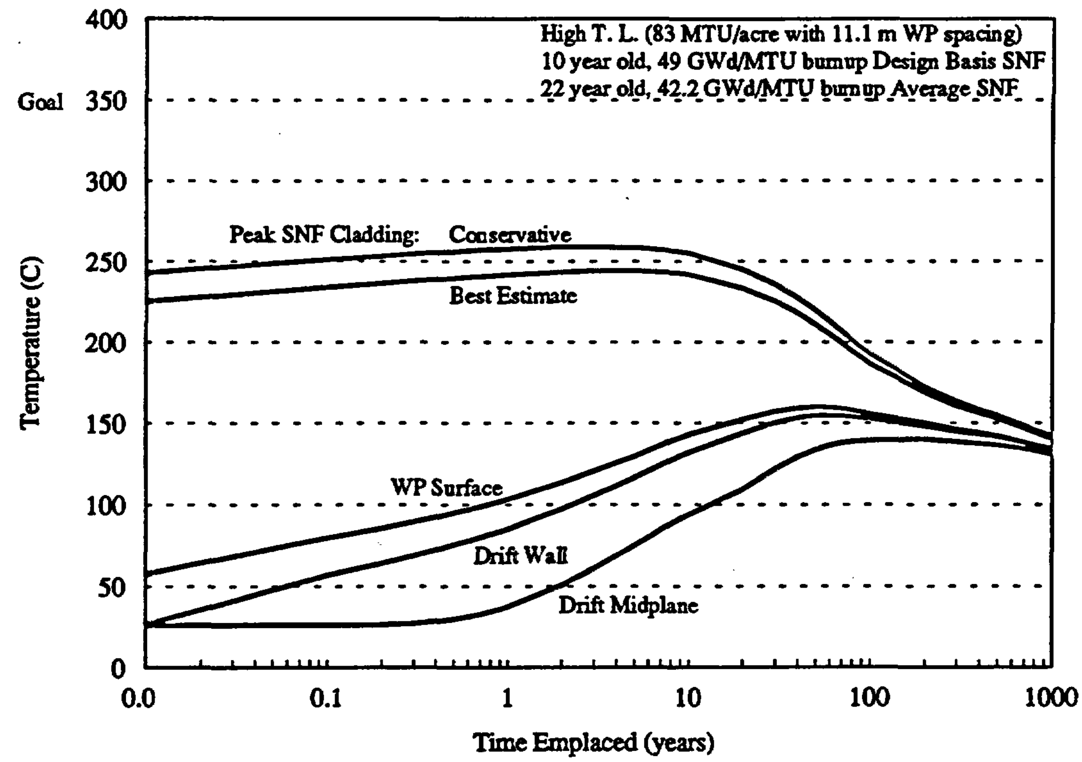

Figure 6.5-42. 24 BWR Multi-Purpose Canister, High Thermal Load (#2), MGDS Design Basis Fuel

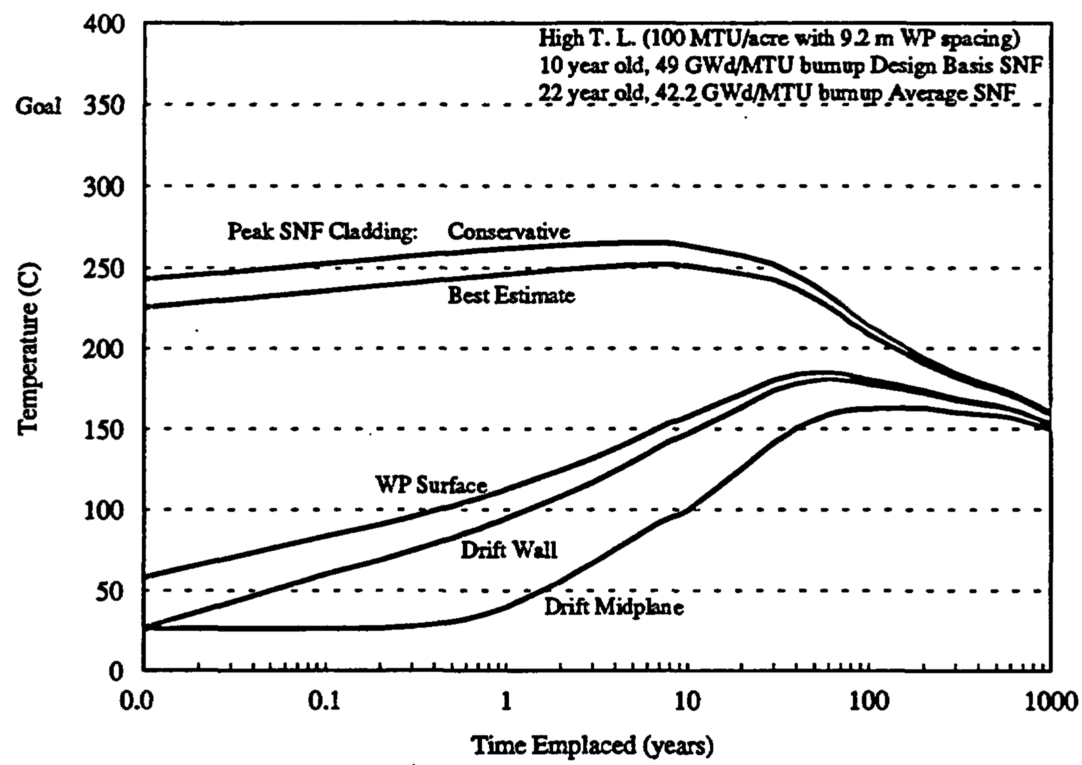

Figure 6.5-43. 24 BWR Multi-Purpose Canister, High Thermal Load (#1), MGDS Design Basis Fuel

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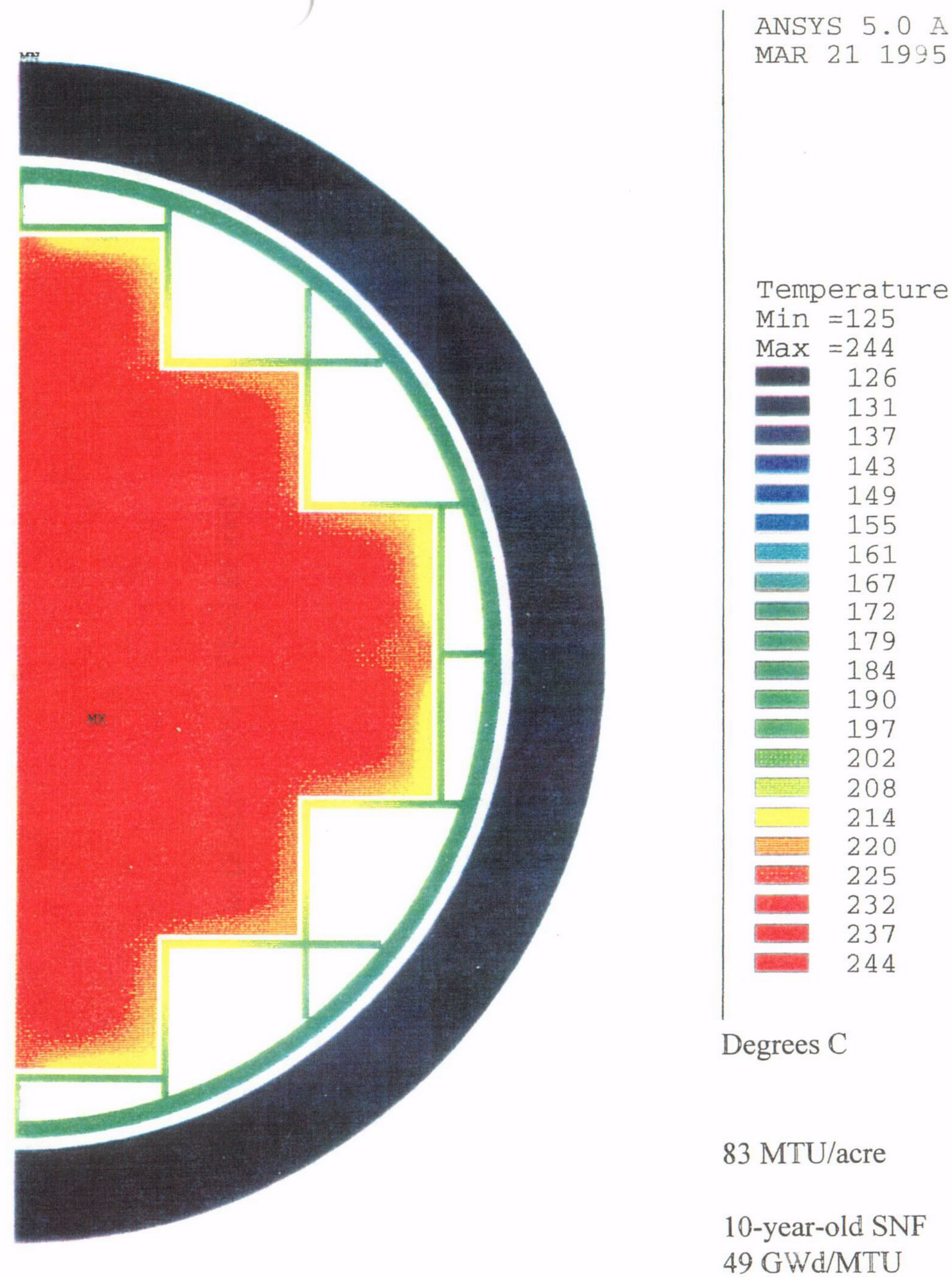


Figure 6.5-44. 24 BWR Multi-Purpose Canister Peak Temperatures at 4 Years

C01

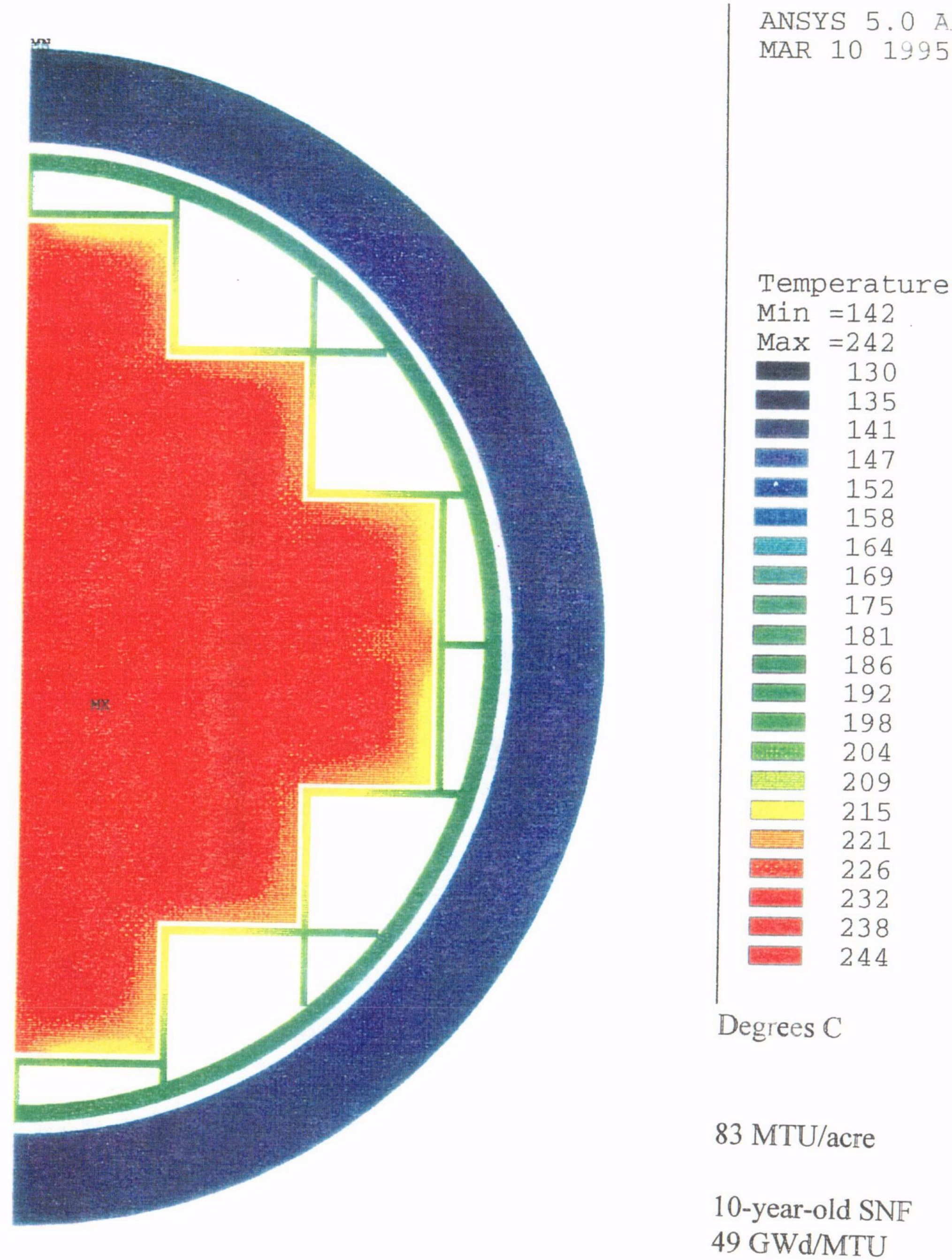
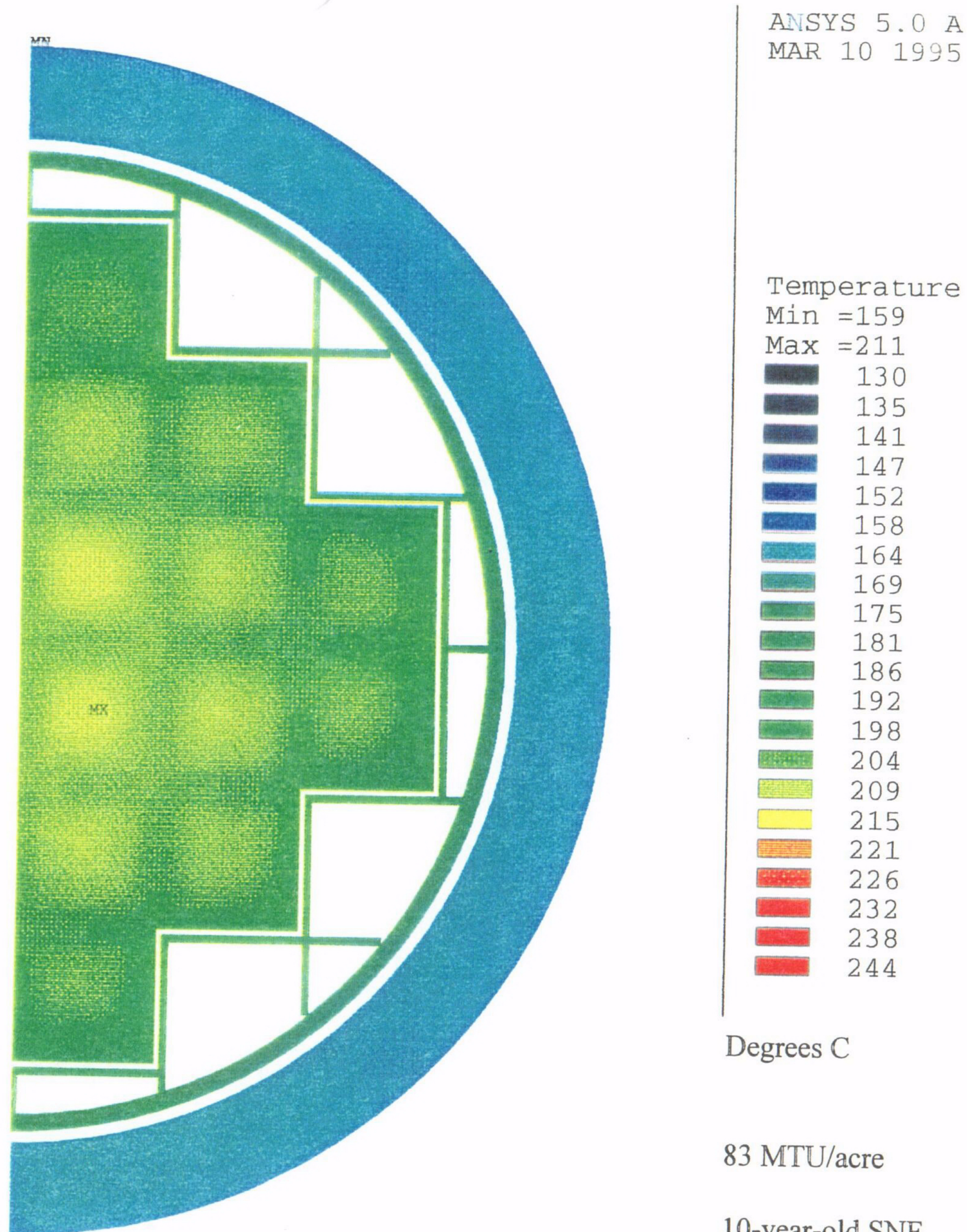


Figure 6.5-45. 24 BWR Multi-Purpose Canister Temperatures at 10 Years

C02



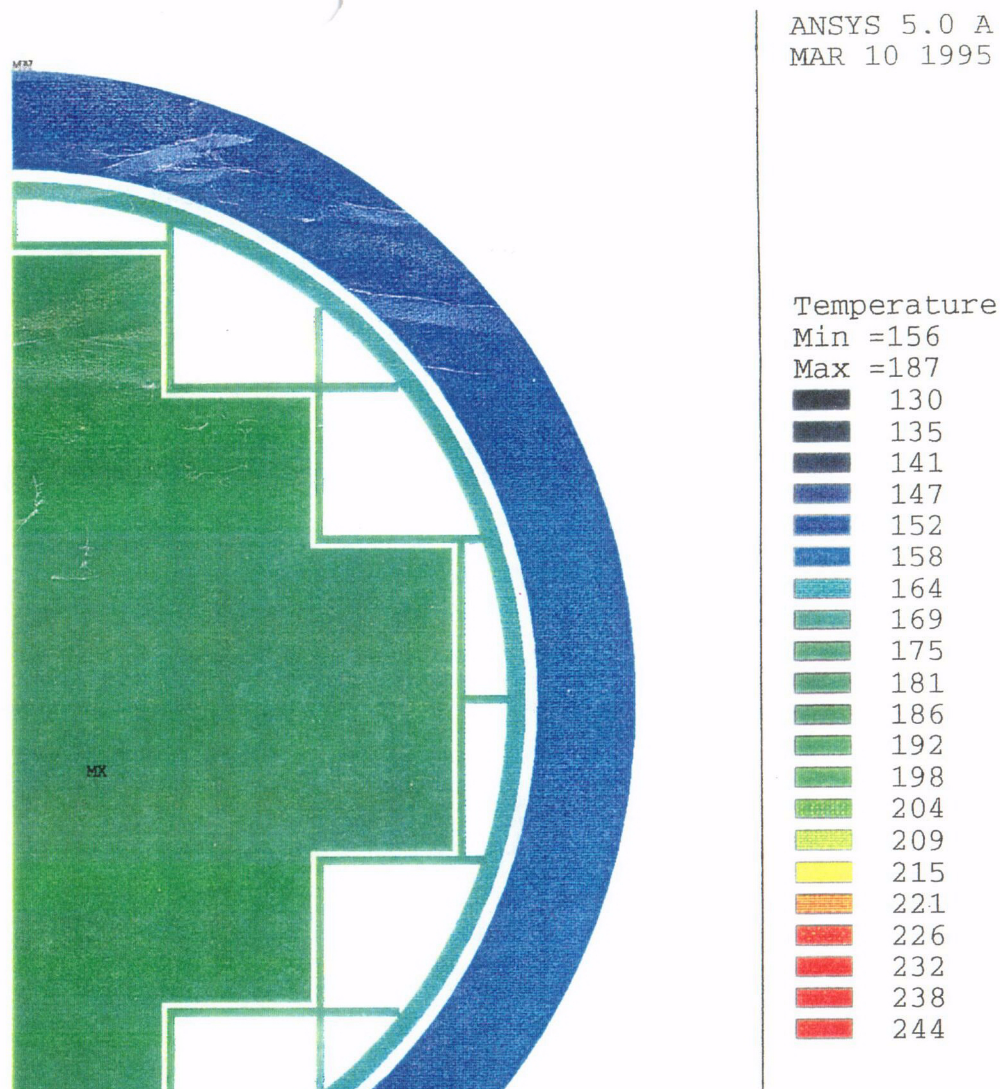
Degrees C

83 MTU/acre

10-year-old SNF
49 GWd/MTU

C03

Figure 6.5-46. 24 BWR Multi-Purpose Canister Temperatures at 50 Years



Degrees C

83 MTU/acre

10-year-old SNF
49 GWd/MTU

C04

Figure 6.5-47. 24 BWR Multi-Purpose Canister Temperatures at 100 Years

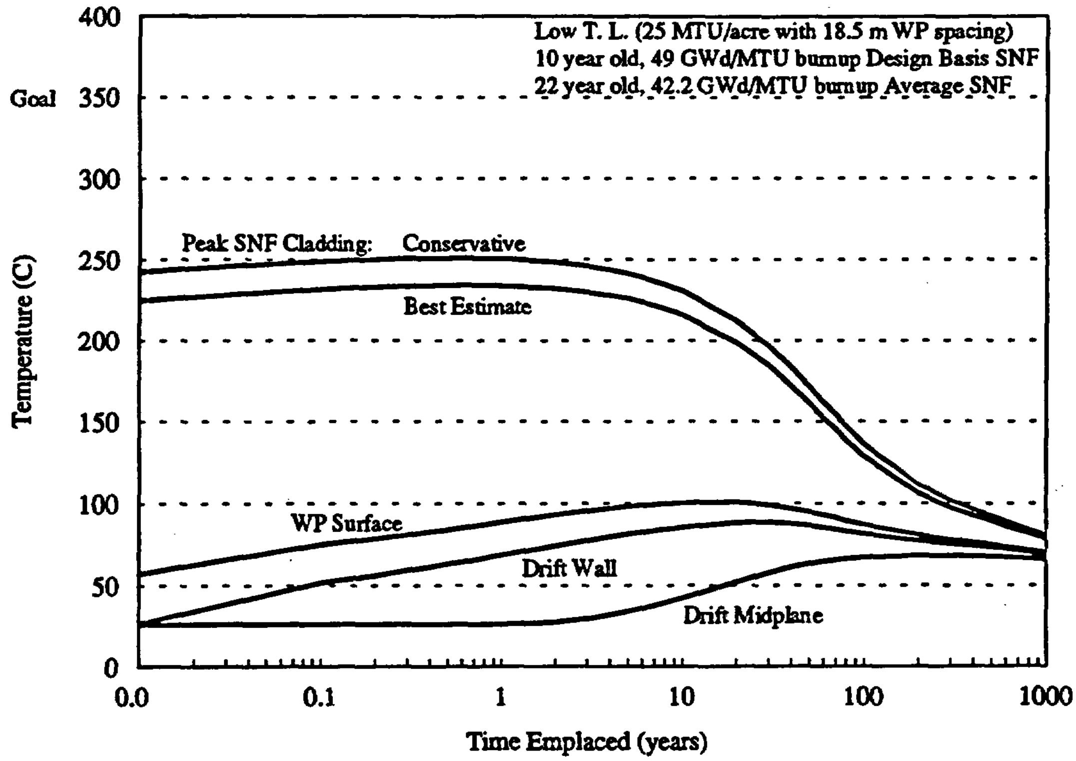


Figure 6.5-48. 24 BWR Multi-Purpose Canister, Low Thermal Load (#1), MGDS Design Basis Fuel

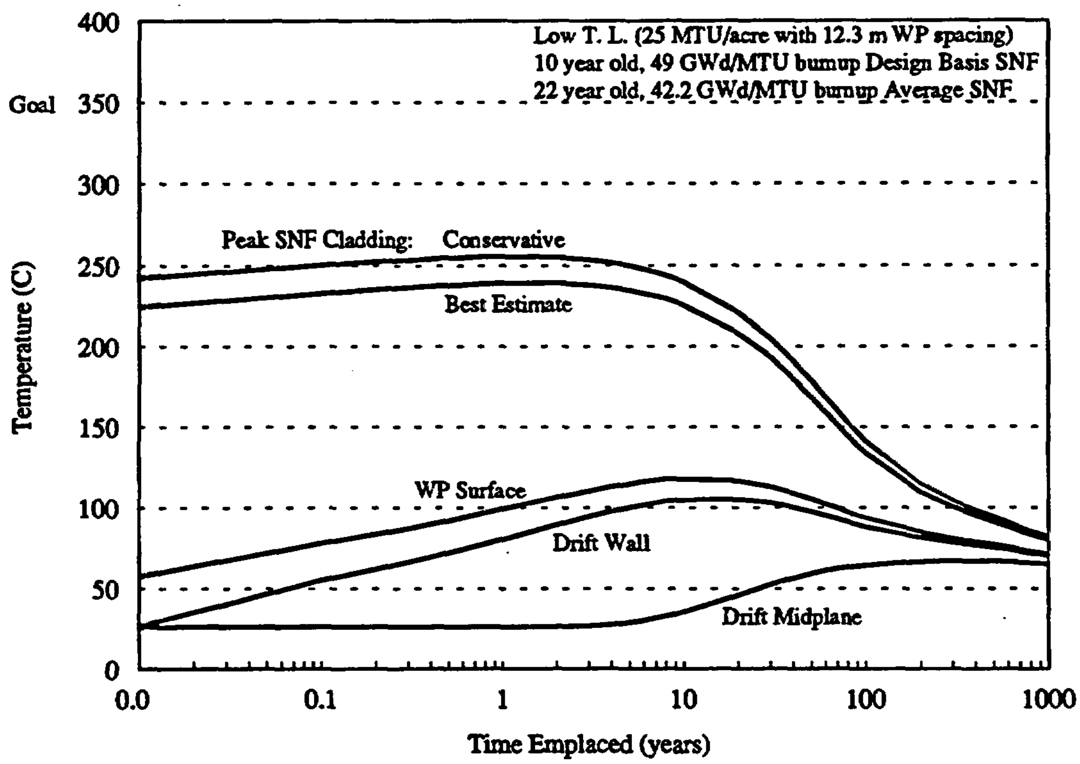


Figure 6.5-49. 24 BWR Multi-Purpose Canister, Low Thermal Load (#2), MGDS Design Basis Fuel

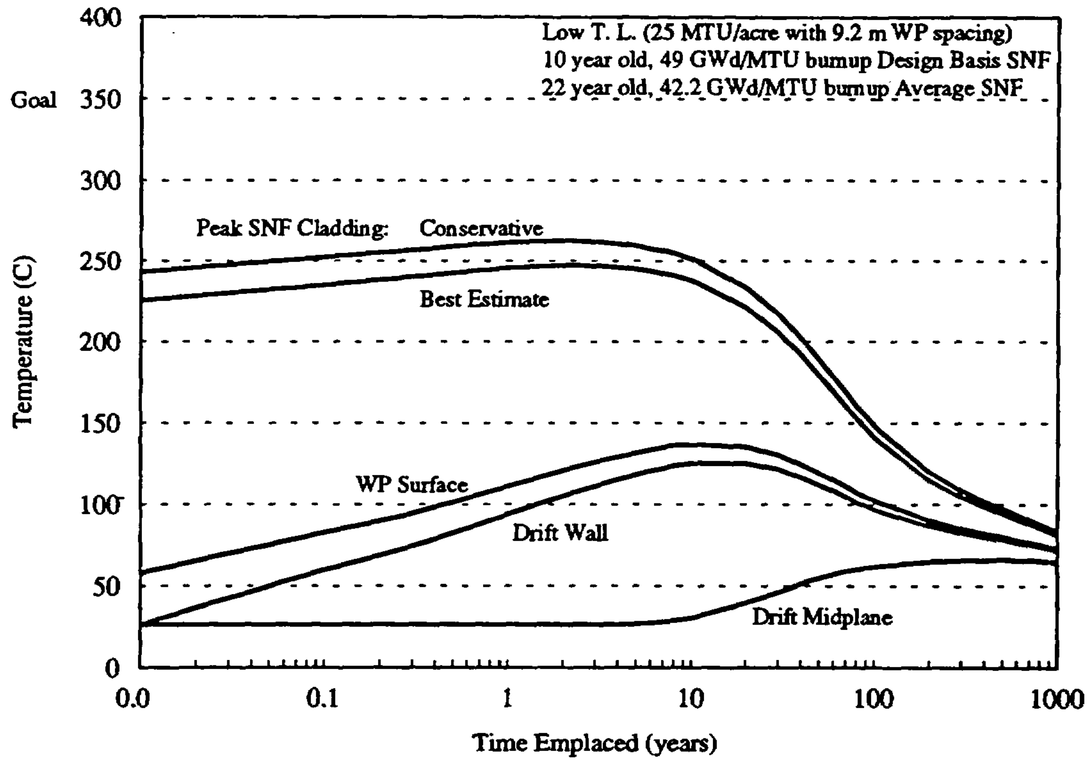


Figure 6.5-50. 24 BWR Multi-Purpose Canister, Low Thermal Load (#3), MGDS Design Basis Fuel

6.5.2 Structural Analysis

The WP must be shown to comply with all regulations and requirements that govern the containment of radionuclides. The regulations that the design must meet include 10 CFR 20, 10 CFR 60, and 40 CFR 191. These requirements state that the WP must remain intact as a unit for containing SNF and provide for the safe handling of the waste, at least until the end of the period of retrievability. The repository, and therefore the WPs, must be designed to preserve the option of waste retrieval throughout this period. The WP must also be capable of sustaining normal handling, packaging, and operational loads without loss of containment. Furthermore, it must be shown that the WP can survive design-basis accidents either without loss of containment or with a limited release of radionuclides.

The most damaging loads the WP must be able to endure are the accident loads. The accident scenarios analyzed for the WPs are a WP 2-m drop, a WP slap down, and a starter tunnel rock fall onto the WP. To be conservative, the acceptance criteria will be to show that there is no breaching of the WP barriers, excluding the skirts (the extensions of the outer barrier cylinder used for handling purposes). Material failure in the outer barrier skirts is acceptable because the skirts are not part of the containment barrier. Plastic deformation of the barriers is also acceptable during these accidents provided that there is no breaching of the material. Four different WPs are analyzed for this accident: the 21 PWR MPC, the 40 BWR MPC, the 24 BWR MPC, and the 12 PWR MPC. Throughout Section 6.5.2, these WPs will be referred to as the 21 PWR, the 40 BWR, the 24 BWR, and the 12 PWR.

The material properties used in the calculations are given in Table 6.5-6. Material behavior is approximated in the model with elastic/plastic bilinear stress-strain curves for the materials. The calculations are performed with room temperature properties because A 516 properties at elevated temperatures were not available. Mechanical properties of Type 316L stainless steel and Alloy 825 have little temperature dependence for the temperatures of interest. ASTM A 516 carbon steel has decreasing strength but increasing ductility at high temperatures (below the material creep temperature). Ductile materials tend to absorb more energy due to impact loads. Therefore, this assumption has only a minor effect on the results.

Table 6.5-6. Material Properties Table for Multi-Purpose Canister Waste Package Models

Material	Temp.	Yield Strength S_y	Ultimate Tensile Strength S_u	Elastic Modulus E	Poisson's Ratio υ	Percent Elongation
316L	20°C	172 MPa[a]	482 MPa[a]	195 GPa[b]	0.25[c]	40[a]
A 516 Grade 55	20°C	205 MPa[d]	515 MPa[d]	206 GPa[e]	0.30[e]	27[d]
Alloy 825	20°C	338 MPa[f]	662 MPa[f]	206 GPa[f]	0.42[f]	45[f]
Rock*	-	-	-	32.7 GPa[g]	0.22[g]	-

Table Notes:

* The Yield strength (S_y), ultimate tensile strength (S_u), and percent elongation are used to determine plastic properties and S_u is used to determine acceptability of the stress levels. For conservatism the rock is assumed not to fail. Therefore, the stresses in the rock are not evaluated and no plastic properties of the rock are included in the analysis. Thus S_y and S_u are not required to perform this analysis.

a. Holt, John M., Harold Mindlin, and C.Y. Ho 1992. (pp. 3-5)
b. American Society of Mechanical Engineers (ASME) 1992. (Section II, Part D, Subpart 2, Table TM-1)
c. *INCO Alloys International* 1963.
d. American Society for Testing and Materials (ASTM) A 516/A 516M 1990. (p. 321)
e. *Metals Handbook, Tenth Edition, Volume 1, Properties and Selection: Irons, Steels, and High-Performance Alloys*. 1990 (p. 374)
f. *INCO Alloys International* 1992. (p. 3)
g. YMP 1994a

6.5.2.1 Waste Package 2-m Drop Accident Analysis

Prior to licensing, the WP must be shown to be capable of surviving certain accident conditions without resulting in a breach of radionuclide containment. One possibly damaging accident is a drop of the WP. The maximum height to which the WP is assumed to be lifted during transportation is 2 m (CRWMS M&O 1995n). Therefore, this height is selected for the WP drop analysis.

This hypothetical accident condition is a free drop of the WP from a height of 2 m onto a flat, essentially unyielding horizontal surface. The horizontal surface cannot be perfectly unyielding because that would require a surface with infinite stiffness. Because such a surface cannot be modeled, these analyses use the next best option, an essentially unyielding surface which has a very high stiffness. The case analyzed is one in which the WP strikes the ground at an angle that puts the center of mass directly above the point of impact. These conditions cause the highest loads on the WP outer barrier and therefore are expected to cause the most damage. The models described and the analysis results given in Section 6.5.2.1 are taken from *Scoping Evaluation to Explore - Two Meter Drop Finite-Element Analysis of Multi-Purpose Canister Waste Packages* (CRWMS M&O 1995aj).

6.5.2.1.1 21 PWR Multi-Purpose Canister

6.5.2.1.1.1 Description of Finite-Element Model for Waste Package 2-m Drop

The ANSYS 5.0A finite-element analysis code is used for the structural analysis of the WP 2-m drop accident scenario. Three-dimensional brick elements are used in modeling all parts of the WP. Some capabilities of the element used to perform this nonlinear transient dynamic analysis are plasticity, large deflection, and large strain.

The model developed for the 2-m drop analysis of the 21 PWR is a one-half symmetry three-dimensional finite-element model, shown in Figure 6.5-51. The analysis incorporates the MPC basket assembly including structural angles, SNF assembly weight, MPC shell, and MPC disposal container. Material properties of the basket assembly, MPC shell, and MPC disposal container (including inner and outer barriers) are defined in the model with their individual property tables. The MPC shell and basket assembly are given properties of 316L stainless steel, the inner barrier is given properties of Alloy 825, and the outer barrier is given properties of A 516.

The WP is symmetric about a plane that runs along the longitudinal axis, so only half of the WP is modeled, and symmetry boundary constraints are placed on the symmetry plane. This approach has been verified on a simple model, which shows that stress distribution results obtained for both half and full models are identical in terms of normal and shear stress components. Stress plots are provided in Figures 6.5-52 and 6.5-53.

Taking advantage of the symmetry reduces the model size, allowing more detail in the half that is modeled. However, even with the half model, the element mesh must remain somewhat coarse to keep run times down and maintain manageability of the output files. Therefore, some other simplifications have been made to the model.

- The model is constructed as if the inner and outer barriers are fabricated as one piece. It is not yet known if the WP will be fabricated in this manner, but using this assumption prevents the need for contact elements between the inner and outer barriers. It is best to avoid the use of contact elements wherever possible to prevent convergence problems and to reduce the model size and run times. While the inner and outer barriers are treated as contiguous, there is an element border between the inner and outer barriers, and the appropriate material properties are used for each barrier.

- The gap between the MPC disposal container and the MPC shell is closed by modifying the MPC disposal container inner and outer diameters in accordance with the calculations to keep the moment of inertia constant. By keeping the modulus of elasticity, moment of inertia, and length of the WP the same, the stiffness remains the same. Therefore, the simplification of considering the MPC disposal container and the MPC shell as one piece with different material characteristics will not affect finite-element analysis results.

- The basket assembly is made up of tubes constructed of Al-B sandwiched between layers of 316L stainless steel (CRWMS M&O 1994k). The stress in a structural member made of two or more materials with different moduli of elasticity when subjected to pure bending can be determined by transforming all materials into one homogeneous material (Beer and Johnston 1987). This procedure involves determination of a multiplication factor for inertia. In modeling the WP, the Al-B section of the SNF basket assembly has been transformed into 316L stainless steel by decreasing the thickness of the Al-B by a factor that is the ratio of the elastic modulus of Al-B (properties of Al alloy 5086 assumed) to the elastic modulus of 316L, (71 GPa)/(195 GPa) = 0.364 (ASME 1992). The thickness of the inner and outer parts of the fuel cell walls made of 316L stainless steel were added to the thickness of the transformed section to determine a total thickness for the tube walls. Adjacent tube walls were also merged to reduce the number of elements in the basket.

- No structural credit is taken for the depleted uranium of the shield plug or the honeycomb steel spacer. These parts have been considered as gaps between the lids and subtracted from the equivalent material thickness. The complexity of structural members along the longitudinal axis has also been simplified by combining all gaps and the lids and shield plug and modeling a single equivalent lid and single gap at each end of the WP.

- The SNF load has been calculated for one-half of the MPC, and incorporated into the basket assembly and MPC shell total mass. This has been accomplished by increasing the density of the 316L stainless steel in proportion to the mass for a constant volume of MPC shell and basket assembly.

The simplifications made in this model are deemed reasonable because for conceptual design, overall system response is more important than the discrete calculation of stresses in the various radii and corners that a finer mesh allows. These discrete calculations will be performed at a later design stage.

The only force acting on the WP is its own mass times gravity. Maximum linear momentum transfer of the WP will occur along a vertical plane passing through its center of gravity, so the WP has been rotated 17.41° with respect to the vertical plane. This puts the center of mass directly above the contact point with the ground, causing the WP to be in an orientation that will result in maximum plastic deformation on the outer barrier skirt. The distance from the flat surface to the WP bottom edge is 2 m.

6.5.2.1.1.2 Barrier Response

As is described in the previous section, the drop considered is a corner drop with the WP rotated 17.41° from vertical. A corner drop can be separated into two individual drop events: initial impact and slap down. For impact angles near vertical, the initial impact dominates and the WP response resembles the condition of an end drop. For impact angles near horizontal, the slap down phase dominates and the assembly response resembles the condition of a side drop (SNL 1992a, pp. 53-64). A corner drop of 17.41° with respect to the vertical axis resembles the condition of an end drop and results in the largest deformation pattern on the barrier. Thus, the effect of second impact on the opposite end of the WP will not be critical when compared to the first impact. The slap down case is analyzed separately and the results are given in Section 6.5.2.2.

The skirt on the outer barrier behaves as an impact limiter during the WP 2-m corner-drop accident scenario. The linear momentum of the WP in the vertical direction results in maximum damage on the skirt in terms of elastic and plastic deformations. Figure 6.5-54 shows the principal stress distribution (S_1) on the WP and on the locally deformed region in the vicinity of the contact area with the flat surface.

An outer barrier breach can occur by ductile tearing as a result of excessive stress or strain. Sharp curvature bending of the impacted WP corner is the reason for very high stress magnitudes which may result in ductile rupture. However, since the skirt functions as an impact limiter in this accident condition, failure is expected at the skirt location of the MPC disposal container. Failure is acceptable in the skirt because a failure in this region will not cause loss of radionuclide containment. As can be observed in Figure 6.5-55, the effect of impact is not critical in any regions of the outer barrier other than the skirt.

The inner surface of the skirt makes a 90° angle with the WP outer lid at the location where the skirt is extended from the container. This sharp corner could be eliminated by introducing a corner radius during the manufacturing process. This might slightly reduce the excessive deformation of the skirt by increasing its stiffness. However, the effect of a corner radius would still not be able to prevent the skirt from deforming plastically because the dynamic loading on the skirt is very high.

The transient simulation of dynamic loading during the drop was terminated after the first impact since the second impact was previously determined to be less critical than the first. It should also be noted that the time step of the finite-element analysis from which the results are taken was selected to include the maximum deformation pattern on the containment barrier.

The maximum stress and strain magnitudes were obtained in the vicinity of the outer barrier skirt where the impact causes local ductile rupture. The maximum principal stress magnitude on the WP is 777 MPa, see Figure 6.5-54. When the ultimate tensile strength of the outer barrier (515 MPa) is compared to the maximum first principal stress (777 MPa), it can be concluded that there is a localized material failure around the region of impact in the skirt. Figure 6.5-55 shows that there is no material failure in any part of outer barrier other than in the skirt since the maximum first principal stress (509 MPa) is less than the ultimate tensile strength of the outer barrier material (515 MPa). In the rest of the finite-element model, the stress does not exceed 54.8 MPa, see Figure 6.5-56, which is below the yield strength of all materials considered for this design analysis. The analysis of the basket assembly is addressed in the following section.

It is concluded that the 2-m WP drop accident will cause ductile rupture of the skirt. Excluding the skirt, the inner and outer barriers will experience plastic deformation but will not rupture. The rest of the WP will retain its structural integrity since the stress magnitudes are below the yield strength in these regions.

6.5.2.1.1.3 Spent Fuel Assembly Basket Response

Of the various basket components, the cells are the most critical items. They must not be permitted to deform plastically during accident conditions in order to ensure that they maintain the geometric spacing of the SNF (CRWMS M&O 1994j). Because the cells are part of the MPC, they have already been analyzed for certain loading conditions (CRWMS M&O 1994k). In the analyses performed here, the intent is to show that the loading on the basket components are less than those analyzed previously, and therefore are acceptable.

Previous structural analysis of the MPC is based upon a $60g$ design limit for a 9-m drop accident scenario (CRWMS M&O 1994k). The design was found to be adequate for the $60g$ load, so any lower load will not cause damage that would be considered detrimental to the proper functioning of the basket.

Based on this information, the maximum load on the basket is calculated in terms of the number of g's it will experience. Because of some local deformation in the skirt on the outer barrier, the basket assembly will experience a different maximum loading than the WP in its entirety. For that reason, the section of the basket having the highest stress was extracted from the entire WP model to be addressed separately.

The section of the basket removed is the bottom center tube. Because of the one-half symmetric model, this is the half tube on the symmetry edge. The maximum vertical force on this tube is obtained as part of the ANSYS output for the 2-m drop case. The weight of the section is calculated from the dimensions and density used in the ANSYS model. The maximum g load is then calculated by dividing the maximum vertical force on the tube by the weight of the tube. The maximum load on this tube of the basket is found to be $46.1g$. This is less than the $60g$ load limit, so the basket of the 21 PWR MPC can withstand a WP 2-m corner-drop accident.

6.5.2.1.2 40 BWR Multi-Purpose Canister

6.5.2.1.2.1 Description of Finite-Element Model for Waste Package 2-m Drop

Because the mass and area moment of inertia characteristics for the 40 BWR are similar to those for the 21 PWR, one finite-element model has been considered to represent both WP concepts. Therefore, correlations between the 21 PWR and the 40 BWR are made where applicable.

6.5.2.1.2.2 Barrier Response

The mass of the 40 BWR is slightly less than that of the 21 PWR, whereas the area moment of inertia is higher. Therefore, deformation patterns and stress distributions obtained for the 21 PWR are considered to be limiting. Thus, the worst case containment barrier stresses for a large WP (i.e., 125 ton) are presented in Section 6.5.2.1.1.2.

6.5.2.1.2.3 Spent Fuel Assembly Basket Response

The basket of the 40 BWR has tubes that are smaller than the tubes in the 21 PWR but have the same thickness. Therefore, it is expected that the basket of the 40 BWR can withstand more load than the basket of the 21 PWR without failing. However, the loading on the basket of the 40 BWR is not the same as the loading on the basket of the 21 PWR. A correlation developed using conservation of energy indicates that the *g* load on the basket is inversely proportional to the square root of the mass. The reason for this relation is that a lower mass results in a smaller skirt deflection during impact, and therefore, higher deceleration.

$$g\ load = \frac{F}{W} = \frac{a}{g} = \sqrt{\frac{2kh}{mg}} \qquad \text{(Equation 6.5-1)}$$

where: F = force
W = weight
a = acceleration
g = gravitational acceleration
k = interface stiffness
h = drop height
m = mass

This correlation is based on perfectly elastic impact, so accuracy will decrease as the difference in plastic deformation between the two cases increases. Therefore, this correlation is only used to compare WPs of approximately the same size. The 40 BWR has a slightly lower mass than the 21 PWR, and therefore, a slightly higher *g* load is present on the basket. The difference in mass of the 21 PWR and the 40 BWR is small (less than 1 percent, see Table 6.1-4), and according to the calculation, the maximum load on the basket of the 40 BWR MPC is 46.3*g*. This is less than the load limit of 60*g*, so the basket of the 40 BWR MPC can withstand a WP 2-m corner-drop accident.

6.5.2.1.3 24 BWR Multi-Purpose Canister

6.5.2.1.3.1 Description of Finite-Element Model for Waste Package 2-m Drop

The model developed for the 2-m drop analysis of the 24 BWR is a one-half symmetry three-dimensional finite-element model, shown in Figure 6.5-57. The analysis incorporates the MPC disposal container, MPC shell, basket assembly, and SNF weight as they were previously modeled for the 21 PWR configuration. However, there are a few differences in the geometry of the finite-element mesh. The containment barrier diameters are smaller, the fuel assembly tubes have a smaller side length, and the number of fuel assembly tubes is higher.

Like the 21 PWR, the 24 BWR is symmetric about a plane that runs along the longitudinal axis, so only half of the WP is modeled, and symmetry boundary constraints are placed on the symmetry plane. Taking advantage of the symmetry reduces the model size, allowing more detail in the half that is modeled. However, even with the half model, the element mesh must remain somewhat coarse to keep run times down and maintain manageability of the output files. The same simplifications made for the 21 PWR model, see Section 6.5.2.1.1.1, have been made to this model with the exception of the following:

- The Al-B section of the SNF basket assembly is not included in the model of the 24 BWR, since the modulus of elasticity and the density have nearly the same ratios for Al-B and 316L stainless steel.

  $E_{Al} = 71$ GPa (ASME 1992, Section II, Part D, Subpart 2, Table TM-2)

  $E_{SS} = 195$ GPa (ASME 1992, Section II, Part D, Subpart 2, Table TM-1)

  $E_{Al}/E_{SS} = 71 \text{ GPa}/195 \text{ GPa} = 0.364$

  $\rho_{Al} = 2657 \text{ kg/m}^3$ (ASME 1992, Section II, Part D, Subpart 2, Table NF-2)

  $\rho_{SS} = 7953 \text{ kg/m}^3$ (Holt, Mindlin, and Ho 1992)

  $\rho_{Al}/\rho_{SS} = (2657 \text{ kg/m}^3)/(7953 \text{ kg/m}^3) = 0.334$

  Thus, no structural credit is taken for the Al-B and no penalty is taken for its mass. This simplification was realized after completing the three-dimensional finite-element analysis of the 21 PWR WP 2-m drop accident scenario. This approach reduced the time spent for the model development of the 24 BWR. The WP 2-m drop accident scenario of the 21 PWR was not rerun with the Al-B section excluded since inclusion of the Al-B is also valid. This simplification is tested by performing calculations on the relation between the total mass of the Al-B section and resulting stress magnitudes. The difference in the total mass of the 24 BWR with and without the simplification is less than 2.2 percent and changes the stresses less than 1.1 percent. Because this is likely to be smaller than the accuracy of the solution, this difference is neglected.

Maximum linear momentum transfer of the WP will occur along a vertical plane passing through its center of gravity, so the WP has been rotated 15° with respect to the vertical plane. This puts the center of mass directly above the contact point with the ground, causing the WP to be in an orientation that will result in maximum plastic deformation of the outer barrier skirt. The distance from the flat surface to the WP bottom edge is 2 m.

6.5.2.1.3.2 Barrier Response

As is described in the previous section, the drop considered is a corner drop with the WP rotated 15° from vertical. A corner drop can be comprised of two separate drop events: initial impact and slap down. For WP impact angles near vertical, the initial impact dominates and the WP response resembles the-condition of an end drop. For impact angles near horizontal, the slap down phase dominates and the assembly response resembles the condition of a side drop (SNL 1992a, pp. 53-64). Corner drop loading of 15° with respect to the vertical axis resembles the condition of an end drop and results in the largest deformation pattern on the barrier. Thus, the effect of second impact on the opposite end of the WP will not be critical when compared to the first impact. The slap down case is analyzed separately and the results are given in Section 6.5.2.2.

The skirt on the outer barrier behaves as an impact limiter during the WP 2-m corner-drop accident scenario. The linear momentum of the WP in the vertical direction results in maximum damage on the skirt in terms of elastic and plastic deformations. Figure 6.5-58 shows the principal stress distribution (S_1) on the WP and on the locally deformed region in the vicinity of the contact area with the flat surface.

An outer barrier breach can occur by ductile tearing as a result of excessive stress or strain. Sharp curvature bending of the impacted WP corner is the reason for very high stress magnitudes, which may result in ductile rupture. Since the skirt functions as an impact limiter in this accident condition, failure is expected at this location of the MPC disposal container. Failure is acceptable in the skirt because a failure in this region will not cause loss of radionuclide containment. As can be observed in Figure 6.5-59, the effect of impact is not critical in any regions of the outer barrier other than the skirt.

The transient simulation of dynamic loading during the drop was terminated after the first impact since the second impact was previously determined to be less critical than the first. It should also be noted that the time step of the finite-element analysis from which the results are taken was selected to include the maximum deformation pattern on the containment barrier.

The mass of the 24 BWR is nearly 19 metric tons less than the mass of the 21 PWR configuration, see Table 6.1-4. Considerable differences in the mass and geometry cause different stress magnitudes and distribution after the impact of the 24 BWR on the flat surface. The maximum stress at the impact region is 1160 MPa, see Figure 6.5-58. This magnitude of stress causes ductile rupture of the material in the skirt. The mechanism of this failure and its location raise a discussion similar to that made for the 21 PWR.

When the ultimate tensile strength of the outer barrier (515 MPa) is compared to the first principal stress value (1160 MPa), it can be concluded that there is a localized material breach around the region of impact in the skirt. Figure 6.5-59 shows that there is no material breach or ductile rupture in any part of the outer barrier other than the skirt since the maximum first principal stress (417 MPa) is less than the ultimate tensile strength of the outer barrier material (515 MPa). In the rest of the finite-element model, the stress does not exceed 60.9 MPa, which is below the yield strength of all materials considered for this design analysis, see Figure 6.5-60. The analysis of the basket assembly is addressed in the following section.

It is concluded that the WP 2-m drop accident will cause ductile rupture of the skirt. Excluding the skirt, the inner and outer barriers will experience plastic deformation but will not rupture. The rest of the WP will retain its structural integrity since the stress magnitudes are below the yield strength in these regions.

6.5.2.1.3.3 Spent Fuel Assembly Basket Response

The 24 BWR is smaller and lighter than the previously analyzed WPs, so it experiences loads that are significantly different from those calculated for the 21 PWR and 40 BWR. Because of some local deformation in the skirt on the outer barrier, the basket assembly will experience a different maximum loading than the WP in its entirety. For that reason, an individual tube of the SNF assembly basket is extracted from the 24 BWR model to be analyzed for vertical loads. The basket of the 24 BWR has been analyzed to the same loading limits, 60*g* (CRWMS M&O 1994k), as the larger MPCs, so that will be the loading limit to which the basket forces are compared to verify structural adequacy.

The section of the basket removed is the bottom tube of the one-half symmetry model. The maximum vertical force on this tube is obtained as part of the ANSYS output for the 2-m drop case. The weight of the section was calculated from the dimensions and density used in the ANSYS model. The maximum *g* load is then calculated by dividing the maximum vertical force on the tube by the weight of the tube. The maximum load on this tube of the basket is 44.3*g*. This is less than the 60*g* load limit, so the basket of the 24 BWR MPC can withstand a WP 2-m corner-drop accident.

6.5.2.1.4 12 PWR Multi-Purpose Canister

6.5.2.1.4.1 Description of Finite-Element Model for Waste Package 2-m Drop

Because the mass and area moment of inertia characteristics for the 12 PWR are similar to those for the 24 BWR, one finite-element model has been considered to represent both WP concepts. Structural complexity and geometric details of the 12 PWR MPC basket made its modeling more difficult than modeling of the 24 BWR. Because the model size prevented detailed modeling of the basket and loading of the MPC basket was already addressed in *MPC Conceptual Design Report* (CRWMS M&O 1994k), modeling of the simpler design was chosen. Therefore, correlations between the 24 BWR model and the 12 PWR are made where applicable.

6.5.2.1.4.2 Barrier Response

The 12 PWR and the 24 BWR configurations have nearly the same mass (the 12 PWR MPC WP has approximately 1.4 percent more mass, see Table 6.1-4), and there is a very small difference in terms of the area moments of inertia of the basket assemblies. Therefore, deformation patterns and stress distributions obtained for the 24 BWR are very close to those of the 12 PWR. Maximum stresses in the 12 PWR will be within a few percent of the maximum stresses in the 24 BWR. Because there is sufficient margin for stresses in the 24 BWR, there will still be sufficient margin for the 12 PWR.

6.5.2.1.4.3 Spent Fuel Assembly Basket Response

The 12 PWR has a slightly higher mass than the 24 BWR, and they have the same diameter, so the 12 PWR is considered to be limiting for outer barrier stresses and loads. However, the correlation used in Section 6.5.2.1.2.3 to compare the 21 PWR and the 40 BWR indicates that the *g* load is inversely proportional to the square root of the mass. The difference in mass of the 24 BWR and 12 PWR is small (approximately 1.4 percent), and according to the calculation, the maximum load on the basket of the 12 PWR MPC is 44*g*. This is slightly less than the *g* load on the basket of the 24 BWR MPC, so the *g* load on the 24 BWR MPC basket is limiting. Therefore, it is concluded that the basket of the 12 PWR MPC can withstand a WP 2-m corner-drop accident.

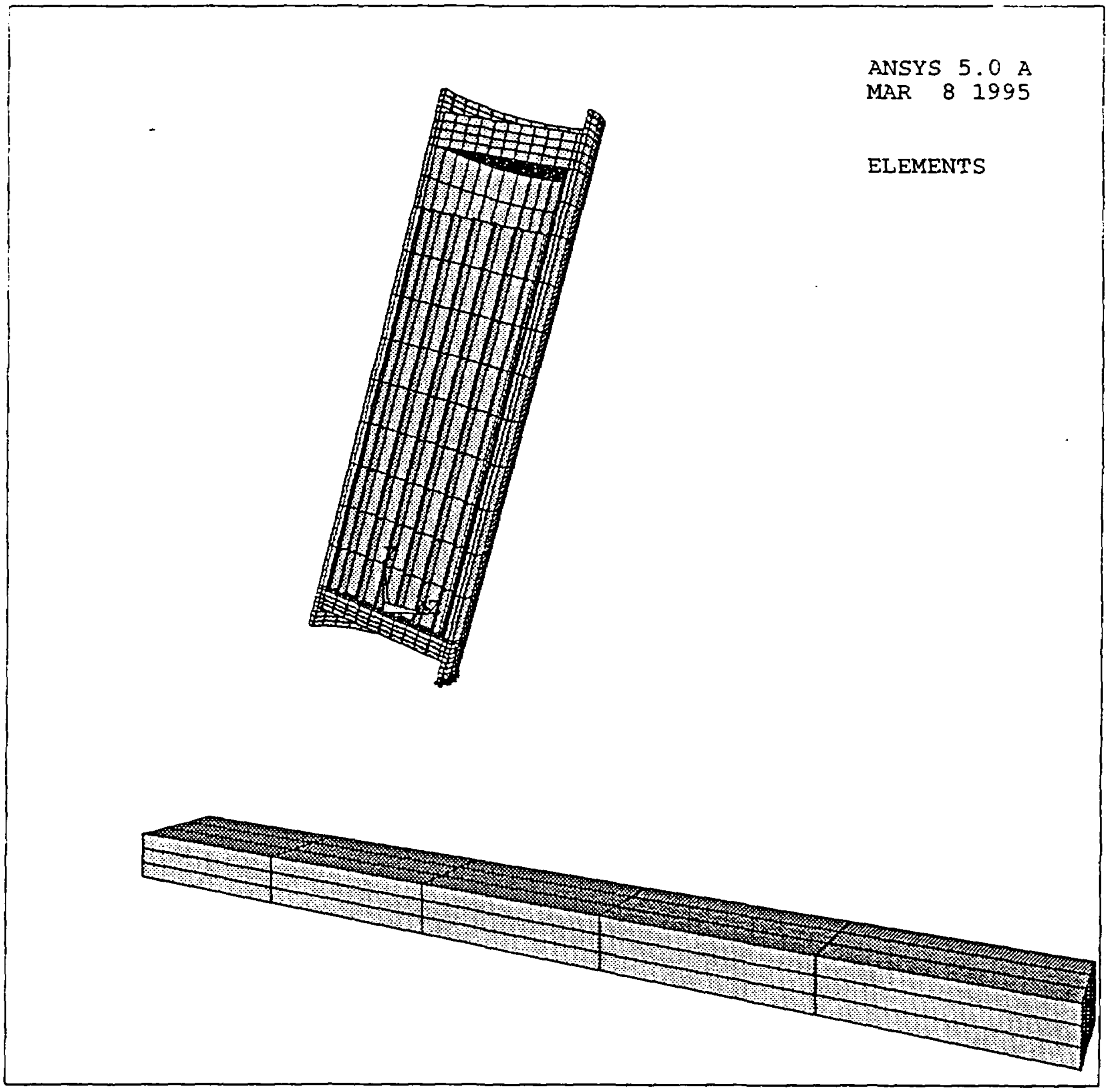


Figure 6.5-51. 21 PWR Multi-Purpose Canister Waste Package 2-m Drop Model

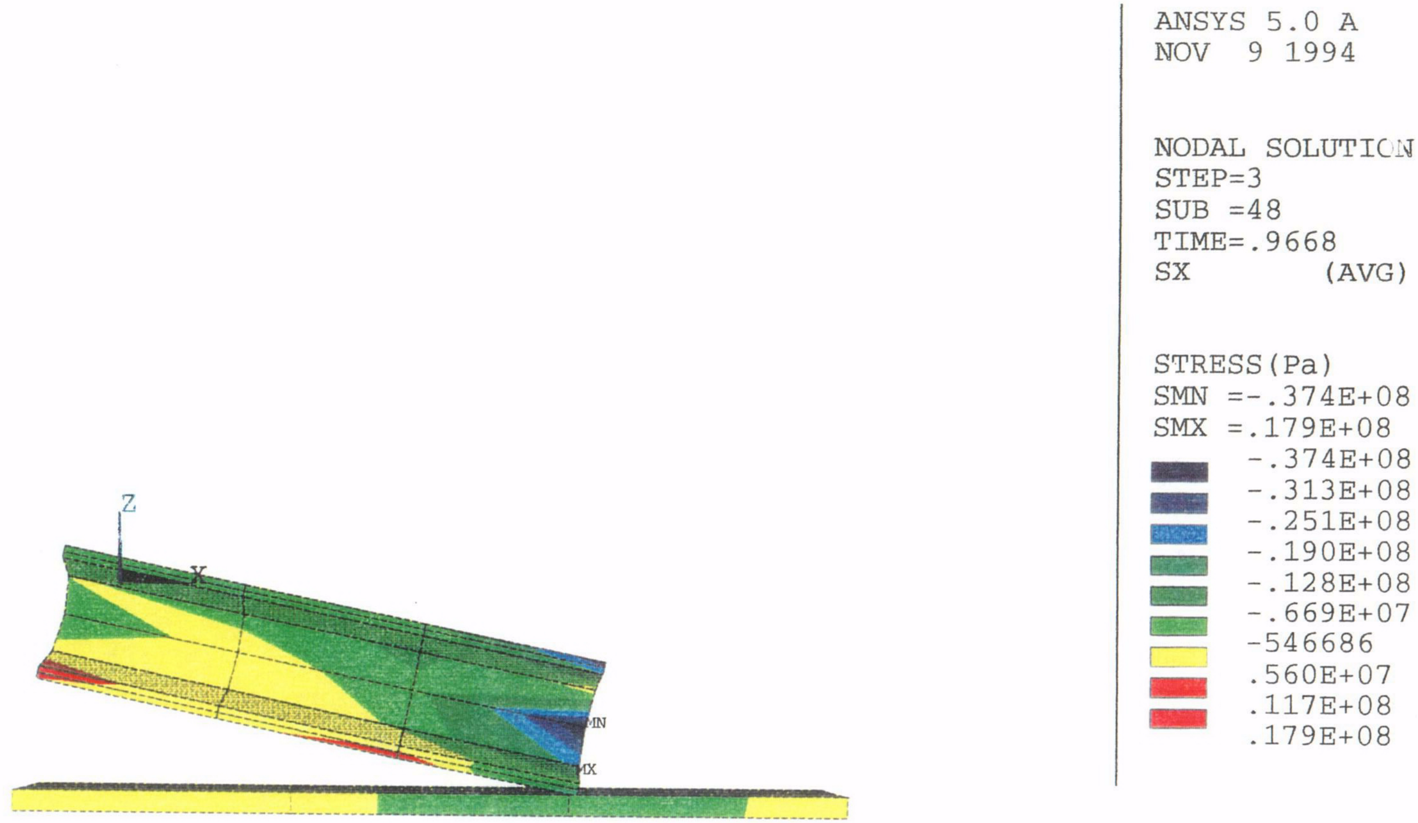


Figure 6.5-52. Half Model Stress Plot Example

C05

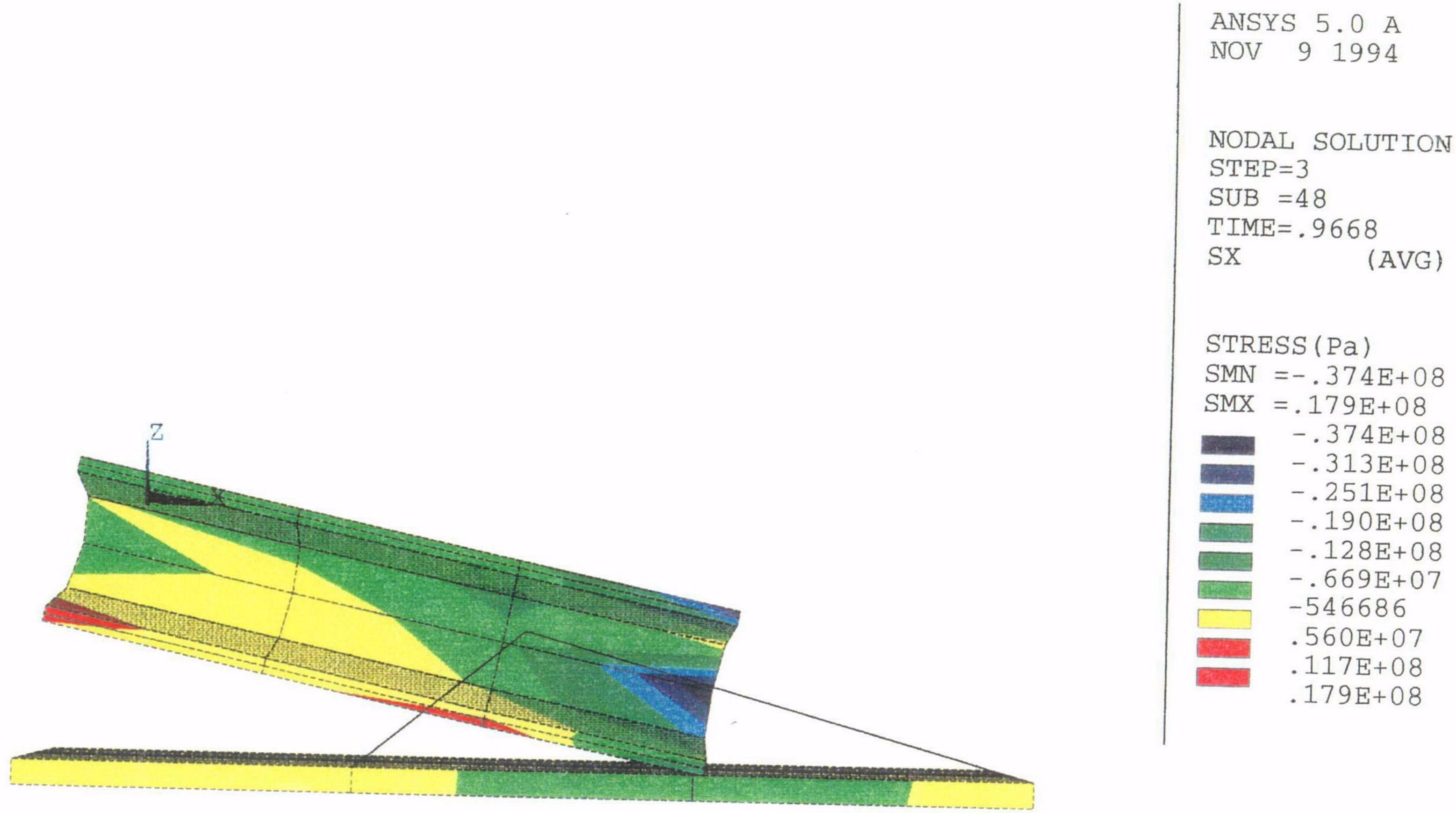


Figure 6.5-53. Full Model Stress Plot Example

C06

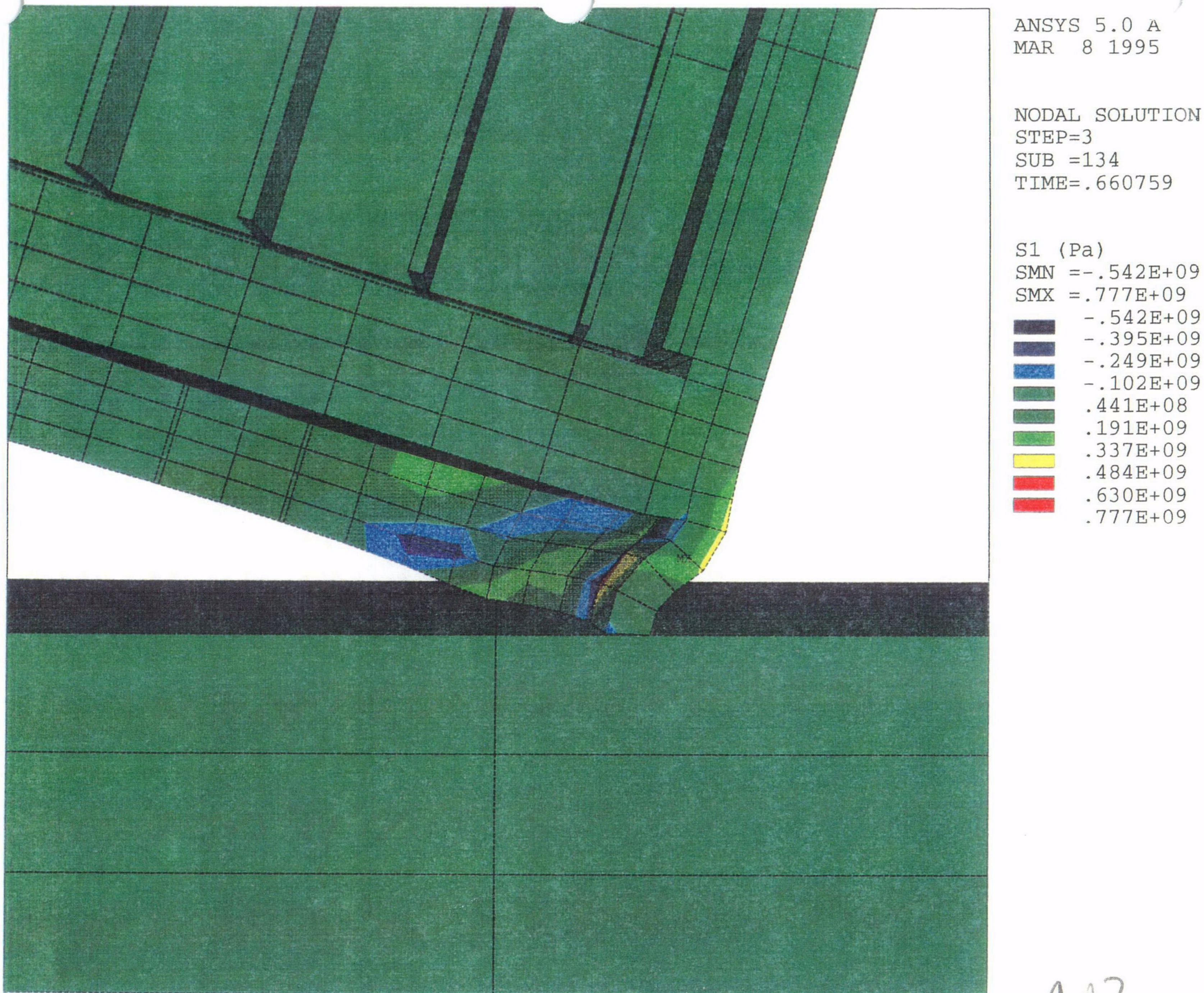


Figure 6.5-54. 21 PWR Multi-Purpose Canister Waste Package 2-m Drop Stress Contour

C07

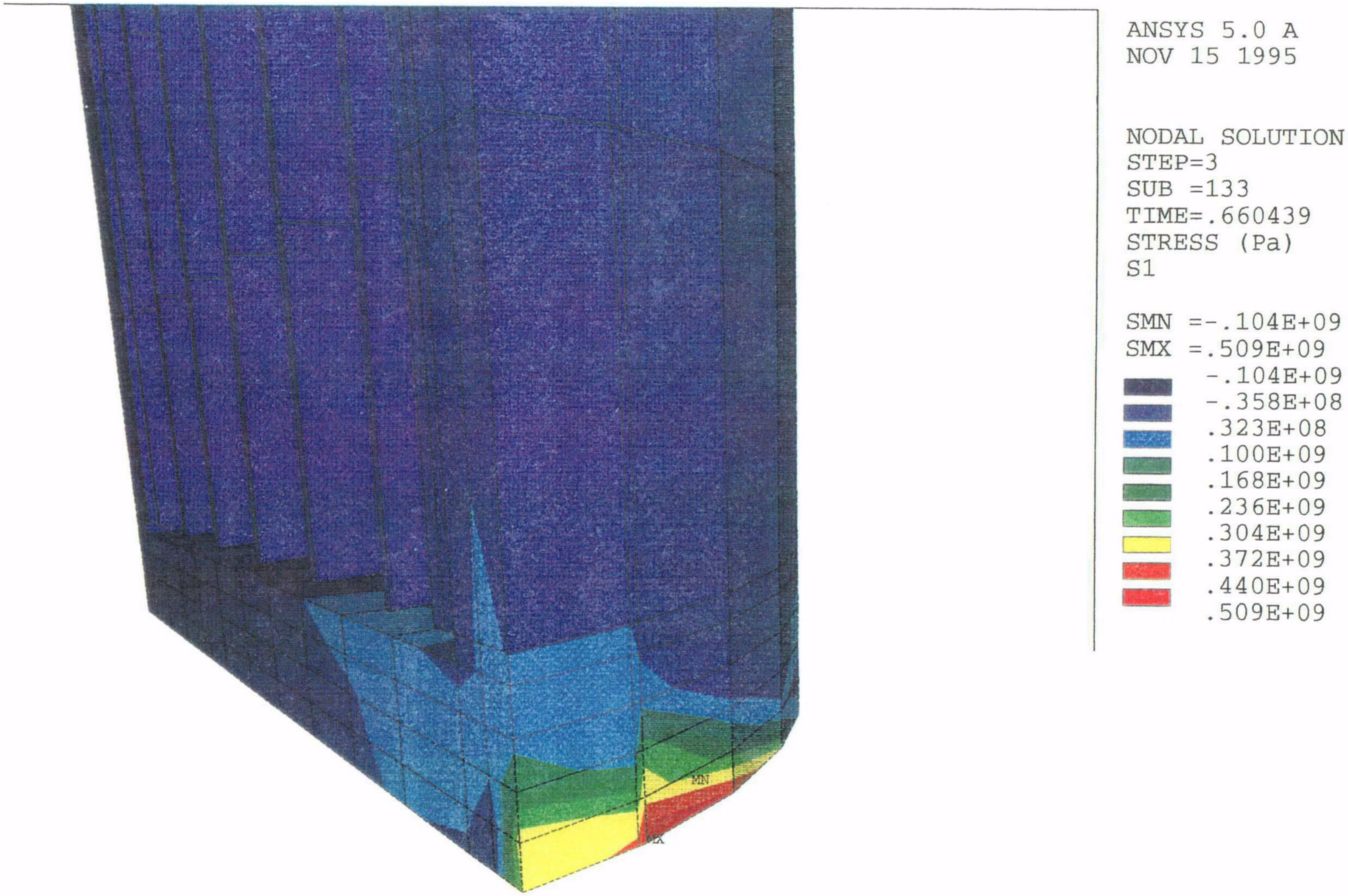


Figure 6.5-55. 21 PWR Multi-Purpose Canister Waste Package 2-m Drop Stress Contour (Without Skirt)

C08

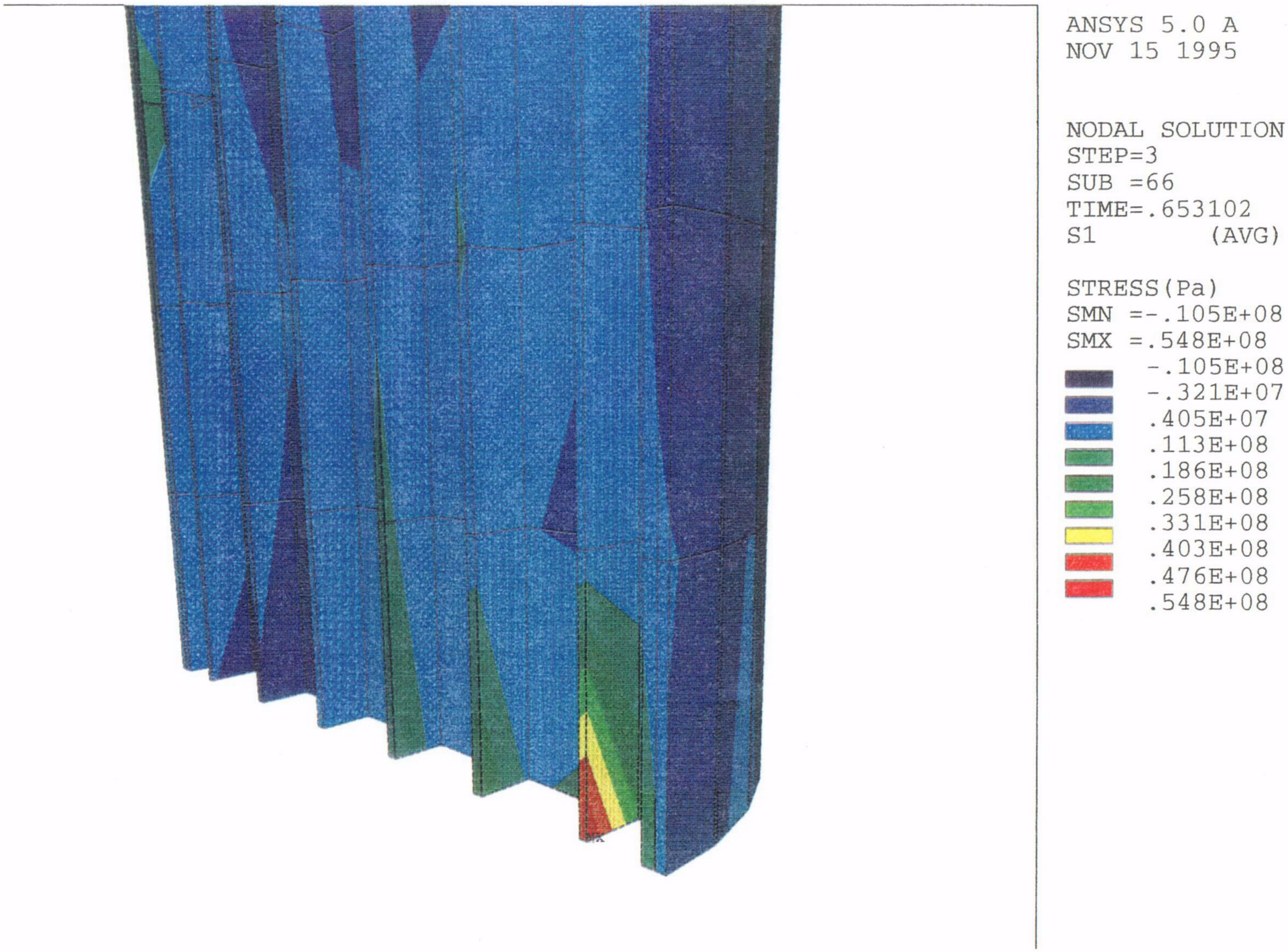


Figure 6.5-56. 21 PWR Multi-Purpose Canister Waste Package 2-m Drop Stress Contour (Without Outer Barrier)

C09

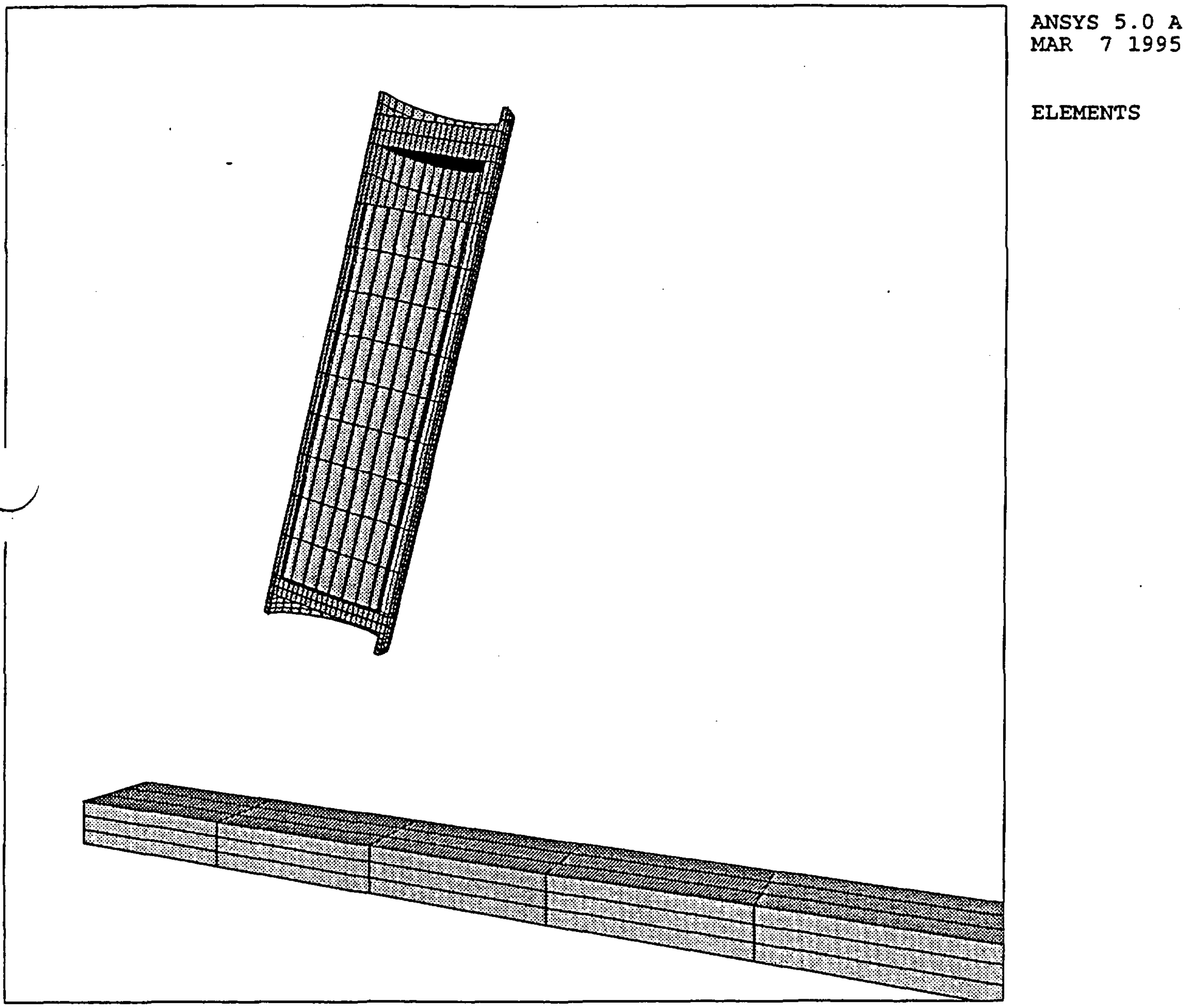
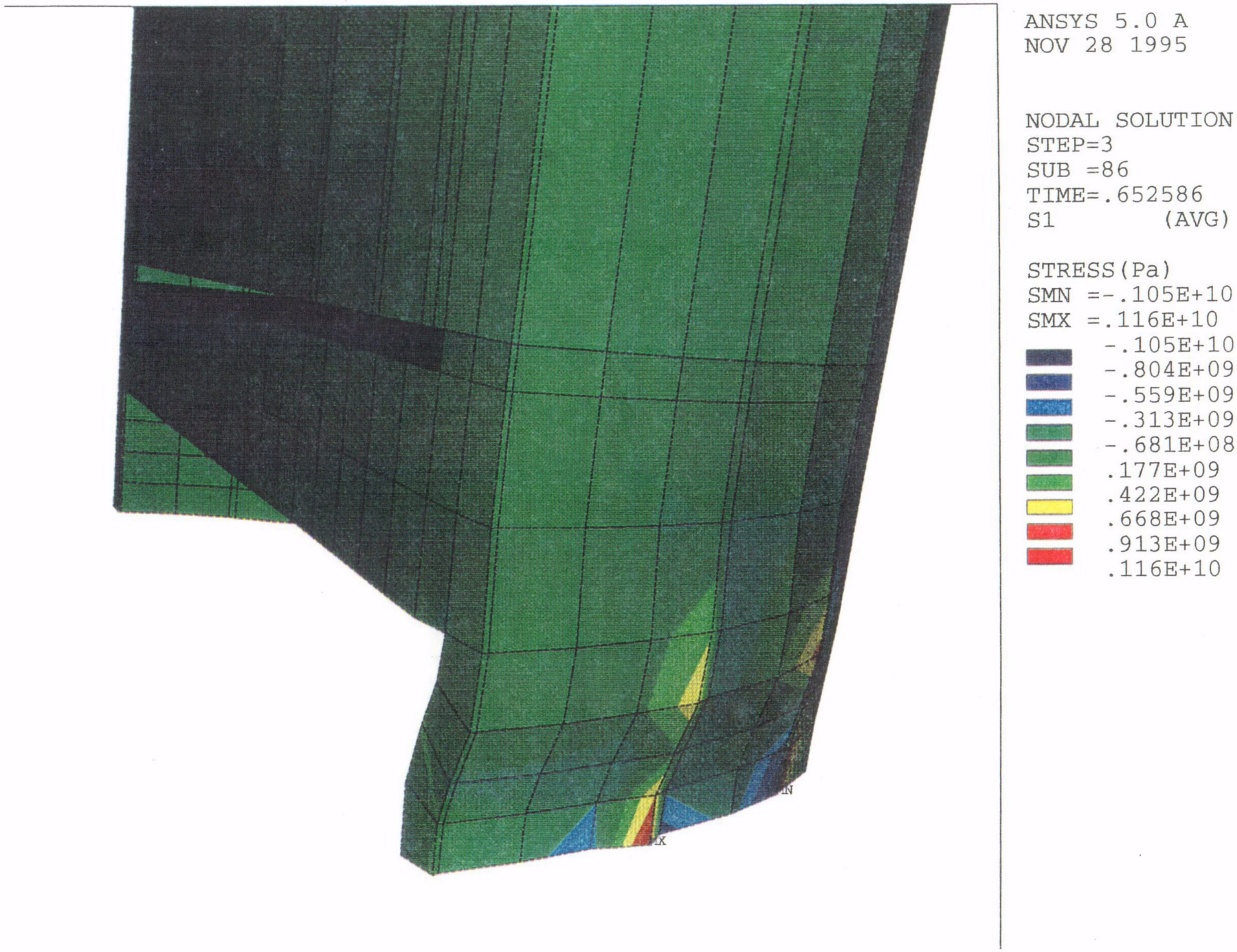


Figure 6.5-57. 24 BWR Multi-Purpose Canister Waste Package 2-m Drop Model



C 10

Figure 6.5-58. 24 BWR Multi-Purpose Canister Waste Package 2-m Drop Stress Contour

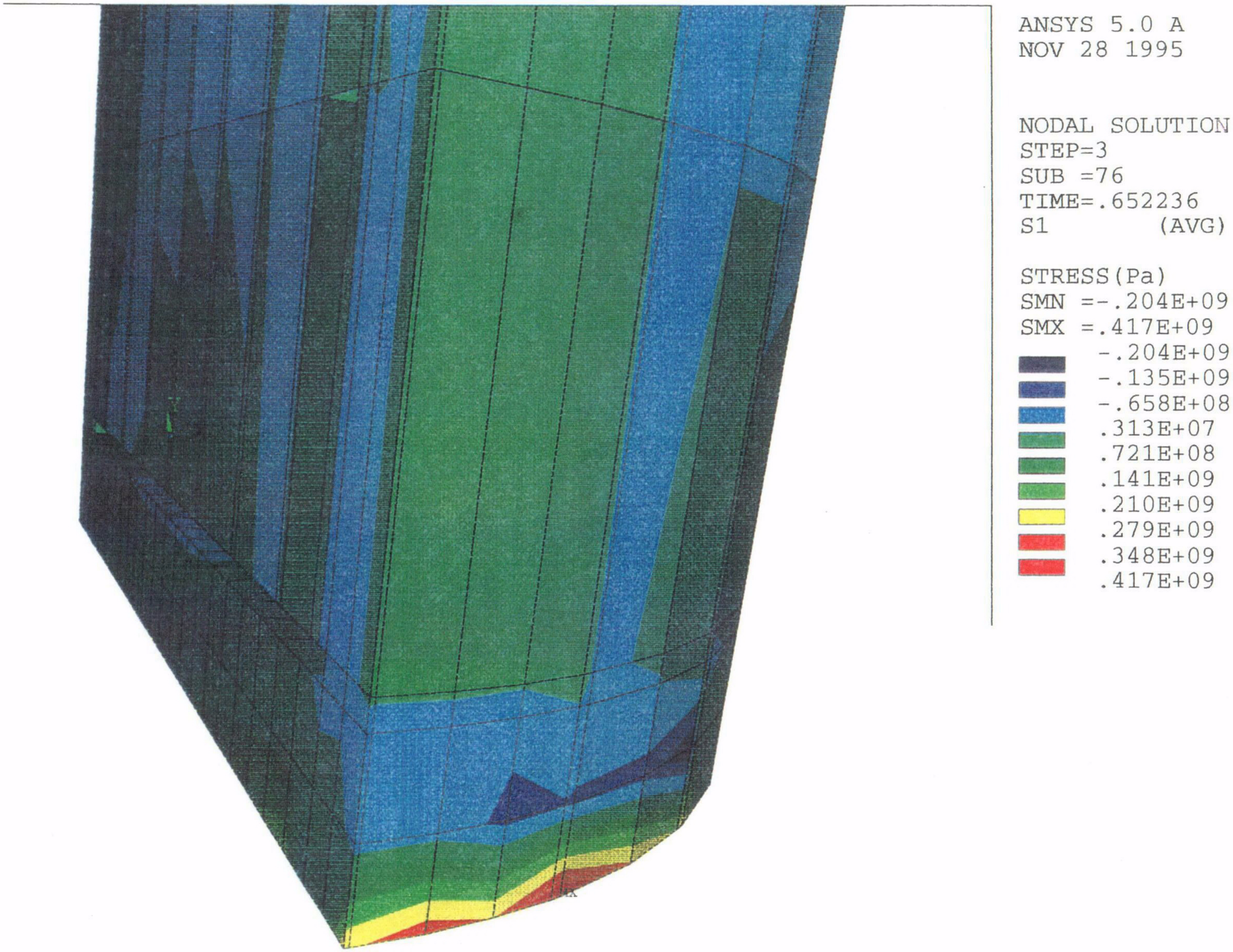
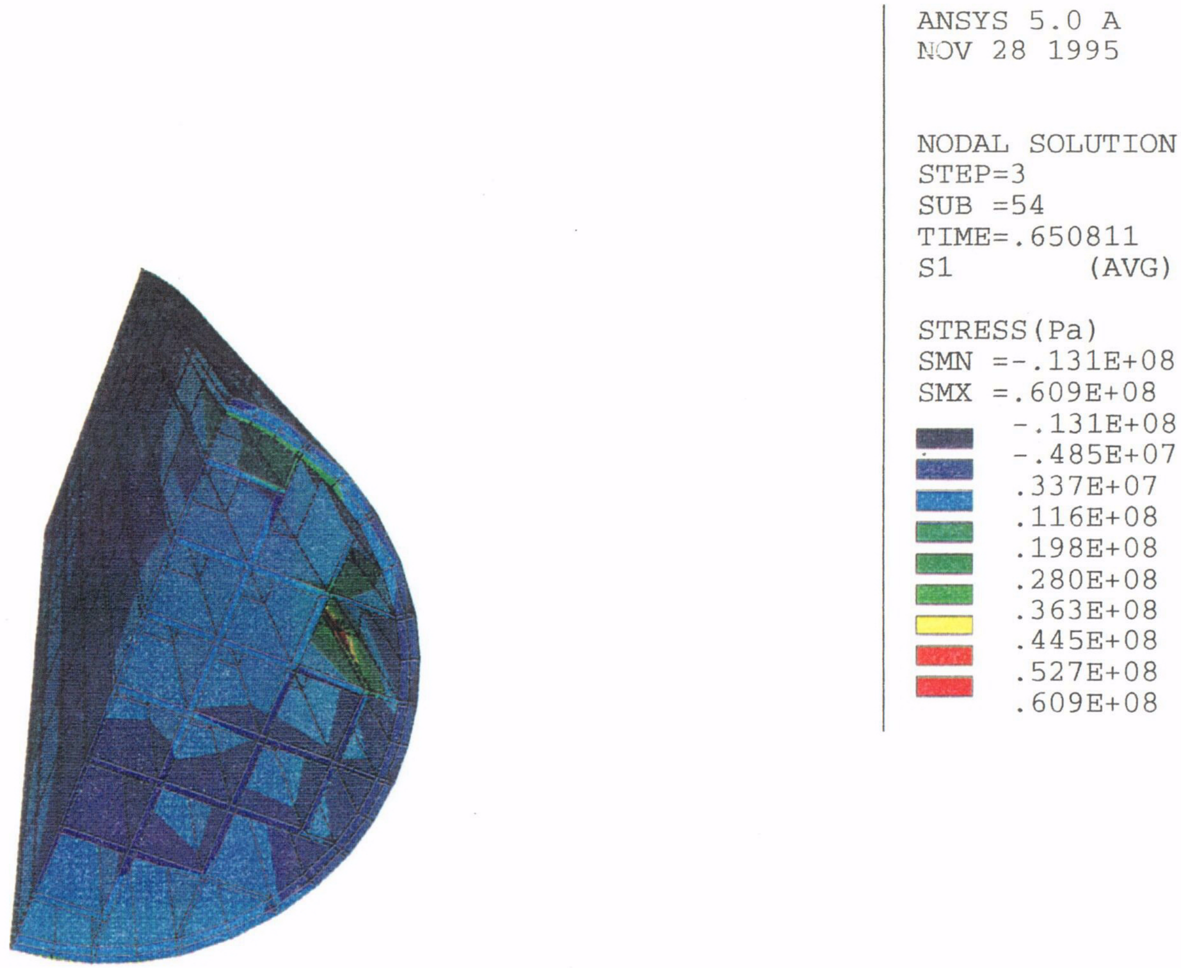


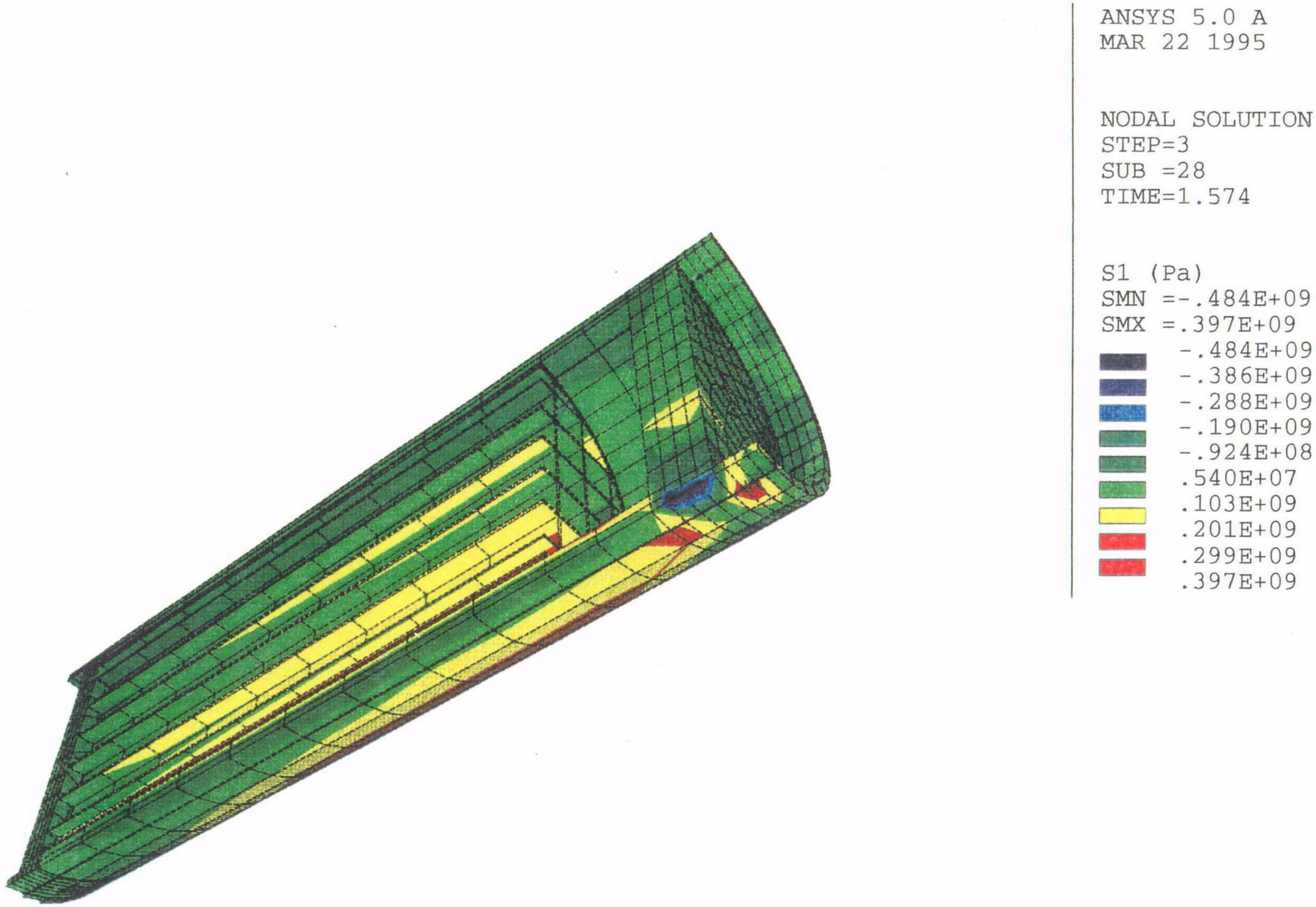
Figure 6.5-59. 24 BWR Multi-Purpose Canister Waste Package 2-m Drop Stress Contour (Without Skirt)

C11



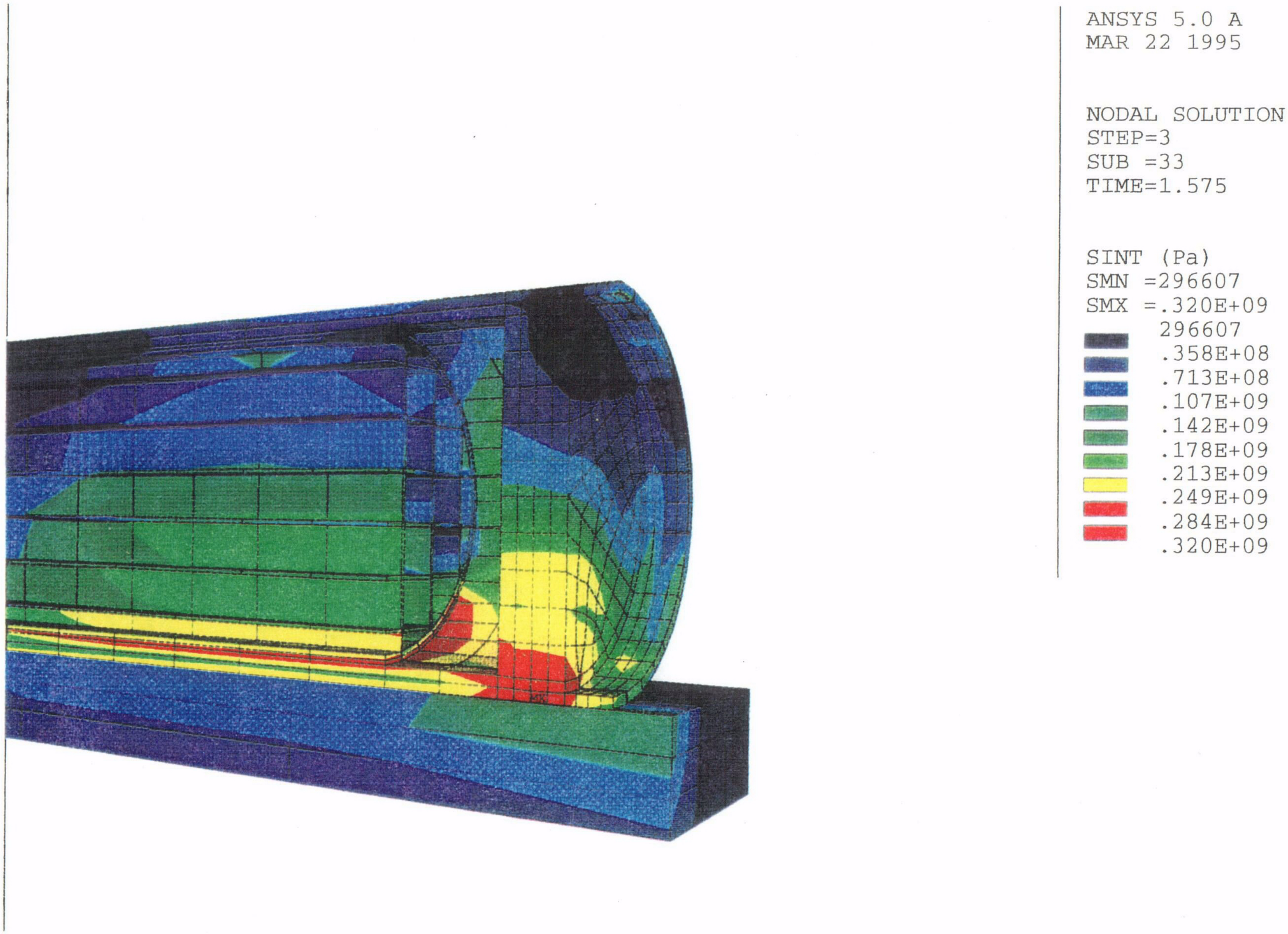
C12

Figure 6.5-60. 24 BWR Multi-Purpose Canister Waste Package 2-m Drop Stress Contour (Without Outer Barrier)



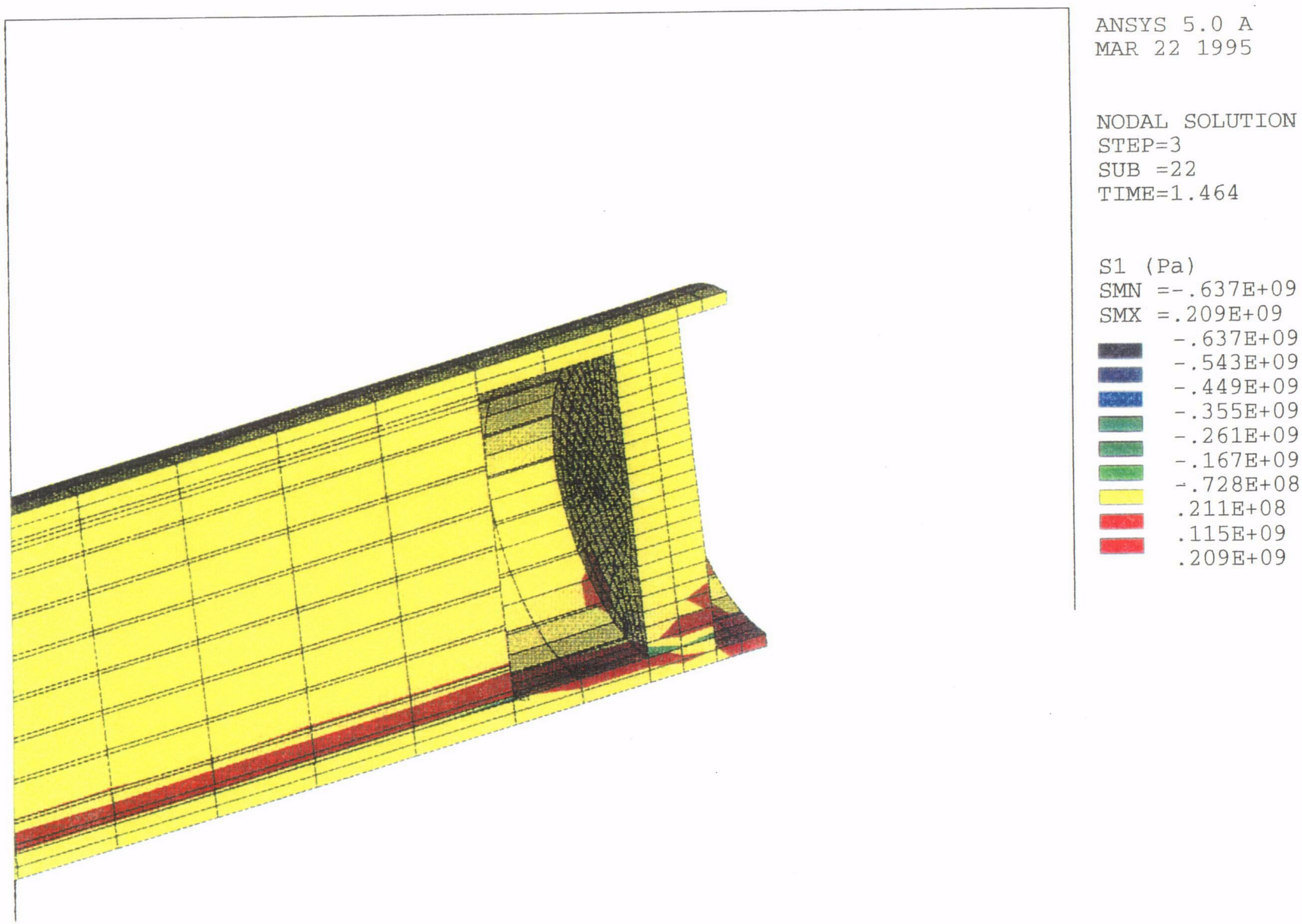
C13

Figure 6.5-62. 21 PWR Multi-Purpose Canister Waste Package Slap Down Stress Contour



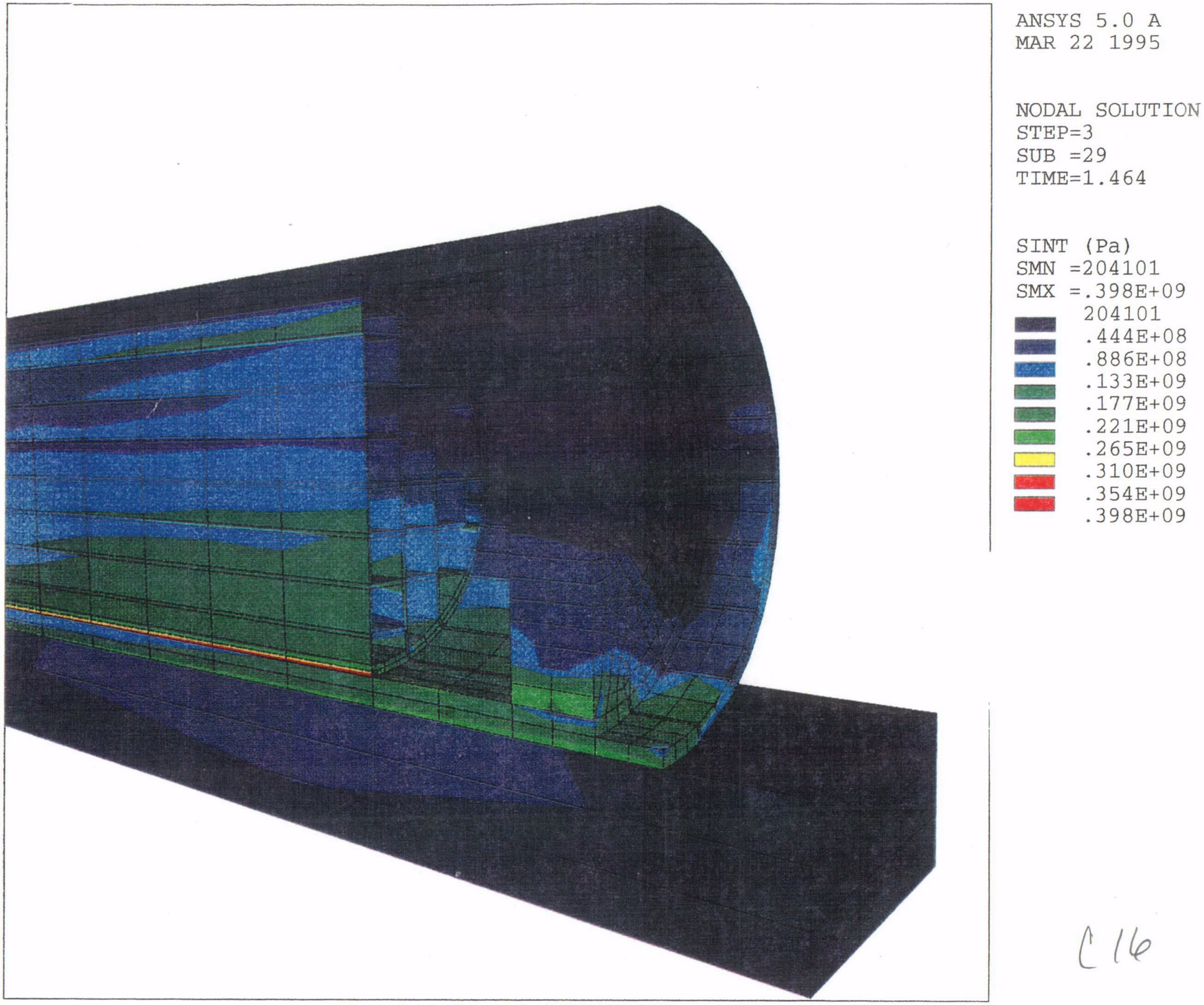
C 14

Figure 6.5-63. 21 PWR Multi-Purpose Canister Waste Package Slap Down Stress Intensity Contour



C15

Figure 6.5-64. 24 BWR Multi-Purpose Canister Waste Package Slap Down Stress Contour


ANSYS 5.0 A
MAR 22 1995
NODAL SOLUTION
STEP=3
SUB =29
TIME=1.464
SINT (Pa)
SMN =204101
SMX =.398E+09
204101
.444E+08
.886E+08
.133E+09
.177E+09
.221E+09
.265E+09
.310E+09
.354E+09
.398E+09

C16

6.5.2.2 Waste Package Slap Down Accident Analysis

A corner drop can be separated into two individual drop events: initial impact and slap down. For impact angles near horizontal, the slap down phase dominates and the assembly response resembles the condition of a side drop (SNL 1992a, pp. 53-64). Initial impact is analyzed in Section 6.5.2.1. The hypothetical slap down accident condition analyzed here is that of a WP striking a flat, essentially unyielding horizontal surface. The models described and the analysis results given in Section 6.5.2.2 are taken from *Scoping Evaluation to Explore - Slap Down Finite-Element Analysis of Multi-Purpose Canister Waste Packages* (CRWMS M&O 1995ak).

6.5.2.2.1 21 PWR Multi-Purpose Canister

6.5.2.2.1.1 Description of Finite-Element Model for Slap Down

A one-half symmetric, three-dimensional finite-element model of a 21 PWR MPC WP has been previously developed for the WP 2-m drop analysis. The slap down event is simulated by making use of the same model with a few modifications. The previously mentioned assumptions and discussions on the 21 PWR finite-element model are the same for the slap down event, see Section 6.5.2.1.1.1.

The initial WP orientation is with its axis at an angle of 30° with respect to vertical. In theory, the maximum impact velocity is obtained if the WP is released from the angle at which slap down is impending, 17.41°. However, to obtain a finite-element solution, a larger initial angle is required to begin the process. Therefore, an angle of 30° was chosen because it satisfied both computational and theoretical requirements. Potential energy calculations indicate that the energy lost due to the use of 30° instead of 17.41° is less than 3.5 percent, leading to a difference in the stresses of less than 1.8 percent. Because this is likely to be smaller than the accuracy of the solution, this difference is neglected. See Figure 6.5-61 for the initial model geometry.

The WP is initially in contact with the impact surface. Constraints placed on the WP at the point it contacts the impact surface cause the WP to experience a fixed axis rotation. The rotation is initiated by the moment about the contact point caused by the weight of the WP. The subsequent impact begins with a line contact between the WP and the flat surface.

To define a contact surface between the WP and the flat surface, contact elements have been defined along the two surfaces where the impact is expected to occur. The same contact stiffness value has been applied to these contact elements as in the case of the WP 2-m drop finite-element model.

6.5.2.2.1.2 Barrier Response

A fixed axis rotation of the WP is followed by an impact on the flat surface. Since the WP is initially at rest, having an angle of 30° with respect to vertical, the potential energy is converted into kinetic energy before the impact, and the kinetic energy is transformed into strain energy by elastic and plastic deformations on the WP after the impact. Deformation on the flat surface is negligible when compared to the WP response at the impact region because the stiffness value assigned to the flat surface is larger than the WP stiffness.

A linearly varying deceleration distribution between the pivot and the opposite end of the WP leads to a loading distribution of similar characteristics. Thus, the dynamic load, deformation, and stress magnitudes on the WP vary from one end to the other.

Figure 6.5-62 illustrates the first principal stress distribution on the WP. As can be seen in the figure, the maximum first principal stress magnitude is 397 MPa and is located in the skirt at the end of the WP away from the initial contact point. The maximum tensile stress is less than the ultimate tensile strength of the outer barrier material, so it can be concluded that the slap down accident will not result in a breach of the outer barrier.

6.5.2.2.1.3 Spent Fuel Assembly Basket Response

The stresses in the MPC basket for the slap down accident are read from the results of the ANSYS analysis. The maximum stress intensity for membrane plus bending stress is above the yield strength of the material, 172 MPa, but less than the ultimate tensile strength, 482 MPa, in some localized areas away from the pivot point of the canister, see Figure 6.5-63. This indicates that some localized plastic deformation may take place in the walls of the basket. For transportation, plastic deformation in the MPC basket is not acceptable (CRWMS M&O 1994j). However, for disposal, the internal structure requirements indicate only that the internal structure must provide separation of the waste forms to prevent criticality, provide mechanical stability of the waste form, and withstand handling, emplacement, and retrieval loads. To meet these requirements, it must be shown that there is no rupturing of the material, i.e., the ultimate tensile strength is not exceeded. Therefore, for disposal, the basket performance is acceptable (YMP 1994b).

There are several factors that may contribute to the high membrane plus bending stress readings in the basket. The basket has sharp, 90° corners. Small fillets in the corners would reduce the high stress concentration. Also, the coarseness of the basket in the model limits the ability of ANSYS to differentiate between peak stress and membrane plus bending stress. This may cause the reading of the membrane plus bending stress to be higher than the actual stress.

The MPC modeled in this analysis is the conceptual design. When the final design is complete, a more detailed model of the MPC basket will be required to better determine the basket response.

6.5.2.2.2 40 BWR Multi-Purpose Canister

6.5.2.2.2.1 Description of Finite-Element Model for Slap Down

Because the mass of the 40 BWR is slightly less than that of the 21 PWR, whereas the area moment of inertia is higher than that of the 21 PWR configuration, a separate model for the 40 BWR is not constructed.

6.5.2.2.2.2 Barrier Response

Since the mass is slightly less and the area moment of inertia is higher for the 40 BWR than for the 21 PWR configuration, deformation patterns and stress distributions obtained for the 21 PWR are considered to be the limiting results. Worst case containment barrier stresses are presented in Section 6.5.2.2.1.2.

6.5.2.2.2.3 Spent Fuel Assembly Basket Response

The size, mass, and area moment of inertia are all very close between the 21 PWR and the 40 BWR. Therefore, the loads on the tubes of the basket assembly during slap down will be nearly the same. However, the 40 BWR MPC basket assembly has a larger number of tubes that have a smaller side length and the same wall thickness. Therefore, the basket of the 40 BWR will be able to take more load than the basket of the 21 PWR. For this reason, the basket stresses of the 21 PWR are considered limiting for the 125-ton WPs, see Section 6.5.2.2.1.3.

6.5.2.2.3 24 BWR Multi-Purpose Canister

6.5.2.2.3.1 Description of Finite-Element Model for Slap Down

A one-half symmetric, three-dimensional finite-element model of the 75-ton, 24 BWR MPC WP has been previously developed for the WP 2-m drop analysis. The slap down event is simulated by making use of the same model with a few modifications. The previously mentioned assumptions and discussions on the 24 BWR finite-element model are the same for the slap down event, see Section 6.5.2.1.3.1.

The initial WP orientation is with its axis at an angle of 30° with respect to vertical. In theory, the maximum impact velocity is obtained if the WP is released from the angle at which slap down is impending, 15°. However, to obtain a finite-element solution, a larger initial angle is required to begin the process. Therefore, an angle of 30° was chosen because it satisfied both computational and theoretical requirements. Potential energy calculations indicate that the energy lost due to the use of 30° instead of 15° is less than 4.5 percent, leading to a difference in the stresses of less than 2.3 percent. Because this is likely to be smaller than the accuracy of the solution, this difference is neglected. The initial orientation of the WP in the model is similar to that shown in Figure 6.5-61 for the 21 PWR.

The WP is initially in contact with the impact surface. Constraints placed on the WP at the point it contacts the impact surface cause the WP to experience a fixed axis rotation. The rotation is initiated by the moment about the contact point caused by the weight of the WP. The subsequent impact begins with a line contact between the WP and the flat surface.

The finite-element models of the small (75-ton) and large (125-ton) MPC configurations are similar in terms of the simulation process. However, the size of the WPs and basket internal structures are different, which plays an important role in the structural response of the WP to the slap down accident condition.

6.5.2.2.3.2 Barrier Response

A fixed axis rotation of the WP is followed by an impact on the flat surface. The barrier response mechanism is similar to the barrier response explained previously in the analysis of the 21 PWR configuration.

Figure 6.5-64 illustrates the first principal stress distribution on the WP. As can be seen from the same figure, the maximum stress magnitude in the outer barrier is 209 MPa and is located in the skirt on the inner surface of the outer barrier above the top of the basket and below the lid. The maximum tensile stress is less than the ultimate tensile strength of the outer barrier material (515 MPa), so it can be concluded that the slap down accident will not result in a breach of the outer barrier.

6.5.2.2.3.3 Spent Fuel Assembly Basket Response

The stresses in the MPC basket for the slap down accident are read from the results of the ANSYS analysis. The maximum stress intensity for membrane plus bending stress is above the yield strength of the material, 172 MPa, in some localized areas away from the pivot point of the canister, see Figure 6.5-65. This indicates that some localized plastic deformation may take place in the walls of the basket. For transportation, plastic deformation in the MPC basket is not acceptable (CRWMS M&O 1994j). However, for disposal, the internal structure requirements indicate only that the internal structure must provide separation of the waste forms to prevent criticality, provide mechanical stability of the waste form, and withstand handling, emplacement, and retrieval loads. To meet these requirements, it must be shown that there is no rupturing of the material, i.e., the ultimate tensile strength is not exceeded. Therefore, for disposal, the basket performance is acceptable (YMP 1994b).

The same factors that may affect the stress reading in the 21 PWR slap down analysis are present for the 24 BWR slap down analysis. Again, when the final MPC design is complete, a more detailed model of the MPC basket will be required to better determine the basket response.

6.5.2.2.4 12 PWR Multi-Purpose Canister

6.5.2.2.4.1 Description of Finite-Element Model for Slap Down

Because the 12 PWR and 24 BWR are nearly the same in terms of size, mass, and area moment of inertia, a separate model is not developed for the 12 PWR.

6.5.2.2.4.2 Barrier Response

The masses of the 12 PWR and 24 BWR configurations are nearly the same, and the difference in terms of the inertia of the basket assemblies is very small. Therefore, deformation patterns and stress distributions obtained for the 24 BWR are very close to those of the 12 PWR. Maximum stresses in the 12 PWR will be within a few percent of the maximum stresses in the 24 BWR. There is sufficient margin for stresses in the 24 BWR WP, thus, given the similarity of the designs, there will still be sufficient margin for stresses for the 12 PWR WP.

6.5.2.2.4.3 Spent Fuel Assembly Basket Response

Because of the similarities between the 12 PWR and the 24 BWR, the loads on the tubes of the basket assembly during slap down will be nearly the same. The 12 PWR is of the flux trap design, which includes additional members between the tubes. These additional members take up some of the fuel load during impact and also distribute the load more evenly between the basket tubes. This effect more than makes up for the additional strength provided in the 24 BWR by the larger number of tubes with smaller side lengths. Thus, the stresses in the basket of the 24 BWR are considered limiting for the small (75-ton) WPs. Worst case basket assembly stresses are presented in Section 6.5.2.2.3.3.

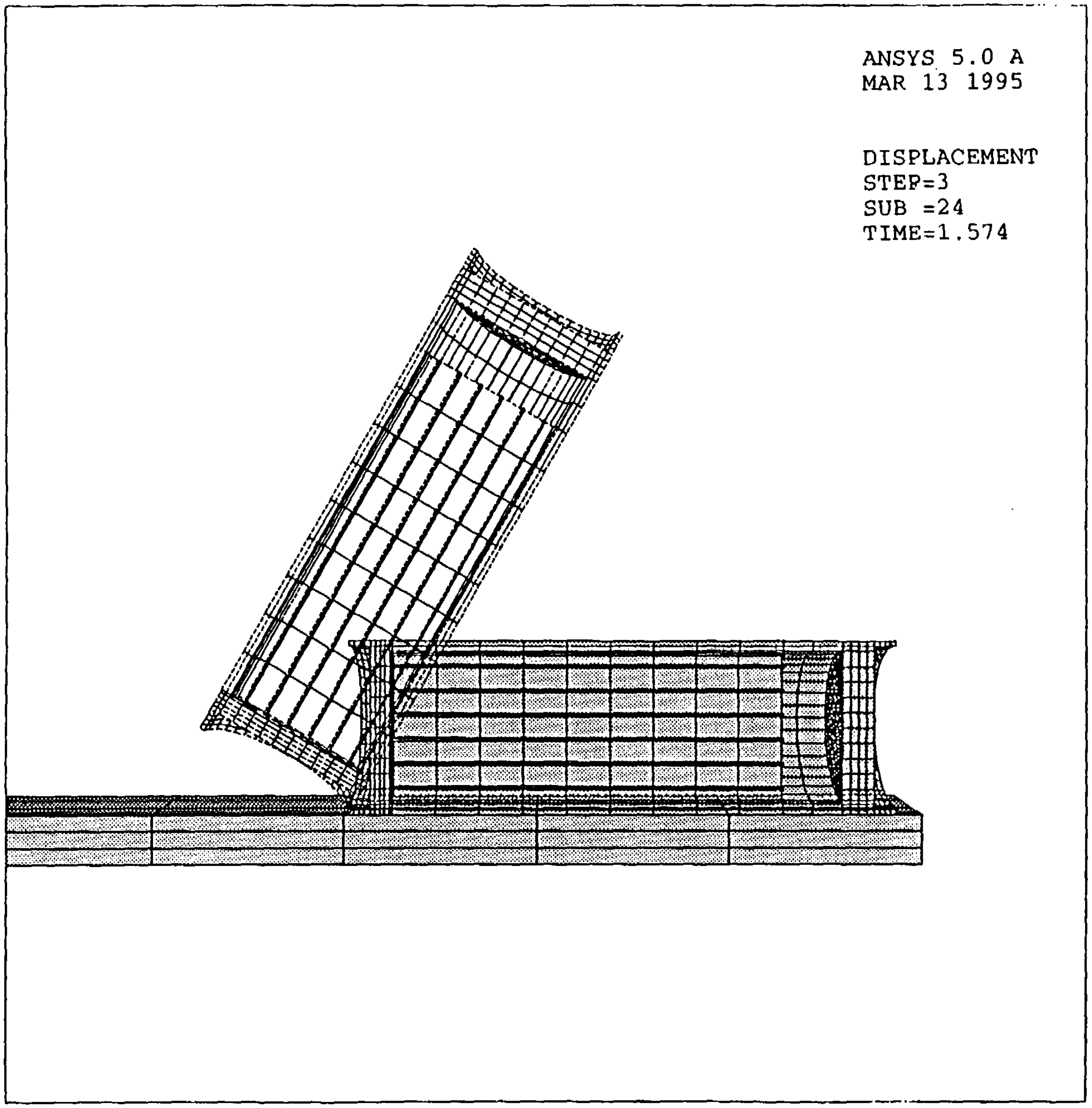
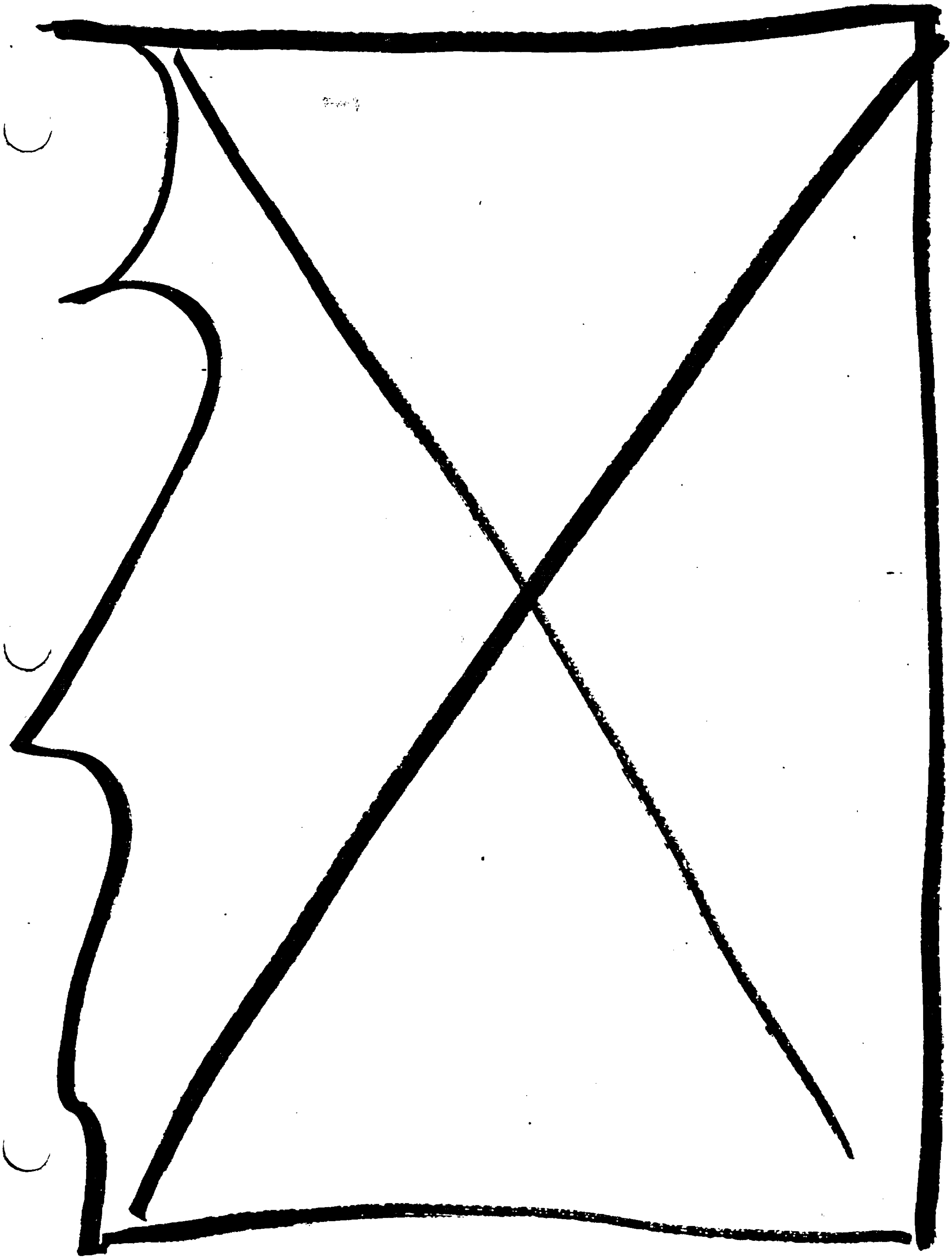
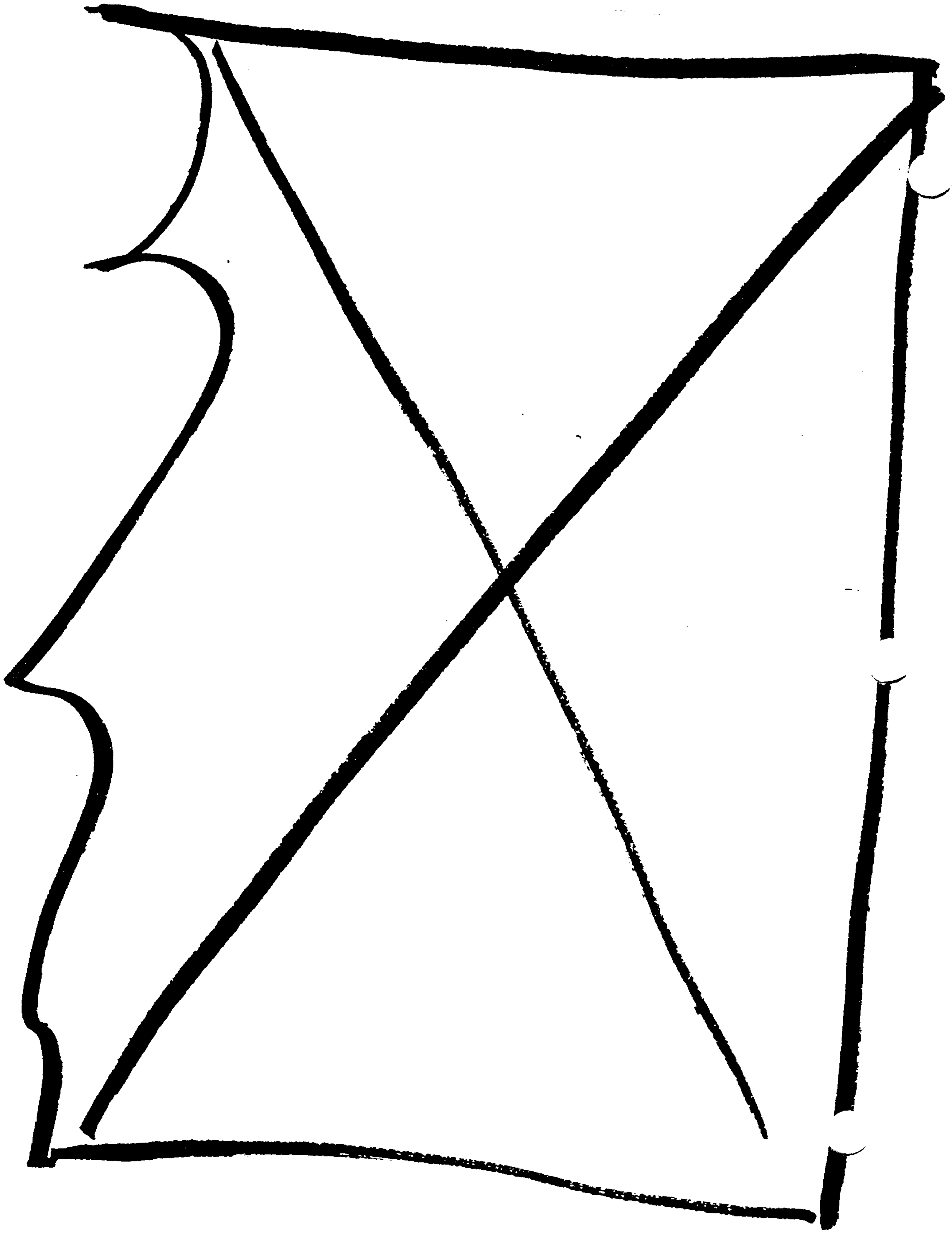


Figure 6.5-61. 21 PWR Multi-Purpose Canister Waste Package Slap Down Model





6.5.2.3 Analysis of Rock Fall onto Waste Package

The large multibarrier WP is designed to be drift emplaced and thus may be subjected to a tunnel collapse or an individual rock fall. The rock load could be the result of rock expansion and rock fall, or by rock instability caused by a phase transformation in the rock.

A rock fall could potentially occur in the starter tunnel or main drift while the WP is in transit, or in the emplacement drift after the WP has been emplaced. The main difference between these cases is the height from which the rock falls. The starter tunnel height is greater than either the emplacement drift or main drift height, so the starter tunnel rock fall is considered to be limiting.

The following conservative assumptions are made for the rock fall analysis in the starter tunnel:

- The WP is assumed to be sitting on the invert of the starter tunnel. It will actually be on a transporter, raising it some distance above the invert, thus reducing the fall height.

- The WP is assumed to be completely unprotected. It will actually be inside a shielded transporter, which may provide some impact protection.

The probability of a rock fall onto the WP in the starter tunnel is low for the following reasons:

- The WP is not stored in the starter tunnel, but rather, just passes through. Therefore, the WP spends very little time there, reducing the probability of an accident.

- Rock bolts and shotcrete are used to support the roof of the starter tunnel, decreasing the probability of a rock fall.

The models described in Section 6.5.2.3 are taken from *Scoping Evaluation to Explore - Rock Fall Finite-Element Analysis and Analytical Evaluations on Interlocking Basket Waste Packages* (CRWMS M&O 1995al), and the analysis results are taken from correlations made in *Scoping Evaluation to Explore - Rock Fall Accident Condition Analysis on Multi-Purpose Canister Waste Packages Correlated from Interlocking Basket Waste Package Design Analysis* (CRWMS M&O 1995am).

6.5.2.3.1 Correlation Method for Small (75-Ton) and Large (125-Ton) Multi-Purpose Canister Configurations

The results of a nonlinear dynamic simulation of a large rock impacting a metallic multibarrier WP have been obtained in a previous analysis. Basic assumptions and properties of this analysis will be discussed briefly to provide a basis for correlation calculations.

A three-dimensional finite-element model of the 12 PWR UCF Interlocking Basket WP (including inner barrier, outer barrier, and basket assembly) has been developed in accordance with early WP conceptual designs. Throughout the rest of Section 6.5.2.3, this WP will be referred to as the 12 UCF. In the model, it is assumed that all of the kinetic energy of the falling rock is imparted onto the WP as mechanical energy (i.e., the rock does not shatter, no heat is produced) and the deflection of the rock is negligible compared to the deflection of the WP. Another important assumption is the modeling of the impacting rock as a sphere. This geometry was selected because the impact of a sphere will result in a global distribution of stress onto the WP, whereas a sharp wedge geometry would deform the pointed region of the rock as a result of the high stress concentration at the impact point. The rock deformation would result in the sharp wedge geometry being less damaging to the WP than would the spherical geometry. It should be noted that ANSYS cannot model a perfect sphere because element surfaces are flat. In the modeling of the rock, there are small points at node locations, but their effects are minor.

The simulation of the rock impact on the WP includes both elastic and plastic deformations. Material behavior is approximated by incorporating bilinear stress-strain curves into the ANSYS WP material properties. A transient dynamic analysis solution is produced with gravitational acceleration as the only load on the system. Displacement constraints at support locations prevent vertical motion and horizontal motion normal to the axis of the WP. The support locations are at the extreme ends of the outer barrier, which in this model does not include the skirt, giving the longest unsupported length of the canister.

The maximum-normal-stress theory is based on a comparison of the ultimate tensile and compressive strengths of the materials with the maximum values of tensile and compressive stresses they experience during an impact. Having evaluated the results of the finite-element analysis of the 12 UCF in accordance with the maximum-normal-stress theory, it is determined that the WP is able to withstand the maximum dynamic loading from the fall of a 19,100 kg rock through 8.4 m without breaching the containment barriers. This fall height corresponds to a rock falling from the roof of the starter tunnel, the maximum height from which a rock can fall onto the WP.

To develop a correlation factor between the results of UCF and MPC concepts, an analytical solution is needed for the same problem. A quasi-static approach can be used to simplify the dynamic load into a static, concentrated force applied at midspan on the outer and inner barriers. Figure 6.5-66 illustrates the WP, applied force, shear force, and bending moment diagrams.

The potential energy of the rock is converted into kinetic energy by the time it falls to the top surface of the outer barrier. Assuming that total energy is spent for deflection of the inner and outer barriers, which is a conservative approach, strain energy stored in the barrier will be equal to the kinetic energy of the rock. Derivation of the strain energy relation in terms of applied force and stiffness of the material is presented in *Finite-Element Analysis of Rock Fall on Uncanistered Fuel Waste Package Designs* (CRWMS M&O 1995au).

Kinetic Energy = Strain Energy

$$\frac{1}{2}mv^2 = \frac{P^2L^3}{96EI} \qquad \text{(Equation 6.5-2)}$$

where: m = mass of rock
v = velocity of rock
P = force on barriers
L = length of basket assembly
E = modulus of elasticity of barrier materials
I = area moment of inertia of barriers

It is also known that

$\sigma = M \times c/I$, and $M = P \times L/4$ for this problem. "σ" is the bending stress, "c" is the outer barrier outer radius, and "M" is the bending moment. Placing these relations into the Equation 6.5-2 and considering L, c, I, and E as constant values, the following is obtained:

$$\frac{1}{2}mV^2 = \left(\frac{4\sigma I}{cL}\right)^2 \frac{L^3}{96\ EI} \qquad \text{(Equation 6.5-3)}$$

The magnitude of the force calculated from these formulations has been used to obtain the maximum value of the bending moment and the bending stress on the surface of the barrier. When the maximum stress magnitudes of analytical and finite-element solutions are compared, a difference is expected due to the advantage of using an elastic-plastic approach in the finite-element analysis over the pure elastic analytical solution.

The relations between the 12 UCF and the MPC WPs are made by holding the maximum stress constant between the 12 UCF and the MPC WPs and solving for the rock size required to cause that stress in the MPC WPs.

Input parameters of this analytical model are the modulus of elasticity, length of the WP, impact velocity and mass of the rock, and area moment of inertia of the WP. The modulus of elasticity, length, and velocity are constants for each MPC configuration in the analytical expressions. However, the value of mass will be varied so as to produce the previously mentioned bending stress magnitude with an inertia input value calculated for each MPC.

The area moment of inertia of each MPC configuration has been calculated for two orientations of the basket to determine the minimum value of inertia that would result in the largest magnitude of stress on the WP. The two orientations are with the basket sides vertical and horizontal (non-rotated) and with the basket rotated 45°. The resultant inertia values for each MPC configuration will be presented in the following sections.

6.5.2.3.2 21 PWR Multi-Purpose Canister Waste Package Barrier Response to Rock Fall Accident

The analytical model assumes that the normal stress varies linearly with the distance from the neutral surface. A compressive normal stress state is active above the neutral axis, and a tensile normal stress state is active below the neutral axis. Furthermore, the magnitudes of maximum compressive and tensile stresses are identical due to the symmetric geometry of the WP cross-sectional area with respect to the plane passing through the center of the WP.

The finite-element evaluations are based on a slightly different mechanism of impact and proceeding stress distribution on the finite-element model. Plastic deformation in a material leads to a nonlinear stress profile across the material thickness. The impacted region of the WP experiences localized plastic deformation due to high stress concentration. Since the analytical formulations will be used only for correlating the results of the finite element analysis performed on the 12 UCF with MPC configurations, the accuracy of the finite-element method will be reflected on the MPC configuration results.

Moment of inertia calculations of the 45° rotated 21 PWR configuration result in a value of 0.2928 m^4. The modulus of elasticity, WP length, and the impact velocity of the rock are substituted into the energy equation together with this moment of inertia value. The rock fall height is conservatively assumed to be the same for the 21 PWR as it was for the 12 UCF, 8.4 m. The mass of the falling rock is varied to produce a resultant stress magnitude previously determined as the limit for not causing any breach on the 12 UCF. The critical rock mass is determined as 32,100 kg for the 21 PWR configuration.

6.5.2.3.3 40 BWR Multi-Purpose Canister Waste Package Barrier Response to Rock Fall Accident

The total area moment of inertia of the 40 BWR is calculated as 0.3026 m^4. The rock fall height is conservatively assumed to be the same for the 40 BWR as it was for the 12 UCF, 8.4 m. Having the modulus of elasticity, WP length, and impact velocity of the rock substituted into the conservation of energy equation, the critical mass of the rock is determined as 33,200 kg.

When the results of the 21 PWR are compared to the 40 BWR configuration in terms of inertia and critical rock mass values, it is observed that the maximum allowable rock mass increases as the area moment of inertia of the WP becomes larger. Due to the fact that there is a small difference in the moment of inertia values between the 21 PWR and the 40 BWR WPs, the difference in the calculated critical mass values is also very small.

In conclusion, deformation patterns and stress distributions due to the rock fall accident scenario for the 21 PWR are considered to be the limiting results since the moment of inertia of the 21 PWR is slightly lower than the moment of inertia of the 40 BWR configuration.

6.5.2.3.4 24 BWR Multi-Purpose Canister Waste Package Barrier Response to Rock Fall Accident

The MPC disposal container and MPC shell inner and outer diameters are smaller for the small (75-ton) MPC configurations than for the large (125-ton) MPC configurations. Considerably smaller moment of inertia values are calculated as a result. Analytically derived conservation of energy equations indicate that any change in the moment of inertia value is inversely proportional to the maximum stress magnitude created on the WP due to the rock drop loading condition. Thus, the maximum rock mass values are expected to be smaller for the small MPC WPs than for the large MPC WPs.

The rock fall height is conservatively assumed to be the same for the 24 BWR as it was for the 12 UCF, 8.4 m. The moment of inertia for the 24 BWR is 0.1662 m^4. When incorporated into the energy equation, the critical rock mass is determined as 25,200 kg.

6.5.2.3.5 12 PWR Multi-Purpose Canister Waste Package Barrier Response to Rock Fall Accident

Inertia calculations of the 45° rotated 12 PWR configuration resulted in a value of 0.1634 m^4. The rock fall height is conservatively assumed to be the same for the 12 PWR as it was for the 12 UCF, 8.4 m. The modulus of elasticity, WP length, and impact velocity of the rock have been substituted into the energy equation together with this inertia value. The same analytical procedure is followed by varying the mass of the falling rock to produce a resultant stress magnitude previously determined as the limit for not causing any breach on the 12 UCF. The critical rock mass is determined as 24,800 kg for the 12 PWR.

It is noted that a smaller inertia value of the 12 PWR compared to the 24 BWR results in a smaller critical rock mass for the 12 PWR. This result is consistent with the previous discussion of the 21 PWR and the 40 BWR configurations in Section 6.5.2.3.3.

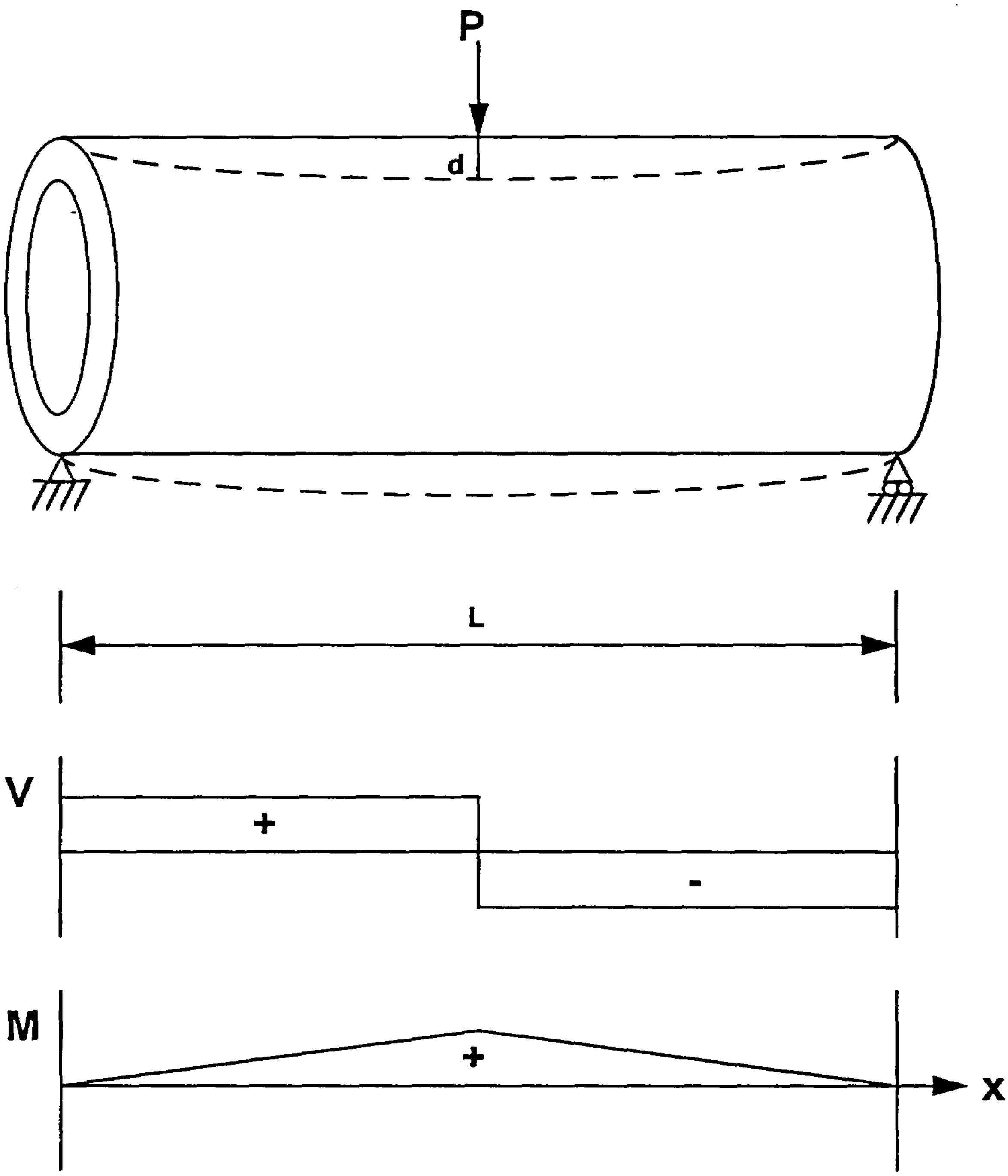


Figure 6.5-66. Waste Package Applied Force, Shear Force, and Bending Moment Diagrams.

6.5.2.4 Conclusions Drawn from Structural Analysis

The WP 2-m drop accident scenario results in a localized material failure around the region of impact in the skirt. However, there is no material failure in any region of the inner or outer barriers other than in the outer barrier skirt. It is also concluded that the maximum principal stress in the basket assembly is below the material yield strength. Therefore, the overall WP response reveals that there will be no loss of radionuclide containment due to a WP drop from a height of 2 m.

The slap down accident scenario does not result in material failure in any part of the outer barrier. The maximum stress intensity for membrane plus bending stress is above the yield strength of the material in some localized areas of the basket assembly. This indicates that some localized plastic deformation may take place in the walls of the basket assembly. For transportation, plastic deformation in the MPC basket is not acceptable (CRWMS M&O 1994k). However, for disposal, the internal structure requirements indicate only that the internal structure must provide separation of the waste forms to prevent criticality, provide mechanical stability of the waste form, and withstand handling, emplacement, and retrieval loads. To meet these requirements, it must be shown that there is no rupturing of the material, i.e., the ultimate tensile strength is not exceeded. Therefore, for disposal, the basket performance is acceptable (YMP 1994b). The MPC modeled in this analysis is the conceptual design. When the final design is complete, a more detailed model of the MPC basket will be required to better determine the basket response to a WP slap down.

A rock fall accident scenario was previously performed for the 12 PWR UCF WP. Using a correlation between the 12 UCF and the MPC WP configurations, the maximum rock mass that can fall from the roof of the starter tunnel without breaching the WP for each MPC WP is calculated. The limiting MPC WP configuration is determined as the 12 PWR with a critical rock mass of 24,800 kg. The maximum rock size which can fall onto the WP is being determined.

6.5.3 Criticality Analysis

The MPC designs analyzed are taken from the *MPC Conceptual Design Report* (CRWMS M&O 1994k). The MPC conceptual designs incorporate boron neutron absorbers in the form of aluminum boron alloy (Al-B) control panels in fuel baskets to provide a criticality control function. The use of the Al-B provides the capability to adjust the neutron absorber concentrations over a wide range to allow for burnup, or it may be decreased for a basket intended to hold low enrichment first core fuel assemblies for storage and transportation.

Two types of fuel baskets are used for the MPC versions to control criticality in the *MPC Conceptual Design Report*: a flux trap design type and a burnup credit design type. The flux trap design relies on control panels that are separated by a gap filled with air (water for accident scenarios). The purpose of the water gap is to thermalize fast neutrons so that the neutron absorber in the control panels will more effectively absorb neutrons than a single solid plate. The burnup credit design type allows a greater fuel payload with a given MPC diameter by removing the flux trap water gap and taking credit for the decrease in SNF reactivity compared to fresh fuel. Credit for fixed neutron absorber material or flux traps is acceptable for storage and transportation, and was assumed to be acceptable in the *MPC Conceptual Design Report* (CRWMS M&O 1994k) for long-term WP criticality control at the MGDS. Issues related to these criticality control methods for long-term disposal are discussed below and are detailed in Section 8, Performance Analysis.

6.5.3.1 Codes, Biases, and Isotopic Inventory Generation

The analysis method employed to ensure criticality control for disposal in the MGDS uses the MCNP computer program, Version 4.2, developed by Los Alamos National Laboratory (ORNL 1992). MCNP is a three-dimensional Monte Carlo particle transport program with a generalized geometry capability that allows the development of much more detailed, accurate models of the systems of interest than combinatorial geometry based programs. MCNP is used to calculate the effective multiplication factors, k_{eff}, for the various designs and configuration of interest. MCNP has been thoroughly benchmarked against criticality experiments and other criticality computer programs. MCNP, version 4.2, has been approved (verified and validated) for QA criticality work according to the CRWMS M&O Quality Administrative Procedures.

An associated continuous energy cross-section set based on ENDF/B-V is used by MCNP. This library provides much more detail than multi-group cross-section sets. The continuous energy cross-section library is not spectrum weighted (biased) and is, therefore, not limited in its applicability.

There are biases and uncertainties, provided in terms of difference in k_{eff}, associated with a criticality calculation. How these biases and uncertainties are treated in criticality calculations is covered in the American National Standard *Criticality Safety Criteria for the Handling, Storage, and Transportation of LWR Fuel Outside Reactors* (ANSI/ANS-8.17). The fresh fuel bias and uncertainty for MCNP stated in terms of the difference in k_{eff} is approximately 0.015 (CRWMS M&O 1994i). The preliminary SNF bias and uncertainty is approximately 0.06 (CRWMS M&O 1994i). The SNF bias and uncertainty is higher because of additional factors such as isotopics and axial effects.

The Characteristics Database (ORNL 1993b) was used to provide fuel assembly information, control rod assembly information, and SNF isotopic composition information. The Charactersitics Database has been qualified for quality assurance work under the M&O program.

6.5.3.2 MPC Modeling Assumptions

The reference PWR fuel assembly selected for conceptual MPC development is the B&W 15x15 fuel type, which has been established as one of the more reactive PWR fuel designs under intact fuel assembly and fixed MPC geometry conditions (CRWMS M&O 1994k). The reference BWR fuel assembly is the GE-5 8x8 fuel type, which has been established as one of the more reactive BWR fuel designs under intact fuel assembly and fixed MPC geometry conditions (CRWMS M&O 1994k).

Credit is taken for the inherent neutron absorption capability of fixed structural components within the MPC internal basket array. No credit is taken for irradiated control rods or burnable absorbers; only approved fixed, full-length supplemental neutron absorbers are assumed present. As discussed in Section 8, credit as a criticality control material is not taken for the Al-B in the MGDS because its corrosion rate is high compared to that of other basket materials. The material is modeled as only aluminum for base cases, but calculations to demonstrate the worth of boron are performed.

Each fuel assembly is treated as a heterogeneous system with the fuel pins, control rod guide tubes, and instrument guide tube modeled explicitly. One-quarter of the MPC and disposal container are modeled laterally and the assemblies and container are modeled from the centerline up (reflected on the centerline) axially. A sketch of the typical modeling detail is shown in Figure 6.5-67.

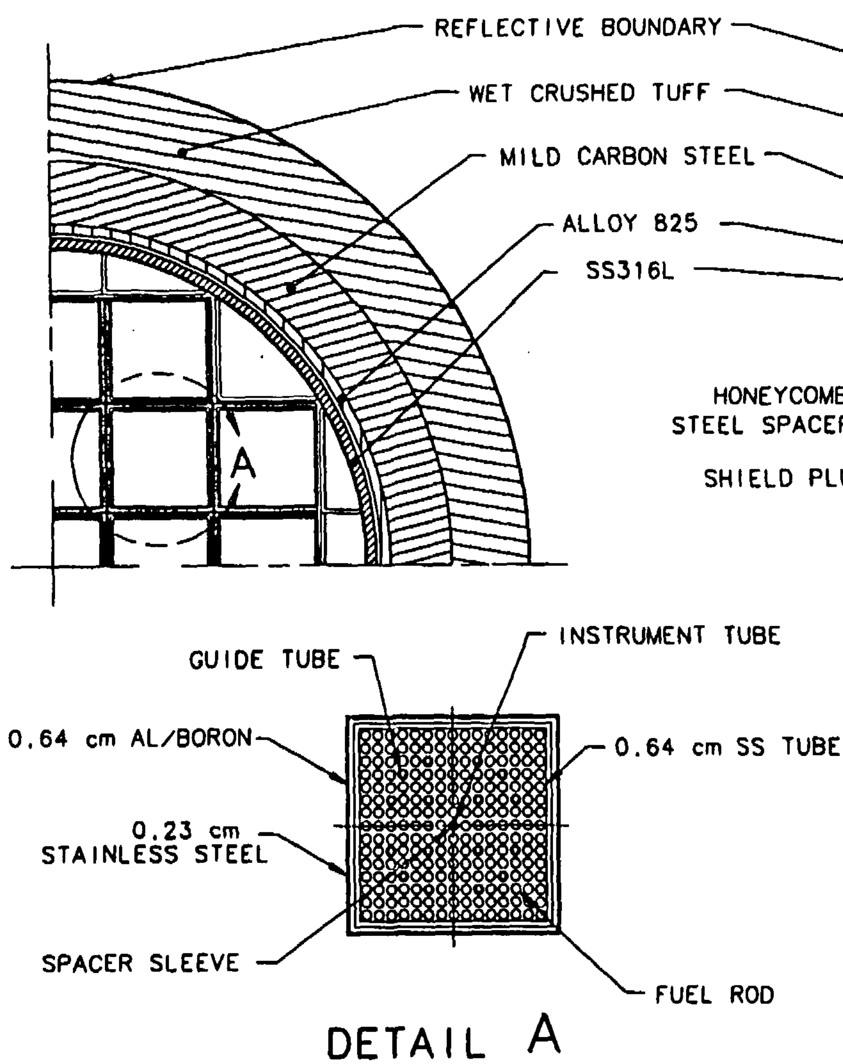
For SNF, the isotopes were reviewed and the isotope importance was determined by generating macroscopic neutron absorption cross-sections in both the thermal and fast/resonance ranges for each of the isotopes being examined. Based on the results of the long-term isotope importance evaluations, a preliminary list of "Principal Isotopes" for long-term criticality control was determined (CRWMS M&O 1994i). The preliminary 31 principal isotopes are shown in Table 6.5-7.

An additional limit on the number of isotopes in the model was their availability in the MCNP cross-section libraries. Nine of the identified principal isotopes are not available in the standard MCNP cross-section sets. The unavailable isotopes are Mo-95, Tc-99, Ru-101, Nd-143, Nd-145, Sm-147, Sm-150, Sm-151, and Sm-152. The 22 available isotopes are used in the MCNP models. The nine unavailable isotopes are absorbers, and, therefore, not accounting for them is conservative.

Table 6.5-7. Preliminary List of Principal Long-Term Burnup Credit Isotopes

O-16	Mo-95	Tc-99	Ru-101	Rh-103
Ag-109	Nd-143	Nd-145	Sm-147	Sm-149
Sm-150	Sm-151	Sm-152	Eu-151	Eu-153
Gd-155	U-233	U-234	U-235	U-236
U-238	Np-237	Pu-238	Pu-239	Pu-240
Pu-241	Pu-242	Am-241	Am-242m	Am-243
Cm-245	-	-	-	-

The criticality evaluations have been performed for various design basis SNF characteristics including 3.75 percent U-235 enrichment, 37 GWd/MTU burnup (40 GWd/MTU for preliminary work), as used by the monitored retrievable storage/MPC group; 1.80 percent U-235 enrichment, fresh fuel, which was established by the monitored retrievable storage/MPC group to conservatively bound the previously listed design basis; and 3 percent U-235 enrichment, 20 GWd/MTU burnup (21 GWd/MTU for BWR), as established by the M&O, and discussed in Section 5. In addition, cases were run for 3.75 percent U-235 enrichment and 3 percent U-235 enrichment fresh fuel to demonstrate the effects of burnup credit. The assembly axial and radial burnup variations and differential loading effects have not been accounted for in the models. These effects are accounted for conservatively with the bias for PWR models, but could be more significant for the BWR models.



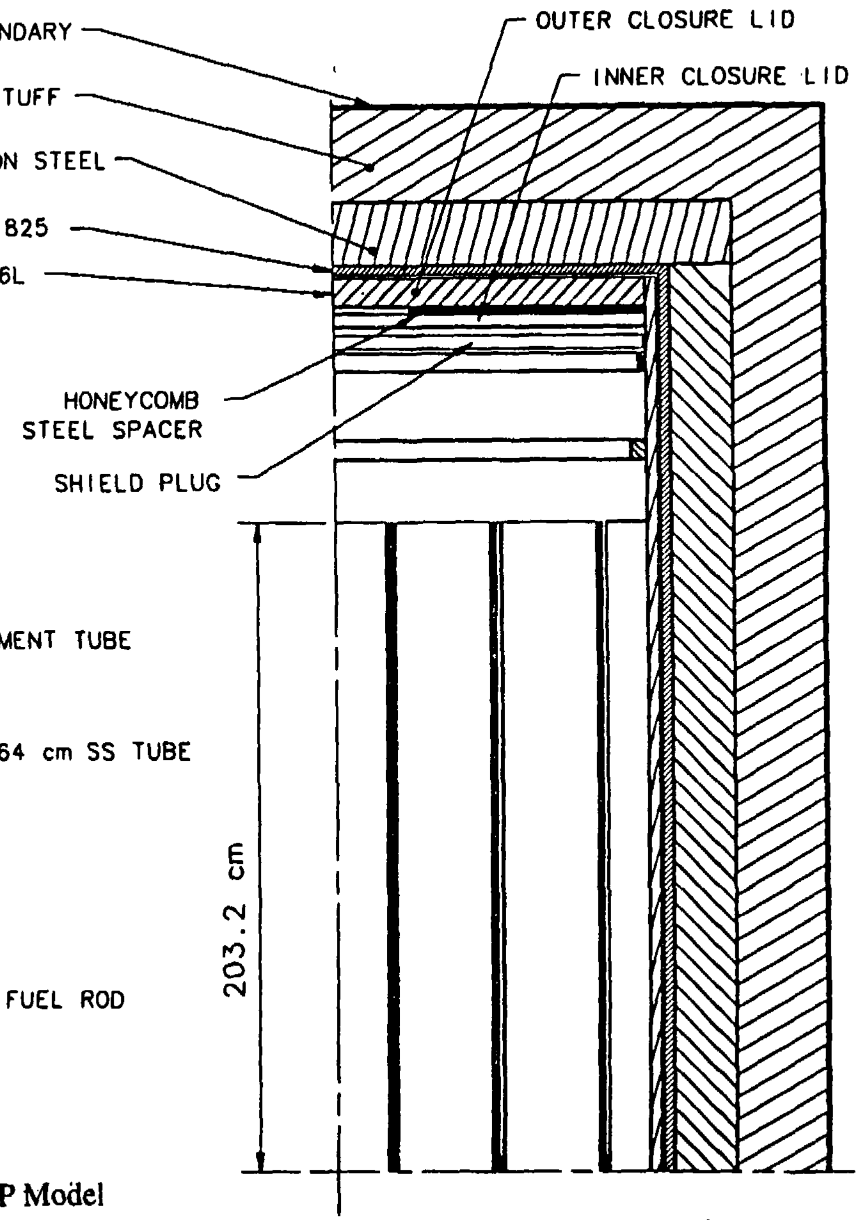


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Figure 6.5-67. MCNP Model

6.5.3.3 Large (125-Ton) Multi-Purpose Canister

This section provides the results of the analyses for PWR and BWR configurations. The 21 assembly PWR design as presented in the *MPC Conceptual Design Report* (CRWMS M&O 1994k) will likely require additional criticality control measures over those discussed in that report in order to meet the MGDS *Engineered Barrier Design Requirements Document* (YMP 1994b) criticality requirements. The 40 assembly BWR design does meet acceptance criteria with burnup credit and would likely meet acceptance criteria with the proposed additional criticality control measures if burnup credit was not accepted by the Nuclear Regulatory Commission.

The inner and outer barriers of the disposal container, into which the MPC is placed, are shown in Appendix B. -

6.5.3.3.1 21 PWR Multi-Purpose Canister

The 21 PWR fuel assembly basket is formed by 21 tubes stacked in a regular array on a nominal 25.436 cm (10.014 inch) center-to-center spacing. The fuel cell opening provided is 22.352 cm (8.8 inches). The fuel cell tubes are formed from 0.635 cm (0.25 inch) thick Al-B plates sandwiched between a 0.635 cm (0.25 inch) thick stainless steel inner wall, and a 0.238 cm (0.094 inch) thick stainless steel outer wrapper. The minimum B-10 loading of the Al-B is indicated to be 0.014 gm/cm^2 (approximately1 weight percent B-10), of which only 75 percent is assumed present to account for variances and burnup in the monitored retreivable storage/MPC analysis (CRWMS M&O 1994k).

The cross-sectional view in the X-Y plane of the physical MCNP model is shown in Figure 6.5-68 and was generated with the MCNP plotting capability.

6.5.3.3.1.1 Standard Evaluation

The aluminum boron alloy is modeled only as aluminum Alloy 1100 because its corrosion rate is higher than the structural materials in the basket. The k_{eff} ($\pm 2\sigma$) values calculated for the different design basis enrichments are as follows (CRWMS M&O 1996c)

3.00% U-235 enrichment, 20 GWd/MTU, five-year decay	0.9818 ± 0.0067
3.75% U-235 enrichment, 37 GWd/MTU, five-year decay	0.9172 ± 0.0066
1.80% U-235 enrichment, 0 GWd/MTU	0.9873 ± 0.0057
3.00% U-235 enrichment, 0 GWd/MTU	1.1492 ± 0.0055
3.75% U-235 enrichment, 0 GWd/MTU	1.2007 ± 0.0062

The burnup credit reactivity worth { $(k_2 - k_1) / (k_1 * k_2)$ } is -0.15 and -0.26 for the 3 percent enriched and 3.75 percent enriched fuels, respectively. The 1.80 percent U-235 enrichment, 0 GWd/MTU does bound the burned fuels. Note that with biases applied, none of these cases meet the 0.95 value limit established in 10 CFR 60.131 (b)(7). The 21 assembly conceptual MPC would require additional criticality control measures, such as the addition of filler material at the MGDS surface facility, to meet the *Engineered Barrier Design Requirements Document* criticality requirements for MGDS.

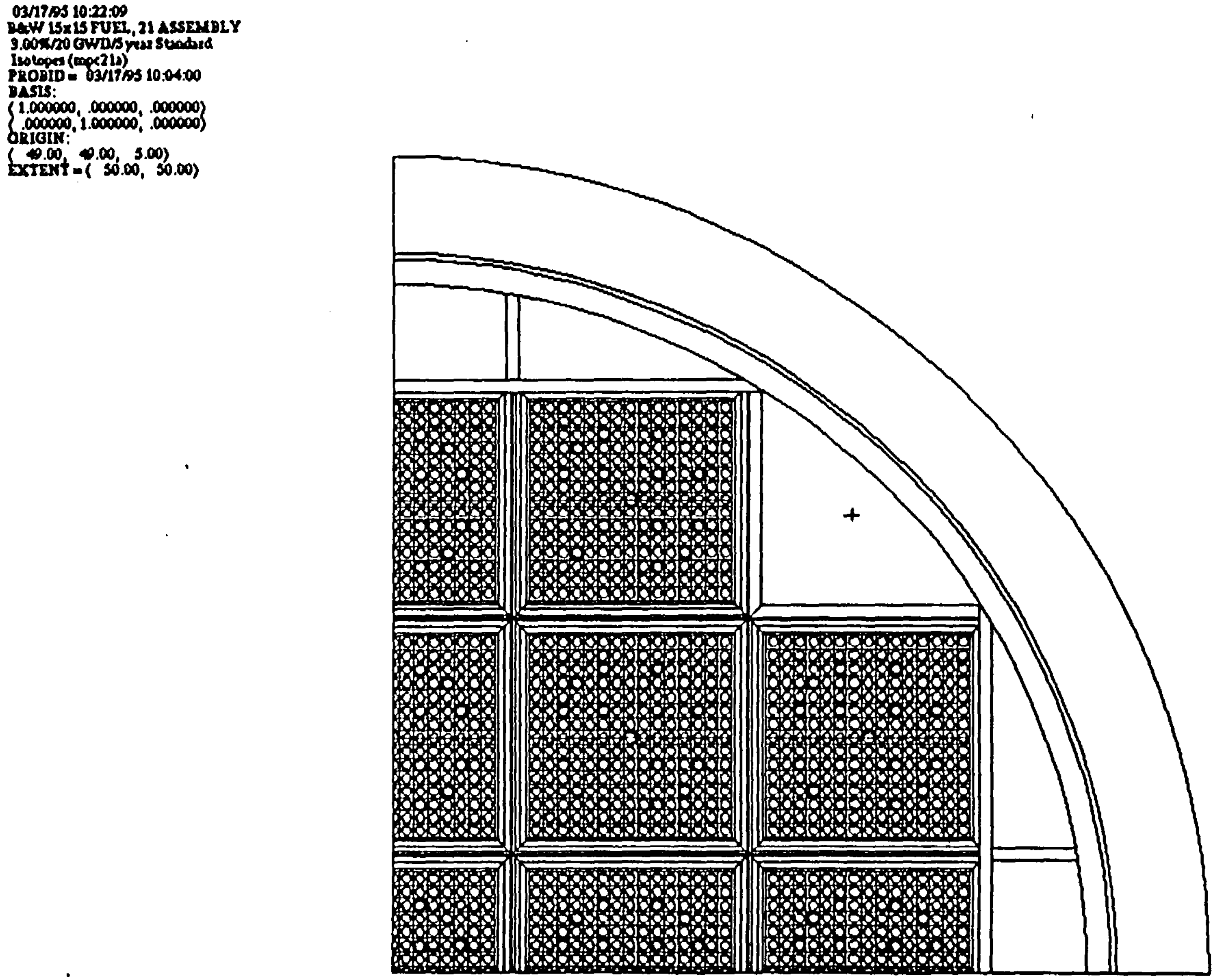

Figure 6.5-68. MCNP Cross-Sectional View of the 21 PWR Assembly Multi-Purpose Canister Waste Package

To demonstrate the effect of different isotopes, three additional calculations were run (CRWMS M&O 1996c) corresponding to the 3 percent U-235 enrichment, 20 GWd/MTU, five-year decay case. The fission product and actinide isotopic makeup of the SNF as modeled is shown in Table 6.5-8. The k_{eff} value for the Actinides Only, Reduced Actinides, and U/Pu Only cases are shown below. The reactivity value for the Actinides Only, Reduced Actinides, and U/Pu Only cases were calculated to be 2.38E-2, 2.76E-2, and 9.70E-2, respectively. These cases serve to demonstrate the effects of different levels of burnup credit.

Actinides Only	1.0053 ± 0.0080
Reduced Actinides	1.0091 ± 0.0077
U/Pu Only	1.0851 ± 0.0048

A "normal" condition case was run with no water moderation using the 3 percent U-235 enrichment, 20 GWd/MTU, and five-year decay, SNF providing a k_{eff} of 0.2738 ± 0.0022.

Table 6.5-8. Spent Nuclear Fuel Composition Makeup for Isotopic Importance Demonstration Cases

Isotopes	**Base Case**	**Actinides Only**	**Reduced Actinides**	**U/Pu Only**
	Included Isotopes are indicated with an X.			
Rh-103	X	-	-	-
Ag-109	X	-	-	-
Sm-140	X	-	-	-
Eu-151	X	-	-	-
Eu-153	X	-	-	-
Gd-155	X	-	-	-
U-233	X	X	-	-
U-234	X	X	X	-
U-235	X	X	X	X
U-236	X	X	X	-
U-238	X	X	X	X
Np-237	X	X	-	-
Pu-238	X	X	X	-
Pu-239	X	X	X	X
Pu-240	X	X	X	-
Pu-241	X	X	X	X
Pu-242	X	X	X	-
Am-241	X	X	X	-
Am-242m	X	X	-	-
Am-243	X	X	-	-
Cm-245	X	X	-	-

6.5.3.3.1.2 Time Effects

The long-term time effects on the criticality potential of SNF in a WP are a criticality concern unique to the MGDS. To calculate the time effect on criticality, isotopic composition information at different time steps was required. The required isotopic compositions for the fuel characteristics at 23 selected decay times were gathered from the Characteristics Database, Radiological Database. Reports of the isotopic composition of the fuel at the 23 times were generated from the Radiological Database, then used to generate number densities that were entered into MCNP model files/input decks. Each fuel characteristics time step represents an MCNP model file/input deck. The results from the MCNP runs are consolidated to form a plot of the change in the criticality potential of the MPC WP k_{eff} over time. The results (CRWMS M&O 1994i) are shown in Figures 6.5-69 and 6.5-70 for the 3 percent U-235 enrichment, 20 GWd/MTU and the 3.75 percent U-235 enrichment, 40 GWd/MTU cases, respectively.

For the first approximately 200 years after discharge, the criticality potential of SNF decreases as the Pu-241 (13.2-year half-life) fissile material decays. From approximately 200 years to approximately 20,000 years, the criticality potential of the SNF increases as Pu-240 (6,580-year half-life) and the medium half-life neutron absorbers decay. After the approximately 20,000-year local peak, the criticality potential of the SNF decreases again as the Pu-239 (24,400-year half-life) fissile material decays.

The advantage of using burnup credit for long-term criticality control can be seen in Figure 6.5-71 (CRWMS M&O 1994i). Figure 6.5-71 plots k_{eff} of 3.75 percent enrichment U-235 and 0 GWd/MTU burnup fresh fuel and 3.75 percent initial enrichment U-235 and 40 GWd/MTU burnup SNF for the 21 PWR Burnup Credit WP design. The differences between the two curves (approximately 0.25 to 0.30) is the criticality control potential of burnup credit.

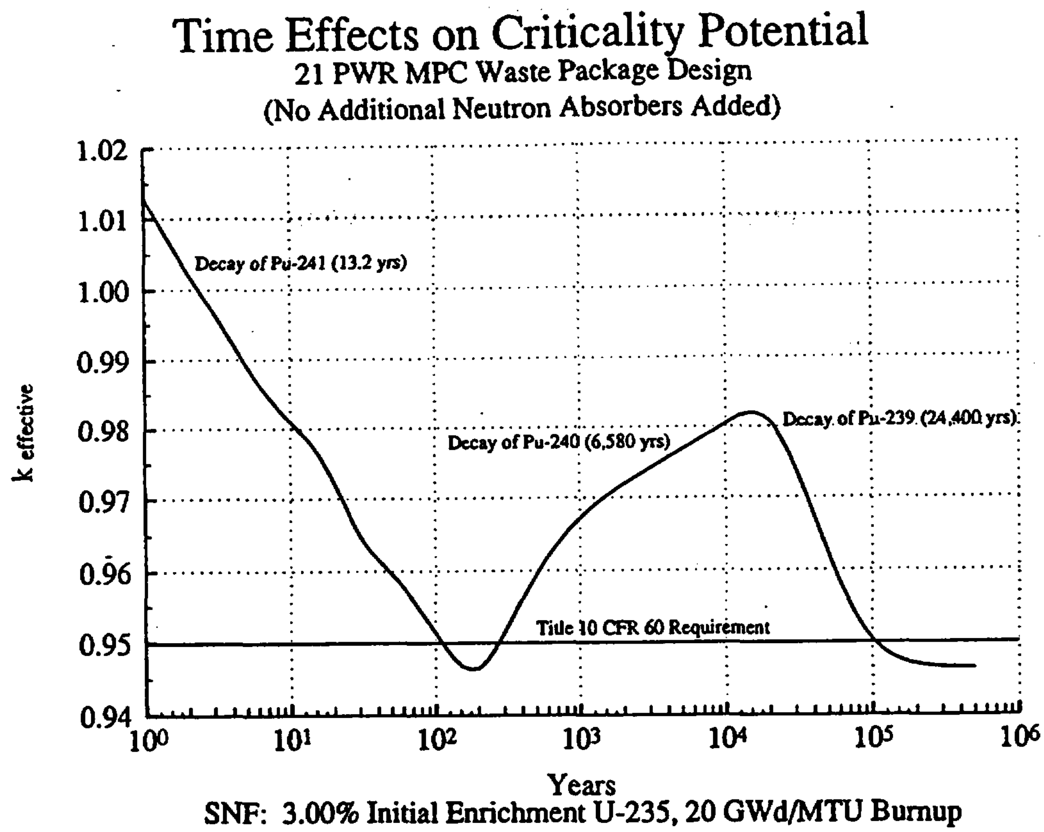

Figure 6.5-69. Time Effects on Criticality Potential, 21 PWR Multi-Purpose Canister Waste Package, MGDS Design Basis Spent Nuclear Fuel

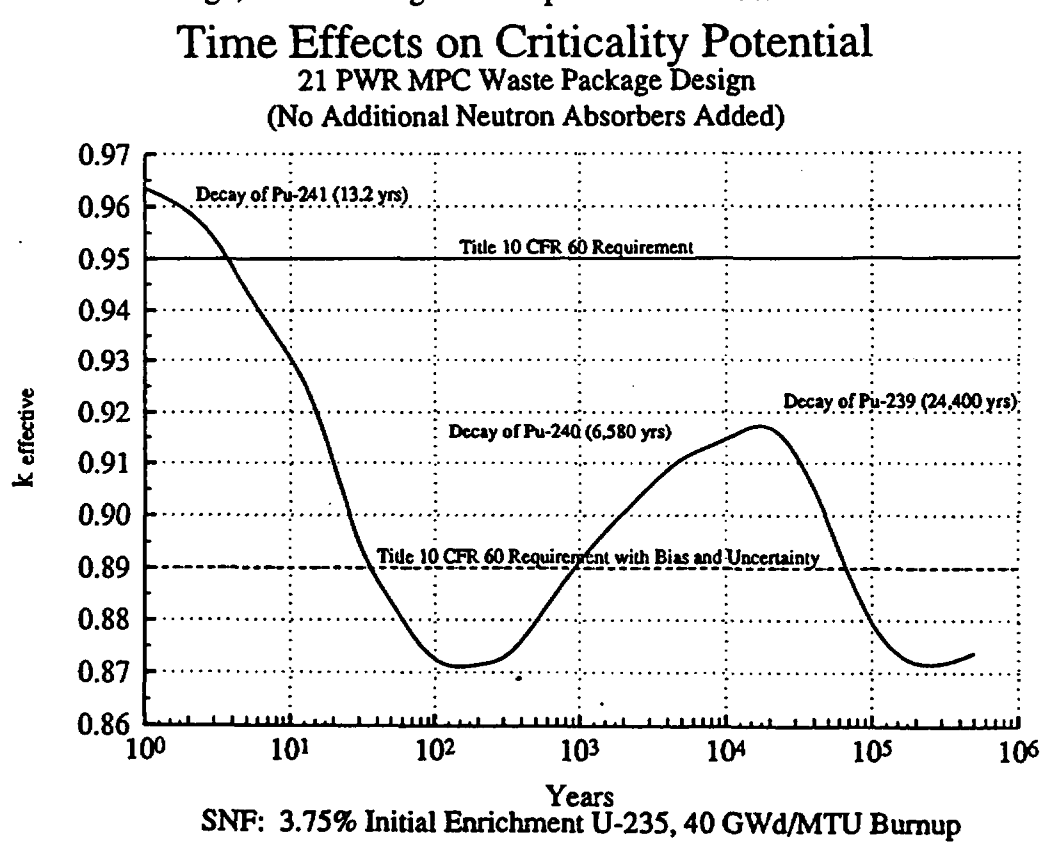

Figure 6.5-70. Time Effects on Criticality Potential, 21 PWR Multi-Purpose Canister Waste Package, Multi-Purpose Canister Design Basis Spent Nuclear Fuel

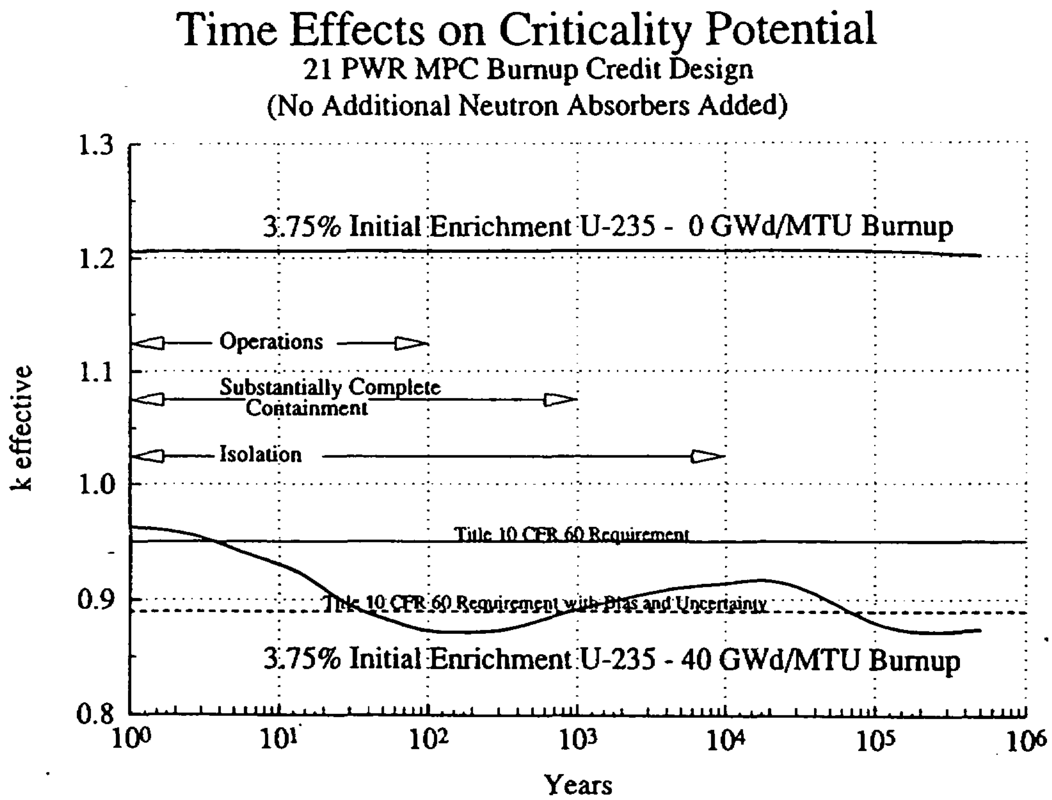

Figure 6.5-71. Time Effects on Criticality Potential, 21 PWR Multi-Purpose Canister Waste Package, Fresh and Spent Multi-Purpose Canister Spent Nuclear Fuel

6.5.3.3.1.3 Criticality Control Methods

A. Control Panels

Aluminum-boron control panels are incorporated into the basket design in the MPC conceptual design, but as mentioned earlier, may not be acceptable for long-term credit because of higher corrosion rates than the basket structural and support materials. Stainless steel boron alloy has much better long-term material performance and so is preferred by the WP development group. The stainless steel base material allows the addition up to 2.25 percent boron by weight in the stainless steel 304B (SS-B) material (Carpenter Technology Corp. 1993). To retain near peak mechanical properties, boron contents less than-or equal to 1.75 weight percent are preferred. Zirconium-hafnium Alloy 702 is another alternative material. Zirconium-hafnium alloys offer the potential for better long-term performance than aluminum-boron and SS-B; the hafnium is not leachable from the zirconium and occurs naturally in zirconium deposits. The reduction in k_{eff} for the three materials of equal thickness used as control panels in the 21 PWR MPC WP as a function of absorber weight percent are shown in Figure 6.5-72 (CRWMS M&O 1994i).

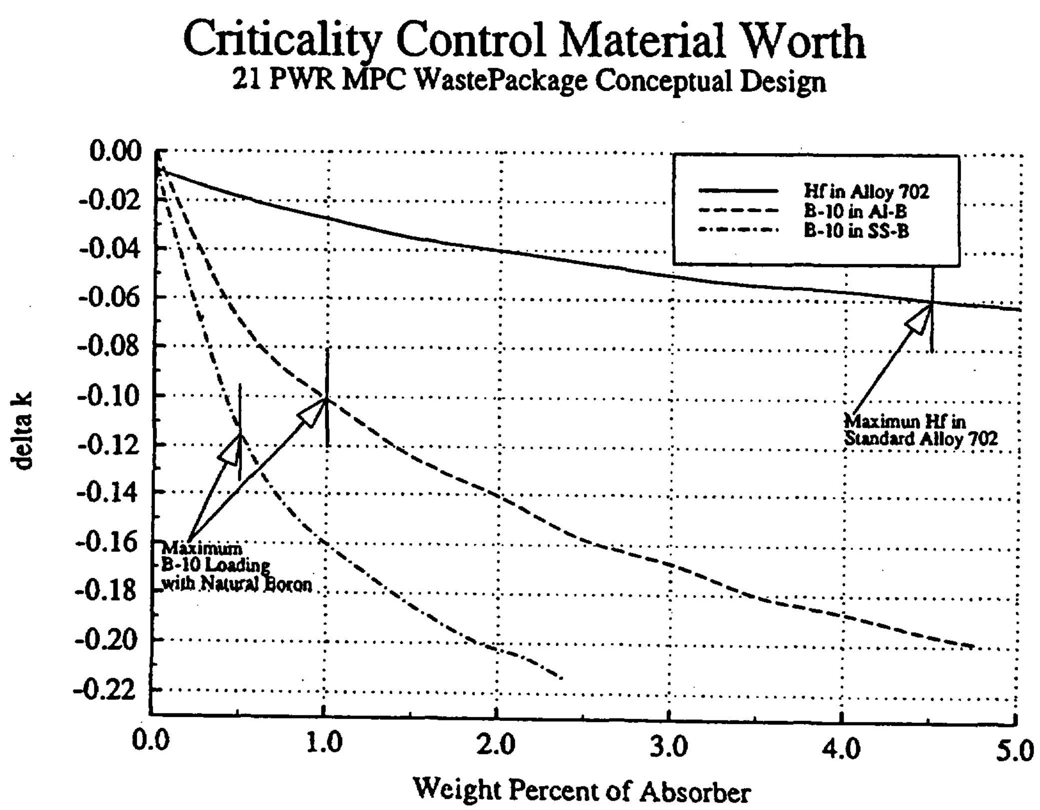

Figure 6.5-72. Criticality Control Material Worth, 21 PWR Multi-Purpose Canister Waste Package

The criticality control potential of control panels is limited by the interrelated properties of panel thickness and the allowable weight percentage of boron (or other absorber) in the panel's base material. Control panels alone (no burnup credit) do not have sufficient criticality control potential to control criticality in the larger capacity WPs for all SNF without them becoming very expensive because of excessive thickness and neutron absorber loading. With control panels containing natural boron, the k_{eff} for the 21 PWR Burnup Credit WP conceptual designs can be lowered up to approximately 0.10. Using control panels with fully B-10 enriched boron, k_{eff} can be lowered by up to approximately 0.20 for the 21 PWR Burnup Credit WP conceptual design. Figure 6.5-73 shows the criticality control potential of the Al-B control panels in the MPC (CRWMS M&O 1994i).

Without long-term material tests to confirm the removal rate of boron, some initial engineering estimates have been made for the loss rate of B-10. These evaluations indicate that approximately 20 percent of the B-10 could be removed from the SS-B over the 10,000 years of disposal isolation (CRWMS M&O 1994i). Assuming no loss of boron over the entire time period indicates that there is no problem in controlling criticality, as shown in Figure 6.5-74. With the loss of B-10, a problem develops in the later years, as shown in Figure 6.5-75. If 25 percent additional B-10 is included initially, criticality control is shown for the entire time period, as shown in Figure 6.5-76. These calculations were performed using Al-B and assuming the boron removal rates of SS-B (CRWMS M&O 1994i). These calculations indicate that, even if credit for the Al-B could be taken, enriched boron must be used in order to reduce k_{eff} below regulatory requirements when time effects are taken into account.

B. Control Rods

The WPD Group has examined the use of new reactor control rod assemblies, spent reactor control rod assemblies, and control rod assemblies made especially for WPs (CRWMS M&O 1994i). The disposable control rod assemblies are the leading candidate and have a Δk range between 0.06 and 0.33. Preliminary results are shown in Table 6.5-9 for full length disposable control rod assemblies in all fuel assemblies of the 21 PWR MPC Burnup Credit Conceptual Design. Control rods are modeled in all guide tubes of B&W 15x15, 3.75 percent enrichment U-235, 0 GWd/MTU fuel. The reductions in k_{eff} reported in the table are the differences between a case without control rods (water filled guide tubes) and cases with the listed types of control rods. Note that these results do not include the reduction in effectiveness due to long-term degradation and neutron absorber depletion.

Table 6.5-9 shows that disposable control rod assemblies are an excellent method of criticality control. However, there is a difficulty in that not all fuel assemblies are capable of accepting control rods. Information from the Waste Acceptance Group and Edison Electric Institute projections indicates that there are approximately 2400 PWR assemblies (approximately 2 percent of total number) that will be unable to accept control rod assemblies.

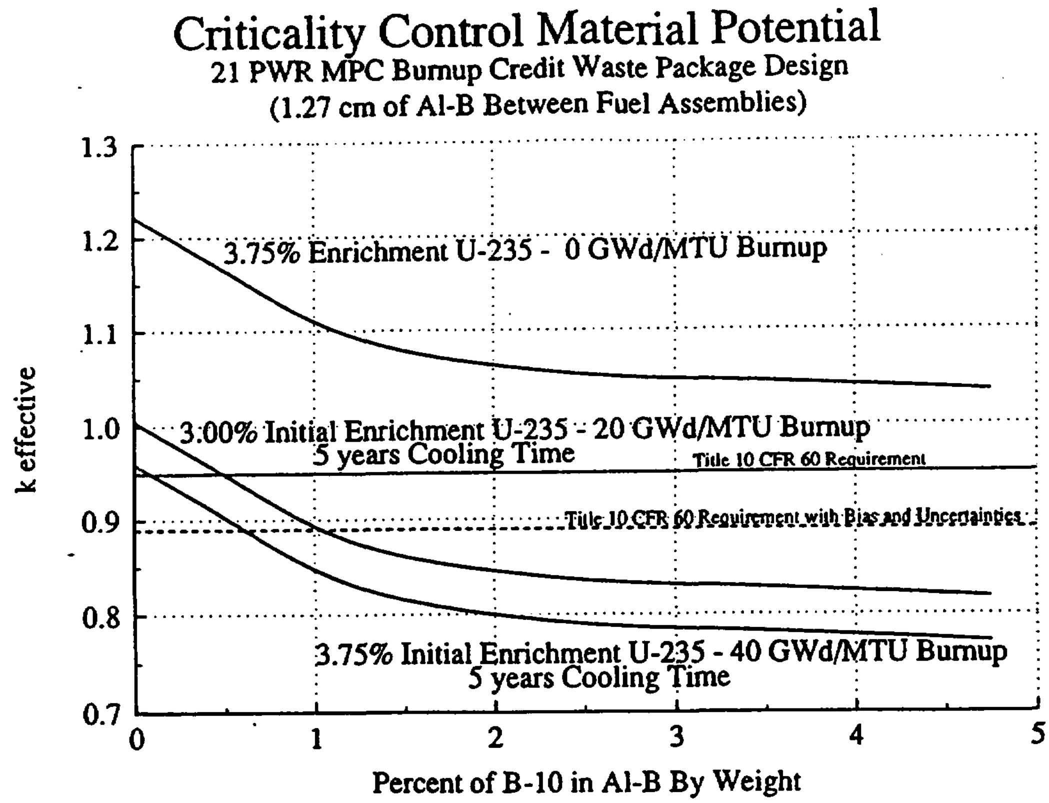

Figure 6.5-73. Criticality Control Material Potential, Al-B Control Panels, 21 PWR Multi-Purpose Canister Waste Package

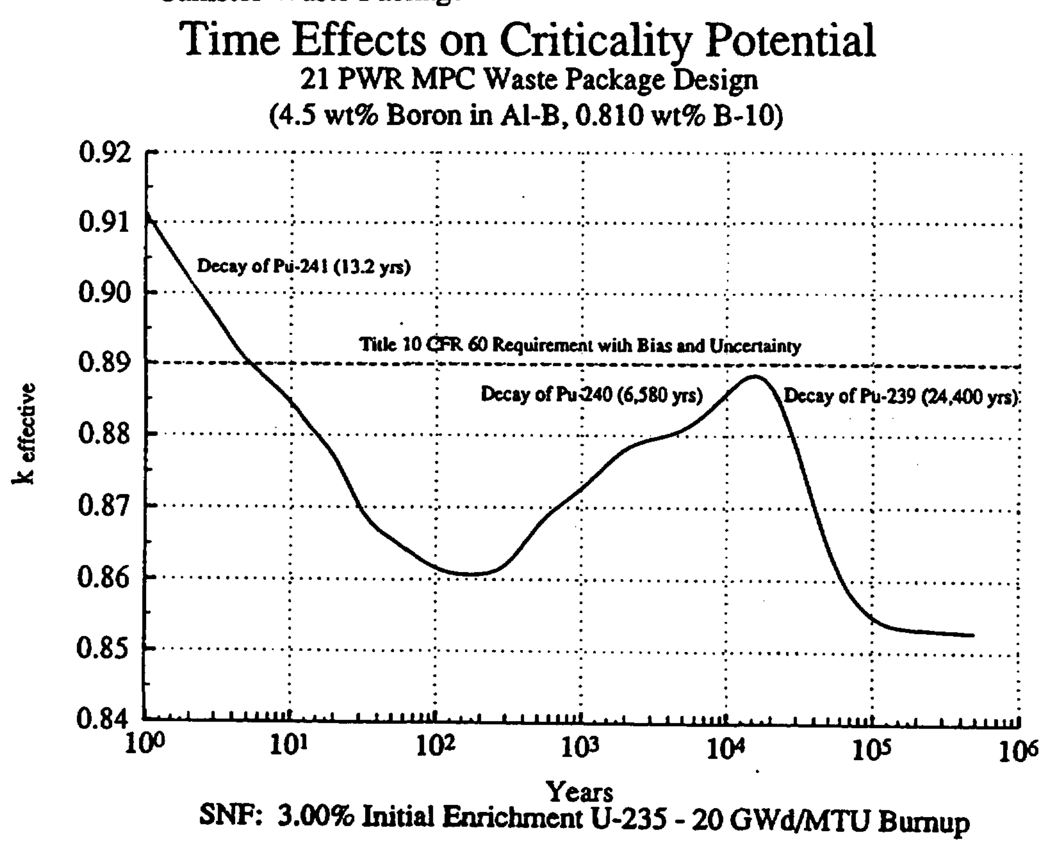

Figure 6.5-74. Time Effects on Criticality Potential, w/o B Removal, 21 PWR Multi-Purpose Canister Waste Package, MGDS Design Basis Fuel

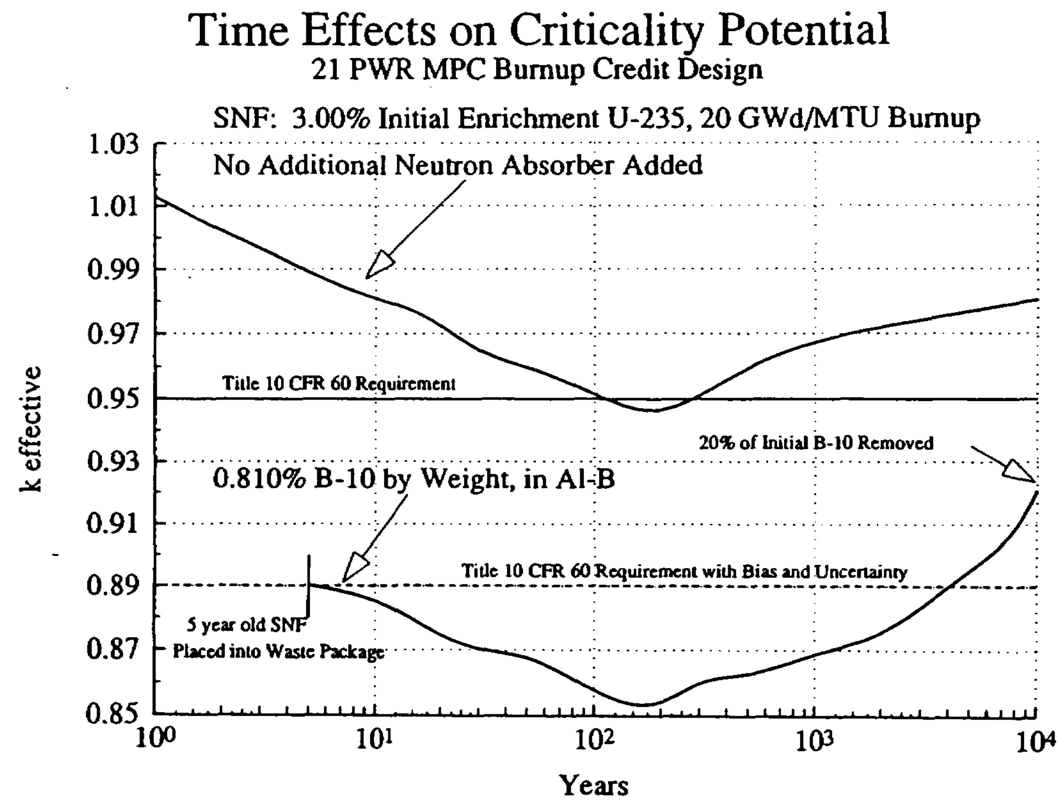

Figure 6.5-75. Time Effects on Criticality Potential, w/ B Removal, 21 PWR Multi-Purpose Canister Waste Package, MGDS Design Basis Fuel

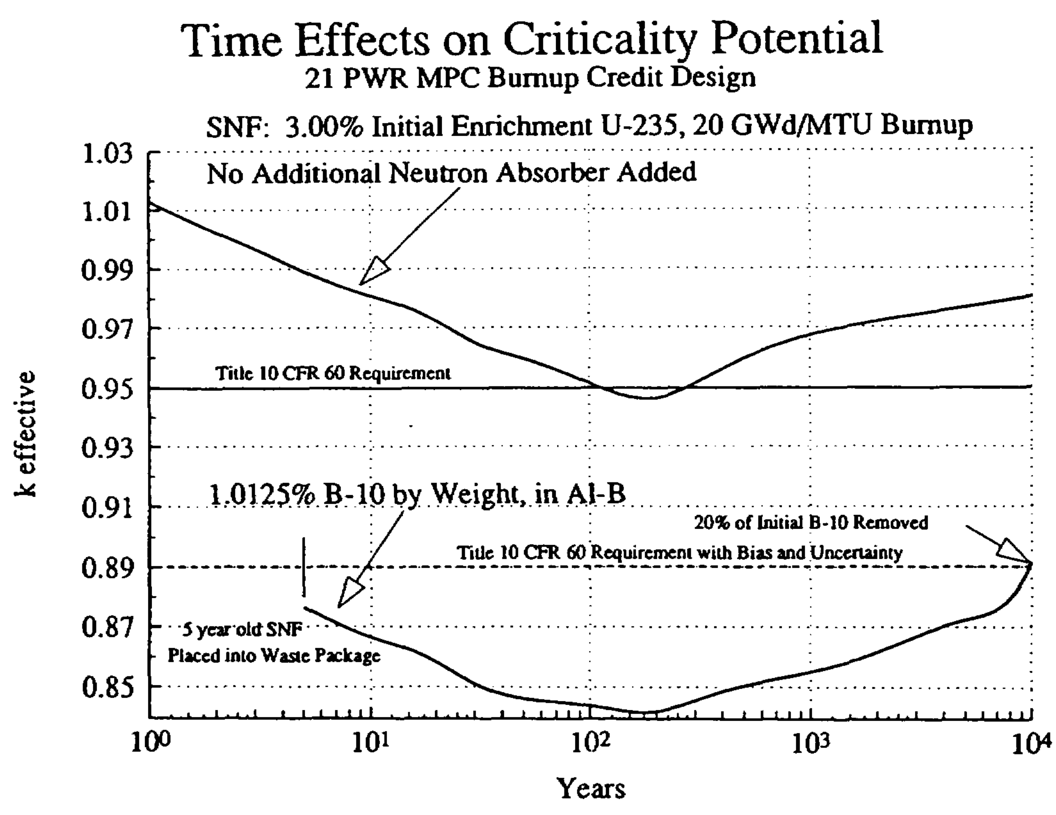

Figure 6.5-76. Time Effects on Criticality Potential, w/o B Removal & 20 percent Increase, 21 PWR Multi-Purpose Canister Waste Package

Table 6.5-9. Criticality Control Potential of Disposable Control Rod Assemblies in a 21 Assembly Multi-Purpose Canister Waste Package

Control Rod Material	Decrease in k_{eff}
Unborated Stainless Steel (SS 316L)	0.06
Borated Stainless Steel (SS 304B6A, natural boron)	0.22
Borated Stainless Steel (SS 304B6A, enriched boron)	0.30
Boron Carbide (B_4C pellets, natural boron)	0.33
Borosilicate Glass (Code 7740, Pyrex)	0.23
Zircaloy	0.01
Zirconium Alloy 702	0.05
Zirconium Alloy + 10 weight percent hafnium	0.09
70-30 hafnium-zirconium Alloy	0.24
Spent silver-indium-cadmium control rod (77% silver, 11% indium, 9% cadmium, and 3% tin representing approximately 3.5×10^{21}nvt fluence over the entire rod length)	0.26

C. Filler Material - Moderator Displacement

An analysis examined the use of filler material for moderator displacement criticality control, to obtain a measure of the benefit of this approach. This criticality analysis basically assumed a worst-case circumstance of (1) no burnup credit, (2) no neutron absorber materials, and (3) the sudden catastrophic breaching and flooding of the WP. Iron shot was chosen as the moderator-displacing filler material. For this analysis, all the free space within the WP/canister (i.e., among the fuel rods in an assembly, inside the empty guide tubes, and around the outside of the basket) was filled uniformly (smeared properties) with various compositions of iron shot and water. Various mixtures were examined because the effectiveness of moderator displacement criticality control depends on the quantity of moderator displaced by the filler. This in turn depends on the quantity of filler that may be emplaced into the WP/canister. The total quantity of iron shot filler is dependent on the density of the bulk material and the percentage fill.

Results (without bias and uncertainty applied) of the investigations are presented in Figure 6.6-77 (CRWMS M&O, 1996c). For the case of graded shot and 85 percent fill, k_{eff} is about 0.81. For the case of mixed grade shot and 100 percent fill, k_{eff} is about 0.70. The acceptable limit is not exceeded for fresh fuel until approximately 58 percent of the space is filled with water. These preliminary results do show a significant reduction of criticality potential sufficient to reduce the k_{eff} for the design basis SNF below the acceptable limit with only a 10 to 25 percent displacement of water. A relatively low level of moderator displacement will adequately supplement burnup credit and/or supplemental neutron absorber credit to meet regulatory requirements.

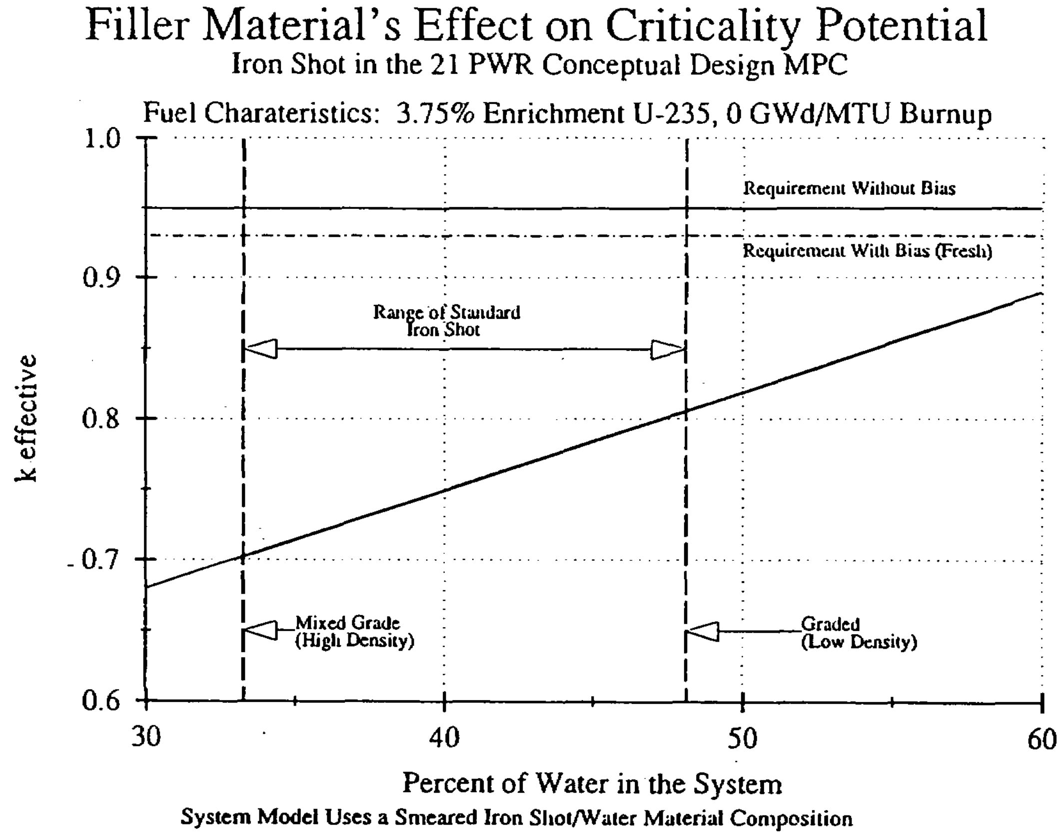

Figure 6.5-77. Criticality Control Potential of Filler Material, 21 PWR Multi-Purpose Canister Waste Package, Multi-Purpose Canister Design Basis Fuel

6.5.3.3.1.4 Basket Degradation

The effect of collapsing the basket structure is also of interest for criticality control in the long term. A model of a PWR WP was run at five years decay in which the water gap between the assembly and the inner basket wall (approximately 0.36 cm) was eliminated all around to simulate the collapse of the basket array. The assemblies were conservatively modeled as centered in the WP and the structural material was maintained resulting in a k_{eff} of 0.9737 ± 0.0043. A more realistic case representing the collapse of the basket array in one dimension and a shift of the array to the bottom (assuming WP on side) was calculated providing a k_{eff} of 0.9724 ± 0.0039. These calculations indicate that the effect of minor geometric changes is insignificant. The extreme effect of removing the basket material and collapsing the assemblies to the bottom of the disposal container (horizontal emplacement) was investigated producing a k_{eff} of 1.1218 ± 0.0054. This last configuration provides a bounding result for a configuration which is unlikely. Future probabilistic analyses will indicate the true limiting configurations.

6.5.3.3.2 40 BWR Multi-Purpose Canister

The 40 BWR fuel assembly basket is formed by 40 tubes stacked in a regular array on a nominal 18.256 cm (7.187 inch) center-to-center spacing. The fuel cell opening provided is 15.240 cm (6 inches). The fuel cell tubes are formed from 0.635 cm (0.25 inch) thick Al-B plates sandwiched between a 0.635 cm (0.25 inch) thick stainless steel inner wall, and a 0.238 cm (0.094 inch) thick stainless steel outer wrapper (CRWMS M&O, 1994k).

The cross-sectional view in the X-Y plane of the physical MCNP model is shown in Figure 6.5-78 and was generated with the MCNP plotting capability.

The assembly axial and radial burnup variations and differential loading effects have not been accounted for in the models. These effects could be significant for the BWR models and will be thoroughly investigated in the future. The axial effects will be examined in detail in the future.

6.5.3.3.2.1 Standard Evaluation

The aluminum boron alloy is modeled only as aluminum Alloy 1100 because its corrosion rate is higher than the structural materials in the basket. The k_{eff} ($\pm 2\sigma$) values calculated for the different design basis enrichments are as follows (CRWMS M&O, 1996d).

3.00% U-235 enrichment, 21 GWd/MTU, five-year decay	0.8031 ± 0.0070
3.00% U-235 enrichment, 0 GWd/MTU	0.9625 ± 0.0069
3.75% U-235 enrichment, 0 GWd/MTU	1.0134 ± 0.0077

The burnup credit reactivity worth is -0.206 for the 3 percent enriched fuel. Note that burnup credit is required to meet the 0.95 (0.89 with bias and uncertainty) value limit established in 10 CFR 60.131(b)(7). With further criticality control measures, the 40 BWR assembly MPC design would likely meet the requirement for MGDS for the DBF with no burnup credit.

A "normal" condition case was run with no water moderation using the 3 percent U-235 enrichment, 21 GWd/MTU, and five-year decay, SNF providing a k_{eff} of 0.2505 ± 0.0018.

For similar enrichment and burnup, the 40 BWR assembly MPC contains less than 75 percent as much fuel as the 21 PWR assembly MPC. This accounts for the much lower k_{eff} values calculated for the BWR MPC.

6.5.3.3.2.2 Time Effects

The long-term time effects for the design basis BWR fuels are similar to those described in Section 6.5.3.3.1.2 for the 21 PWR assembly MPC based on evaluation of results for eight time steps (CRWMS M&O, 1996d). The design basis U-235 enrichments and burnup are essentially the same, and only neutron spectrum effects contribute to differences, producing slightly higher plutonium inventories in the BWR. The results for the BWR fuel follow the same general trend as the PWR fuel with a decrease in reactivity out to 200 years, an increase from 200 years to 20,000 years, and a decrease from 20,000 years to 1 million years.

6.5.3.3.2.3 Criticality Control Methods

The use of control panels containing boron would have a similar percentage effect on k_{eff} as that reported for the 21 assembly PWR MPC in Section 6.5.3.3.1.3 Part A.

BWR assemblies do not have the control rod guide tubes to allow the insertion of control rods, prohibiting the use of this criticality control option.

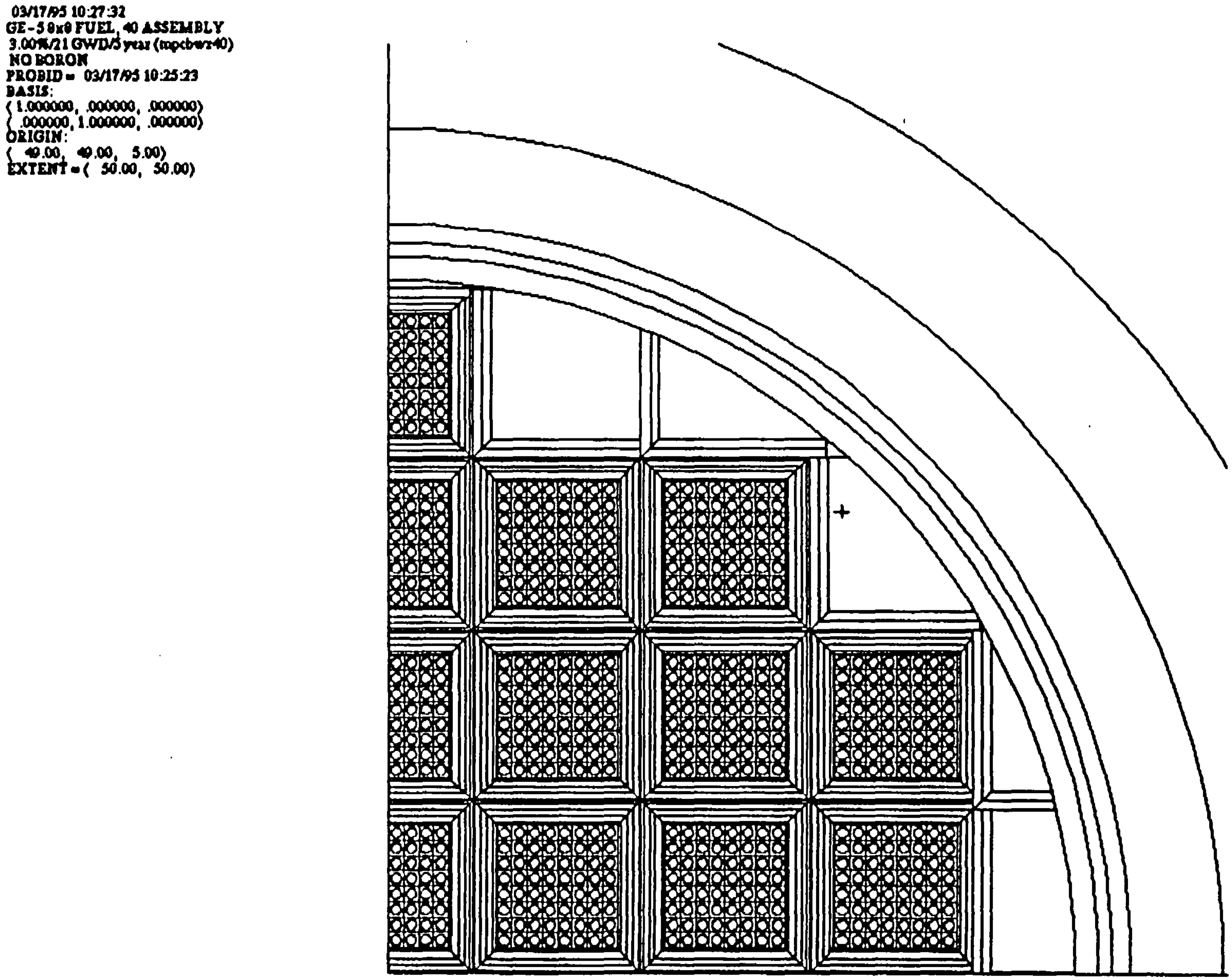

Figure 6.5-78. MCNP Cross-Sectional View of the 40 BWR Assembly Multi-Purpose Canister Waste Package

The use of filler material to displace moderators would have a similar percentage effect on k_{eff} as that reported for the 21 assembly PWR MPC in Section 6.5.3.3.1.3 Part C.

6.5.3.3.2.4 Basket Degradation

The effect of collapsing the basket structure is also of interest for criticality control in the long term. A model of the BWR WP was run at five years decay in which the water gap between the assembly shroud and the inner basket wall (approximately 0.75 cm) was eliminated all around to simulate the collapse of the basket array. The assemblies were conservatively modeled as centered in the WP and all structural material was maintained resulting in a k_{eff} of 0.8420 ± 0.0052. A more realistic case representing the collapse of the basket array in one dimension and a shift of the array to the bottom (assuming WP on side) was calculated for the BWR SNF providing a k_{eff} of 0.8198 ± 0.0057 (CRWMS M&O, 1996d). The increase in k_{eff} due to collapse of the basket is higher for the BWR design than the PWR design because of the greater tolerance (gap) between the assembly and basket walls. Even in the conservative collapse case, a significant margin was maintained for this design in meeting acceptance criteria.

6.5.3.4 Small (75-Ton) Multi-Purpose Canister

This section provides the results of the analyses for PWR and BWR configurations. The 12 assembly PWR design will likely require additional criticality control measures beyond those discussed in the MPC Conceptual Design Report in order to meet the MGDS *Engineered Barrier Design Requirements Document* criticality requirements. The 24 assembly BWR design, like the 40 assembly BWR design, would meet acceptance criteria with burnup credit and would likely meet acceptance criteria with additional criticality control measures even if burnup credit is not accepted by the Nuclear Regulatory Commission.

The inner and outer barriers of the disposal container, into which the MPC is placed, are shown in Appendix B.

6.5.3.4.1 12 PWR Multi-Purpose Canister

The 12 PWR fuel assembly capacity basket is shown in Figure 6.1-2. The basket array is formed by 1.27 cm (0.5 inch) thick structural steel members and a stacked arrangement of fuel cells. The array formed includes "flux trap" criticality control design features. A symmetrical configuration of three fuel cells is placed in each of the four quadrants formed by a thick stainless steel structural support cruciform. Water gaps are formed between the 1.27 cm (0.5 inch) thick structural steel parallel plates by 2.54 cm (1 inch) spacer rods. The fuel cell opening provided is 22.860 cm (9 inches). The fuel cell tubes are formed from 0.272 cm (0.107 inch) thick Al-B sheet sandwiched between a 0.238 cm (0.094 inch) thick stainless steel inner wall, and a 0.238 cm (0.094 inch) thick stainless steel outer wrapper. Water gaps are between some fuel cells by 2.54 cm (1 inch) "ladder" spacers to form flux traps between adjacent fuel cells not separated by thick structural stainless steel formed water gaps (CRWMS M&O, 1994k).

Flux traps are not considered acceptable for disposal criticality control because of long-term material performance concerns. Flux trap designs depend on the structural materials to form the separative water gaps, which control criticality. If the flux trap's separative gap collapses, the neutron interaction between fuel assemblies increases the criticality potential of the system. The structural performance of materials is one of the first physical properties lost as the materials degrade. After the containment barriers have failed, the basket structural materials will likely lose their structural integrity over the period of isolation, and a flux trap will likely be unable to provide criticality control. Flux traps will be assumed collapsed unless first demonstrated by analysis to remain intact.

Flux trap designs are assumed to be filled with fresh fuel up to some initial enrichment limit. The reactivity increase due to collapse of the flux traps would cause the k_{eff} to go over the 0.95 limit if the package k_{eff} was at or near this limit with intact flux traps. Flux traps can be combined with other criticality control measures to form hybrid strategies that may provide criticality control for disposal (i.e., provide sufficient neutron absorber to control criticality with collapsed flux traps).

The cross-sectional view in the X-Y plane of the physical MCNP model for both the flux trap and collapsed flux trap cases is shown in Figures 6.5-79 and 6.5-80 and was generated with the MCNP plotting capability.

6.5.3.4.1.1 Standard Evaluation

The aluminum boron alloy is modeled only as aluminum alloy 1100 because its corrosion rate is higher than the structural materials in the basket. The following models are based on collapsed flux traps. The k_{eff} ($\pm 2\sigma$) values calculated for the different design basis enrichments are as follow (CRWMS M&O, 1996e).

3.00% U-235 enrichment, 20 GWd/MTU, five-year decay	0.9328 ± 0.0069
3.75% U-235 enrichment, 37 GWd/MTU, five-year decay	0.8820 ± 0.0071
1.80% U-235 enrichment, 0 GWd/MTU	0.9581 ± 0.0086
3.00% U-235 enrichment, 0 GWd/MTU	1.1000 ± 0.0079
3.75% U-235 enrichment, 0 GWd/MTU	1.1562 ± 0.0086

The burnup credit reactivity worth is -0.16 and -0.27 for the 3 percent enriched and 3.75 percent enriched fuels, respectively. The 1.80 percent U-235 enrichment, 0 GWd/MTU does bound the burned fuels. Note that with biases applied, the 3 percent enriched burnup credit case does not meet the 0.95 value limit established in 10 CFR 60.131(b)(7), but the 3.75 percent burnup credit case does. Additional criticality control measures, such as the addition of filler material at the MGDS surface facility, offer the potential for the 12 assembly MPC design to meet the requirement for the MGDS.

A "normal" condition case was run with no water moderation using the 3 percent U-235 enrichment, 20 GWd/MTU, and five-year decay, SNF providing a k_{eff} of 0.2355 ± 0.0021.

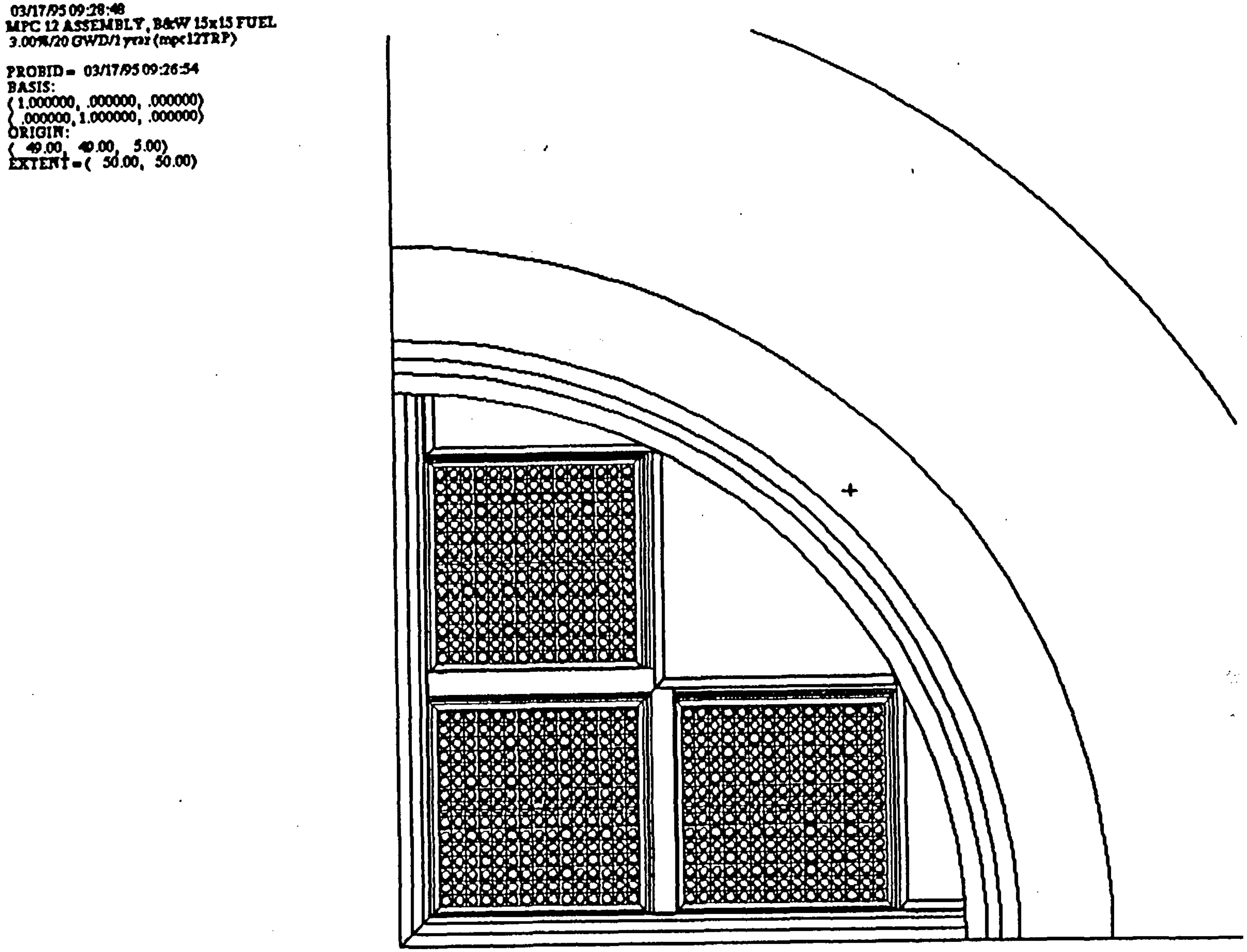

Figure 6.5-79. MCNP Cross-Sectional View of the 12 PWR Assembly Multi-Purpose Canister Waste Package w/ Flux Traps

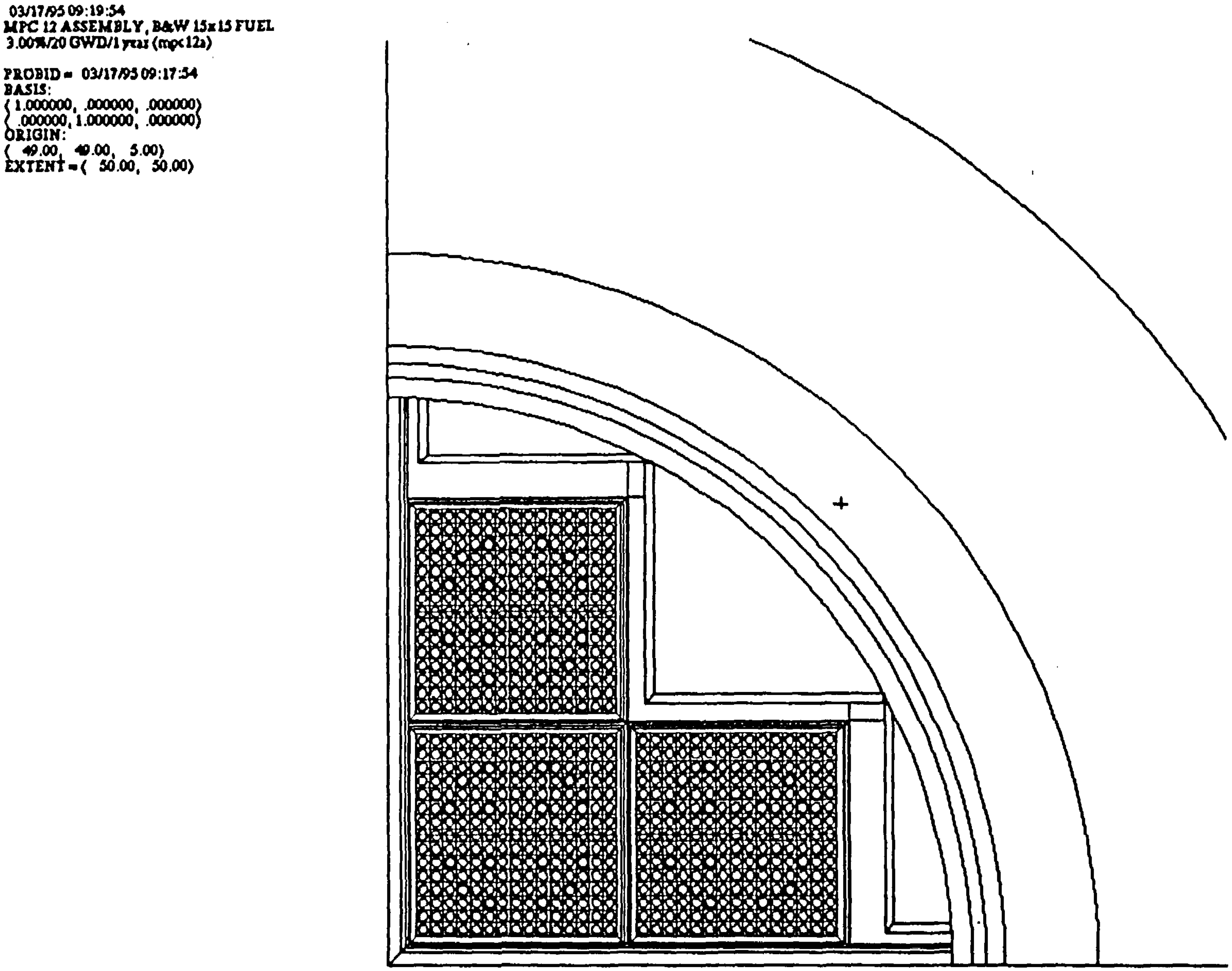

Figure 6.5-80. MCNP Cross-Sectional View of the 12 PWR Assembly Multi-Purpose Canister Waste Package w/ 'apsed Flux Traps

To demonstrate the reactivity worth of the flux traps, a case was run with flux traps for 3 percent U-235 enrichment, 20 GWd/MTU, and five-year decay, providing a k_{eff} of 0.8421 ± 0.0077. The reactivity worth of the flux trap is -0.12. The burnup credit reactivity worth is -0.16 and -0.27 for the 3 percent enriched and 3.75 percent enriched fuels, respectively. The 1.80 percent U-235 enrichment, 0 GWd/MTU does bound the burned fuels. Note that with biases applied, the 3 percent enriched burnup credit case does not meet the 0.95 value limit established in 10 CFR 60.131(b)(7), but the 3.75 percent burnup credit case does. Additional criticality control measures offer the potential for the 12 assembly MPC design to meet the requirement for the MGDS.

A "normal" condition case was run with no water moderation using the 3 percent U-235 enrichment, 20 GWd/MTU, and five-year decay, SNF providing a k_{eff} of 0.2355 ± 0.0021.

To demonstrate the reactivity worth of the flux traps, a case was run with flux traps for 3 percent U-235 enrichment, 20 GWd/MTU, and five-year decay, providing a k_{eff} of 0.8421 ± 0.0077. The reactivity worth of the flux trap is -0.12.

6.5.3.4.1.2 Time Effects

To calculate the long-term time effect on criticality, isotopic compositions for the fuel characteristics at 12 selected decay times were gathered from the Characteristics Database, Radiological Database. Reports of the isotopic composition of the fuel at the 12 times were generated from the Radiological Database, then used to generate number densities that were entered into MCNP model input decks. Each time step for the fuel characteristics represents an MCNP model input deck. The results from the MCNP runs are consolidated to form a plot of the change in the criticality of the MPC WP (k_{eff}) over time. The results are shown in Figure 6.5-81 for the 3 percent U-235 enrichment, 20 GWd/MTU MGDS design basis; and the 3.75 percent U-235 enrichment, 37 GWd/MTU MPC design basis (CRWMS M&O, 1996e).

As also shown in Section 6.5.3.3.1.2 for the 21 assembly PWR MPC, for the first approximately 200 years after discharge, the criticality potential of SNF decreases as the Pu-241 (13.2-year half-life) fissile material decays. From approximately 200 years to approximately 20,000 years, the criticality potential of the SNF increases as Pu-240 (6,580-year half-life) and the medium half-life neutron absorbers decay. After the approximately 20,000-year local peak, the criticality potential of the SNF decreases again as the Pu-239 (24,400-year half-life) fissile material decays.

6.5.3.4.1.3 Criticality Control Methods

The use of control panels containing boron has a similar percentage effect on k_{eff} as that reported for the 21 assembly PWR MPC in Section 6.5.3.3.1.3 Part A as shown in Figure 6.5-82. The use of control rods would have a similar percentage effect on k_{eff} as that reported for the 21 assembly PWR MPC in Section 6.5.3.3.1.3 Part B. The use of filler material to displace moderator would have a similar percentage effect on k_{eff} as that reported for the 21 assembly PWR MPC in Section 6.5.3.3.1.3 Part C.

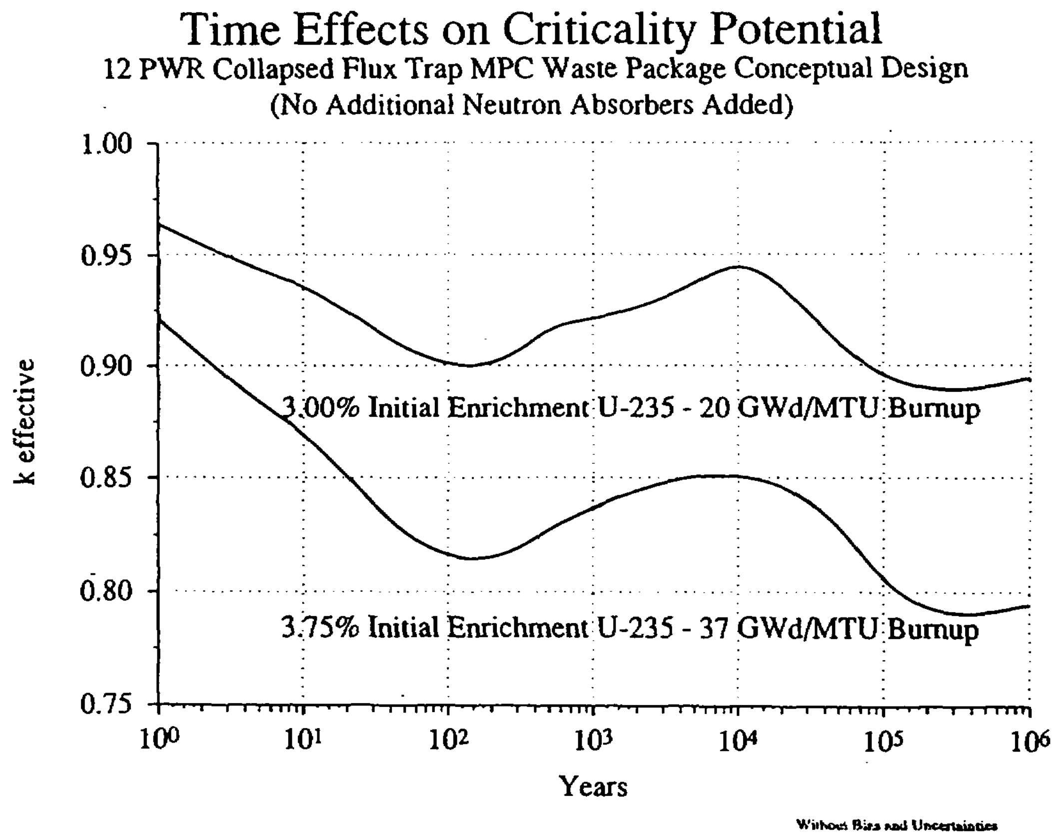

Figure 6.5-81. Time Effects on Criticality Potential, 12 PWR Multi-Purpose Canister Waste Package w/ Collapsed Flux Traps

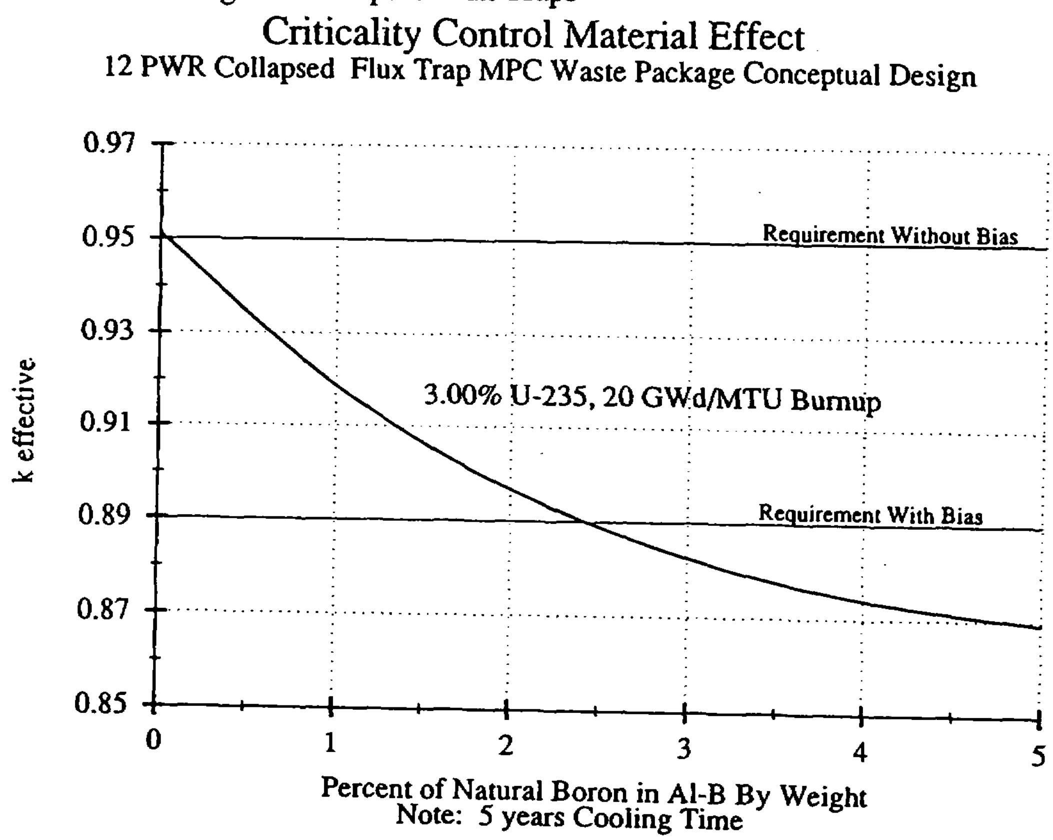

Figure 6.5-82. Aluminum-Boron Criticality Control Panel Effect, 12 PWR Multi-Purpose Canister Waste Package w/ Collapsed Flux Traps, MGDS Design Basis Fuel

6.5.3.4.2 24 BWR Multi-Purpose Canister

The 24 BWR fuel assembly basket array is formed by 24 tubes stacked in a regular array on a nominal 18.256 cm (7.187 inch) center-to-center spacing. The fuel cell opening provided is 15.240 cm (6 inches). The fuel cell tubes are formed from 0.635 cm (0.25 inch) thick Al-B plates sandwiched between a 0.635 cm (0.25 inch) thick stainless steel inner wall, and a 0.238 cm (0.094 inch) thick stainless steel outer wrapper (CRWMS M&O, 1994k).

A model has not been developed to date for the 24 BWR assembly MPC.

6.5.3.4.2.1 Standard Evaluation

Proportionally, the 24 BWR assembly MPC is slightly more heavily loaded than the 40 BWR assembly MPC compared to the corresponding PWR designs. The 24 BWR assembly MPC is loaded with approximately 75 percent as much fuel as the 12 PWR assembly MPC, for similar initial enrichments and burnup. With burnup credit, the 24 BWR assembly design would meet the requirements for MGDS, based on analysis and comparison of the results for the 21 and 12 PWR assembly designs and the 40 BWR design presented above. Additional criticality control measures would be required for the 24 BWR assembly design to meet MGDS requirements without burnup credit.

6.5.3.4.2.2 Time Effects

The long-term time effects for the design basis BWR fuels are expected to be similar to those described in Section 6.5.3.3.1.2 for the 21 PWR assembly MPC. The design basis U-235 enrichments and burnup are essentially the same, and only neutron spectrum effects will contribute to differences, producing slightly higher plutonium inventories in the BWR.

6.5.3.4.2.3 Criticality Control Methods

The reactivity effect of the various control methods are expected to be similar to those discussed for the PWR MPC disposal containers.

6.5.4 Shielding Analysis

With the exception of the MPC top end shield plug, the MPC performs no direct, stand-alone, shielding functions. However, credit for the MPC shell is taken in MPC shielding calculations to account for its indirect shielding effects.

For the MGDS, the MPC disposal container provides some direct shielding function. The MPC is placed into the disposal container either in its entirety or after removal of the upper shield plug section of the MPC. The disposal containers are designed primarily for containment with materials that also provide shielding. No materials whose only purpose is shielding are located in the WP disposal containers.

A basic property of the WP is high radioactivity. Because limiting the dose to workers is a major radiation protection goal (as low as reasonably achievable) two methods for limiting the dose rate to workers have been devised: shielding and remote handling. With these two methods, three strategies for limiting the dose rate from WPs are being considered: use of a shielding WP sleeve, use of a self-shielding containment barrier, and use of an emplacement transport shield.

The shielding sleeve strategy places shield materials, specific to the shielding needs of the source in a "sleeve" around the outer containment barriers of the disposal container. In the case of SNF WPs, a concrete shielding sleeve has been considered. The shielding sleeve strategy would not necessarily need remote handling and could allow manned inspection of the WPs.

The self-shielding containment barrier strategy thickens the outer containment barrier to reduce the dose rate. The metallic outer barriers for the SNF WPs are not optimum neutron shields. Thus, this strategy would require remote handling unless the additional weight and cost of an extremely thick containment barrier are acceptable.

The emplacement transport shield strategy uses the incidental shielding of the WP and a shield carried on the WP emplacement transporter to reduce the dose rate. The transport shield can have the optimized shield materials that are not allowed around, or in, the emplaced WP. This strategy relies upon remote handling operations.

An option of combining the self-shielded WP and transporter shield strategies is possible. The idea is that the two strategies in tandem would provide the necessary shielding. However, the disadvantages for both strategies still apply.

An initial target dose rate of 20 mrem/hr at 2 m from the shield surface was established for WP radiation shielding (CRWMS M&O, 1994i). This limiting dose rate is based upon the 10 CFR 20.1201(a)(i) 5 rem annual occupational dose limit for 250 work days per year (2,000 hours per year) and allowing up to one hour per day in the radiation field. The base 250 work days per year does not account for vacation, holidays, or training time so some additional margin of safety exists for the target dose rate.

The target dose rate listed above was lowered to 10 mrem/hr at 2 m for more recent calculations (two-dimensional). This dose rate is the same as that specified by the transportation regulation from 10 CFR 71 and would provide a lower bounds for application to repository operations. This dose rate would allow more direct access to WPs as far as radiation protection is concerned and is consistent with as-low-as-reasonably-achievable goals.

6.5.4.1 Codes, Isotopic Inventory, and Source Terms

One-dimensional shielding analysis has been performed using the ANISNBW computer program Version 4.5HP on Hewlet Packard 9000 730/735 workstations. ANISNBW is a B&W Nuclear Technologies version of the ANISN-W one-dimensional discrete ordinates transport program developed at Oak Ridge National Laboratory (BWNT, 1993a).

Two-dimensional shielding analysis has been performed using the TORT-DORT Version 2.12.14 computer program. TORT-DORT is a two- or three-dimensional discrete ordinates transport code developed at Oak Ridge National Laboratory (ORNL, 1995).

Multi-group neutron and photon cross-section data was provided by the BUGLE-80 (ORNL, 1980) library or BUGLE-93 (ORNL, 1994).

SNF radionuclide inventories and multiple energy group photon and neutron source terms have been calculated with ORIGEN2 (BWNT, 1991). The SCALE package, featuring ORIGEN-S (ORNL, 1993a), is being qualified for QA radiation shielding work according to the CRWMS M&O Quality Administrative Procedures and will be used in the future.

6.5.4.2 Modeling Assumptions

Unlike the *MPC Conceptual Design Report* (CRWMS M&O 1994k), which focused on loading, transportation, and storage issues, this analysis is concerned with emplacement and disposal in the MGDS. The internal area of the MPC is homogenized in the shielding model into four zones: fuel/basket, outer basket ring, spacer/support ring, and a gap ring. The applicability of this scheme was validated through use of a detailed R-theta model (CRWMS M&O, 1995aa). In the one-dimensional cases, the WP is modeled as infinitely long, and doses are conservatively overestimated. In the two-dimensional cases, axial zones are modeled for fuel, end fittings, etc., and the actual radial and axial WP dimensions are used for the bottom half of the WP (end opposite the shield plug in the MPC).

The shielding analyses reported herein were performed over a significant span of time with changing requirements and have been performed for various design basis SNF characteristics including the original MGDS PWR design basis of 4.6 percent U-235 enrichment, 58 GWd/MTU burnup, 16 years cooling time (CRWMS M&O, 1994i); the current MGDS PWR design basis of 4.2 percent U-235 enrichment, 48.086 GWd/MTU burnup, 10 years cooling time as discussed in Section 5; the MPC PWR design basis of 3.75 percent U-235 enrichment, 40 GWd/MTU, 20 years cooling time (CRWMS M&O, 1994k); the MGDS PWR average of 3.92 percent U-235 enrichment, 42.21 GWd/MTU, 22.48 years cooling time (CRWMS M&O, 1994i); the original MGDS BWR design basis of 3.59 percent U-235 enrichment, 44.5 GWd/MTU, 10 years cooling (CRWMS M&O, 1994i); the current MGDS BWR design basis of 3.74 percent U-235 enrichment, 49 GWd/MTU, 10 years cooling discussed in Section 5; and a bounding BWR set of 3.74 percent U-235 enrichment, 50 GWd/MTU, 10 years cooling (CRWMS M&O, 1994i).

6.5.4.3 Large (125-Ton) Multi-Purpose Canister

This section provides the results of the shielding analysis for the BWR and PWR configurations. The dose rates are approximately the same outside the disposal container for both configurations. Supplemental shielding is required for handling or performing operations in the proximity of the emplaced containers to meet a preliminary target dose rates of 10 and 20 mrem/hr at 2 m. The BWR WP establishes the bounding transporter shield thicknesses.

The inner and outer barrier of the disposal container are shown in Appendix B.

6.5.4.3.1 21 PWR Multi-Purpose Canister

6.5.4.3.1.1 Disposal Container Outer Barrier

Based on one-dimensional calculations, the dose rates at 2 m for MPC disposal containers with a 10 cm (3.94 inches) and a 20 cm (7.87 inches) outer barrier thicknesses are shown in Table 6.5-10 (CRWMS M&O, 1994o).

Table 6.5-10. 21 PWR Assembly Multi-Purpose Canister Disposal Container Dose Rates at 2 m - One-Dimensional Calculations

Outer Barrier Thickness (cm)	SNF Design Basis	Dose Rate (rem/hr)
10	Original MGDS	3.30
20	Original MGDS	0.44
10	Multi-Purpose Canister	1.90
20	Multi-Purpose Canister	0.23

Figures 6.5-83 and 6.5-84 illustrate the curves for dose rates at various distances from the outer barrier of 10 and 20 cm thickness, respectively. Dose rates are shown for both the MPC and original MGDS design basis fuels. The 21 PWR MPC disposal container requires approximately a 38 cm (15 inch) thick shielded outer barrier to achieve the target dose rates of 20 mrem/hr at 2 m from the surface for the MPC design basis fuel (CRWMS M&O, 1994i). A radial dose profile for a 21 PWR WP design with a shielding outer barrier thickness of 38 cm is shown in Figure 6.5-85 (CRWMS M&O, 1994i). Based on the attenuation indicated in Figure 6.5-86, an additional outer barrier thickness of 3 to 4 cm is required to achieve the target dose rate with the original MGDS design basis SNF.

Based on two-dimensional calculations, the dose rates at the WP surface and at 2 m for the 21 PWR assembly MPC disposal containers with 10 cm (3.94 inches) outer barrier thicknesses are shown in Table 6.5-11 (CRWMS M&O, 1996f) for the current design basis fuel. Figures 6.5-86 and 6.5-87 illustrate the curves for dose rates at various distances from the axial and radial centerline of the container for the PWR package.

Table 6.5-11. 21 PWR Assembly Multi-Purpose Canister Disposal Container Dose Rates - Two-Dimensional Calculations

Axis	Dose Rate (rem/hr)	
	Surface	2 m
Radial	6.5	2.5
Axial	12.2	2.8

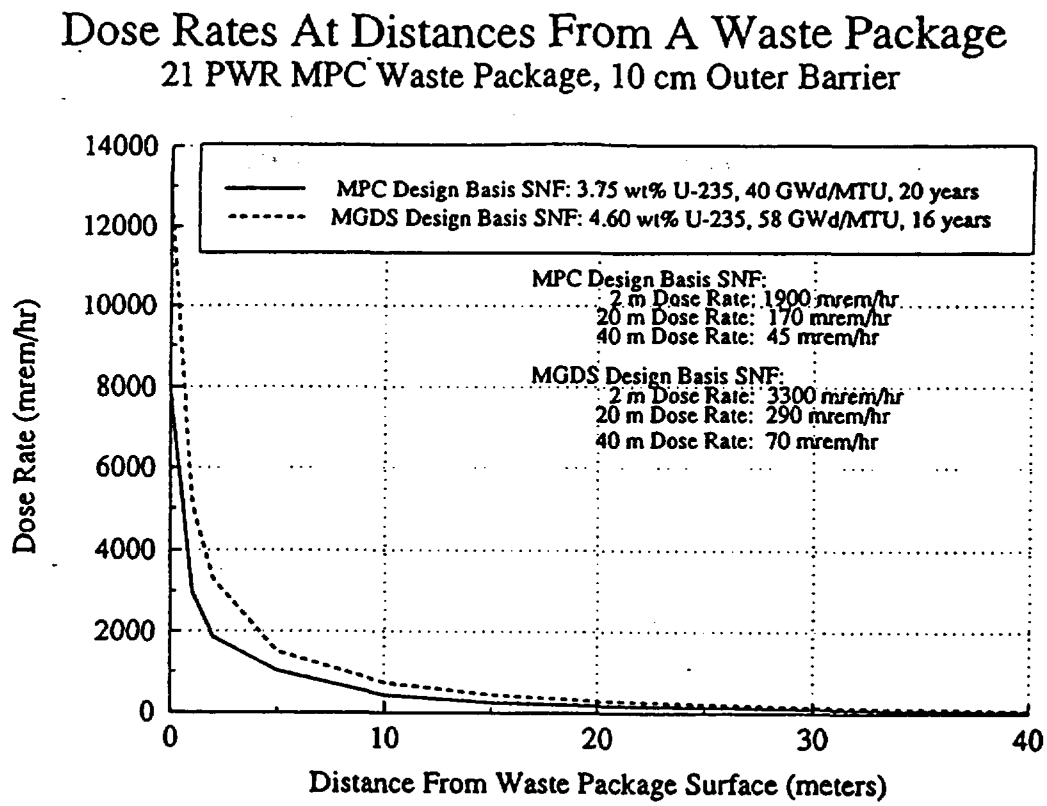

Figure 6.5-83. Dose Rates Away from 10 cm Outer Barrier, 21 PWR Multi-Purpose Canister Waste Package

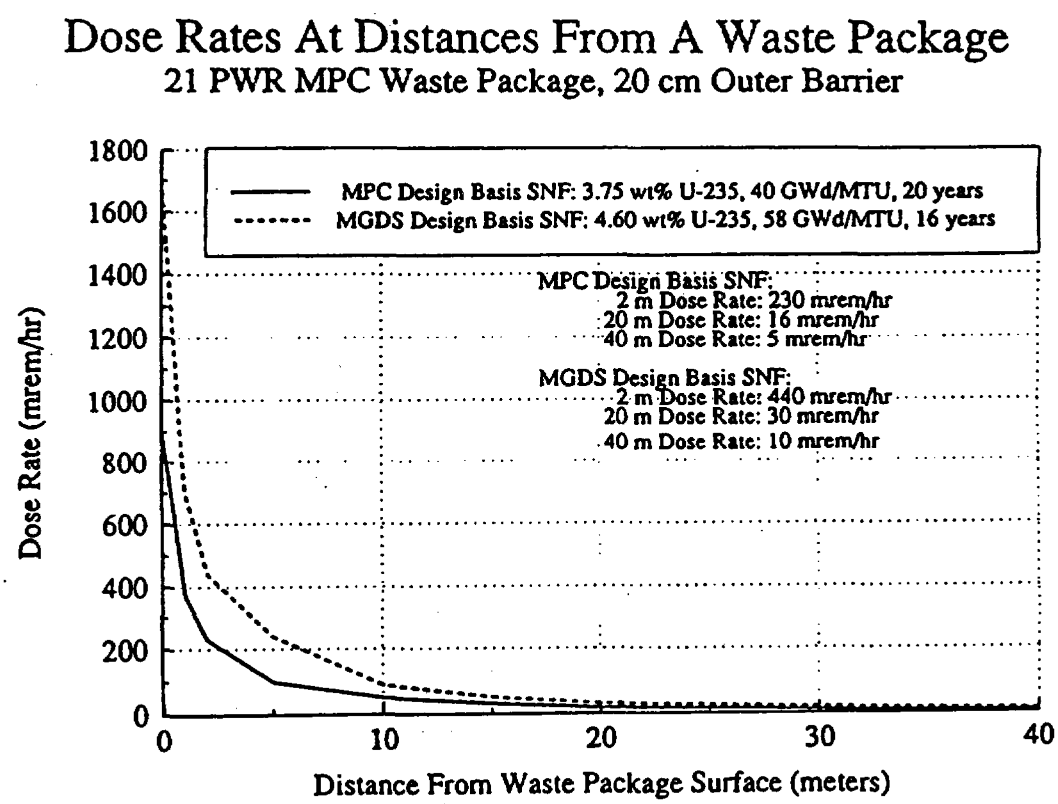

Figure 6.5-84. Dose Rates Away from 20 cm Outer Barrier, 21 PWR Multi-Purpose Canister Waste Package

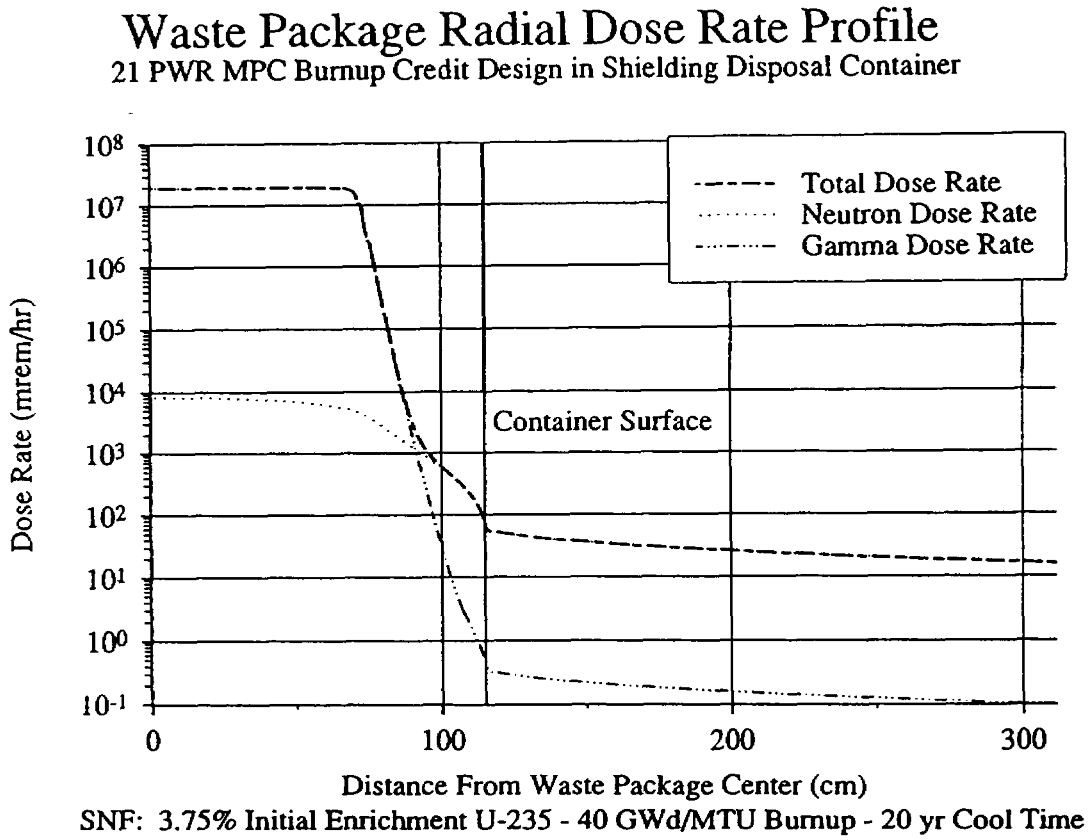

Figure 6.5-85. Multi-Purpose Canister 21 PWR Waste Package Radial Dose Rate Profile, Multi-Purpose Canister Design Basis Fuel

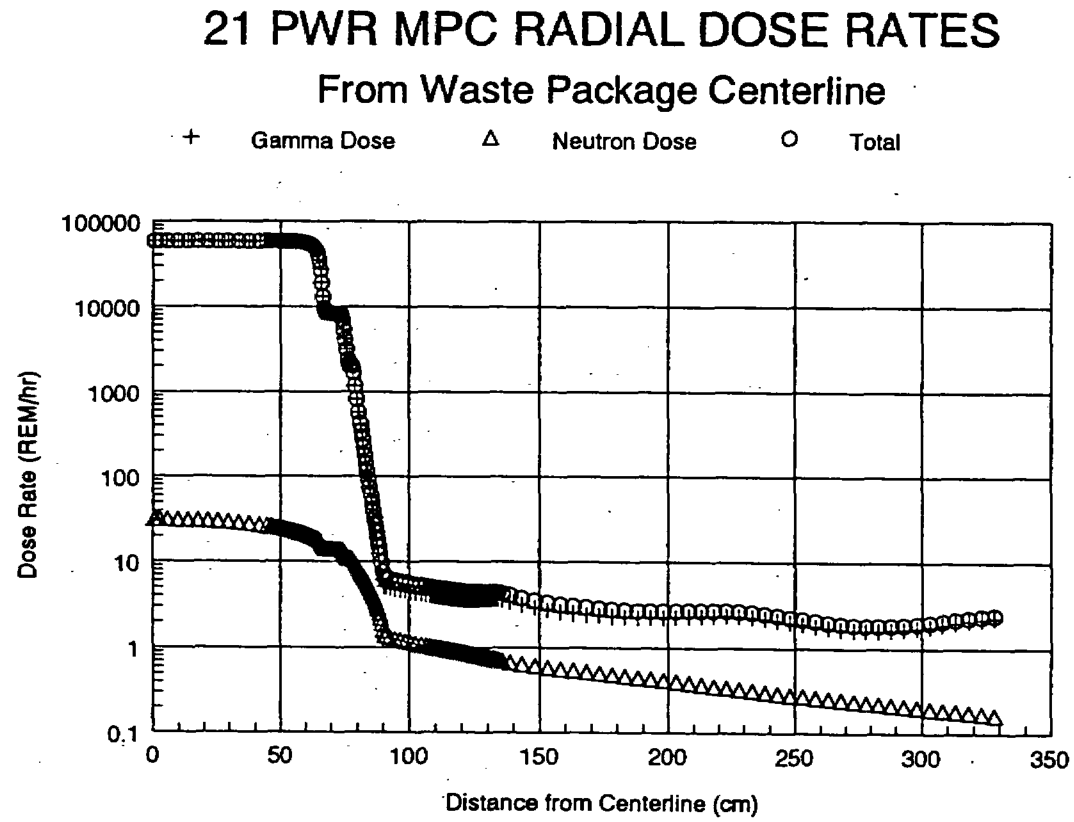

Figure 6.5-86. Multi-Purpose Canister 21 PWR Waste Package Radial Dose Rate Profile, MGDS Design Basis Fuel - Two-Dimension

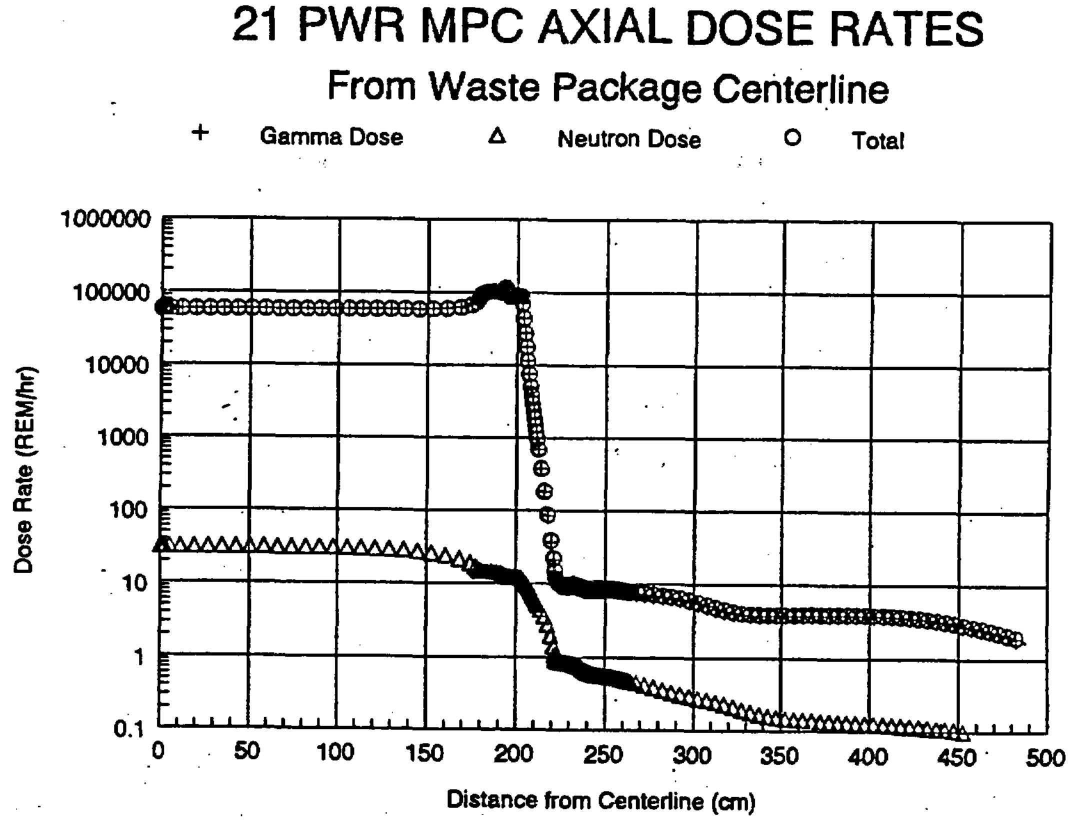

Figure 6.5-87. Multi-Purpose Canister 21 PWR Waste Package Axial Dose Rate Profile, MGDS Design Basis Fuel - Two-Dimension

6.5.4.3.1.2 Emplacement Transport Shield

The emplacement transport shield strategy places the necessary shielding on the emplacement transporter. This transporter would have sufficient shielding to limit the dose rate to an individual to the target levels. Once emplaced underground, the WP would not have sufficient intrinsic shielding to allow unprotected human access to the emplacement drifts.

The conceptual design of the WP emplacement transport shield is as follows: A standard three-layer, or sandwich, approach to shielding was used, with a layer of neutron shielding material between layers of gamma shielding material. The gamma shield, consisting of a dense material, should be close to the package to reduce the shield weight. Secondary gamma rays from (n,γ) reactions in the neutron shield require a gamma shield outside the neutron shield. These shield material layers can be arranged in various configurations (e.g., cylinder, rectangle, and half-cylinder). A sketch of the WP and transport shield for the cylindrical arrangement is shown in Figure 6.5-88.

The initial materials chosen for the transport shield model were 5 percent boron-polyethylene for the neutron shield and lead for the gamma shields. These materials were chosen because they are commonly used in the nuclear industry and their use is well proven as shielding materials. The shielding layers are encased in a thin (0.5 cm) outer layer of 316L stainless steel to provide minimal structural support for the shields and containment for the lead. A 30 cm air gap between the WP and the transport shield for handling equipment and additional structural support was included radially. A 20 cm air gap between the WP and the transport shield was included axially.

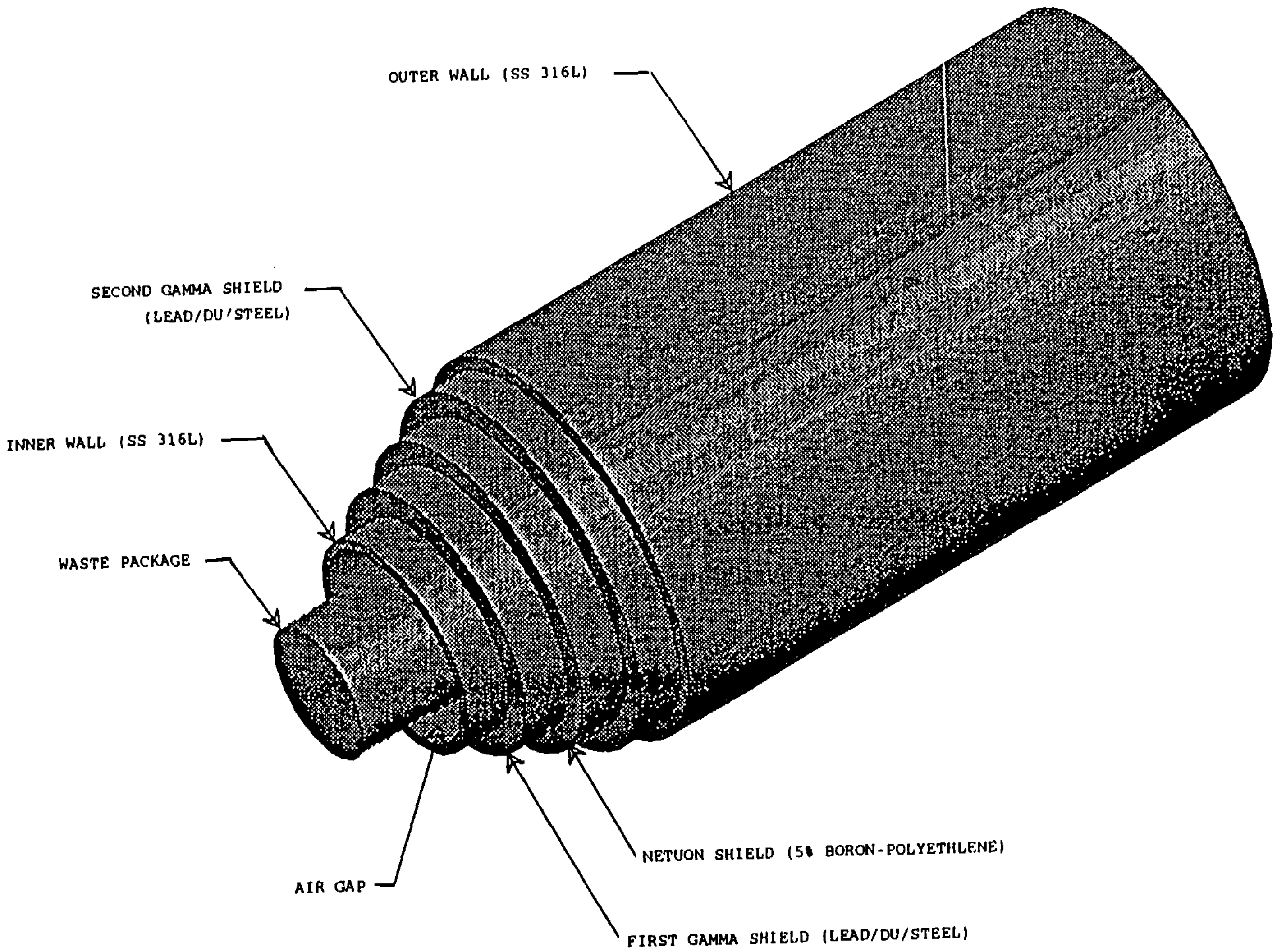

Figure 6.5-88. Conceptual Waste Package and Transport Shield Arrangement

Two-dimensional calculations were performed to determine the minimum shielding thickness in both the axial and radial directions for the transport shield (CRWMS M&O, 1996f). The radial transport shield thicknesses required to meet the target dose rate of 10 mrem/hr at 2 m for a 21 PWR assembly WP with the current shielding design basis fuel, are 5.5 cm (2.17 inches) of lead for the first gamma shield, 9 cm (3.54 inches) of 5 weight percent boron-polyethylene for the neutron shield, and 1 cm (0.39 inches) of lead for the second gamma shield. In the axial direction the thicknesses are 7 cm (2.76 inches) of lead for the first gamma shield, 7 cm (2.76 inches) of 5 weight percent boron-polyethylene for the neutron shield, and 1 cm (0.39 inches) of lead for the second gamma shield. The transport shield weight is 57 tonnes (63 tons).

6.5.4.3.1.3 Shielding Sleeve

A shielding sleeve is an alternative shielding concept and is presented here for reference only. The WP shielding sleeve strategy places a shielding sleeve around the disposal barriers/container of the standard unshielded WP. The shielding sleeve is not part of the containment barriers and so does not have the long-term material performance requirement that the containment barriers have. Without the long-term material performance requirements, the shield sleeve can be made out of standard shielding materials. This strategy would provide radiological protection for workers during emplacement and performance confirmation.

The SNF neutron source component tends to dominate the selection of the WP shielding sleeve material. When sufficient neutron shielding material is placed in the sleeve, no additional gamma-ray shielding material is needed. Based on one-dimensional calculations, the 21 PWR assembly MPC WP requires about a 42 cm (16½ inch) thick concrete shielding sleeve to achieve the target dose rate of 10 mrem/hr at 2 m carrying the original MGDS design basis fuel (CRWMS M&O, 1994i). The 2 m dose rates for both the MGDS average SNF and the MPC design basis SNF for the above configuration is 4 mrem/hr (CRWMS M&O, 1994i). The radial dose profiles for a 21 PWR MPC conceptual design WP in a shielding sleeve with the MGDS design basis SNF and the MPC design basis SNF are shown in Figure 6.5-89 (CRWMS M&O, 1994o).

The discussion of shielding sleeve thickness above does not take into account the environment in which the sleeve is to be placed. The hot dry environment envisioned in many of the repository thermal scenarios will drive the water out of the concrete shield material, lowering its effectiveness. The 42-cm concrete shielding sleeve, with its water removed, would have a 2-m dose rate of approximately 25 mrem/hr instead of the 10 mrem/hr it provided initially. To prevent loss of water from the shield, the concrete would need to be encapsulated in steel. A 1 cm thick steel wall around the concrete is sufficient to retain the water. The encapsulated concrete shielding sleeve (40 cm of concrete and 2 cm of carbon steel) provides shielding approximately equivalent to the 42 cm thick all concrete shield (with water) (CRWMS M&O, 1994o).

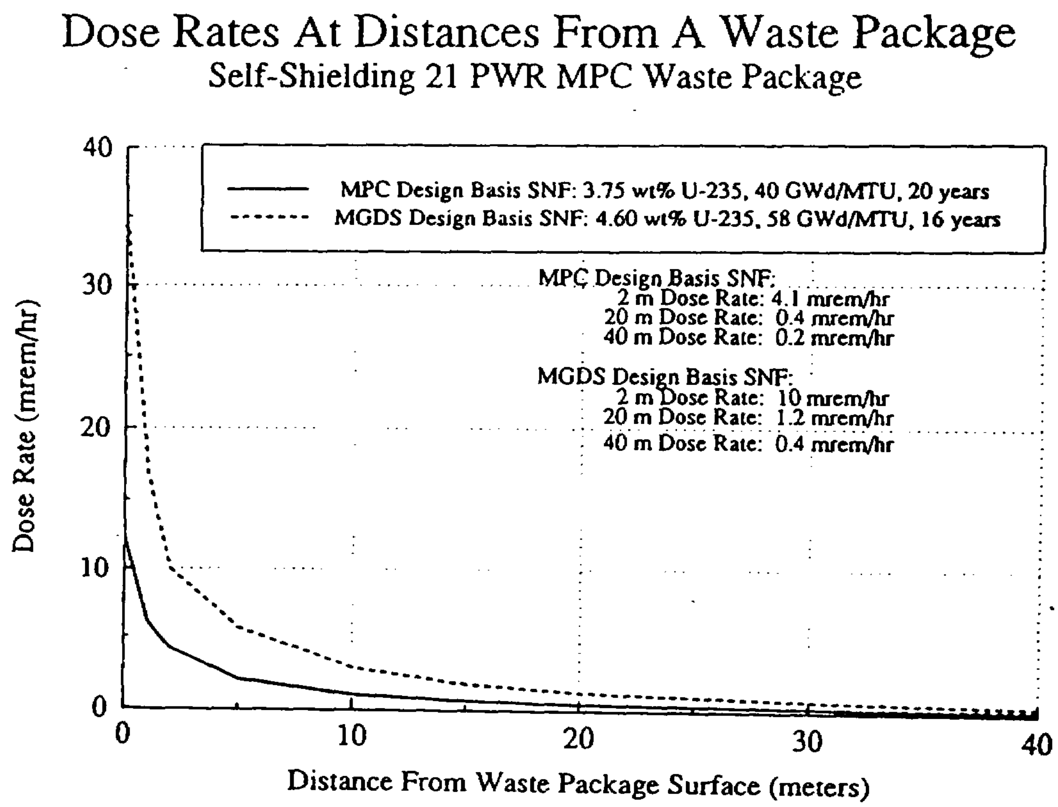

Figure 6.5-89. Dose Rates Outside of Shielding Sleeve, 21 PWR Multi-Purpose Canister Waste Package

6.5.4.3.2 40 BWR Multi-Purpose Canister

6.5.4.3.2.1 Disposal Container Outer Barrier

Based on one-dimensional calculations, the dose rates at 2 m for a 40 BWR assembly MPC disposal container with outer barrier thicknesses of 10 and 20 cm are shown in Table 6.5-12 (CRWMS M&O, 1994o). Figures 6.5-90 and 6.5-91 illustrate the curves for dose rates at various distances from the outer barrier at 10 and 20 cm thickness, respectively for the original MGDS DBF. As indicated previously, the 21 PWR MPC disposal container requires about a 42 cm (15 inch) thick shielded outer barrier to achieve the target dose rates of 20 mrem/hr at 2 m from the surface for the original MGDS DBF. A similar outer barrier thickness would be required to achieve the target dose rate with the MGDS BWR design basis SNF.

Table 6.5-12. 40 BWR Assembly Multi-Purpose Canister Disposal Container Dose Rates at 2 m - One-Dimensional Calculations

Outer Barrier Thickness (cm)	SNF Design Basis	Dose Rate (rem/hr)
10	Original MGDS	3.60
20	Original MGDS	0.48

Based on two-dimensional calculations, the dose rates at the WP surface and at 2 m for the 40 BWR assembly MPC disposal containers with 10 cm (3.94 inches) outer barrier thicknesses are shown in

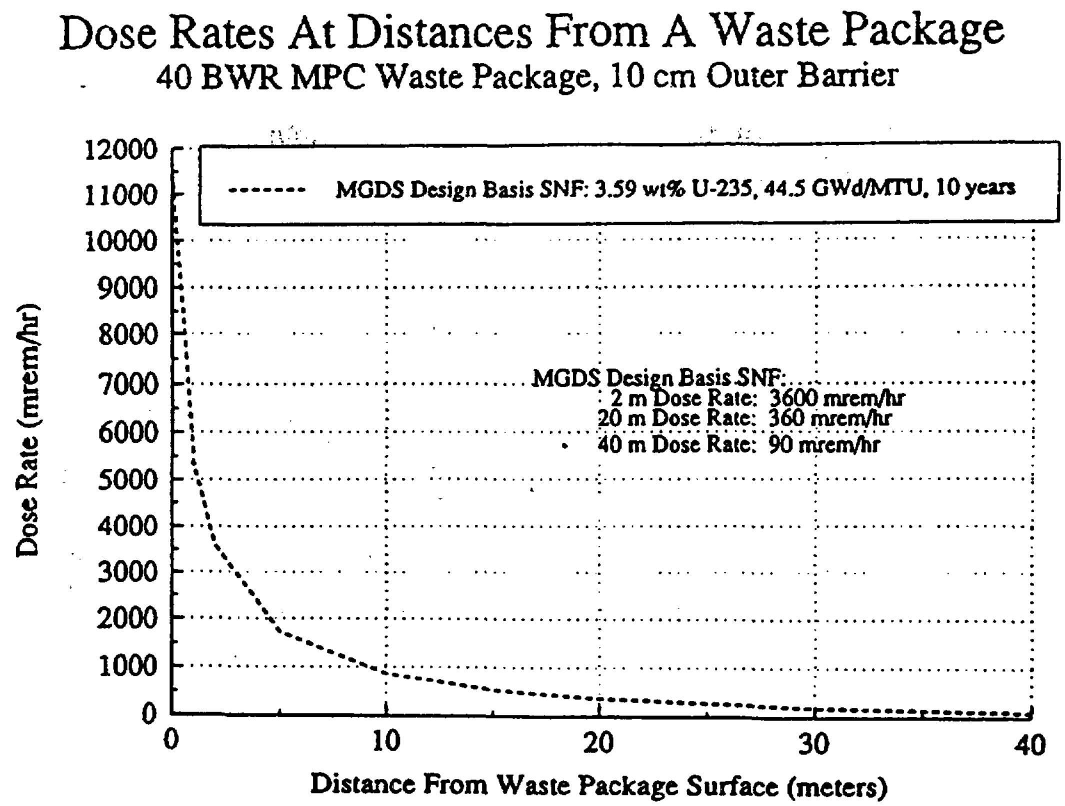

Figure 6.5-90. Dose Rates Away from 10 cm Outer Barrier, 40 BWR Multi-Purpose Canister Waste Package, MGDS Design Basis Fuel

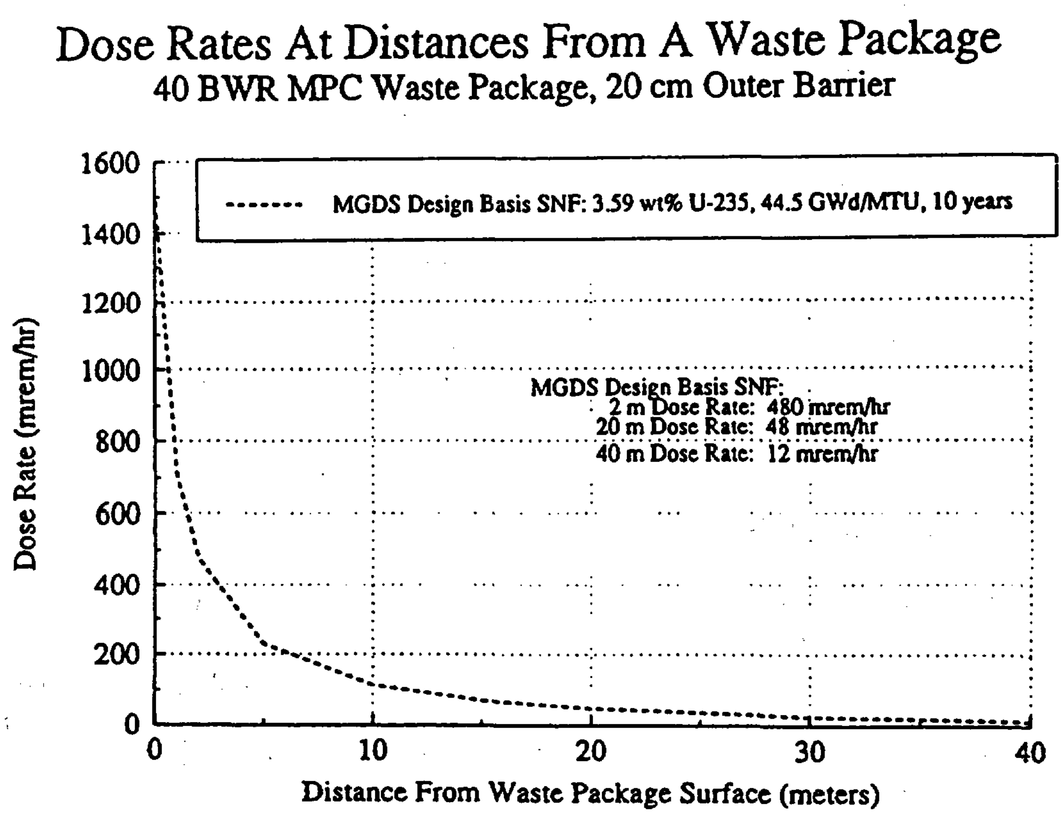

Figure 6.5-91. Dose Rates Away from 20 cm Outer Barrier, 40 BWR Multi-Purpose Canister Waste Package, MGDS Design Basis Fuel

Table 6.5-13 (CRWMS M&O 1996f) for the current DBF. Figures 6.5-92 and 6.5-93 illustrate the curves for dose rates at various distances from the axial and radial centerline of the container for the BWR package.

Table 6.5-13. 40 Boiling Water Reactor Assembly Multi-Purpose Canister Disposal Container Dose Rates - Two Dimensional Calculations

Axis	Dose Rate (rem/hr)	
	Surface	2 m
Radial	7.5	2.4
Axial	29.1	5.4

6.5.4.3.2.2 Emplacement Transport Shield

Two-dimensional calculations were performed to determine the minimum shielding thickness in both the axial and radial directions for the transport shield (CRWMS M&O 1996f). The radial transport shield thicknesses required to meet the target dose rate of 10 mrem/hr at 2 m for a 40 BWR assembly WP with the current shielding DBF, are 6 cm (2.36 inches) of lead for the first gamma shield, 9 cm (3.54 inches) of 5 percent boron-polyethylene for the neutron shield, and 1 cm (0.39 inches) of lead for the second gamma shield. In the axial direction the thicknesses are 8 cm (3.15 inches) of lead for the first gamma shield, 7 cm (2.76 inches) of 5 percent boron-polyethylene for the neutron shield, and 1 cm of lead for the second gamma shield. The transport shield weight is 61 tonnes (67 tons).

6.5.4.3.2.3 Shielding Sleeve

Like the 21 PWR assembly MPC WP, the 40 BWR assembly MPC WP will require about a 42-cm (16½-inch) thick concrete shielding sleeve to achieve the target dose rate of 10 mrem/hr at 2 m carrying the MGDS DBF (CRWMS M&O 1994o). With the bounding BWR fuel (similar to current DBF), the dose rate will be 17 mrem/hr for the same shield. A radial dose profile for a 40 BWR MPC conceptual design WP in a shield sleeve with the MGDS design basis SNF is shown in Figure 6.5-94 (CRWMS M&O 1994o).

6.5.4.4 Small (75-Ton) Multi-Purpose Canister

This section provides the results of the shielding analysis for the BWR and PWR configurations for the small (75-ton) MPC. The dose rates are approximately the same outside the disposal container for both configurations. Supplemental shielding is required for handling or for performing operations in the proximity of the emplaced containers to meet a preliminary target dose rate of 20 mrem/hr at 2 m. The dose rates for the 75-ton MPC disposal container are approximately 20 percent lower than those for the 125-ton container.

The inner and outer barrier are shown in Appendix B.

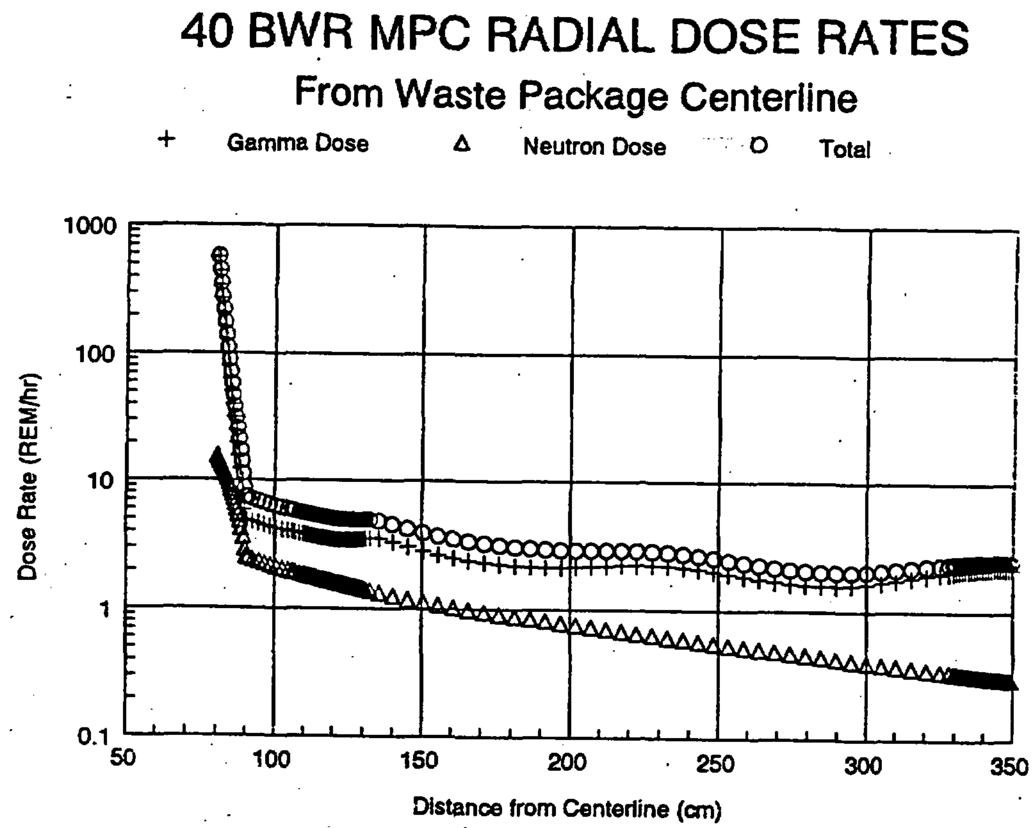

Figure 6.5-92. Multi-Purpose Canister 40 BWR Waste Package Radial Dose Rate Profile, MGDS Design Basis Fuel - Two-Dimension

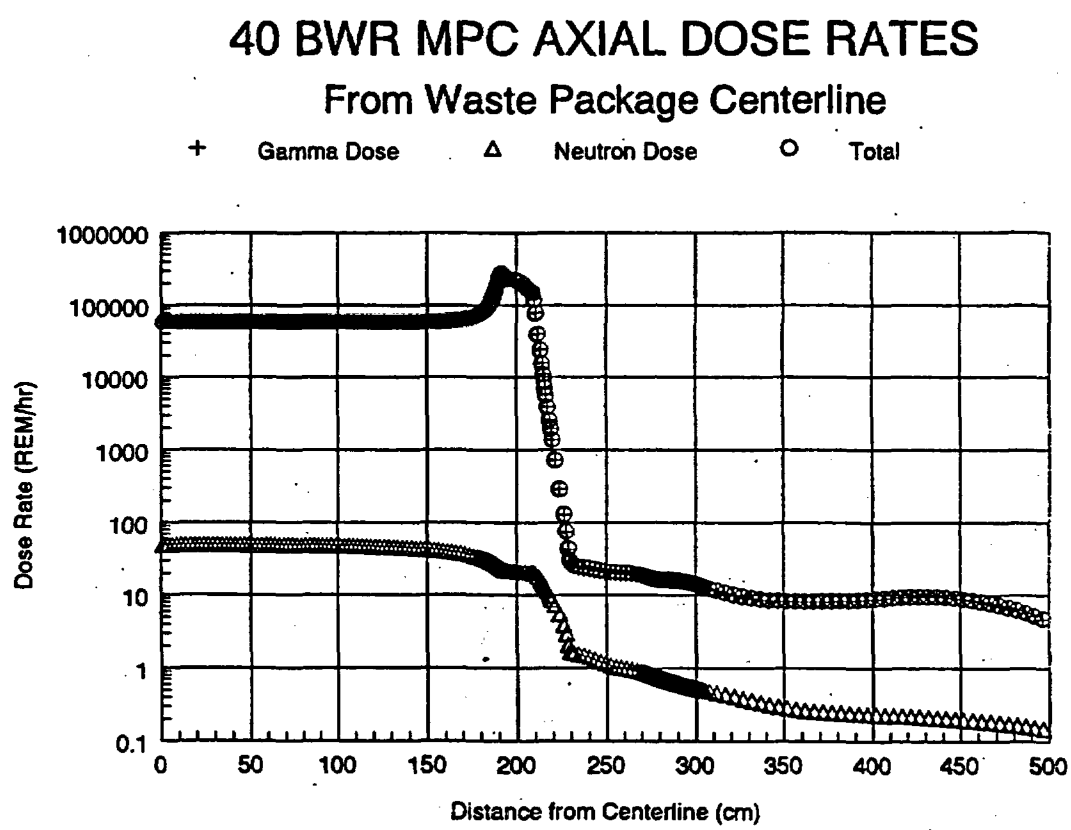

Figure 6.5-93. Multi-Purpose Canister 40 BWR Waste Package Axial Dose Rate Profile, MGDS Design Basis Fuel - Two-Dimension

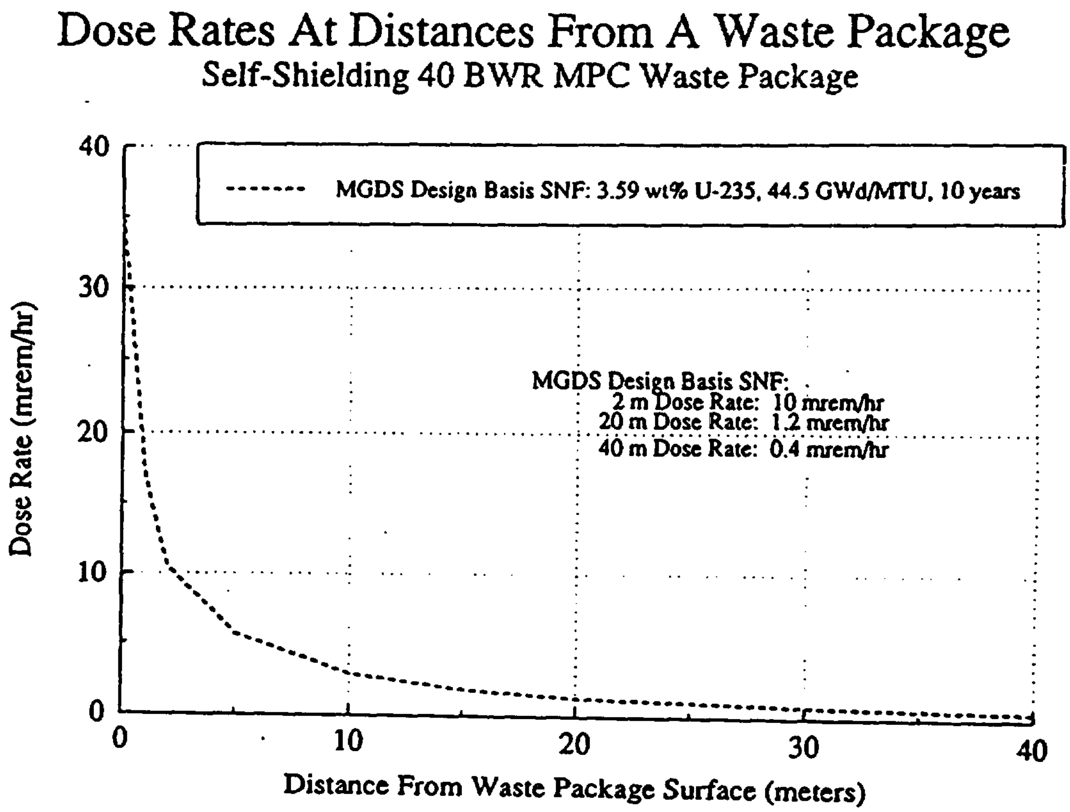

Figure 6.5-94. Dose Rates Outside of Shielding Sleeve, 40 BWR Multi-Purpose Canister Waste Package

6.5.4.4.1 12 PWR Multi-Purpose Canister

6.5.4.4.1.1 Disposal Container Outer Barrier

Based on one-dimensional calculations, the dose rates at 2 m for 12 PWR assembly MPC disposal containers with 10 and 20 cm outer barrier thicknesses are shown in Table 6.5-14 (CRWMS M&O, 1994o). Figures 6.5-95 and 6.5-96 illustrate the curves for dose rates at various distances from the outer barrier at 10 and 20 cm thickness, respectively. Dose rates are shown for the original MGDS and MPC DBFs. The 21 PWR MPC disposal container requires an approximately 38-cm (15-inch)

Table 6.5-14. 12 PWR Assembly Multi-Purpose Canister Disposal Container Dose Rates at 2 m - One-Dimensional Calculations

Outer Barrier Thickness (cm)	SNF Design Basis	Dose Rate (rem/hr)
10	Original MGDS	2.60
20	Original MGDS	0.38
10	Multi-Purpose Canister	1.60
20	Multi-Purpose Canister	0.20

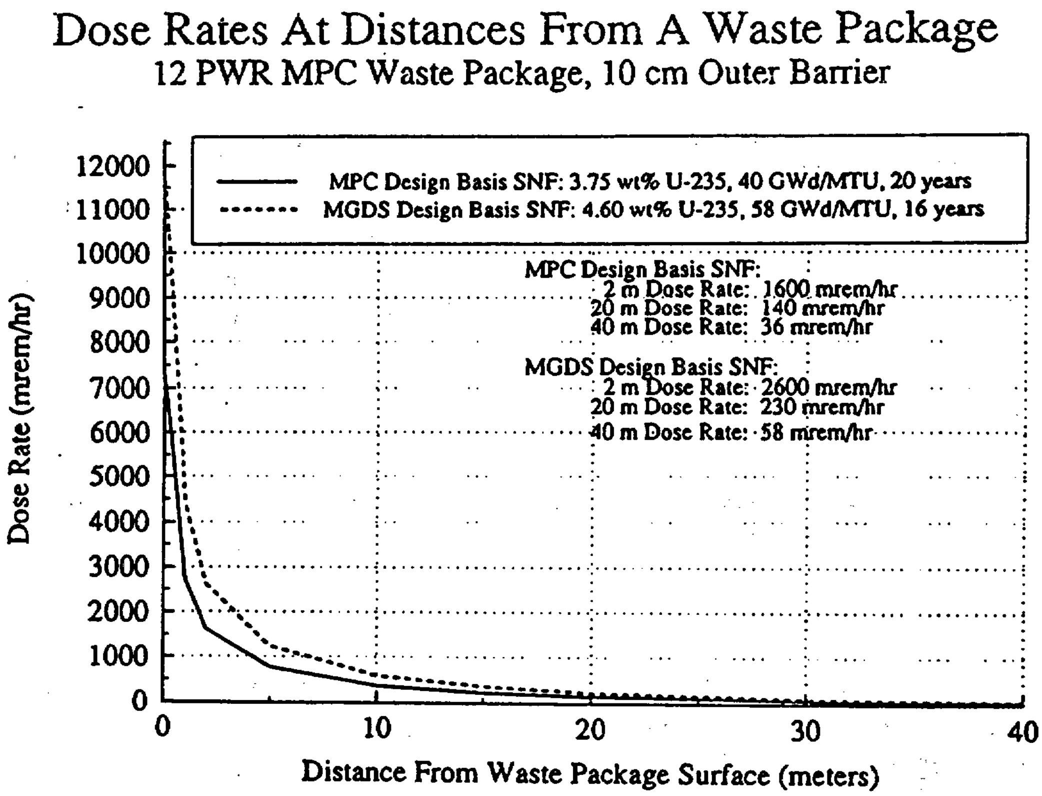

Figure 6.5-95. Dose Rates Away from 10 cm Outer Barrier 12 PWR Multi-Purpose Canister Waste Package, MGDS and Multi-Purpose Canister Design Basis Fuel

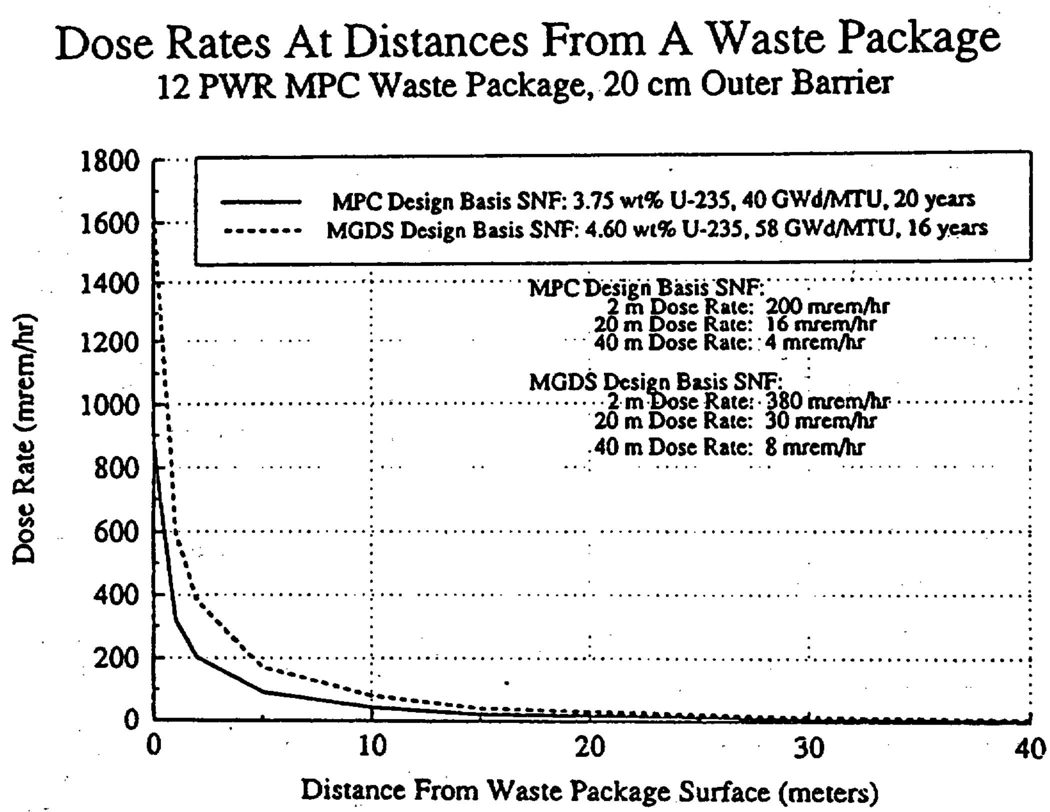

Figure 6.5-96. Dose Rates Away from 20 cm Outer Barrier, 12 PWR Multi-Purpose Canister Waste Package, MGDS and Multi-Purpose Canister Design Basis Fuel

thick shielded outer barrier to achieve the target dose rates of 20 mrem/hr at 2 m from the surface for the MPC DBF (CRWMS M&O, 1994i). Based on comparison of the unshielded dose rates at 2 m, the outer barrier thickness required to achieve the target dose rate with the 12 PWR MPC disposal container would be less. The 12 PWR MPC dose rate is 21 percent less than that for the 21 PWR MPC; the thickness saving is approximately 1.5 cm based on the attenuation indicated in Figure 6.5-85.

6.5.4.4.1.2 Emplacement Transport Shield

The transport shield thicknesses required to meet the target dose rate for the 12 assembly PWR MPC disposal container will be less than that for the 21 assembly PWR case.

6.5.4.4.1.3 Shielding Sleeve

Like the 21 PWR assembly MPC WP, the 12 PWR assembly MPC WP requires an approximately 42 cm (16½ inch) thick concrete shielding sleeve to achieve the target dose rate of 10 mrem/hr at 2 m carrying the original MGDS DBF. The dose rate for both the MGDS average SNF and the MPC design basis SNF for the above configuration is 4 mrem/hr (CRWMS M&O, 1994o). A radial dose profile for a 12 PWR MPC conceptual design WP in a shielding sleeve with the MGDS design basis SNF is shown in Figure 6.5-97 (CRWMS M&O, 1994o).

6.5.4.4.2 24 BWR Multi-Purpose Canister

6.5.4.4.2.1 Disposal Container Outer Barrier

Based on one-dimensional calculations, the dose rates at 2 m for 24 BWR assembly MPC disposal containers with 10 and 20 cm outer barrier thicknesses are shown in Table 6.5-15 (CRWMS M&O, 1994o). Figures 6.5-98 and 6.5-99 illustrate the curves for dose rates at various distances from the outer barrier for 10 and 20 cm thicknesses, respectively. Dose rates are shown for the MGDS DBF. The 21 PWR MPC disposal container requires approximately a 38 cm (15-inch) thick shielded outer barrier to achieve the target dose rates of 20 mrem/hr at 2 m from the surface for the MPC DBF (CRWMS M&O, 1994i). Like the 12 PWR assembly WP design, which has similar unshielded dose rates at 2 m, the outer barrier thickness required for the 24 BWR assembly MPC design will be lower.

Table 6.5-15. 24 BWR Assembly Multi-Purpose Canister Disposal Container Dose Rates at 2 m - One-Dimensional Calculations

Outer Barrier Thickness (cm)	SNF Design Basis	Dose Rate (mrem/hr)
10	Original MGDS	2.40
20	Original MGDS	0.34

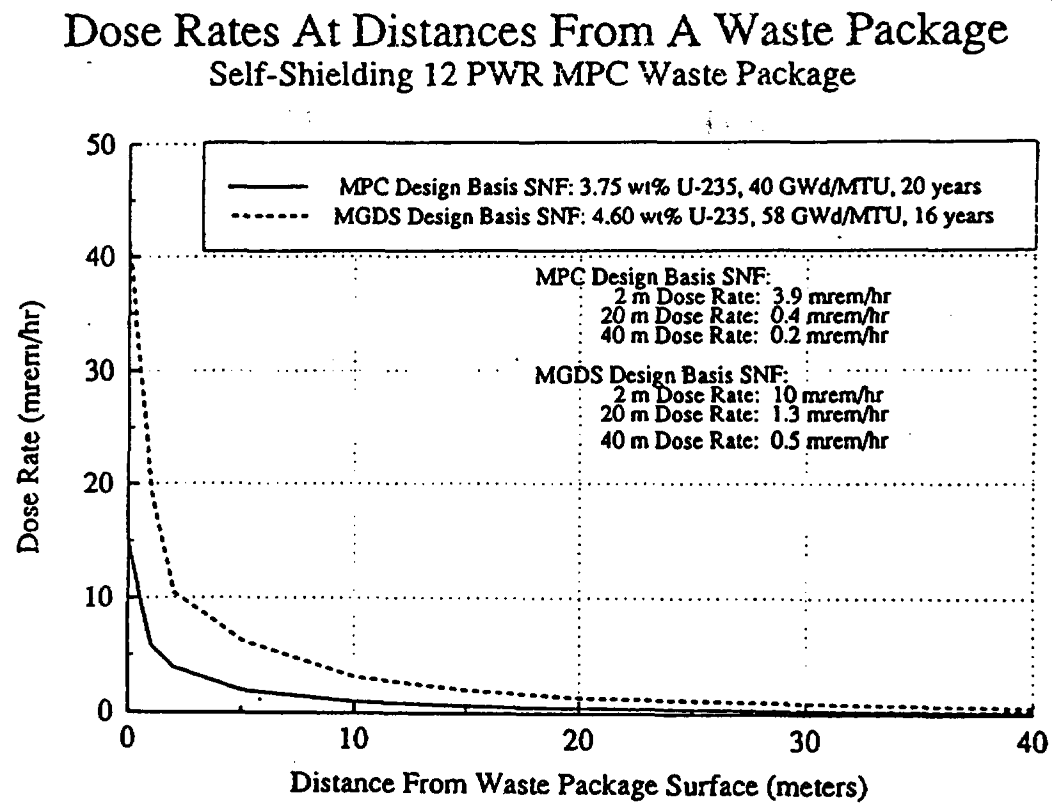

Figure 6.5-97. Dose Rates Outside of Shielding Sleeve, 12 PWR Multi-Purpose Canister Waste Package

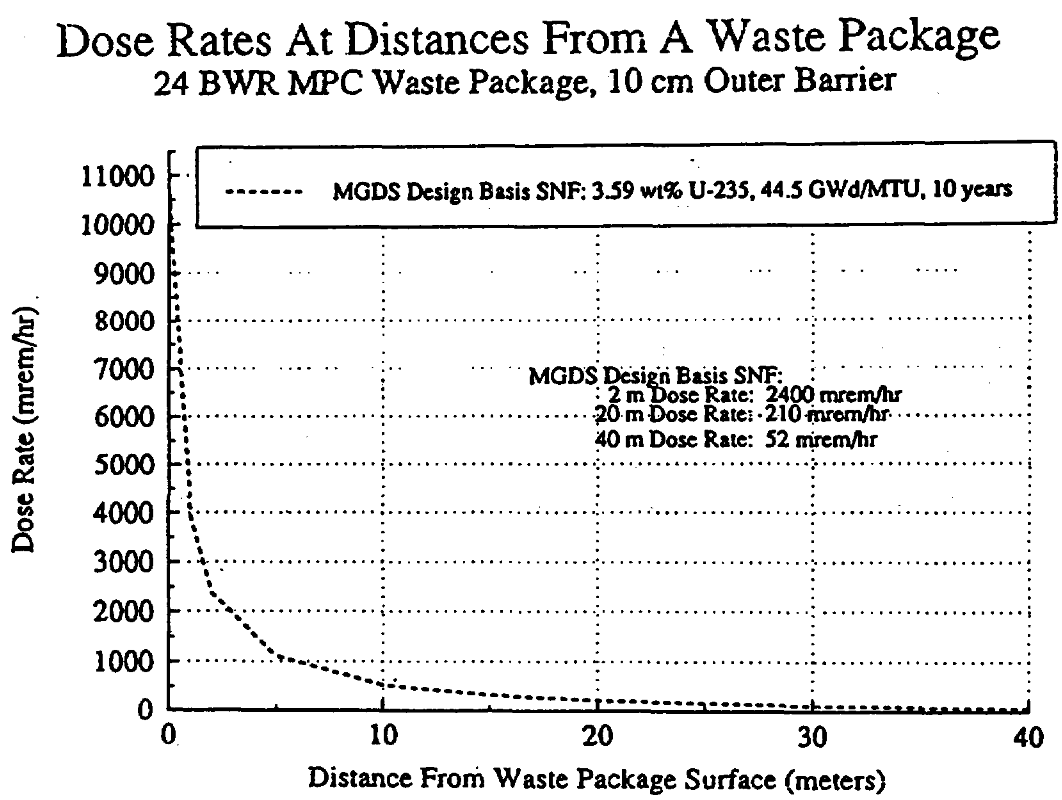

Figure 6.5-98. Dose Rates Away from 10 cm Outer Barrier, 24 BWR Multi-Purpose Canister Waste Package, MGDS Design Basis Fuel

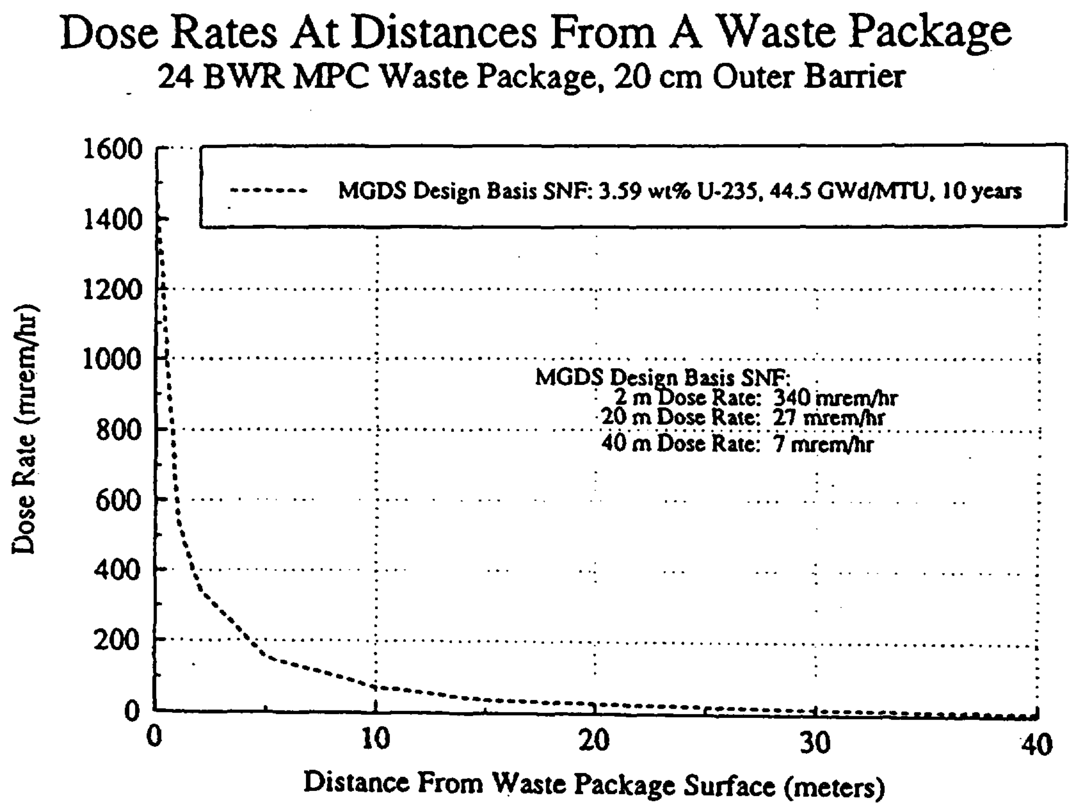

Figure 6.5-99. Dose Rates Away from 20 cm Outer Barrier 24 BWR Multi-Purpose Canister Waste Package, MGDS Design Basis Fuel

6.5.4.4.2.2 Emplacement Transport Shield

The transport shield thicknesses required to meet the target dose rate for the 24 assembly BWR MPC disposal container will be less than that for the 21 assembly PWR case.

6.5.4.4.2.3 Shielding Sleeve

Like the 21 PWR assembly MPC WP, the 24 BWR assembly MPC WP requires an approximately 42 cm (16½ inch) thick concrete shielding sleeve to achieve the target dose rate of 10 mrem/hr at 2 m carrying the MGDS DBF (CRWMS M&O, 1994o). With the bounding BWR fuel, the dose rate will be 15 mrem/hr for the same shield. A radial dose profile for a 24 BWR MPC conceptual design WP in a shielding sleeve with the MGDS design basis SNF is shown in Figure 6.5-100 (CRWMS M&O, 1994o).

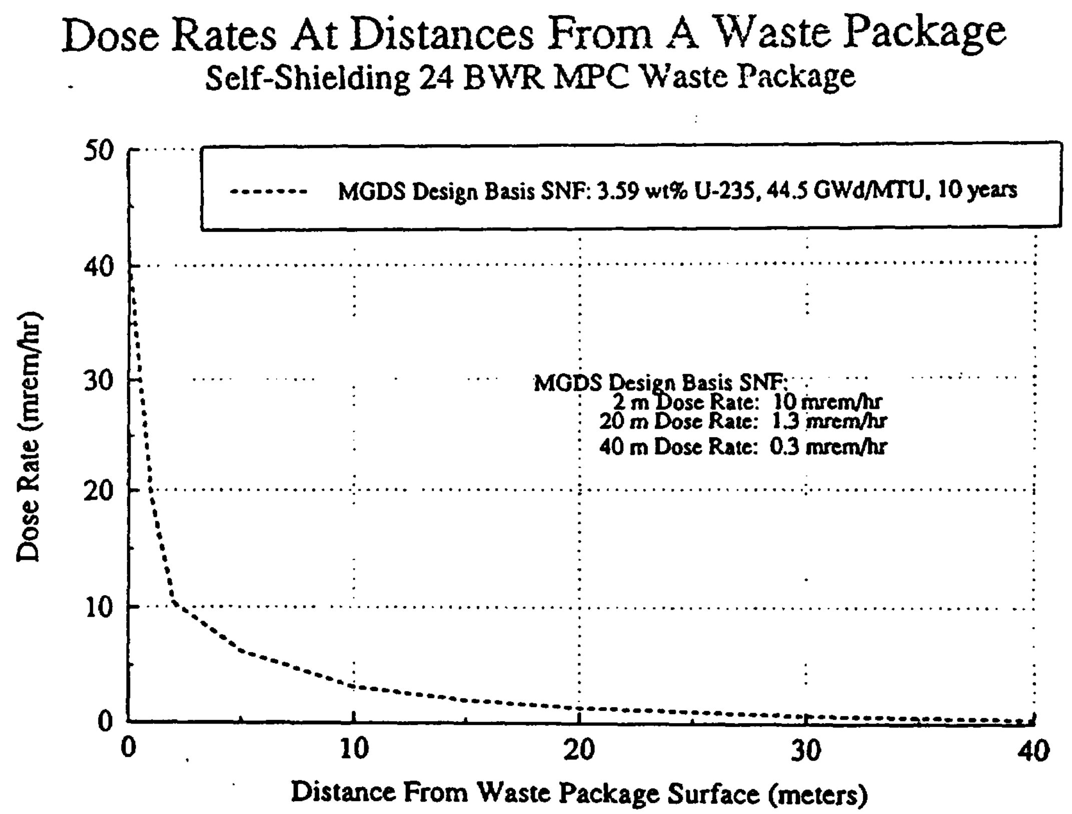

Figure 6.5-100. Dose Rates Outside of Shielding Sleeve, 24 BWR Multi-Purpose Canister Waste Package

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7. PROBABILISTIC EVALUATION AND DESIGN BASIS EVENTS

7.1 INTRODUCTION

Probabilistic evaluation involves the estimation of the probability that an undesirable event will occur by (1) defining the various sequences of failures (or processes) that can produce the event, (2) estimating the probability of each failure (or process), and (3) combining the individual probabilities to determine the probability of each sequence that may lead to the final, undesirable event. When combined with an assessment of the consequences of the undesirable event, a determination of "risk" can be made. This assessment of probability and consequences is typically referred to as Probabilistic Risk Assessment. Both probabilistic evaluation and Probabilistic Risk Assessment are used throughout industry, particularly in the nuclear power generation, for evaluating the importance of plant systems and components, identifying component failures or failure sequences that are dominant contributors to severe accident likelihood and/or public risk, and guiding design change and maintenance decisions to minimize the probability of these failures and risks.

In the Waste Package (WP) design process, probabilistic evaluation techniques are being used to evaluate design options primarily by estimating the likelihood of undesirable events. For this application, the probabilistic methods are appropriate and effective analysis tools because of:

- The degree of uncertainty in many of the design inputs.
- The complexity and number of possible design conditions resulting from these uncertainties.
- The need to design to realistic, rather than incredible (i.e., ultra-conservative) conditions.
- The need to assess vulnerabilities in the current conceptual design and identify possible improvements.

Furthermore, probabilistic evacuations will provide a part of the basis for license application by complementing the use of deterministic methods for demonstrating that certain design requirements applicable to the WP have been met (NRC 1995).

Section 7.2 will briefly summarize the methodology and application of probabilistic evaluation for screening the list of preclosure Mined Geologic Disposal System (MGDS) design basis events (which is described in more detail in Section 10.1 of Volume II of this document) to identify WP design basis events. These are the set of off-normal events which the waste package design is intended to withstand without suffering a significant reduction in performance. Section 7.3 describes the postclosure probabilistic evaluation methodology. Section 7.4 applies the postclosure methodology to waste package internal criticality, which is a summary of work reported in more detail in referenced documents. Section 7.5 is a brief discussion of how the methodology will be applied to postclosure external criticality.

7.2 PROBABILISTIC EVALUATION FOR PRECLOSURE

7.2.1 WP Design Basis Events

This section summarizes methods used and the resulting list of WP Design Basis Events (DBEs) for use in the next phase of WP design. These methods were developed based on a review of the Mined Geologic Disposal System Advanced Conceptual Design (MGDS ACD) design. The detailed evaluation, with more complete descriptions of the WP DBEs and preliminary estimates of their frequency of occurrence is given in the *Waste Package Off-Normal and Accident Scenario Report* (CRWMS M&O 1996n).

7.2.1.1 Analysis Methodology

As discussed in Section 10.1 of Volume II, a preliminary list of potential MGDS DBEs has been developed. This preliminary list of DBEs has been systematically developed from a *Preliminary Hazards Analysis* (CRWMS M&O 1996m), followed by a screening out of events that are not applicable to Yucca Mountain or to the preclosure phase of the project, and finishing with a preliminary evaluation of the remaining events with respect to event credibility, likelihood of the event causing a radiological release, and whether the event is limiting for its type. This section will discuss the subset of these potential MGDS DBEs which may have significant impact to the performance of the waste package.

The methodology used for WP DBE development is essentially the same as that which was used for the MGDS DBE analysis presented in Volume II Section 10.1.3.1, except as noted here. The *Preliminary Hazards Analysis* for the surface and subsurface facilities (CRMWS M&O 1996m) provided the initial list of hazards to be considered by reviewing hazards lists developed for an earlier *Repository Conceptual Design* (SNL 1992b; SNL 1987b) and assessing their applicability to the current MGDS ACD design. In addition to identifying internal events, the *Preliminary Hazards Analysis* performed a pre-screening of the external event list, and screened out certain events in accordance with the following two criteria:

1. Events that are not applicable to the Yucca Mountain Site, and

2. Events that are not applicable to the preclosure phase of the Yucca Mountain Project.

Events that are eliminated in this manner are listed in the notes for Table 10.1-1 in Volume II.

For the identification of WP DBEs, a third criterion will also be used to screen external hazards which was not used in Volume II Section 10.1 for the MGDS in general. A hazard not screened by the above two criteria may be eliminated from consideration in the WP design because the MGDS Surface and/or Sub-Surface facilities are designed such that the hazard would have no adverse effect on the WP (e.g., the Waste Handling Building and transporter design should protect WP from design basis hazards such as tornado/extreme wind generated missiles or lightning). Final verification of the adequacy of such an assumption will be made as the MGDS design matures and further analysis of DBEs is performed.

Events involving the WP which were not eliminated by the above screening are reviewed and grouped into the following categories based on their general effect on the WP:

1. Falling objects
2. WP drop events
3. Collision/crushing events
4. Missile hazards
5. Seismic hazards
6. Internal pressurization
7. Thermal hazards
8. Chemical contamination
9. Flooding.

These categories intentionally differ from those used in the MGDS DBE analysis in Volume II Section 10.1, where internal and external events were grouped according to location (surface or subsurface). As the WP is the primary item of protection for this evaluation, the categories selected facilitate a more detailed definition of the type and magnitude of potential loads on the WP, including some which are unique to the WP (e.g., internal pressurization) and are not part of the MGDS list in Volume II, Section 10.1. The location of the event only becomes important if the performance of the WP is adversely affected (outside vs. inside Waste Handling Building or Subsurface). The event numbering system defined in Volume II Section 10.1 has been maintained to allow easy cross-referencing between the two lists.

Finally, WP related initiating events not eliminated during the above process are subjected to a screening based on the estimated frequency of occurrence of the event. Events are assigned one of the following qualitative "frequency category" designators based on the estimated frequency:

Category 1: Those initiating events that are reasonably likely to occur regularly, moderately frequently, or one or more times before permanent closure of the geologic repository operations area (Frequency greater than or equal to 10^{-2} per year).

Category 2: Other initiating events that are considered unlikely, but sufficiently credible to warrant consideration, taking into account the potential for significant radiological impacts on health and safety of the public (Frequency greater than 10^{-6} per year).

Category NC: Those initiating events that are considered to be non-credible during the preclosure phase (Frequency less than 10^{-6} per year).

These categories are consistent with those discussed in Volume II, Section 10.1.3.1, as well as proposed changes to 10 CFR 60. The 10^{-6} events/year threshold for credibility is consistent with previous U. S. Nuclear Regulatory Commission, U. S. Department of Energy (DOE), and commercial nuclear industry precedent. For those WP related events in Categories 1 or 2, the general effects on the WP are briefly discussed, as well as whether a specific or similar analysis of the effects of the event on the WP was performed for the ACD.

7.2.1.2 Preliminary Hazards Analysis

A summary of the preliminary hazards analysis which was performed for the MGDS surface and subsurface facilities (CRWMS M&O 1996m) is provided in Section 10.1 of Volume II and will not be repeated here. Internal events identified by the preliminary hazards analysis were carried forward to the frequency analysis discussed in Section 7.2.1.3. External events that were eliminated during the preliminary hazards analysis pre-screening, because they were not applicable to the Yucca Mountain site or the preclosure phase of the MGDS, are listed in the notes for Table 10.1-1 in Volume II. As discussed in Section 7.2.1.1, several external events were also screened out based on the assumption that the MGDS surface or subsurface facilities would mitigate the effects of the event to the extent that there would be no adverse impact to the performance of the WP. The external events screened on this basis include:

1. Seismic related failure of other systems, structures, and components
2. Flooding
3. Lightning
5. Weather extremes
6. Toxic gas
7. Tornado (and associated missile hazards)
8. Extreme wind (and associated missile hazards)
9. Industrial activity accident
10. Military accident
11. Commercial aircraft crash
12. Intentional intrusion (sabotage)
13. Loss of off-site power
14. Thermal loading

Final verification of the adequacy of this assumption will be made as the MGDS design matures and further analysis of DBEs is performed. The remaining external hazards which were not screened out include:

1. Seismic events
2. Volcanic ashfall
3. Sandstorm
4. Range fire
5. Rockfall

These remaining external events will be discussed under the appropriate category of this evaluation in Section 7.2.1.3.

7.2.1.3 Identification of WP Design Basis Events

This section summarizes the categorization of WP related internal events, and external events not screened out as discussed in Section 7.2.1.2, by general WP effect and frequency of occurrence. This summary is presented in Table 7-1. As discussed in Section 7.2.1.1, the internal and unscreened external events from the *Preliminary Hazards Analysis* (CRWMS M&O 1996m) were recategorized according to their general effect on the WP. Descriptions of the events and the potential loads imposed on the WP are also provided in sufficient detail to allow the effects to be analyzed in detail (if the event is credible) during the next phase of WP design. The sources for the potential WP loads, and the probabilistic evaluations performed to estimate frequencies used to define the frequency categories (1, 2, or NC) assigned to the events in Table 7-1, are provided in the *Waste Package Off-Normal* and *Accident Scenario Report* (CRWMS M&O 1996n). The last two columns of the table identify whether an event is considered in a WP ACD design analysis, indicate the Volume III section where the analysis is summarized (if one has been performed) and provide additional comments on the estimated effects of the event.

Table 7-1. Waste Package Design Basis Event Summary

WP Event Category	WP Event Description[(A)]	WP Load Description[(A)]	MGDS Event No. [(B)]	Frequency Category [(A)]	Evaluated for ACD[(C)]	Estimated Effect
1. Falling Objects	Dropped SNF Assembly CF-Case (Less bare fuel handling)	Max. 963 kg assembly drops max. 8.9 m to DC bottom or ≈4.3 m to DC basket top	S-1.22	2	N	Effect on WP basket not previously evaluated. For no-CF case, event is likely and may require consideration in criticality evaluations (one unlikely event plus all likely events). Note: ACD assumes CF-case.
	Dropped SNF Assembly No-CF Case (all bare fuel handling)		S-1.22	1	N	
	Rockfall [spherical for conservative analysis, or flat bottomed tuff wedge or slab with steel set (if used) and/or rockbolt for realistic analysis]	5MT rock for Cat. 2 25 MT rock for Cat. NC falls 2.1 - 2.6 m onto WP top	SS-1.12, SS-1.13, 8.38	2/NC	P (see Sect. 6.3.4.3, 6.4.2.3, 6.5.2.3)	Critical rock mass currently evaluated for starter tunnel height (8 m) only. Analysis required for emplacement drift height and rock geometry.
	WP slaps down onto previously fallen WP in Waste Handling Building - Both WP falls induced by seismic event of sufficient magnitude	Max. 48MT unfilled WP slaps down and strikes other WP edge on or broadside.	8.1a	2/NC	N	Effect on WP barriers, basket, or fuel/glass not previously evaluated. Event may be non-credible if a) WP seismic supports are provided in Waste Handling Building staging area, or b) analysis demonstrates that seismic event will not cause WP slap down.
	Automatic Center of Gravity Lift Fixture Drop	≈1000 kg 5 m drop	S-1.24	2/NC	P (see Sect. 6.3.4.3, 6.4.2.3, 6.5.2.3)	Effect on WP barriers, basket, or fuel/glass not previously evaluated for these objects. Expected to be bounded by current analyses which indicate most vulnerable WP (DHLW) can handle 4 MT spherical mass dropped from a height of 8m onto horz. WP.
	Emplacement Drift Steel Set	≈500 kg 2.1-2.6 m drop	SS-1.14	2/NC		

Table 7-1. Waste Package Design Basis Event Summary (Continued)

WP Event Category	WP Event Description[(A)]	WP Load Description[(A)]	MGDS Event No. [(B)]	Frequency Category [(A)]	Evaluated for ACD[(C)]	Estimated Effect
2. WP Drop Events	Horizontal Drop by Gantry	1.68 m horizontal drop	S-1.7	2	N	Effect on WP barriers, basket, fuel or glass & canisters not previously evaluated
		0.37 m horizontal drop	S-1.7	2	N	
	Vertical Drop by Crane	0.46 m vertical drop	S-1.6	2	Y (see Sect. 6.3.4.1, 6.4.2.1, 6.5.2.1)	No effect on barriers, basket, fuel or glass canisters expected based on existing 2m drop analyses. Effect on glass not previously evaluated. Possible recovery problems due to damaged WP skirt.
		0.15 m vertical drop	S-1.6	2		
	Slap Down	Vertical drop by Crane (only 12PWR/24BWR WPs are capable of tipping based on lift height and critical angle for tipping)	S-1.11	2	Y (see Sect. 6.3.4.2, 6.4.2.2, 6.5.2.2)	No breach of WP barriers or glass canisters expected based on existing analyses if WP strikes flat surface. Possible damage to basket, fuel or glass. Slap down onto other objects not yet evaluated.
		WP tips over from vertical position due to seismic event of sufficient magnitude	8.1a	2	P (same as above)	
2. WP Drop Events (continued)	Puncture Hazards	0.37 m horizontal drop onto rail line running perpendicular to WP axis	S-1.13	2	N	Effect on WP barriers, basket, fuel or glass & canisters not previously evaluated
3. Collision/ Crushing Events	Transporter Derailment/Collision	WP and cart collide head-on with unyielding object at 8 km/hr	S-1.26 S-1.27 S-1.28 SS-1.4 SS-1.5 SS-1.15	½	N	Effect on WP barriers, basket, fuel or glass & canisters not previously evaluated. Three possible scenarios identified. Effects of transporter derailments/collisions expected to bound those for emplacement carts and Waste Handling Building gantries.
		WP and cart ejected from transporter at maximum speed of 8 km/hr and at a height of ≈1.28 m				
		WP, cart, and transporter tip sideways and slap down. WP in horz. position in transporter with its centerline ≈2.5 m above the rails.				

Table 7-1. Waste Package Design Basis Event Summary (Continued)

WP Event Category	WP Event Description[A]	WP Load Description[A]	MGDS Event No. [B]	Frequency Category [A]	Evaluated for ACD[C]	Estimated Effect
3. Collision/ Crushing Events (Continued)	Transporter Runaway	Sequences required for this event would include failures of several components or multiple human errors.	SS-1.6	NC	N	N/A Event is currently considered non-credible until further review of more detailed design can be performed.
	Shield Door Closes On WP	Gantry, carts and transporter prevent shield doors from contacting WP	S-1.20, SS-1.10, SS-1.11	NC	N/A	N/A Event is considered non-credible
4. Missile Hazards	Missiles resulting from failure of pressurized components	0.5 kg valve stem; 1 cm diameter velocity = 73.6 m/s	S-3.1	2	N	Effect on WP barriers, basket fuel or glass & canisters not previously evaluated
5. Seismic Hazards	Seismic Events	0.66 g acceleration (differs from Controlled Design Assumptions; see 7.5.1 for more detail)	8.1a	2	N	Effect on WP barriers, basket fuel or glass & canisters not previously evaluated
	Shearing Due To Fault Displacement	Max. 28 cm fault displacement (Bow Ridge)	8.1b	2	N	WP reorientation. Maximum credible displacement insufficient to shear load emplaced WP.
6. Internal Pressurization	Fuel Cladding Rupture Due To WP Horizontal Drop or Slap-down	Max. PWR WP 86 psig Max. BWR WP 63 psig	N/A (unique to WP)	2	N	Based on estimate of maximum internal pressure, breach of barriers not expected, but not yet formally evaluated.

Table 7-1. Waste Package Design Basis Event Summary (Continued)

WP Event Category	WP Event Description[A]	WP Load Description[A]	MGDS Event No. [B]	Frequency Category [A]	Evaluated for ACD[C]	Estimated Effect
7. Thermal Hazards	Fire	Electrical/Cabling Fire Range Fire ASTM E119-95 Std. Fire Time (min.) Temp (°C) 0 20 5 538 10 704 30 843 60 927	S-4.1, SS-4.1, 8.16	1/2	N	Effect on WP barriers, basket fuel, or glass & canisters not previously evaluated. If evaluation desired the Standard Fire defined in ASTM E119-95 could be used to gauge WP response to fire (clad or glass temps., thermal stresses), for later comparison with future MGDS fire hazards analysis.
7. Thermal Hazards (continued)	Thermal Cycling (due to WP retrieval)	Max. Drift ΔT=117°C 2 cycles possible Min. Drift ΔT=32°C 99 cycles possible	SS-5.1	1/2/NC	N	Possible thermal fatigue of WP components. Frequency may vary from 1 to NC based on retrieval needs. Effects not previously evaluated but expected to be negligible.
	Loss of Heat Sink (i.e., burial)	Emplaced WP buried by rockfall	8.38	2/NC	P (see Sect. 6.2.1.1.3)	Effects equivalent to early backfill. Existing backfill analysis indicates many days or weeks before peak clad temp. limit exceeded depending on the thermal conductivity of the backfill. Further evaluation may be required.
		Transporter burial by ashfall	8.4.2	NC	N/A	N/A Event is considered non-credible
		Transporter burial by sandstorm	8.7	NC	N/A	

Table 7-1. Waste Package Design Basis Event Summary (Continued)

WP Event Category	WP Event Description[(A)]	WP Load Description[(A)]	MGDS Event No. (B)	Frequency Category (A)	Evaluated for ACD[(C)]	Estimated Effect
8. Chemical Contamination	Chemical contaminants (lubricants, oils, etc.) with potential for affecting post-closure performance	None Identified By PHA (CRWMS M&O 1996m)	N/A (unique to WP)	NC	N/A	N/A Event is currently considered non-credible until further review of more detailed facility design can be performed.
9. Flooding	Decon Unit Floods UCF WP during loading	Flooding of loaded UCF DC - Potential for Criticality	S-3.2	2	Y (see Sect. 6.3.5, 6.4.3, 6.5.3)	No effect expected based on existing criticality analyses.

Table 7-1 Notes:

A - Further detail and bases for the loads and event frequencies is provided in the *WP Off-Normal and Accident Scenario Report* (CRWMS 1996n).

B - In some cases, different surface and subsurface internal events in Volume II, Section 10.1 share the same numeric identifier. These events are distinguished here by an S for surface and an SS for sub-surface.

C - The status of analysis of each of the events for WP ACD is indicated as follows:

Y(es) - The event has been previously evaluated during WP ACD design activities, and found to have no adverse affect on the performance of the WP. However, these events may require reanalysis if the WP designs deviate significantly from the ACD design during the next phase of WP design.

P(artial) - A similar event was evaluated during WP ACD, but further evaluation will be required during the preliminary design phase to adequately demonstrate that the event will not adversely affect the performance of the WP.

N(o) - The event has not yet been evaluated.

7.2.2 Probabilistic Evaluation of WP Manufacturing Defects

Previous Performance Assessments (SNL 1995a) have considered the scenario of a WP with a through-wall manufacturing defect. Recent preclosure DBE analyses have also indicated a desire to evaluate the effects of a "non-mechanistic" WP failure, possibly due to pre-existing through-wall defect in the WP. In both cases, such a WP is considered to be breached even in the absence of a specific off-normal event at the MGDS. The previous assumptions of the fraction of packages assumed to contain such a defect have ranged from 0.05 percent up to 1 percent of the total WP population (SNL 1995a). However, in all cases, the assumptions appear to have been completely arbitrary and were generally intended to provide added conservatism to the evaluation. The purpose of this section is to review the available literature on manufacturing defects to develop a more realistic estimate for future analyses.

WP manufacturing defects can be postulated to occur during the two welding processes to which the WP is subject: welding of base metal sections and/or inner barrier cladding during fabrication of the disposal container, or welding of the lids onto the WP to seal it after spent nuclear fuel (SNF) has been loaded. The types of defects which may occur include cracks, lack-of-fusion, porosity, and slag inclusions. For the purposes of this assessment, all will be treated as a localized reduction in the wall thickness of a barrier. In addition, no distinction will be made between surface breaking and non-surface breaking defects, as corrosion of the barriers will eventually expose any defect not initially at the surface. Finally, while the probability that both an inner and outer barrier weld defect will concurrently occur in the same cross-sectional area of the WP is extremely small, this aspect will be neglected to further add to the conservatism of the estimate.

Several studies of weld defect depth distributions and densities have been performed in the past. Chapman (1993) has developed a computer simulation for predicting defect density and depth distributions for post-inspection welds, and validated this simulation against actual data from nuclear pressure vessel welds. For the 217.5 mm thick sub-arc welds of the Midland reactor vessel, the probability density function produced by this simulation predicted that the probability that a given defect depth was greater than or equal to 50 percent ($\approx$109 mm) through-wall was $\approx 2\text{x}10^{-7}$, with the probability continuing to decrease for larger defect depths. This was found to be conservative compared to actual Midland vessel data. Other distributions developed for 25.4 mm and 51 mm nuclear welds found the probability of 50 percent through-wall defects to be $\approx 2\text{x}10^{-3}$ and $\approx 1\text{x}10^{-5}$, respectively. Making the assumption that the Midland 50 percent probability applies to 100 percent through-wall defects for a sub-arc welded WP outer barrier, and using the simulation's conservative (by a factor of 9) prediction of 390 defects per m^3 of Midland weld material, this suggests a conservative probability of $7.8\text{x}10^{-5}$ through-wall defects per m^3 of weld.

Current estimates (CRWMS M&O 1995x) indicate that a WP outer barrier will require $\approx$0.019 m^3 of weld material (21 PWR/40 BWR estimate). This allows a deterministic estimate of $\approx 1.5\text{x}10^{-6}$ through-wall defects per *outer barrier* for each WP. Assuming the same frequency of through-wall defects per m^3 of weld material applies to the Alloy 825 inner barrier, and assuming the inner barrier is clad on the outer barrier resulting in 0.498 m^3 of weld material (CRWMS M&O 1995x), the frequency of a through-wall defect in the inner barrier is estimated to be $3.9\text{x}10^{-5}$ per inner barrier for each WP. Neglecting the additional low probability of both inner and outer barrier defects concurrently occurring in the same cross-section of weld material, this provides for a conservative estimate of 5.8×10^{-11} through-wall manufacturing defects per WP.

Based on the *Controlled Design Assumptions Key Assumption* 003 of 12,037 total WPs (CRWMS M&O 1995n), and the frequency of WP through-wall defects estimated above, the probability that there will be one WP with a through-wall manufacturing defect in the MGDS is conservatively estimated to be 7.0×10^{-7}. On a per year basis, assuming an average of 502 WPs per year, the frequency of encountering a WP breach due to a manufacturing defect is estimated to be 2.9×10^{-8} events per year. Therefore, while the occurrence of such a WP defect may be assumed for the purposes of providing conservative analyses, it is to be considered a non-credible event based on the criteria discussed in Section 7.1.1.1 above.

7.3 POSTCLOSURE PROBABILISTIC EVALUATION METHODOLOGY

During all operations in the nuclear fuel cycle, there exists the potential for uncontrolled criticality given the necessary conditions. Recognizing this, the Nuclear Regulatory Commission requires that design features and operating procedures be established to preclude criticality. For waste disposal, this mandate is contained in the 10 CFR 60.131(b)(7), stating that a "nuclear criticality event shall not be possible unless at least two unlikely, independent, and concurrent or sequential changes have occurred in the conditions essential to nuclear criticality safety." For the *Engineered Barrier System, Engineered Barrier Design Requirements Document* (YMP 1994b) Requirement 3.7.1.3.A reiterates verbatim the 10 CFR 60 requirement. Furthermore, to encourage conservative design practice, 10 CFR 60 (and the *Engineered Barrier Design Requirements Document*) requires that "the calculated effective multiplication factor (k_{eff}) must be sufficiently below unity to show at least a 5 percent margin after allowance for the bias in the method of calculation and the uncertainty in the experiments used to validate the method of calculation." As a result of this design philosophy, several events or conditions must occur before a criticality event is possible in any phase of the repository. For a loaded and sealed WP, these are:

1. Breach of all WP barriers. Some mechanisms that lead to this event are conditional on the presence of water.

2. A source of water or other moderator with access to the waste form. Moderator access is generally conditional on Event 1.

3. A low quantity of neutron absorber with specification fuel loaded, or a moderate quantity of neutron absorber with non-specification (low burnup and/or high initial enrichment) fuel loaded.

4. Fuel geometry that provides for sufficient moderation/reflection. For filled WPs, filler material (if used) must be partially or wholly displaced by moderator.

5. Fuel that is isotopically capable of criticality given a source of moderator, an appropriate geometry, and insufficient neutron absorbers.

To satisfy 10 CFR 60.131(b)(7), the likelihood of all five events/conditions occurring must be evaluated for both the preclosure and postclosure phases of the repository. In the preclosure phase, the likelihood of a criticality is characterized by the independent occurrence of the above events. Sequences of dependent events (i.e., corrosion) are extremely unlikely due to the relatively small amount of time elapsed in the preclosure phase. However, during postclosure, the likelihood of dependent sequences of Events 1 through 4 increases as more time passes, while many of the possible causes of independent failure during preclosure (i.e., human error) are no longer present. Event 5 is partially independent of the other four events in terms of the fuel isotopics, while the fuel cladding is relatively intact. However, the criticality potential for fuel with given isotopic characteristics is highly dependent on Event 4. In addition, once the fuel cladding has degraded sufficiently, aqueous or diffusion transport will provide a mechanism, in addition to that of radioactive decay, for changing the fuel isotopics. As a result of these dependencies, the methodologies used to evaluate the likelihood of criticality in the preclosure and postclosure phases will be different.

Analyses will begin with a qualitative Failure Modes and Effects Analysis (FMEA). The FMEA involves the identification of failure modes for components of the WP, the initiating conditions or mechanisms that can cause the failure modes to occur, and the effects of the failures on other components of the WP and the overall performance of WP criticality controls. The effect of each identified failure mode will be related to one or more of the five events/conditions listed above.

For postclosure, it is convenient to subdivide criticality events into two types: events that may occur internal to the WP structure, and events that may occur external to the WP once the structure has degraded sufficiently to allow transport of the fissile isotopes and/or neutron absorbers. The FMEA for postclosure internal criticality primarily focuses on identifying those failure modes that result in the occurrence of one or more of the five events/conditions listed above. The FMEA serves to clearly identify any failure mode dependencies (i.e., water acts both as a moderator and corrosion enhancer) and failure modes that are conditional on other failures (i.e., neutron absorber cannot leach until WP is breached). The outcome is a comprehensive listing of failure modes for WP components, and the effects that the failure mode will have on the WP configuration, the failure of other components within the WP, and the criticality potential of the package. The FMEA for postclosure external criticality will focus primarily on identifying the possible pathways for fissile radionuclide transport and the potential critical configurations that could be formed in the natural environment. Both internal and external FMEAs will serve as the framework for development of quantitative probabilistic models for internal and external criticality. The end states of the internal model will serve as inputs to the external model.

Quantitative models for postclosure criticality are necessary to allow determination of the effects of various design decisions on the overall likelihood of a criticality event. These models must use probabilistic methods due to the uncertainty in the time to occurrence of WP failure modes and the mechanisms and conditions that produce them. The first step in developing the probabilistic model is defining probability density functions (PDFs) for each failure mode to describe the probability that the failure mode will occur at a given time. This is done by using available data or models of the mechanisms that can cause the failure mode. For conditional failure modes, which require the prior occurrence of another failure mode or initiating event, the PDFs are developed assuming the event or preceding failure has already occurred (called conditional PDFs).

Next, the PDF for the occurrence of a specific configuration likely to produce criticality of a WP can be defined as the convolution of the PDFs of all failure modes in the sequence leading to that change. For example, if f_1 is the PDF for an initiating condition (e.g., an environmental condition), f_2 is the PDF for a failure mode (e.g., WP barrier breach), which is conditional on f_1, and f_3 is the PDF for a second failure mode (e.g., neutron absorber removal), which is conditional on f_2, then the PDF for the time to occurrence of all three failures is mathematically represented as:

$$f(t)=\int_0^t\int_0^{t_2} f_1(t_1)f_2(t_2)f_3(t-t_1-t_2)dt_1dt_2 \qquad \text{(Equation 7-1)}$$

The use of this method to estimate the time to occurrence of a single configuration for which a fraction of the WPs may be critical is demonstrated in Section 7.3.

In reality, there are a multitude of possible intermediate geometries that can occur in the time period between the intact and fully degraded WP. The above method can be expanded to model the parallel degradation of all components interior to the WP by determining the conditional PDFs and performing the convolution for the sequences of failures that lead to failure of each component. In this manner, the degraded WP geometries that are likely to exist, and the times that they are likely to be present, emerge. This information can then be supplied to the designers to reduce the number of degraded WP geometries that must undergo deterministic neutronics analysis.

In turn, the resulting deterministic analyses of k_{eff} for the credible geometries can be used to determine the fraction of WPs that will be capable of criticality. This fraction is time-dependent for any geometry that spans a few thousand years due to changing isotopics. This step is also demonstrated for the configuration evaluated in Section 7.4. Using this information and the probabilistic configuration model discussed above, the probability of a criticality event can be estimated. Design changes, specific loading schemes, or filler addition to specific packages can then be recommended as necessary to reduce the probability of criticality below a predetermined limit.

Quantitative modeling of external criticality potential will require a somewhat different approach. As discussed previously, concern with external criticality focus on the times after the dissolution of the WP. This will require modeling of the dissolution, transport, and reprecipitation of fissile radionuclides.

7.4 PROBABILISTIC EVALUATION OF POSTCLOSURE INTERNAL CRITICALITY EVENTS

This section presents a demonstration of the method discussed in Section 7.3 for determining the probability that a given geometric configuration exists, and the probability of a criticality as a result of this configuration. The configurations evaluated are breached, but otherwise geometrically intact 21 PWR uncanistered fuel (UCF) and canistered fuel (CF) WPs. A similar evaluation for WPs containing defense high-level waste (DHLW) glass has not been performed because they are physically incapable of an internal criticality event (see Section 7.5.3).

The system boundary for this analysis includes the emplaced WP and the emplacement drift in the vicinity of the package. The WP outer barrier for both the UCF and CF designs was 100 mm (3.94 in.) of carbon steel. The inner barrier for both designs was 20 mm (0.79 in.) of Incoloy Alloy 825 corrosion-resistant material. The CF shell was assumed to be 25.4 mm (1.0 in.) of Type 316L stainless steel (CRWMS M&O 1994k). Present waste package designs provide a total of 10 mm (0.39 in.) of borated Type 316L stainless steel between adjacent assemblies. All interior spaces were assumed to be filled with an inert gas, and no filler material was used.

The local emplacement environment is the horizontal in-drift emplacement concept using a low-thermal loading (24.2 metric tons per acre[MTU]/acre) strategy and 4.27 m (14 ft.) drifts (CRWMS M&O 1994b). This was the preferred thermal loading concept at the time the analysis was performed. The assumption of low thermal loading means that liquid water would be physically capable of contacting the WPs within a few hundred years after emplacement (thus providing an environment for enhanced corrosion and supplying a source of moderator once the WP has breached). Therefore, within the present understanding of the Yucca Mountain hydro-thermal processes, evaluating the UCF and CF WPs with a low thermal load is a conservative assumption with respect to criticality. It was also assumed that backfilling of the emplacement drifts has not been performed.

7.4.1 Failure Modes and Effects Analysis and Fault Tree Development

A fault tree is developed to estimate the frequency of a criticality event for a WP. To identify the basic events of the fault tree, an FMEA has been performed for the WP and the emplacement drift in the immediate vicinity. A level of detail has been chosen that is appropriate given the conceptual nature of the design. The objective of the FMEA is to provide a systematic, qualitative review of the WP design to identify the effects of failures on the geometry and criticality potential of the WP.

This analysis considers only water moderated criticality internal to the WP. It has been shown that unmoderated criticality is impossible for intact light water reactor fuel with fissile content less than 5 percent (Knief 1993). Water is the only moderator present in the WP environment that can enter a relatively intact WP.

The WP can be broken down into five basic components: Corrosion Allowance Outer Barrier, Corrosion Resistant Inner Barrier, Basket, Fuel Assembly, and Fuel Rods. In the case of the CF WP, the CF canister shell adds a sixth component. The emplacement drift can be broken down into two components: Drift and WP Supports. These WP and emplacement drift components represent the level of detail for which the FMEA was conducted.

The FMEA table is divided into eight headings; these are described in Table 7-2. The results of the FMEA are presented in Table 7-3. For the events in the credible sequences leading to criticality, event probabilities and PDFs will be developed in Section 7.4.3. The fault tree developed from the FMEA is shown in Figure 7-1 and the logic represented will be the same for both the UCF and CF WPs. The methodology and symbols used in the construction of the fault tree are as suggested in the *Fault Tree Handbook* (NRC 1981). The fault tree was plotted using CAFTA Version 2.3 fault tree analysis software (SAIC 1993). In addition to a one-line description, each gate, basic event, and conditional event is uniquely labeled with an identifier. These identifiers will be used for each basic event of the fault tree quantified in Section 7.4.2 and for the fault tree quantification in Section 7.4.3.

Table 7-2. Description of FMEA Headings

FMEA Heading Name	FMEA Heading Description
Component	The components of the WP or emplacement drift for which the failure modes were investigated.
Component Function	The intended function(s) for each component being evaluated.
Failure Mode	The failure mode(s) of consideration for each component.
Failure Mechanisms	Physical mechanisms capable of causing the indicated failure mode.
Required Conditions	Conditions such as a specific environment or previous failure of another component required to cause the failure mechanism.
Failure Effects	The effects of the failure on other components of the WP.
Criticality Potential Effects (CPE)	The qualitative effect on the potential for internal WP criticality. These are recorded on the table as: + failure mode favorable to criticality - failure mode unfavorable to criticality N condition has no effect on criticality
Plans for Modeling Failure Mode/Mechanism (MOD)	Indication of when or if failure mode/mechanism will be modeled for probabilistic evaluations of internal criticality. These are recorded on the table as: P Presently Modeled F May Be Modeled In Future Analyses N Will Not Be Modeled
Comment	Comments to provide additional information or explanation.

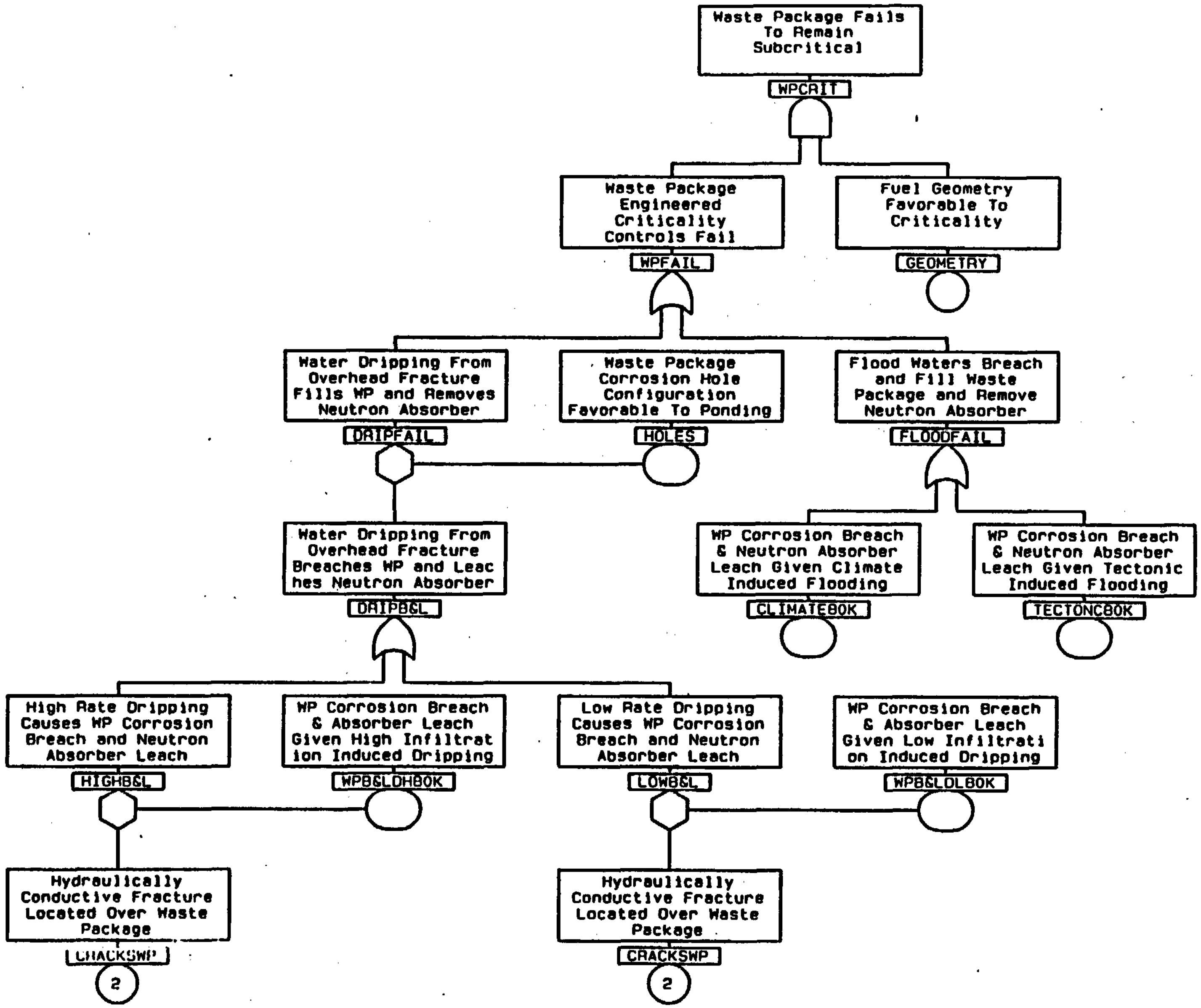

Figure 7-1. Postclosure WP Internal Criticality Fault Tree

Table 7-3. Waste Package Postclosure Criticality Failure Modes and Effects Analysis

Component	Component Function	Failure Modes	Failure Mechanisms	Required Conditions	Failure Effects	CPE	MOD	Comments
Emplacement Drift & Near-field Environment.	Provide an environment which ensures long WP life by limiting contact with water and other hazards.	Fails to prevent flooding of repository.	Water table rises to repository horizon.	Drastic climate change towards wetter conditions.	Submersion of all or most waste packages. Provides conditions necessary for enhanced corrosion of waste package components. Immediate filling of interior volume and moderation upon breach. See Comment.	+	P	Both mechanisms require millions of years to produce the failure mode. The geologic record indicates that the water table has not been at the same level as the repository horizon during the 10-13 million years since Yucca Mountain was formed [National Research Council 1992]. However, several hundred million years ago, the area that is now southern Nevada was covered by a shallow sea, so this cannot be completely ruled out over that time span.
			Tectonic trends lower repository to level of water table.	Repository block subsidence.				
		Fails to prevent localized flooding of drift.	Drift collapse creates local dam, causing infiltrating water to pool.	Infiltrating water (surface or returning pore-space water) AND Collapsed Drift.	No effect. See Comment.	N	N	Fallen rock acts as "french drain," preventing water from pooling. Inverts may also contain channels to direct water away.
		Fails to prevent wetting of package component surfaces.	Water film forms on rubble from collapsed drift roof in contact with waste package.	Drift temperature below boiling AND Collapsed drift AND High Humidity.	Localized wetting of waste package. May be intermittent or continuous depending on flow/drip rate. Provides conditions for enhanced corrosion of waste package components. Except for humid atmosphere, provides potential source of moderator to interior volume of package. See comments for applicable mechanisms.	+	N	Contact with rubble alone should not result in enhanced corrosion as tests performed using Paintbrush tuff indicated that even at 100% humidity, the surface of the tuff remained air-dry, with no water film [Conca 1990]. In addition, film would not supply a sufficient amount of water to fill and moderate a breached package.
			Rubble from collapsed drift ceiling creates pathway from hydraulically conductive fracture to waste package.	Infiltrating water AND Collapsed drift AND Drift temperature below boiling.			F	Constant flow would be required. Tests performed using Paintbrush tuff indicate that a sustained flow rate of 1ml/hr would only raise volumetric water content of the tuff to ≈4%. Under no flow conditions, initially wet tuff with 4% volume water content became air-dry in a few days [Conca 1990].

Table 7-3. Waste Package Postclosure Criticality Failure Modes and Effects Analysis (Continued)

Component	Component Function	Failure Modes	Failure Mechanisms	Required Conditions	Failure Effects	CPE	MOD	Comments
Emplacement Drift & Near-field Environment. (continued)	Provide an environment which ensures long WP life by limiting contact with water and other hazards. (continued)	Fails to prevent wetting of package component surfaces. (continued)	Saturated flowing fracture located over waste package concentrates infiltrating water onto package.	Drift temperature below boiling AND Infiltrating water.	Localized wetting of waste package. May be intermittent or continuous depending on flow/drip rate. Provides conditions for enhanced corrosion of waste package components. Except for humid atmosphere, provides potential source of moderator to interior volume of package. See comments for applicable mechanisms. (continued)	+	P	Only a small fraction of packages would have flowing fractures over them. Flow rate dependent on size of fracture, degree of concentration of infiltrating water, and amount of water infiltrating from surface, and/or returning from condensate cap.
			Human intrusion in the form of drilling creates fast pathway for surface water.	Reason for drilling. AND Package located under or near borehole.			F	Only a few packages could be affected by human intrusion. A recent PA estimated the probability that one or more packages would be affected by a human intrusion event in 10,000 years to be 7.9×10^{-2} [SNL 1995a]. In addition, a drill bit which intersects the WP would damage the geometry of the fuel in a manner which is unfavorable to criticality. To provide favorable conditions, drilling would have to cease at the point of penetrating the inner barrier, or before.
			High humidity environment creates condensed water film on surface of package components exposed to air.	WP component surface temperature below boiling AND High Humidity.			F	Water film from humid atmosphere capable of causing corrosion, but provides insufficient source of water for filling and moderation of package interior.
		Fails to prevent excessive mechanical loading of waste package.	Rock fall incident on waste package.	Drift support failure OR Severe seismic event.	Mechanical loading of waste package. Possible breach of WP barriers depending on amount of applied stress and degree of barrier degradation. See comments.	+	F	Waste package breach only of concern for criticality if sufficient amounts of water are available to fill and moderate interior. Significant absorber loss also necessary. See section 7.1.1 for discussion on evaluation of rock-induced loading.

Table 7-3. Waste Package Postclosure Criticality Failure Modes and Effects Analysis (Continued)

Component	Component Function	Failure Modes	Failure Mechanisms	Required Conditions	Failure Effects	CPE	MOD	Comments
Waste Package Supports	Prevent increased corrosion at waste package/support interface.	Support materials incompatible with outer barrier materials under certain conditions.	Galvanic corrosion, saline water attack (typically affects rebar), crevice corrosion, presence of corrosion enhancing species.	Aqueous conditions AND Incompatible material.	Localized corrosion of lower portions of waste package outer barrier. See comments.	– OR +	F	Breach of the lower portion of the package only favorable to criticality for flooded drift conditions (very unlikely). For non-flooded conditions (most-likely) corrosion of lower portions creates pathway for drainage of WP interior. This failure mode cannot be evaluated in detail because a specific WP support method (cart vs. pedestal) has not been selected at this time.
	Prevent movement of waste package.	Fails to prevent movement.	Seismic event of sufficient magnitude causes waste package to fall off of support.	Severe seismic event.	Mechanical loading of waste package if it falls off of support. Possible breach of WP barriers depending on amount of applied stress and degree of barrier degradation. See comments.	+ OR –	F	Waste package breach only of concern for criticality if sufficient amounts of water are available to fill interior, orientation of WP prevents drainage through holes in bottom (if any) and significant absorber loss has occurred.
Corrosion Allowance Outer Barrier	Contain and isolate waste from environment and prevent intrusion of water to interior components.	Outer barrier breached (may be discretized into thickness remaining and area breached for future evaluations).	Manufacturing defect	Occurrence of through-wall defect during welding AND Pre-service inspection fails to find defect.	Allows degradation of inner barrier. Area of inner barrier affected dependent on size of outer barrier breaches and design of inner barrier (e.g., clad inner barrier exposes area under pit only). See comments for applicable mechanisms.	+	F	Based on thickness of weld, number of weld beads, and degree of pre-service inspection for nuclear components, estimated probability of such a defect through carbon steel barrier only would be ≈ 10^{-6}/package (see section 7.1.2).
			Excessive mechanical load	{Rockfall OR Seismic event w/ WP support failure} AND Partially degraded package.			F	Ability of package to withstand rockfall decreases with loss of outer barrier thickness. Size of breach dependent on difference between applied and allowable stress. Expected to be << 1% of WPs breached by this mode by 1000 years.
			General corrosion	Aqueous or humid environment.			P	Results in bulk thinning of barrier. Primarily of concern for determining structural response of package to mechanical loading (i.e., rockfall) and widening of pits. Pitting expected to produce first breach.

Table 7-3. Waste Package Postclosure Critic[illegible] Failure Modes and Effects Analysis (Continued)

Component	Component Function	Failure Modes	Failure Mechanisms	Required Conditions	Failure Effects	CPE	MOD	Comments
Corrosion Allowance Outer Barrier (continued)	Contain and isolate waste from environment and prevent intrusion of water to interior components. (continued)	Outer barrier breached (may be discretized into thickness remaining and area breached for future evaluations). (continued)	Pitting and crevice corrosion	Aqueous or humid environment.	Allows degradation of inner barrier. Area of inner barrier affected dependent on size of outer barrier breaches and design of inner barrier (e.g., clad inner barrier exposes area under pit only). See comments for applicable mechanisms. (continued)	+	P	Pitting expected to be dominant mechanism for pin hole breach of outer barrier. Effects of microbiologically influenced corrosion may increase susceptibility at lower temperatures (60-80°C). Crevice corrosion may occur at waste package/support interface. However, this would result in the creation of drainage holes, which would be beneficial from a criticality standpoint.
			Galvanic corrosion	Aqueous or humid environment AND galvanic couple with more noble metal.			N	Current design information does not indicate that a noble material will be available for galvanic coupling until package is breached. Only potential source prior to breach would possibly be from railcar emplacement concept, if used.
			Stress corrosion cracking	sensitized material AND aqueous environment with aggressive species (Cl, S, etc.) AND tensile stresses in excess of ½ yield.			N	General literature indicates that carbon steels are not susceptible to stress corrosion cracking [NRC 1987].

Table 7-3. Waste Package Postclosure Criticality Failure Modes and Effects Analysis (Continued)

Component	Component Function	Failure Modes	Failure Mechanisms	Required Conditions	Failure Effects	CPE	MOD	Comments
Corrosion Allowance Outer Barrier (continued)	Contain and isolate waste from environment and prevent intrusion of water to interior components. (continued)	Structural failure.	Galvanic corrosion	Aqueous or humid environment AND Outer barrier breach allowing galvanic couple with inner barrier,	Barrier fails to contribute to structural support of package weight. If all barriers have succumbed to this failure mode, fuel collapses into a pile on the invert, mixed with remnants of barrier, basket, filler (if used), fuel assembly structures, and corrosion products from all of the above. Promotes further consolidation of fuel. See comments for applicable mechanisms.		F	Galvanic coupling of carbon steel outer barrier with Alloy 825 inner barrier possible once outer barrier is breached. Outer barrier is expected to act as a sacrificial anode for protection of inner barrier.
			General corrosion	Aqueous or humid environment.			F	No comments.

Table 7-3. Waste Package Postclosure Critic ailure Modes and Effects Analysis (Continued)

Component	Component Function	Failure Modes	Failure Mechanisms	Required Conditions	Failure Effects	C P E	M O D	Comments
Corrosion resistant Inner Barrier	Contain and isolate waste from environment and prevent intrusion of water to interior.	Inner barrier breached (may be discretized into thickness remaining and area breached for future evaluations)	General corrosion	Aqueous environment. AND Breach of outer barrier.	Allows degradation of interior components (interior inner barrier surface, basket, fuel assembly spacer grids, & fuel cladding). Allows potential moderator access to interior (UCF only). Slow filling of breached WP if continuous supply of liquid water available in sufficient quantity. Rate of filling dependent on flow rate, size of corrosion holes on upper surfaces, and rate of water removal via evaporation, and corrosion holes which may form on the lower inside surface of the WP. See comments for applicable mechanisms.	+	F	Inner barrier material chosen for its resistance to general corrosion. Not expected to be dominant mechanism for breach of inner barrier. However, general corrosion would be expected to produce a gradual increase in the radius of pits which do penetrate the inner barrier, increasing the area available for water entry/exit.
			Pitting Corrosion	Aqueous environment AND Breach of outer barrier.			P	Pitting expected to be dominant mechanism for pin hole breach of inner barrier. Effects of microbiologically influenced corrosion may increase susceptibility at lower temperatures (60-80°C). Crevice corrosion may occur following inner barrier breach for CF waste package only on bottom inner portion of barrier at the crevice created by the waste package/CF container shell interface. However, this would result in the creation of drainage holes, which would be beneficial from a criticality standpoint.

Table 7-3. Waste Package Postclosure Criticality Failure Modes and Effects Analysis (Continued)

Component	Component Function	Failure Modes	Failure Mechanisms	Required Conditions	Failure Effects	CPE	MOD	Comments
Corrosion Resistant Inner Barrier (continued)	Contain and isolate waste from environment and prevent intrusion of water to interior. (continued)	Inner Barrier Breached (may be discretized into thickness remaining and area breached for future evaluations) (continued)	Stress corrosion cracking	sensitized material AND aqueous environment with aggressive species (Cl, S, etc.) AND tensile stresses in excess of ½ yield AND Breach of outer barrier.	Allows degradation of interior components (interior inner barrier surface, basket, fuel assembly spacer grids, & fuel cladding). Allows potential moderator access to interior (UCF only). Slow filling of breached WP if continuous supply of liquid water available. Rate of filling dependent on flow rate, size of corrosion holes on upper surfaces, and rate of water removal via evaporation, formation of corrosion holes on bottom surfaces of WP. See comments for applicable mechanisms.	+	N	Immersion studies of Alloy 825 in J-13 well water have failed to produce SCC [NRC 1991]. However, these same tests have shown evidence of slight pitting. In addition, if the inner barrier is clad on the interior surface of the carbon steel outer barrier, it will be in compression due to the higher thermal expansion coefficient of the Alloy 825 while the WP is still at a high temperature (stresses will become tensile as WP cools) [INCO 1992]. Therefore, it is expected that pitting would cause failure of the barrier before SCC.
			Excessive mechanical load	Rockfall OR Seismic event w/ WP support failure.			F	The inner barrier alone will not be able to withstand a rockfall as massive as the complete package can handle. A large enough rockfall occurring after the outer barrier has ceased to provide protection in this respect, may cause a breach of the inner barrier. A breach of the inner barrier before the outer barrier as a result of a rock fall is unlikely due to the higher tensile strength of Alloy 825 [see section 6.5.4.3.4.2].
			Manufacturing defect	Occurrence of through-wall defect during welding AND Pre-service inspection fails to find defect.			F	If the inner barrier is clad on the outer barrier using a sub-arc welding process, the likelihood of a through wall defect may be higher than that given for the outer barrier.

Table 7-3. Waste Package Postclosure Critical Failure Modes and Effects Analysis (Continued)

Component	Component Function	Failure Modes	Failure Mechanisms	Required Conditions	Failure Effects	CPE	MOD	Comments
Corrosion Resistant Inner Barrier (continued)	Contain and isolate waste from environment and prevent intrusion of water to interior. (continued)	Structural failure.	General corrosion	Aqueous environment AND Breach of outer barrier	Barrier fails to provide structural support for package weight. If all barriers have succumbed to this failure mode, fuel collapses into a pile on the invert, mixed with remnants of barrier, basket, filler (if used), fuel assembly structures, and corrosion products from all of the above.		F	Deterioration of the inner barrier will eventually result in its inability to support the weight of the enclosed fuel and basket structure if the other barrier(s) have also failed. This will likely represent the final barrier to the general collapse of the waste package structure. In the absence of other plausible mechanisms (such as SCC), general corrosion will most likely be the cause.
			Stress corrosion cracking	sensitized material AND aqueous environment with aggressive species (Cl, S, etc.) AND tensile stresses in excess of ½ yield AND Breach of outer barrier.		–	F	Immersion studies of Alloy 825 in J-13 well water have failed to produce SCC [NRC 1991], and have shown that this material is more resistant to SCC than stainless steels [ANL 1992]. However, while stresses in the inner barrier may be compressive at high temperature, they may gradually become tensile as the inner barrier cools. The higher thermal expansion coefficient of Alloy 825, indicates that it will also contract quicker than the carbon steel, thus introducing tensile stresses if the outer barrier remains structurally intact once the package has cooled. Pit walls may also provide regions of stress concentration to serve as initiating points. Further work is necessary to determine if sufficient stresses, and the other conditions necessary for SCC, are available.

Table 7-3. Waste Package Postclosure Criticality Failure Modes and Effects Analysis (Continued)

Component	Component Function	Failure Modes	Failure Mechanisms	Required Conditions	Failure Effects	CPE	MOD	Comments
CF container shell (CF WP only)	Contain and isolate waste from environment and prevent intrusion of water to interior.	CF container shell breached	General corrosion	Aqueous environment AND Breach of outer and inner barriers	Allows degradation of interior components (interior inner barrier surface, basket, fuel assembly spacer grids, and fuel cladding). Allows potential moderator access to interior. Slow filling of breached WP if continuous supply of liquid water available. Rate of filling dependent on flow rate, size of corrosion holes on upper surfaces, and rate of water removal via evaporation, formation of corrosion holes on bottom surfaces of WP. See comments for applicable mechanisms.	+	N	Tests have shown stainless steel to be very resistant to general corrosion in J-13 well water [NRC 1987 1990b & 1991]. CF container will likely be breached by localized corrosion first.
			Pitting corrosion	Corrosive conditions (aqueous environment) AND Breach of outer and inner barriers			P	304 and 316 stainless steels have been shown to pit in J-13 well water environments [NRC 1987 1990b & 1991]. Pitting has occurred in specimens where SCC has not. The effects of microbiologically influenced corrosion may also increase susceptibility at lower temperatures (60-80°C).
			Stress corrosion cracking	Sensitized material AND Aqueous environment with aggressive species (Cl, S, etc.) AND Tensile stresses in excess of ½ yield AND Breach of outer and inner barriers			F	Testing of 304 and 304L stainless steels in vapor phase and alternate immersion J-13 well water, with addition of H_2O_2 to simulate radiolysis, found occurrence of SCC [NRC 1990b 1991]. Slow strain rate and fatigue crack growth rate testing in J-13 well water has demonstrated that the 316 stainless steel material intended for use in this application is more resistant to SCC [ANL 1992]. However, these tests only measure relative susceptibility and do not make a determination as to whether SCC will actually occur in the given environment in the absence of cyclic stresses.

Table 7-3. Waste Package Postclosure Criticality Failure Modes and Effects Analysis (Continued)

Component	Component Function	Failure Modes	Failure Mechanisms	Required Conditions	Failure Effects	CPE	MOD	Comments
CF container shell (CF WP only) (continued)	Contain and isolate waste from environment and prevent intrusion of water to interior. (continued)	CF container shell breached (continued)	Excessive mechanical load	{Rockfall OR seismic event w/ WP support failure} AND Outer and inner Barrier Failure	Allows degradation of interior components (interior inner barrier surface, basket, fuel assembly spacer grids, and fuel cladding). Allows potential moderator access to interior. Slow filling of breached WP if continuous supply of liquid water available. Rate of filling dependent on flow rate, size of corrosion holes on upper surfaces, and rate of water removal via evaporation, formation of corrosion holes on bottom surfaces of WP. See comments for applicable mechanisms.	+	F	The CF container shell alone will not be able to withstand a rockfall as massive as the complete package can handle. However, both the outer and inner barriers will be breached long before they lose their structural strength. Therefore, it is likely that the CF container shell will be breached by other mechanisms first.
			Manufacturing defect	Occurrence of through-wall defect during welding AND Pre-service inspection fails to find defect			F	As the CF container shell is a rolled cylinder with welded seams, and is He leak tested at the utility site after inverting, the likelihood of a through-wall defect should be similar to that of the outer barrier.
		Structural failure	General corrosion	aqueous environment AND Breach of outer and inner barriers.	Barrier fails to provide structural support for package weight If all barriers have succumbed to this failure mode, fuel collapses into a pile on the invert, mixed with remnants of barrier, basket, filler (if used), fuel assembly structures, and corrosion products from all of the above.	–	F	Tests have shown stainless steel to be very resistant to general corrosion in J-13 well water [NRC 1987 1990b & 1991]. SCC will probably degrade the structural strength of the CF container shell before gradual reduction of the shell thickness by general corrosion can do so.

Table 7-3. Waste Package Postclosure Criticality Failure Modes and Effects Analysis (Continued)

Component	Component Function	Failure Modes	Failure Mechanisms	Required Conditions	Failure Effects	CPE	MOD	Comments
CF container shell (CF WP only) (continued)	Contain and isolate waste from environment and prevent intrusion of water to interior. (continued)	Structural failure (continued)	Stress corrosion cracking	sensitized material AND aqueous environment with aggressive species (Cl, S, etc.) AND tensile stresses in excess of ½ yield AND Breach of outer and inner barriers.	Barrier fails to provide structural support for package weight If all barriers have succumbed to this failure mode, fuel collapses into a pile on the invert, mixed with remnants of barrier, basket, filler (if used), fuel assembly structures, and corrosion products from all of the above. (continued)	–	F	Testing of 304 and 304L stainless steels in J-13 well water, with addition of H_2O_2 to simulate radiolysis, found occurrence of SCC [NRC 1990b 1991]. Slow strain rate and fatigue crack growth rate testing in J-13 well water has demonstrated that the 316 stainless steel material intended for use in this application is more resistant to SCC [ANL 1992]. However, these tests only measure relative susceptibility and do not make a determination as to whether SCC will actually occur in the given environment in the absence of cyclic stresses. Effects of microbiologically influenced corrosion may increase susceptibility at lower temperatures (60-80°C).
Basket Structure Note: Burnup Credit Design only. Flux Trap design will be evaluated once sufficient design information becomes available	Maintain spacing between intact fuel assemblies.	Basket tube "floor" fails by shearing from static load of single assembly.	General corrosion	aqueous environment AND Breach of all WP barriers	Tube "floor" shears, eliminating the gap between fuel assembly in tube and the one below it. Assembly is separated from the one below it by only the thickness of the basket material sandwiched between them. Eventually, collapse of all tubes will leave assemblies stacked on top of each other, separated only by remaining basket material (compression to ≈90% of original volume).	+	N	General corrosion rate results in slow thinning of basket tube walls. Localized corrosion may result in quicker penetration in specific areas that would cause shearing before it would result from general corrosion. Preliminary structural calculations for the UCF-WP indicate that shearing will occur from the weight of one fuel assembly when basket tube wall thicknesses have been reduced to less than 1 mm [CRWMS M&O 1995ag]. Assuming that all tube walls corrode at the same rate, buckling of the tube walls below will cause a fuel assembly to fall before shearing of the tube "floor" can occur.

Table 7-3. Waste Package Postclosure Critica[l F]ailure Modes and Effects Analysis (Continued)

Component	Component Function	Failure Modes	Failure Mechanisms	Required Conditions	Failure Effects	CPE	MOD	Comments
Basket Structure Note: Burnup Credit Design only. Flux Trap design will be evaluated once sufficient design information becomes available (continued)	Maintain spacing between intact fuel assemblies. (continued)	Basket tube "floor" fails by shearing from static load of single assembly. (continued)	Stress corrosion cracking,	Sensitized material AND Aqueous environment with aggressive species (Cl, S, etc.) AND Tensile stresses in excess of ½ yield AND Breach of all WP barriers	Tube "floor" shears, eliminating the gap between fuel assembly in tube and the one below it. Assembly is separated from the one below it by only the thickness of the basket material sandwiched between them. Eventually, collapse of all tubes will leave assemblies stacked on top of each other, separated only by remaining basket material (compression to ≈90% of original volume). (continued)	+	F	Testing of 304 and 304L stainless steels in J-13 well water, with addition of H_2O_2 to simulate radiolysis, found occurrence of SCC [NRC 1990b & 1991]. Slow strain rate and fatigue crack growth rate testing in J-13 well water has demonstrated that the 316 stainless steel material intended for use in this application is more resistant to SCC [ANL 1992]. However, these tests only measure relative susceptibility and do not make a determination as to whether SCC will actually occur in the given environment in the absence of cyclic stresses. Also, no testing has been performed to determine if introduction of boron into 316 stainless steel will have a detrimental or beneficial effect on SCC resistance. Effects of microbiologically influenced corrosion may increase susceptibility at lower temperatures (60-80°C).
			Pitting or crevice corrosion	Aqueous environment AND Breach of all WP barriers			F	Perforation due to pitting corrosion not sufficient to cause shearing due to small diameter of pits. However, crevice corrosion may be possible at tube corners depending on fabrication method, and in space between tubes. Buildup of corrosion products between tubes could cause increased stresses and denting of tubes. Effects of microbiologically influenced corrosion may increase susceptibility at lower temperatures (60-80°C).

Table 7-3. Waste Package Postclosure Criticality Failure Modes and Effects Analysis (Continued)

Component	Component Function	Failure Modes	Failure Mechanisms	Required Conditions	Failure Effects	CPE	MOD	Comments
Basket Structure Note: Burnup Credit Design only. Flux Trap design will be evaluated once sufficient design information becomes available (continued)	Maintain spacing between intact fuel assemblies. (continued)	Basket tube side-walls fail by buckling.	Stress corrosion cracking	Sensitized material AND Aqueous environment with aggressive species (Cl, S, etc.) AND Tensile stresses in excess of ½ yield	Tube side-walls buckle, causing a reduction or elimination in the gap between fuel assembly in tube and top of tube. Eventually, collapse of all tubes will leave assemblies stacked on top of each other, separated only by remaining basket material (compression to ≈90% of original volume).		F	Preliminary structural analyses for the 21 PWR UCF WP indicate that buckling of tube side walls due to the weight of the fuel assembly above will not result until the thickness has been reduced from 5 mm to 1.22 mm [CRWMS M&O 1995ag].
			General corrosion	Aqueous environment AND Breach of all WP barriers		+	F	Preliminary structural calculations indicate that buckling of side walls for maximally loaded (bottom-most) tube requires less material loss than shearing of tube "floor" supporting load of single assembly. [CRWMS M&O 1995ag].
			Pitting, and crevice corrosion	Aqueous environment AND Breach of all WP barriers			N	Localized corrosion would be expected to result in buckling before shearing, assuming all tube walls are attacked equally.
			Excessive mechanical loading.	Rockfall of sufficient mass impacting WP.			F	Preliminary structural calculations indicate that rockfall of insufficient mass to cause breach of WP barriers may still produce deformation of basket which results in buckling of some tubes (see section 6.5.4.3.2.3).
		Tube "floor" undergoes tensile failure due to static load of single assembly.	General corrosion	Aqueous environment AND Breach of all WP barriers	No Effect. See comments	N	N	Preliminary structural calculations for the 21 PWR UCF WP indicate that tube "floor" tensile failure requires the greater amount of material loss (reduction to 0.9 mm wall thickness) than shearing and buckling failures [CRWMS M&O 1995ag]. Therefore, this failure mode is not expected to occur.

Table 7-3. Waste Package Postclosure Criticality Failure Modes and Effects Analysis (Continued)

Component	Component Function	Failure Modes	Failure Mechanisms	Required Conditions	Failure Effects	CPE	MOD	Comments
Basket Structure Note: Burnup Credit Design only. Flux Trap design will be evaluated once sufficient design information becomes available (continued)	Maintain spacing between intact fuel assemblies. (continued)	Tube "floor" undergoes tensile failure due to static load of single assembly. (continued)	Stress corrosion cracking	Sensitized material AND Aqueous environment with aggressive species (Cl, S, etc.) AND Tensile stresses in excess of ½ yield AND Breach of all WP barriers	No Effect. See comments (continued)		N	Preliminary structural calculations for the 21 PWR UCF WP indicate that tube "floor" tensile failure requires greater amount of material loss than shearing and buckling failures [CRWMS M&O 1995ag].
			Pitting corrosion	Aqueous environment AND Breach of all WP barriers		N	N	Preliminary structural calculations for the 21 PWR UCF WP indicate that tube "floor" tensile failure requires greater amount of material loss than shearing and buckling failures [CRWMS M&O 1995ag].
			Excessive mechanical loading	Rockfall OR seismic event w/ WP support failure			N	Preliminary structural calculations for the 21 PWR UCF WP indicate that tube "floor" tensile failure requires greater amount of material loss than shearing and buckling failures [CRWMS M&O 1995ag].
	Maintain sufficient neutron absorber between intact assemblies to prevent criticality under moderated conditions.	Insufficient neutron absorber available between assemblies to maintain sub-criticality under moderated conditions with fuel that is isotopically capable of criticality.	General corrosion of basket material leaches boron.	Aqueous environment AND Breach of all WP barriers.	Criticality may occur if package interior is moderated, fuel is in appropriate geometry (intact fuel assemblies), and fuel is isotopically capable of criticality.	+	P	Amount of boron removed by general corrosion of the containing matrix would be proportional to the thickness of material lost. This assumes that the boron is uniformly distributed throughout the containing matrix, and that no selective leaching occurs. This is currently considered the dominant mechanism for removal of neutron absorber.

Table 7-3. Waste Package Postclosure Criticality Failure Modes and Effects Analysis (Continued)

Component	Component Function	Failure Modes	Failure Mechanisms	Required Conditions	Failure Effects	CPE	MOD	Comments
Basket Structure Note: Burnup Credit Design only. Flux Trap design will be evaluated once sufficient design information becomes available (continued)	Maintain sufficient neutron absorber between intact assemblies to prevent criticality under moderated conditions. (continued)	Insufficient neutron absorber available between assemblies to maintain sub-criticality under moderated conditions with fuel that is isotopically capable of criticality. (continued)	Stress corrosion cracking of basket material leaches boron.	Sensitized material AND Aqueous environment with aggressive species (Cl, S, etc.) AND Tensile stresses in excess of ½ yield AND Breach of all WP barriers. AND Boron concentrated at grain boundaries rather than evenly distributed throughout material.	Criticality may occur if package interior is moderated, fuel is in appropriate geometry (intact fuel assemblies), and fuel is isotopically capable of criticality. (continued)	+	F	SCC would only be expected to directly result in significant loss of boron if boron was found to be concentrated at the grain boundaries. However, not all grain boundaries would be attacked, and thus not all boron would be removed by this mechanism. If boron is not concentrated at grain boundaries, SCC will only fracture the containing matrix into smaller pieces, which would be expected to stay in place if sandwiched between two fuel assemblies. However, side walls could eventually crumble leaving little or no borated material between upper adjacent assemblies.
			Basket material loaded with insufficient absorber during fabrication.	Failure to alloy in sufficient boron during manufacture of basket material AND Failure of material chemistry certification process to detect low boron concentration.			F	This would require failure to add sufficient boron during alloying and failure of the material certification process to identify this deficiency. Due to the strict procedural controls on these activities for nuclear components, this is expected to be less likely than a through-wall defect in the outer barrier.

Table 7-3. Waste Package Postclosure Critica[illegible] Failure Modes and Effects Analysis (Continued)

Component	Component Function	Failure Modes	Failure Mechanisms	Required Conditions	Failure Effects	CPE	MOD	Comments
Fuel Assembly Spacer Grids	Maintain fuel rod pitch. (Note: Loss of this function post-closure is beneficial)	Fails to maintain fuel rods in original pitch.	Stress corrosion cracking	Sensitized material AND Aqueous environment with aggressive species (Cl, S, etc.) AND Tensile stresses in excess of ½ yield AND Breach of all WP barriers	Fuel rods consolidate to the bottom of the tube or package. Final compression with total basket collapse is to 30% of original volume, but in intermediate states of compression, between 80% and 30%, are possible depending on number of fuel assemblies and number of basket tubes collapsed.		F	Studies of zircaloy fuel cladding performance in inert, dry storage environments indicate that neither tensile stresses nor aggressive species known to cause SCC in zircaloy (Cl, I, Cs) are available in sufficient amount to produce SCC [PNL 1987a]. Therefore, it is not expected that the spacers will fail by this mechanism. Some old fuel was fabricated with Inconel or stainless steel spacers. These spacers may be susceptible to SCC (see basket collapse comments).
			Delayed hydride cracking	Aqueous environment AND Breach of all WP barriers			N	Studies on zircaloy cladding indicate that conditions required for delayed hydride cracking are more severe than for SCC. Failure by this mechanism is unlikely. [PNL 1987a]
			Crevice corrosion	Aqueous environment AND Breach of all WP barriers			F	Stainless steel and inconel spacers may be susceptible to crevice corrosion at the spacer/fuel rod interface. Zircaloy spacers would be expected to be resistant to this mechanism.
			Excessive mechanical loading	Rockfall OR Weight of upper assemblies			F	Rockfall on a package with degraded barriers may be sufficient to crush some fuel assemblies depending on the mass of the rock and the location of the assembly. The weight of upper assemblies in a collapsed basket may also cause this to occur to assemblies in the lower part of the WP.
			General corrosion	Aqueous environment AND Breach of all WP barriers			F	Results in slow thinning of spacer ligaments supporting fuel rods until the weight of the rod can no longer be supported. Expected to be the dominant mechanism if SCC is eliminated. Rate of thinning for zircaloy spacers much less than that for stainless steel.

Table 7-3. Waste Package Postclosure Criticality Failure Modes and Effects Analysis (Continued)

Component	Component Function	Failure Modes	Failure Mechanisms	Required Conditions	Failure Effects	CPE	MOD	Comments
Fuel Rod Cladding	Prevent transport of contained radionuclides	Gross structural failure	Stress corrosion cracking	Sensitized material AND Aqueous environment with aggressive species (Cl, Cs, I, etc...) AND Tensile stresses in excess of ½ yield AND Breach of all WP barriers	Maximum consolidation of fuel given collapse of basket tubes. Fissile materials available for transport given gross structural failure or corrosion holes in lower portion of all WP barriers. See comments for applicable mechanisms.	-	F	Studies of fuel performance in inert, dry storage indicate that neither tensile stresses nor aggressive species known to cause SCC in zircaloy (Cl, I, Cs) are available in sufficient amount to produce SCC [PNL 1987a]. Therefore, it is not expected that zircaloy cladding will fail by this mechanism. Older fuel assemblies with stainless steel cladding may be susceptible to SCC.
			Delayed hydride cracking	aqueous environment AND Breach of all WP barriers			N	Studies indicates that conditions required for delayed hydride cracking are more severe than for SCC. Cladding would fail by SCC first. [PNL 1987a]
			General corrosion	aqueous environment AND Breach of all WP barriers			F	This would result in gradual thinning of the cladding wall thickness. Studies indicate that general corrosion rate is very low for zircaloy cladding [CRWMS M&O 1995m] and would likely be similar to that of the stainless steel basket material for stainless steel cladding. In the absence of SCC, or splitting due to fuel oxidation, this would be expected to be the dominant mechanism for general collapse of the fuel rod structure.

Table 7-3. Waste Package Postclosure Criticality Failure Modes and Effects Analysis (Continued)

Component	Component Function	Failure Modes	Failure Mechanisms	Required Conditions	Failure Effects	CPE	MOD	Comments
Fuel Rod Cladding	Prevent transport of contained radionuclides	Gross structural failure	Creep rupture	High temperature, Cladding stress, AND Internal pressure	Maximum consolidation of fuel given collapse of basket tubes. Fissile materials available for transport given gross structural failure or corrosion holes in lower portion of all WP barriers. See comments for applicable mechanisms.		F	Zircaloy cladding can fail by creep rupture under a narrow band of temperature and internal stress. The cladding fails by pinhole rupture and mechanism ceases due to loss of internal pressure [PNL 1987a; CRWMS M&O 1995m].
			Splitting	Breached cladding AND High Cladding Temperature			F	Cladding may split, or "unzip," once initially breached as a result of swelling of the fuel pellets due to oxidation of the UO_2 to U_4O_9 and then to U_3O_8. However, this mechanism requires cladding temperatures in excess of 400K as well as an oxidizing environment and a pre-existing cladding breach to initiate. The conditions necessary to cause this mechanism will only be available for packages which contain failed fuel rods and that are breached within the first few hundred years [CRWMS M&O 1995m].
			Excessive Mechanical Loading	Rockfall			F	Rockfall on a package with severely degraded barriers may be sufficient to crush or sever some fuel rods depending on the mass of the rock.
			Pre-existing cladding defect	Failure during reactor operation, storage, or transportation.			F	Less than 1% of fuel rods are expected to contain pre-existing defects which resulted during reactor operations, transportation, or dry-storage [CRWMS M&O 1995m].

7.4.2 Definition of Event Probabilities

The following sections provide a detailed description of the estimation of the basic event probabilities and the PDFs used as input for the fault tree quantification. This analysis is described in detail in supporting design analyses (CRWMS M&O 1995v and 1995w). Event identifiers used to abbreviate the full description in the analysis of the fault tree are given in parentheses. Event probabilities have been summarized in Table 7-4 for specific times at which the fault tree was quantified.

7.4.2.1 Flow Defining Events

The four initiating events defined in Table 7-3 describe a state of water flow into or flooding of the system. It is assumed that this state continues indefinitely once initiated. This assumption is considered conservative since it is possible that any increased state of flow or flood will eventually revert to something like the original state before the enhanced corrosion mechanism has completed the corrosion of the WP component (barrier or basket). This assumption also simplifies the analysis by allowing a single event to represent a variable process. The time-to- occurrence of these events is characterized by a separate PDF for each event, denoted by f_1. The description of flow defining events is intended to be qualitative in nature. The effects of the various flow defining events on the WP are treated more quantitatively in Section 7.4.2.2.

The first of the four flow defining events, low infiltration, is defined as the currently estimated nominal infiltration rate of 0.1 mm/year (0.004 in./year) (Flint and Flint 1994) for surface water to the repository horizon (but not necessarily incident on a WP). This low infiltration rate would have to be increased by a factor of 10 to supply the volume of water necessary to accomplish the degradation of the WP barriers, filling of the WP interior, and leaching of the neutron absorber. It is postulated that such an enhanced flow rate may occur in the next 10,000 years. Therefore, the time-to-occurrence of the low infiltration event will be represented by a uniform distribution between 1,000 and 10,000 years, which can be expressed in units of per year as:

$$f_1(t) = 1/9000 \qquad 1000 < t < 10000 \qquad \text{(Equation 7-2)}$$

The high infiltration event is considered to be an order of magnitude greater than the low infiltration in terms of infiltration rate. Such a high rate would require a large increase in rainfall due to significant climate change, which is may possibly occur in the next 2,000 to 100,000 years (a range chosen to include several glacial extremes). As with the low infiltration case, a uniform distribution is used, again expressed in units of per year:

$$f_1(t) = 1/98000 \qquad 2000 < t < 100000 \qquad \text{(Equation 7-3)}$$

Table 7-4. Summary of WP Fault Tree Event Probabilities as a Function of Time

Time Emplaced (years)	Basic and Conditional Event Probabilities					
	Holes	Crackswp	Geometry	Climate & Tectonic	WPB&LDL	WPB&LDH
UCF WP Barriers Provide Temporary Protection Against Moderator Entry						
20,000	1.00×10^{-2}	6.95×10^{-2}	1.00	0	0	1.16×10^{-5}
40,000	1.00×10^{-2}	6.95×10^{-2}	1.00	7.89×10^{-8}	6.90×10^{-6}	1.43×10^{-3}
80,000	1.00×10^{-2}	6.95×10^{-2}	1.00	1.20×10^{-6}	2.48×10^{-2}	1.44×10^{-2}
UCF WP Barriers Given No Credit For Preventing Moderator Entry						
20,000	1.00×10^{-2}	6.95×10^{-2}	1.00	0	3.13×10^{-3}	2.45×10^{-4}
40,000	1.00×10^{-2}	6.95×10^{-2}	1.00	1.62×10^{-7}	2.22×10^{-2}	3.29×10^{-3}
80,000	1.00×10^{-2}	6.95×10^{-2}	1.00	1.55×10^{-6}	4.71×10^{-2}	1.85×10^{-2}
CF WP Barriers Provide Temporary Protection Against Moderator Entry						
20,000	1.00×10^{-2}	7.45×10^{-2}	1.00	0	0	3.74×10^{-3}
40,000	1.00×10^{-2}	7.45×10^{-2}	1.00	2.01×10^{-7}	2.25×10^{-3}	1.64×10^{-2}
80,000	1.00×10^{-2}	7.45×10^{-2}	1.00	1.97×10^{-6}	5.48×10^{-2}	3.76×10^{-2}
CF WP Barriers Given No Credit For Preventing Moderator Entry						
20,000	1.00×10^{-2}	7.45×10^{-2}	1.00	1.64×10^{-7}	6.50×10^{-2}	1.21×10^{-3}
40,000	1.00×10^{-2}	7.45×10^{-2}	1.00	7.40×10^{-7}	6.50×10^{-2}	2.47×10^{-2}
80,000	1.00×10^{-2}	7.45×10^{-2}	1.00	2.97×10^{-6}	6.50×10^{-2}	4.59×10^{-2}

Finally, flooding of the repository would require some event that could cause the water table to rise to the repository horizon. As no evidence of such an extreme rise has ever occurred (National Research Council 1992), it is postulated that such a rise would require a severe tectonic event or a drastic change to a much wetter climate. For this assessment, it is assumed that such events are approximately as likely as the large volcanic activity that produced Yucca Mountain and the neighboring topographic features 10 to 13 million years ago (see Table 7-3). Therefore, the occurrence of flooding by either of the above two events will be represented by a monotonically increasing triangular distribution with the upper limit at 10,000,000 years:

$$f_1(t) = 2x10^{-14}t \qquad 10{,}000<t<10{,}000{,}000 \qquad \text{(Equation 7-4)}$$

where t is expressed in years and f_1 is expressed in units of per year. The occurrence of such an event as described above is considered non-credible prior to 10,000 years, thus the above PDF is normalized such that t = 0 at 10,000 years.

7.4.2.2 Corrosion Events

In this analysis, criticality cannot occur until the WP barriers have been breached by corrosion, allowing moderator entry and leaching of the basket material containing the neutron absorber. The PDF for corrosion penetration of the WP barriers will be designated as f_2. The PDF for corrosion of the basket structure containing the neutron absorber is designated as f_4 for both the UCF and CF WP. The PDF for the corrosion of the CF canister shell is designated f_3; note that the UCF WP does not have a third shell and thus, does not need to have an f_3 PDF defined.

The functional form of the PDFs for corrosion will be assumed to be the three-parameter Weibull distribution, which is often used in reliability analysis to model corrosion related failures (Modares 1993). The PDF of the Weibull distribution is given by:

$$f(t) = \frac{\beta}{\alpha}\left(\frac{t-\theta}{\alpha}\right)^{\beta-1}\exp\left[-\left(\frac{t-\theta}{\alpha}\right)^{\beta}\right] \qquad \text{(Equation 7-5)}$$

where α, β, and θ represent the scale, shape, and location parameters respectively (all >0) and $t \geq \theta$. The associated Weibull CDF is given by:

$$F(t) = 1 - \exp\left[-\left(\frac{t-\theta}{\alpha}\right)^{\beta}\right] \qquad \text{(Equation 7-6)}$$

for $t \geq \theta$. For values of $t<\theta$, both f(t) and F(t) equal zero.

7.4.2.2.1 Barrier Corrosion

Distributions describing the time to penetration of the WP barriers have been developed for two basic conditions: intermittent and continuous wetting. These distributions are derived from data and models currently available on the pitting corrosion behavior of carbon steel and Alloy 825 barrier materials, and the predicted WP surface temperature histories for various locations in a low thermal loading repository (CRWMS M&O 1995v and 1995w). The Weibull distribution f_2 (failure of the WP barriers) parameters for two basic environmental conditions, intermittent and continuous wetting of the WP barrier, are summarized in Table 7-5.

Table 7-5. Weibull Parameters for WP Barrier Corrosion PDFs

Condition	α	β	θ
Intermittent Wetting	5030	1.7	30,000
Continuous Wetting	425	0.9	8100

As mentioned above, the CF WP also required the use of an additional PDF (f_3) for corrosion of the Type 316L stainless steel shell. As before, this PDF was developed using currently available pitting corrosion data for this material (CRWMS M&O 1995v and 1995w). The CF canister shell Weibull distribution f_3 parameters for the two basic environmental conditions, intermittent and continuous wetting of the CF shell, are summarized in Table 7-6.

Table 7-6. Weibull Parameters for CF Shell Corrosion PDFs (CF WP Only)

Condition	α	β	θ
Intermittent Wetting	3571	1.0	3962
Continuous Wetting	1786	1.0	1981

The intermittent wetting PDFs were used for the low infiltration sequence, and the continuous wetting PDFs were used for the high infiltration and flooding sequences.

7.4.2.2.2 Neutron Absorber Leaching

The neutron absorber leaching PDFs, f_4, for the UCF and CF WPs, were developed from currently available data on the general corrosion behavior of Type 316 stainless steel and aluminum Alloy 1100, respectively (CRWMS M&O 1995v and 1995w). In developing the distribution, it was conservatively assumed that all neutron absorber was lost by the time 60 percent of the basket material thickness was removed by general corrosion. The Weibull distribution parameters for the two basic environmental conditions, intermittent and continuous wetting of the basket material, are summarized in Table 7-7.

Table 7-7. Weibull Parameters for Neutron Absorber Leach PDFs

Condition	UCF WP			CF WP		
	α	β	θ	α	β	θ
Intermittent Wetting	39343	2.1	4800	320	2.1	286
Continuous Wetting	19671	2.1	2400	160	2.1	143

The intermittent wetting PDFs were used in the convolution of the low and high infiltration sequences, and the continuous wetting PDFs were used for the flooding sequences.

7.4.2.3 Fraction of Fuel With Sufficient Fissile Material

After all the hazard events that are necessary for a criticality event (breach of all barriers, absorber leach, and internal flooding) have occurred, there is still one fundamental requirement for each scenario: the SNF must have the right combination of initial enrichment and burnup to be capable of criticality. The criticality capability of a WP full of SNF assemblies having specific characteristics in a given geometry is determined by the value of k_{eff}. The physical requirement for subcriticality is $k_{eff}<1.0$. For licensing calculations, it is required by 10 CFR 60.131 that $k_{eff}\leq0.95$, to provide a 5 percent safety factor. In addition, there is usually an additional amount (typically up to 0.06) to be subtracted to account for bias and uncertainty in the calculational method. However, for this analysis, a k_{eff} of 0.95 will be used as the criterion for determining criticality potential. This provides a conservative probabilistic estimate of what could actually happen, but not necessarily conservative enough to license a WP with respect to a deterministic estimate of worst case performance.

The deterministic calculation of k_{eff} as a function of time for the MGDS design basis PWR fuel assemblies has been previously discussed in Sections 6.3.3.3.1 and 6.5.3.3 for the 21 PWR UCF and CF WP designs respectively. This information was used in conjunction with a statistical tabulation of SNF with respect to k_∞ (CRWMS M&O 1994e) to determine the fraction of fuel capable of exceeding the k_{eff} criterion of 0.95 as a function of time (CRWMS M&O 1995v and 1995w). The results are shown in Figure 7-2. This factor will be used as a multiplier on each of the three conditional breach and leach PDFs produced in Section 7.4.2.4 for the CF and UCF WPs to determine the corresponding breached, leached, and criticality capable cumulative probability plots.

7.4.2.4 Convolution of PDFs

The PDF for a given sequence of flow, breach, and leach events was obtained from the convolution of f_1, f_2, and f_4 for the UCF WP, and f_1, f_2, f_3, and f_4 for the CF WP. This convolution is computed by a Monte-Carlo numerical integration that used a Mathcad 5.0+ worksheet to randomly sample the CDF for each distribution and sum the times to reach the defined flow (or flood) condition, to breach the WP, and to leach 60 percent of the boron. Two hundred and fifty thousand trials were performed (CRWMS M&O 1995v and 1995w). Table 7-8 below summarizes the distributions used in the convolution of each sequence (shown with an X). The resulting PDF for a given sequence was then adjusted using the appropriate curve for the time-dependent fraction of fuel capable of exceeding the 5 percent k_{eff} criterion, and numerically integrated to produce a CDF. Probabilities of occurrence for each of the four conditional breach, leach, and criticality capable event sequences were obtained from these CDFs at 20,000, 40,000, and 80,000 years after emplacement, and are summarized in Table 7-4.

In addition, due to the current level of uncertainty on the pitting corrosion behavior of Alloy 825, it was decided to investigate a worst-case scenario in which the WP barriers were penetrated in a relatively short period of time compared to the other events in the sequence. This was performed for each of the event sequences by simply eliminating f_2 from the convolution, effectively producing conditional breach and leach distributions that consider the WP barriers to be instantly breached upon the occurrence of the initiating event. The results are summarized in Table 7-4.

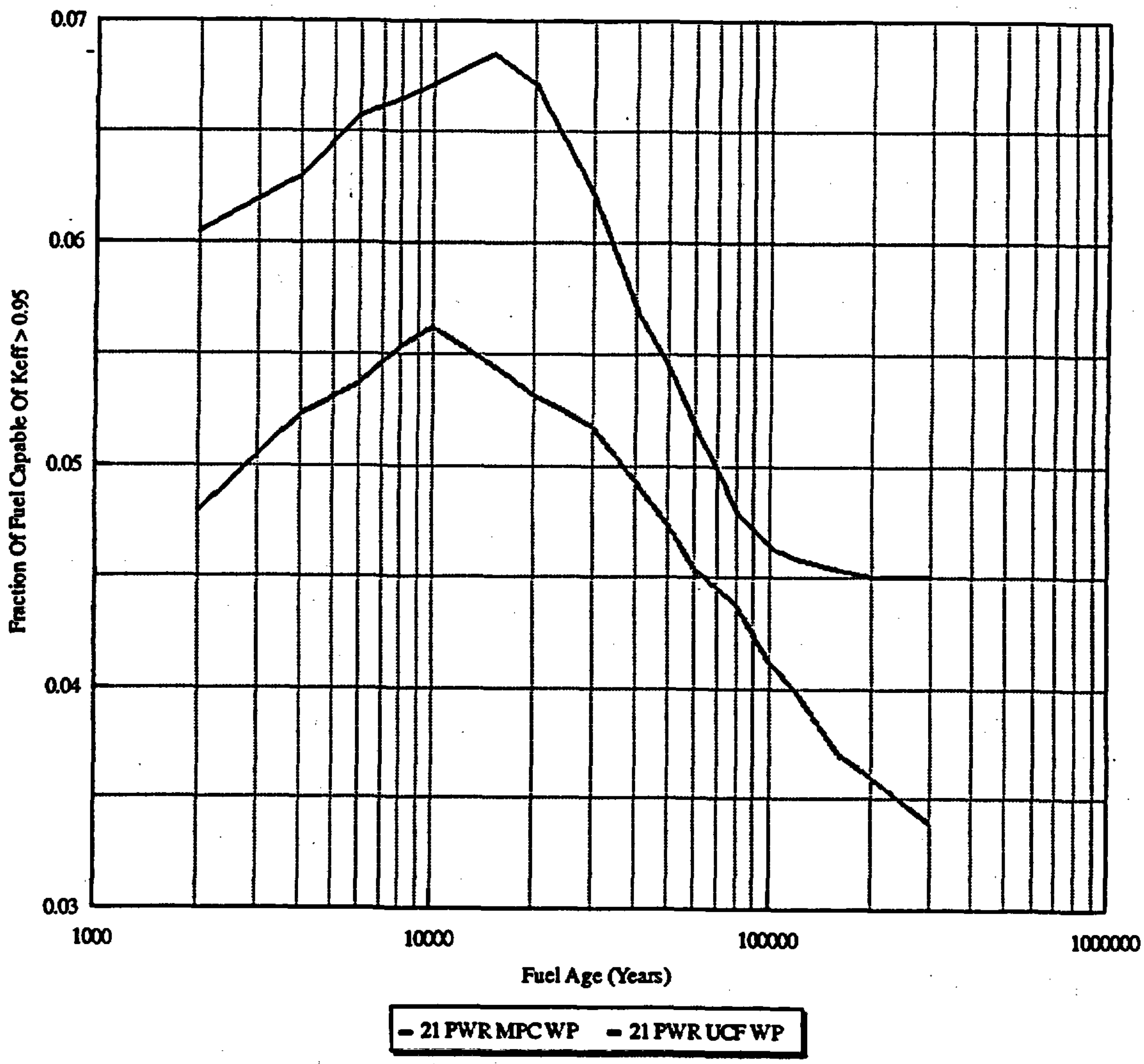

Figure 7-2. Fraction of SNF Capable of $k_{eff}>0.95$ as a Function of Time (Intact 21 PWR CF and UCF WP Designs without Additional Neutron Absorber)

Table 7-8. Convolutions Performed To Obtain Conditional Fault Tree Event PDFs

PDF Used In Convolution		Barrier			No-Barrier		
		wpb&ldl	wpb&ldh	Climate & Tectonic	wpb&ldl	wpb&ldh	Climate & Tectonic
Flow Defining Events $f_1(t)$	Low Infiltration	X			X		
	High Infiltration		X			X	
	Flooding			X			X
WP Barrier Breach $f_2(t)$	Int. Wetting	X					
	Cont. Wetting		X	X			
CF Breach $f_3(t)$ (CF WP only)	Int. Wetting	X					
	Cont. Wetting		X	X			
Absorber Leach $f_4(t)$	Int. Wetting	X	X		X	X	
	Cont. Wetting			X			X

7.4.2.5 Probability of Flow Concentration on WP

In order for the sequences involving water infiltration (as defined above) to be effective in corroding and filling a WP, the nominal infiltration flow must be concentrated over some localized position on the package (typically by the location of a flowing fracture). This localized flow serves both to generate the corrosion hole and to channel the water into that hole, from where it can fill the lower half of the package and leach the neutron absorber. Based on currently available information about the occurrence of fractures in the TSw2 rock unit, the probability that a WP has at least one fracture over it that is capable of concentrating the infiltration flow (fault tree basic event "crackswp") has been estimated as 0.070 for the UCF WP and 0.075 for the CF WP (CRWMS M&O 1995v and 1995w). This probability is assumed to be a property of the repository, which remains constant over at least 100,000 years. Therefore, it was not included in the convolutions performed in Section 7.4.2.4.

7.4.2.6 Probability of Sufficient Moderator in WP

For the overhead dripping scenarios, there must be holes around the middle of the package, but not the lower part. The most likely location is on the upper surface that has the greatest exposure to dripping water. The conditional probability of such a hole configuration (fault tree basic event "holes"), given that there is sufficient corrosion to produce the holes in the first place, is the product of the conditional probability of holes around the middle, given that there are holes in the top, and the conditional probability of no holes in the lower half, given that there are holes around the middle. Both of the above conditional probabilities have been estimated to be 0.1. This is actually quite conservative for the latter probability, since half of the weld around the lid will be in the lower, sub-merged, half of the horizontal package, and this weld is more likely to corrode and leave a hole to prevent ponding. On the other hand, there is a possibility that the leached/corroded material could plug up such holes, so that subsequent ponding could be supported even if the initial hole configuration were not favorable to ponding.

7.4.2.7 Probability That SNF Maintains Geometry Required For Criticality

Since criticality of SNF assemblies will require nearly full moderation, there can be no criticality if the basket and assembly hardware fail in such a way that the fuel rods can collapse into a consolidated configuration that does not permit sufficient water between the rods. Such a collapse would generally require the corrosion of the fuel cladding or spacer grids in each assembly. However, it has been conservatively assumed that the fuel assemblies will always maintain a geometry that supports optimal moderation for the time frame covered by the current analysis. Therefore, this event ("geometry" in the fault tree) has been assigned a probability of 1.0.

7.4.3 Fault Tree Analysis

In this section, the basic and time-dependent conditional event probabilities developed in Section 7.4.2 are used as input to quantify the fault tree developed in Section 7.2.2. The fault tree was evaluated at the times after emplacement for which conditional event probabilities were quoted in Table 7-4. The fault tree top event (cumulative probability of criticality per package) was quantified using CAFTA Version 2.3 fault tree analysis software (SAIC 1993). The results of the quantification at each point in time are given in Table 7-9 for the UCF WP, with and without credit for the WP barriers. Similarly, the results of the fault tree quantification for the CF WP are given in Table 7-10. At each point in time, the individual sequences (known as "cutsets") that contribute to the top event probability are listed in the table in the order of their contribution to the total probability.

7.4.4 Comparison and Conclusions

With the top event quantified at several points in time for both the UCF and CF WPs, the time effects on the cumulative probability of a criticality event can be evaluated. Figure 7-4 displays the cumulative per package probability of criticality for both the barrier and no-barrier cases, which were evaluated for the UCF and CF WPs. The number of WP criticalities expected to occur by a given time can be approximated from Figure 7-3 simply by multiplying the cumulative probability at that time by the number of packages. In particular, for a repository assumed to contain 10,000 CF WPs, the expected number of CF WP criticalities that would have occurred by 40,000 years is 0.14 with the WP barriers credited, and 0.68 without crediting the WP barriers. In contrast, the expected number of criticalities at 40,000 years for an assumed 10,000 UCF WP repository would be 0.012 with barrier credit, and 0.18 without barrier credit.

Table 7-9. Summary of Top Event Probabilities and Cutsets for UCF WP

Time (Years)	Top Event Probability	Cutset Probabilities and Event Sequences
(with Barrier Credit)		
20,000	8.07E-09	8.07E-09 Crackswp Geometry Holes WPB&LDH20K
40,000	1.15E-06	9.92E-07 Crackswp Geometry Holes WPB&LDH40K
		4.80E-09 Crackswp Geometry Holes WPB&LDL40K
		7.89E-08 Tectonc40K Geometry
		7.89E-08 Climate40K Geometry
80,000	2.96E-05	1.72E-05 Crackswp Geometry Holes WPB&LDL80K
		1.00E-05 Crackswp Geometry Holes WPB&LDH80K
		1.20E-06 Tectonc80K Geometry
		1.20E-06 Climate80K Geometry
(without Barrier Credit)		
20,000	2.34E-06	2.17E-06 Crackswp Geometry Holes WPB&LDL20K
		1.70E-07 Crackswp Geometry Holes WPB&LDH20K
40,000	1.80E-05	1.54E-05 Crackswp Geometry Holes WPB&LDL40K
		2.28E-06 Crackswp Geometry Holes WPB&LDH40K
		1.62E-07 Tectonc40K Geometry
		1.62E-07 Climate40K Geometry
80,000	4.86E-05	3.27E-05 Crackswp Geometry Holes WPB&LDL80K
		1.28E-05 Crackswp Geometry Holes WPB&LDH80K
		1.55E-06 Tectonc80K Geometry
		1.55E-06 Climate80K Geometry

Table 7-10. Summary of Top Event Probabilities and Cutsets for CF WP

Time (Years)	Top Event Probability	Cutset Probabilities and Event Sequences				
(with Barrier Credit)						
20,000	2.79E-06	2.79E-06	Crackswp	Geometry	Holes	WPB&LDH20K
40,000	1.43E-05	1.22E-05	Crackswp	Geometry	Holes	WPB&LDH40K
		1.68E-06	Crackswp	Geometry	Holes	WPB&LDL40K
		2.01E-07	Tectonc40K	Geometry		
		2.01E-07	Climate40K	Geometry		
80,000	7.28E-05	4.09E-05	Crackswp	Geometry	Holes	WPB&LDL80K
		2.80E-05	Crackswp	Geometry	Holes	WPB&LDH80K
		1.97E-06	Tectonc80K	Geometry		
		1.97E-06	Climate80K	Geometry		
(without Barrier Credit)						
20,000	5.77E-05	4.84E-05	Crackswp	Geometry	Holes	WPB&LDL20K
		8.99E-05	Crackswp	Geometry	Holes	WPB&LDH20K
		1.64E-07	Tectonc20K	Geometry		
		1.64E-07	Climate20K	Geometry		
40,000	6.83E-05	4.84E-05	Crackswp	Geometry	Holes	WPB&LDL40K
		1.84E-05	Crackswp	Geometry	Holes	WPB&LDH40K
		7.40E-07	Tectonc40K	Geometry		
		7.40E-07	Climate40K	Geometry		
80,000	8.86E-05	4.84E-05	Crackswp	Geometry	Holes	WPB&LDL80K
		3.42E-05	Crackswp	Geometry	Holes	WPB&LDH80K
		2.97E-06	Tectonc80K	Geometry		
		2.97E-06	Climate80K	Geometry		

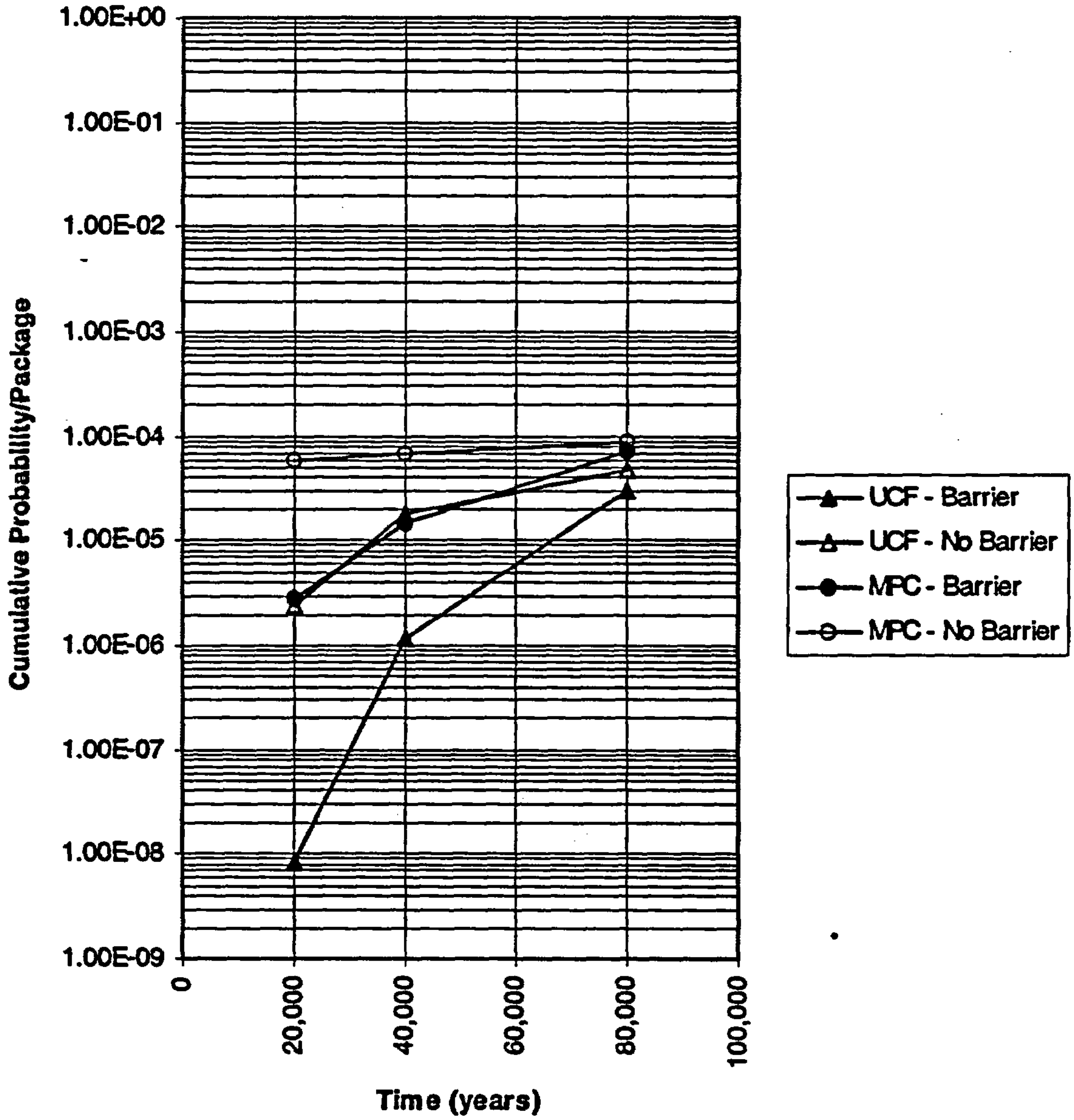

Figure 7-3. WP Internal Criticality Event Cumulative Probability per Package as a Function of Time (21 PWR CF and UCF WPs)

The difference in probability of long-term criticality between CF WP and UCF WP is primarily due to the assumption that the CF WP basket absorber material will be borated aluminum Alloy 1100, while the UCF WP has a more corrosion resistant borated stainless steel basket. This is because the mean time to uniformly corrode 60 percent of the borated aluminum (and thus remove the neutron absorber) is estimated to be only a few hundred years after breach of the WP barriers, whereas several thousand years were required to uniformly remove 60 percent of the borated SS steel. At earlier times, the relative disadvantage of the CF WP is much larger. It is interesting to note that the relative advantage of the UCF WP is greater for the low infiltration case than for high infiltration. This is because basket leach time (which is the primary difference between the UCF WP and the CF WP) for the UCF WP at high infiltration is only half of what it is at low infiltration. It is evident that refinements in the estimates of corrosion time PDFs and/or flow event PDFs could significantly affect the estimate of the relative advantage of the UCF WP.

It is clear from a review of the cutsets presented in Tables 7-9 and 7-10 that the dominant sequences contributing to the rise in the probability of criticality during the first 80,000 years are those involving water dripping on a WP from an overhead fracture. This suggests that actions be taken to identify and avoid placement of WPs in such areas would significantly reduce the probability that a WP would be located under such a fracture, and thus reduce the rate and degree to which the overall WP criticality probability rises. These conclusions, however, are subject to validation and/or refinement of the assumptions made in the analysis regarding flowing fracture frequency.

It is also evident from the cumulative distributions shown in Figure 7-3 that the rate at which the barrier is assumed to be breached has a significant effect on the rate at which the criticality probability rises over the first 80,000 years, but little effect thereafter. The effect in the early years is primarily due to the uncertainty in the time-to-breach of the WPs located below flowing fractures. However, in the later years, further increases in the probability of WP criticality are primarily governed by the occurrence of events that produce repository flooding. As the time frame for occurrence of these events is on the order of several million years, and the range uncertainty in barrier performance spans at most only a few thousand years, there is little effect on the overall probability of criticality due to sequences initiated by flooding.

7.4.5 Future Work

As is evident from comparison of the simple model detailed above with the number of component failure modes presented in the FMEA (Table 7-3), additional modeling work remains to be performed. Once the WP barriers have been breached and the internal inert environment is lost, all of the components interior to the package will begin to degrade simultaneously. The current evaluation only examines the effects of boron loss on the criticality potential of an otherwise permanently intact basket structure. In reality, the same mechanisms that cause the boron loss will also cause the collapse of the basket structure long before most of the boron has been removed from between the assemblies. This will result in a slight reactivity insertion, but not as great as if all of the boron had been removed. Similarly, degradation mechanisms will also be acting on the fuel assemblies and the lower portion of the WP in a manner that will have a negative effect on WP k_{eff}. Failure of the fuel assembly spacer grids will result in consolidation of the fuel rods and the eventual formation of corrosion holes in the lower portion of the package will prevent ponding. Future evaluations will apply accepted corrosion models for the materials of each WP component to determine the PDF for each credible component failure mode in the FMEA. In this manner, the WP geometries most likely to exist at a given point in time may be identified from the spectrum of possible configurations for deterministic evaluation of their criticality potential.

The current evaluation also conservatively assumes that all boron has been removed from the basket once 60 percent of the tube wall thickness has corroded away. This assumption allowed the use of existing curves detailing the time dependence of k_{eff} for non-borated, as-built CF and UCF WP geometries to develop the time-dependent fraction of fuel capable of exceeding 0.95 k_{eff}. In reality, the fraction of fuel capable of exceeding 0.95 k_{eff} will be proportional to the amount of boron removed. The current evaluation also assumes that the entire population of SNF is indiscriminately loaded into one type of package, with no derating or blending of any packages. In reality, the application of loading curves will result in many low burnup, high initial enrichment assemblies either blended with high burnup low enrichment assemblies, or possibly simply loaded into packages that are only filled to 1/3 or ½ capacity before being sealed. Credit for loading curves will greatly reduce the fraction of fuel that will be capable of criticality in all geometries, both degraded and undegraded.

Finally, the current evaluation makes some generalized assumptions about the environment. This evaluation assumes that intruding water contacting a WP is a single event, and that this contact is continuous for all time once it has begun. In reality, many factors will affect the time when liquid water will be available to the WP, the amount that will be available, and the duration of the flow. In addition, degradation of WP components may occur as a result of high relative humidity prior to the availability of significantly flowing source of water. Explicit consideration of the possible variation in the rate at which water contacts the WP, as well as the aperture size of WP breaches, will be necessary to determine a PDF for the time to fill the WP.

7.5 PROBABILISTIC EVALUATION OF POSTCLOSURE EXTERNAL CRITICALITY EVENTS

Potential external configurations will follow general collapse of the WP structure and involve dissolution, transport, and re-precipitation of fissile nuclei, possibly in concentrated form and/or separated from the absorbers originally present in the SNF or WP. Examples of possible external configurations include:

- Plutonium is precipitated along the drift floor with all absorbers (boron, fission products, and actinides) removed.
- Plutonium in particulate form is transported from several packages and concentrated at a low point in the drift, possibly where water drains into a fracture that is too narrow to permit penetration by the particles.
- Plutonium is transported into fractures in the drift floor and precipitates on the fracture walls, thereby forming a network of plutonium coated fractures. Parametric criticality analyses will be required for probable fracture configurations, taking into account the moderating capability of tuff, ground water, and combinations of the two.

For the next revision of the WP criticality risk analysis, the probability and consequences of the above processes will be quantified by survey of expert judgement (both published and unpublished), particularly with respect to the following issues:

- Corrosion of metals (barriers, basket, etc.)
- Erosion and dissolution of glass
- Processes for the removal of ions from a porous matrix
- Transport of nuclides (separating absorbers from fissile materials being the most serious).

The results of these studies will be presented in terms of statistical summaries (probability distributions, expected criticalities, etc.), which reflect the irreducible uncertainties in the following:

- Infiltration rate as a function of time
- Concentration of infiltration by fractures
- Effectiveness of backfill in diverting fracture drippings
- Fracture of near field rock by the WP heat pulse
- Precipitation on fracture walls (of unknown asperity)
- Concentration from several packages
- Dissolution rates/solubilities of essential components (particularly neutron absorbers).

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8. PERFORMANCE ANALYSIS

This section presents Waste Package (WP) and Engineered Barrier Segment performance analyses. These analyses are closely focused in that they are design related and generally consider individual processes and degradation of a single component. Many of these performance analyses are developed in collaboration with Engineered Barrier System/Near Field Environment Performance Assessment and serve as input to the Total System Performance Assessments.

Key performance parameters of materials and/or components must be modeled with adequate confidence to show compliance with applicable requirements. These models should be deterministic and/or mechanistic to provide confidence in their validity over repository time periods.

The models for performance analysis and performance assessment are placed in a model hierarchy. At the base of the hierarchy, and providing the technical basis for the performance calculations, are the models that quantitatively characterize the performance parameters or responses of the materials or design to the repository environment. At higher levels, these performance parameter models may be simplified, but must remain defensible at the mechanistic model level. The higher-level analyses provide feedback for testing and design. The performance assessment effort is thus closely linked to the design effort, particularly for the selection of materials, material geometries, and environmental scenarios.

The following sections describe advances made in WP performance analysis. Components discussed include the backfill, invert, engineered components of the WP, spent fuel cladding, and control rods. Additional work on oxidation and dissolution of waste forms is described in Section 4.3.

The work reported was part of advanced conceptual design for the WP and Engineered Barrier Segment. At the end of advanced conceptual design, a more definitive design will be available. This will permit more detailed and complete analysis to be undertaken in the phase of the design process leading to viability assessment. These future investigations will include more complete process-level models. In addition, sensitivity and uncertainty studies will be undertaken in parallel and consistently with Total System Performance Assessment evaluations. Since limited data and conceptual designs were available at the time of this work, conclusions or judgments about waste package performance are necessarily provisional and subject to revision.

8.1 UNDERGROUND FACILITY COMPONENTS

The Engineered Barrier Segment, includes backfill (if present) and invert, as well as the WP. Types of backfills and their effects on WP life are discussed here. Several types of backfill have been proposed. An evaluation in 1994 concluded that the best approaches are to use no backfill or a backfill of crushed tuff (CRWMS M&O 1994g). More elaborate backfills, with sloping layers of fine and coarse material, have been considered as a means for diverting dripping groundwater from the waste, but appear to be impractical to construct in an emplacement drift. Evaluations of backfill are summarized in this section (CRWMS M&O 1994g). Work on backfill is ongoing, particularly as regards its effect on long-term release (CRWMS M&O 1995ae). Future reports are expected to incorporate more recent results.

New WPs are quite strong, but degradation of the barriers will gradually reduce their strength. For a degraded WP, rock fall appears to be the most likely cause of WP failure by mechanical stress. The backfill is not strong enough to function as ground support, but it nevertheless protects against rock fall. Protection occurs by four mechanisms: (1) reducing the distance through which rocks can fall and thus limiting the kinetic energy they acquire, (2) absorbing energy by crushing and flowing, (3) distributing the force of the rock fall over a larger area of the WP, and (4) transferring force to the drift walls and thus reducing the forces applied to the WP.

Backfills can also have strong effects on heat dissipation. With no backfill, radiation will be the primary mechanism for transporting heat from the WP to the drift wall, and the temperature drop between these two surfaces will be small. Backfill would block the radiation, however, so substantially higher WP temperatures are expected if a backfill is used, especially a particulate backfill that is emplaced early. A large temperature drop between the WP and drift wall will help control atmospheric corrosion. At high relative humidities, thin films of liquid water form on metal surfaces and corrosion can occur. A reduction of relative humidity at the WP will result if there is a temperature drop between the surface of the WP and the nearest liquid water. For no backfill or a gravel backfill, the nearest liquid water is likely to be at the drift wall. A gravel backfill will insulate the WP, increase the temperature drop, and decrease the relative humidity at the WP.

Backfill has the potential for controlling transport of radionuclides to the host rock. Radionuclides can move through the backfill by either advection or diffusion. Both can occur even in unsaturated media. There, transport from one particle to another occurs through thin, interconnected films of water on the surfaces of the particles. Measurements by Conca (Conca and Wright 1992) show that aqueous diffusion through adsorbed water is very slow, especially at low saturations, so diffusive aqueous release rates through a gravel backfill will be very low. Diffusive transport is also limited with no backfill because the only diffusion paths are through or on the surface of the waste package support. For any backfill, advective transport is possible if liquid water drips onto the waste.

Formal lists of candidate invert materials have not been compiled, but materials under consideration include crushed tuff and concrete emplaced as precast sections. Selection of invert material will interface with subsurface design. For example, if WPs are emplaced on railcars, a crushed tuff invert might require crossties between the rails, but the rails might be fastened directly to a concrete invert.

The invert could perform a variety of functions. It must support the WP and any emplacement hardware during the preclosure phase, and it should also provide support during the postclosure phase or at least break down predictably. To control radionuclide releases, it is beneficial to have the invert limit diffusion of nuclides to the host rock and to provide drainage to limit contact between water and the waste packages. Additional increases in performance might be achieved by the use of additives to promote sorption of radionuclides.

8.2 WP ENGINEERED COMPONENTS

WP components that have been analyzed include the containment barriers, fuel basket, fill gas, control rods, and spent nuclear fuel (SNF). Work on the engineered components is described in this section; work on waste forms is described in Sections 8.3 and 8.4.

8.2.1 Containment Barrier Corrosion

Models for dry oxidation and aqueous corrosion were developed by Stahl (CRWMS M&O 1993j) on the basis of information from the engineering literature. The data were for carbon steel or similar corrosion-allowance materials. Dry oxidation data were for exposure to air; aqueous corrosion data were for exposure to a variety of natural waters. Stahl developed equations that give penetration directly as a function of exposure time and temperature. As written, they are applicable only to exposures under constant conditions. The recommended penetration equations are as follows:

For high-temperature dry oxidation, $P = 178.7 \cdot t^{0.33} e^{-6870/T}$ (Equation 8-1)

For general corrosion for $T < 358$ K (85°C), $P = 2{,}525 \cdot t^{0.47} e^{-2850/T}$ (Equation 8-2)

For pitting corrosion, $P = 10{,}100 \cdot t^{0.47} e^{-2850/T}$ (Equation 8-3)

Here, P is the penetration depth in millimeters, t is the time of exposure in years, and T is the exposure temperature in kelvins. For temperatures between 358 K and the boiling point, Equation 8-2 is expected to overestimate the corrosion rate because the solubility of oxygen in water decreases. The smaller time exponent for dry oxidation indicates that the oxide formed under these conditions is more protective than that formed during aqueous corrosion. Because of the protective nature of the oxide formed during dry oxidation and the small rate constant, dry oxidation is negligible for repository conditions. Although these equations have proven useful, future work must consider more recent models such as those reported in *Total System Performance Assessment 1995* (CRWMS M&O 1995u).

The containers will be subject to aqueous corrosion if they are exposed to below-boiling conditions with water or water films available. Thus, the corrosion will be a strong function of the thermal loading. Temperature profiles for several WP and drift spacings have been generated through the use of a finite-element code that includes details of the WP design (CRWMS M&O 1994i). One particular set of calculations considers a 21 pressurized water reactor (PWR) assembly WP with 42.2 GWd/MTU (gigawatt-days per metric ton of uranium) burnup fuel and 22 years cooling time. Three repository thermal loadings were analyzed: 4.9 W/m^2, 14.1 W/m^2, and 24.7 W/m^2. These represent low, medium, and high thermal loads. The medium thermal load is similar to that in the *Site Characterization Plan* (DOE 1988). The medium thermal load is the worst for WP integrity in that a large fraction of the early postemplacement time falls within the temperature range from 60°C to the boiling point, in which aggressive conditions prevail.

For the high thermal loading, the temperature profile does not enter the aggressive zone until thousands of years have passed. This helps ensure that the containment requirement is met, and contributes to meeting the controlled release requirement. For the low thermal loading, the temperature passes through the aggressive zone early in the postemplacement period. It is possible to mitigate this condition through long-term cooling of the waste before emplacement. However, the low thermal load will create conditions conducive to microbiologically influenced corrosion.

The potential for microbiologically influenced corrosion was evaluated in the *Initial Summary Report for Repository/Waste Package Advanced Conceptual Design* (CRWMS M&O 1994i).

Mesophilic heterotrophs are usually active in the range of 10°C to about 45°C, with the peak at about 20°C. These are likely to be the major contributors to the potential for microbiologically influenced corrosion at Yucca Mountain because they are known to populate tunnels near Yucca Mountain and because temperatures at long times favor them. Thermophilic heterotrophs are active in the range of 45°C to the boiling point of water. These species have not been found near Yucca Mountain, but their potential appearance cannot be ruled out. For microbiologically influenced corrosion to occur, the microorganisms must establish a biofilm on the surface, which requires moisture and an energy source. The organisms can create microenvironments conducive to their survival, many times in conjunction (synergism) with other microorganisms. Many microorganisms are resistant to radiation fields. Most materials are susceptible to microbiologically influenced corrosion. Titanium and titanium alloys appear to be an exception. In all cases, the impact of microbiologically influenced corrosion can be reduced by drying out the rock surrounding the waste package.

The thermal loading of the repository, then, influences the environment that the WPs and engineered barriers will experience and, thus, the thermal zones through which the WPs pass. Current understanding is that high thermal loads will delay the onset of aggressive conditions for corrosion, thus helping to ensure that the containment requirements are met. Radionuclide decay will also reduce the potential releases of the short-lived isotopes. Intermediate to low thermal loads ensure that the WP is in a relatively aggressive environment during the containment period. This requires that the likelihood of microbiologically influenced corrosion be established under these conditions and, if likely, requires that materials and designs be developed that resist or reduce this corrosion.

Radiolysis may cause some increase in the aggressiveness of the near-field environment. Hydroxyl radicals, hydrogen peroxide, nitrogen oxides, nitric acid, and other reactive species can be produced from humid air (Reed and Van Konynenburg 1991). Preliminary studies indicate effect on corrosion will be small. As is shown in Figure 6.3-95, absorbed dose rates at the surface of a uncanistered spent fuel (UCF) WP are not expected to exceed 24 rad/hour (carbon steel), even for 10-year-old fuel. Absorbed dose rates will be lower for canistered fuel (CF) WPs because of the extra layer of shielding provided by the CF container, and dose rates will decrease with time for all types of WPs. To provide detectable effects, much larger dose rates are common in experiments on radiolytic corrosion. Experiments by Reed and Van Konynenburg used dose rates of 11,000 to 21,000 rad/hr. Anantatmula et al. (Anantatmula 1987) mentions several experiments on radiolytic corrosion. At a dose rate of 300,000 rad/hr, an enhancement of corrosion rate of roughly 2.5 times was observed in iron-base alloys. At dose rates of 440 to 5500 rad/hour, radiation did not have a significant effect on corrosion of alloy steel or cast carbon steel. For an air/steam environment at 250°C, corrosion of both iron-base and copper-base materials was increased by "an order of magnitude" at 10,000 rad/hour and 5 times at 100 rad/hour; but, at 150°C, only the copper-base materials showed corrosion enhancement. Anantatmula concludes "Gamma radiation does not have a significant effect on corrosion of copper-base and iron-base materials in the aqueous environment for the *Site Characterization Plan* Conceptual Design." This conclusion is significant because the dose rates for the thin-walled containers that he considered are up to 240 rad/hour. Since this is well above what is expected for the current WPs with their much thicker walls, gamma radiation will also not have a significant effect on WP corrosion.

It should be pointed out that the experiments described above are for gamma radiation only, whereas the radiation at the surface of a WP will be a mixture of photons and neutrons. It is possible that there will be an additional effect on the corrosion rate as a result of atomic displacements within the containment barriers. However, this effect is expected to be small for two reasons. First, the absorbed dose rate due to neutrons is small. Figure 6.3-95 indicates that it will not exceed 0.2 rad/hour (carbon steel). Second, an analysis of neutron-induced embrittlement of the waste containment barriers shows that the effect of neutron irradiation on the ductile-brittle transition temperature is very small (CRWMS M&O 1995y). Since the ductile-brittle transition temperature is strongly dependent on microstructure, it is unlikely that corrosion will be significantly affected.

In addition to the corrosion considerations described above, calculations were performed for degradation of a corrosion-allowance barrier in humid air. For these calculations, temperature and humidity histories were based on Buscheck's models (IHLRWMC, 1994c). The conditions were used as input to an equation for corrosion rate as a function of temperature, relative humidity, and thickness of corrosion product layer. The equation is a generalization of Stahl's equation in which an exponential dependence of corrosion rate on relative humidity is postulated. The equation has no explicit time dependence, so it can be applied to exposures with time-dependent temperature and humidity. The results have been reported in more detail in the *Updated Report on Preliminary RIP/YMIM Analysis of Designs* (CRWMS M&O, 1995m).

Three calculations were performed. Each provides slightly different insights. In the first calculation, temperatures and humidities were taken from a drift-scale thermohydrologic model in which the WPs were represented by infinite cylindrical heat sources in open drifts. The model underestimates temperatures and overestimates humidities to some degree because the WPs are smeared into infinite cylinders. Nevertheless, the model predicts rapid dry-out at the WP surface and suppression of corrosion is for hundreds to thousands of years.

The second calculation used results from a mountain-scale thermohydrologic model, with the temperature and humidity corrected to account for local heating by the WPs. This calculation provides information on how corrosion behavior will vary from one location to another. The results generally agreed with those from the first calculation in that corrosion was suppressed for hundreds to thousands of years. For a low thermal load, a moderate corrosion rate is seen for all packages, regardless of position. At short times, local heating keeps the WPs dry and suppresses corrosion. At longer times, humidity and corrosion rates rise for all packages. The results for a high thermal load are strikingly different. For most of the packages, corrosion performance is excellent because repository-scale drying keeps humidity and corrosion rate low for a long time. At the extreme edge of the repository, however, larger corrosion rates are seen because humid conditions return earlier. Nevertheless, both thermal loads provide acceptable performance.

The third calculation was like the second, but the heat output of the WPs was varied to simulate different burnups, different fuel ages, and packages with high-level waste (HLW) glass rather than SNF. The calculation determined the effects of variations in WP heat outputs. Variations in burnup resulted in modest changes in corrosion depths. Varying the age of the emplaced fuel from 10 to 50 years resulted in negligible changes in corrosion depths. For HLW glass, the corrosion rates are higher than those for SNF because the local heating effect decreases more quickly.

The results of these calculations support existing container designs in that containment for longer than 1000 years is predicted.

8.2.2 Repository Integration Program/Yucca Mountain Integrating Model Analyses

The work reported in this section includes results from two programs for modeling of WP degradation: RIP (Repository Integration Program) 3.13 by Golder Associates, and YMIM (Yucca Mountain Integrating Model) 2.0 by Lawrence Livermore National Laboratory. These programs are used to evaluate the containment and control of release provided by various designs.

The purpose of this effort was not to reproduce the results of *Total System Performance Assessment 1993* (CRWMS M&O 1994f) or *Total System Performance Assessment 1995* (CRWMS M&O 1995u). Instead, it was to develop a model that can be used to contrast the performance of different types of WPs. Results of the effort have been published elsewhere (CRWMS M&O 1995m). The work reported in Sections 8.2.2.1 and 8.2.2.2 is based on *Total System Performance Assessment 1993*, which was current when the work was performed.

8.2.2.1 Repository Integration Program Evaluations

Several RIP evaluations were performed during 1994. These were based on the data for *Total System Performance Assessment 1993*, which was current at the time. *Total System Performance Assessment 1993* considered several models of the proposed repository. The primary differences between these models are the thermal loadings, thicknesses of inner and outer barriers, and the treatment of initiation and rate of corrosion. Only the description of the Engineered Barrier Segment was used in this work. The Repository Integration Program treats containment by the inner and outer barriers of the disposal container. No credit is taken for any of the fuel cladding or HLW pour canisters.

Twelve Repository Integration Program calculations were performed to determine the effects of different corrosion rates. Following *Total System Performance Assessment 1993*, the outer barrier was taken to have a thickness of 100 mm and the Stahl model of corrosion, as described in Section 8.2.1, was used, with initiation when the rock saturation reaches 0.08. Calculations were performed at three repository thermal loads: 28.2 W/m^2, 14.1 W/m^2, and 7.04 W/m^2 (114 kW/acre, 57 kW/acre, and 28.5 kW/acre). For each thermal load, calculations were run with (1) corrosion rates as specified in *Total System Performance Assessment 1993*, and (2) half that corrosion rate. For each of these, calculations were run with (1) an inner barrier that fails immediately after the outer barrier fails, and (2) an inner barrier that fails 1,000 years after the outer barrier. The calculations simulated 10,000 years of disposal.

Compliance with the substantially complete containment requirement of 10 CFR 60.113 is sensitive to the assumptions about corrosion. All calculations predicted no container failures during the first 300 years. Only two calculations showed failures at 1,000 years, those at thermal loadings of 28.2 W/m^2 and 14.1 W/m^2 (with the larger corrosion rate and an inner barrier that fails quickly). The fractions of containers failed were 23 percent and 48 percent, respectively. The large number of failures in these calculations shows that if the current models for corrosion and failure are accurate, achievement of substantially complete containment is strongly dependent on corrosion rate.

Releases from all WPs, normalized to the limits of 40 CFR 191, over 10,000 years are shown in Table 8-1.

It is seen that the addition of a 1,000-year inner barrier provides only a modest decrease in the mean normalized release, but decreasing the corrosion rate of the outer barrier provides a more substantial increase. The intermediate thermal load allows the largest releases, and the high and low thermal loads give very similar releases. Several of the calculations gave normalized releases that exceed the (remanded) limits of 40 CFR 191. However, that result simply reflects the conservatism of the model. The more recent analyses of *Total System Performance Assessment 1995* (CRWMS M&O 1995u) are more realistic and give lower releases.

The values given above are subject to substantial uncertainty. For example, at a thermal load of 28.2 W/m^2, the packages do not corrode until rewetting occurs. Upon rewetting, however, the packages experience high corrosion rates for a long time. Package failures are thus closely spaced in time and a rapid increase in release is calculated. Such abrupt releases may well be an artifact of the model. Because of heterogeneities in the rock, rewetting of the packages will be spread over time, so releases should be more gradual.

Table 8-1. Normalized Releases from WPs Over 10,000 Years

Thermal Load, W/m^2	Corrosion Rate	Inner Barrier Life, yr	WPs Failed at 1,000 yr	Mean Normalized Release
28.2	high	0	23%	2.74
28.2	high	1,000	0	2.14
28.2	low	0	0	0.44
28.2	low	1,000	0	0.31
14.1	high	0	48%	8.44
14.1	high	1,000	0	7.24
14.1	low	0	0	3.90
14.1	low	1,000	0	3.10
7.04	high	0	0	2.70
7.04	high	1,000	0	2.07
7.04	low	0	0	0
7.04	low	1,000	0	0

8.2.2.2 Yucca Mountain Integrating Model Evaluations

Containment lifetime is of great interest in WP development. Several Yucca Mountain Integrating Model calculations of containment life were performed. The calculations focused on containment lifetime and used data files based on sample files supplied by Lawrence Livermore National Laboratory.

The calculations use Lamont's model for pitting failure of the inner barrier (CRWMS M&O 1994f). Alloy selection is important in WP design. To simulate the effect of changing the pitting resistance of the inner barrier, different parameters were used in the pitting model. For realism, the thickness of the inner barrier was taken to be 20 mm in these calculations. The results are summarized in Table 8-2. The descriptive names "high," "medium," and "low" are as used by Lamont to describe susceptibility to pitting.

Table 8-2. Results of Pitting Model

Pit Growth Increment, µm, at		Median Failure Time, yr	Descriptive Name
70°C	100°C		
300	30,000	12,500	high
100	10,000	13,300	medium-high
30	3,000	15,500	medium
10	1,000	19,900	medium-low
3	300	39,000	low
1	100	$> 10^6$	very low

The Yucca Mountain Integrating Model predicts that the corrosion allowance barrier will be intact for about 12,300 years. According to the *Total System Performance Assessment 1993* data set for a mass loading of 28.2 W/m^2 (114 kW/acre), the temperature of the waste package is about 87°C at that time. Thus, a corrosion-resistant barrier with high susceptibility to pitting gives little benefit (only about 200 years) in terms of extending containment life. In contrast, a corrosion-resistant barrier with low or very low susceptibility to pitting gives very great extensions of containment lifetime. At least part of the increase is due to the Arrhenius dependence of pit growth increment on temperature. At long times, the decreasing temperature results in very small pit penetration rates.

The results of these analyses have large variation in containment lifetime. The variation results from uncertainty in environment and materials properties that had not been resolved at the time of these calculations.

8.2.3 Fill Gas

Four aspects of helium release from the proposed high-level nuclear waste repository at Yucca Mountain have been considered: (1) What are the chemical and radiological consequences of releasing helium to the environment? (2) How fast will helium escape from a WP and how will helium release affect thermal performance? (3) Will helium release from fuel rods affect thermal performance? (4) How much helium will be produced by α decay and what are the effects of helium production? These questions are discussed in more detail elsewhere (CRWMS M&O 1993c). Conclusions are as follows:

Since helium is chemically inert and not radioactive, its release will not have any chemical or radiological consequences nor is it expected to enhance or inhibit the transport of ^{14}C or other radionuclides.

Helium leak specifications for the waste package have not been set, but typical specifications (CRWMS M&O 1994j) are so stringent that the rate at which helium will escape from an intact WP will be very low. Only a small fraction of the helium will escape during the period of substantially complete containment (1,000 years); nearly all will be retained until the time of WP breach. If air replaces the helium fill gas, the peak temperature of the waste will increase, and for WPs that fail very early, additional degradation of the cladding or fuel may result. But if a package lasts through the period of substantially complete containment, the heat generation rate will be so low at the time of breach that little temperature rise will result.

Since the thermal conductivity of gases is essentially independent of pressure, helium release from fuel rods will not affect thermal performance.

The rate of helium production by α decay will be low. The production over 10^6 years will be on the same order of magnitude as the initial helium inventory of a WP.

It is concluded that these studies provide no evidence that helium will impair waste package performance.

8.3 FUEL CLADDING

Fuel cladding has substantial promise as a barrier because most cladding is made of zircaloy, which is resistant to aqueous corrosion. Only about 1 percent of the fuel rods dating from the 1970s or earlier had leaks upon discharge from the reactor, and more recent fuel rods provide better containment, with only about 0.02 to 0.05 percent of rods having leaks upon discharge (BNL 1990).

Two types of cladding failure may be distinguished: perforation and gross failure. Perforation is the less severe of the two and tends to occur if the degradation mechanism is driven by gas pressure inside the fuel rod. When perforation occurs, the gas inside the rod escapes and the degradation mechanism becomes inactive. Perforations are small, with hole dimensions on the order of micrometers. As a result, perforations may become plugged with products of corrosion, which could limit the access of outside gas or water to the fuel. Gross failure implies much larger holes or cracks in the cladding and may be caused, for example, by general corrosion. In an oxidizing environment, perforation may result in gross failure later. Oxygen can enter the fuel rod through the perforation, leading to oxidation of UO_2 to U_4O_9 or U_3O_8, depending on temperature and time. Oxidation to U_3O_8 causes the fuel to increase significantly in volume. Such volume increases can split the cladding (EPRI 1986).

It has been suggested that limiting the temperature of the cladding to 350°C will suffice to control degradation. In this report we assess degradation mechanisms and attempt to predict the durability of the cladding as a function of temperature. The model of diffusion-controlled cavity growth described below does not specify an absolute maximum temperature to which the cladding may be safely exposed. Instead, it specifies that damage accumulates at a temperature-dependent rate and that failure occurs when enough damage has accumulated.

The rule of thumb that maximum cladding temperatures should be kept below 350°C was examined and found to be roughly correct. But for very slow cooling, the 350°C rule may not be conservative. Such conditions might arise if emplacement drifts are backfilled some time after waste emplacement. Therefore, the use of damage accumulation integrals is recommended in evaluating thermal loadings.

Only a few experimental efforts have considered the failure of fuel rods at high temperature because of internal pressure. Einziger and Kohli (1984) stored irradiated fuel rods at 323°C for 2,101 hours and studied the effects of temperature and pressure on cladding creep. The duration of the experiment was too short to allow confident extrapolation to the duration of waste disposal. However, the experiment did reveal the reorientation of hydrides. The hydrides were originally circumferential, but after dissolution at high temperature under tensile stress, the hydride reprecipitated on grain boundaries. The new precipitates were largely radial. Hydride reorientation is discussed in more detail below.

In *Results of Simulated Abnormal Heating Elements for Full-Length Nuclear Fuel Rods* (PNL 1983b), Pacific Northwest Laboratory describes an experiment in which unirradiated fuel rods were heated until they failed. The author was interested in studying the overheating of rods in dry storage. The times are much shorter and the temperatures much higher than those for disposal. However, the results do provide information on cladding failure modes.

Several mechanisms have been proposed for the failure of fuel rod cladding. These include creep rupture, stress corrosion cracking, delayed hydrogen cracking, radial reorientation of hydrides, hydrogen redistribution, irradiation embrittlement, oxidation of the cladding, and strain rate embrittlement. *Control of Degradation of Spent LWR Fuel During Dry Storage in an Inert Atmosphere* (PNL 1987a) includes a detailed assessment of the importance of these modes for cladding degradation during dry storage. The authors concluded that creep rupture is the primary mechanism of failure. The Nuclear Regulatory Commission reached similar conclusions in their evaluation of storage casks for dry storage of spent fuel (NRC 1985b). In disposal, the conditions are similar to those for storage. Accordingly, the dominant mode of degradation in disposal should be creep rupture.

8.3.1 Creep Rupture

Einziger and Kohli (1984) used a conservative Larson-Miller analysis of creep life to predict a creep rupture life of 100 years at 305°C for irradiated cladding. Because it is impractical to perform experiments of such long duration, predictions like these require extrapolations from short-term behavior measured at high temperatures. In making such extrapolations, there is always the danger that the dominant degradation mechanism will change, and that the degradation rate will be underestimated. If the degradation mechanisms can be readily described, it is preferable to use mechanistic models instead of empirical models, and to have the choice of models dependent on temperature and stress.

Four mechanisms have been suggested for creep failure (PNL 1986a): ductile transgranular fracture, triple-point cracking, power-law cavity growth, and diffusion-controlled cavity growth. Under the conditions expected for storage or disposal, diffusion-controlled cavity growth is expected to be the primary failure mechanism (BNL 1990; PNL 1987a). The Nuclear Regulatory Commission agrees with this conclusion, as is shown by their calculations for storage casks in which they calculated temperature limits from a model for diffusion-controlled cavity growth (NRC 1985b).

For any mechanism of creep rupture, failure is slow and the driving force for degradation is completely removed when failure occurs. Accordingly, creep rupture leads to perforation rather than to gross rupture.

Standard treatments of cladding life use the method of damage accumulation: the amount of damage at time t, $D(t)$, is given by

$$D(t) = \int_0^t \frac{d\tau}{L(\tau)} \qquad \text{(Equation 8-4)}$$

where L is the lifetime of the material under the conditions that prevail at time τ. When $D(t)=1$, the material fails.

A crucial factor in prediction of cladding life is the hoop stress. The maximum hoop stress σ occurs at the inner surface of the cladding. Note that σ will depend on the temperature of the rod, initial pressurization, burnup, and degree of fission gas release.

Mechanistic models of diffusion-controlled cavity growth yield the following equation for creep rupture life L under constant conditions:

$$L = \frac{n\lambda^3 kT}{\delta D_{gb}\Omega\sigma m} \qquad \text{(Equation 8-5)}$$

where n is a dimensionless constant that depends on the shape of the cavity and the diffusion geometry, λ is the cavity spacing (m), k is Boltzmann's constant (J/K), T is temperature (K), δ is the effective thickness of the grain boundary for diffusion (m), D_{gb} is the grain boundary diffusivity (m^2/s), Ω is the atomic volume (m^3), σ is the hoop stress (Pa), and m is a dimensionless constant that depends on microstructure. Values of the factors in Equation 8-5 have been discussed in "Prediction of Cladding Life in Waste Package Environments" (IHLRWMC 1994b).

In many treatments of diffusion-controlled cavity growth, it is assumed that the cladding is subject to uniaxial tensile stress and all the grain boundaries are normal to the axis of the stress. Such a model is applicable to cladding only if all the grain boundaries lie in radial planes, whereas cold-forming of cladding produces grains that are flattened in the radial direction. Typical grain boundaries are thus favorably oriented for resisting diffusion-controlled cavity growth. The factor m takes the effects of microstructure into account. Growth of cavities is possible only if the traction across the grain boundary has a normal tensile component. The value of m reflects the average normal traction on the grain boundaries. Values have been calculated for various shapes. For equiaxed grains, $m = 0.500$. For flattened grains with geometries that are typical for cold-worked cladding, $m = 0.165$.

With the information given above we can calculate the lifetime L for various temperatures. Sample results are given in Table 8-3.

Table 8-3. Cladding Lifetime for Various Temperatures

Temperature, °C	Lifetime, yr
200	120,000
250	5,000
300	360
325	110
350	40
375	15
400	6.1

These results are for isothermal exposure at the specified temperature. For varying temperatures, the method of damage accumulation (Equation 8-4) applies. Since we attempted to make conservative estimates of each parameter, these predictions should be conservative.

These methods were used to calculate the degradation of cladding from a simulated spent fuel thermal history (IHLRWMC 1994b). Pressurized water reactor fuel assemblies with 40 GWd/MTU(gigawatt days per metric ton of uranium) burnup and 461 kg of heavy metal per assembly were assumed to be stored in a fuel pool for five years, then stored in a CASTOR V/21 dry storage cask for five years, and finally emplaced in a repository. The repository contained 21-pressurized-water-reactor-assembly packages in 7.62-m (25-ft) drifts with a thermal loading of 19.8 W/m^3 (80 kW/acre). The temperature (CRWMS M&O 1993h) and fraction of cladding life consumed are plotted in Figure 8-1. Here the amount of cladding life consumed is the amount of damage D in Equation 8-4; when $D = 1$, the material fails. A small amount of damage (slightly over 5 percent) accumulates during the five years of dry storage. Because of the rapid decrease in temperature during storage, prolonging dry storage by an additional 40 years produces only a very small amount of additional damage. Because of the low temperature and low rate of damage accumulation, calculations for dry storage were stopped at that time. However, if the waste package is emplaced in a repository, the temperature and rate of damage accumulation increase. Almost all of the damage occurs during the first 100 years; as the temperature drops, damage accumulates very slowly. The total damage accumulated at 10,000 years is about 0.46. This value is sufficiently small to raise hopes that cladding may be effective as a barrier to release. If the model for cladding degradation is conservative or temperatures are lower than those used here, cladding should be an effective barrier. However, if temperatures are significantly higher than those used here, numerous cladding breaches may occur.

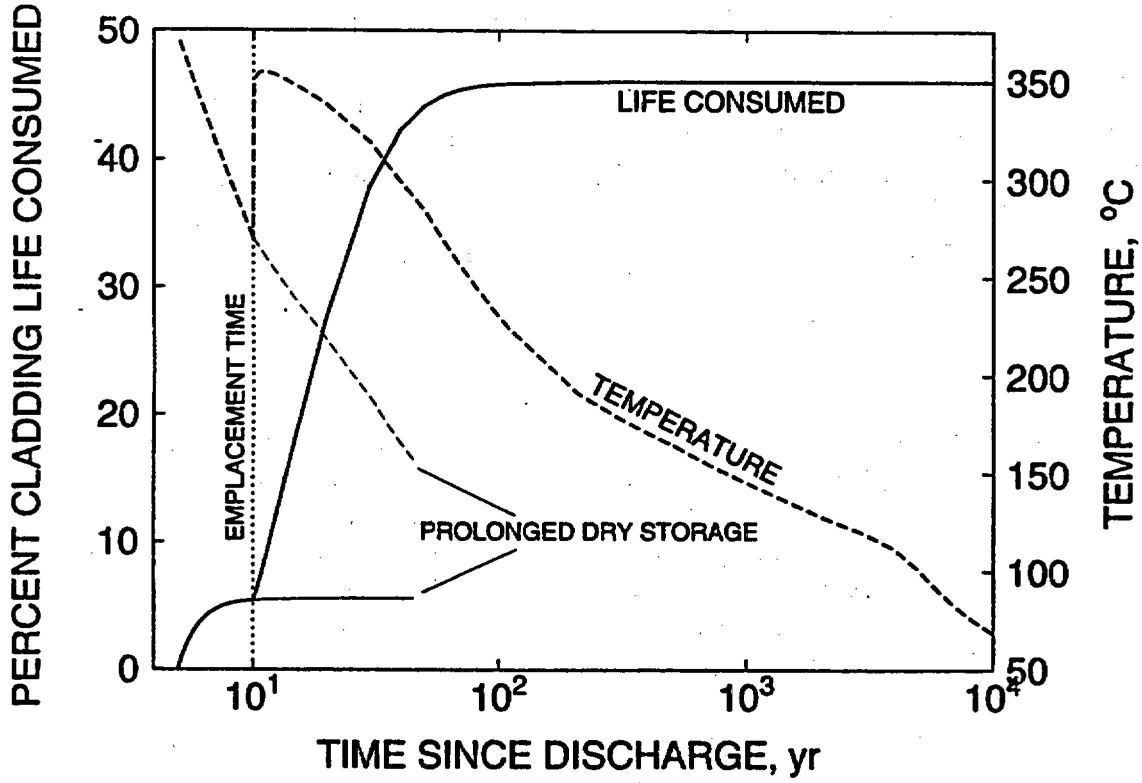

Figure 8-1. Temperature and Fraction of Cladding Life Consumed as Functions of Time for One Repository Design

8.3.2 Hydride Reorientation

Cladding corrosion during reactor operation releases hydrogen, which can be absorbed by the cladding. Hydrogen has a large solubility in zircaloy; at 350°C, 120 ppm of hydrogen can dissolve (BNL 1990). When the cladding cools, the solubility decreases and dissolved hydrogen precipitates on grain boundaries as hydrides. The hydrides are brittle and can provide paths for brittle fracture of the cladding. Zircaloy grains are typically flattened in the radial direction, so in unstressed material the hydrides are mostly circumferential. But if the hydrides precipitate while the material is under stress, the hydrides preferentially precipitate along planes normal to the largest principal tensile stress. For pressurized fuel rods, the preferred direction is radial, and radial hydrides tend to promote cracking of the cladding.

It is difficult to assess the effects of hydride reorientation on the life of cladding, as there is very little information available except that from Einzinger and Kohli (1984). The conditions for their experiment were similar to those expected for disposal of spent fuel: the temperature was 323°C and the hoop stress in the cladding was 145 MPa. This stress is somewhat higher than that expected for typical fuel rods, but the hydrogen content of the cladding, about 90 ppm, was not unusual. At the end of the experiment, in which the rods were held at temperature for 2,101 hours (about 88 days), substantial reorientation of hydrides was observed. The hydrogen would have dissolved while the rods were at temperature, and the reoriented hydrides presumably precipitated during cooldown. Therefore, even a much shorter time might suffice to reorient the hydrides.

In some regions, the radial hydrides extended almost completely through the wall thickness. Although the hydrides did not cause failure of the cladding, it seems that the ductility and durability of the cladding must be greatly reduced. The expected behavior in a repository is that hydrides will dissolve when the fuel heats up and reprecipitate as it cools. Additional study of hydride reorientation appears to be needed if credit is to be taken for the cladding.

8.3.3 Mitigating Processes

Not all processes degrade the material of the cladding. Apart from the degradation mechanisms discussed above, there are also mechanisms that improve the condition of the material or reduce driving forces for degradation. Notable among these is creep, which can reduce the driving force for diffusion-controlled cavity growth. Even small amounts of creep can produce substantial changes in pressure. Einziger and Kohli (1984) state that a hoop creep strain of 2×10^{-4} will typically reduce the internal pressure about 1 percent. For unirradiated zircaloy, the increases in radius can be large. In tests of fuel rods that were furnace-heated at 10 to 56 K/hour, the cladding radii increased by 10 to 36 percent before rupture occurred (PNL 1983b).

A second process that tends to improve the condition of the material is annealing. Irradiation is expected to produce both dislocations and point defects, with a resulting increase in hardness and decrease in ductility. At sufficiently high temperatures, the defects can be annealed out, with a resulting decrease in hardness and, presumably, an increase in ductility. The annealing rate depends strongly on temperature. At 500°C, annealing is essentially complete in 10 days, but at 400°C it takes about 50 years (PNL 1987a). Annealing at 350°C takes much longer, perhaps 5,000 years. The annealing times are so long that using decay heat to anneal the cladding in the repository is not practical.

It is not clear how much annealing would occur inside a waste container or how much benefit might be obtained from annealing. Einziger and Kohli (1984) estimate that a storage life of 100 years may be possible at 400 to 440°C for annealed rods.

8.3.4 Oxidation

Fuel cladding is potentially significant as a barrier that will help to provide compliance with the requirements for controlled release of radionuclides. Even perforated cladding will provide significant confinement because the perforations are typically quite small. In contrast, grossly failed cladding would allow relatively easy release of radionuclides from the fuel. Under certain conditions, oxidation of the fuel cladding and pellets could cause such gross failure. A study was recently performed to evaluate the significance of oxidation for degradation of SNF (CRWMS M&O 1995m).

The vast majority (more than 99 percent) of the fuel rods placed in the repository will be intact, with complete integrity of the cladding. A small fraction of the fuel rods will be perforated, i.e., they will have a tiny hole or crack in the cladding. Intact fuel cladding can perforate by creep rupture if it is stored at a high temperature, but it is generally agreed that a creep rupture failure will produce only a small perforation. Once the cladding is perforated, the gas pressure equalizes between the inside and outside of the cladding, and creep degradation ceases. If the rod is stored in an inert atmosphere, as in an intact disposal container, no additional degradation occurs. However, if the atmosphere is oxidizing, two types of oxidation become possible. The first and simpler type is oxidation of the cladding, which could gradually thin the cladding until it is gone. The second type is oxidation of the fuel pellets. There are two steps in this oxidation process. First, the fuel oxidizes locally in the vicinity of the perforation. The resulting volume increase of the fuel eventually leads to formation of a macroscopic split in the cladding. Second, after the split forms, continued oxidation of the fuel pellets near the ends of the split causes the cladding to gradually "unzip," i.e., split axially. These two steps are called "split initiation" and "split propagation" here. The first step lasts until U_3O_8 begins to form; the second step corresponds to oxidation of the remainder of the fuel to U_3O_8.

Einziger (IHLRWMC 1994a) has presented equations for the rate of oxidation of fuel cladding, the time necessary for oxidation of fuel pellets to cause a macroscopic split to develop in perforated cladding, and the speed of propagation for such a split, all as functions of temperature. These equations apply for exposure to air. The equations were combined with information on fuel rod geometry to determine characteristic times for the various degradation processes for a typical pressurized water reactor fuel assembly. The results are reported in more detail in the *Updated Report on Preliminary RIP/YMIM Analysis of Designs* (CRWMS M&O 1995m).

For mass loadings being considered by the project and emplacement without backfill, fuel temperatures will be low enough that oxidation of the cladding will cause negligible damage even if the disposal container provides no protection. For emplacement with immediate placement of backfill, destruction of the cladding by oxidation is possible only if the disposal containers fail very early. Such degradation can be controlled by delaying emplacement of backfill.

For perforated fuel rods, temperatures are sufficiently high at short times that some protection by the disposal container is necessary. For a low mass loading (5.93 kg U/m^2) and no backfill, the fuel must be protected from oxidation for about 35 years to prevent the cladding from splitting at some later time; for a high mass loading (20.5 kg U/m^2), the fuel must be protected from oxidation for about 130 years. These times are extended to 150 years and 510 years, respectively, if a crushed tuff backfill is used.

Since the requirements for protection by the disposal containers are small, intact fuel rods are expected to remain intact unless perforation occurs by some other mechanism, such as creep rupture. Fuel rods with perforations are not expected to split unless the disposal container fails very early.

Cladding oxidation and splitting of the cladding as a result of fuel pellet oxidation are significant only at high temperatures. However, while the fuel rods are hot enough for significant degradation to occur, the surface of the disposal container will be above the boiling point. As a result, failure of the disposal container by corrosion is unlikely during this period.

8.4 CONTROL RODS

Under current regulations, a HLW repository must comply with the Resource Conservation and Recovery Act of 1976. The Resource Conservation and Recovery Act and its implementing regulations impose strict controls on the release of cadmium from solid waste. Spent fuel contains significant concentrations of cadmium as a fission product. Some spent fuel assemblies also contain control rod assemblies in which the neutron absorber is an alloy of 80 percent silver, 15 percent indium, and 5 percent cadmium (by weight). If these used control rods are left in the fuel assemblies at disposal, or if additional assemblies of the same type are added to provide additional criticality control, they will provide an additional cadmium inventory. The inventories of cadmium in control rods and as fission products are comparable. Preliminary corrosion calculations suggest that neither control rods nor spent fuel pellets is a hazardous material under the Resource Conservation and Recovery Act (CRWMS M&O 1994l).

The Toxicity Characteristic Leaching Procedure, defined in 40 CFR 261, is used to determine whether a material is hazardous because of its toxicity. In this test, the material is exposed to an aqueous solution of acetic acid or acetic acid and sodium hydroxide. At the end of the test, the fluid is analyzed. The material is hazardous if the cadmium content of the fluid is greater than 1 mg/L. To achieve that concentration of cadmium requires a corrosion rate of 0.4 μm/day. That corrosion rate is much larger than what is expected. Shreir (1976) states that silver is thermodynamically stable in acetic acid and is widely used to handle sodium hydroxide. Since silver is the primary component of the alloy, and the concentration of cadmium is so small, dealloying and leaching of cadmium appears to be unlikely. It is concluded that silver-indium-cadmium control rod alloys will *not* be classified as hazardous waste.

In an Environmental Protection Agency contractor report (Cohen 1990), bounding estimates were made for releases from pressurized-water reactor assemblies without control rods and boiling water reactor fuel assemblies. The concentrations of toxic elements were well below the Resource Conservation and Recovery Act limit. Thus, spent fuel assemblies without control rods are not hazardous materials under the Resource Conservation and Recovery Act.

9. DEVELOPMENT TASKS AND ISSUES

The purpose of the Waste Package Engineering Development program is to develop nuclear waste disposal container designs that the Nuclear Regulatory Commission will find acceptable and will license for disposal of spent nuclear fuel and defense high-level waste (DHLW) glass within a tuff repository. These activities include development of waste package fabrication techniques, closure weld techniques, nondestructive examination development techniques, and filler material testing. The *Waste Package Engineering Development Task Plan* (CRWMS M&O 1993b) satisfies the requirement for a description of planned waste package (WP) development activities, as specified in the *Waste Package Implementation Plan* (YMP 1995). The *Development Task Plan* is intended to cover the entire engineering development period, which will continue through conclusion of the License Application Design phase; however, the current edition of the plan will require periodic revision and updating in response to evolution of the waste package final design.

As a consequence of historical usage, the term waste package is frequently used to refer to the Waste Package Subsystem Configuration Item. An empty waste package is a disposal container (Disposal Container Subsystem Element). Waste container is a generic term for a Waste Package Subsystem without the Shielding Subsystem Element and the Packing and Absorbent Materials Subsystem Element (i.e., a waste container is a sealed disposal container with the uncanistered or canistered waste form placed therein [and possibly filler material]) (see Figure 3-5).

The definition of disposal container includes not only the multiple containment barriers which make up the waste container shell, but also includes any internals. To differentiate between the disposal container designs for canistered fuel (CF) and high-level waste (HLW) canistered waste forms (both being simple right cylinders), and the uncanistered spent nuclear fuel disposal container designs (right cylinder with basket therein), the latter are entitled "disposal container with basket." The basket includes the basket grid, structural supports, and any thermal shunts.

Shielding may or may not be finally included as a design feature of the WP; current designs do not include shielding beyond that shielding inherently resulting from the heavy walls of the disposal containers and the CF shell. Packing and other absorbent materials may be provided to immediately surround the waste containers as emplaced in the repository (current designs do not include packing and other absorbent materials, nor do they include backfill).

Within this report, the term "waste package or WP" is often used to refer to what may more properly be termed the waste container or the waste disposal container; i.e., usage of the term "waste package or WP" herein often excludes any packing and other absorbent materials and any shielding.

For the purposes of this Advanced Conceptual design (ACD), it is assumed that the Civilian Radioactive Waste Management System program is configured such that the majority of spent nuclear fuel will arrive at the repository as CF, which will be used only for intact, non-failed fuel. All CF canisters will arrive at the repository by rail. However, not all nuclear facilities will be able to accommodate dual purpose canisters or a CF system; thus shipment of spent nuclear fuel from these few remaining sites will result in some uncanistered fuel (UCF) being received at the repository (all UCF will arrive at the repository by truck). This spent nuclear fuel will be placed in UCF disposal containers with baskets. The HLW glass in sealed metal pour canisters will arrive at the repository by rail.

Because of the variety of wastes to be emplaced, there will be several types of WPs/disposal containers. The waste will arrive at the repository in several different forms: spent nuclear fuel contained within large and small CF canisters, different types of uncanistered spent nuclear fuel, and canistered HLW glass. Although some of the HLW glass will not be defense-related, the disposal container design for HLW glass pour canisters is nonetheless referred to as the DHLW disposal container. Each waste type will be loaded into a disposal container designed for that particular type of waste. The multibarrier container portion of the UCF, CF , and DHLW disposal containers are of similar design and serve the same function—that of isolating the waste for the prescribed period of time.

For purposes of this report, the CF configuration and physical characteristics were taken to be those of the original multi-purpose canister conceptual designs (CRWMS M&O 1994k). The CF conceptual designs result in two CF canister sizes, large and small, differing in capacity and mass (essentially differing only in diameter), which necessitated development of two CF disposal container conceptual designs. Several UCF disposal container designs have been developed with capacities generally corresponding to the CF conceptual designs; however, based on projections for UCF delivery, only the larger UCF disposal container designs would be used in the repository. The large UCF disposal container and the large CF disposal containers are of somewhat similar diameter and length, whereas the DHLW disposal container is of similar diameter but shorter in length.

Spent nuclear fuel waste package design activity is limited to intact, nondefective fuel assemblies from commercial light water power reactors, whether arriving at the repository as UCF or CF. Design variations have been deferred until later which would accommodate canistered failed fuel assemblies, oversized fuel assemblies, DOE-owned spent nuclear fuel including Navy fuel, etc.

As part of the container development process, evaluation of each disposal container concept will be based on both technical feasibility and cost-effectiveness of the manufacturing processes (i.e., disposal container fabrication, closure, and inspection).

The specific engineering development tasks involve test and evaluation of full or reduced-scale sections of various WP design concepts. The tasks will focus on key manufacturing uncertainties specific to each design concept. As the manufacturing processes and nondestructive examination techniques are developed, and as the results of the prototype testing become available, proposed process specifications will be developed and preliminary fabrication drawings will be prepared.

Manufacturing studies will include prototype fabrication and an engineering test series that will be completed on the prototype units and results incorporated into the designs. The test series will determine whether the WP design requirements have been met.

9.1 WASTE PACKAGE MANUFACTURING PROCESS DEVELOPMENT

The purpose of the Waste Package Engineering Development tasks is to perform the requisite engineering development and manufacturing process development for fabrication, closure, and inspection of Mined Geologic Disposal System (MGDS) waste disposal packages. These waste disposal packages will be the variety of WP/disposal container design configurations put forth for the disposal of both HLW and spent nuclear fuel. The engineering development tasks will complement not only the WP design evaluations, but also the Engineered Barrier Segment and MGDS system design and Performance Assessment evaluations. These tasks focus on the WP design configurations and sizes that have evolved from the various discipline conceptual design studies such as thermal, neutronics, handling, and emplacement.

The sizes, wall thicknesses, and materials of construction of the containers require consideration of a range of manufacturing processes, as a result of limitations of those various processes. Similarly, a range of container closure design configurations and welding techniques are also being considered, driven by concerns of weld-induced stress minimization, the possible requirement for postweld stress reduction treatment, and nondestructive examination inspection capability and limitations. Technical feasibility and cost-effectiveness of the various fabrication and closure concepts have been the focus of engineering development tasks during the ACD phase of the program, and will continue throughout the next design phase.

Two development programs are currently being conducted and both will be completed in FY 1996. The waste package closure weld development program, described in the *Waste Package Closure Development Technical Guidelines Document* (CRWMS M&O, 1995aa), is nearing completion. The results are expected to present a welding technique suitable for performing the remote closure welds. The waste package filler placement test program, described in the *Spent Nuclear Fuel Waste Package Filler Testing Technical Guidelines Document* (CRWMS M&O, 1995ac), is in progress; two different dummy PWR fuel assemblies have been fabricated, along with a test fixture. Testing will utilize different sizes of filler (steel shot); test results are expected to demonstrate the size needed to accomplish a high percentage fill of the free space in and around a fuel assembly.

The engineering development process is based on a proven Industrial Engineering process (Roberts 1983). At present there are five identified Waste Package Engineering Development tasks:

1. Container fabrication, including stress minimization.

2. Remote closure welds, including weld-induced stress minimization.

3. Remote nondestructive examination of the closure welds.

4. Remote in-service-inspection.

5. WP internal filler material infiltration/uniform distribution.

The engineering development activities include

1. Preparation of a Technical Guidelines Document for each individual task, including test plan and interface requirements.

2. Review of prospective development organization facilities and/or engineering test laboratories, leading to selection of the development organization.

3. Concurrence review of the Technical Guidelines Document with the development organization, including test plan and interface requirements.

4. Establishment of needed Quality Assurance approvals for the development organization.

5. Technical management of the process development and testing performed by the development organization.

6. Creation of draft process specifications, based on results of the development tasks.

7. Preparation of the development task final development test report. Include the draft process specifications in the final development test report.

The following sections discuss the first three tasks and current status. Discussion of remote closure and remote nondestructive examinations are combined. The remote in-service inspection task has not yet begun. The use of filler material in spent nuclear fuel WPs is discussed in Section 9.5. A filler placement development activity is being initiated, based on steel shot as the filler material.

9.2 TECHNICAL GUIDELINES DOCUMENTS AND TEST PLANS

The following Technical Guidelines Documents have been created for the WP development tasks of remote closure and nondestructive examination, and for placement of filler material:

- *Waste Package Closure Development Technical Guidelines Document* (CRWMS M&O 1995aa)

- *Nondestructive Examination Development Technical Guidelines Document* (CRWMS M&O 1995ab)

- *Spent Nuclear Fuel Waste Package Filler Testing Technical Guidelines Document* (CRWMS M&O 1995ac)

The Technical Guidelines Documents describe the task objective, scope, requirements, background, and required developmental testing activities. These development task activities will also include review of prospective development organization facilities and/or engineering development test laboratories; concurrence review of the guidelines with the selected development organization, including test plan and interface requirements; establishment or approval of quality assurance requirements, and subsequent technical management of the process development and testing. The development organization is required to provide periodic status reports and a final development test report.

The final goal of each Waste Package Engineering Development task must be demonstration of a successful process, with sufficient confidence that the process will be suitable for the intended end application (fabrication, closure, etc.). Draft process specifications will be prepared based on results of the development task; these draft specifications should be included in the final development test report.

9.3 WASTE PACKAGE FABRICATION, ASSEMBLY, AND INSPECTION

The conceptual designs for CF disposal containers, DHLW disposal container, and UCF disposal containers are illustrated in the figures presented in Appendix B.

As stated earlier, the disposal containers will not be designed and built to American Society of Mechanical Engineers pressure vessel codes, as the disposal containers do not function as pressure vessels. Disposal container materials will conform to American Society for Testing and Materials standards, and may be specified to be American Society of Mechanical Engineers Code Case materials, but there will be no blanket requirement to use Code Case materials in the disposal containers.

The presently selected WP fabrication technique for the empty container assembly is to first roll and weld the plate to form rings for the outer barrier cylindrical shell. As an alternative, the feasibility and cost of forged rings will be investigated. It is anticipated that the size of the rings that could be fabricated will be large enough that only two rings will be necessary to form the outer barrier cylinder, joined with a circumferential weld. Possibly the thinner, shorter DHLW disposal container outer barrier could be formed as a single ring.

The presently preferred method of forming the inner barrier is by weld deposition cladding inside the cylinder, building up the necessary thickness through multiple weld passes. The bottom lid(s) could either be clad, or two separate lids individually welded in place during fabrication. The top lids would be separate non-clad pieces. Both the bottom lids and top lids have been sized slightly thicker than the wall to ensure that the weld effective throat exceeds wall thickness. Details of the empty container assembly fabrication steps are as follows:

1. Outer barrier cylinder fabrication and stress relief heat treatment.

2. Skim cut (machine cut) of inner diameter preparatory to nondestructive examination; also serves to true up the cylinder inner diameter.

3. Nondestructive examination of cylinder weld seams (radiographic, ultrasonic, etc.), and magnetic particle testing of cylinder inside surface.

4. Inner barrier weld cladding operation.

5. Post cladding stress relief heat treatment.

6. Skim cut of cylinder clad inner diameter preparatory to nondestructive examination of cladding; also perform weld preparation machining for the inner and outer lid weld joints (both bottom and top).

7. Nondestructive examination of cladding (dye penetrant, ultrasonic).

8. Weld bottom lids in place, and examination bottom lid welds.

9. Stress relief heat treatment of container assembly.

As part of fabrication, the outer barrier inside surface would be machined with a skim cut before cladding is performed; the skim cut serves to produce a suitable surface for nondestructive examination and a necessary true inner cylinder profile upon which to deposit the inner barrier weld cladding. The resulting inner surface of the built up inner barrier would also be machined with a skim cut to produce a true cylindrical surface. A bonus of this fabrication technique is that a small predictable cylindricity will result, beneficial to all of the disposal container and WP designs. Consequential distortions during final heat treatment can normally be avoided by performing the heat treatment with the container in the upright position.

9.3.1 Disposal Container Cylindricity

Cylindricity is defined as the condition of a surface of revolution in which all points of the surface are equidistant from a common axis. The definition of cylindricity tolerance specifies a tolerance zone bounded by two concentric cylinders within which the surface must lie. The term "cylindricity" as used herein would more properly be termed "cylindricity tolerance."

Selection of appropriate values for disposal container cylindricity would depend on the selected method of fabrication for the inner barrier (container). For example, the DHLW disposal container design concept evolved before the clad inner barrier concept was adopted, thus a larger value for cylindricity was assumed. The CF disposal container conceptual design uses the cladding fabrication method, such that a smaller value is assumed even though a CF canister is over 50 percent longer than a HLW canister. The subject of cylindricity will be investigated further to establish disposal container cylindricity specifications that are attainable and yet are economically reasonable.

The diametral clearance between the CF canister and disposal container is presently assumed to be 30 mm. The *MPC Subsystem Design Procurement Specification* (CRWMS M&O 1994j) stipulated an MPC outer cylindricity of 0.65 in. (16.5 mm) on the radius; by difference this would allow a CF disposal container cylindricity of 30-16.5=13.5 mm. Based on the clad inner barrier fabrication concept, this should be ample.

The earlier DHLW disposal container design assumed a somewhat larger cylindricity of 24 mm, prior to adoption of the clad inner barrier fabrication concept. The HLW canisters are nominally 610 mm diameter with 30 mm cylindricity; the statistical envelope of several canisters clustered together might be expected to be slightly less than the geometric envelope comprised of several tangent 640 mm diameter circles. At this point, it could be assumed that a cylindricity less than or equal to 5 mm may be adequate for the DHLW disposal container.

9.3.2 Closure Welding Background

The leading welding method for the waste container closure welds is automated, remote, narrow groove (narrow-gap) welding (CRWMS M&O 1993b; Eichhorn and Borowka 1990), with the expectation of using the same equipment for both inner and outer barrier closure welds. Stresses in the waste container closure region originate in the welding process, due both to material heating and to material shrinkage upon cooling. Thus the more potentially rewarding approach will be to choose welding techniques that inherently minimize stresses induced during welding, rather than to perform postweld stress relief. Zero-clearance, high-energy processes such as electron beam welding most nearly achieve the goal of stress minimization. Narrow groove welding is relatively superior to V-groove processes when applied to heavy sections, due to reduced heat input resulting from reduced weld volume and weld times.

Automated welding processes (preprogrammed computer controlled) are now clearly recognized as having desirable features in terms of improved efficiency and output by virtue of increased deposition rates, higher arc efficiency, and fewer weld defects. Computer monitoring and control of process variables enhances weld quality. (Automated processes do not involve human interaction in the control process; however, process monitoring may be performed by both computer analysis and human observation, at which point human intervention in the process would be possible.) The benefits of automated narrow groove welding techniques may be best realized when applied to thicker sections to reduce joint volumes, reduce/control distortion, and increase joint completion rates.

9.3.3 Nondestructive Examination Background

Nondestructive examination during empty container fabrication will be performed by conventional means typically used in heavy industry, such as radiographic, dye penetrant, ultrasonic, and/or magnetic particle testing. Selection of examination methods suitable for WP remote closure operation in a radioactive environment is a development activity about to begin; expected techniques are ultrasound, and possibly dye penetrants and/or magnetic particle testing for surface defects. Limitations of the feasible nondestructive examination techniques could force reconfiguration of the inner and/or outer closure joint configurations (i.e., inner and/or outer closures to be other than the selected conceptual design flat plate lid configurations). For this reason, the waste container closure and nondestructive examination development tasks need to be performed somewhat concurrently to provide feedback of results from one task to the other.

9.4 WASTE PACKAGE/WASTE CONTAINER CLOSURE AND INSPECTION

Installation of the waste container closure lids will take place in a hot cell at the MGDS repository surface facility Waste Handling Building after loading of the CF disposal container, UCF disposal container, or DHLW disposal container. Each of two closure lids must be separately, remotely welded into place, remotely inspected, and the weld joints possibly stress relieved (heat treatment or other means) to complete the envelope for each corrosion barrier.

The primary development concerns are the interrelated concerns of closure joint design configurations. Joint design configuration must be remotely fabricable by a proven welding technique, must be remotely inspectable in an environment of high radiation and potentially elevated temperatures resulting from the nuclear waste, and should result in the lowest possible postweld tensile stress conditions. Of the many techniques that will be considered during development, the narrow groove weld does hold forth considerable promise as being already developed for thick sections and remote operation.

Various standard industrial remote closure welding processes have been investigated, considering technological readiness and process adaptability to the specifics of this application, the quality of closure welds (weld integrity, and good mechanical properties of the welds and the heat affected zones), economy and time involved in making the closure welds (high deposition rate and minimizing the amount of weld filler material), and viable methods for repair of defective welds or for container replacement if weld repair should be unfeasible. Additional considerations include fully automatic remote closure welding equipment, the ability to use the same equipment for both the thin inner weld and the thick outer weld, and inherent tolerance or hardening capability of the welding equipment to the anticipated levels of radiation exposure.

Closure of either the CF disposal container, UCF disposal container, or DHLW disposal container will be the same, except that the radiation intensity in the area of the closure would be less for the CF case due to the presence of the CF shield plug and lids. As may be seen from WP conceptual designs such as shown on Figure B.2-6, the inner and outer lid welds will be flat position narrow groove welds. The advantages of narrow groove weld are (1) improved economy and closure times as a result of minimum filler metal use, (2) good mechanical properties in the weld and heat-affected zone due to relatively low heat input, (3) fully automatic operation in all welding positions, and (4) reduction of weld-induced stresses and/or distortion.

The narrow groove weld technique is normally associated with steel thicknesses in the range of 25 to 300 mm, or even greater. Narrow groove (narrow-gap) gas-metal arc welding deposition rates of up to 8 kg/hr have been demonstrated. The narrow groove width (generally 10 to 20 mm) and the potentially proportionally greater depth of the groove have promoted the development of a number of automated process-oriented welding head guidance systems. These already-developed, remote, automated systems essentially constitute the equipment needed for remote waste container closure welding, except that the equipment would need to be radiation hardened.

9.4.1 Selected Narrow Groove Welding Process

Preliminary screenings of various standard industrial remote closure welding processes have been performed to select reasonable candidate method(s) suitable for support of ACD activities. The narrow groove welding process is chosen for the features of reduced weld material volume, and correspondingly reduced closure welding times (CRWMS M&O 1993b; Eichhorn and Borowka 1990). The leading methods for waste container closure narrow groove welding are gas metal arc welding, or tungsten-metal arc welding, also known as tungsten inert gas welding. Initial waste container closure weld development work is focussing on automatic gas tungsten arc welding narrow groove welding.

The narrow groove welding process has been selected over V-groove processes because of applicability to heavy sections, welding time reduction due to greatly reduced weld joint volume, and the advanced state of the development of narrow groove welding remote guidance and control (LLNL 1990a). The sides of the narrow groove welding weld groove are normally U-shaped (slightly tapered, near-vertical side walls and a rounded bottom). Even though gas tungsten arc welding deposition rates are less than gas metal arc welding, gas tungsten arc welding has presently been chosen to begin closure weld development testing, as gas tungsten arc welding has a history of producing the highest quality welds, and is commonly used in the nuclear industry. Furthermore, automatic gas tungsten arc welding groove widths may be very narrow; the development program is using 3/8 in. (approximately 10 mm), which has a marked affect upon the amount of filler material required for weld completion. A recently developed narrow groove welding system using gas metal arc welding, with completely integrated process and guidance control (Eichhorn and Borowka 1990), is described later.

The gas metal arc welding process would normally be preferred over gas tungsten arc welding due to higher deposition rates, which may range up to five times higher. Oconee experience has produced deposition rates of 0.5 to 0.9 kg/hr (1 to 2 lb/hr) using gas tungsten arc welding; the narrow groove welding non-optimized gas metal arc welding demonstration (single-wire) process results described herein (Eichhorn and Borowka 1990) produced deposition rates of 2.8 to 3.1 kg/hr (6 to 7 lb/hr). Rates approaching 9 kg/hr have been reported for two-wire gas metal arc welding; this would correspond to a single-wire rate approaching 4.5 kg/hr (Metals Handbook, eighth edition 1971). However, different welding techniques cannot be ranked solely by comparative deposition rate; in the final analysis, it is the time to complete the weld that is a far more important measure. It is this latter measure that becomes important for design of the repository surface facility Waste Handling Building, having a direct bearing on the required number of parallel welding stations.

Use of automatic gas tungsten arc welding during the closure weld development program, with a very narrow groove and possibly improved deposition rate, is anticipated to somewhat narrow the apparent disparity of gas tungsten arc welding as compared to gas metal arc welding, when measured in terms of time to complete the weld. As closure weld development progresses, consideration will be given to using the superior quality gas tungsten arc welding for both the inner lid weld and for the outer lid root pass weld, and then use of the higher deposition rate gas metal arc welding to complete the thick outer lid weld.

9.4.2 Electron Beam Welding Process

Selection of the high-energy, single-pass vacuum electron beam welding process has for the present been preempted by the relatively slower multiple-pass narrow groove welding gas tungsten arc welding and gas metal arc welding processes. (Single-pass indicates that a full-depth weld is completed in a single pass through the close-fitted butt joint.) This decision was predicated on (1) the comparative technological readiness of narrow groove welding gas metal arc welding, and (2) designs for the Waste Handling Building studies would be more conservative as a result of factoring in the longer weld times for narrow groove welding gas metal arc welding or gas tungsten arc welding (slower throughput).

Although choice of electron beam welding has been dismissed, electron beam welding does have some very desirable features: only a single weld pass is required to complete a weld, the near-zero-gap weld requires no filler material and minimizes shrinkage, the intense almost-instant local heating results in far less heat input and very substantial reduction in size of the heat-affected zone, and weld joint distortion is nearly eliminated due to performing a full-penetration weld in a single pass. However, other features of electron beam welding are less desirable, including high equipment costs; requirements for high power, a vacuum chamber, and close tolerance for the joint (nuclear waste heat could distort the joint); the need to manage beam energy emanating from the weld back-side (would affect joint design); and the difficult aspects of weld initiation and termination which may affect joint design.

9.4.3 Nondestructive Examination

Nondestructive examination comprises a range of test methods for detecting discontinuities (flaws) in a material without causing damage. There are five main nondestructive examination test methods:

- Penetrant (dye) testing
- Magnetic particle testing
- Eddy-current testing
- Radiographic testing
- Ultrasonic testing

Radiographic testing would not be considered due to the internal radiation emissions.

Any discussion of nondestructive examination requires a clear definition of the term "defect." A defect is a discontinuity (flaw) that creates a substantial risk of failure in a component or structure during its service life. Although the aim of nondestructive examination is to detect defects, it provides evidence of flaws as well. The significance of these flaws and whether they are actually defects must be determined (i.e., every detectable flaw will not necessarily fit the definition of a defect). Since the waste package barriers serve as corrosion barriers, and do not function as a pressure vessel, the definition of a defect will relate to the consequence of the defect upon the integrity and durability of the corrosion barriers.

The first three nondestructive examination methods are limited to detection of surface-breaking, or possibly near-surface, flaws. To be of use for full volume closure weld inspection, the method would have to be employed between successive welding passes. To use either of these nondestructive examination methods is an unlikely prospect; such inspection would be performed within the confines of the narrow weld groove, and any possible remaining inspection material residue would have to be satisfactorily managed so as to avoid any subsequent impairment of weld quality. Inspection after each pass has definite advantages, as repair could be affected before that pass is overlaid. However, note that further limitations of each method may preclude consideration of its use (these limitations are discussed briefly in the American Society for Metals Handbook (ASM 1993; pp. 1081-1088). Should one of these methods prove feasible, it is possible that the rate of inspection could be as rapid as the welding speed forward, in which case nondestructive examination would be concluded somewhat coincidentally with the welding, rather than performed postweld.

Ultrasonic nondestructive examination may be found most feasible, as long as the technique can be successfully applied to the WP joint configurations and section thicknesses for each material. It is presumed that ultrasonic nondestructive examination would be performed postweld; as a result, this assumption has been adopted for purposes of estimating the total time for waste container closure operations.

9.4.4 Advanced Narrow-Gap Welding

The following describes an example of an advanced narrow-gap (narrow groove) welding system recently developed in Germany (Eichhorn and Borowka 1990) with completely integrated process and guidance control, with no external measuring devices, and controlled only from measurement information taken directly from the electrical process variables. This system ensures not only correct weld head positioning, but also uniform buildup of weld layers despite changes in gap width (regulation of fill ratio by varying welding speed). The compact welding head structure results in a gap width of only 10 to 15 mm as being adequate for workpiece thicknesses up to 200 mm. The demonstration weld head, applied to a low-alloy steel workpiece 100 mm thick with 12 mm groove width, produced a weld "with the exception of the root pass, the seam is perfect" (Eichhorn and Borowka 1990; the root defect was knowingly accepted to limit testing cost).

The single-wire rotating-tube welding method produces an oscillating movement of the arc from side to side in the gap. Low-spatter, pulsed-arc metal transfer is used, which is suitable over a wide range of operation. The aim of the welding head guidance system is to deduce correcting signals from the electrical process variables for lateral positioning and distancing of the welding head (the arc itself being used as a feedback transducer to form the distance profile of the welding spot, as the arc scans the welding site as part of the oscillating movement). The oscillating motion and tailoring of the pulsed-arc on-time profile permits dwell at the sidewalls to ensure adequate penetration.

The system employs the gas metal arc welding process, which is sometimes also identified as metal-inert gas welding. Gas metal arc welding can be accomplished in all positions; continuous wire feed enables long welds to be deposited without starts and stops; and minimal postweld cleaning is required because of the absence of heavy slag.

Eichhorn and Borowka state that the welding head guidance system results in "noticeable reduction in the equipment costs...and considerably shortens the welding downtimes, as a result of its integrated process control and quality assurance functions."

The following narrow groove welding process parameters are taken from a paper by Eichhorn and Borowka (1990) (the parameters were not optimized with regard to deposition rate):

Workpiece thickness	100 mm
Narrow-gap width	12 mm
Welding speed forward	110 mm/minute
Weld buildup rate	4.5 to 5 mm/pass
Workpiece material	St 52-3 (presumably German [DIN] 1.7138, 52MnCrB3, similar to SAE 50B50 low-alloy steel)

9.4.5 Waste Package/Waste Container Closure Operations Total Time Estimates

The foregoing narrow groove welding parameters, with the exception of weld buildup rate, have been applied to several WP sizes to calculate estimated welding times. The same weld buildup rate has been assumed for C71500 cupronickel as for A 516 carbon steel. The selected weld buildup rate has been reduced to 3.18 mm/pass (1/8 in./pass), based on welding experience, to adjust the calculated uninterrupted welding time estimates for the net affect of various factors, both positive and negative, such as

- The aforementioned parameters were not optimized for weld deposition rate.
- Normal interruptions to operation, e.g., connecting a new coil of wire.
- Improvements may be expected over this 1989-90 technology.
- Parameters suitable for the inner barrier nickel alloy may be different than for steel.
- WP lid (weld) thicknesses may change as the conceptual designs evolve.

Estimates of welding operation times presented in this section are for the high thermal load/dry repository design conditions. Welding time calculations are based on conservatively oversized lid thicknesses of 25 mm for the inner lid and 110 mm for the outer lid, versus nominal barrier thicknesses of 20 mm and 100 mm (55 mm lid thickness versus 50 mm nominal barrier for the DHLW outer barrier). Choosing the conservative weld buildup rate of 3.18 mm/pass results in total number of passes per lid for closure of 8 and 35 for the inner and outer lids, respectively (18 passes for the DHLW outer lid). Welding time and nondestructive examination time estimates are listed in Table 9-1.

Total time for closure operations includes time for welding, nondestructive examination, and all other hot cell serial closure operations that must be performed in addition to the welding and nondestructive examination. Time estimates for the other operations are detailed in the following subsection, and include equipment mounting and dismounting, joint preheat and/or heat treatment and cooldown, and time required for quality control inspection and documentation. Total times for closure operations are in Table 9-2.

The engineering estimates for nondestructive examination time are presently assumed to occur postweld and to require a time interval proportional to weld time (which relates to weld circumference and material thickness) plus a constant interval. The assumed factors are (1) inner lid on destructive examination = 1 times weld time plus four hours, and (2) outer lid nondestructive examination = 0.75 times weld time plus four hours. The estimated times include quality control inspection and documentation. The nondestructive examination time estimates are unsubstantiated engineering estimates and lack the methodical justification such as presented for weld times.

Section 9.4.6 presents more information on WP design details. It also details weld preheat requirements for the WP materials of construction and requirements for postweld heat treatment. Also, note that the time estimates are made for only those operations that must take place serially within the waste container closure hot cell, beginning with the CF disposal container, UCF disposal container, or DHLW disposal container already filled, ready for placement of the inner lid.

Table 9-1. Time Estimates for WP/Waste Container Closure Operations: Welding and Nondestructive Examination

Waste Package	Inner Lid Weld Diameter (mm)	Outer Lid Weld Diameter (mm)	Calculated Inner Lid Weld Time (hrs)	Calculated Outer Lid Weld Time (hrs)	Sum of Calculated Weld Time (hrs)	Time for Quality Control Inspection & Documentation (hrs) (1)
Large CF WP	1568	1602	6.0	26.7	32.7	6
Small CF WP	1297	1331	4.9	22.2	27.1	6
DHLW WP	1575	1615	6.0	13.8	19.8	6
Large UCF WP (21PWR)	1395	1435	5.3	23.8	29.1	6
Small UCF WP (12PWR)	1064	1104	4.1	18.3	22.4	6

(1) Engineering estimate, total for both inner and outer welds.

Table 9-2. Time Estimates for WP/Waste Container Closure Operations: Other

Waste Package	Total Weld Time w/QC Inspection and Documentation (hrs)	Inner Lid Weld NDE Time (hrs) (1)	Outer Lid Weld NDE Time (hrs) (1)	Inner Lid other Weld Operations (hrs) (2)	Outer Lid other Weld Operations (hrs) (2)	Total Time for Closure Operations (hrs)
Large CF WP	38.7	10.0	24.0	5	35	113
Small CF WP	33.1	8.9	20.6	5	35	103
DHLW WP	25.8	10.0	14.3	5	35	90
Large UCF WP (21PWR)	35.1	9.3	21.9	5	35	106
Small UCF WP (12PWR)	28.4	8.1	17.7	5	35	94

(1) Engineering estimate, including quality control inspection and documentation.
(2) Engineering estimate, including quality control inspection and documentation; Section 9.4.6.

The possible use of filler material in the UCF disposal containers and CF canisters is an option yet to be decided. In the event that some of the CF canisters must be opened and filler material added (optional), reclosure welding times for the CF may be assumed to be similar to those of the inner lid of the CF WP, plus the assumed additional time for nondestructive examination and other operations. However, the expected scenario for opening, filling, and reclosing CF would result in these operations being performed separately from, and in parallel with, waste container closure.

9.4.6 Waste Package/Waste Container Closure Serial Operations Time Estimates

The following time estimations are based on multibarrier UCF disposal containers and CF disposal containers comprised of a 20 mm inner barrier of Alloy 825 and a 100 mm outer barrier of A 516 carbon steel, and the DHLW disposal container comprised of a 20 mm inner barrier of Alloy 825 and a 50 mm outer barrier of C71500 cupronickel. Time estimates for the various operations involved in closure, except for individual welding times and weld inspection times, have been assumed to be the same for both the large and small WPs (e.g., 21 PWR and 12 PWR sizes). The time estimates are best judgment engineering estimates, except for the narrow groove welding time estimates. Welding time estimates may be traced back to the demonstrated work (Eichhorn and Borowka 1990), as detailed in the preceding subsection.

The time estimates are made only for those operations that must take place serially within the waste container closure hot cell, beginning with the UCF disposal container, CF disposal container, or DHLW disposal container already filled, ready for inner lid placement. Any operations performed preceding inner barrier lid placement, such as final preparation of weld surfaces, have been excluded. The option to add filler material at the MGDS to some/all of the UCF disposal containers or CF canisters has not been included in the time estimates, as the requirement to use filler material has yet to be established.

9.4.6.1 Preheat Temperature

Required preheat temperatures are 16°C (60°F) for Alloy 825 and C71500, and 107°C (225°F) for A 516 (ASTM 1990). (Note: In the following, the word shell refers to the disposal container without lid, as received from the fabricator.) The inner shell is assumed to be preconditioned to ≥16°C without the necessity for joint preheat within the hot cell (contributing conditions would include thermal conditioning if required before moving into the hot cell, plus heating due to heat output from the WP contents).

WP shell temperatures have the potential to be affected by heat output of the waste once the waste has been placed within the disposal container. The large WPs are limited to a heat output of 14.2 kW; considering the mass of the shell and this maximum value of heat output, the idealized heating rate of a perfectly insulated shell would be less than 5°C/hr. The reality of the situation will be considerably different than the foregoing idealized calculation, but the calculation serves to illustrate the rather modest affect on thermal conditioning that could be attributed to waste-generated heat.

Maintaining the DHLW disposal container C71500 outer barrier weld area temperature within the required preheat temperature range should not be a problem.

However, since the inner barrier is clad to the outer barrier, the outer barrier carbon steel weld lands will experience substantial heating during the inner lid welding operation. This heating may be sufficient to raise the carbon steel weld area temperature to at least the A 516 preheat minimum of greater than or equal to 107°C. Actually, the solution should be as simple as predetermination of an empty disposal container/WP precondition temperature, which would ensure that the minimum preheat temperature of greater than or equal to 107°C is attained as a consequence of all of the steps that precede welding the outer lid in place. Otherwise, the shell top end may require additional welding preheat. Due to the wide range allowed for A 516 weld preheat temperature, it is not anticipated that any cooling of the shell top end would be required. The A 516 top lid will require welding preheat, heated in a separate parallel operation before being placed in position.

The circumstance of considerable heat output from most of the filled waste containers will require that some cooling be provided to the hot cell, since the waste container surface temperature will need to be kept below approximately 200°C. Furthermore, design features will be necessary to ventilate welding fumes, which will include the welding cover gases. As a result, continued heat input to the outer barrier closure joint may possibly be required to maintain preheat temperature greater than or equal to 107°C. For example, it will require about 45 minutes to make a single weld pass (one revolution) around a large CF disposal container closure joint, so it is possible that welding heat input alone might be insufficient to maintain preheat temperature.

9.4.6.2 Postweld Heat Treatment

No postweld heat treatment of Alloy 825 is required to maintain or restore corrosion resistance; the same appears to be true for the C71500 cupronickel. The thick A 516 carbon steel outer barrier may require postweld stress relief heat treatment in the range of 525°C to 675°C. Localized heating of the closure joint region may be achieved using resistance heating or induction heating techniques; however, the heat treatment equipment must be designed to prevent possible overheating of the waste container contents. Various alternative techniques to reduce weld-induced stresses, such as temper beads or subharmonic vibration, will be evaluated during the closure weld development program, with the objective of eliminating any need for postweld heat treatment of the A 516 outer barrier weld joint.

9.4.6.3 Time Estimates

The time estimates for those serial operations that must be performed in the hot cell to accomplish waste container closure, excluding time for welding and for nondestructive examination, are listed in Table 9-3. Unless otherwise indicated, the time for any of the listed operation steps is a total of time for the action plus time required for quality control inspection and documentation. A further refinement would find outer barrier postweld heat treatment time to be somewhat dependent on waste container size, the large waste container having 20 to 30 percent greater weld circumference than the small package.

9.5 WASTE PACKAGE INTERNAL FILLER MATERIAL

Within Section 9.5, the terms WP or waste container should be interpreted to mean the spent nuclear fuel-containing item, which therefore includes both the CF canisters and the UCF disposal containers, but excludes DHLW disposal containers.

As part of the MGDS WP design and development activities, it has been determined that addition of material (to fill the free spaces within selected CF canisters and/or UCF WPs) may be needed for criticality control. Use of filler material could also be beneficial to long-term containment and isolation by inhibiting corrosion and inhibiting release of radionuclides. The benefits to be derived

Table 9-3. Time Estimates for WP/Waste Container Closure and Inspection Operations

Operation	Time Estimate, hours
Inner closure lid	
Lid placement	2
Mount/position welding equipment	2
Perform closure weld(1)	-
Dismount welding equipment	1
NDE inspection of closure weld(2)	-
Subtotal	5+
Outer closure lid	
Mount-preheat heaters(3)	1
Preheat shell top end (closure joint)	2
Preheated lid placement	2
Mount/position welding equipment	2
Perform closure weld(1)	-
Dismount welding equipment	1
Dismount preheat heater	1
Pre-NDE cooldown (in addition to prior two steps)	0
NDE inspection of closure weld(2)	-
Mount heat treatment heater(3)	1
Postweld joint heat treatment	20
Cooldown (in addition to prior step)	4
Dismount heat treatment heater	1
Subtotal	35+

Notes:
(1) Narrow groove closure weld times appear in the previous subsection.
(2) Nondestructive examination inspection time estimates appear in the previous subsection; the particular technique is yet to be determined.
(3) The preheat heater and heat treatment heater may or may not be the same set of equipment; however, it is assumed that the heater must be removed and the joint allowed to cool somewhat before performing nondestructive examination.

through the addition of filler material will have to be satisfactorily established by analyses, including performance analysis, before the decision can be made whether filler materials should be used in any case, and then under what circumstances or conditions would filler be added to a given spent nuclear fuel WP.

Preliminary results of both thermal investigations and neutronics investigations relative to usage of a specific candidate filler material (graded iron or steel shot) have been reported in *Initial Review/Analysis of Thermal and Neutronic Characteristics of Potential MPC/WP Filler Materials* (CRWMS M&O 1994c). Further evaluations are planned.

A draft analysis has been prepared entitled *Analysis of MPC Access Requirements for Addition of Filler Materials* (CRWMS M&O 1995ad). The design analysis serves as input to the CF design, stipulating CF design requirements imposed by MGDS so as to not preclude opening of a CF canister at the repository, adding filler material, and resealing the CF canister.

A filler development test program has been initiated, to be performed in accordance with the *Spent Nuclear Fuel Waste Package Filler Testing Technical Guidelines Document* (CRWMS M&O 1995ac). The development test program is discussed in Section 9.5.3.

9.5.1 Objective/Purpose of Using Filler Material

The use of WP internal filler material versus filling the free space with an inert gas is a waste disposal design issue yet to be resolved. The choice will be determined by the benefits or penalties related to use of filler materials, as derived from future engineering studies and performance analysis. Filler materials may be solids placed while in a liquid state such as low-melting-temperature metals, graded granular solids such as small iron shot, or fine materials such as dry cementitious mixes. Cementitious material would be placed in the dry, unreacted state; at such time as the barriers might breach, any water entering the WP would react with the material, causing it to solidify.

However, program requirements prescribe that the design of the waste packages and the repository permit retrievability of waste packages until time of repository closure. Given this programmatic requirement, a material such as a solid placed while in a liquid state would almost certainly be excluded from consideration, as extraction of spent fuel assemblies encased within the solid material would be extremely difficult. Granular solids or dry cementitious mixes should be expected to make extraction of spent fuel assemblies more difficult, but still manageable; note that the retrievability period is sufficiently brief that no waste packages would have been breached, so a cemetitious material would still be in the dry unreacted state upon retrieval. Although unacceptable due to current programmatic requirements, these materials will nonetheless be included in the following discussions which develop and compare candidate filler materials, regarding their physical characteristics and attributes.

To be effective, criticality control by moderator displacement necessitates that the amount of filler be maximized to minimize the amount of void space remaining, which could then fill with moderator upon breach of the corrosion barriers, and breach of the CF shell, if present. Thus, designs of the spent nuclear fuel basket and other internal structures for both CF canisters and UCF disposal containers (when oriented vertically) must be configured: (1) to provide access to essentially all free spaces, including free spaces within any flux trap basket designs, and (2) to permit achievement of a certain minimum percentage free volume fill with filler material (the selected value is 85 percent fill, based on loose as-poured filler bulk density). Free volume is defined as the CF canister or UCF disposal container internal volume less displacement volume of all objects therein (e.g., fuel assemblies, basket components). Void volume is the difference between free volume and actual volume displaced by the filler material (e.g., includes interstitial volume between filler particles plus volume devoid of filler material).

Neutronics and thermal studies of filler material usage, such as that discussed in the *Initial Review/Analysis of Thermal and Neutronics Characteristics of Potential MPC/WP Filler Materials* (CRWMS M&O 1994c) must compare use of various candidate filler materials versus only an inert gas in the free spaces within the WP. The primary purposes of adding filler to a WP are chemical buffering for radionuclides, and providing or contributing to long-term criticality control of the spent nuclear fuel by the moderator displacement criticality control method. It may be expected that those WPs containing spent nuclear fuel with lower criticality potential would be exempt from addition of filler material.

Addition of filler material would take place only at the MGDS, added remotely to a CF canister. For the CF case, the CF canister would be cut open at the MGDS, then filler would be added, and the CF canister resealed. A means of measuring emplaced quantity of filler material would be required to establish that placement of the proper total quantity was actually accomplished, to confirm a minimum percentage fill of free space within the canister. Current Surface Facility designs are based on the CDA assumption that only the first procurement quantities of CF canisters may require addition of filler material. The Surface Facility is not designed to add filler to the balance of the spent fuel waste packages, whether CF canisters or UCF disposal containers.

The choice to use WP filler materials would be based on the necessity to achieve one or more of the technical objectives appearing in the following list; different candidate materials would achieve differing objectives within the list:

1. Filler material could aid in criticality control by moderator displacement; in the event of repository flooding and a breach of the disposal container, the maximum amount of water that could be present in the container would be substantially diminished by the presence of the filler material.

2. Filler material could act as a chemical buffer for radionuclides in the event of water intrusion into the WP upon breach of the disposal container.

3. Filler material could provide cathodic protection in the event of water intrusion into the WP upon breach of the disposal container by selection of a filler material having the highest electrochemical activity in comparison to other materials present in the WP.

4. Filler material could function as mechanical packing to inhibit movement (collapse) of other materials internal to the WP (fuel rods, fuel pellets, and/or basket materials).

5. Filler material could improve thermal conductance, hence improving heat transfer and decreasing fuel rod temperatures.

6. Filler material could function as a neutron absorber.

However, use of filler material would significantly increase WP mass and cost. The use of filler material would also add significant capital and operating costs and complexity to the waste handling operations, depending on the number of waste packages requiring filler and the type of filler material selected. There is also the potential of significant costs associated with disposal of the filler material as low level waste, should the option to retrieve waste packages actually occur.

9.5.2 Candidate Filler Materials

Selection of candidate filler materials must consider the effects that the presence of such material would have on the spent nuclear fuel rod cladding temperatures, as compared to having the free space filled with only an inert gas. This concern would include the brief interval of filler material placement, as well as the extended waste disposal containment period. Filler materials placed in a molten state should have melting temperatures somewhat above the disposal period fuel cladding temperature limit so that the filler would remain in the solid state while in the repository. A brief, modest excursion above this cladding temperature limit as a result of molten filler material placement may not cause consequential cladding damage, as such damage is a cumulative process and depends on a time-at-temperature integrated effect, in addition to the absolute temperature. However, such a material is not considered to be a viable candidate due to the program requirement to maintain the option of waste package retrieval and spent nuclear fuel recovery.

Many candidate filler materials may be expected to increase WP internal thermal resistance. In that instance, cladding temperature limitations may require (1) that the basket thermal conductance be enhanced accordingly to offset the thermal resistance increase due to the filler, or (2) a reduction in maximum permissible WP size (fewer spent nuclear fuel assemblies).

Desirable attributes of candidate filler materials would include ability to displace water from the WP interior free spaces, cathodic protection, chemical buffering of radionuclides, higher thermal conductivity, ease and rapidity of filler emplacement including assurance of attaining minimum acceptable percent free space fill, melting temperature within an acceptable range if the material is to be placed in the molten state, lower density, naturally plentiful, and inexpensive for the required material purity. Additional attributes include minimizing overall system costs, minimizing operational complexities regarding achievement of required percentage fill, minimizing secondary waste generation, and minimizing impacts on performance confirmation and the option for retrievability. Sample materials that might be considered as candidate WP filler materials are listed in Table 9-4.

Individual physical characteristics within this group of candidate materials are quite varied. The thermal conductivity of natural magnetite would categorize it as a thermal insulator. Borosilicate glass appears to be an even better insulator. Iron shot has about an order of magnitude higher thermal conductivity than magnetite, but is still quite low. Tin is not sufficiently plentiful. Lead is toxic, very heavy, and can cause embrittlement of other metal components. Unalloyed zinc is

Table 9-4. Sample Materials for Consideration as Candidate WP Filler Materials

Material	Melting Point, °C	Specific Gravity	Thermal Conductivity (k) W/m·K	Comments
Tin (place molten)	232.0	7.31	64	Not plentiful, no U.S. source
Lead (place molten)	327.5	11.35	34.6	Considered toxic, very heavy
Zinc (place molten)	419.6	7.13	115	Use lower temperature Zn-4Al alloy instead
Zn-4Al alloy (place molten)	381-387	6.6	113	AG40B die cast alloy, inexpensive
Magnetite, Fe_3O_4 natural ore (granular, graded)		2.67(1)	0.284(2)	Used in nuclear testing at the Nevada Test Site, "flows like water"
Iron shot (graded) or steel shot (graded)		4.8-5.3(1) (same)	~1-4(2) (lower)	Ratioed from thermal k of pure iron=80.3, using (solid/granular ratio of k for magnetite of ~20:1)
Chemically resistant borosilicate glass (e.g., Pyrex™)		2.40 (solid) ~1.6 (beads)	1.25 (solid) ~0.03 (est. for beads)	Boron content would aid in criticality control

(1) Granular form, not monolithic.
(2) Conductivity with heavier fill gas (air, nitrogen, argon); bulk conductivity would increase with He as fill gas.

rejected as it has been determined that the zinc will interact with the Zircaloy fuel cladding material. Possibly the Zn-4Al alloy would be acceptable from that standpoint, as it would have a reduced tendency to interact with the cladding.

Of those candidates listed above, the best choices appear to be zinc alloy and iron shot. Zinc alloy mass would be about 50 percent higher, but the zinc alloy conductivity is between one and two orders of magnitude higher. Zinc is a plentiful, low-cost material. The zinc alloy melting temperature of less than 400°C would not be considered harmful to the fuel rod cladding due to the brief interval at this temperature. Other zinc-aluminum alloys could be chosen if a somewhat higher melting temperature were desired. However, use of a molten filler material that solidifies on placement is presumed to be unacceptable from the standpoint of the WP retrievability requirement; if retrieval were for the purpose of spent nuclear fuel reprocessing, having the spent nuclear fuel cast in place within the WP would certainly complicate spent nuclear fuel removal.

Iron shot has been chosen as the first candidate filler material to be investigated. Characteristics of iron shot that lead to this choice include relative ease of placement (near-spherical shot "flows" readily), commercially available in a variety of graded sizes, inexpensive, a plentiful natural resource, a reactive anodic material providing protection to the fuel cladding and to 316 stainless steel components, would inhibit radionuclide release, and contributes to a small degree to neutron absorption. In anticipation that filler material thermal conductivity could be of concern, iron shot is presently preferred over steel shot due to higher thermal conductivity. The choice of iron shot is not exhaustive or exclusive, rather it is based on engineering judgment.

9.5.3 Development Testing

WP filler material development testing has been included as part of WP design and development, as stated in the *Waste Package Implementation Plan* (YMP 1995) and the *Waste Package Engineering Development Task Plan* (CRWMS M&O 1993b). Filler material development work will be applicable to both the UCF WP and the CF engineering development activities. A filler development test program is scheduled was begun early in fiscal year 1996, being performed in accordance with the *Spent Nuclear Fuel Waste Package Filler Testing Technical Guidelines Document* (CRWMS M&O 1995ac).

The objective of the testing will be to determine procedures for addition of filler material to a WP before geologic disposal. The goal in adding filler material is to achieve a high percentage fill of the free space around the simulated spent nuclear fuel assembly. Small-diameter iron/steel shot has been chosen as the candidate spent nuclear fuel WP filler material to be used in the filler placement portion of the test program (either iron or steel shot is acceptable for the placement testing). Such shot is commercially available, inexpensive, will provide ease of handling, and achieves a number of the technical objectives that have been established for filler materials so as to benefit criticality control and/or inhibit release of radionuclides.

The required development activities, as described in the *Technical Guidelines Document*, are execution of the test plans and preparation and submittal of associated test reports upon completion of the testing programs. The primary test plan will detail a testing program to investigate placement of steel shot filler material into a partial, simulated pressurized water reactor (PWR) spent nuclear fuel basket cell, such as within a WP or CF basket. The secondary test plan will determine selected physical properties of the bulk shot; bulk thermal conductivity over a range of temperatures for various shot samples, bulk density, angle of repose measurement, and determination of any additional filler material properties at elevated temperatures as may be required. The shot volumetric placement portion of this testing program may use either steel or iron shot, providing the size is correct. The physical properties portion of this testing program will determine the needed properties of both near-pure iron shot and steel shot. No filler placement testing or physical properties testing is planned for the HLW glass type of WPs.

As noted in the *Waste Package Engineering Development Task Plan* (CRWMS M&O 1993b), a development testing program is required to determine procedures for addition of filler material, as necessary to achieve a high percentage fill of the free space around the simulated spent nuclear fuel assembly. The testing program will fabricate and use two different dimensionally accurate dummy PWR nuclear fuel assemblies. Each assembly will be installed in turn within a partial, simulated spent nuclear fuel basket test fixture equivalent in size to a single cell within a canister or UCF disposal container basket. Testing will be performed with various shot sizes, with the goal of achieving at least 85 percent free space fill (excludes filler interstitial void space) under passive gravity-fed fill conditions. The 85 percent figure is an engineering estimate believed to be reasonably achievable. Methods to improve fill percentage above the 85 percent level will be investigated.

9.5.4 Iron Shot Filler Material in a 21 PWR Canister Waste Package

Initial neutronics and thermal investigations (CRWMS M&O 1994c) regarding use of filler were based on the large multibarrier WP design, namely the 21 PWR case. The specific design is the *21 PWR Burnup Credit MPC Conceptual Design* (CRWMS M&O 1994k). The 21 PWR case presents a higher thermal output and higher criticality potential than the companion 40 BWR case, or any of the smaller WPs.

9.5.4.1 Neutronics Investigations

Results of the neutronics investigations were reported in the *Initial Summary Report for Repository/Waste Package Advanced Conceptual Design* (CRWMS M&O 1994i). The analysis investigated spent nuclear fuel criticality control by the moderator displacement criticality control method, to obtain a measure of the benefit of this approach. A system model was used to investigate the circumstance of water (moderator) intrusion into a breached WP, flooding to completely fill the available free space (all free space filled uniformly with various compositions of near-pure iron shot and water). This analysis considered no other criticality control measures in conjunction with the moderator displacement method.

The effectiveness of moderator displacement criticality control depends on the quantity of moderator displaced by the filler, which in turn depends on the quantity of filler that may be emplaced into the CF canister WP. The total quantity of iron shot filler is dependent on the density of the bulk material and the percentage fill.

Based on manufacturer's data for graded shot (near-uniform size, see Section 9.5.5), bulk density of about 4,800 kg/m^3 would indicate a bulk interstitial void fraction of about 39 percent (based on iron density of 7,870 kg/m^3). By mixing grades of shot, bulk density may be increased to about 5,250 kg/m^3, which would reduce the bulk void fraction to about 33 percent. However, the possibility of selecting mixed grade shot would have to be deferred until successful resolution of the potential problem of size separation and stratification (see Section 9.5.5), which could result from WP handling and emplacement in the repository.

The fraction of free space fill chosen for the neutronics analysis ranged between the development target goal of 85 percent fill and a maximum fill of 100 percent. Combining the foregoing densities and fill percentages resulted in available space for moderator ranging between 48 and 33 percent (i.e., 1.0-0.85*[1.0-0.39]]=0.48, and 1.0-1.0*[1.0-0.33]]=0.33). The analysis assumes access to all spaces within the WP/canister; i.e., none of the spaces between the basket and the container inside diameter were sealed off, since corrosion through that barrier would allow the sealed-off space to fill with moderator.

Basically, the shot filler conditions discussed above do not result in sufficient moderator displacement. The results do show a significant reduction of criticality potential, but that reduction alone is not sufficient for the conservative design basis spent nuclear fuel conditions chosen (fresh fuel, no burnup). This moderator displacement criticality control approach may be viable for spent nuclear fuel with lower reactivity, or when used in combination with burnup credit and/or credit for supplemental neutron absorber materials. Refinement of the criticality analysis model is planned, which may alter these results and conclusions.

The foregoing criticality analyses basically assumed a worst-case circumstance of (1) no burnup credit, (2) no neutron absorber materials, and (3) the sudden catastrophic breaching and flooding of the WP. Subsequent to this investigation, Performance Analysis will assess the probability of such an extreme occurrence. The far more likely circumstance would be a non-catastrophic WP breach resulting in slow intrusion of moderator (water), at a rate so slow as to result in coincident oxidation of the iron shot. The slow intrusion of water and resultant conversion of iron to iron oxide would slowly reduce the remaining internal void space (space which potentially could be subsequently rapidly filled with moderator), thereby slowly reducing criticality potential even further. A secondary aspect of oxidation of the iron shot is the slow creation of a swelling, welding mass of iron oxide that would function as mechanical packing to inhibit movement (collapse) of other materials internal to the WP (fuel rods, fuel pellets, and/or basket materials). Furthermore, the swelling mass of iron oxide would provide a degree of chemical buffering, which would inhibit further corrosion.

9.5.4.2 Thermal Investigations

Results of the thermal investigations were reported in the *Initial Summary Report for Repository/ Waste Package ACD* (CRWMS M&O 1994i). The initial thermal investigations regarding WP filler material focused on the effects of changes in thermal resistance of the various pathways within (1) the fuel assemblies, (2) the basket (being the primary thermal shunt to the WP shell), and (3) the spaces between the basket and the WP barriers. The effect of adding filler material to the free spaces is to alter the operative heat transfer mechanisms, from the combination of radiation plus gas conduction, to conduction only in the iron shot bulk filler material.

No source of information was discovered to provide actual values for iron shot bulk thermal conductivity. Prior work at the Nevada Test Site (RSN 1992) had occasioned the measurement of thermal conductivity of roughly-spherical granular magnetite, a naturally occurring iron oxide ore (Fe_3O_4). The granular magnetite exhibited a thermal conductivity approximately 1/20th that of the crystalline material. Assuming that the same ratio might apply to iron shot versus iron (pure iron conductivity = 80.3 W/m·K) would suggest iron shot conductivity in the range of 4 W/m·K. The thermal calculation was therefore based on a non-conservative iron shot thermal conductivity of 4 W/m·K, as well as a more conservative lower value of 1 W/m·K.

The calculated thermal effect of using iron shot indicates that peak cladding temperature would decrease somewhat for the higher assumed filler thermal conductivity, but would increase slightly for the lower assumed filler conductivity. Results of these calculations demonstrate that a filler material with a reasonable conductivity can reduce internal temperatures compared to the case of helium fill gas only. However, if the filler material bulk thermal conductivity is poor (such as the conservative 1 W/m·K case), internal temperatures can be increased by the addition of filler material. This effect can seriously impact large CF canister WPs, where internal temperatures can approach maximum thermal limitation goals.

Steel shot is more commonly available than near-pure iron shot; however, nominal thermal conductivity of steel is generally only 2/3 that of pure iron. As a result of the aforementioned concerns regarding filler bulk thermal conductivity, attention has focused more on the commercially available 99.8 percent iron shot. The filler development program thermal conductivity determination will be testing both the 99.8 percent iron shot and commercial steel shot.

9.5.4.3 Preliminary Results

From the preliminary criticality calculations, moderator displacement alone does not achieve the needed level of control for the assumed conservative design basis fuel (fresh fuel, no burnup). Other criticality control measures would be needed in conjunction with iron shot filler, such as adoption of burnup credit and/or credit for supplemental neutron absorber materials.

Preliminary thermal investigations of the effect of iron shot filler material indicate that WP internal thermal conductance appears to possibly be improved. However, this cannot be definitively stated due to the uncertainty of bulk iron shot thermal conductivity. The bounding values of the estimated thermal conductivity resulted in one calculated peak fuel cladding temperature slightly above, and the other slightly below, the peak cladding temperature calculated for the reference case with no filler material (filled with helium gas only).

These preliminary results indicate that iron shot should continue to be considered a potentially viable filler material. The investigations must be continued and expanded upon. Collection of the physical properties of the filler material will be developed during the filler development testing program described in Section 9.5.3. After the physical properties are better defined, detailed evaluations can be performed to better quantify filler material worth.

9.5.5 Iron Shot Filler Material

The present choice of filler material is graded near-pure iron shot, in the nominal range of 0.5 to 1.5 mm diameter (SAE Size No. S170 to S460, SAE Specification J444). The purpose of development testing will be to demonstrate achieving a high level of free space filling about an spent nuclear fuel assembly resting within a test cell (simulated PWR basket cell). The goal will be demonstration of at least 85 percent free space fill; that is, shot to fill the given percentage of the total free space, but not counting the interstitial void space within the bulk shot.

9.5.5.1 Shot Sizing

Shot is produced by atomizing molten metal, wherein the droplets assume nearly spherical shape before solidification. Shot size may be as high as 6 mm (approximately 1/4 in.) with current production hardware. The product is normally graded into various sizes. According to Mr. Reeves of Ervin Industries, newer production techniques are available to produce quite small shot (approximately 0.4 mm and smaller) with more uniform size distribution, improving yield within the nominal size range.

To avoid any tendency of shot size stratification within the WP free space, present thinking is to limit shot size to a fairly narrow grading size band (apropos the question of why the larger nuts rise to the top of the can during handling). Cost of the graded shot would depend somewhat on the ability of the process to provide a reasonable yield in that size range, as the rejected shot would have to be recycled back to the process. It is recognized that mixed grade shot provides a denser bulk material (approximately 10 percent), which reduces bulk interstitial void space.

The present filler test program is being limited to testing of different sizes of graded shot; there is no plan at present to pursue examination of mixed grade shot until the likelihood of using filler becomes more apparent. At that point, the relative neutronic benefit of increased bulk density (reduction of void space) would be examined more closely, and a shot production process cost analysis would be performed. Based on the outcome of the neutronic and economic analyses, a decision would be made whether to expand the filler development testing program to study mixed grade shot (aspects of filler placement, bulk density, bulk thermal conductivity, and tendency for size stratification).

SAE has specifications for shot screenings, such as Specification J444. Typically, any specified shot size number has three or four combinations of screen sizes, each with a related percentage of shot that must pass or not pass through. The central two screen sizes bound the bulk (75 to 80 percent) of the shot in that SAE size; and the average of the two screen sizes would be roughly the nominal shot size. For SAE J444 size numbers of S230 and larger, the ratio of the central two screen sizes is approximately 1.4, whereas below that size the ratio is closer to 1.7. The effect of the larger ratio for the smaller SAE sizes is that shot bulk density would be somewhat higher as a result of the greater variation in actual shot size.

The filler development testing program will test two shot grades, size S330 with the bulk of the shot falling between 0.85 mm and 1.18 mm diameter, and size S230 with the bulk of the shot falling between 0.60 mm and 0.85 mm diameter.

Graded steel shot density is indicated to be in the range of 4,800 kg/m^3 (300 lb/ft^3); mixed grade may run 10 percent higher. Calculated complete fill of the single-cell test fixture (D=223.8 mm, L=4572 mm [D=8.81", L=180"]), deducting for a 15x15 BWR PWR fuel assembly estimated volume of 0.09 m^3 (3 ft^3), is estimated as 667 kg (1470 lb) for the above-mentioned density. Mr. Stevers of Metaltec Steel Abrasive Company stated that the graded steel shot 1994 market price was $0.50/kg ($0.225/lb) plus shipping cost, for 1-ton lots.

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10. REFERENCES

10.1 DOCUMENT REFERENCES

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APPENDIX A

ACRONYMS AND GLOSSARY

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APPENDIX A. ACRONYMS AND GLOSSARY

A.1 ACRONYMS AND ABBREVIATIONS

ACD	Advanced Conceptual Design
ANS	American Nuclear Society
ANSI	American National Standards Institute, Inc.
ASM	American Society for Metals
ASME	American Society of Mechanical Engineers
ASTM	American Society for Testing and Materials
AWS	American Welding Society
BNL	Brookhaven National Laboratory
B&W	B&W Fuel Company
BWR	Boiling Water Reactor
C	Celsius
CDA	Controlled Design Assumptions Document
CDR	Conceptual Design Report
CF	Canistered Fuel
CFR	Code of Federal Regulations
CH	Calico Hills
CHn	Calico Hills Non-welded
COBRA-SFS	Name of a Computer Program
COYOTE	Name of a Computer Program
CRWMS	Civilian Radioactive Waste Management System
DBF	Design Basis Fuel
DCWP	Design Concept Assumptions
DHLW	Defense High-Level Waste
DOE	U.S. Department of Energy
EBDRD	Engineered Barrier Design Requirements Document
EG&G	EG&G, Inc.
EPA	U.S. Environmental Protection Agency
FIDAP	Fluid Dynamics Analysis Program (Computer Program)
FMEA	Failure Modes and Effects Analysis
FY	Fiscal Year
GWd/MTU	Gigawatt-Day per Metric Ton of Uranium
HEATING	Name of a Computer Program
HLW	High-level Waste
HP	Hewlett Packard
I-DEAS	Integrated Design Engineering Analysis Software (Computer Program)
kW	Kilowatt
LANL	Los Alamos National Laboratory
LBL	Lawrence Berkeley Laboratory
LLNL	Lawrence Livermore National Laboratory
LWR	Light Water Reactor
MATPRO	Name of a Computer Program
MCNP	Monte Carlo Neutron and Photon Transport (Computer Program)

MGDS	Mined Geologic Disposal System
MGDS-RD	Mined Geologic Disposal System Requirements Document
M&O	Management and Operating Contractor
MPa	Megapascals
MPC	Multi-Purpose Canister
MTU	Metric Tons of Uranium
MWd	Megawatt Days
NEPA	National Environmental Policy Act of 1969
NLP	Nevada Site Administrative Line Procedure
NRC	U.S. Nuclear Regulatory Commission
OCRWM	Office of Civilian Radioactive Waste Management
ORNL	Oak Ridge National Laboratory
PATRAN	Name of a Computer Program
PNL	Pacific Northwest Laboratory
PTn	Paintbrush Tuff Non-welded
PWR	Pressurized Water Reactor
QA	Quality Assurance
QAP	Quality Administrative Procedure
RDRD	Repository Design Requirements Document
RIP	Repository Integration Program (Computer Program)
RSIC	Radiation Shielding Information Center
RSN	Raytheon Services Nevada
SAE	Society of Automotive Engineers
SCALE	Standardized Computer Analysis for Licensing Evaluations (Computer Program)
SIP	Scientific Investigation Plan
SNF	Spent Nuclear Fuel
SNL	Sandia National Laboratories
SRP	Savannah River Plant
SRS	Savannah River Site
TBD	To Be Determined
TBR	To Be Resolved
TBV	To Be Verified
TOPAZ	Name of a Computer Program
TS	Topopah Spring
TSw	Topopah Spring welded
TSw2	Topopah Spring welded unit 2
TSw3	Topopah Spring welded unit 3 (Vitrophyre tuff)
UCF	Uncanistered Fuel
UCRL	University of California Research Laboratory
UNS	Unified Numbering System for Metals and Alloys
V-TOUGH	Vectorized Transport of Unsaturated Groundwater and Heat (Computer Program)
WP	Waste Package
YMIM	Yucca Mountain Integrating Model
YMP	Yucca Mountain Site Characterization Project
YMSCO	Yucca Mountain Site Characterization Office

A.2 GLOSSARY

This glossary contains the meaning of the specialized terms used in this volume of the report. The references in square brackets at the end of a definition are the highest level document which contains that definition verbatim.

Anticipated processes and events are those natural processes and events that are reasonably likely to occur during the period the intended performance objective must be achieved. To the extent reasonable in the light of the geologic record, it shall be assumed that those processes operating in the geologic setting during the Quaternary Period continue to operate, but with the perturbation caused by the presence of emplaced radioactive waste superimposed thereon. [10 CFR 60.2]

Architecture is that part of the physical system to be built, found, or selected to perform a function subject to its stated requirements.

As low as is reasonably achievable means making every reasonable effort to maintain exposures to radiation as far below the dose limits in 10 CFR 20 as is practical consistent with the purpose for which the licensed activity is undertaken, taking into account the state of technology, the economics of improvements in relation to state of technology, the economics of improvements in relation to benefits to the public health and safety, and other societal and socioeconomic considerations, and in relation to utilization of nuclear energy and licensed materials in the public interest. [10 CFR 20.1003]

Backfill is a material used to fill the space previously created by excavation or drilling, such as in a shaft or borehole.

Barrier is any material or structure that prevents or substantially delays movement of water or radionuclides. [10 CFR 60.2]

Burnup credit is an approach used in criticality control evaluations which accounts for the reduction in criticality potential associated with spent nuclear fuel relative to that of fresh fuel. Burnup credit reflects the net depletion of fissile material and the creation of neutron absorbing isotopes during the fission reaction. Burnup credit is one of the licensing issues which will be addressed in the Topical Reports submitted to the U.S. Nuclear Regulatory Commission.

Canister is a metal receptacle with the following purpose: (1) for solidified high-level radioactive waste, its purpose is a pour mold, and (2) for spent fuel, it may provide structural support for loose rods, nonfuel components, or confinement of radionuclides during preclosure operations.

Cask is a container for shipping or storing spent nuclear fuel and/or high-level waste that meets all applicable regulatory requirements.

Canistered Fuel Disposal Container, CI BBAB00000. The Canistered Fuel Disposal Container Subsystem Element includes all items that form a disposal container for a canistered SNF waste form which is a small CF or a large CF. This subsystem element includes the small CF disposal container component and the large CF disposal container component. The CF disposal container includes but is not limited to multiple containment barriers including multiple closure lids.

Civilian Radioactive Waste Management System is the composite of the sites, and all facilities, systems, equipment, materials, information, activities, and the personnel required to perform those activities necessary to manage radioactive waste disposal.

Cladding is the metal cylinder that surrounds the uranium pellets.

Container is the component of the waste package that is placed around the waste form or the canistered waste form to perform the function of containing radionuclides.

Containment is the confinement of radioactive waste within a designated boundary. [10 CFR 60.2]

Criticality control is the suite of measures taken to maintain nuclear fuel, including spent fuel, in a subcritical condition during storage, transportation and disposal, so that no self-sustaining nuclear chain reaction can occur. Subcriticality is assured by loading spent fuel that meets certain requirements related to fuel age, enrichment, and reduction in nuclear fuel reactivity through burnup.

DHLW Disposal Container, CI BBAC00000. The Defense High-Level Waste (DHLW) Disposal Container Subsystem Element includes all items which form a disposal container for high-level process waste forms packaged in waste canisters originating from Savannah River, Hanford, Idaho National Engineering Laboratory, West Valley, and any other designated locations supplying process waste for disposal. The DHLW disposal container includes but is not limited to multiple containment barriers including multiple closure lids, and internal structure.

Disposal is the isolation of radioactive wastes from the accessible environment. [10 CFR 60.2] Disposal means the emplacement in a repository of high-level radioactive waste, spent nuclear fuel, or other highly radioactive material with no foreseeable intent of recovery, whether or not such emplacement permits the recovery of such waste. [10 CFR 961.11][NWPA Section 2(9)]

Disposal container is a vessel consisting of the barrier materials and internal components designed to meet disposal requirements, into which the uncanistered or canistered waste form will be placed.

Disposal system is any combination of engineered and natural barriers that isolate spent nuclear fuel or radioactive waste after disposal. [40 CFR 191.12(a)]

Drift is a nearly horizontal mine passageway driven on or parallel to the course of a vein or rock stratum or a small crosscut in a mine.

Emplacement Drift Backfill Materials Subsystem Element includes all backfill materials placed in the waste emplacement drifts a an engineered barrier for the purpose of containing and isolating the waste from the accessible environment. Backfill will be used to retard the migration of radionuclides from the waste package to the geologic setting. It may also be placed in peaked layers to provide a barrier which prevents water from contacting the waste package.

Emplacement Drift Invert Subsystem Element consists of the material or inverted arch placed at the bottom of the emplacement drift to provide a floor with a flat surface. The Invert includes the invert materials placed in the waste emplacement drifts as an engineered barrier for the purpose of containing and isolating the waste from the accessible environment. The invert will retard the migration of radionuclides from the waste package to the geologic setting.

Emplacement Drift Openings Subsystem Element includes the space from which rock has been excavated, specifically, where waste is to be emplaced and specifically excluding all other excavated spaces.

Engineered barrier system is the waste packages and the underground facility. [10 CFR 60.2]

Engineered Barrier Segment, CI BB0000000. The Engineered Barrier Segment includes the Waste Package Subsystem and the Underground Facility Subsystem. The major components of the Engineered Barrier Segment shall contribute to the assigned function, Isolate Waste, by containing waste in the waste packages during the prescribed containment period, and then by limiting the release of radionuclides during the post-contaiment period.

The Waste Package Subsystem includes the uncanistered fuel, canistered fuel, and defense high-level waste disposal containers, filler materials, shielding, packing and absorbent materials, and waste package support subsystem elements. The Underground Facility Subsystem includes the emplacement drift openings, emplacement drift backfill materials, and emplacement drift invert subsystem elements.

Excavation extraction ratio is in a horizontal plane through the widest part of repository excavations, the area of excavations divided by the total area.

Filler Materials, CI BBAD00000. The Filler Materials Subsystem Element includes all filler materials used to fill the free space remaining in disposal containers after loading the high-level nuclear waste. Filler materials may used for neutron absorption, moderator displacement, chemical buffering, or radionuclide retardation. The most likely application would be the addition of filler material to selected SNF waste package disposal containers, i.e., UCF, CF, or dual purpose canisters, for the purpose of moderator displacement to aid in criticality control. Filler material may also be added to DHLW waste package disposal containers. Filler materials, if used, will be added to the waste packages disposal containers only at the repository.

Function is a primary statement of purpose; a definition of what a system or subsystem must accomplish to meet the system mission.

Functional analysis is the first step in the Systems Engineering process that defines a baseline of functions and function performance requirements that must be met in order to adequately accomplish the operation, support, test, and production requirements of a system.

Functional interface is the interaction between functions, as in the flow of material or information between a sequence of activities.

Geologic repository is a system which is intended to be used for, or may be used for, the disposal of radioactive wastes in excavated geologic media. A geologic repository includes (1) the geologic repository operations area, and (2) the portion of the geologic setting that provides isolation of the radioactive waste. [10 CFR 60.2]

Geologic repository operations area (GROA) is a high-level radioactive waste facility that is part of a geologic repository, including both surface and subsurface areas, where waste handling activities are conducted. [10 CFR 60.2]

High-level radioactive waste (HLW) means (1) the highly radioactive material resulting from the reprocessing of spent nuclear fuel, including liquid waste produced directly in reprocessing and any solid material derived from such liquid waste that contains fission products in sufficient concentrations; and (2) other highly radioactive material that the Nuclear Regulatory Commission, consistent with existing law, determines by rule requires permanent isolation. The CRWMS will only accept solidified HLW. For the purposes of this document, HLW is vitrified borosilicate glass cast in a stainless steel canister. [NWPA Section 2(12)] [10 CFR 72.3] [10 CFR 960.2] [10 CFR 961.11] [MGDS-RD]

(Items) **Important to Waste Isolation** means the natural and engineered barriers that are relied on for achieving the postclosure performance objectives in 10 CFR 60 Subpart E.

Institutional Barrier System consists of the active and passive institutional controls.

Active institutional controls include (1) controlling access to the MGDS by any means other than passive institutional controls, (2) performing maintenance operations or remedial actions at a site, (3) controlling or cleaning up releases from a site, or (4) Monitoring parameters related to disposal system performance.

Passive institutional controls include (1) permanent markers placed at a disposal site, (2) public records and archives, (3) government ownership and regulations regarding land or resource use, and (4) other means of preserving knowledge about the location, design, and contents of a disposal system. (TBR) [40 CFR 191.02]

Isolation is inhibiting the transport of radioactive material so that amounts and concentrations of this material entering the accessible environment will be kept within prescribed limits. [10 CFR 60.2]

Mission critical refers to those systems, structures, and components (and related activities) whole importance to the successful accomplishment of the CRWMS mission is determined by management to warrant the selected application of QA Program controls.

Multi-purpose canister refers to a sealed, metallic container maintaining multiple spent nuclear fuel assemblies in a dry, inert environment and overpacked separately and uniquely for the various system elements of storage, transportation, and disposal. (See definition of waste form.)

Off-normal are abnormal or unplanned events or conditions that adversely affect, potentially affect, or are indicative of degradation in, the safety, security, environmental or health protection performance or operation of a facility.

Package means the packaging together with its radioactive contents as presented for transport. [10 CFR 71.4]

Packaging means the assembly of components necessary to ensure compliance with the packaging requirements of 10 CFR 71. It may consist of one or more receptacles, absorbent materials, spacing structures, thermal insulation, radiation shielding, and devices for cooling or absorbing mechanical shocks. The vehicle, tie-down system, and auxiliary equipment may be designated as part of the packaging. [10 CFR 71.4]

Packing and Absorbent Materials, CI BBAF00000. The Packing and Absorbent Materials Subsystem Element includes any items or materials immediately surrounding an individual waste container that inhibit the release of radionuclides to the accessible environment.

Performance assessment means any analysis that predicts the behavior of a system or a component of a system under a given set of constant or transient conditions.

Permanent closure is final backfilling of the underground facility and the sealing of shafts and boreholes. [10 CFR 60.2]

Physical system means the CRWMS consisting of the composite of the sites, and all facilities, systems, equipment, materials, information, activities, and the personnel required to perform those activities necessary to manage waste disposal.

Postclosure means the period of time after the permanent closure of the geologic repository.

Preclosure means the period of time before and during the permanent closure of the geologic repository.

Radioactive waste or waste is HLW and other radioactive materials other than HLW that are received for emplacement in a geologic repository. [10 CFR 60.2]

Repository is any system licensed by the Commission that is intended to be used for, or may be used for, the permanent deep geologic disposal of high-level radioactive waste and spent nuclear fuel, whether or not such system is designed to permit the recovery, for a limited period during initial operation, of any materials placed in such system. Such term includes both surface and subsurface areas at which high-level radioactive waste and spent nuclear fuel handling activities are conducted. [NWPA]

Retrieval is the act of intentionally removing radioactive waste from the underground location at which the waste had been previously emplaced for disposal. [10 CFR 60.2]

Shielding, CI BBAE00000. The Shielding Subsystem Element includes any material that provides radiation protection, beyond the limited shielding inherently provided by the disposal container, which will be disposed of as part of the waste package. This configuration item excludes any shielding that is not an integral part of the waste package (i.e., overpacks necessary for transport or for use within containment buildings where waste containers are handled or stored).

Spent nuclear fuel (SNF) is fuel which has been withdrawn from a nuclear reactor following irradiation, the constituent elements of which have not separated by reprocessing. [Specifically in this document, SNF includes (1) intact, non-defective fuel assemblies; (2) failed fuel assemblies in canisters; (3) fuel assemblies in canisters; (4) consolidated fuel rods in canisters; (5) non-fuel assembly hardware inserted in PWR fuel assemblies, including, but not limited to, control rod assemblies, burnable poison assemblies, thimble plug assemblies, neutron source assemblies, instrumentation assemblies; (6) fuel channels attached to boiling water reactor fuel assemblies; and (7) non-fuel assembly hardware and structural parts of assemblies resulting from consolidation in canisters.] [NWPA Section 2(23)][10 CFR 961.11]

Unanticipated processes and events mean those processes and events affecting the geologic setting that are judged not to be reasonably likely to occur during the period the intended performance objective must be achieved, but which are nevertheless sufficiently credible to warrant consideration. Unanticipated processes and events may be either natural processes or events or processes and events initiated by human activities other than those activities licensed under this part. Processes and events may be either natural processes or events or processes and events initiated by human activities other than those activities licensed under this part. Processes and events initiated by human activities may only be found to be sufficiently credible to warrant consideration if it is assumed that (1) the monuments provided by this part are sufficiently permanent to serve their intended purpose; (2) the value to future generations of potential resources within the site can be assessed adequately under the applicable provisions of this part; (3) an understanding of the nature of radioactivity, and an appreciation of its hazards, have been retained in some functioning institutions; (4) institutions are able to assess risk and to take remedial action at a level of social organization and technological competence equivalent to, or superior to, that which was applied in initiating the processes or events concerned; and (5) relevant records are preserved, and remain accessible, for several hundred years after permanent closure. [10 CFR 60.2]

UCF Disposal Container with Basket, CI BBAA00000. The Uncanistered Fuel (UCF) Disposal Container with Basket Subsystem Element is a disposal container containing a fuel basket. The UCF disposal container is employed only at the repository for the disposal of uncanistered (bare) commercial PWR and BWR spent nuclear fuel (SNF) assemblies. Such assemblies would originate from either SNF sent to the repository in bare fuel transportation casks, or the contents of any dual purpose canisters which are determined to be unsuitable for disposal. The UCF disposal container includes but is not limited to multiple containment barriers including multiple closure lids, basket members, optional neutron absorber material, optional thermal shunts, and internal supports for the basket. The containment barriers consist of corrosion-allowance and/or corrosion-resistant materials. Criticality control alternatives include but are not limited to neutron absorber material alloyed with the basket material, addition of neutron absorbing panels or control rods, and/or addition of filler material for moderator displacement to aid in criticality control.

Underground facility is the underground structure, including openings and backfill materials, but excluding shafts, boreholes, and their seals. [10 CFR 60.2]

Underground Facility, CI BBD000000. The Underground Facility Subsystem is that portion of the Engineered Barrier Segment that has been allocated the primary function of limiting radionuclide transport.

The Underground Facility Subsystem includes the following Subsystem Elements: Emplacement Drift Openings, Emplacement Drift Backfill Materials, and Emplacement Drift Invert.

Unrestricted area means any area, access to which is not controlled by the licensee for purposes of protection of individuals from exposure to radiation and radioactive materials, and any area used for residential quarters.

Waste container is a sealed disposal container with the uncanistered or canistered waste form (and possibly filler material) placed therein.

Waste form is the radioactive waste materials and any encapsulating or stabilizing matrix. [10 CFR 60.2] A loaded multi-purpose canister is a canistered waste form. [MGDS-RD]

Waste package means the waste form and any containers, shielding, packing and other absorbent materials immediately surrounding an individual waste container. [10 CFR 60.2]

Waste Package, CI BBA000000. The Waste Package Subsystem includes any waste form containers, shielding, and packing and absorbent materials immediately surrounding an individual disposal container. The multibarrier disposal containers will be used for geologic disposal of high-level radioactive waste forms, limited to intact irradiated reactor fuel assemblies from pressurized water reactors, boiling water reactors, and vitrified glass or other solid process high-level waste forms in canisters. The multibarrier disposal containers will consist of multiple layers of corrosion-allowance and/or corrosion-resistant materials.

The Waste Package Subsystem includes the following Subsystem Elements: UCF Disposal Container and Basket, Canistered Fuel Disposal Container, DHLW Disposal Container, Filler Materials, Shielding, Packing and Absorbent Materials, and Waste Package Support.

Waste Package Support, CI BBAG00000. The Waste Package Support Subsystem Element includes the components necessary to support and stabilize the waste container when emplaced in the repository. These components are those items which (1) are in immediate contact with the emplaced disposal container (or shield, if included), and (2) will remain permanently emplaced in the drift with the waste package. The items in this subsystem include but are not limited to cradles used to support the disposal container/shield and any associated items to restrain movement of the disposal container/shield.

APPENDIX B

ENGINEERED BARRIER SEGMENT/WASTE PACKAGE CONCEPTUAL DESIGNS

BOUND SEPARATELY