Revision 1
6.6.3-11 .... Revision 1
6.6.3-12 .... Revision 1

List of Effective Pages (Continued)

6.6.3-13 .... Revision 1
6.6.3-14 .... Revision 1
6.6.3-15 .... Revision 1
6.6.3-16 .... Revision 1
6.6.3-17 .... Revision 1
6.6.3-18 .... Revision 1
6.6.3-19 .... Revision 1
6.6.3-20 .... Revision 1
6.6.3-21 .... Revision 1
6.6.3-22 .... Revision 1
6.6.3-23 .... Revision 1
6.6.3-24 .... Revision 1
6.6.3-25 .... Revision 1
6.6.3-26 .... Revision 1
6.6.3-27 .... Revision 1
6.6.3-28 .... Revision 1
6.6.3-29 .... Revision 1
6.6.3-30 .... Revision 1
6.6.3-31 .... Revision 1
6.6.3-32 .... Revision 1
6.6.3-33 .... Revision 1
6.6.3-34 .... Revision 1
6.7-1 .... Revision 1
6.7-2 .... Revision 1

Chapter 7

7-i .... Revision 1
7-ii .... Revision 1
7-1 .... Revision 1
7-2 .... Revision 1
7-3 .... Revision 1
7.1-1 .... Revision 1
7.1-2 .... Revision 1
7.1-3 .... Revision 1
7.1-4 .... Revision UMST-04A
7.1-5 .... Revision UMST-04A
7.1-6 .... Revision 1
7.1-7 .... Revision 1
7.2-1 .... Revision 1
7.2-2 .... Revision 1
7.3-1 .... Revision 1
7.3-2 .... Revision 1
7.3-3 .... Revision UMST-04A
7.3-4 .... Revision UMST-04A
7.4-1 .... Revision 1
7.5-1 .... Revision 1

Chapter 8

8-i .... Revision 1
8-ii .... Revision 1
8-iii .... Revision 1
8-1 .... Revision 1
8.1-1 .... Revision 1
8.1-2 .... Revision 1
8.1-3 .... Revision 1
8.1-4 .... Revision 1
8.1-5 .... Revision 1
8.1-6 .... Revision 1
8.1-7 .... Revision 1
8.1-8 .... Revision 1
8.1-9 .... Revision 1
8.1-10 .... Revision 1
8.1-11 .... Revision 1
8.1-12 .... Revision 1

List of Effective Pages (Continued)

8.1-13 .... Revision 1
8.1-14 .... Revision 1
8.1-15 .... Revision 1
8.1-16 .... Revision 1
8.2-1 .... Revision 1
8.2-2 .... Revision 1
8.2-3 .... Revision 1
8.2-4 .... Revision 1
8.2-5 .... Revision 1
8.3-1 .... Revision 1
8.3-2 .... Revision 1
8.3-3 .... Revision 1
8.3-4 .... Revision 1
8.3-5 .... Revision 1
8.3-6 .... Revision 1
8.3-7 .... Revision 1
8.3-8 .... Revision 1

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Chapter 1

Table 1-1 Terminology (continued)

-spacer	Installed on the tie rod between the support disks (BWR only) or between the support disks and upper and lower weldments (BWR and PWR) to properly position the disks and provide axial support for the support disks.
- split spacer	Installed on the tie rod between the support and heat transfer disks to properly position the disks and provide axial support for the support disks and heat transfer disks in the PWR and BWR baskets.
Canister spacer	Stainless steel or aluminum components that position the canister in the Universal Transport Cask cavity during transport. Spacers are used for canister Classes 1, 2, 4, or 5.
Transfer cask	A shielded lifting device used for handling of the Transportable Storage Canister during loading of spent fuel or GTCC waste, canister closure operations, and transfer of the canister into or out of the Universal Transport Cask, or into or out of the vertical concrete cask during storage operations. The Transfer Cask is described in the Safety Analysis Report for the onsite storage (10 CFR 72) components of the UMS®, Docket No. 72-1015.
Vertical concrete cask	The cask used to store the Transportable Storage Canister containing spent fuel or GTCC waste. The Vertical Concrete Cask is described in the Safety Analysis Report for the on-site storage (10 CFR 72) components of the UMS®, Docket No. 72-1015.
Maine Yankee Fuel Can	A specially designed stainless steel screened can sized to hold an intact fuel assembly, consolidated fuel or damaged fuel. The can screens permit draining and drying, while precluding the release of gross particulates into the canister cavity. The Maine Yankee Fuel Can may only be loaded into a Class 1 Canister.

Table 1-1 Terminology (continued)

High Burnup Fuel

A Maine Yankee fuel assembly having a burnup between 45,000 and 50,000 MWD/MTU, which must be preferentially loaded in periphery positions in the basket.

Damaged Fuel

Spent fuel that manifests any of the following conditions that result in either compromise of cladding confinement integrity or rearrangement (reconfiguration) of fuel bundle geometry:

1) A fuel assembly that contains known or suspected cladding defects greater than pinhole leaks or hairline cracks that have the potential for release of significant amounts of fuel particles into the cask.

2) A fuel assembly that:
   a. is damaged in such a manner as to impair its structural integrity;
   b. has missing or displaced structural components such as grid spacers; or
   c. cannot be handled using normal (i.e., crane and grapple) handling methods.

3) Fuel that is no longer in the form of an intact fuel assembly and consists of, or contains, debris such as loose fuel pellets, rod segments, etc.

Except as described in the following discussion, the only damaged fuel that is authorized for storage and transport in the NAC-UMS® System is Maine Yankee Site-Specific PWR fuel. Damaged fuel is placed in a Maine Yankee Fuel Can and loaded into a basket corner position. Note that PWR intact fuel assemblies with one or more grid spacers missing, or damaged such that the unsupported length of the fuel rods does not exceed 60 inches, or with end fitting damage such as damaged or missing hold-down springs are authorized to be loaded as intact fuel as long as they can be handled safely by normal means.

1.2.1.2.7 PWR Fuel Basket

The PWR fuel basket is contained within the Transportable Storage Canister and is constructed of Type 304 stainless steel. The fuel basket design is a right-circular cylinder configuration with square fuel tubes laterally supported by a series of support disks. The basket design parameters for the transport of the three classes of PWR fuel are provided in Table 1.2-3. The Class 1, 2 or 3 fuel baskets incorporate 30, 32 or 34 support disks, respectively. The disks are retained by a top nut and supported by spacers on tie rods at eight locations. The top nut is torqued at installation to provide a solid load path in compression between the support disks. The 0.50-in. thick, 65.50-in. diameter support disks are fabricated of SA 693, Type 630, 17-4PH stainless steel. The disks are spaced axially at 4.92 in. center-to-center and contain square holes for the fuel tubes.

The 1.25-in. thick top and 1.0-in. thick bottom weldments are fabricated from Type 304 stainless steel and are geometrically similar to the support disks. The tie rods have a 2.5-in. length of 1 5/8 - 8 UN thread at the upper end and are fabricated from SA-479, Type 304 stainless steel. The top nut is fabricated from a 3.5-in. diameter bar, and the spacers are fabricated from 2.5-in. pipe XXS, Type 304 stainless steel. The fuel tubes are fabricated from A-240, Type 304 stainless steel and support the encased neutron absorber sheet on each of the four sides. The neutron absorber provides criticality control in the basket. No structural credit is taken for the stainless steel tubes for structural strength of the basket.

Each PWR fuel basket can hold 24 PWR fuel assemblies in an aligned configuration in 8.80-in. ID square fuel tubes, which have 0.141-in. thick walls (including neutron absorber and cladding). The hole in the top weldment is 8.75 in. by 8.75 inches. The hole in the bottom weldment is 8.65 in. by 8.65 inches. The flange outer surface at the top of the fuel tube is 9.65 in. by 9.65 inches. The basket design traps the fuel tube between the top and bottom weldments, thereby preventing axial movement of the fuel tube. The fuel tubes are separated by webs in the support disks with widths of 0.9 in., 1.0 in., and 1.5 in., depending on location.

The PWR basket design incorporates Type 6061-T651 aluminum alloy heat transfer disks to enhance heat transfer in the basket. Thirty support disks are contained in the Class 1 fuel basket. Class 2 and 3 fuel baskets contain 32 and 34 disks, respectively. The heat transfer disks are spaced and supported by the tie rods and spacers, which also support and locate the support disks. The heat transfer disks, located at the center of the axial spacing between the support disks, are sized to eliminate contact with the cask inner shell and basket tie rods due to differential thermal expansion.

The Transportable Storage Canister is designed to facilitate filling with water and subsequent draining. Water fills and drains freely between the basket disks through three separate paths. One path is the gaps that exist between the disks and canister shell. The second path is through the gaps between the fuel tubes and disk that surrounds the fuel tubes. The third path is through three 1.3-in. diameter holes in each of the disks that are intended to provide additional paths for water flow between disks. The basket bottom weldment supports the fuel tubes above the canister bottom plate. The fuel tubes are open at the top and bottom ends, allowing the free flow of water from the bottom of the fuel tube. The bottom weldment is positioned by supports 1.0 inch above the canister bottom to facilitate water flow to the drain line. These design features ensure that water flows freely in the basket so that the canister fills and drains freely.

1.2.1.2.8 BWR Fuel Basket

Like the PWR fuel basket, the BWR basket is contained within the stainless steel Transportable Storage Canister. The BWR fuel basket is also a right-circular cylinder configuration with square fuel tubes laterally supported by a series of support disks (40 disks for the Class 4 fuel basket and 41 disks for the Class 5 fuel basket). The basket design parameters for the transport of the two classes of BWR fuel are provided in Table 1.2-3. The support disks are retained by cylindrical spacers on tie rods at six locations. The top nut is torqued at installation to provide a solid load path in compression between the support disks. The 0.625-in. thick, 65.50-in. diameter support disks are fabricated of SA 533, Type B, Class 2 carbon steel. The disks are spaced axially at 3.8-in. center-to-center and contain square holes for the fuel tubes.

The 1.0-in. thick top and bottom weldments are fabricated from Type 304 stainless steel, and are geometrically similar to the support disks. The tie rods have a 2.5-in. length of 1 5/8 - 8 UN thread at the upper end and are fabricated from Type 304 stainless steel. The top nut and spacers are fabricated from 3-in. diameter bars of Type 304 stainless steel. The fuel tubes are also fabricated from Type 304 stainless steel and support the encased neutron absorber sheet. Three types of tubes are designed to contain BWR fuel: tubes with neutron absorber on two sides, tubes with neutron absorber one side, and tubes with no neutron absorber. No structural credit is taken for the stainless steel tubes for structural strength of the basket or support of the fuel assemblies.

Table 1.2-1 Design Characteristics of the Universal Transport Cask and Components (continued)

Design Characteristic	Value (in)	Material
Fuel basket		
• PWR Top Weldment	1.25	Type 304 Stainless Steel
• PWR Bottom Weldment	1.0	Type 304 Stainless Steel
• BWR Top and Bottom Weldment	1.0	Type 304 Stainless Steel
• Support disks thickness		
- PWR	0.5	Type 17-4 PH Stainless Steel
- BWR	0.625	SA-533,Type B Class 2 Carbon Steel
• Heat transfer disk thickness	0.5	Type 6061-T651 Aluminum Alloy
• Fuel tube dimensions		
- PWR (inside)	8.8 x 8.8	Type 304 Stainless Steel Encasing neutron absorber
- BWR Standard (inside)	5.9 x 5.9	Type 304 Stainless Steel Encasing neutron absorber
- BWR Enlarged (inside)	6.05 x 6.05	Type 304 Stainless Steel Encasing neutron absorber
• Spacers(s) diameter	2.875	Type 304 Stainless Steel
• Tie rod diameter		
PWR	1-5/8	Type 304 Stainless Steel
BWR	1-5/8	Type 304 Stainless Steel

Table 1.2-2 Transportable Storage Canister Design Parameters

Canister Parameter	Value
Internal fuel cavity length (in)	
Class 1 (PWR)	163.3
Class 2 (PWR)	172.4
Class 3 (PWR)	180.0
Class 4 (BWR)	173.8
Class 5 (BWR)	178.6
Overall length (in)	
Class 1 (PWR)	175.1
Class 2 (PWR)	184.2
Class 3 (PWR)	191.8
Class 4 (BWR)	185.6
Class 5 (BWR)	190.4
Internal cavity diameter (in)	65.8
Fuel cell opening (in-square) top weldment	
Classes – 1 - 3	8.75
Classes – 4 - 5 (Standard)	5.75
Classes – 4 - 5 (Oversized — 4 places)	5.90
Shell (in)	
Outer Diameter	67.06
Thickness	0.625
Structural lid thickness (in)	3
Shield lid thickness (in)	7
Neutron absorber	0.011 g/cm^2 ^{10}B (BWR) 0.025 g/cm^2 ^{10}B (PWR)

1.3.4 License Drawings

This section contains the License Drawings pertinent to the Universal Transport Cask. The dimensions indicated on the drawings are generally limited to one significant digit past the decimal point. Note that analysis of systems or components may present dimensions with additional significant digits based on more detailed engineering drawings.

Drawing No.	Rev. No.	Title
790-209	1	Impact Limiter Assembly-Upper, Cask, NAC-UMS®
790-210	1	Impact Limiter Assembly-Lower, Cask, NAC-UMS®
790-500	4	Assembly, Universal Transport Cask, Overpack, NAC-UMS®
790-501	3	Canister/Basket Assembly Table, NAC-UMS®
790-502	6	Cask Body, Transport Cask, NAC-UMS®
790-503	2	Lid Assembly, NAC-UMS® Cask
790-504	2	Port Coverplate Assembly, NAC-UMS®
790-505	2	Lifting Trunnion, NAC-UMS®
790-508	2	Misc. Details, Transport Cask, NAC-UMS®
790-509	2	Nameplates - NAC-UMS®
790-516	2	Package Assembly, Universal Transport Cask (UTC), NAC-UMS®
790-519	1	Package Assembly, Transport, Universal Transport Cask (UTC), NAC-UMS®
790-520	2	Spacers, Universal Transport Cask (UTC), NAC-UMS®
790-570	4	Fuel Basket Assembly, 56 Element BWR, NAC-UMS®
790-571	3	Bottom Weldment, Fuel Basket, 56 Element BWR, NAC-UMS®

Licensed Drawings (Continued)

Drawing No.	Rev. No.	Title
790-572	4	Top Weldment, Fuel Basket, 56 Element BWR, NAC-UMS®
790-573	7	Support Disk and Misc. Basket Details, 56 Element BWR, NAC-UMS®
790-574	3	Heat Transfer Disk Fuel Basket, 56 Element BWR, NAC-UMS®
790-575	10	BWR Fuel Tube, NAC-UMS®
790-581	9	PWR Fuel Tube, NAC-UMS®
790-582	11	Shell Weldment, Canister, NAC-UMS®
790-583	7	Assembly, Drain Tube, Canister, NAC-UMS®
790-584	17	Details, Canister, NAC-UMS®
790-585	16	Transportable Storage Canister (TSC), NAC-UMS®
790-587	1	Spacer Shim, Canister, NAC-UMS®
790-591	6	Bottom Weldment, Fuel Basket, 24 Element PWR, NAC-UMS®
790-592	8	Top Weldment, Fuel Basket, 24 Element PWR, NAC-UMS®
790-593	7	Support Disk and Misc. Basket Details, 24 Element PWR, NAC-UMS®
790-594	2	Heat Transfer Disk, Fuel Basket, 24 Element PWR, NAC-UMS®
790-595	9	Fuel Basket Assembly, 24 Element PWR, NAC-UMS®
790-605	11	BWR Fuel Tube, Over-Sized Fuel, NAC-UMS®
790-611	5	GTCC Waste Basket, Maine Yankee, NAC-UMS®
790-612	8	GTCC Waste Canister, Maine Yankee, NAC-UMS®
412-501	4	Spent Fuel Can Assembly, Maine Yankee (MY), NAC-UMS®
412-502	5	Fuel Can Details, Maine Yankee (MY), NAC-UMS®

FIGURE WITHHELD UNDER 10 CFR 2.390

UNLESS OTHERWISE STATED

DIMENSIONING AND TOLERANCING SHALL BE PER ASME Y14.5M-94, UNSPECIFIED DIMENSIONAL TOLERANCES ARE SHOWN BELOW.

ALL THREAD DEPTH CALLOUTS ARE TO BE CONSIDERED AS A MIN. DEPTH OF PERFECT THREADS. THE ACTUAL DEPTH OF THE THREADS IS NOT SUBJECT TO TOLERANCE CONTROLS.

.XXX	UNDER 3	±.003
.XXX	3–12	±.005
.XXX	OVER 12	±.010
.XX	UNDER 6	±.02
.XX	6–18	±.03
.XX	OVER 18	±.06
.X	ALL	±.1
FRACTIONAL		±1/8
ANGLES		±0.5°

WEIGHTS ARE APPROXIMATE AND ARE TO BE USED FOR HANDLING PURPOSES ONLY

ALL DIMENSIONS ARE IN INCHES

BORDER SIZE: F (40 X 28)

ALL UNSPECIFIED TOOL RADII: .015 – .030

BREAK ALL SHARP CORNERS .015 – .030

MACHINED SURFACES TO BE ∇ OR BETTER

NEXT ASSEMBLY: 790-570

DRAWING TYPE: LICENSE

GROUP	NAME	DATE
PREPARER	R. Walker	9-22-04
CHECKER	[illegible]	9-23-04
PROJECT MANAGER	[illegible]	9/23/04
ENGINEERING	[illegible]	9/24/04
LICENSING	[illegible]	9-23-04
QUALITY	[illegible]	9/23/04

BWR FUEL TUBE, NAC-UMS®

PROJECT 790 | DRAWING 575 | REV 10

SCALE 1/1 | WEIGHT | SH 1 OF 2 | 9:54AM 9-22-2004

FIGURE WITHHELD UNDER 10 CFR 2.390

NAC INTERNATIONAL			
BWR FUEL TUBE, NAC-UMS®			
PROJECT 790		DRAWING 575	REV 10
SCALE 1/1	WEIGHT	SH 2 OF 2	

A

FIGURE WITHHELD UNDER 10 CFR 2.390

UNLESS OTHERWISE STATED

DIMENSIONING AND TOLERANCING SHALL BE PER ASME Y14.5M-94. UNSPECIFIED DIMENSIONAL TOLERANCES ARE SHOWN BELOW.

ALL THREAD DEPTH CALLOUTS ARE TO BE CONSIDERED AS A MIN. DEPTH OF PERFECT THREADS. THE ACTUAL DEPTH OF THE THREADS IS NOT SUBJECT TO TOLERANCE CONTROLS.

.XXX	UNDER 3	±.003
	3-12	±.005
	OVER 12	±.010
.XX	UNDER 6	±.02
	6-18	±.03
	OVER 18	±.05
.X	ALL	±.1
FRACTIONAL ±1/8		
ANGLES ±0.5°		

WEIGHTS ARE APPROXIMATE AND ARE TO BE USED FOR HANDLING PURPOSES ONLY

ALL DIMENSIONS ARE IN INCHES

BORDER SIZE: F (40 X 28)

ALL UNSPECIFIED TOOL RADII: .015 - .030

BREAK ALL SHARP CORNERS .015 - .030

MACHINED SURFACES TO BE ▽ OR BETTER

NEXT ASSEMBLY: 790-595

DRAWING TYPE: LICENSE

GROUP	NAME	DATE
PREPARER	R. [illegible]	9-22-04
CHECKER	[illegible]	9-23-04
PROJECT MANAGER	[illegible]	9/23/04
ENGINEERING	[illegible]	9/23/04
LICENSING	JC Thompson	9-23-04
QUALITY	[illegible]	9/23/04

PWR FUEL TUBE,
NAC-UMS®

PROJECT 790 | DRAWING 581 | REV 9

SCALE FULL | WEIGHT | SH 1 OF 2 | 9:54AM 9-23-2004

FIGURE WITHHELD UNDER 10 CFR 2.390

NAC INTERNATIONAL				
PWR FUEL TUBE, NAC-UMS®				
PROJECT 790		DRAWING 581		REV 9
SCALE FULL	WEIGHT	SH 2 OF 2		8:34AM 9-22-2004

FIGURE WITHHELD UNDER 10 CFR 2.390

QUANTITY

UNLESS OTHERWISE STATED

DIMENSIONING AND TOLERANCING SHALL BE PER ASME Y14.5M-94. UNSPECIFIED DIMENSIONAL TOLERANCES ARE SHOWN BELOW.

ALL THREAD DEPTH CALLOUTS ARE TO BE CONSIDERED AS A MIN. DEPTH OF PERFECT THREADS. THE ACTUAL DEPTH OF THE THREADS IS NOT SUBJECT TO TOLERANCE CONTROLS.

.XXX	UNDER 3	±.003	WEIGHTS ARE APPROXIMATE AND ARE TO BE USED FOR HANDLING PURPOSES ONLY
	3-12	±.005	ALL DIMENSIONS ARE IN INCHES
	OVER 12	±.010	BORDER SIZE: F (40 X 28)
.XX	UNDER 6	±.02	ALL UNSPECIFIED TOOL RADII: .015 - .030
	6-18	±.03	BREAK ALL SHARP CORNERS .015 - .030
	OVER 18	±.06	MACHINED SURFACES TO BE ∇ OR BETTER
.X	ALL	±.1	NEXT ASSEMBLY: 790-570
FRACTIONAL ±1/8			
ANGLES ±0.5°			DRAWING TYPE: LICENSE

GROUP	NAME	DATE
PREPARER	R. Walker	9-22-04
CHECKER	[illegible]	9-23-04
PROJECT MANAGER	[illegible]	9/23/04
ENGINEERING	[illegible]	9/23/04
LICENSING	JC Thompson	9-23-04
QUALITY	[illegible]	9/23/04

NAC INTERNATIONAL

BWR FUEL TUBE,
OVER-SIZED FUEL,
NAC-UMS®

PROJECT 790		DRAWING 605	REV 11
SCALE 1/1	WEIGHT	SH 1 OF 2	8:54AM 9-22-2004

FIGURE WITHHELD UNDER 10 CFR 2.390

NAC INTERNATIONAL

BWR FUEL TUBE,
OVER-SIZED FUEL,
NAC-UMS®

PROJECT 790		DRAWING 605			REV 11
SCALE 1/1	WEIGHT	SH 2	OF 2		9:34AM 9-22-2004

FIGURE WITHHELD UNDER 10 CFR 2.390

UNLESS OTHERWISE STATED

DIMENSIONING AND TOLERANCING SHALL BE PER ASME Y14.5M-94. UNSPECIFIED DIMENSIONAL TOLERANCES ARE SHOWN BELOW.

ALL THREAD DEPTH CALLOUTS ARE TO BE CONSIDERED AS A MIN. DEPTH OF PERFECT THREADS. THE ACTUAL DEPTH OF THE THREADS IS NOT SUBJECT TO TOLERANCE CONTROLS.

.XXX	UNDER 3	±.003
	3-12	±.005
	OVER 12	±.010
.XX	UNDER 6	±.02
	6-18	±.03
	OVER 18	±.06
.X	ALL	±.1
FRACTIONAL		±1/8
ANGLES		±0.5°

WEIGHTS ARE APPROXIMATE AND ARE TO BE USED FOR HANDLING PURPOSES ONLY

ALL DIMENSIONS ARE IN INCHES

BORDER SIZE: F (40 X 28)

ALL UNSPECIFIED TOOL RADII: .015 - .030

BREAK ALL SHARP CORNERS .015 - .030

MACHINED SURFACES TO BE ▽ OR BETTER

NEXT ASSEMBLY: 412-501

DRAWING TYPE: LICENSE

GROUP	NAME	DATE
PREPARER	R. [illegible]	9-17-04
CHECKER	[illegible]	9-17-04
PROJECT MANAGER	[illegible]	9/21/04
ENGINEERING	[illegible]	9/21/04
LICENSING	[illegible]	9/22/04
QUALITY	[illegible]	9/23/04

FUEL CAN DETAILS
MAINE YANKEE (MY)
NAC-UMS®

PROJECT 412 | DRAWING 502 | REV 5

SCALE FULL | WEIGHT NOTED | SH 1 OF 6 | 10:01AM 9-17-2004

FIGURE WITHHELD UNDER 10 CFR 2.390

NAC INTERNATIONAL

FUEL CAN DETAILS
MAINE YANKEE (MY)
NAC-UMS®

PROJECT 412		DRAWING 502	REV 5
SCALE FULL	WEIGHT NOTED	SH 2 OF 6	[illegible] 9-16-2004

FIGURE WITHHELD UNDER 10 CFR 2.390

NAC INTERNATIONAL

FUEL CAN DETAILS
MAINE YANKEE (MY)
NAC–UMS®

PROJECT 412		DRAWING 502	REV 5
SCALE FULL	WEIGHT NOTED	SH 3 OF 6	

FIGURE WITHHELD UNDER 10 CFR 2.390

NAC INTERNATIONAL

FUEL CAN DETAILS
MAINE YANKEE (MY)
NAC-UMS®

PROJECT 412		DRAWING 502	REV 5
SCALE FULL	WEIGHT NOTED	SH 4 OF 6	9-16-2004

FIGURE WITHHELD UNDER 10 CFR 2.390

NAC INTERNATIONAL

FUEL CAN DETAILS MAINE YANKEE (MY) NAC–UMS®			
PROJECT 412		DRAWING 502	REV 5
SCALE FULL	WEIGHT NOTED	SH 5 OF 6	

FIGURE WITHHELD UNDER 10 CFR 2.390

NAC INTERNATIONAL			
FUEL CAN DETAILS MAINE YANKEE (MY) NAC–UMS®			
PROJECT 412		DRAWING 502	REV 5
SCALE FULL	WEIGHT NOTED	SH 6 OF 6	3:45PM 6-16-2004

Chapter 2

The bearing stress evaluation is presented for the regions under the shield lid and structural lid welds (see Sections 9, 10 and 11 in Figure 2.6.12.3-1) for the governing normal condition (one-foot side drop). The averaged stress components, Sx, in Sections 9, 10 and 11 of Table 2.6.12.6-2, are considered the bearing stresses at the weld region. The stress is averaged only for the region of contact between the canister shell and concrete cask inner shell. The region of contact is determined based on the stress component, S_x, obtained from the finite element analysis results at the circumferential locations of 0°, 9° and 18° (measured from the plane of symmetry of the finite element model). When S_x is in compression (or negative), the canister is in contact with the cask inner shell. The compressive stresses corresponding to the region of contact are averaged using the trapezoidal averaging method. As shown in Table 2.6.12.6-2, the maximum bearing stress ($\sigma_{bearing}$) of -17.1 ksi occurs at Section 11.

The margin of safety is:

$$MS = \frac{S_y}{\sigma_{bearing}} - 1 = +0.16$$

where:

S_y = 19.95 ksi, the yield stress of 304L at the peak temperature of 266°F in the lid region of the canister shell.

The ANSYS finite element model of the canister conservatively assumes that the cask inner shell is rigid. Therefore, the analysis underpredicts the actual contact between the canister shell and cask inner shell. With a larger contact area, stresses in the canister shell will be smaller. Therefore, the analysis results are conservative. In accordance with ASME Section III, NB-3227.1(a): *"The average bearing stress for resistance to crushing under the maximum load, experienced as a result of design loads, test loads, or any Service Loadings, except those for which Level D Limits are designated, shall be limited to S_y at temperature..."* Therefore, the bearing stress evaluation is based on the averaged compressive stresses over the contact region (determined by the conservative finite element model) between the canister shell and cask inner shell.

2.6.12.12 Canister Buckling Evaluation for 1-Foot End Drop

Code Case N-284-1 [12] of the ASME Boiler and Pressure Vessel Code is used to analyze the PWR canister for the normal condition 1-foot end drop (both top and bottom end drops). The evaluation requirements of Regulatory Guide 7.6, Paragraph C.5, are shown to be satisfied by the results of the buckling interaction equation calculations of Code Case N-284-1. The canister buckling design criteria are described in Section 2.1.2.5.3.

The data considered for the buckling evaluation includes shell geometry parameters, shell fabrication tolerances, shell material properties, theoretical elastic buckling stress values for the shell, and membrane stress components in the shell. The internal stress field that controls the buckling of a cylindrical shell consists of the longitudinal (axial) membrane, circumferential (hoop) membrane, and in-plane shear stresses. These stresses may exist singly or in combination, depending on the applied loading. Only these three stress components are considered in the buckling analysis.

A 20 g deceleration load was used for all the 1-ft drop canister analyses that are presented in Sections 2.6.12.4 through 2.6.12.9. The 20 g-load bounds all 1-ft deceleration loads for all other drop angles. The top- and bottom-end drops result in the largest potential for canister shell buckling and, therefore, are the two load cases presented here. The side drop load case is not considered a credible buckling mode of the canister shell and is, therefore, not presented here.

The stress results from the canister analysis are screened for the maximum values of the longitudinal compression, circumferential compression, or in-plane shear stresses for the 1-ft drop cases (top- and bottom-end drops) with and without pressure. For each loading case, the largest of each of the three stress components anywhere regardless of location within the PWR canister shell are combined. To these maximum stress components are added the maximum stresses from the hot and cold thermal cases (Tables 2.6.12.3-1 and 2.6.12.3-2). Combining the

September 2004
Revision UMST-04B

NAC-UMS

Universal Multi-Purpose Cask System

SAFETY ANALYSIS REPORT

Volume 2 of 2
Docket No. 71-9270

NAC INTERNATIONAL

Atlanta Corporate Headquarters: 3930 East Jones Bridge Road, Norcross, Georgia 30092 USA
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List of Effective Pages

List of Effective Pages

Master Table of Contents

i .... Revision 1
ii .... Revision 1
iii .... Revision 1
iv .... Revision 1
v.................................... Revision 1
vi .... Revision 1
vii .... Revision 1
viii .... Revision 1
ix .... Revision 1
x .... Revision 1
xi .... Revision 1
xii .... Revision 1
xiii .... Revision 1
xiv .... Revision 1
xv .... Revision 1
xvi .... Revision 1
xvii .... Revision 1
xviii .... Revision 1

Chapter 1

1-i .... Revision 1
1-ii .... Revision 1
1-1 .... Revision 1
1-2 .... Revision 1
1-3 .... Revision 1
1-4 .... Revision 1
1-5 .... Revision 1
1-6 .... Revision UMST-04A
1-7 .... Revision UMST-04B
1-8 .... Revision UMST-04B
1.1-1 .... Revision UMST-04A
1.1-2 .... Revision 1
1.1-3 .... Revision UMST-04A
1.1-4 .... Revision UMST-04A
1.1-5 .... Revision 1
1.1-6 .... Revision 1
1.2-1 .... Revision UMST-04A
1.2-2 .... Revision 1
1.2-3 .... Revision 1
1.2-4 .... Revision 1
1.2-5 .... Revision 1
1.2-6 .... Revision 1
1.2-7 .... Revision 1
1.2-8 .... Revision 1
1.2-9 .... Revision UMST-04B
1.2-10 .... Revision UMST-04B
1.2-11 .... Revision UMST-04A
1.2-12 .... Revision 1
1.2-13 .... Revision UMST-04A
1.2-14 .... Revision UMST-04A
1.2-15 .... Revision 1
1.2-16 .... Revision 1
1.2-17 .... Revision 1
1.2-18 .... Revision 1
1.2-19 .... Revision UMST-04A
1.2-20 .... Revision UMST-04B
1.2-21 .... Revision UMST-04A
1.2-22 .... Revision 1
1.2-23 .... Revision 1
1.2-24 .... Revision 1
1.2-25 .... Revision 1
1.3-1 .... Revision 1
1.3.1-1 .... Revision 1

List of Effective Pages (Continued)

1.3.1-2 .... Revision UMST-04A
1.3.1-3 .... Revision 1
1.3.1-4 .... Revision 1
1.3.1-5 .... Revision 1
1.3.1-6 .... Revision 1
1.3.1-7 .... Revision 1
1.3.1-8 .... Revision 1
1.3.1-9 .... Revision 1
1.3.1-10 .... Revision 1
1.3.1-11 .... Revision 1
1.3.1-12 .... Revision 1
1.3.2-1 .... Revision 1
1.3.3-1 .... Revision 1
1.3.4-1 .... Revision UMST-04A
1.3.4-2 .... Revision UMST-04B

License Drawings
35 drawings
Revised as shown in Chapter 1

Chapter 2

2-i .... Revision 1
2-ii .... Revision 1
2-iii .... Revision 1
2-iv .... Revision 1
2-v .... Revision 1
2-vi .... Revision 1
2-vii .... Revision 1
2-viii .... Revision 1
2-ix .... Revision 1
2-x .... Revision 1
2-xi .... Revision 1
2-xii .... Revision 1
2-xiii .... Revision 1
2-xiv .... Revision 1
2-xv .... Revision 1
2-xvi .... Revision 1
2-xvii .... Revision 1
2-xviii .... Revision 1
2-xix .... Revision 1
2-xx .... Revision 1
2-xxi .... Revision 1
2-xxii .... Revision 1
2-xxiii .... Revision 1
2-xxiv .... Revision 1
2-xxv .... Revision 1
2-xxvi .... Revision 1
2-xxvii .... Revision 1
2-xxviii .... Revision 1
2-xxix .... Revision 1
2-xxx .... Revision 1
2-xxxi .... Revision 1
2-xxxii .... Revision 1
2-xxxiii .... Revision 1
2-xxxiv .... Revision UMST-04A
2-1 .... Revision 1
2.1-1 .... Revision 1
2.1-2 .... Revision 1
2.1-3 .... Revision 1
2.1-4 .... Revision 1
2.1-5 .... Revision UMST-04A
2.1-6 .... Revision UMST-04A
2.1-7 .... Revision 1
2.1-8 .... Revision 1
2.1-9 .... Revision 1

List of Effective Pages (Continued)

2.1-10 .... Revision 1
2.1-11 .... Revision 1
2.1-12 .... Revision UMST-04A
2.1-13 .... Revision 1
2.1-14 .... Revision 1
2.1-15 .... Revision 1
2.1-16 .... Revision 1
2.1-17 .... Revision 1
2.1-18 .... Revision 1
2.1-19 .... Revision 1
2.1-20 .... Revision 1
2.1-21 .... Revision 1
2.1-22 .... Revision 1
2.1-23 .... Revision 1
2.1-24 .... Revision 1
2.2-1 .... Revision 1
2.2-2 .... Revision 1
2.2-3 .... Revision 1
2.2-4 .... Revision 1
2.3-1 .... Revision 1
2.3-2 .... Revision UMST-04A
2.3-3 .... Revision 1
2.3-4 .... Revision 1
2.3-5 .... Revision 1
2.3-6 .... Revision 1
2.3-7 .... Revision 1
2.3-8 .... Revision 1
2.3-9 .... Revision 1
2.3-10 .... Revision 1
2.3-11 .... Revision 1
2.3-12 .... Revision 1
2.3-13 .... Revision 1
2.3-14 .... Revision 1
2.3-15 .... Revision 1
2.3-16 .... Revision 1
2.3-17 .... Revision 1
2.3-18 .... Revision 1
2.3-19 .... Revision 1
2.3-20 .... Revision 1
2.3-21 .... Revision 1
2.3-22 .... Revision 1
2.4-1 .... Revision 1
2.4-2 .... Revision 1
2.4-3 .... Revision 1
2.4-4 .... Revision 1
2.4-5 .... Revision 1
2.4-6 .... Revision 1
2.4-7 .... Revision 1
2.4-8 .... Revision 1
2.4-9 .... Revision 1
2.5-1 .... Revision 1
2.5-2 .... Revision 1
2.5-3 .... Revision 1
2.5-4 .... Revision 1
2.5-5 .... Revision 1
2.5-6 .... Revision 1
2.5-7 .... Revision 1
2.5-8 .... Revision 1
2.5-9 .... Revision 1
2.5-10 .... Revision 1
2.5-11 .... Revision 1
2.5-12 .... Revision 1
2.5-13 .... Revision 1
2.5-14 .... Revision 1
2.5-15 .... Revision 1
2.5-16 .... Revision 1

List of Effective Pages (Continued)

2.5-17 .... Revision 1
2.5-18 .... Revision 1
2.5-19 .... Revision 1
2.5-20 .... Revision 1
2.5-21 .... Revision 1
2.5-22 .... Revision 1
2.5-23 .... Revision 1
2.5-24 .... Revision 1
2.5-25 .... Revision 1
2.5-26 .... Revision 1
2.5-27 .... Revision 1
2.5-28 .... Revision 1
2.5-29 .... Revision 1
2.5-30 .... Revision 1
2.5-31 .... Revision 1
2.5-32 .... Revision 1
2.5-33 .... Revision 1
2.5-34 .... Revision 1
2.5-35 .... Revision 1
2.5-36 .... Revision 1
2.5-37 .... Revision 1
2.5-38 .... Revision 1
2.5-39 .... Revision 1
2.5-40 .... Revision 1
2.5-41 .... Revision 1
2.5-42 .... Revision 1
2.5-43 .... Revision 1
2.5-44 .... Revision 1
2.5-45 .... Revision 1
2.5-46 .... Revision 1
2.5-47 .... Revision 1
2.5-48 .... Revision 1
2.5-49 .... Revision 1
2.5-50 .... Revision 1
2.5-51 .... Revision 1
2.5-52 .... Revision 1
2.6-1 .... Revision 1
2.6-2 .... Revision 1
2.6-3 .... Revision 1
2.6-4 .... Revision 1
2.6-5 .... Revision 1
2.6-6 .... Revision 1
2.6-7 .... Revision 1
2.6-8 .... Revision 1
2.6-9 .... Revision 1
2.6-10 .... Revision 1
2.6-11 .... Revision 1
2.6-12 .... Revision 1
2.6-13 .... Revision 1
2.6-14 .... Revision 1
2.6-15 .... Revision 1
2.6-16 .... Revision 1
2.6-17 .... Revision 1
2.6-18 .... Revision 1
2.6-19 .... Revision 1
2.6-20 .... Revision 1
2.6-21 .... Revision 1
2.6-22 .... Revision 1
2.6-23 .... Revision 1
2.6-24 .... Revision 1
2.6-25 .... Revision 1
2.6-26 .... Revision 1
2.6-27 .... Revision 1
2.6-28 .... Revision 1
2.6-29 .... Revision 1
2.6-30 .... Revision 1

List of Effective Pages (Continued)

2.6-31 .... Revision 1
2.6-32 .... Revision 1
2.6-33 .... Revision 1
2.6-34 .... Revision 1
2.6-35 .... Revision 1
2.6-36 .... Revision 1
2.6-37 .... Revision 1
2.6-38 .... Revision 1
2.6-39 .... Revision 1
2.6-40 .... Revision 1
2.6-41 .... Revision 1
2.6-42 .... Revision 1
2.6-43 .... Revision 1
2.6-44 .... Revision 1
2.6-45 .... Revision 1
2.6-46 .... Revision 1
2.6-47 .... Revision 1
2.6-48 .... Revision 1
2.6-49 .... Revision 1
2.6-50 .... Revision 1
2.6-51 .... Revision 1
2.6-52 .... Revision 1
2.6-53 .... Revision 1
2.6-54 .... Revision 1
2.6-55 .... Revision 1
2.6-56 .... Revision 1
2.6-57 .... Revision 1
2.6-58 .... Revision 1
2.6-59 .... Revision 1
2.6-60 .... Revision 1
2.6-61 .... Revision 1
2.6-62 .... Revision 1
2.6-63 .... Revision 1
2.6-64 .... Revision 1
2.6-65 .... Revision 1
2.6-66 .... Revision 1
2.6-67 .... Revision 1
2.6-68 .... Revision 1
2.6-69 .... Revision 1
2.6-70 .... Revision 1
2.6-71 .... Revision 1
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List of Effective Pages (Continued)

2.6-97 .... Revision 1
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2.6-160 .... Revision UMST-04A
2.6-161 .... Revision UMST-04A
2.6-162 .... Revision 1

List of Effective Pages (Continued)

2.6-163 .... Revision UMST-04A
2.6-164 .... Revision UMST-04A
2.6-165 .... Revision UMST-04A
2.6-166 .... Revision UMST-04A
2.6-167 .... Revision UMST-04A
2.6-168 .... Revision UMST-04A
2.6-169 .... Revision UMST-04B
2.6-170 .... Revision UMST-04A
2.6-171 .... Revision 1
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List of Effective Pages (Continued)

2.6-229 .... Revision 1
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List of Effective Pages (Continued)

2.6-295 .... Revision 1
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List of Effective Pages (Continued)

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List of Effective Pages (Continued)

2.7-45 .... Revision 1
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List of Effective Pages (Continued)

2.7-111 .... Revision 1
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List of Effective Pages (Continued)

2.7-177 .... Revision 1
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List of Effective Pages (Continued)

2.10.2-10 .... Revision 1
2.10.2-11 .... Revision 1
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2.10.3-46 .... Revision 1
2.10.3-47 .... Revision 1
2.10.3-48 .... Revision 1

7 Model Drawings
Shown at the end of Section 2.10.3.7

2.10.4-1 .... Revision 1
2.10.4-2 .... Revision 1
2.10.4-3 .... Revision 1
2.10.4-4 .... Revision 1
2.10.4-5 .... Revision 1
2.10.4-6 .... Revision 1

List of Effective Pages (Continued)

2.10.4-7 .... Revision 1
2.10.4-8 .... Revision 1
2.10.4-9 .... Revision 1
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2.11.1-4 .... Revision UMST-04A
2.11.1-5 .... Revision UMST-04A
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2.11.1-8 .... Revision 1
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2.11.1-22 .... Revision UMST-04A
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2.12-1 .... Revision 1
2.12-2 .... Revision 1
2.12-3 .... Revision 1
2.12-4 .... Revision 1
2.12-5 .... Revision 1

List of Effective Pages (Continued)

Chapter 3

3-i .... Revision 1
3-ii .... Revision 1
3-iii ....Revision UMST-04A
3-iv .... Revision 1
3-v .... Revision 1
3-vi .... Revision UMST-04B
3-vii .... Revision 1
3-viii .... Revision 1
3.1-1 .... Revision 1
3.1-2 .... Revision 1
3.1-3 .... Revision 1
3.1-4 .... Revision 1
3.1-5 .... Revision 1
3.1-6 .... Revision 1
3.2-1 .... Revision 1
3.2-2 .... Revision 1
3.2-3 .... Revision 1
3.2-4 .... Revision 1
3.2-5 .... Revision 1
3.2-6 .... Revision 1
3.2-7 .... Revision 1
3.2-8 .... Revision 1
3.2-9 .... Revision 1
3.2-10 .... Revision 1
3.2-11 .... Revision 1
3.2-12 .... Revision 1
3.2-13 .... Revision 1
3.2-14 .... Revision 1
3.2-15 .... Revision 1
3.2-16 .... Revision 1
3.2-17 .... Revision 1
3.2-18 .... Revision 1
3.2-19 .... Revision UMST-04B
3.3-1 .... Revision 1
3.3-2 .... Revision 1
3.3-3 .... Revision 1
3.4-1 .... Revision 1
3.4-2 .... Revision 1
3.4-3 .... Revision UMST-04A
3.4-4 .... Revision UMST-04A
3.4-5 .... Revision UMST-04A
3.4-6 .... Revision 1
3.4-7 .... Revision UMST-04A
3.4-8 .... Revision 1
3.4-9 .... Revision 1
3.4-10 .... Revision 1
3.4-11 .... Revision 1
3.4-12 .... Revision 1
3.4-13 .... Revision UMST-04A
3.4-14 .... Revision UMST-04A
3.4-15 .... Revision 1
3.4-16 .... Revision UMST-04A
3.4-17 .... Revision UMST-04A
3.4-18 .... Revision UMST-04A
3.4-19 .... Revision UMST-04A
3.4-20 .... Revision UMST-04A
3.4-21 .... Revision UMST-04A
3.4-22 .... Revision UMST-04A
3.4-23 .... Revision 1
3.4-24 .... Revision 1
3.4-25 .... Revision 1
3.4-26 .... Revision 1
3.4-27 .... Revision 1

List of Effective Pages (Continued)

3.4-28 .... Revision 1
3.4-29 .... Revision 1
3.4-30 .... Revision 1
3.4-31 .... Revision 1
3.4-32 .... Revision 1
3.4-33 .... Revision 1
3.4-34 .... Revision 1
3.4-35 .... Revision 1
3.4-36 .... Revision 1
3.4-37 .... Revision 1
3.4-38 .... Revision 1
3.4-39 .... Revision 1
3.4-40 .... Revision 1
3.4-41 .... Revision 1
3.4-42 .... Revision 1
3.4-43 .... Revision UMST-04A
3.4-44 .... Revision UMST-04A
3.4-45 .... Revision 1
3.4-46 .... Revision 1
3.4-47 .... Revision 1
3.4-48 .... Revision 1
3.4-49 .... Revision 1
3.4-50 .... Revision 1
3.4-51 .... Revision 1
3.4-52 .... Revision 1
3.4-53 .... Revision 1
3.4-54 .... Revision 1
3.4-55 .... Revision 1
3.4-56 .... Revision 1
3.4-57 .... Revision 1
3.4-58 .... Revision 1
3.4-59 .... Revision 1
3.4-60 .... Revision 1
3.5-1 .... Revision 1
3.5-2 .... Revision 1
3.5-3 .... Revision 1
3.5-4 .... Revision 1
3.5-5 .... Revision 1
3.5-6 .... Revision 1
3.5-7 .... Revision 1
3.5-8 .... Revision 1
3.5-9 .... Revision 1
3.5-10 .... Revision 1
3.5-11 .... Revision 1
3.5-12 .... Revision 1
3.5-13 .... Revision 1
3.5-14 .... Revision 1
3.5-15 .... Revision 1
3.5-16 .... Revision 1
3.5-17 .... Revision 1
3.5-18 .... Revision 1
3.5-19 .... Revision 1
3.5-20 .... Revision 1
3.5-21 .... Revision 1
3.5-22 .... Revision 1
3.5-23 .... Revision 1
3.5-24 .... Revision 1
3.5-25 .... Revision 1
3.5-26 .... Revision 1
3.5-27 .... Revision 1
3.5-28 .... Revision 1
3.6-1 .... Revision UMST-04A
3.6-2 .... Revision 1
3.6-3 .... Revision 1
3.6-4 .... Revision 1
3.6-5 .... Revision 1

List of Effective Pages (Continued)

3.6-6 .... Revision 1
3.6-7 .... Revision UMST-04A
3.6-8 .... Revision UMST-04B
3.6-9 .... Revision 1
3.6-10 .... Revision 1
3.6-11 .... Revision 1
3.6-12 .... Revision 1
3.6-13 .... Revision 1
3.6-14 .... Revision 1
3.6-15 .... Revision 1
3.6-16 .... Revision 1
3.7-1 .... Revision 1
3.7-2 .... Revision 1
3.7-3 .... Revision 1

Chapter 4

4-i .... Revision 1
4-ii .... Revision 1
4-iii .... Revision 1
4-iv .... Revision 1
4-1 .... Revision 1
4-2 .... Revision 1
4.1-1 .... Revision 1
4.1-2 .... Revision 1
4.1-3 .... Revision 1
4.1-4 .... Revision 1
4.1-5 .... Revision 1
4.1-6 .... Revision 1
4.1-7 .... Revision 1
4.2-1 .... Revision 1
4.2-2 .... Revision 1
4.2-3 .... Revision 1
4.2-4 .... Revision 1
4.2-5 .... Revision UMST-04B
4.2-6 .... Revision 1
4.2-7 .... Revision 1
4.2-8 .... Revision 1
4.2-9 .... Revision 1
4.2-10 .... Revision 1
4.2-11 .... Revision 1
4.3-1 .... Revision 1
4.3-2 .... Revision 1
4.3-3 .... Revision 1
4.3-4 .... Revision 1
4.3-5 .... Revision 1
4.4-1 .... Revision 1
4.5-1 .... Revision 1
4.5.1-1 .... Revision UMST-04B
4.5.2-1 .... Revision 1
4.5.2-2 .... Revision 1
4.5.2-3 .... Revision 1
4.5.2-4 .... Revision 1
4.5.2-5 .... Revision 1
4.5.2-6 .... Revision 1
4.5.2-7 .... Revision 1
4.5.2-8 .... Revision 1
4.5.2-9 .... Revision 1
4.5.2-10 .... Revision 1
4.5.2-11 .... Revision 1
4.5.3-1 .... Revision 1
4.5.3-2 .... Revision 1
4.5.3-3 .... Revision 1
4.5.3-4 .... Revision 1
4.5.3-5 .... Revision 1
4.5.3-6 .... Revision 1

List of Effective Pages (Continued)

4.5.3-7 .... Revision 1
4.5.3-8 .... Revision 1
4.5.3-9 .... Revision 1
4.5.3-10 .... Revision 1
4.5.3-11 .... Revision 1
4.5.3-12 .... Revision 1
4.5.3-13 .... Revision 1
4.6-1 .... Revision 1

Chapter 5

5-i .... Revision 1
5-ii .... Revision 1
5-iii .... Revision 1
5-iv .... Revision 1
5-v .... Revision 1
5-vi .... Revision 1
5-vii .... Revision 1
5-viii .... Revision 1
5-ix .... Revision 1
5-1 .... Revision 1
5-2 .... Revision 1
5-3 .... Revision 1
5.1-1 .... Revision UMST-04A
5.1-2 .... Revision 1
5.1-3 .... Revision 1
5.1-4 .... Revision 1
5.1-5 .... Revision 1
5.1-6 .... Revision 1
5.1-7 .... Revision 1
5.1-8 .... Revision 1
5.1-9 .... Revision 1
5.1-10 .... Revision 1
5.2-1 .... Revision 1
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5.2-3 .... Revision 1
5.2-4 .... Revision 1
5.2-5 .... Revision 1
5.2-6 .... Revision 1
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5.2-8 .... Revision 1
5.2-9 .... Revision 1
5.2-10 .... Revision 1
5.2-11 .... Revision 1
5.2-12 .... Revision 1
5.2-13 .... Revision 1
5.2-14 .... Revision 1
5.2-15 .... Revision 1
5.2-16 .... Revision 1
5.2-17 .... Revision 1
5.2-18 .... Revision 1
5.2-19 .... Revision 1
5.2-20 .... Revision 1
5.2-21 .... Revision 1
5.2-22 .... Revision 1
5.2-23 .... Revision 1
5.2-24 .... Revision 1
5.2-25 .... Revision 1
5.2-26 .... Revision 1
5.3-1 .... Revision 1
5.3-2 .... Revision 1
5.3-3 .... Revision 1
5.3-4 .... Revision 1
5.3-5 .... Revision 1
5.3-6 .... Revision 1
5.3-7 .... Revision 1

List of Effective Pages (Continued)

5.3-8 .... Revision 1
5.3-9 .... Revision 1
5.3-10 .... Revision 1
5.3-11 .... Revision 1
5.3-12 .... Revision 1
5.3-13 .... Revision 1
5.3-14 .... Revision 1
5.3-15 .... Revision 1
5.3-16 .... Revision 1
5.3-17 .... Revision 1
5.3-18 .... Revision 1
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5.3-22 .... Revision 1
5.3-23 .... Revision 1
5.3-24 .... Revision 1
5.3-25 .... Revision 1
5.3-26 .... Revision 1
5.3-27 .... Revision 1
5.3-28 .... Revision 1
5.3-29 .... Revision 1
5.3-30 .... Revision 1
5.3-31 .... Revision 1
5.3-32 .... Revision 1
5.3-33 .... Revision 1
5.3-34 .... Revision 1
5.4-1 .... Revision 1
5.4-2 .... Revision 1
5.4-3 .... Revision 1
5.4-4 .... Revision 1
5.4-5 .... Revision 1
5.4-6 .... Revision 1
5.4-7 .... Revision 1
5.4-8 .... Revision 1
5.4-9 .... Revision 1
5.4-10 .... Revision 1
5.4-11 .... Revision 1
5.4-12 .... Revision 1
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5.4-14 .... Revision 1
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5.4-16 .... Revision 1
5.4-17 .... Revision 1
5.4-18 .... Revision 1
5.4-19 .... Revision 1
5.4-20 .... Revision 1
5.4-21 .... Revision 1
5.4-22 .... Revision 1
5.4-23 .... Revision 1
5.4-24 .... Revision 1
5.4-25 .... Revision 1
5.4-26 .... Revision 1
5.5-1 .... Revision 1
5.5.1-1 .... Revision 1
5.5.1-2 .... Revision 1
5.5.1-3 .... Revision 1
5.5.1-4 .... Revision 1
5.5.1-5 .... Revision 1
5.5.1-6 .... Revision 1
5.5.1-7 .... Revision 1
5.5.1-8 .... Revision 1
5.5.1-9 .... Revision 1
5.5.1-10 .... Revision 1
5.5.1-11 .... Revision UMST-04A
5.5.1-12 .... Revision 1

List of Effective Pages (Continued)

5.5.1-13 .... Revision 1
5.5.1-14 .... Revision 1
5.5.1-15 .... Revision 1
5.5.1-16 .... Revision 1
5.5.1-17 .... Revision 1
5.5.1-18 .... Revision 1
5.5.1-19 .... Revision 1
5.5.1-20 .... Revision 1
5.5.1-21 .... Revision 1
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5.5.1-27 .... Revision 1
5.5.1-28 .... Revision 1
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5.5.1-32 .... Revision 1
5.5.1-33 .... Revision 1
5.5.1-34 .... Revision 1
5.5.1-35 .... Revision 1
5.5.1-36 .... Revision 1
5.5.1-37 .... Revision 1
5.5.1-38 .... Revision 1
5.5.1-39 .... Revision 1
5.5.1-40 .... Revision 1
5.5.1-41 .... Revision 1
5.5.2-1 .... Revision 1
5.5.2-2 .... Revision 1
5.5.2-3 .... Revision 1
5.5.3-1 .... Revision 1
5.5.3-2 .... Revision 1
5.5.3-3 .... Revision 1
5.5.3-4 .... Revision 1
5.5.3-5 .... Revision 1
5.5.3-6 .... Revision 1
5.5.3-7 .... Revision 1
5.5.3-8 .... Revision 1
5.5.3-9 .... Revision 1
5.5.3-10 .... Revision 1
5.5.3-11 .... Revision 1
5.5.3-12 .... Revision 1
5.5.3-13 .... Revision 1
5.5.3-14 .... Revision 1
5.5.3-15 .... Revision 1
5.5.3-16 .... Revision 1
5.5.3-17 .... Revision 1
5.5.3-18 .... Revision 1
5.5.3-19 .... Revision 1
5.5.3-20 .... Revision 1
5.5.3-21 .... Revision 1
5.5.3-22 .... Revision 1
5.5.3-23 .... Revision 1
5.5.3-24 .... Revision 1
5.5.3-25 .... Revision 1
5.5.3-26 .... Revision 1
5.5.3-27 .... Revision 1
5.5.3-28 .... Revision 1
5.5.3-29 .... Revision 1
5.5.3-30 .... Revision 1
5.5.3-31 .... Revision 1
5.5.3-32 .... Revision 1
5.5.3-33 .... Revision 1
5.5.3-34 .... Revision 1

List of Effective Pages (Continued)

5.5.3-35 .... Revision 1
5.5.3-36 .... Revision 1
5.5.3-37 .... Revision 1
5.5.3-38 .... Revision 1
5.5.3-39 .... Revision 1
5.5.3-40 .... Revision 1
5.5.3-41 .... Revision 1
5.5.3-42 .... Revision 1
5.5.3-43 .... Revision 1
5.5.3-44 .... Revision 1
5.5.3-45 .... Revision 1
5.5.3-46 .... Revision 1
5.5.3-47 .... Revision 1
5.5.3-48 .... Revision 1
5.5.3-49 .... Revision 1
5.5.3-50 .... Revision 1
5.6-1 .... Revision 1
5.6-2 .... Revision 1

Chapter 6

6-i .... Revision 1
6-ii .... Revision 1
6-iii .... Revision 1
6-iv .... Revision 1
6-v .... Revision 1
6-vi .... Revision 1
6-vii .... Revision 1
6-viii .... Revision UMST-04A
6.1-1 .... Revision UMST-04A
6.1-2 .... Revision UMST-04A
6.1-3 .... Revision UMST-04A
6.2-1 .... Revision UMST-04A
6.2-2 .... Revision 1
6.2-3 .... Revision 1
6.2-4 .... Revision 1
6.3-1 .... Revision 1
6.3-2 .... Revision 1
6.3-3 .... Revision 1
6.3-4 .... Revision UMST-04A
6.3-5 .... Revision 1
6.3-6 .... Revision 1
6.3-7 .... Revision 1
6.3-8 .... Revision 1
6.3-9 .... Revision 1
6.3-10 .... Revision 1
6.3-11 .... Revision 1
6.3-12 .... Revision 1
6.3-13 .... Revision 1
6.3-14 .... Revision 1
6.4-1 .... Revision 1
6.4-2 .... Revision UMST-04A
6.4-3 .... Revision UMST-04A
6.4-4 .... Revision UMST-04A
6.4-5 .... Revision UMST-04A
6.4-6 .... Revision 1
6.4-7 .... Revision 1
6.4-8 .... Revision 1
6.4-9 .... Revision 1
6.4-10 .... Revision 1
6.4-11 .... Revision 1
6.4-12 .... Revision 1
6.4-13 .... Revision 1
6.4-14 .... Revision 1
6.4-15 .... Revision 1
6.4-16 .... Revision 1

List of Effective Pages (Continued)

6.4-17 .... Revision 1
6.4-18 .... Revision 1
6.4-19 .... Revision 1
6.4-20 .... Revision UMST-04B
6.4-21 .... Revision UMST-04B
6.4-22 .... Revision UMST-04B
6.4-23 .... Revision UMST-04B
6.4-24 .... Revision UMST-04B
6.4-25 .... Revision 1
6.4-26 .... Revision 1
6.4-27 .... Revision 1
6.4-28 .... Revision 1
6.4-29 .... Revision 1
6.4-30 .... Revision 1
6.4-31 .... Revision 1
6.4-32 .... Revision 1
6.4-33 .... Revision 1
6.4-34 .... Revision 1
6.4-35 .... Revision 1
6.4-36 .... Revision 1
6.4-37 .... Revision 1
6.4-38 .... Revision 1
6.4-39 .... Revision UMST-04A
6.5-1 .... Revision 1
6.5-2 .... Revision 1
6.5-3 .... Revision 1
6.5-4 .... Revision 1
6.5-5 .... Revision 1
6.5-6 .... Revision 1
6.5-7 .... Revision 1
6.5-8 .... Revision 1
6.5-9 .... Revision 1
6.5-10 .... Revision 1
6.5-11 .... Revision 1
6.5-12 .... Revision 1
6.5-13 .... Revision 1
6.5-14 .... Revision 1
6.5-15 .... Revision 1
6.5-16 .... Revision 1
6.5-17 .... Revision 1
6.5-18 .... Revision 1
6.5-19 .... Revision 1
6.5-20 .... Revision 1
6.5-21 .... Revision 1
6.5-22 .... Revision 1
6.5-23 .... Revision 1
6.5-24 .... Revision 1
6.5-25 .... Revision 1
6.5-26 .... Revision 1
6.5-27 .... Revision 1
6.5-28 .... Revision 1
6.5-29 .... Revision 1
6.5-30 .... Revision 1
6.5-31 .... Revision 1
6.5-32 .... Revision 1
6.5-33 .... Revision 1
6.5-34 .... Revision 1
6.5-35 .... Revision 1
6.5-36 .... Revision 1
6.5-37 .... Revision 1
6.5-38 .... Revision 1
6.5-39 .... Revision 1
6.5-40 .... Revision 1
6.5-41 .... Revision 1
6.5-42 .... Revision 1
6.5-43 .... Revision 1

List of Effective Pages (Continued)

6.5-44 .... Revision 1
6.6-1 .... Revision 1
6.6.1-1 .... Revision 1
6.6.1-2 .... Revision 1
6.6.1-3 .... Revision 1
6.6.1-4 .... Revision 1
6.6.1-5 .... Revision 1
6.6.1-6 .... Revision 1
6.6.1-7 .... Revision UMST-04B
6.6.1-8 .... Revision UMST-04B
6.6.1-9 .... Revision UMST-04A
6.6.1-10 .... Revision 1
6.6.1-11 .... Revision 1
6.6.1-12 .... Revision 1
6.6.1-13 .... Revision 1
6.6.1-14 .... Revision 1
6.6.1-15 .... Revision 1
6.6.1-16 .... Revision 1
6.6.1-17 .... Revision 1
6.6.1-18 .... Revision 1
6.6.1-19 .... Revision 1
6.6.1-20 .... Revision 1
6.6.1-21 .... Revision 1
6.6.1-22 .... Revision UMST-04A
6.6.1-23 .... Revision 1
6.6.1-24 .... Revision 1
6.6.2-1 .... Revision 1
6.6.2-2 .... Revision 1
6.6.2-3 .... Revision 1
6.6.2-4 .... Revision 1
6.6.2-5 .... Revision 1
6.6.2-6 .... Revision 1
6.6.2-7 .... Revision 1
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6.6.2-20 .... Revision 1
6.6.2-21 .... Revision 1
6.6.2-22 .... Revision 1
6.6.2-23 .... Revision 1
6.6.2-24 .... Revision 1
6.6.2-25 .... Revision 1
6.6.2-26 .... Revision 1
6.6.2-27 .... Revision 1
6.6.2-28 .... Revision 1
6.6.2-29 .... Revision 1
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6.6.2-31 .... Revision 1
6.6.2-32 .... Revision 1
6.6.2-33 .... Revision 1
6.6.2-34 .... Revision 1
6.6.2-35 .... Revision 1
6.6.2-36 .... Revision 1
6.6.2-37 .... Revision 1
6.6.2-38 .... Revision 1
6.6.2-39 .... Revision 1
6.6.2-40 .... Revision 1

List of Effective Pages (Continued)

6.6.2-41 .... Revision 1
6.6.2-42 .... Revision 1
6.6.2-43 .... Revision 1
6.6.2-44 .... Revision 1
6.6.2-45 .... Revision 1
6.6.2-46 .... Revision 1
6.6.2-47 .... Revision 1
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6.6.2-53 .... Revision 1
6.6.2-54 .... Revision 1
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6.6.2-56 .... Revision 1
6.6.2-57 .... Revision 1
6.6.2-58 .... Revision 1
6.6.2-59 .... Revision 1
6.6.2-60 .... Revision 1
6.6.2-61 .... Revision 1
6.6.2-62 .... Revision 1
6.6.2-63 .... Revision 1
6.6.2-64 .... Revision 1
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6.6.2-66 .... Revision 1
6.6.2-67 .... Revision 1
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6.6.2-72 .... Revision 1
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6.6.2-92 .... Revision 1
6.6.2-93 .... Revision 1
6.6.2-94 .... Revision 1
6.6.3-1 .... Revision 1
6.6.3-2 .... Revision 1
6.6.3-3 .... Revision 1
6.6.3-4 .... Revision 1
6.6.3-5 .... Revision 1
6.6.3-6 .... Revision 1
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6.6.3-8 .... Revision 1
6.6.3-9 .... Revision 1
6.6.3-10 .... Revision 1
6.6.3-11 .... Revision 1
6.6.3-12 .... Revision 1

List of Effective Pages (Continued)

6.6.3-13 .... Revision 1
6.6.3-14 .... Revision 1
6.6.3-15 .... Revision 1
6.6.3-16 .... Revision 1
6.6.3-17 .... Revision 1
6.6.3-18 .... Revision 1
6.6.3-19 .... Revision 1
6.6.3-20 .... Revision 1
6.6.3-21 .... Revision 1
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6.6.3-26 .... Revision 1
6.6.3-27 .... Revision 1
6.6.3-28 .... Revision 1
6.6.3-29 .... Revision 1
6.6.3-30 .... Revision 1
6.6.3-31 .... Revision 1
6.6.3-32 .... Revision 1
6.6.3-33 .... Revision 1
6.6.3-34 .... Revision 1
6.7-1 .... Revision 1
6.7-2 .... Revision 1

Chapter 7

7-i .... Revision 1
7-ii .... Revision 1
7-1 .... Revision 1
7-2 .... Revision 1
7-3 .... Revision 1
7.1-1 .... Revision 1
7.1-2 .... Revision 1
7.1-3 .... Revision 1
7.1-4 .... Revision UMST-04A
7.1-5 .... Revision UMST-04A
7.1-6 .... Revision 1
7.1-7 .... Revision 1
7.2-1 .... Revision 1
7.2-2 .... Revision 1
7.3-1 .... Revision 1
7.3-2 .... Revision 1
7.3-3 .... Revision UMST-04A
7.3-4 .... Revision UMST-04A
7.4-1 .... Revision 1
7.5-1 .... Revision 1

Chapter 8

8-i .... Revision 1
8-ii .... Revision 1
8-iii .... Revision 1
8-1 .... Revision 1
8.1-1 .... Revision 1
8.1-2 .... Revision 1
8.1-3 .... Revision 1
8.1-4 .... Revision 1
8.1-5 .... Revision 1
8.1-6 .... Revision 1
8.1-7 .... Revision 1
8.1-8 .... Revision 1
8.1-9 .... Revision 1
8.1-10 .... Revision 1
8.1-11 .... Revision 1
8.1-12 .... Revision 1

List of Effective Pages (Continued)

8.1-13 .... Revision 1
8.1-14 .... Revision 1
8.1-15 .... Revision 1
8.1-16 .... Revision 1
8.2-1 .... Revision 1
8.2-2 .... Revision 1
8.2-3 .... Revision 1
8.2-4 .... Revision 1
8.2-5 .... Revision 1
8.3-1 .... Revision 1
8.3-2 .... Revision 1
8.3-3 .... Revision 1
8.3-4 .... Revision 1
8.3-5 .... Revision 1
8.3-6 .... Revision 1
8.3-7 .... Revision 1
8.3-8 .... Revision 1

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Chapter 3

List of Figures (Continued)

Figure 3.5-12 Hypothetical Accident Conditions Maximum Lead Temperature History (BWR) 3.5-18

Figure 3.5-13 Hypothetical Accident Conditions Maximum Neutron Shield Exterior Temperature History (BWR) 3.5-19

Figure 3.5-14 Hypothetical Accident Conditions Maximum Cask Inner Shell Temperature History (BWR) 3.5-20

Figure 3.5-15 Hypothetical Accident Conditions Maximum Cask Outer Shell Temperature History (BWR) 3.5-21

Figure 3.5-16 Hypothetical Accident Conditions Maximum Lower Neutron Shield Temperature History (BWR) 3.5-22

Figure 3.5-17 Hypothetical Accident Conditions Maximum Lower Drain Port O-Ring Temperature History (BWR) 3.5-23

Figure 3.5-18 Hypothetical Accident Conditions Maximum Cask Lid Vent Port O-Ring Temperature History (BWR) 3.5-24

Figure 3.5-19 Hypothetical Accident Conditions Maximum Cask Lid O-Rings Temperature History (BWR) 3.5-25

Figure 3.6.1.1-1 Fuel Assembly Quarter Symmetry Model for Maine Yankee Consolidated Fuel 3.6-9

Figure 3.6.1.1-2 Maine Yankee Three-Dimensional Periodic Canister Internal Model 3.6-10

Figure 3.6.1.1-3 Evaluated Locations for a Maine Yankee Consolidated Fuel Lattice in the PWR Fuel Basket 3.6-11

Figure 3.6.1.1-4 Damaged Fuel Region in the Three-Dimensional Cask Model (PWR) 3.6-12

Figure 3.6.1.2-1 Two-Dimensional Axisymmetric Thermal Model of the Maine Yankee GTCC Waste Basket/Canister/Transport Cask 3.6-15

List of Tables

Table 3.1-1 Thermal Analysis Bounding Conditions - Normal Conditions of Transport.... 3.1-6

Table 3.2-1 Thermal Properties of Solid Neutron Shield (NS-4-FR) 3.2-6

Table 3.2-2 Thermal Properties of Stainless Steel 3.2-7

Table 3.2-3 Thermal Properties of Carbon Steel 3.2-8

Table 3.2-4 Thermal Properties of Chemical Copper Lead 3.2-9

Table 3.2-5 Thermal Properties of Type 6061-T651 Aluminum Alloy 3.2-10

Table 3.2-6 Thermal Properties of Helium 3.2-11

Table 3.2-7 Thermal Properties of Dry Air 3.2-12

Table 3.2-8 Thermal Properties of Copper 3.2-13

Table 3.2-9 Thermal Properties of Zircaloy and Zircaloy-4 Cladding 3.2-14

Table 3.2-10 Thermal Properties of Fuel (UO_2) 3.2-15

Table 3.2-11 Thermal Properties of BORAL Composite Sheet 3.2-16

Table 3.2-12 Thermal Properties of Redwood (Air Dry) 3.2-17

Table 3.2-13 Thermal Properties of Fiberfrax Ceramic Fiber Paper 3.2-18

Table 3.2-14 Gaps Within the Universal Transport Cask 3.2-19

Table 3.4-1 Maximum Component Temperatures - Normal Conditions of Transport, Maximum Decay Heat, Maximum Ambient Temperature 3.4-50

Table 3.4-2 Maximum Component Temperatures - Normal Conditions of Transport, Maximum Decay Heat, Minimum Ambient Temperature 3.4-51

Table 3.4-3 Universal Transport Cask Thermal Performance Summary For Component Operating Temperature 3.4-52

Table 3.2-14 Gaps Within the Universal Transport Cask

Gap Location	Gap (in)	
	Cask with PWR Fuel Canister	**Cask with BWR Fuel Canister**
Gap between support disk and canister shell	0.155	0.155
Gap between canister and inner shell	0.275	0.275
Gap between lead gamma shield and inner shell	0.015	0.015
Gap between heat transfer disk and canister shell	0.280	0.315
Gap between canister bottom plate and top of spacer	0.125	0.25
Gap between bottom forging and spacer	*	0.25
Gap between spacer and canister bottom plate	0.125	0.25

* Only the base disk of the spacer is modeled in the PWR cask analysis. The cylindrical shells attached to the base disk are neglected.

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7. Standard fuel assemblies that have fuel rods removed from the lattice.

There is one fuel assembly that has 107 rods removed. This fuel assembly has a heat load of 70 watts (only 8% of the design basis heat load of 0.83 kW). For the rest of fuel assemblies that have fuel rods removed from the lattice, the maximum number of removed fuel rods is 14, which is 8% (14/176) of the total number of rods in one fuel assembly. The maximum heat load for any one of these fuel assemblies is conservatively determined to be 0.63 kW. This heat load is 24% less than the design basis heat load of 0.83 kW. Therefore, the thermal performance for the configuration that contains standard fuel assemblies bounds that of the fuel assemblies with removed rods.

8. Damaged Fuel Assemblies

Damaged fuel assemblies are standard fuel assemblies with fuel rods that have known or suspected cladding defects greater than hairline cracks or pinhole leaks. Each damaged fuel assembly will be placed in one of the two configurations of the Maine Yankee fuel can. The primary function of the fuel can is to confine fuel material within the can and to facilitate handling and retrievability. The Maine Yankee fuel can is shown in Drawings 412-501 and 412-502. The placement of the loaded fuel cans is restricted by operating procedures and/or Technical Specifications to loading into the four fuel tube positions at the periphery of the fuel basket as shown in Figure 3.6.1.1-4. The heat load for each damaged fuel assembly is limited to the design basis heat load of 0.833 kW (20 kW/24).

A steady-state thermal analysis is performed using the three-dimensional cask model described in Section 3.4.1.1.1 simulating 100% failure of the damaged fuel rods held in the Maine Yankee fuel can. The canister is assumed to contain twenty (20) design basis PWR fuel assemblies and damaged fuel assemblies in fuel cans in each of the four corner positions.

A debris compaction length of 104 inches is considered in the analysis based on the volume of fuel rods and a 50% compaction of the debris. Additionally, this 104-inch debris region is assumed to be located at the center of the active fuel region of the design basis PWR fuel assemblies, as shown in Figure 3.6.1.1-4. The entire heat load for a single fuel assembly (i.e., 0.833 kW) is considered to be concentrated in the debris region. The effective thermal conductivities for the design basis PWR fuel assembly (Section 3.4.1.1.2) are used for the debris

region. This is conservative, since the debris (100% failed rods) is expected to have a higher density (better conduction) and more surface area (better radiation) than an intact fuel assembly. In addition, the thermal conductivity of helium is used for the remainder of the active fuel length. Boundary conditions corresponding to normal transport are used at the outer surface of the cask (see Section 3.4.1.1.1). The results of the steady-state thermal analysis for 100% fuel rod, fuel cladding and guide tube failure are:

Description	Maximum Temperature (°F)			
	Fuel Cladding	Damaged Fuel	Support Disk	Heat Transfer Disk
Configuration with damaged fuel loaded in four basket corner locations	682	633	618	614
Design basis PWR fuel	673	N/A	608	605
Allowable	750	N/A	650	700

As shown by the previous data, the maximum temperatures for the fuel cladding, damaged fuel assembly, support disks, and heat transfer disks for the configuration with damaged fuel loaded in four (4) basket corner locations are within the allowable temperature range. Additionally, the maximum temperature of the support disk remains bounded by that used in the structural analyses of the fuel basket.

The effect of the compaction of the damaged fuel is most significant for the interior of the basket, and this effect is determined to be 10 °F, as shown in the table above. For the cask body closure lid seal, the effect of the damaged fuel is expected to be insignificant, since the transportable storage canister shield and structural lids, representing a thickness of 10 inches of steel, separate the fuel from the cask body closure lid seals. The canister lids act to spread any concentration of heat from the damaged fuel. The port cover seals are even more remote from the damaged fuel than the cask body lid seals and, therefore, are not considered to be affected by the damaged fuel.

Damaged high burnup fuel must be loaded into one of two configurations of Maine Yankee damaged fuel cans. Based on guidance provided in ISG-15, high burnup fuel assemblies having more than 1% of rods with oxide layers greater than 80 microns, or more than 3% of rods with oxide layers greater than 70 microns, are classified as damaged. Consistent with the containment analysis, a basket release fraction of 20% is applied. This release fraction accounts for up to 12 high burnup assemblies, including up to four classified as damaged. Applying this release fraction to the pressure evaluation in Section 3.4.4.1 yields a normal conditions cask pressure of 15.61 psig, calculated using B&W 17×17 Mark C fuel assembly parameters.

Chapter 4

Mixture A_2 values are determined for gas, volatile, fine, and crud mixtures and are then combined for a total cask mixture A_2 value. Tables 4.2-2 and 4.2-3 provide the source term and A_2 values per group for PWR and BWR cask systems release rate calculations.

Maximum Allowable Leak Rates

On the basis of the methodology discussed above, the maximum allowable leak rates for the casks containing standard or high burnup PWR and BWR standard fuel under normal conditions of transport are calculated to be 5.5×10^{-6} and 1.3×10^{-5} ref cm^3/sec, respectively (Table 4.2-4).

The maximum allowable release rates are more restrictive for the cask containing high burnup damaged PWR assemblies because of the higher failure rate associated with ISG-15. This ISG specifies a high burnup fuel rod failure rate of 50% in normal conditions for fuel rods with an oxide layer thickness greater than 70 microns. In accordance with ISG-15, no more than 3% of the rods in a high burnup assembly may contain oxide layers over 70 microns without invoking the requirement for canning of the assembly. The worst-case containment analysis loading is, therefore, 12 intact standard assemblies failing at 3%, 8 high burnup assemblies classified as intact (at a nominal 3% failure rate) with an additional 50% failure of the 3% high burnup rods exceeding the 70 micron oxide layer, and 50% failure of rods inside the damaged fuel cans. As significantly damaged standard or high burnup fuel may be loaded into the damaged fuel cans, a conservative 100% normal condition failure rate is applied to four assemblies. The effective failure rate for normal conditions of transport is, therefore, 19.7% ($12 \times 3\% + 8 \times 4.5\% + 4 \times 100\%$). This configuration is bounded by the 20% average release fraction applied to a full canister load of high burnup assemblies.

4.2.1.2 Correlation of Allowable Leak Rates to Air Standard

The volumetric gas leak rate is independent of transport cask pressure and temperature. The maximum allowable release must be correlated with air standard leak rates, which depend on gas temperatures, pressures, and leakage path length and diameter. This correlation requires calculation of the capillary opening diameter through which the flow occurs. Depending on pressure and condition of the flow, a combination of continuum and molecular flow occurs.

Continuum flow and molecular flow equations are obtained from NUREG/CR-6487, Section 2. Both continuum and molecular flow rate equations presented below are adjusted to upstream flow rate in accordance with NUREG/CR-6487 and ANSI N14.5-1997.

The continuum volumetric flow rate of the gas (cm^3/sec), L_c, is given by:

$$L_c = \frac{2.48x10^6 D^4}{a\mu}(P_u - P_d) * \frac{P_a}{P_u} = F_c * (P_u - P_d) * \frac{P_a}{P_u}$$

where:

F_c	=	coefficient for continuum flow [cm^3/atm-s]
D	=	capillary diameter [cm]
a	=	capillary length [cm]
μ	=	fluid viscosity [cP]
P_u	=	upstream pressure [atm] - pressure inside containment
P_d	=	downstream pressure [atm] - pressure outside containment

and, the molecular volumetric flow rate of the gas (cm^3/sec), L_m, is given by:

$$L_m = \frac{3.81x10^3 D^3 \sqrt{\frac{T}{M}}}{aP_a}(P_u - P_d) * \frac{P_a}{P_u} = F_m * (P_u - P_d) * \frac{P_a}{P_u}$$

where:

L_m	=	is the volumetric flow rate of gas at P_a [cm^3/sec]
F_m	=	is the coefficient for molecular flow [cm^3/atm-s]
D	=	is the capillary diameter [cm]
T	=	is the gas temperature [K]
M	=	is the gas molecular weight [g/mole]
P_a	=	is the average pressure $(P_u+P_d)/2$ [atm]
P_u	=	is the upstream pressure [atm]
P_d	=	is the downstream pressure [atm].
a	=	capillary diameter [cm]

For this analysis, the gas temperature used for molecular flow analysis is identical to the upstream temperature. Pressures and temperatures for PWR and BWR system normal operating conditions are summarized in Table 4.2-5. Based on the pressure, temperature and allowable leakage rate (L_N) the capillary diameter of the leak is determined. The calculated capillary diameter is then used to determine the air standard leak rate and helium test leak rate. Air standard condition leak rates are determined for air leaking from 1 atmosphere to 0.01

4.5.1 Containment Evaluation for Site Specific Contents

4.5.1.1 Containment Evaluation for Maine Yankee Contents

Pressure and radionuclide content of the Maine Yankee 14×14 fuel assemblies are bounded by the larger B&W 15×15 assembly employed in the containment evaluations presented in Sections 4.2 and 4.3. The larger fission mass of the B&W assembly produces higher fission gas inventories for a fixed burnup. The Maine Yankee fuel assemblies, with non-fuel components or in consolidated or damaged form (up to four consolidated or damaged assemblies per canister) displaces less free volume than the B&W fuel assembly forming the design basis for the containment analysis, and, therefore, results in lower pressures at a fixed decay heat.

Maine Yankee fuel with up to 50,000 MWD/MTU may be loaded in the transportable storage canister. ISG-15 requires a containment evaluation assuming 50% failure of high burnup fuel with an oxide layer thickness above 70 microns. ISG-15 also requires that fuel assemblies with more than 3% of rods with an oxide layer over 70 microns are considered as damaged and will, therefore, be placed in a damaged fuel can. High burnup fuel must be loaded in the outer fuel loading positions of the basket. The PWR basket has 24 fuel loading positions, including 12 outer positions.

The PWR containment analysis in Section 4.2.1 assumes 4 PWR (high burnup) fuel assemblies failing at 100%, 8 high burnup fuel assemblies failing at 4.5% (3% standard failure fraction plus 50% of the 3% high burnup rods), and the remaining 12 assemblies failing at 3%. This results in 20% of the fuel rods failing in normal conditions and bounds the presence of four Maine Yankee damaged fuel cans of either configuration (see Drawings 412-501 and 412-502) in the basket. The PWR leak rate calculation is based on the higher failure fraction.

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Chapter 6

- BWR fuel rods are either tie rods connecting the top and bottom nozzles, or are rods manufactured with an external spring between the top of the fuel rod and the top nozzle tie plate. For this evaluation, all external springs are ignored. Therefore, all BWR fuel rods are allowed to shift axially into contact with the top tie plate.
- Within BWR fuel rods, the fuel is assumed to shift upward into the plenum region. Each plenum region contains a plenum spring. Detailed structural analyses have shown that during a 60g top end impact, the BWR plenum spring will compress and rebound 1.729 inches. The fuel material in the rods is assumed to shift and remain in contact with the compressed plenum spring. A review of plenum length and spring data for various rod designs indicates a minimum of 13% of the BWR plenum space is occupied by a solid height plenum spring. The final height of the plenum spring is calculated from the sum of the solid height of the spring and a spring rebound height of 1.729 inches.
- Detailed structural analyses of the BWR assembly have also shown that during a 60g top end impact, the lifting bail will deform. The maximum BWR bail deformation was calculated to be 2.371 inches.

Therefore, spacing from the assembly top to the active fuel region is controlled by the top end-fitting height, the fuel rod end-plug height and the distance the active fuel moves into the top plenum. In the case of BWR rods, the end-plug height includes only the portion of the plug below the tie plate when the fuel rod is shifted up. The distance between the top of the fuel assembly and the active fuel region for the bounding (minimum offset) fuel types in each canister class is presented in Table 6.4-19. Also included in Table 6.4-19 is the Class 5 Exxon/ANF 9x9 assembly, since this assembly represents the maximum reactivity radial lattice geometry.

In the third stage of the evaluation, the maximum active fuel exposure (or minimum coverage) for each fuel type is determined by simply subtracting the active fuel offset from the neutron absorber offset. A positive value indicates active fuel is exposed (i.e., neutron absorber does not cover the entire active fuel region).

As shown in in Table 6.4-19, the maximum lengths of exposed BWR fuel result for the Exxon/ANF 9x9 and GE 8x8 BWR fuel assemblies at 4.312 inches. For further conservatism in the criticality analysis, the active fuel exposure length is increased to 7.625 inches for BWR fuel assemblies.

As previously mentioned, the maximum length of exposed fuel is not obtained from the BWR fuel assembly defined as having the maximum reactivity in Section 6.4. The maximum reactivity

BWR assembly documented in the SAR is the UMS® Class 5 Exxon/ANF 9x9 (79 fuel rod) assembly for BWR canisters. Therefore, rather than analyzing each fuel type with its specific exposed fuel height, the evaluated exposed fuel height of 7.625 inches (BWR) is applied to the Class 5 Exxon/ANF 9x9 BWR assembly.

To model the 7.625-inch exposed fuel length for the Class 5 Exxon/ANF 9 × 9 BWR fuel assembly criticality evaluation, the following model parameters are specified:

- Fuel assemblies are shifted to the shield lid.
- Fuel rod and fuel assembly are reconfigured for a total distance between the top of the fuel assembly and the active fuel region of 9.265 inches. This is accomplished by:
  - Spacing the fuel rods from the top tie plate by the external spring.
  - Moving the active fuel region to the midpoint of the plenum.
  - Reducing the top nozzle height by 4.7 inches.
- Basket geometry is modified to a 16.89-inch distance between the shield lid and the neutron absorber top. This is accomplished by:
  - Locating the basket and tubes at the bottom of the canister cavity.
  - Reducing the BWR neutron absorber sheet length by 3.009 inches (based on 3.0 inches of absorber added during final design iterations and 0.009 inch that accounts for differences in exposed fuel between the Exxon 9×9 BWR/3-4 and BWR/4-6 assembly types).

A distance between the top of the assembly and the active fuel region of 9.265 inches and a neutron absorber offset of 16.89 inches result in a 7.625-inch fuel exposure when the assembly is shifted to the shield lid.

<u>Evaluation of PWR System Top End Impact</u>

Axial shifting of the contents of the Transportable Storage Canister (TSC) occurs as a result of a top end impact load condition for the transport cask containing a loaded TSC. In this scenario of contents shifting, the fuel assembly and the basket shift upward to contact the canister lid. The distance between the canister lid and the neutron absorber sheets, which are attached to the fuel tubes, and the distance between the top of the fuel assembly and the active fuel region are required to establish the height of active fuel exposed beyond the neutron absorber for any given assembly. Exposure of the active fuel in any specific fuel type occurs if the minimum distance between the top of the assembly and the top of the active fuel region is less than the maximum distance from the canister lid to the top of the neutron absorber sheet. The exposed fuel height evaluation is performed for each PWR fuel assembly type that is proposed to be loaded into the

UMS® canister. The calculation is divided into three stages: calculation of the neutron absorber offset, determination of the active fuel offset, and calculation of the fuel exposure.

In stage one, the maximum distance between the top of the neutron absorber sheet and the canister lid is determined for each UMS® PWR canister class. The maximum distance provides the greatest fuel exposure, when considering a shifted fuel assembly. This distance depends on the canister class specific weldment, basket, tube and neutron absorber lengths; the relative location of the neutron absorber on the fuel tube; and tolerances associated with the basket components. The maximum distances for a PWR basket shifted to the canister lid appear in Table 6.4-19.

In the second stage of the analysis, the minimum distances between the canister lid and the fuel assembly, and between the top of the fuel assembly and the active fuel region are determined for each PWR fuel type. Since the fuel assembly is shifted to contact the canister lid, the distance between the lid and the fuel assembly is always zero (no credit is taken for any offset produced by the PWR leaf springs). The active fuel shifting condition in the fuel assembly assumes that:

- In PWR fuel assemblies, where a space exists between the fuel rod end-cap and the end-fitting, the fuel rods are shifted within the grid until contact is made with the top end-fitting (zero gap).
- Within the PWR fuel rods, the active fuel is assumed to shift upward into the plenum region. Each plenum region contains a spring. Detailed structural analysis has shown that during a 60g top-end impact, the PWR spring will compress and rebound a minimum of 2.39 inches. The fuel material in the rods is assumed to shift and remain in contact with the fully compressed plenum spring. A review of plenum length and spring data for various rod designs indicates a minimum of 31% of the PWR plenum space is occupied by a solid height plenum spring. The 2.39-inch spring rebound is added to the compressed spring height.
- Detailed structural analyses of the PWR assembly have shown that during a 60g top end impact, no significant damage to the top end-fitting (i.e., no height reduction) occurs.

Therefore, spacing from the assembly top to the active fuel region is controlled by the end-fitting height, the fuel rod end-plug height and the distance the active fuel moves into the top plenum.

The distance between the top of the fuel assembly and the active fuel region for typical fuel types in each canister class is presented in Table 6.4-19. Also included in Table 6.4-19 is the Westinghouse 17×17 OFA assembly, since this assembly represents the maximum reactivity radial lattice geometry.

In the third stage of the evaluation, the maximum active fuel exposure (or minimum coverage) for each fuel type is determined by simply subtracting the active fuel offset from the neutron absorber offset. A positive value indicates active fuel is exposed (i.e., neutron absorber does not cover the entire active fuel region), while a negative value indicates full active fuel coverage and additional coverage to that extent.

As shown in Table 6.4-19, the maximum lengths of exposed PWR fuel result for the Westinghouse 15x15 PWR fuel assembly at 0.782 inch. For further conservatism in the criticality analysis, the active fuel exposure length is increased to 4.52 inches for PWR fuel assemblies.

As previously mentioned, the maximum length of exposed fuel is not obtained from the PWR fuel assembly defined as having the maximum reactivity in Section 6.4. The maximum reactivity PWR assembly documented in the SAR is the Westinghouse 17×17 OFA assembly for PWR canisters. Therefore, rather than analyzing each fuel type with its specific exposed fuel height, the evaluated exposed fuel height of 4.52 inches (PWR) is applied to the Westinghouse 17×17 OFA PWR assembly.

To model the 4.52-inch exposed fuel length for the Westinghouse 17×17 OFA PWR fuel assembly criticality evaluation, the following model parameters are specified:

- Fuel assemblies are shifted to the shield lid.
- Fuel rod and fuel assembly are reconfigured for a total distance between the top of the fuel assembly and the active fuel region of 5.67 inches. This is accomplished by:
  - Shifting the fuel rods to contact the top end-fitting.
  - Moving the active fuel region to the midpoint of the plenum.
  - Reducing the top nozzle height by 4.7 inches.
- Basket geometry is modified to a 10.19-inch distance between the shield lid and the neutron absorber top. This is accomplished by:
  - Locating the basket and tubes at the bottom of the canister cavity.
  - Reducing the PWR neutron absorber sheet length by 0.815 inch.

A distance between the top of the assembly and the active fuel region of 5.67 inches and a neutron absorber offset of 10.19 inches result in a 4.52-inch fuel exposure when the assembly is shifted to the shield lid.

Fuel Assembly Minimum Intact Hardware Dimension Limits

Based on limiting the exposed height of active fuel to 4.52 inches for the PWR fuel assemblies and to 7.625 inches for the BWR fuel assemblies, intact fuel assembly hardware limits are defined to assure compliance with the safety basis of the analysis. These limits consider zero PWR top end-fitting deformation, a PWR plenum spring rebound height of 2.39 in, 2.371 in of BWR top end-fitting (lifting bail) deformation and a BWR plenum spring rebound height of 1.729 inches. The limits for each UMS® canister class containing PWR fuel are calculated by subtracting the sum of the exposed fuel height (4.52 in) and the plenum spring rebound (2.39 in) from the distance between the canister lid and the top of the neutron absorber. The limits for each UMS® canister class containing BWR fuel are calculated by subtracting the sum of the height of exposed fuel, 7.625 inches, and the plenum spring rebound height, 1.729 inches, from the sum of the lifting bail deformation, 2.371 inches, and the distance between the canister lid and the top of the neutron absorber. These resulting limits are provided in Table 6.4-20.

Each minimum axial assembly dimension is a generic limit that all fuel types loaded into a particular can's class shall meet. Compliance with these limits will ensure that the exposed fuel heights evaluated and found to result in a subcritical system will not be exceeded. For PWR fuel, the minimum intact assembly hardware dimension above the active fuel shall be calculated by summing the top end-fitting height, the top end cap height, and the solid height of the plenum spring. For BWR fuel, the minimum axial assembly dimension above the active fuel shall be calculated by summing the intact top end-fitting height, the portion of the top end-cap height below the tie plate when the fuel rod is shifted up, and the solid height of the plenum spring. Tolerances on these components shall be conservatively considered when calculating the subject dimension.

Evaluation of Bottom End Axial Fuel Shifting

Similar to the top end evaluation, a bounding hypothetical axial fuel-shifting condition is considered in which all of the fuel rods are shifted to the bottom of each assembly. For PWR fuel assemblies with a lower plenum, the fuel within every rod is assumed to shift downward to contact a fully compressed plenum spring. Each fuel assembly is assumed to remain in contact with the canister bottom plate. The basket dimensions used assume conservative tolerances, and

the basket is conservatively assumed to be shifted upward to contact the canister shield lid. This bounding axial shifting scenario results in the maximum distance from the canister bottom plate to the lower end of the neutron absorber panels. For all UMS® PWR canister classes, this distance is limited to 5.22 inches. For all UMS® BWR canister classes, the distance is limited to 8.19 inches. However, all PWR and BWR fuel assembly types have rod bottom end caps, tie plates and/or components of the bottom end-fitting/nozzle that will not deform to a total height of less than 0.7 inches. Consequently, the top end axial fuel shifting condition, which considers exposed fuel lengths of 4.52 inches for PWR fuel and 7.625 inches for BWR fuel, bounds the bottom end axial fuel shifting condition.

End Impact Accident Condition Effect on System Reactivity

The bounding PWR system end impact event does not significantly affect the reactivity of the system. Therefore, poison sheet coverage is adequate for all allowed PWR fuel contents of the UMS® system. The bounding BWR end impact event increases the reactivity of the system. This increase in reactivity, a Δk_{eff} of 0.0249, is added to the most reactive accident condition system reactivity, $k_{eff} \pm 2\sigma$, of 0.9108 ± 0.0008 as determined in Section 6.4.3.3, to establish a maximum BWR system reactivity, $k_{eff} \pm 2\sigma$, of 0.9357 ± 0.0008. This value is less than the USL of 0.9426 identified in Section 6.5.5. Including code bias, code bias uncertainty, and statistical uncertainty (2σ) in accordance with Equation 6 in Section 6.5.3, the resulting system k_s of 0.9497 is less than 0.95. Thus, poison sheet coverage is adequate for all allowed BWR fuel contents of the UMS® system.

6.4.6 Regulatory Compliance

The licensing requirements for criticality analyses are provided in 10 CFR 71.55 and 10 CFR 71.59 for shipment of radioactive material.

10 CFR 71.55 and 10 CFR 71.59 require that the fissile material package be subcritical under any credible condition, e.g., optimum interior/exterior moderation and reflection and credible configuration of the material. A criticality transport index is to be assigned to the fissile material package. This transport index must be based on the number of packages (casks in this context) remaining subcritical in an array configuration.

Additional requirements imposed include the reduction in poison plate ^{10}B from 100 to 75 percent and water in the pellet-to-cladding gap.

loading restriction, the Westinghouse 17 x 17 OFA fuel criticality evaluation is bounding.

6.6.1.1.8 Damaged Fuel and Fuel Debris in the Maine Yankee Fuel Can

Damaged fuel assemblies are placed in one of two configurations of the Maine Yankee fuel can prior to loading in the basket (see Drawings 412-501 and 412-502). Both configurations of the the Maine Yankee fuel can have screened openings in the baseplate and the lid to permit drainage, vacuum drying and inerting. This evaluation conservatively considers 100% of the fuel rods in the fuel can as damaged. For conservatism, the fuel can configuration with the larger internal cross-section is used in these analyses.

Fuel debris must be loaded in a rod or tube structure that is subsequently loaded into one of two configurations of the Maine Yankee fuel can. The mass of fuel debris placed in the rod or tube is restricted to the mass equivalent of a fuel rod of an intact fuel assembly.

The Maine Yankee spent fuel inventory includes fuel assemblies with fuel rods inserted in the guide tubes of the assembly. If the integrity of the cladding of the fuel rods in the guide tubes cannot be ascertained, then those fuel rods are assumed to be damaged.

Damaged Fuel Evaluation

All of the spent fuel classified as damaged and all of the spent fuel not in its original lattice are stored in one of two configurations of the Maine Yankee fuel can. This fuel is analyzed using a 100% fuel rod failure assumption. The screened fuel can is designed to preclude the release of pellets and gross particulates into the canister cavity. Evaluation of the canister with four Maine Yankee fuel cans containing CE 14×14 fuel assemblies that have up to 176 damaged fuel rods, or consolidated fuel consisting of up to 289 fuel rods, considers 100% dispersal of the fuel from these rods within the fuel can. Either configuration of the Maine Yankee fuel can is restricted to loading in one of the four corner positions of the basket.

All loose fuel in each analysis is modeled as a homogeneous mixture of fuel and water, of which the volume fractions of the fuel versus the water are varied from 0-100%. By varying the fuel fraction up to 100%, this evaluation addresses fuel masses significantly larger than those available in a standard or consolidated fuel assembly. First, loose fuel from damaged fuel rods within a fuel assembly is evaluated between the remaining rods of the most reactive missing rod array. The results of this analysis, provided in Table 6.6.1.1-9, show a slight decrease in the reactivity of the system. Adding fuel to the already optimized H/U ratio of the bounding missing rod array reduces the reactivity of the system as this effectively returns the system to an undermoderated state. Second, loose fuel is considered above and below the active fuel region of

this most reactive missing rod array. This analysis is performed within a finite cask model. The results of this study, provided in Table 6.6.1.1-10, show that any possible mixture combination of fuel and water above and below the active fuel region and, hence, above and below the neutron absorber sheet coverage, will not significantly increase the reactivity of the system beyond that of the missing rod array. Loose fuel is also considered to replace all contents of the Maine Yankee fuel can with the internal square dimension of 8.52 inches in each four corner fuel tube location. As shown in Table 6.6.1.1-11, the results of this study, which include modeling of the Maine Yankee fuel can with the internal square dimension of 8.52 inches, show that any mixture of fuel and water within this cavity will not significantly increase the reactivity of the system beyond that of the missing rod array.

Damaged fuel within the fuel can may also result from a loss of integrity of a consolidated fuel assembly. As described in Section 6.6.1.1.7, the consolidated assembly missing rod study shows that a potentially higher reactivity heterogeneous configuration does not increase the overall reactivity of the system beyond that of loading 24 Westinghouse 17×17 OFA assemblies when this configuration is restricted to the four corner locations. The homogeneous mixture study of loose fuel and water replacing the contents of the Maine Yankee fuel can having an internal square dimension of 8.52 inches (in each of the four corner fuel tube locations) considers more fuel than is present in the 289 fuel rod consolidated assembly. This study shows that a homogeneous mixture at an optimal H/U ratio within the fuel can also does not affect the reactivity of the system.

Preferential flooding during the transport cask hypothetical accident flooding could result in different moderator densities existing in the transportable storage canister (canister) cavity and damaged fuel can cavity. To investigate the impact of preferential flooding of the canister and damaged fuel can cavity, various moderator densities were evaluated in the damaged fuel can while retaining a fully flooded canister, and vice versa. As shown in Table 6.6.1.1-12, any reduction in moderator density in either the canister or damaged fuel can will result in reduced system reactivities.

The transport cask loaded with the Westinghouse 17×17 OFA fuel assemblies is subcritical. Therefore, it is inherent that a statistically equivalent, or less reactive, canister loading of four Maine Yankee fuel cans containing assemblies with up to 176 damaged rods, or consolidated assemblies with up to 289 rods and 20 of the most reactive Maine Yankee fuel assemblies, is also subcritical. Therefore, assemblies with up to 176 damaged rods and consolidated assemblies with up to 289 rods are allowable contents as long as they are loaded into one of two configurations of the Maine Yankee damaged fuel cans.