71-9322

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16 March 2006
E&L-011-06

Jill Caverly
Project Manager
Spent Fuel Project Office
Office of Nuclear Material Safety and Safeguards, NMSS
U.S. Nuclear Regulatory Commission
Washington, DC 20555-0001

Reference 1: Request for Additional Information dated 21 February 2006(Docket 71-9322, TAC L23931)
Reference 2: Request for Special Package Authorization, E&L-118-05, dated 16 December 2005

Dear Ms. Caverly:

Subject: Request for Special Package Authorization for the LACBWR Reactor Pressure Vessel Package

Duratek respectfully submits the attached responses to your Request for Additional Information (RAI) (Ref. 1) concerning our application for a transportation package for the Dairyland Power Cooperative's La Crosse Boiling Water Reactor Pressure Vessel. Duratek will submit, under a separate letter (E&L-010-06), proprietary information responding to the RAI. Responses to the RAI items required revisions to Chapters 1, 2, 4, 5, and 8 of the application. Revisions to those chapters are attached. With the exception of Chapter 1, the entire chapter has been revised to replace that previously submitted (Ref. 2). For Chapter 1, the text, pages 1-1 through 1-7, have been revised but the drawings, Appendix 1.3, have not changed and have not been resubmitted.

There are several attachments to this letter, listed below:

<u>Attachment 1</u>	Response to Request for Additional Information
<u>Attachment 2</u>	Cover page, rev.1
<u>Attachment 3</u>	Chapter 1, rev.1 (pages 1-1 through 1-7)
<u>Attachment 4</u>	Chapter 2, rev.1
<u>Attachment 5</u>	Chapter 4, rev.1
<u>Attachment 6</u>	Chapter 5, rev.1
<u>Attachment 7</u>	Chapter 8, rev.1

Should you or members of your staff have questions about the response, please contact Mark Whittaker at (803) 758-1898.

Sincerely,

Patrick L. Paquin
General Manager – Engineering & Licensing

Attachments: As stated

NMSS01

Attachment 1
Response to Request for Additional Information

Response to Request for Additional Information dated 21 February 2006

Chapter 1

1-1 State the maximum number of injection ports allowed in the upper assembly of the containment shell.

The description of the containment boundary given in SAR Ch. 4.1.1 states that the upper subassembly of the containment shell will have at least four injection ports to facilitate filling the annulus between the containment shell and the reactor pressure vessel (RPV) with medium density cellular concrete (MDCC). The SAR goes on to state that more injection ports may be added as necessary, but does not state the maximum number of injection ports that may be added. The maximum number of injection ports in the upper assembly of the containment shell is necessary to provide a complete description of the containment boundary.

This information is necessary to satisfy the requirements of 10 CFR Part 71.33(a)(5)(iv).

Note: Correct Table 1-1. In the activity column the total is less than the sum of the individual nuclides (This is assumed to be a typographical error in the total).

Response:
The maximum number of grout injection ports allowed is six. See page 4-3 of the revised SAR. Table 1-1 has been corrected. See page 1-4 of the revised SAR.

Chapter 2

2-1 Clarify the geometric inconsistency on pp. 2-1 of the SAR.

Geometric inconsistency – Text on pp. 2-1 states that the upper shell outer diameter (OD) is 125.5" (inner diameter (ID) = 124") and the lower shell OD is 121.5" (ID=120"). It is unclear whether a gap between the upper and lower shell is intentional.

This information is needed to determine compliance with 10 CFR Part 71.7.

Response:
The dimensions provided on page 2-1 show that there is a ½" gap between the upper and lower shells to allow assembly in the field.

2-2 Demonstrate that the down ending of the vessel will not damage the steel shell.

There are no calculations or rationale demonstrating that down ending of the vessel will not produce damage to the steel shell base material near the trunnions.

This information is needed to determine compliance with 10 CFR Part 71.45.

Response:
As noted in Section 2.5.1 (pg. 2-7), an evaluation of the package for handling (lifting, turning etc.) will be performed and documented per the Duratek QA program to ensure there is no damage to the packaging during its preparation at the site.

2-3 Specify the maximum number of plugs and cover plates that are necessary. Also, provide more detail on how the plugs and cover plates are attached.

On drawing C-068-163041-004 Rev. 0, Note 2 states that the "final number of holes to be determined." This statement is too vague to make a determination of fact regarding safety especially coupled with the lack of detailed information concerning the implementation of the steel plug and cover plate weldment. There is no information that indicates whether the steel plug is welded, loose, or threaded prior to the attachment of the fillet welded cover plates to the outer shell.

This information is needed to determine compliance with 10 CFR Part 71.7 and 10 CFR Part 71.33.

Response:
The maximum number of grout injection ports allowed is six. See page 4-3 of the revised SAR. Drawing C-068-163041-004, ITEM-8A and ITEM-8B, shows that the round stock is welded to the plate with a 5/16" weld to form the grout plug. Drawing C-068-163041-002, Sheet 1, shows that the grout plug is welded to the side of the canister with a 3/8" weld and to the cover plate with a 5/16" weld.

2-4 Clarify the following statement:

On pp. 2-2, it states "For inelastic drop analysis, the acceptance criteria are set in such a way that rupture of the material is prevented." This statement is inconsistent with the analytical results that show rupture, partial or otherwise, does occur.

This information is needed to determine compliance with 10 CFR Part 71.7, 10 CFR Part 71.71(c)(7), and 10 CFR Part 71.73(c)(1).

Response:
The statement has been revised. See page 2-2 of the revised SAR.

2-5 Demonstrate that a single fully integrated element through the thickness of the outer shell is adequate to capture bending effects accurately.

A single element through the thickness of the solid element outer shell subjected to bending will provide spurious results because the tensile and compressive stresses cannot be resolved accurately within the element.

This information is needed to determine compliance with 10 CFR Part 71.7, 10 CFR Part 71.71 (c)(7), 10 CFR Part 71.73(c)(3), and 10 CFR Part 71.71(c)(3).

Response:
A sensitivity analysis was performed that shows a single element is acceptable. The analysis has been added to the proprietary calculation ST-517. The revised calculation will be provided under a separate submittal as a proprietary document.

2-6 Identify where the use of the ASME code proved impractical for the containment boundary.

The test of the SAR (pp. 2-5 for example) indicates that his may be the case, but does not identify what portions of the code are impractical.

This information is needed to determine compliance with 10 CFR Part 71.7.

Response:
Section 2.1.4, pages 2-3&-4, and Section 2.3.1 and 2.3.2, pages 2-5&-6, have been revised to define the use of the ASME code.

2-7 Provide more information on the material properties of LDCC and MDCC including mechanical test information.

The current submittal provides information on de3nsity for these materials only. This is not sufficient for making a determination regarding the safety of this package.

This information is needed to determine compliance with 10 CFR Part 71.7 and 71.33 (5).

Response:
Section 2.2.1, page 2-4 & -5, has been revised to provide the material properties.

2-8 Provide calculations or other methodology demonstrating shock loading related to transport and the associated fatigue effects are negligible.

The characterization of this package as a "fully welded steel structure that does not have any flexible component" and that is "monolithic" in nature, does not preclude it from experiencing shock effects due to such events as coupling-decoupling of rail cars.

This information is needed to determine compliance with 10 CFR Part 71.71(c)(5).

Response:
Section 2.6.5, pages 2-14 and 2-15, has been revised to discuss shock loading.

2-9 Provide basis for using Nelm's equation and relevant supporting materials.

No basis is provided that indicates this analytical approach is appropriate for use in this case. Staff cannot make a determination that the approach is sufficiently broad in scope as to accurately predict the impact and penetration effects of the blunt impact of a 6-inch diameter steel bar.

This information is needed to determine compliance with 10 CFR Part 71.7 and 10 CFR Part 71.73(c).

Response:
Appendix 2-4 has been added to provide the basis for Nelm's equation.

2-10 Provide a basis for using the Ballistics research lab formula and relevant supporting materials.

No basis is provided that indicates that this analytical approach is appropriate for use in this case. Staff cannot make a determination that the approach is sufficiently broad in scope to accurately predict the impact and penetration effects of a 13 lb rod into a concrete backed steel shell.

This information is needed to determine compliance with 10 CFR Part 71.7.

Response:
Appendix 2-5 has been added to provide the basis for the Ballistics research lab formula.

2-11 Justify that hydrogen gas generated in service is less than required to form a flammable mixture and is insufficient to be considered a safety concern. Compute the available free space, the maximum estimated amount of hydrogen that would be available to release into that space during the service period and the amount of available oxidant for combustion of the hydrogen.

It is not clear that the package is made of materials that ensure that there will be no significant chemical, galvanic, or other reaction among the packaging contents or components. This information will assist in ensuring that there is no safety concern related to the formation of an explosive mixture in the package.

This information is needed to determine compliance with 10 CFR Part 71.43(d).

Response:
RADCALC was used to calculate the hydrogen generation in the LACBWR RPVP. These results, which show that the amount is acceptable, were included in revised Section 4.2.3, page 4-4 of the revised SAR. Material compatibility is also discussed in the revised Section 4.2.3.

2-12 Section 2.3.2 Examination - List the exceptions to the ASME B&PV Code.

The application requires the use of ASME B&PV ND-5000 as the examination criteria; however, the applicant states that the requirements will be followed as much as practicable. Details concerning the portions of the code that are not practicable should be explained.

This information is needed for completeness of the review and to meet the requirements of 10 CFR Part 71.7(a).

Response:
Section 2.3.2, page 2-5, has been revised to specify weld examination criteria.

Chapter 4

4-1 Justify that the hydrogen gas generated in service is less than required to form a flammable mixture.

SAR p.4-4 states that the only potential for hydrogen generation is radiolytic decomposition of water in the LDCC in the region of the activated core materials, and that water is removed from the concrete during the curing process. The SAR does not state how much water is removed or how the applicant will ensure that enough water was removed such that an acceptable low amount of hydrogen will be generated. Demonstrate that hydrogen generation rate within the package is not sufficient to result in ignition of the hydrogen, and that the containment boundary will not be compromised.

This information is necessary to demonstrate compliance with 10 CFR Part 71.43(d).

Response:
RADCALC was used to calculate the hydrogen generation in the LACBWR RPVP. These results, which show that the amount is acceptable, were substituted for the previous text on page 4-4 of the revised SAR.

4-2 Justify assumptions stated in SAR Ch. 4.4 used to quantify the potential release under hypothetical accident conditions (HAC), and state whether the assumptions are the limiting case.

SAR Ch. 4.4 states the following assumptions: (1) the dispersible contents mixes with 25% of the LDCC, and (2) 5% of the LDCC is dispersed during HAC; however, no justification is provided to support these assumptions. State whether these assumptions are the limiting case; if they are not, state (within reason) the assumptions leading to the maximum release under HAC.

This information is necessary to demonstrate compliance with 10 CFR Part 71.51(a)(2).

Response:
The SAR was revised to identify the calculation as a bounding calculation with additional justification of the assumptions. See pages 4-7 through 4-9 of the revised SAR.

4-3 Explain the derivation of the activity and A_2 values for potentially dispersible material presented in Section 1.2.2 of the SAR and used to quantify the potential release under hypothetical accident conditions (HAC).

Values for potentially dispersible activity (1.61 Ci) and the associated A_2 value (3.43 Ci) are given in the last paragraph of SAR Section 1.2.2, but data to verify these values are not provided.

This information is necessary to demonstrate compliance with 10 CFR 71.51(a)(2).

Response:
Table 1-2, which shows the derivation of the dispersible source term, was added to the SAR along with explanatory text. See pages 1-4 and 1-5 of the revised SAR.

Chapter 5

5-1 Provide a complete copy of the "LACBWR RPV Activated Materials Report" (Reference 1-1).

This reference contains information needed to confirm values presented in the SAR and to evaluate the methodology used for the activation calculations.

This information is necessary to determine compliance with 10 CFR 71.47.

Response:
A complete copy of the report is provided on the enclosed CD.

5-2 Provide the input file used for the ORIGEN source term calculation, or a representative file if separate calculations were done for separate components (see Question 5-1 above).

The source activity is the key parameter for determining the dose rate at any point outside this package. This is needed for NRC staff to be able to confirm that the source term provide.

This information is necessary to determine compliance with 10 CFR 71.47.

Response:
The ORIGEN files are included in the report provided as stated above.

5-3 Make the assumed activity values in the text and table in Section 5.2.1 the same, or clarify why they are different.

The source activity is the key parameter for determining the dose rate at any point outside the package. This document should be internally consistent in the value chosen for modeling,

This information is necessary to determine compliance with 10 CFR 71.47

Response:
The text in Section 5.2.1 has been revised. See page 5-2 of the revised SAR.

5-4 Describe how the radiation surveys presented in Appendix 5.5.2 were performed. Include the difference between the readings labeled "towards reactor vessel" and "towards thermal shield".

These measurements are used as the basis of the assumed source terms for the shielding evaluations. The method used to collect the data needs to be clear.

This information is necessary to determine compliance with 10 CFR 71.47.

Response:
Section 5.2 has been revised to include a description of the survey method. See page 5-2 of the revised SAR.

Chapter 8

8-1 In SAR Ch. 8.1.2, clarify the examinations that will be performed to inspect the filed weld joining the upper and lower containment shell assemblies.

SAR CH. 8.1.2 describes weld examinations that will be performed to inspect the shop welds and the plug assembly fillet welds. However, it is unclear whether the discussion includes the weld examinations indicated on drawing C-068-153041-002 that will be performed to inspect the weld joining the upper and lower containment shell assemblies.

This information is necessary to demonstrate compliance with 10 CFR Parts 71.85(a) and 71.33.

Response:
Section 8.1.2 was revised to specify that the "field weld", i.e., the weld joining the upper and lower containment shells, will be inspected visually, by MT, and by UT. See page 8-2 of the revised SAR.

Attachment 2

Cover page, rev.1

SAFETY ANALYSIS REPORT

FOR

MODEL LACBWR RPVP

REVISION 1

March 2006

DURATEK, INC.
CORPORATE HEADQUARTERS
140 STONERIDGE DRIVE
COLUMBIA, SOUTH CAROLINA 29210

Attachment 3

Chapter 1, rev.1 (pages 1-1 through 1-7)

1.0 GENERAL INFORMATION

This chapter of the La Crosse Boiling Water Reactor (LACBWR) Reactor Pressure Vessel Package (RPVP) Safety Analysis Report (SAR) presents a general description of the packaging and its contents. This application requests a special package authorization for the shipment by Dairyland Power of the LACBWR RPVP per 10 CFR 71.41(d).

1.1 INTRODUCTION

The (LACBWR) is owned and was operated by Dairyland Power Cooperative (DPC) of La Crosse, Wisconsin.

LACBWR was a nuclear power plant of nominal 50 Mw electrical output, which utilized a forced-circulation, direct-cycle boiling-water reactor as its heat source. The plant is located on the east bank of the Mississippi River in Vernon County, Wisconsin, approximately 1 mile south of the village of Genoa, Wisconsin, and approximately 19 miles south of the city of La Crosse, Wisconsin.

LACBWR achieved initial criticality on July 11, 1967, and the low power testing program was completed by September 1967. In November 1967, the power testing program began. The power testing program culminated in a 28-day power run between August 14 and September 13, 1969.

DPC operated the facility as a base-load plant on its system since November 1, 1969, when the AEC accepted the facility from Allis-Chalmers, until LACBWR was permanently shut down on April 30, 1987. During this time the reactor was critical for a total of 103,287.5 hours.

As part of the decommissioning of LACBWR, the intact reactor vessel will be removed from the reactor building, packaged for transport, and shipped, primarily by rail, to the Barnwell LLW Facility for disposal. The LACBWR RPVP described in this submittal will be transported a single time from its location near Genoa, Wisconsin to the Barnwell LLW Facility.

1.2 PACKAGE DESCRIPTION

1.2.1 Packaging

The LACBWR reactor vessel packaging consists of a steel canister surrounding the reactor pressure vessel, with the annulus between the vessel and the canister filled with medium-density concrete, as shown in the drawings in Appendix 1.3. The canister is formed of a 1.5" steel cylindrical shell with end plates of 4" steel plate. The completed package is 39' 7" long with an outer diameter of 10' 6". The total design weight of the package is 624,500 lbs. All joints in the canister are welded forming the containment boundary and providing a tamper-resistant seal. Shielding is welded to the exterior of the canister at the location of the reactor core. The lower section of the canister has a raised flat ring, on which the eight (8) RPV support legs rest when the RPV is placed inside the lower section of the canister. There are no tie-down devices that are a structural part of the packaging and, at the time of shipment, there are no operable lifting attachments that are a structural part of the packaging. The packaging will be fabricated and assembled in accordance with Duratek's NRC approved Part 71 Quality Assurance program.

1.2.2 Contents

Physical Description

The contents of the LACBWR RPVP are the irradiated reactor pressure vessel and the reactor internals. The reactor vessel consists of a cylindrical shell section with a formed integral hemispherical bottom head and a removable hemispherical top head, which is bolted to a mating flange on the vessel shell. The vessel has an overall height of 37', an inside diameter of 99", and a nominal wall thickness of 4" (including 3/16" of integrally bonded stainless steel cladding). The reactor vessel is ferritic steel (ASTM A-302-Gr-B) plate with integrally bonded Type 304L stainless steel cladding. The reactor internals consist of the following: a thermal shield, a core support skirt, a plenum separator plate, a bottom grid assembly, steam separators, a thermal shock shield, a baffle plate structure with a peripheral lip, a steam dryer with support structure, an emergency core spray tube bundle structure combined with fuel hold-down mechanism, control rods, fuel assembly shrouds, and reactor core support structures. The internals are composed primarily of AISI Type 304 stainless steel with certain components also containing zircalloy, inconel, and boron carbide. The voids in the reactor vessel will be filled with low-density cellular concrete (LDCC) prior to cutting the nozzles and lifting the vessel to remove it from the reactor building. The total weight of the filled vessel is 185 tons. All fuel has been previously removed.

Radionuclide Content

In January 2003, the Waste Policy Institute (WPI) issued their report (Ref. 1-1) of an analytical determination of nuclide activation levels in the LACBWR reactor pressure vessel (RPV), reactor internals, and subcomponents. WPI used a dual approach – manual calculation using a simplified reactor model for isotopic irradiation/decay, and a detailed irradiation analysis using the ORIGEN-ARP 2.00 computer code. The resulting activity was decayed from the time of shutdown (April 1987) to January 2003, the assumed time of shipment for the WPI report. The manual calculation results were nearly a factor of two higher than the ORIGEN-ARP results, 15631 vs 8130 total curies. In the activation calculation, the upper bound material percentage for niobium was used from NUREG/CR-6567 (Ref. 2). The listed range for niobium in stainless steel (the predominate material in the reactor internals) is 5-300 ppm, so 300 ppm was used in the activation calculation. Thus, the results for Nb-94 are extremely conservative.

A surface coating evaluation, based on removable contamination samples from the Shutdown Condenser, was performed for the internal surfaces of the RPV and internals. The measured activity per unit area was distributed over the area of the vessel and internals, 1047.47 m^2, to determine the total activity of surface contamination in the vessel. This activity is a small fraction of the total activity.

Three fuel assembly designs (Type I, Type II, and Type III) were used in the LACBWR reactor (Ref. 3). All assemblies were stainless steel clad. Visible fuel rod clad failures were evident in many spent Type I and Type II fuel assemblies. There has been no evidence of any fuel rod clad failures in Type III fuel. During refueling in 1977 and in 1979, after the grossly failed fuel assemblies were moved from the reactor to the spent fuel pool (SFP), several pieces of fuel rod and fuel debris were recovered from the tops of other fuel assemblies and control rods in the reactor and placed in the SFP. During 1977, a significant fraction of the reactor internals, including other fuel assemblies, tops of control rods, below the core, unfueled positions, steam separator down-comer region, etc, was examined and searched for identifiable fuel debris. Very little other than a few small pieces of fuel clad was found, and all were recovered and placed in the SFP. Cladding failures decreased after Cycle 5 (Mar. 1978- Mar. 1979).

After detailed examination of the failed fuel rods, an estimate of the amount of uranium displaced from the failed rods was made. After including the collected debris and the uranium identified in waste shipments sent offsite for disposal, a residual of 58.2 grams of uranium remains (Ref. 4). It is assumed that this material is distributed throughout the primary system and has plated out on the reactor vessel, internals, primary system piping, and other primary system components outside the vessel. The calculated TRU generated from this distributed uranium is shown below (decayed to January 2003), based on the burnup and power levels characteristic of the LACBWR reactor. The activity per unit area conservatively assumes 100% of the activity is distributed only on the reactor vessel and internals. This gives a TRU activity per unit area approximately 100 times the measured contamination values included in the WPI characterization.

Table 1-1
Activity From Residual Uranium

Radionuclide	Grams	Decayed Ci	μCi/cm^2
U-234	1.76E-02	1.09E-04	1.04E-05
U-235	1.13E+00	2.49E-06	2.38E-07
U-236	1.98E-01	1.29E-05	1.23E-06
U-238	5.69E+01	1.93E-05	1.85E-06
Np-237	7.82E-03	5.55E-06	5.30E-07
Pu-238	1.81E-03	2.72E-02	2.59E-03
Pu-239	2.66E-01	1.65E-02	1.57E-03
Pu-240	7.15E-02	1.64E-02	1.57E-03
Pu-241	3.64E-02	1.71E+00	1.63E-01
Pu-242	6.84E-03	2.67E-05	2.54E-06
Am-241	1.26E-03	4.19E-03	4.00E-04
Am-243	9.25E-04	1.85E-04	1.76E-05
Cm-242	3.01E-04	2.52E-11	2.40E-12
Cm-244	1.46E-04	6.48E-03	6.19E-04
Total	5.86E+01	1.78E+00	1.70E-01

The WPI results were updated by substituting the conservative estimate (Table 1-1) of the contamination levels from uranium and TRU that could be present due to the fuel failures that occurred during operation. The resulting activity due to contamination and potentially dispersible is shown in Table 1-2.

Table 1-2

Contamination on LACBWR RPV and Internals (decayed to June 2007)

Radionuclides	WPI Table 10 (Ci)	Residual Uranium (Ci)	Assumed Contamination (Ci)	Decayed (Ci)	Fraction A_2
H-3	1.07E-05		1.07E-05	8.35E-06	7.72E-09
C-14	1.82E-05		1.82E-05	1.82E-05	2.24E-07
Fe-55	4.37E-02		4.37E-02	1.40E-02	1.29E-05
Co-57	5.22E-06		5.22E-06	8.56E-08	3.17E-10
Co-60	1.36E-01		1.36E-01	7.61E-02	7.04E-03
Ni-59	6.87E-04		6.87E-04	6.87E-04	0.00E+00
Ni-63	7.27E-02		7.27E-02	7.05E-02	8.70E-05
Sr-90	8.10E-05		8.10E-05	7.29E-05	8.99E-06
Cs-137	3.28E-04		3.28E-04	2.96E-04	1.83E-05
U-233/234	2.51E-07	1.09E-04	1.09E-04	1.09E-04	6.71E-04
U-235		2.49E-06	2.49E-06	2.49E-06	0.00E+00
U-238	1.11E-07	1.93E-05	1.93E-05	1.93E-05	0.00E+00
Pu-238	1.90E-04	2.72E-02	2.72E-02	2.62E-02	9.71E-01
Pu-239/240	1.23E-04	3.29E-02	3.29E-02	3.29E-02	1.22E+00
Pu-241	4.68E-03	1.71E+00	1.71E+00	1.38E+00	8.53E-01
Cm-242	2.96E-08	2.52E-11	2.52E-11	2.63E-14	9.74E-14
Cm-243/244	7.16E-05	6.48E-03	6.48E-03	5.82E-03	2.15E-01
Pu-242		2.67E-05	2.67E-05	2.67E-05	9.86E-04
Am-241	4.58E-04	4.19E-03	4.19E-03	4.16E-03	1.54E-01
Am-243		1.85E-04	1.85E-04	1.85E-04	6.83E-03
Total				1.61E+00	3.43E+00

Of the total activity in the vessel only 1.61 curies, 3.43 A_2, is from contamination and is potentially dispersible. The rest of the activity is in the activated metal components.

Finally, the activation activities were decayed to the expected date of shipment, i.e., June 1, 2007 and combined with the activities from Table 1-2. The resulting total activity is shown in Table 1-3.

Table 1-3

Total Activity

Radionuclide	Ci	TBq	A_2	Number of A_2s
H-3	8.35E-06	3.09E-07	40	7.72E-09
C-14	1.28E+01	4.73E-01	3	1.58E-01
Fe-55	9.20E+02	3.40E+01	40	8.51E-01
Co-57	8.56E-08	3.17E-09	10	3.17E-10
Co-60	4.32E+03	1.60E+02	0.4	4.00E+02
Ni-59	5.10E+01	1.89E+00	Unlimited	0.00E+00
Ni-63	4.81E+03	1.78E+02	30	5.94E+00
Sr-90	7.29E-05	2.70E-06	0.3	8.99E-06
Nb-94	5.60E-01	2.07E-02	0.7	2.96E-02
Cs-137	2.96E-04	1.10E-05	0.6	1.83E-05
U-233/234	1.09E-04	4.03E-06	0.006	6.71E-04
U-235	2.49E-06	9.21E-08	Unlimited	0.00E+00
U-238	1.93E-05	7.15E-07	Unlimited	0.00E+00
Pu-238	2.62E-02	9.71E-04	0.001	9.71E-01
Pu-239/240	3.29E-02	1.22E-03	0.001	1.22E+00
Pu-241	1.38E+00	5.12E-02	0.06	8.53E-01
Cm-242	2.63E-14	9.74E-16	0.01	9.74E-14
Cm-243	5.82E-03	2.15E-04	0.001	2.15E-01
Pu-242	2.67E-05	9.86E-07	0.001	9.86E-04
Am-241	4.16E-03	1.54E-04	0.001	1.54E-01
Am-243	1.85E-04	6.83E-06	0.001	6.83E-03
Total	1.01E+04	3.75E+02		4.10E+02

The total quantity of fissile material is 1.7 g, which qualifies as "fissile exempt" material. The total decay heat is less than 70 watts.

1.3 APPENDIX

1.3.1 Drawings

C-068-163041-002 "RPV Canister Assembly"

C-068-163041-003 "Lower Shell Assembly"

C-068-163041-004 "Upper Shell Assembly"

1.3.2 References

1-1. LACBWR RPV Activated Materials Report, DPC.0101.02.01, Jan. 2003

1-2 Low-Level Radioactive Waste Classification, and Assessment: Waste Streams and Neutron Activated Metals, NUREG/CR-6567, August 2000

1-3 Response to NRC re: Accounting at Reactors and Wet Spent Fuel Storage Facilities, LAC-13866, March 2005

1-4 Summary Report – Research of Material Displaced from LACBWR Spent Fuel Assemblies, LAC-TR-141, Feb. 2003

Attachment 4

Chapter 2, rev.1

2.0 STRUCTURAL EVALUATION

This chapter presents the structural evaluation of the LACBWR RPV package. The evaluations are performed in accordance with the requirements of 10CFR71 (Reference 2-1) for an exclusive use Type B package. Tables and Figures cited in the text are found in the Appendix.

2.1 DESCRIPTION OF STRUCTURAL DESIGN

2.1.1 Discussion

The LACBWR RPV package (henceforth referred to as the "package" in this SAR) consists of a fully welded canister, fabricated in two parts and field welded together, and the grouted RPV. The package is cylindrical in shape and has a maximum diameter of approximately 11′. The overall length of the package, excluding the remnant of the lifting attachment, is approximately 39′ 7″. The upper part of the canister is made of a 1½″ thick shell, having an outside diameter of 125½″, and a 4″ thick endplate. The lower part of the canister is made of a 1½″ thick shell, having an outside diameter of 121½″, and a 4″ thick endplate. To provide a surface for welding the upper and lower parts of the canister, the lower part of the canister is fitted with a ring that is 3″ thick and has an outside diameter of 131″.

Prior to placement in the canister, the RPV (Reference 2-2), with some of its internal components, as discussed in Chapter 1, is filled with low-density cellular concrete (LDCC). The interstitial space between the canister and the RPV is filled with the medium density cellular concrete (MDCC). Thus, the content of the package is in the form of a monolith that tightly fits inside the canister. The structure of the canister forms the containment boundary of the package.

Supplemental shielding plates are welded to the canister at the location of the core-region of the RPV. These plates do not form the containment boundary but need to remain attached to the canister during the normal operating conditions to meet the shielding requirements of 10CFR71. However, these plates are not needed to meet the dose rate requirements under hypothetical accident conditions. Please see Chapter 5 for the detailed evaluation of the shielding requirements.

Trunnions and other lifting and handling attachments may be welded to the canister to facilitate package handling during the preparation of the package. Any such attachments will be disabled or removed before the shipment. The fill holes provided in the upper shell of the canister and the upper endplate, as discussed in Section 1.2.1 and 7.1.2 are plugged and welded closed. Care was taken to ensure that the attachments, and the fill holes are not located in the region where the package is postulated to be dropped during the normal operating conditions.

The weight of the package components and contents, as well as the package center of gravity is discussed in Section 2.1.3. The fabrication of the package will be in accordance with fabrication specification satisfying the design requirements described in this SAR. Chapter 8 addresses the inspections and examinations that will be performed on the package for compliance with applicable design and regulatory requirements.

2.1.2 Design Criteria

The package is designed to satisfy the requirements of 10CFR71.71 under the normal conditions of transport (NCT) and hypothetical accident conditions (HAC). Compliance with the "General Standards for All Packages" specified in 10 CFR 71.43 and the "Lifting and Tie-Down Standards" specified in 10 CFR 71.45 are discussed in Section 2.4 and 2.5 respectively.

The allowable stresses in the package containment boundary are based on the criteria of Regulatory Guide 7.6 (Reference 2-3). The allowable stresses under normal conditions are:

Primary membrane stresses < S_m

Primary membrane + bending stresses < 1.5 S_m

Where, S_m = Design stress intensity

Based on ASME Code (Reference 2-4), Section III, ND-3000 the design stress intensity is defined to be:

S_m = Lesser of (1 Sy and Su/3.5)

Where, S_y = Material yield stress

S_u = Material ultimate strength

The containment boundary of the LACBWR RPV package is made of ASTM A-516 Gr. 70 material (see Section 2.2.1), for which S_y = 38,000 psi and S_u = 70,000 psi. Therefore,

S_m = 20,000 psi

For inelastic drop analyses, the package evaluation is based on the determination of the material rupture. Since the free drop analyses for the LACBWR RPV package are performed using nonlinear finite element analysis techniques, where the accumulated plastic strains can be calculated, the failure criteria is established based on the material ductility. For the package to remain intact, the total accumulated tensile strains are limited to the minimum elongation in 2″ specimen of the rupture test. The minimum specified elongation for ASTM A-516 Gr. 70 is 21%. The accumulated plastic strain is limited to this value.

The acceptance criterion for prevention of buckling is set such that a minimum safety factor of 3 is achieved on the critical buckling stress under normal loading conditions. The acceptance criteria for prevention of brittle fracture are based on Regulatory Guide 7.11 (Reference 2-5) and its source document NUREG/CR-1815 (Reference 2-6).

2.1.3 Weights and Centers of Gravity

The weight of the various components of the LACBWR RPV package has been evaluated in Reference 2-7. They are summarized here as follows:

Weight of the RPV + LDCC	=	380,000	lb
Canister	=	130,000	lb
Interstitial MDCC	=	99,000	lb
1¼" Supplemental Shield Plates	=	15,000	lb
1¾" Supplemental Shield Plates	=	15,000	lb
Total Package Mass	=	639,000	lb

The C.G. of the package is estimated to be at a distance of 235" from the top surface of the upper endplate.

2.1.4 Identification of Codes and Standards for Package Design

Based on the contents form and amount of radioactivity (normal form, radioactive contents between 3000A2 and 300A2 and not greater than 30,000 Ci), the LACBWR RPV package is categorized as Type-B, Category II package (Reference 2-5). Based on the recommendations of Reference 2-8 the fabrication, examination, and inspection of the containment boundary components of a Type II package should be per ASME B&PV Code Section III, Subsection ND. Part of the ASME Code, however, is not applicable to the design of radioactive material packages. Regulatory Guide 7.6 (Reference 2-3) has incorporated the applicable portion of the code and was used for the design of the LACBWR package.

All the welds on the containment boundary are full penetration welds that meet the ASME code configuration, except the field-weld between the upper and lower portions of the canister, which according to the ASME code needs to be either a double bevel full-penetration weld or a single bevel full penetration weld with backing plate. Because of the geometry of the package both these welds are impractical to make. At this location, a single bevel weld, without a backing plate, has been specified. Based on low stresses in this weld under the internal and external pressure loading, use of this weld configuration is justified.

2.2 MATERIALS

2.2.1 Material Properties and Specifications

RPV

Specification: ASTM A-302, No Grade Specified (Reference 2-2), Assume Grade A
Minimum Yield Strength, S_y = 45,000 psi
Minimum Ultimate Strength, S_u = 75,000 psi
Minimum Elongation, in 2″ specimen, e = 15%

Canister

Specification: ASTM A-516 Gr. 70
Minimum Yield Strength, S_y = 38,000 psi
Minimum Ultimate Strength, S_u = 70,000 psi
Minimum Elongation, in 2″ specimen, e = 21%

Welds

Rod Specification: E-70xx Electrodes
Minimum Ultimate Strength, S_u = 70,000 psi

Concrete

The low-density cellular concrete (LDCC), used to fill the RPV cavity, and the medium density cellular concrete (MDCC), used to fill the interstitial space between the Canister and the RPV, are comprised of Portland cement mixture meeting the guidelines of the specific sections of

ACI 523.3R (Reference 2-9). The mixture shall consist of Portland cement meeting ASTM C 150 standard, small aggregate, and suitable binders to yield the desired flow ability. The mass density of the hardened mixture, as obtained per ASTM C109, shall be 120±10 lb/ft^3 for MDCC and 50±5 lb/ft^3 for LDCC.

2.2.2 Chemical, Galvanic, or Other Reactions

The materials from which the package is fabricated (carbon steel, LDCC and MDCC) along with the contents (the carbon steel RPV) will not cause significant chemical, galvanic or other reaction in air, nitrogen or water atmosphere.

2.2.3 Effects of Radiation on Materials

The materials from which the package is fabricated (carbon steel, LDCC and MDCC) along with the contents (the carbon steel RPV), exhibit no significant degradation of their mechanical properties under the radiation field produced by the RPV.

2.3 FABRICATION AND EXAMINATION

2.3.1 Fabrication

For a Type-B, Category II package Reference 2-8 recommends using ASME B&PV Code, Section III, Subsection ND, as the fabrication criteria. NUREG/CR-3854 (Reference 2-8) has incorporated the portion of the ASME Code applicable to the fabrication of radioactive material packages and has been used for the LACBWR package fabrication criteria.

2.3.2 Examination

For a Type-B, Category II package Reference 2-8 recommends using ASME B&PV ND-5000 as the examination criteria. NUREG/CR-3019 (Reference 2-10) has incorporated the portion of the ASME Code applicable to the examination of radioactive material packages and has been used for the examination criteria. The details of the weld examination for the LACBWR package are provided in Section 8 of this SAR.

2.4 GENERAL REQUIREMENTS FOR ALL PACKAGES

10 CFR 71.43 establishes the general standards for packages. This section identifies these standards and provides the bases that demonstrate compliance.

2.4.1 Minimum Package Size

10 CFR 71.43(a) requires that:

"The smallest overall dimension of a package must not be less than 10 cm (4")."

The smallest overall dimension of the package is the diameter of the lower part of the canister (121.5"), which is larger than 4". Therefore, the minimum package size requirement is satisfied.

2.4.2 Tamper-Indicating Feature

10 CFR 71.43(b) requires that:

"The outside of a package must incorporate a feature, such as a seal, which is not readily breakable, and which, while intact, would be evidence that the package has not been opened by unauthorized persons."

The outside of the package is a totally welded structure. Therefore, the requirement of the tamper-proof feature is satisfied.

2.4.3 Positive Closure

10 CFR 71.43(c) requires that:

"Each package must include a containment system securely closed by a positive fastening device that cannot be opened unintentionally or by a pressure that may arise within the package,"

The package is a totally welded structure. It is subjected to a very low design pressure (MNOP = 7.5 psi). It has been shown in Section 2.6.1.3 that the canister structure is capable of sustaining this pressure with a large margin of safety. Therefore, the requirement of positive closure is satisfied.

2.5 LIFTING AND TIE-DOWN STANDARDS FOR ALL PACKAGES

10 CFR 71.45 specifies the requirements for the lifting and tie-down devices that are "structural parts of the package". The lifting and tie-down devices for the package are designed such that they are "not structural part of the package". Therefore, their design is not a part of the package safety analysis for Part 71 considerations, and the criteria of 10 CFR 71.45 do not apply.

2.5.1 Lifting Devices

Trunnions and lifting attachments may be welded to the canister to facilitate the handling of the package during its preparation for the shipment. These devices must be disabled or removed prior to its shipment of the package. The evaluation of these devices under the site-applicable standards must be performed to ensure that the temporary use of these devices may not impair the package to meet the requirements of this SAR.

2.5.2 Tie-Down Devices

There are no tie-down devices that are "integral part of the package". Therefore, the criteria of 10 CFR 71.45 do not apply.

2.6 NORMAL CONDITIONS OF TRANSPORT

This Chapter demonstrates that the package is structurally adequate to meet the performance requirements of Subpart E of 10 CFR 71 when subjected to NCT as defined in 10 CFR 71.71. Compliance with these requirements is demonstrated by analyses in lieu of testing as allowed by 10 CFR 71.41(a) and Regulatory Guide 7.6 (Reference 2-3).

2.6.1 Heat

The LACBWR RPV package has been analyzed for the hot environment (ambient temperature 100°F) with and without solar insolation using a 1-dimensional analytical model. The details of these analyses are presented in Chapter 3 of the SAR. Total solar insolation of 400 g cal/cm^2 on the horizontal curved surface for 12-hour duration has been used. The internal heat load of the package (70 Watt) is also included in the analysis. The maximum normal operating pressure (MNOP) is established based on the maximum cavity temperature obtained from these analyses.

2.6.1.1 Summary of Pressures and Temperatures

Based on the analyses performed in Chapter 3, the maximum temperature of various component of the package is summarized below (Reference Table 3-2).

Canister temperature..153.7°F
Interstitial concrete temperature .... 154°F
RPV Temperature.. 154.28°F
Temperature gradient through the package wall .... 0.3°F
The design temperature of the package is established to be160°F and the MNOP is 7.5 psig.

2.6.1.2 Differential Thermal Expansion

The canister of the LACBWR RPV package is a welded structure that does not have any thermal insulation and dissimilar metal joints. Under the thermal test conditions the entire canister will rise in temperature to approximately 154°F, with very little (less than 1°F) temperature gradient through its wall. Therefore, under these tests the entire canister will expand uniformly, with little or no thermal stresses.

2.6.1.3 Stress Calculations

The stresses in the package under NCT are mainly due to the internal pressure. As mentioned in the previous section, negligible amount of thermal stresses would result under the thermal tests. The canister of the package is a single-layered steel structure fabricated in two pieces that are welded together with no force-fits and an appropriate amount of pre-heat. Therefore, no appreciable fabrications stresses will be present in the package.

Stresses in the package are calculated under the MNOP as follows:

Stresses in the Wall

Under the design internal pressure the canister will be subjected to hoop and longitudinal stresses in the wall. These stresses can be calculated using the formulas from Roark (Reference 2-11), Table 29, Case 1c.

Under the design pressure, $p = 7.5$ psig

$$\sigma_2 = \frac{pr}{t} = \frac{7.5\times 62.75}{1.5} = 313.8 \text{ psi}$$

$$\sigma_1 = \frac{pr}{2t} = \frac{7.5\times 62.75}{2\times 1.5} = 156.9 \text{ psi}$$

Where,

r = internal radius, for conservativeness external radius of 62.75″ is used

t = thickness of the shell = 1.5″

Because of the discontinuity at the joint where the upper and lower parts of the canister are welded, the stress will be intensified. This joint is similar to a socket -welded joint for which a stress concentration factor of 3 is normally used. To be conservative a stress concentration factor of 3.5 is used in this calculation. Therefore, the maximum stresses are as follows:

$$\sigma_{hoop} = 3.5\times\sigma_2 = 3.5\times 313.8 = 1{,}098 \text{ psi}$$

$$\sigma_{long} = 3.5\times\sigma_1 = 3.5\times 156.9 = 549 \text{ psi}$$

Stresses in the Endplates

Both the top and the bottom endplates of the canister are 4" thick circular plates. The top endplate is also welded with the lifting arrangement, part of which will remain attached to it even when the lifting attachment is rendered ineffective prior to the shipment of the package. Thus, this end will be much stiffer than the lower end, which is analyzed for the maximum stress under the design pressure. The maximum stress in this plate can be calculated by idealizing it as a circular plate, with the simply-supported edge, and uniformly loaded over its surface. Using the formula from Roark (Reference 2-11), Table 24, Case 10, we get:

$$\sigma_{max} = 0.375 \times (3 + \nu) \times q \times (a/t)^2$$

Where,

ν = Poisson's Ratio = 0.3 for steel

q = uniform pressure = 7.5 psi

a = radius of the plate = 62.75″

t = thickness of the plate = 4″

Thus,

$$\sigma_{max} = 0.375 \times (3 + 0.3) \times 7.5 \times (62.75/4)^2 = 2{,}284 \text{ psi}$$

It should be noted that the ASME B&PV Code classifies the stress at the juncture of the endplates and the shell as a secondary stress.

2.6.1.4 Comparison with Allowable Stresses

From the analyses presented in the previous section, the maximum stress in the package under the normal operating conditions is 2,284 psi. Since this is a bending stress, based on the ASME code, it is classified as a primary membrane + bending stress. Conservatively considering it to be primary membrane stress, the allowable stress is 20,000 psi (see Section 2.1.2). Therefore, the factor of safety under the normal operating conditions is:

F.S. = 20,000/2,284 = 8.76

2.6.2 Cold

The LACBWR RPV package has been analyzed for the cold environment (ambient temperature -20°F) with the internal heat load of 70 Watt using a 1-dimensional analytical model. The details of these analyses are presented in Chapter 3 of the SAR. Based on the analyses performed, the maximum temperature of various component of the package is summarized below (Reference Table 3-2).

Canister temperature...-18.55°F

Interstitial concrete temperature .... -18.26°F

RPV Temperature .... -17.97°F

Temperature gradient through the package wall .... 0.29°F

The canister of the LACBWR RPV package is a welded structure that does not have any thermal insulation and dissimilar metal joints. Under the cold test conditions the entire canister will drop in temperature to approximately -19°F, with very little (less than 1°F) temperature gradient through its wall. Therefore, under this test the entire canister will contract uniformly, with little or no thermal stresses.

Although the LACBWR RPV package has been evaluated for the regulatory cold condition requirement of -20°F ambient temperature, the minimum temperature the package can be transported has been set to 0°F. The fracture toughness requirements for various parts of the containment boundary (i.e. the canister) are established based on 0°F lowest service temperature (LST). Provisions of Regulatory Guide 7.11 (Reference 2-5) and NUREG/CR-1815 (Reference 2-6) are used in determination of the nil ductility transition (NDT) temperature for the package material. The ASME Code Section VIII – Division 2, is used to establish NDT test exclusion criteria.

The LACBWR RPV package is a Type B, Category II package. Therefore, the required NDT temperature is determined by using a value of β=0.6 in accordance with the methodology provided in Section 5.2 of NUREG/CR-1815. The NDT temperature for a particular thickness of plate is determined from the following equation.

$T_{NDT} = LST - A$

Where A is the temperature offset obtained from Figure 6 of Reference 2-6 (Provided in Appendix 2-2 of this SAR).

For the 4″ thick endplates, the value of A from Figure 6 is 15°F. Therefore,

T_{NDT} (4″) = 0 – 15 = -15°F

For the 3″ thick welding ring, the value of A from Figure 6 is 0°F. Therefore,

T_{NDT} (3″) = 0 – 0 = 0°F

For the 1½″ shell, the impact test exclusion criteria of the ASME Code exemption criterion is used. Figure AM-218.1 of Reference 2-4 (Provided in Appendix 2-3 of this SAR) gives a set of curves for different materials that specify the LST above which the material is exempted from impact test. For 1½″ thick plate made of A-516 material that has been normalized and conforms to the fine grain practice, the LST is -13°F. Therefore, for 0°F LST, no impact test is required for 1½″ thick plates.

2.6.3 Reduced External Pressure

The reduced external pressure test, specified in 10 CFR 71.71(c)(3), is required to be performed under an external pressure of 3.5 psia. Under this pressure condition the design pressure will result in an internal pressure of 7.5 + 14.7 – 3.5 = 18.7 psig. The stresses, calculated in Section 2.6.1.3, for 7.5 psig may be linearly ratioed to obtain the stresses in the package under the reduced external pressure. Thus, the stresses in the canister under reduced external pressure are:

Maximum stress in the shell = 1,098×18.7/7.5 = 2,738 psi

Maximum stress in the endplates = 2,284×18.7/7.5 = 5,695 psi

Since these are bending stresses, based on the ASME code, they are classified as a primary membrane + bending stresses. Conservatively considering it to be primary membrane stress, the allowable stress is 20,000 psi (see Section 2.1.2). Therefore, the factor of safety under the reduced external pressure loading is:

F.S. = 20,000/5,695 = 3.51

2.6.4 Increased External Pressure

The increased external pressure test, specified in 10 CFR 71.71(c)(4), is required to be performed under an external pressure of 20 psia. Assuming zero internal pressure, the canister will be subjected to an external pressure of 20 psig. The stresses, calculated in Section 2.6.1.3, for 7.5 psig may be linearly ratioed to obtain the stresses in the package under the increased external pressure. Thus, the stresses in the canister under increased external pressure are:

Maximum stress in the shell = 1,098×20/7.5 = 2,928 psi

Maximum stress in the endplates = 2,284×20/7.5 = 6,091 psi

Since these are bending stresses, based on the ASME code, they are classified as a primary membrane + bending stresses. Conservatively considering it to be primary membrane stress, the allowable stress is 20,000 psi (see Section 2.1.2). Therefore, the factor of safety under the increased external pressure loading is:

F.S. = 20,000/6,091 = 3.28

Under the increased external pressure loading, the canister will be subjected to a compressive loading. A closed end cylindrical shell may be susceptible to buckling under this loading condition. The buckling stress of the LACBWR RPV canister is calculated from the formulas of Reference 2-11, Table 35, Case 20.

For, l = length of the cylinder = 480″

r = radius of the cylinder = 62.75″

t = wall thickness = 1.5″

$$\left(\frac{l}{r}\right)^2\left(\frac{r}{t}\right)=\left(\frac{480}{62.75}\right)^2\left(\frac{62.75}{1.5}\right)=2{,}448 \gg 300$$

The critical stress is given by the formula:

For, E = modulus of elasticity for steel = 30×10^6 psi

$$q'=\frac{0.92E}{\left(\frac{l}{r}\right)\left(\frac{r}{t}\right)^{2.5}}=\frac{0.92\times30\times10^6}{\left(\frac{480}{62.75}\right)\left(\frac{62.75}{1.5}\right)^{2.5}}=318.8\text{ psi}$$

The buckling stress is:

$$q_{buckling} = 0.8\times q' = 255\text{ psi}$$

Therefore, the factor of safety against the buckling is:

F.S. = 318.8/20 = 15.9 > 3.0

It should be noted that the buckling stress calculated here is very conservative because of the following reasons.

1. The formula used is for the unsupported shell whereas the canister is filled with the grouted RPV which supports the canister wall under compressive loading.

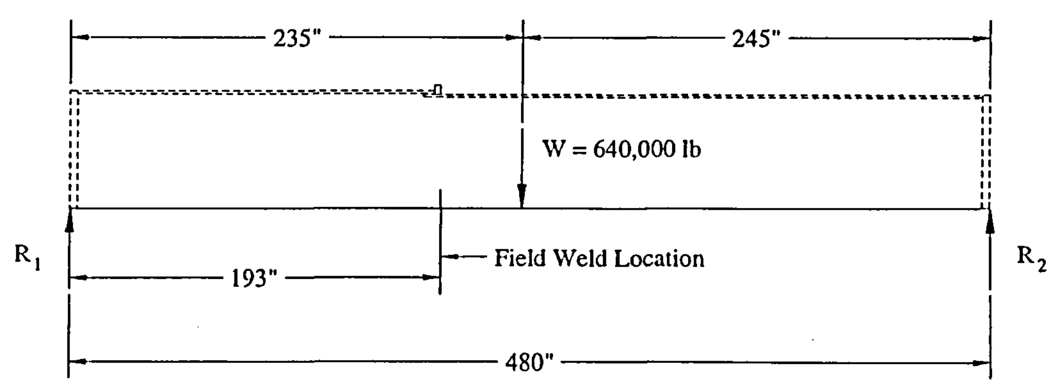
2. The total length of the canister is used for the unsupported length, whereas for the canister, the discontinuity in the upper and lower part of the canister will reduce the effective length of the shell. Therefore, the buckling stress will be higher than that calculated from the above formula.

2.6.5 Vibration

The package will be transported by the railcar on a bolster. It will be placed on a pair of saddles and tied to the bolster with the help of steel cables. Tightly-shimming it to the bolster will provide the longitudinal restraint. Thus, no part of the package will be used as a tie down component that could be subjected to severe shock and vibration loading. The coupling-decoupling of the railcar will subject the package shell to compressive loading due to longitudinal acceleration, and bending due to lateral and vertical acceleration. The only component of the package that could be vulnerable to fatigue is the field weld at the juncture of the upper and lower part of the canister. A conservative evaluation of the package is provided to demonstrate that the shock and vibration associated with the transportation of the package is of no concern for the LACBWR package.

The LACBWR package will be transported on a special train for which the particular railroad company that carries it approves the loading. For the evaluation purpose use the loading specified by the AAR for unlimited interchange packages. These packages need to meet 3g longitudinal, 2g lateral, and 2g vertical loading. Based on the shipping configuration, the vertical loading of 2g could subject the LACBWR package to shock loading. Assume further that the saddle supports are located at the extreme ends of the package. The maximum package weight is 640,000 lb and its C.G. is located at 235" from the top of the canister (see Section 2.1.3).

The maximum stress in the field weld is evaluated with the idealization as shown below:



Package Weight, W = 640,000 lb

$R_1 = 640{,}000 \times 245/480 = 326{,}667$ lb

$R_2 = 640{,}000 \times 235/480 = 313{,}333$ lb

Moment at the field weld location under 2g loading,

$M = 2 \times 326{,}667 \times 193 = 1.261 \times 10^8$ in-lb

The upper shell outside diameter is 125.5" and inside diameter is 122.5". Therefore the moment of inertia of the cross-section about the bending axis is:

$$I = \pi/64\times(125.5^4 - 122.5^4) = 1.1233\times10^6 \text{ in}^4$$

Distance of the extreme fiber,

$$c = 125.5/2 = 62.75 \text{ in}$$

Nominal bending stress,

$$\sigma_b = 1.261\times10^8\times62.75/1.1233\times10^6 = 7{,}044 \text{ psi}$$

Using a stress concentration factor, $K_t = 3.5$ for the geometric discontinuity, the maximum alternating stress is:

$$\sigma_a = 3.5\times7{,}044 = 24{,}654 \text{ psi}$$

From ASME Boiler & Pressure Vessel Code, Section III, Appendix I, Table I-9.1 for materials with ultimate tensile stress less than 80,000 psi, subjected to 24,654 psi alternating stress loading, the allowable number of cycles is 40,000.

The loading considered in this analysis arises due to coupling/decoupling of the railcars and will occur only a few times during a 1,500 miles travel. Conservatively assuming that such a loading occurs once every mile of travel; the number of allowable cycles (40,000) is much larger than the expected number of cycles (1,500). Therefore, it is concluded that the fatigue of the material is not a concern for the LACBWR package during its transportation.

2.6.6 Water Spray

Not applicable, since the package exterior is constructed of steel.

2.6.7 Free Drop

Under the normal conditions of transport (NCT), 10CFR71.71(c)(7) specifies that a free drop test of the package through a distance of 1 ft (for packages weighing more than 33,100 lb) on a flat, essentially unyielding, horizontal surface be performed. Under the normal conditions of transport, the LACBWR RPV package is always oriented in a horizontal orientation. Two orientations of the package, as shown in Figure 2-1, have been considered for this drop test. In the first orientation, the package is totally horizontal; the lowest point of the package is its welding ring, which contacts first with the unyielding surface. In the second orientation, the package axis is inclined at a 5° angle with the horizontal plane; the lowest point of the package is the endplate edge, which contacts first with the unyielding surface.

The demonstration of compliance with the regulatory requirements of the free drop test is accomplished by analytical evaluation as permitted by the regulations (10CFR71.41). Duratek Inc. proprietary document ST-517 (Reference 2-7) provides the details of these analyses. The results from these analyses, in a summary form, are presented in this SAR.

Finite element analysis methods, using the ANSYS/LS-DYNA (Reference 2-12) explicit dynamics computer code, have been employed to simulate the regulatory drop tests. Inelastic behavior of the package components – RPV, concrete, and the canister material is incorporated into the models. Under each drop condition, the finite element model of the package is dropped freely from the specified height on a rigid unyielding surface. The models are analyzed over a sufficiently large time period so that the kinetic energy of the package has been transformed into the internal energy and/or external work. The state of stresses and strain in the canister is observed throughout this period. The failure of the containment material is assumed to occur when the maximum tensile strain reaches the maximum specified elongation at the ultimate tensile strength of that material.

The finite element model is constructed from 3-dimensional 8-node hexahedral solid elements (ANSYS SOLID 164). All the major components of the package have been exclusively represented in the model. Figures 2-2 and 2-3 show the finite element model. The major components of the model are identified in these figures. Since all the bounding orientations considered for the evaluation, are symmetric about the vertical plane, only one-half of the geometry of the package has been modeled. Symmetry boundary conditions have been applied at the cut-plane. The model consists of 9,364 nodes and 5,170 elements.

The results of the analyses of 1-ft drop test simulation are summarized in Table 2-1. A discussion of these results is presented in the following sections.

1-ft Side Drop

The time-history plots of various energy and work quantities for this load case are included in Figure 2-4. Figures 2-5 and 2-6 present the stress intensity and tensile strain contour plots at the instant when these quantities achieve their maximum value. Figures 2-7 and 2-8 present the time-history plot of the attachment load in the 1¼" and 1¾" supplemental shield plates, respectively.

The maximum tensile strain of 10.804% is calculated for this drop test simulation. Since this value is smaller than the allowable value of 21%, it is concluded that no failure of the containment will occur during this drop test. Also, the supplemental shielding will remain attached during the test.

1-ft Inclined (Oblique) Drop

The time-history plots of various energy and work quantities for this load case are included in Figure 2-9. Figures 2-10 and 2-11 present the stress intensity and tensile strain contour plots at the instant when these quantities achieve their maximum value. Figures 2-12 and 2-13 present the time-history plot of the attachment load in the 1¼" and 1¾" supplemental shield plates, respectively.

The maximum tensile strain of 10.597% is calculated for this drop test simulation. Since this value is smaller than the allowable value of 21%, it is concluded that no failure of the containment will occur during this drop test. Also, the supplemental shielding will remain attached during the test.

It should be noted that LACBWR package containment boundary structure (canister) is made of solid steel for which the material properties used in the drop analyses (density, elastic modulus, yield stress and the tangent modulus) do not appreciably change through the temperature range of -20°F and 100°F. Therefore, the initial conditions for the cold and hot environment will not have any significant effect on the results and conclusions of the drop test analyses presented in this SAR.

2.6.8 Corner Drop

Not applicable; the LACBWR RPV package is not a fiberboard, wood, or fissile material package.

2.6.9 Compression

Not applicable; the LACBWR RPV package weighs more than 11,000 lbs.

2.6.10 Penetration

The package is evaluated for the impact of the hemispherical end of a vertical steel cylinder of 1¼" diameter and 13 lb mass, dropped from a height of 40″ onto the exposed surface of the package.

The penetration depth of the 13 lb 1¼″ diameter rod dropped from a height of 40 inch is calculated from the Ballistic Research Laboratories (BRL) formula sited in Reference 2-13. For a steel target, the penetration depth is given by the formula:

$$\left(\frac{e}{d}\right)^{3/2} = \frac{DV_0^{\,2}}{1.12\times10^6\times K_s^{\,2}}$$

Where,

e = penetration depth, inch

d = effective projectile diameter, inch = 1.25″

W = missile weight, lb = 13 lb

D = caliber density of the missile, lb/in^3 = W/d^3

V_0 = striking velocity of the missile, ft/sec

K_s = steel penetrability constant = 1.0

For 40″ drop of the rod, the striking velocity,

V_0 = $(2\times32.2\times40/12)^{0.5}$ = 14.65 ft/sec

D = $13/1.25^3$ = 6.656 lb/in^3

Solving the penetration equation, we get,

$$e = 1.25\times\left(\frac{6.656\times14.65^2}{1.12\times10^6\times1^2}\right)^{2/3} = 0.0147''$$

Since the minimum thickness of the LACBWR RPV canister is 1½ ″, the puncture drop test will not cause any damage to the package.

2.7 HYPOTHETICAL ACCIDENT CONDITIONS

This Section demonstrates that the package is structurally adequate to meet the performance requirements of Subpart E of 10 CFR 71 when subjected to HAC as defined in 10 CFR 71.73. Compliance with these requirements is demonstrated by analyses in lieu of testing as allowed by 10CFR 71.41(a) and Regulatory Guide 7.6 (Reference 2-3).

2.7.1 Free Drop

Under the hypothetical accident conditions (HAC), 10CFR71.73(c)(1) specifies that a free drop test of the package through a distance of 30 ft on a flat, essentially unyielding, horizontal surface be performed. The tests are required to be performed in orientations, which may result in the maximum damage to the package. Three orientations of the package, as shown in Figure 2-14, have been considered for this drop test. In the first orientation, the package is totally horizontal; the lowest point of the package is its welding ring that contacts first with the unyielding surface. In the second orientation, the package axis is inclined at a 5° angle with the horizontal plane; the lowest point of the package is the endplate edge that contacts first with the unyielding surface. In the third orientation the center of gravity (C.G.) of the package is aligned with the lowest point of the package along a vertical axis. Other orientations, including the end drop orientations, are enveloped with these orientations.

The package is analyzed with the help of ANSYS/LS-DYNA finite element model, as described in Section 2.6.7 and shown in Figures 2 and 3, except that the supplemental shielding plates have been assumed to have been detached during all the 30 ft drop tests. These plates are not needed for satisfying the shielding requirements during the HAC events (as shown in Chapter 5). Their removal from the finite element model results in a conservative evaluation of the package under the drop tests.

The finite element models are analyzed over a sufficiently large time period so that the kinetic energy of the package has been transformed into the internal energy and/or external work. During the time interval analyzed, several impacts between various parts of the package and the unyielding surface do take place. For the inclined and the corner-over-CG orientations the so-called "slap-down" effect is automatically included in the analyses. The state of stresses and strain in the canister is observed throughout the analysis period. The failure of the containment material is assumed to occur when the maximum tensile strain reaches the maximum specified elongation at the ultimate tensile strength of that material.

Duratek Inc. proprietary document ST-517 (Reference 2-7) provides the details of the HAC drop test analyses. The results from these analyses, in a summary form, are presented in this SAR.

The results of the analyses of 30-ft drop test simulation are summarized in Table 2-2. A discussion of these results is presented in the following sections.

2.7.1.1 End Drop.

The end drop orientation of the package is enveloped by the three other orientations analyzed in this SAR.

2.7.1.2 Side Drop.

The time-history plots of various energy and work quantities for this load case are included in Figure 2-15. Figures 2-16 and 2-17 present the stress intensity and tensile strain contour plots at the instant when these quantities achieve their maximum value.

The maximum tensile strain of 19.197% is calculated for this drop test simulation. Since this value is smaller than the allowable value of 21%, it is concluded that no failure of the containment will occur during this drop test.

30-ft Inclined Drop

The time-history plots of various energy and work quantities for this load case are included in Figure 2-18. Figures 2-19 and 2-20 present the stress intensity and tensile strain contour plots at the instant when these quantities achieve their maximum value.

The maximum tensile strain of 31.171% is calculated for this drop test simulation. Since this value is larger than the allowable value of 21%, it indicates that there will be a rupture of the containment material. To examine in more details as to where this rupture is expected, the maximum strain contour plot of the package, excluding the elements representing the outer half of the two end plates, is obtained as shown in Figure 2-21. This plot shows that a maximum tensile strain of 13.97% occurs in this part of the package. Therefore, it is concluded that the rupture may occur at the outer face of the endplate near the point of impact but it will extend to less than half the thickness of the plates (i.e. 2"). No failure of the containment is expected during this drop test.

Although the results of the analyses show that no failure of the containment is expected during this drop test, it is nonetheless postulated that the due to material imperfection or other reasons failure of the package may occur at the location of high tensile strains. Clearly such a failure will be limited to a very small region (see Figure 2-20). This region is assumed to be extending to 10° in the circumferential direction on either side of the plane of impact. The consequence of such a failure are addressed in Chapter 4 of this SAR.

2.7.1.3 Corner Drop.

The time-history plots of various energy and work quantities for this load case are included in Figure 2-22. Figures 2-23 and 2-24 present the stress intensity and tensile strain contour plots at the instant when these quantities achieve their maximum value.

The maximum tensile strain of 24.068% is calculated for this drop test simulation. Since this value is larger than the allowable value of 21%, it indicates that there will be a rupture of the containment material. To examine in more details as to where this rupture is expected, the maximum strain contour plot of the package, excluding the elements representing the outer half of the two end plates, is obtained as shown in Figure 2-25. This plot shows that a maximum tensile strain of 18.647% occurs in this part of the package. Therefore, it is concluded that the rupture may occur at the outer face of the endplate near the point of impact but it will extend to less than half the thickness of the plates (i.e. 2″). No failure of the containment is expected during this drop test.

Although the results of the analyses show that no failure of the containment is expected during this drop test, it is nonetheless postulated that the due to material imperfection or other reasons failure of the package may occur at the location of high tensile strains. Clearly such a failure will be limited to a very small region (see Figure 2-24). This region is assumed to be extending to 10° in the circumferential direction on either side of the plane of impact. The consequence of such a failure is addressed in Chapter 4 of this SAR.

2.7.1.4 Oblique Drops.

The shallow angle side drop test evaluation has been included under 2.7.1.2. As discussed under Section 2.7.1 the finite element models are analyzed over a sufficiently large time period. During this period, several impacts between various parts of the package and the unyielding surface do take place. For the inclined and the corner-over-CG orientations the so-called "slap-down" effect is automatically included in the analyses.

2.7.1.5 Summary of Results.

The results of the HAC drop test evaluation is summarized as follows:

- The supplemental shielding plates welded on the lower part of the canister detach from the package.
- The region near the impact point of the package deforms severely but remain within the allowable limits of the plastic strain. However, for conservativeness it is assumed that the welds in the severely stretched region do fail. The failure region is assumed to be extending to 10° in the circumferential direction on either side of the plane of impact.

- The MDCC near the point of impact will crack and crush.
- The RPV and the LDCC will experience relatively small stresses and strains.

2.7.2 Crush

Not applicable; the LACBWR package weighs more than 1,100 lb, its density is larger than 62.4 lb/ft3, and it does not contain greater than 1000 A2 radioactivity.

2.7.3 Puncture

The puncture drop test specified in 10CFR71.73(c)(3) requires that the package be dropped on a 6″ diameter mild steel rod from a height of 40″. The Nelm's Equation (Reference 2-14) predicts that a package weighing W, made with steel having an ultimate strength Su needs a shell thickness t to prevent penetration of the puncture bar, which is given by the formula:

$$t = (W/S_u)^{0.7}$$

For LACBWR RPV package, $W = 639{,}000$ lb, $S_u = 70{,}000$ psi, then,

$$t = (639{,}000/70{,}000)^{0.7} = 4.7''$$

Since the wall of the canister is 1½″ thick, it is predicted that the puncture drop test will result in the bar piercing through the canister shell. The MDCC behind the shell will impede further penetration of the rod. Since the shell of the RPV is 4″ thick, it can be concluded that under this test the wall of the RPV will remain intact.

The consequences of the puncture of the containment boundary under this test are addressed in Chapters 4 and 5.

2.7.4 Thermal

A qualitative evaluation of the LACBWR RPV package under a fully engulfing fire, as specified in 10 CFR 71.73(c)(4), has been performed in Section 3.4 of the SAR.

2.7.4.1 Summary of Pressures and Temperatures

Since the outer component of the package – the canister, is a welded structure that does not have any thermal insulation and dissimilar metal joints, under the fire test, the entire canister will rise to a temperature close to 1475°F, with very little temperature gradient through its wall. Also since the canister has been assumed to have developed cracks in the welds near the point of impact, during the drop tests, and a puncture through its wall during the puncture test, no pressure can develop inside the canister during the fire test.

2.7.4.2 Differential Thermal Expansion

Under the HAC fire test the temperature of the canister uniformly rises, with very little through the wall temperature gradient, the entire canister will expand uniformly under this test. Since there are no thermal insulation and dissimilar metal joints there will be no differential thermal expansion of the package.

2.7.4.3 Stress Calculations

Since under the HAC fire test the temperature of the canister uniformly rises, with very little through the wall temperature gradient, the entire canister will expand uniformly under this test, with very little thermal stress in the canister. Due to the cracks in the weld developed preceding the fire test, the canister will not be able to withhold any pressure. Thus there will be little or no primary stresses in the canister under the fire test conditions.

2.7.4.4 Comparison with Allowable Stresses

As discussed before, under the HAC fire conditions the LACBWR RPV package will not be able to contain any pressure. Therefore, it will develop little or no primary stresses. Also because of the uniformity of the structure (no dissimilar metals, no thermal insulators) the containment boundary of the package, i.e. the canister will uniformly expand under this test, developing no significant thermal stresses.

2.7.5 Immersion — Fissile Material

Not applicable for LACBWR RPV package; since it does not contain fissile material.

2.7.6 Immersion — All Packages

All the Type-B packages are required to meet the water immersion test specified in 10CFR71.73(c)(6). According to which, an undamaged package must be subjected to a pressure of 21.7 psig.

In Section 2.6.4, the LACBWR RPV package has been analyzed under an external pressure of 20 psig for stresses and buckling. The factors of safety calculated in that section can be linearly ratioed to obtain the corresponding factors of safety under the water immersion test. Thus, factor of safety on the stresses is:

$$F.S. = 3.28 \times 20/21.7 = 3.02$$

And, factor of safety against buckling of the shell is:

$$F.S. = 15.9 \times 20/21.7 = 14.65$$

2.7.7 Deep Water Immersion Test (for Type B Packages Containing More than 105 A2)

Not applicable; LACBWR RPV package does not contain more than 105 A2 (see Chapter 1).

2.7.8 Summary of Damage

The summary of damage due to the HAC fire test, which follows the HAC drop and penetration tests is as follows:

- The supplemental shielding plates welded on the lower part of the canister detach from the package.
- The region near the impact point of the package deforms severely.
- Possible, failed region (assumed to be extending to 10° in the circumferential direction on either side of the plane of impact).
- The MDCC near the point of impact will crack and crush.
- Penetration rod pierced through the canister wall, MDCC in the vicinity cracked, pulverized, and or lost. The RPV remains intact.

- The initial cracks in the weld caused during the HAC drop tests may expand during the fire test.

The effect of these damages on the shielding effectiveness is addressed in Chapter 5 and on containment effectiveness in Chapter 4.

2.8 ACCIDENT CONDITIONS FOR AIR TRANSPORT OF PLUTONIUM

Not applicable for LACBWR RPV package; since it neither contains plutonium, nor it is transported by air.

2.9 ACCIDENT CONDITIONS FOR FISSILE MATERIAL PACKAGES FOR AIR TRANSPORT

Not applicable for LACBWR RPV package; since it neither contains fissile material, nor it is transported by air.

2.10 SPECIAL FORM

Not applicable for LACBWR RPV package; since no special form content is included in the package.

2.11 FUEL RODS

Not applicable for LACBWR RPV package; since it does not include fuel rods.

2.12 APPENDIX

The appendix to this section includes the references and applicable pages from reference documents that are not readily available.

LIST OF APPENDICES

Appendix 2-1 – List of References

2-1. Code of Federal Regulations, Title 10, Part 71, *Transportation*.

2-2. Allis-Chalmers Drawing No. 43-501-186-501, *As-Built, LACBWR Reactor Vessel.*

2-3. U.S. NRC Regulatory Guide 7.6, Rev.1, 1978.

2-4. ASME Boiler and Pressure Vessel Code, Addenda through 2005, American Society of Mechanical Engineers.

2-5. U.S. NRC Regulatory Guide 7.11, 1991.

2-6. *Recommendations for Protecting Against Failure by Brittle Fracture in Ferritic Steel Shipping Containers Up to Four Inches Thick*, NUREG/CR-1815.

2-7. Duratek Proprietary Document ST-517, *Structural Analyses of the LACBWR RPV Package Under Various Drop Scenarios.*

2-8. *Fabrication Criteria for Shipping Containers*, NUREG/CR-3854, 1985.

2-9. *Guide for Cellular Concretes Above 50 pcf, and for Aggregate Concrete Above 50 pcf with Compressive Strengths Less Than 2500 psi*, ACI 523.3R-93, American Concrete Institute.

2-10. *Recommended Welding Criteria for Use in the Fabrication of Shipping Containers for Radioactive Materials*, NUREG/CR-3019, 1984.

2-11. *Formulas for Stress and Strain*, Roark and Young, Fifth Edition, McGraw Hill Publication.

2-12. ANSYS Release 9.0 (including the LS-DYNA module), ANSYS Inc., Canonsburg, PA.

2-13. *Structural Analysis and Design of Nuclear Plant Facilities*, ASCE Publication No.58, American Society of Civil Engineers (excerpt included in Appendix 2-4).

2-14. *Radioactive Materials Packaging Handbook*, Design, Operations, and Maintenance, ORNL/M-5003, 1998 (excerpt included in Appendix 2-4).

Appendix 2-2 – Excerpt from Reference 2-6

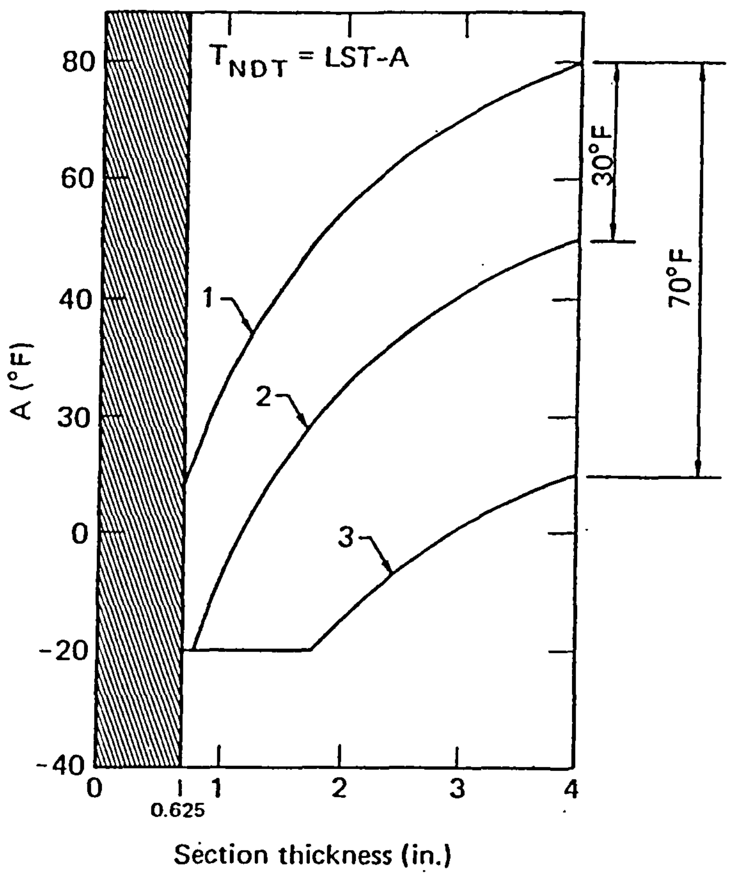


FIG. 6. Design chart for Category II fracture critical components showing reference temperature relative to NDT as a function of section thickness (derived from Fig. 7). Curve 1 is the basic K_{ID}/σ_{yd} curve for $\beta = 0.6$, and represents full dynamic loading with stresses at yield stress level. For effective g loadings of less than approximately 100 g: curve 2, shifted 30°F, may be used for steels with σ_{ys} in the range 60 ksi $\leqslant \sigma_{ys} \leqslant$ 100 ksi; curve 3, shifted 70°F, may be used for steels with σ_{ys} less than 60 ksi.

Appendix 2-3 – Excerpt from Reference 2-4

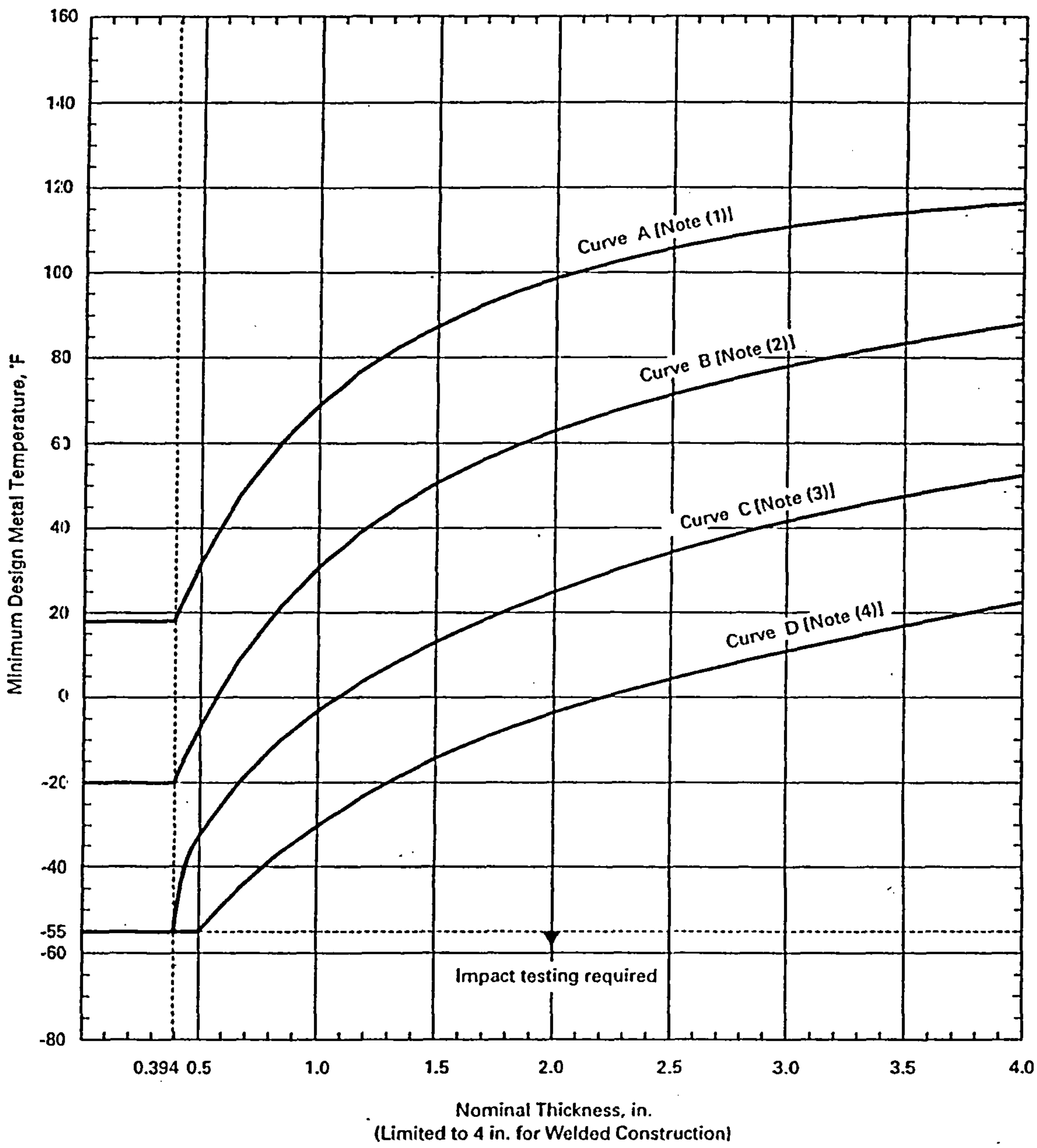


FIG. AM-218.1 IMPACT TEST EXEMPTION CURVES

(Notes to figure follow on next page)

Appendix 2-4 – Excerpt from Reference 2-13

6.4.1.2.7 *Design Considerations.*—It has been recommended(50) that in order to provide a sufficient safety margin, the design thickness, t_d, should be 10% to 20% greater than the threshold value for the phenomenon being prevented. Thus, to prevent perforation:

$$t_d = 1.2e \tag{6.43}$$

If it is necessary to prevent scabbing, then the design thickness should be:

$$t_d = 1.2s \tag{6.44}$$

These design recommendations account for scatter of the test data and uncertainty in the prediction equations.

If one must design to prevent scabbing, then a steel scab plate must be attached to the rear face of the concrete, or else the concrete thickness must satisfy Eq. 6.44. To prevent scabbing through the use of a steel scab plate, the concrete thickness must satisfy Eq. 6.43 and the steel scab plate should meet the following criteria(42):

1. The scab plate must be rigidly anchored to the concrete by embedded anchors whose spacing is based on analysis or testing.

2. All construction joints in the scab plate must be continuous butt joined and groove welded so that the plate does not pull apart at the joints.

3. The inclusion of a thin scab plate will not allow a decrease in the perforation thickness of the concrete slab.

4. An improperly designed or constructed scab plate is worse than no scab plate at all. It is imperative that the scab plate be rigidly attached to the concrete slab at close intervals. Spot welding to the shear steel is insufficient.

For nearly nondeformable missiles (steel slugs, steel pipe, etc.), satisfying Eq. 6.43 to prevent perforation will also automatically prevent a gross punching shear failure, and punching shear does not need to be checked separately. For a soft missile (wood pole, aircraft impact, etc.), perforation will not be induced by missile impact, but a punching shear failure is still possible and should be checked in accordance with Section 6.2.3.

In some instances multiple barriers in series may be used to prevent missile perforation. In this case, the first barrier will be perforated but will slow the missile down, while the second barrier will stop the slowed missile. The residual missile velocity remaining after the missile has perforated the first barrier must be determined. A number of procedures have been proposed for determining this residual missile velocity(9, 13, 42, 89). All are roughly similar to the following simple approach:

$$V_R^{1.8} = V_I^{1.8} - V_P^{1.8}, \text{ for } V_I > V_P \tag{6.45}$$

where V_R is the residual missile velocity; V_I is the missile impact velocity; and V_P is the velocity to just perforate the target (i.e., no residual velocity) as obtained from the perforation thickness equations.

A water barrier also may be used to reduce the missile velocity prior to striking the concrete target. Procedures are available(9) for accounting for the influence of the water barriers on missile velocity.

6.4.1.3 *Steel Targets*—Steel targets, such as pipes and mechanical equipment vessels, ~~may be~~ perforated by an impacting nondeformable steel missile. Sometimes protruding elements of a missile may puncture a steel target while the entire missile does not perforate or pass through the target. The minimum contact area of a missile protrusion should be used to calculate puncture thickness, and the projected area of the entire missile should be used to calculate perforation thickness(9).

The current procedures for determining the puncture or perforation thickness of steel targets is to use primarily empirically-derived formulas which are based upon tests involving the impact of small diameter, high calibre, high density, non-deformable projectiles striking thin steel targets. The two formulas most commonly used are the Ballistic Research Laboratory (BRL) formula and the Stanford Equation.

The Ballistic Research Laboratory Formula(38, 57) can be written as:

$$\left(\frac{e}{d}\right)^{3/2} = \frac{D v_s^2}{1{,}120{,}000 K_s^2} \tag{6.46}$$

where K_s is a steel penetrability constant depending upon the grade of the steel target (usually taken as approximately 1.0). The range of test data parameters used in developing this formula and its range of applicability are both undefined.

The Stanford Equation was developed by the Stanford Research Institute(38, 58) in 1963 and is more thoroughly documented. The formula is based upon tests with the following range of parameters:

$$0.1 \leq \frac{e}{d} \leq 0.8$$

$$2\ \text{lb/in}^3\ (550 \times 10^2\ \text{kg/m}^3) \leq D \leq 12\ \text{lb/in}^3\ (3{,}300 \times 10^2\ \text{kg/m}^3)$$

$$0.062\ \text{in.}\ (1.6\ \text{mm}) \leq d \leq 3.5\ \text{in.}\ (89\ \text{mm})$$

$$70\ \text{fps}\ (21\ \text{m/sec}) \leq v_s \leq 400\ \text{fps}\ (120\ \text{m/sec})$$

$$2\ \text{in.}\ (50\ \text{mm}) \leq B \leq 12\ \text{in.}\ (300\ \text{mm})$$

$$5 \leq \frac{B}{d} \leq 8$$

$$8 \leq \frac{B}{e} \leq 100$$

where B represents the width of plate between rigid supports. The Stanford Equation can be rewritten in the form:

$$\left(\frac{e}{d}\right)^2 + \frac{3}{128}\left(\frac{B}{d}\right)\left(\frac{e}{d}\right) = \frac{0.0452 D v_s^2}{S_t} \tag{6.47}$$

With the possible exception of the ratios (B/d) and (B/e), the range of test parameters from which the Stanford Equation was developed is reasonable for missile impact parameters of interest to the nuclear facility industry for steel targets. To ensure conservatism for (B/d) ratios greater than 8, or (B/e) ratios greater than 100, it is recommended(37) that Eq. 6.47 be modified as follows:

$$\left(\frac{e}{d}\right)^2 + \frac{3F}{128}\left(\frac{e}{d}\right) = \frac{0.0452Dv_s^2}{S_t} \qquad (6.48)$$

where $F = (B/d)$, except $F \leq 8$, and $F \leq 100(e/d)$, whichever is lower.

Over most of the range of available test data, the BRL Formula (Eq. 6.46) and the Modified Stanford Equation (Eq. 6.48) predict similar results. However, for values of (B/d) less than 6, the BRL Formula tends to be unconservative when compared with the test data upon which the Stanford Equation is based.

For the design of steel targets, it is recommended that the minimum design thickness, t_d, be given by:

$$t_d = 1.25e \qquad (6.49)$$

where the perforation thickness, e, is obtained from Eq. 6.48. In determining the perforation thickness, the ultimate tensile strength of the target, S_t, should be reduced by the amount of bilateral tensile stress already in the target(38).

6.4.2 Consideration Of Overall Effects.—In order to develop methods for evaluating overall response, the type of impact may be classified as either "hard" or "soft." The "softness" of a given impact is obviously relative, but in general, soft impact is characterized by significant local deformation of the missile or target in the region of impact, while local deformation under hard impact is small. In the soft impact case, discussed in Subsection 6.4.2.1, the deformation characteristics of the missile or target are used to develop an applied force time history, and analysis for overall response to the force is carried out as for an impulsive load. In the hard impact case, discussed in Subsection 6.4.2.2, energy and momentum balance techniques are used to predict maximum response.

6.4.2.1 *Analysis Of Soft Missile Impact.*—As indicated above, the terminology "soft missile impact" is used here to describe that class of problems characterized by significant local deformation of the missile and/or target structure during impact. Most missile impacts postulated in nuclear power plant design are in fact "soft." Tornado wind-driven objects, such as reinforcing bars and small pipes, and internally generated missiles, such as bolts, valve stems and fragments from rotating equipment, will penetrate into concrete targets upon impact. Missiles such as aircraft, automobiles and wood poles will themselves deform upon striking concrete structures. The two cases of target penetration and missile deformation are treated individually in the following.

6.4.2.1.1 *Missile Penetration Into Target.*—In cases where missile penetration into the target structure upon impact is significant, relationships can be developed for the forcing function applied to the structure by the missile, based on application of the equation of motion during deceleration of the missile. This approach was first proposed in Ref. 53. However, the development herein is somewhat different than that in Ref. 86 to ensure consistency between basic assumptions, resulting in a different form for the forcing function derived.

It is assumed that the velocity varies linearly to zero as a function of time as the missile penetrates the structure. Implicit in this assumption is a constant acceleration, and consequently a constant force of impact. The total kinetic energy of the missile before impact is expended as it penetrates into the structure.

If it is assumed that the overall deflection of the structure during impact is negligible when compared to penetration, then the work done by the missile as it penetrates the structure is equal to the initial kinetic energy. Thus:

$$F_i X = \frac{1}{2}\left(\frac{W v_o^2}{g}\right) \tag{6.50}$$

or

$$F_i = \frac{W v_o^2}{2gX} \tag{6.51}$$

where F_i = force of impact; g = acceleration of gravity; W = weight of missile; v_o = initial velocity of missile; and X = penetration

The assumption that the velocity reduces linearly to zero may be used to determine the time of impulse or duration of the impact force:

$$t_d = \frac{2X}{v_o} \tag{6.52}$$

The value of F_i (Eq. 6.51) and t_d (Eq. 6.52) completely define a rectangular impulse loading applied to the structure. The analysis of the structure for the rectangular impulse can be carried out using the procedures described in Section 6.3 or 6.5. Note that parametric curves for prediction of maximum response of systems subject to rectangular impulse loads are given in Fig. 6.7 through 6.9. The application of the technique is demonstrated in Example Problem 2 following.

In the above "soft target" formulation, the magnitude and duration of the impulse load are dependent upon missile weight and initial velocity, and penetration depth into the target. It has been recommended in the preceding section that the Modified NDRC Formula be used to predict missile penetration depth in concrete targets. A predicted penetration depth greater than about 15% of target thickness is considered sufficient to validate the application of the soft target technique. Conversely, the predicted penetration depth must be less than that to cause perforation or spalling, since these would constitute failure due to local effects. Between these extremes, it is generally conservative to underpredict penetration depth, which will increase magnitude and decrease duration of load. Response to short duration rectangular impulse loads is primarily dependent upon their magnitude, with little sensitivity to duration.

6.4.2.1.2 *Missile Deformation During Impact.*—When deformable missiles such as aircraft, automobiles, whipping pipes, or wood poles impact a structure, both local and overall effects must be considered in design. Local effects, such as punching shear or penetration of aircraft subcomponents, can be evaluated using procedures presented in preceding sections. Methods are described in this section

Appendix 2-5 – Excerpt from Reference 2-14

Adjustments to modeling methods or recalculation of stresses in the region near the boundary condition by using a different model may be used to correct stresses in these regions.

Quasistatic analyses of the cask impact stresses may be less expensive than dynamic analyses, and quasistatic analyses result in calculated stresses which may be directly compared to ASME stress allowables. If dynamic analyses are performed, dynamic allowable stress limits must be calculated and documented. A quasistatic analysis may be adjusted for dynamic effects by scaling the calculated stresses with a "dynamic load factor." Typically, the dynamic load factor for impacts less than 100 *g* range from 1.0 to 1.2. Higher values for the dynamic load factor should be used for loads greater than 100 *g*. The dynamic load factor used in the analyses of the SARP should be confirmed in scale-model drop testing. The stresses in the scale impact tests may be compared with the calculated quasistatic stresses, adjusted by the dynamic load factor, to demonstrate a measure of conservatism.

5.2.1 Pin Puncture

Multiwall casks, consisting of a structural inner wall, a shield annulus, and a structural outer wall, are usually designed to provide required shielding in a package with the minimum weight. A benefit of this construction is that the outer structural wall of a multiwall cask provides protection from pin puncture so that the inner structural wall is not deformed in a puncture impact. Lead shielding effectively isolates the inner structural wall of a multiwall cask. The lead layer acts as the shock-absorbing medium, distributing the energy of the puncture impact, which is concentrated in a 15-cm-diam circle of the outer shell, over a larger area of the inner shell. The puncture shock wave propagates inward from the outer wall and approximates a hemispherical shock pattern. Hence, the energy in the shock wave decreases as the square of the distance traveled inward toward the inner wall. This effect is illustrated in Fig. 5.1. Some nonscale dimpling of the inner structural wall of quarter-scale models can occur in pin puncture because of nonlinear scaling of dynamic shock front. If the scale of a drop test model is 1.4, then the shock transmitted to the inner cask wall is increased by a factor of 16 by the nonlinear effect and reduced by a factor of 4 by the scaling of the puncture pin. The net result is a factor of 4 increase in shock energy transmitted to the inner cask wall. Dimpling of the inner structural wall caused by this effect is acceptable if the deformation does not cause an inteference fit with the cask cavity contents (fuel basket, special form container, or waste container)–which would raise the possibility of damage to the cask contents.

A series of pin puncture tests performed at Oak Ridge National Laboratory were used to develop an empirical equation[4] for the stress in the outer wall of a multiwall cask as a function of the mass of the cask and the thickness of the cask outer wall material. This equation (Nelm's equation) applies to steel-lead-steel cask wall construction. Nelm's equation may be used to demonstrate-pin puncture adequacy for casks with stainless steel or ferritic steel walls, and this equation has been the basis for the puncture analysis of several licensed casks.

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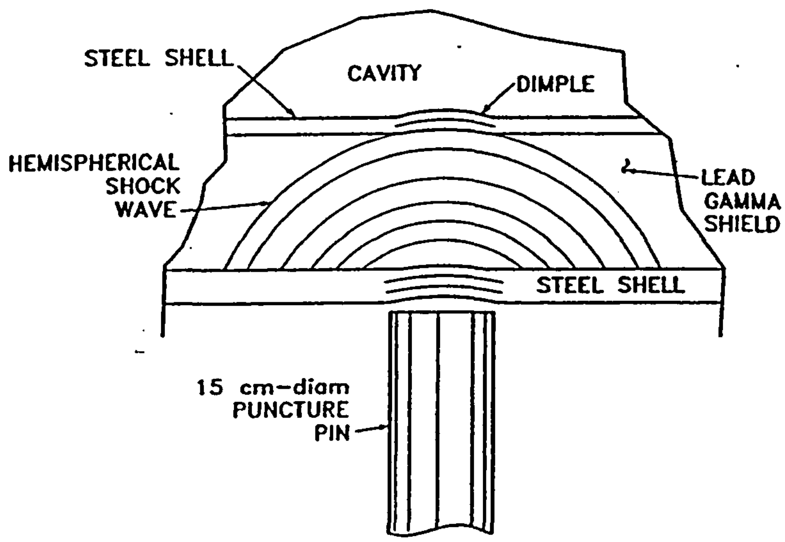


Fig. 5.1. Pin-puncture event.

For hot-rolled carbon steel and stainless steel outer shells, the minimum outer shell thickness required to withstand the punching action of a steel piston is given by Nelms equation:

$$t = (W/S)^{0.71} \,, \tag{5.1}$$

where

t = shell thickness, in.,
W = cask weight, lb, and
S = ultimate tensile strength of the outer shell, psi.

Note that this equation gives acceptable results when the diameter of the package is greater than about 75 cm and when the impact location is not close to a stiffening structure such as a heat transfer fin; for packages having a diameter below 75 cm, one should apply a factor of 1.3 to the resulting outer-shell thickness.

Nelm's equation does not apply to steel-depleted-uranium (DU) steel multiwall constructions because DU has properties similar to those of mild steel. The impact force of a puncture pin is thus transmitted to the inner structural wall in a concentrated form. A relatively thin layer of lead in a steel-DU-lead-steel cask provides puncture shock isolation because of the shock-absorbing effect of the lead and because the deformation of the outer wall must exceed the thickness of the lead annulus before the outer wall directly contacts and loads the DU layer. The minimum thickness of lead required to provide puncture shock isolation is package-design-dependent, but an analysis[5] of typical rail casks showed that a layer of 0.125 cm or more provided isolation. It should be recalled that the 15-cm-diam pin is constant regardless of the size of the cask and that a thin layer of lead would therefore be effective regardless of cask size. The manufacturing process required to provide a lead layer with consistent thickness is difficult and can require vacuum pouring.

Single-wall monolithic casks must withstand pin puncture with only local deformation. Fracture toughness requirements for monilithic casks may be evaluated per NUREGs for shells less than 4 in. (0.1 m) thick[1] and greater than 4 in. (0.1 m) thick.[2]

5.2.2 Bending

Cask shells must resist bending tensile loads without yielding. Compressive bending loads can lead to buckling instability that causes failure of a cylindrical shell before yield is reached. Analyses must show that the containment (inner) shell of a cask does not buckle. ASME Buckling Code Case N-0284 may be used to demonstrate that buckling does not occur. Classical crippling or buckling analyses may be used as an alternative to N-0284, but factors of safety appropriate to the specific cask and calculational methodology must be established. The factors of safety specified in N-0284 are applicable only if the methodology established in the N-0284 is used; these factors of safety may not be used with classical crippling or buckling analyses. Factors of safety are applied directly to the load to account for ovality and variations in shell thickness and must be evaluated together with the buckling stress methodology to ensure conservatism. Hence, the factors of safety of N-0284 may not be excerpted and applied to classical crippling or buckling analyses.

Cylindrical shells are desirable for casks because the stresses on them are axisymmetric. Casks with square or rectangular cross sections have been licensed in the past, but stresses are higher in their corner areas. Square shells with rounded corners reduce these stresses somewhat, and the rounded corners avoid discontinuities in the stress distribution which might be difficult to calculate precisely.

5.2.3 End Forgings

Forgings used to form the ends of casks may contain penetrations for filling or venting and could cause local stress concentrations and weak areas. Robust sections are desirable so that the effect of penetrations is minimized. Closure end forgings cannot take credit for any support that might be provided by bolted closure lid(s) and must be treated as unsupported rings.

The joint between shells and end forging (castings are not favored because of potentially nonuniform material properties) should not be a butt-weld. A portion of the shell material must be part of the end forging, as required by the ASME code, so that the weld joint is between cylindrical sections. Full-penetration welds are required to join sections of the containment boundary.

Weld efficiency factors may not be applied to containment welds. Radiography is required for all containment welds per the ASME code, and derating of weld zone stress allowables through the use of efficiency factors is, therefore, unnecessary.

Transition zones, thicker shell material near the end forgings, can reduce the effect of discontinuity stresses, as illustrated in Fig. 5.2. The taper (slope) of transition zones should be 1:3 (1-cm decrease in a 3-cm length of shell) or shallower as directed by the ASME Boiler & Pressure Vessel Code (B&PV), Sect. III, Division 1, NB-3339.6, and NB-4232.1.

Table 2-1

Summary of Results of 1-ft Drop Test Simulation

Orientation	Quantity	Max. Value	At Time After the Impact
Side Drop	Maximum Stress Intensity (psi)	60,043(1)	0.040 second
	Maximum Principal Stress (psi)	45,526	0.030 second
	Maximum Tensile Strain (%)	10.804(2)	0.315 second
Inclined Drop	Maximum Stress Intensity (psi)	58,661(3)	0.2275 second
	Maximum Principal Stress (psi)	44,787	0.2275 second
	Maximum Tensile Strain (%)	10.597(4)	0.2275 second

NOTES:

(1) See Figure 2-5 for the location of the maximum stress intensity.

(2) See Figure 2-6 for the location of the maximum tensile strain.

(3) See Figure 2-10 for the location of the maximum stress intensity.

(4) See Figure 2-11 for the location of the maximum tensile strain.

Table 2-2
Summary of Results of 30-ft Drop Test Simulation

Orientation	Quantity	Max. Value	At Time After the Impact
Side Drop	Maximum Stress Intensity (psi)	92,532(1)	0.0178 second
	Maximum Principal Stress (psi)	122,561	0.0054 second
	Maximum Tensile Strain (%)	19.197(2)	0.019 second
Inclined Drop	Maximum Stress Intensity (psi)	117,566(3)	0.0645 second
	Maximum Principal Stress (psi)	50,702	0.0645 second
	Maximum Tensile Strain (%)	31.171(4)	0.0705 second
Corner Drop With Slap-down	Maximum Stress Intensity (psi)	96,179(5)	0.030 second
	Maximum Principal Stress (psi)	46,970	0.795 second
	Maximum Tensile Strain (%)	24.068(6)	0.795 second

NOTES:

(1) See Figure 2-16 for the location of the maximum stress intensity.

(2) See Figure 2-17 for the location of the maximum tensile strain.

(3) See Figure 2-19 for the location of the maximum stress intensity.

(4) See Figure 2-20 for the location of the maximum tensile strain. The maximum tensile strain in the weld is less than 13.97% as shown in Figure 2-21.

(5) See Figure 2-23 for the location of the maximum stress intensity.

(6) See Figure 2-24 for the location of the maximum tensile strain. The maximum tensile strain in the weld is less than 18.647% as shown in Figure 2-25.

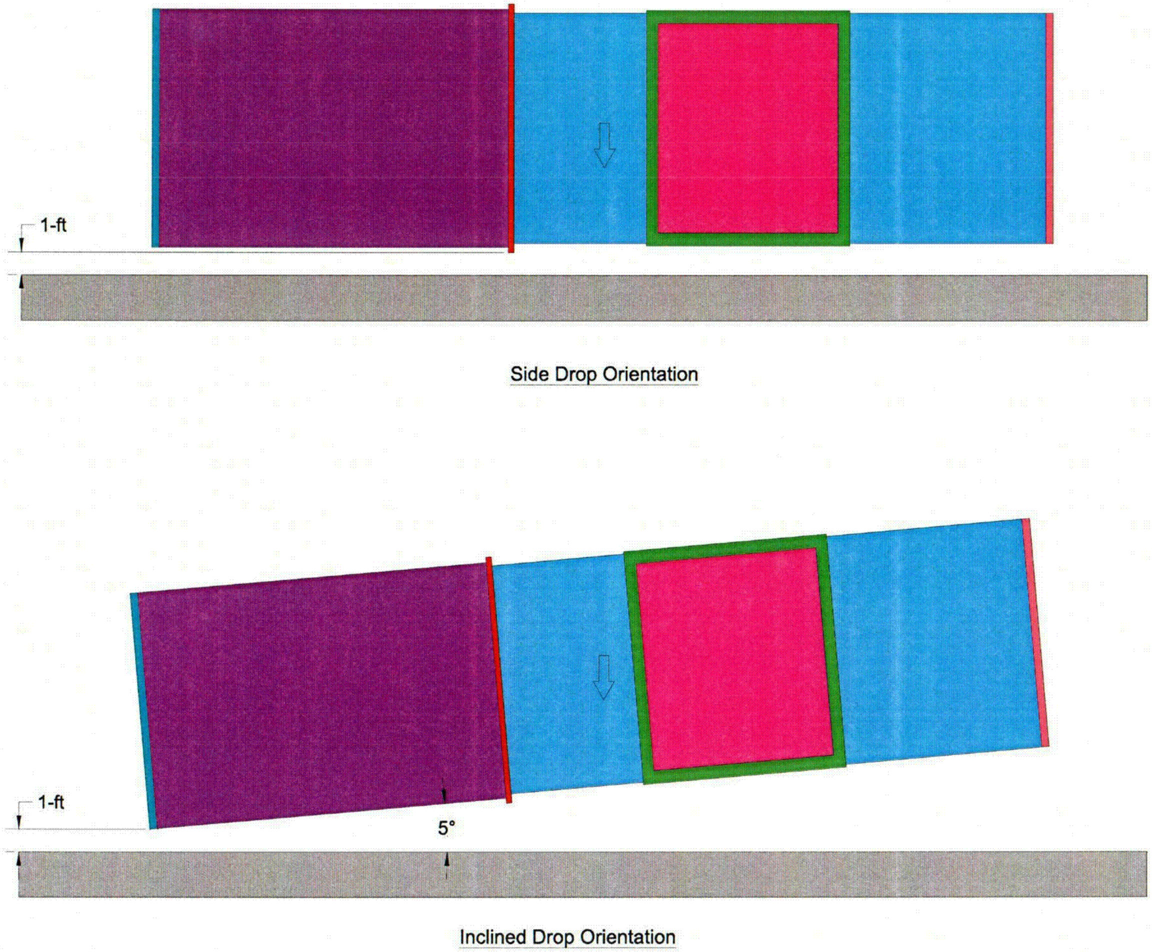


Figure 2-1
Package Orientations Analyzed for the 1-ft Drop Test Simulation

C01

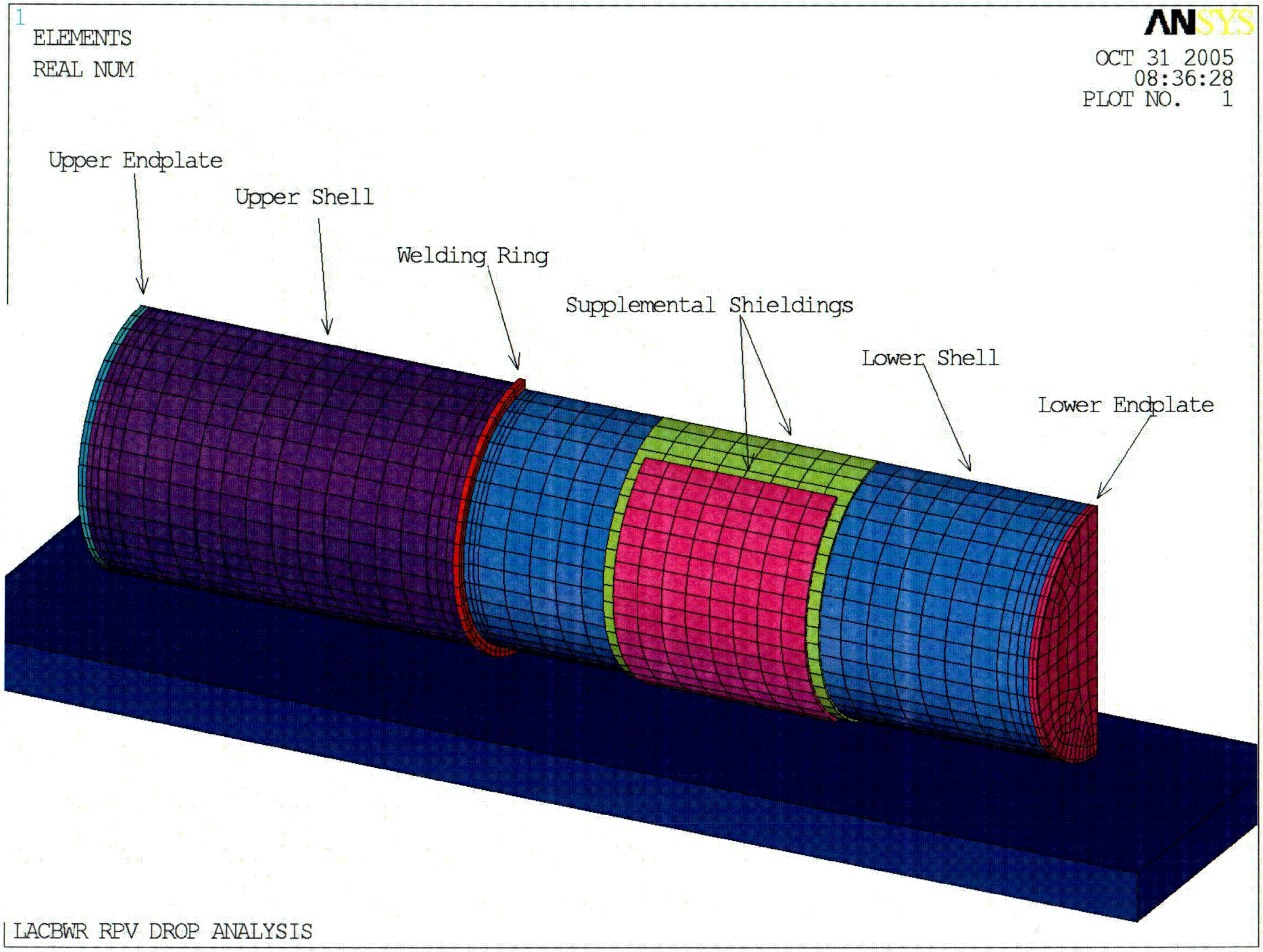


Figure 2-2
Finite Element Model of the LACBWR RPV Package Components

C02

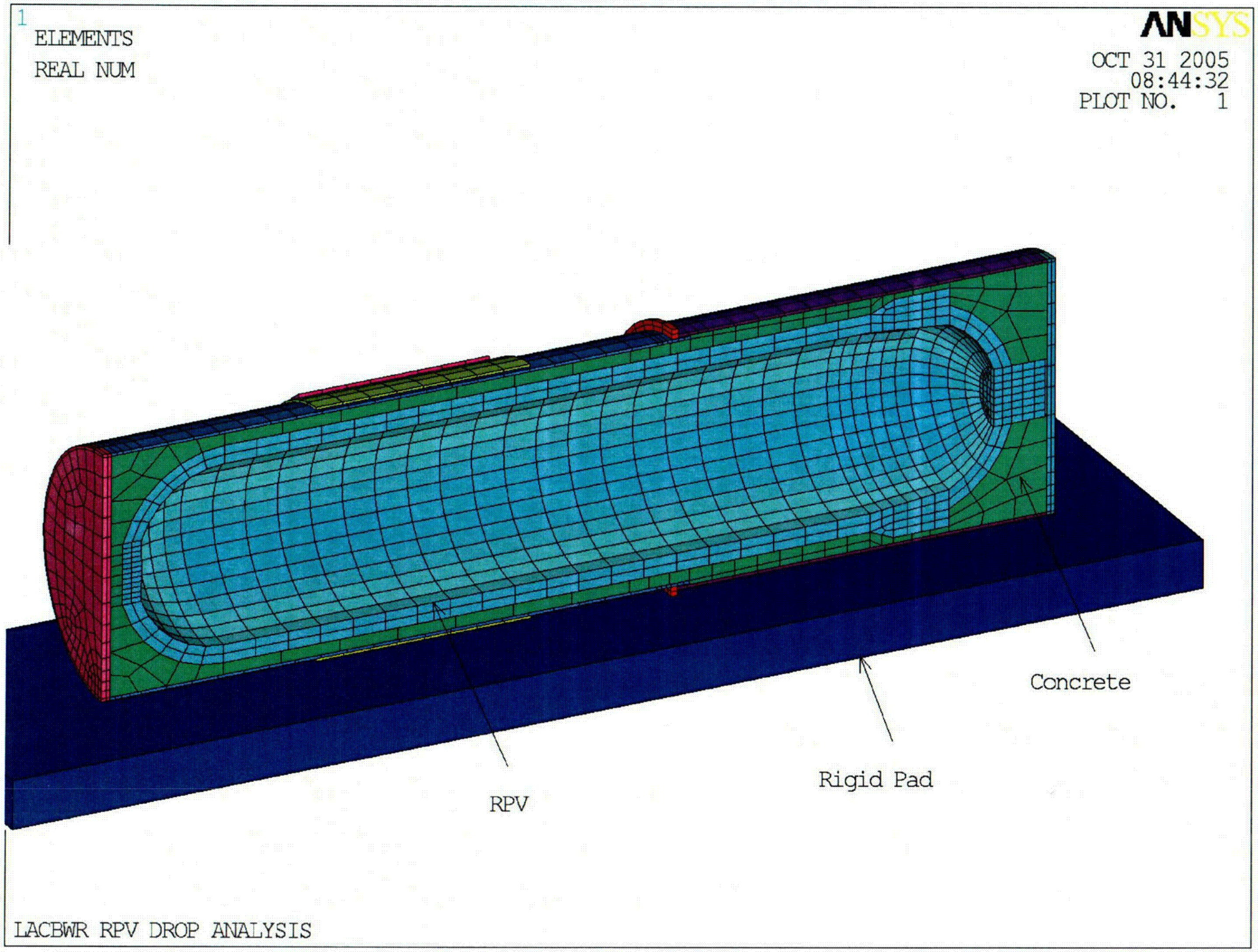


Figure 2-3
Finite Element Model of the LACBWR RPV Package Internals

C03

Figure 2-4
Time-History Plot of Various Quantities – 1-ft Side Drop

C04

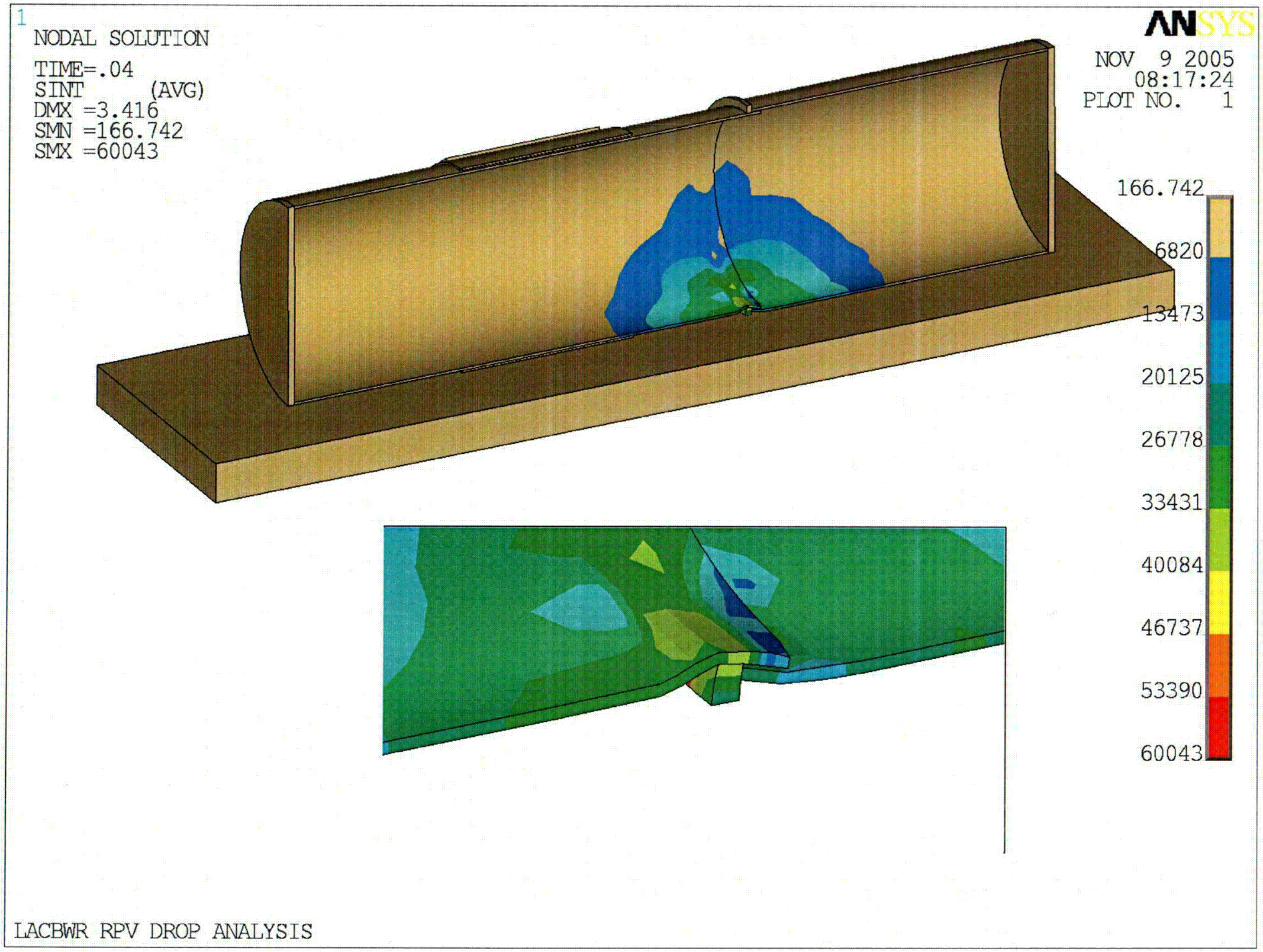


Figure 2-5
Stress Intensity Contour Plot of the Maximum S.I. – 1-ft Side Drop

C05

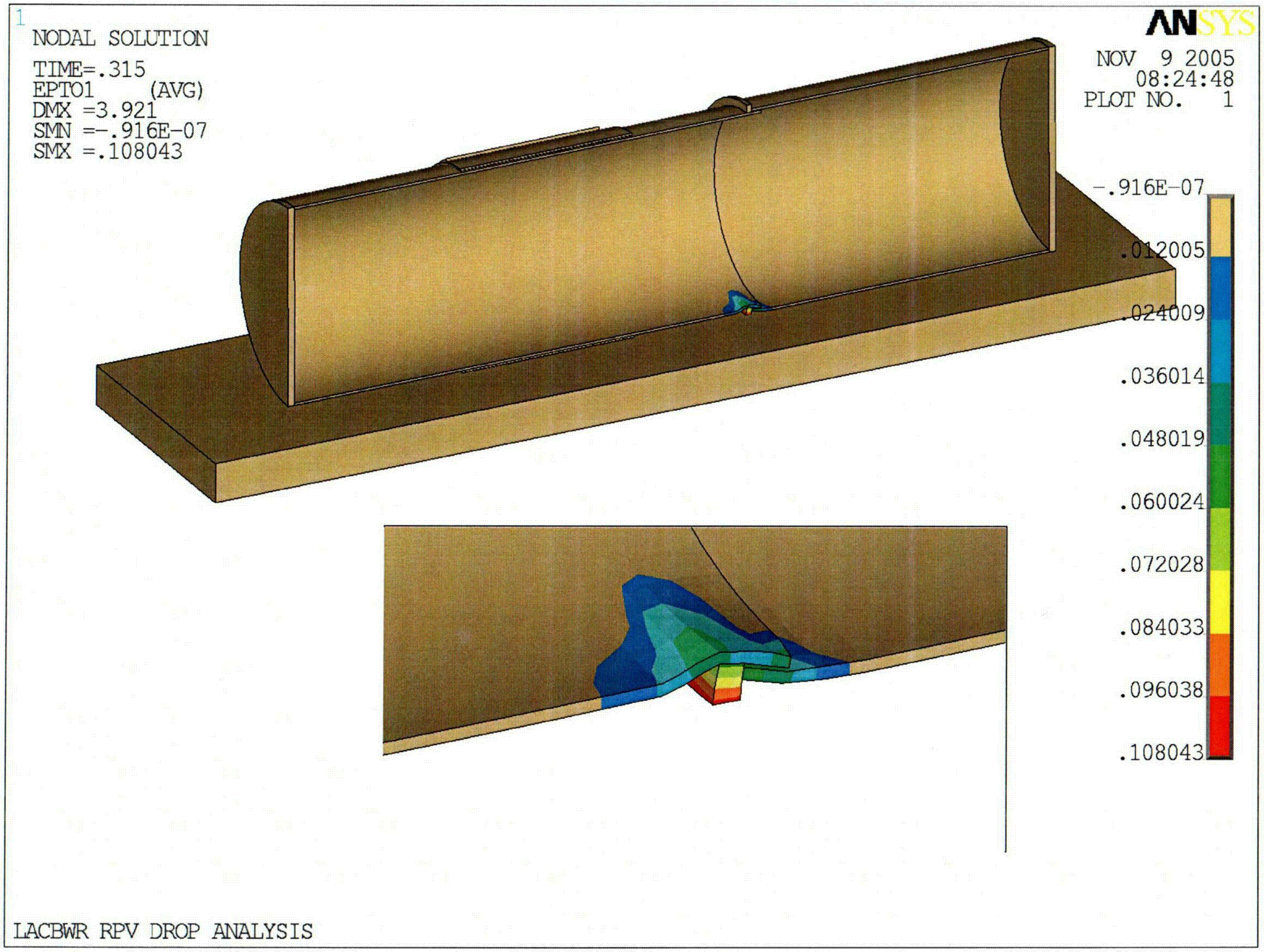


Figure 2-6
Stress Intensity Contour Plot of the Maximum Tensile Strain – 1-ft Side Drop

C06

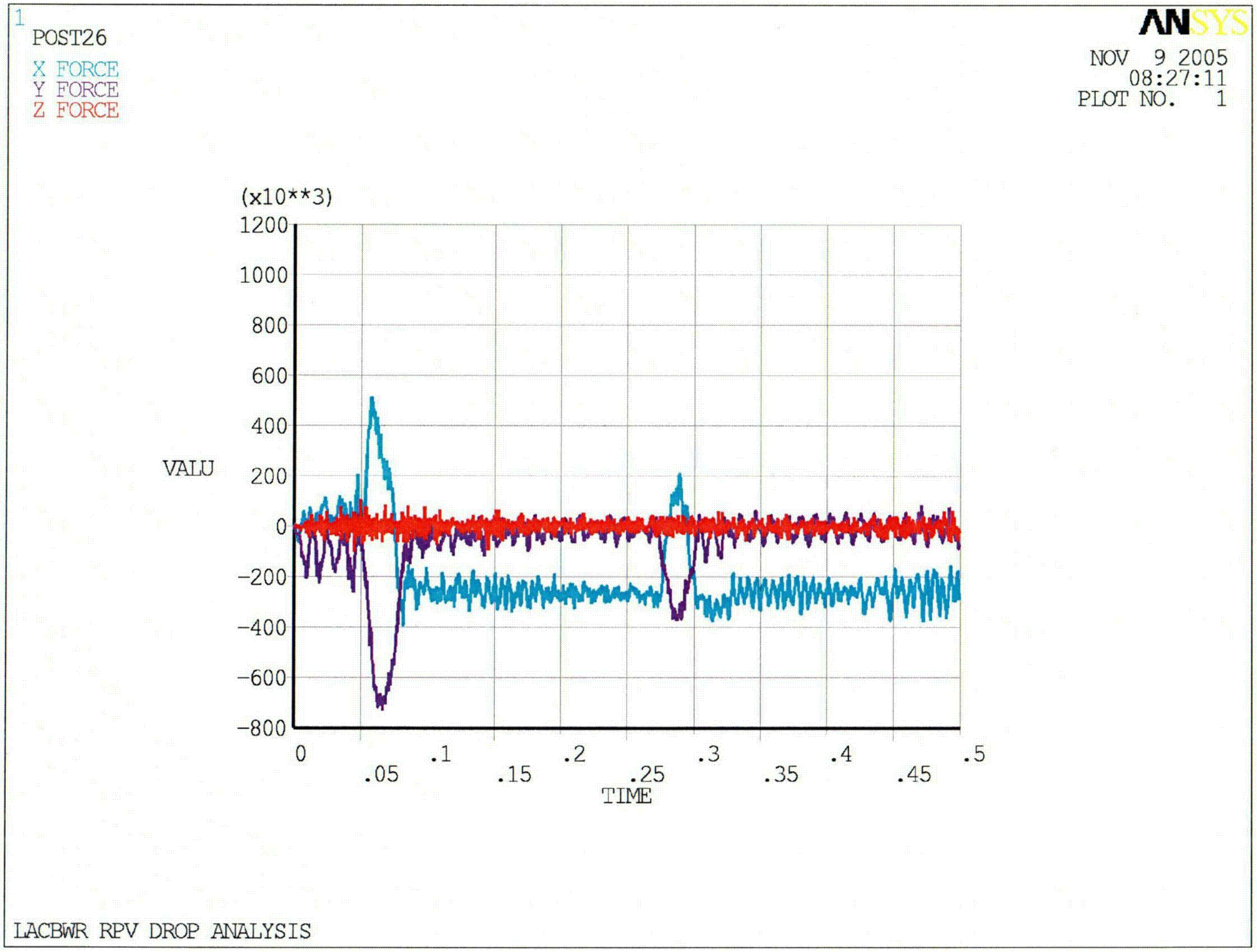


Figure 2-7
Time-History Plot of the Total Forces During 1-ft Side Drop in the 1¼ " Supplemental Shield Plates

C07

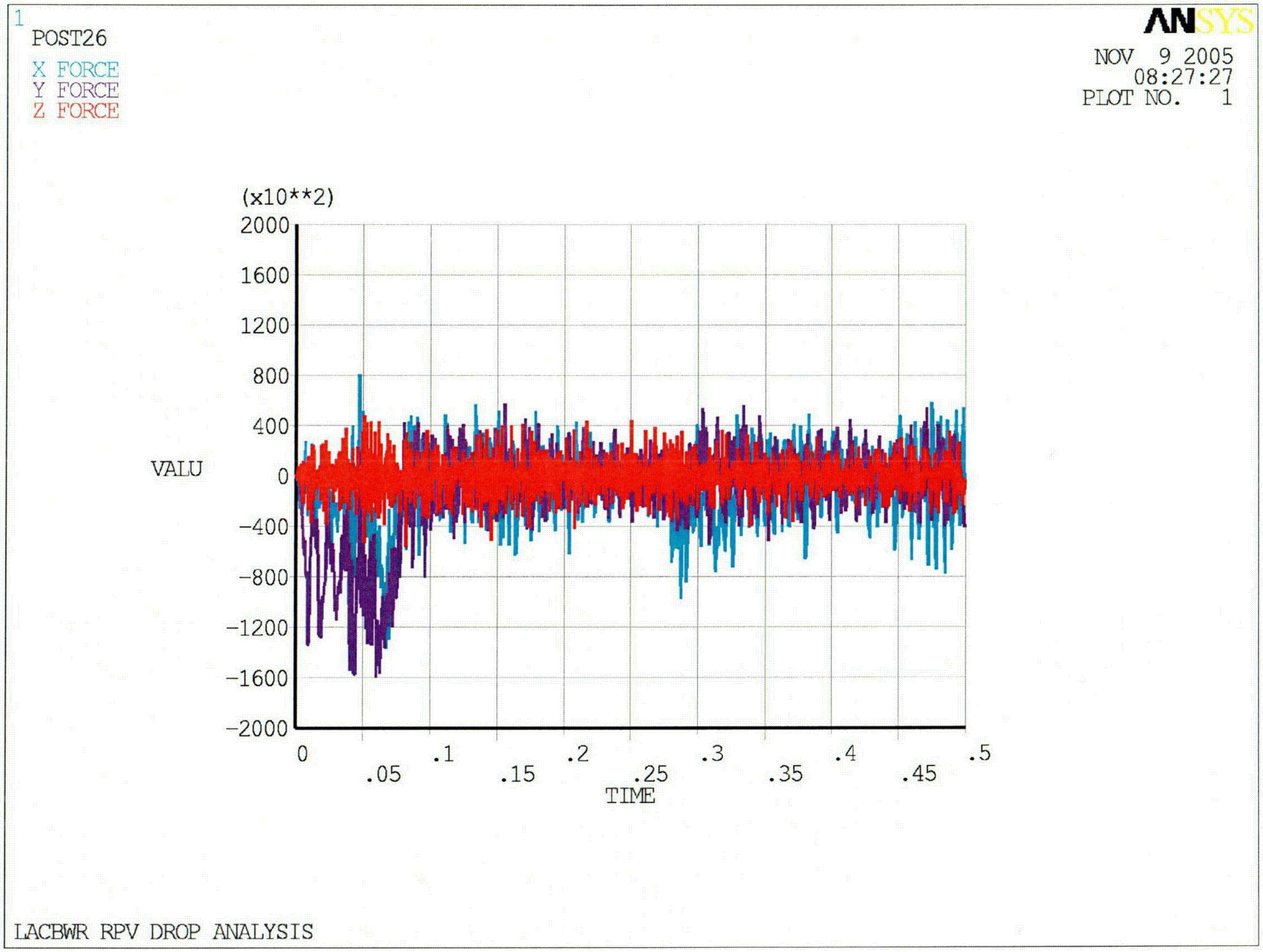


Figure 2-8
Time-History Plot of the Total Forces during 1-ft Inclined Drop in the 1¾ " Supplemental Shield Plates

C08

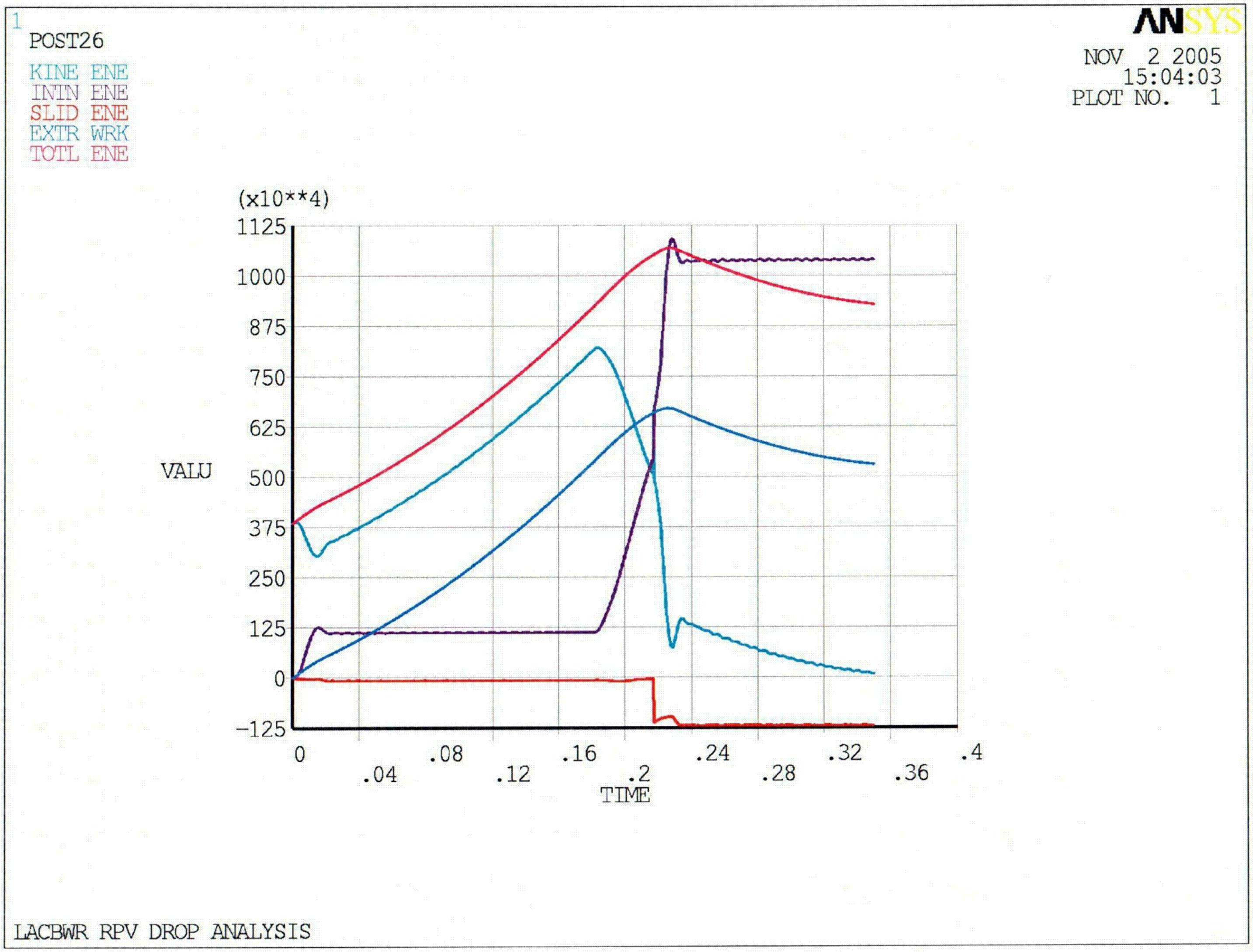


Figure 2-9
Time-History Plot of Various Quantities – 1-ft Inclined Drop

C09

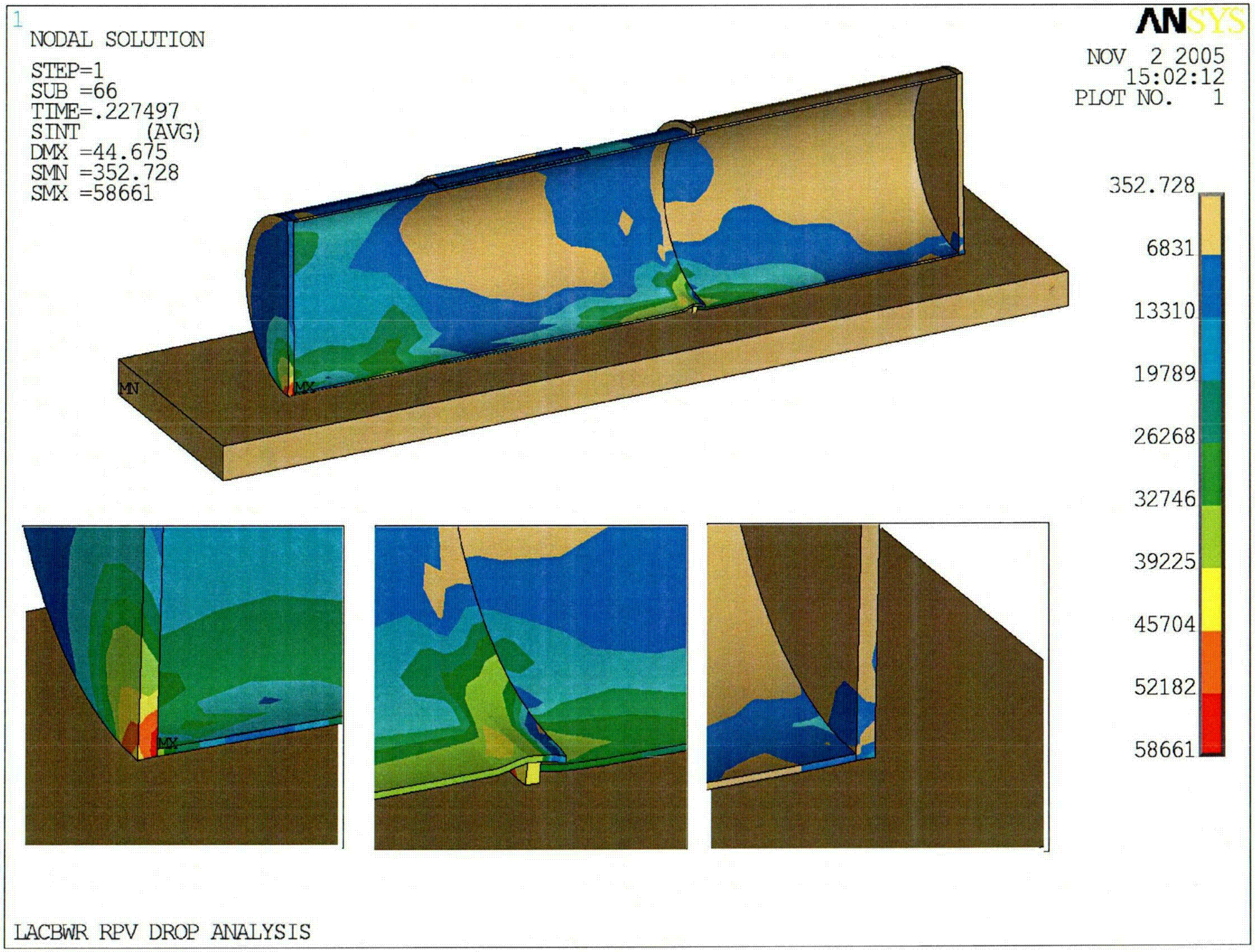


Figure 2-10
Stress Intensity Contour Plot of the Maximum S.I. – 1-ft Inclined Drop

C10

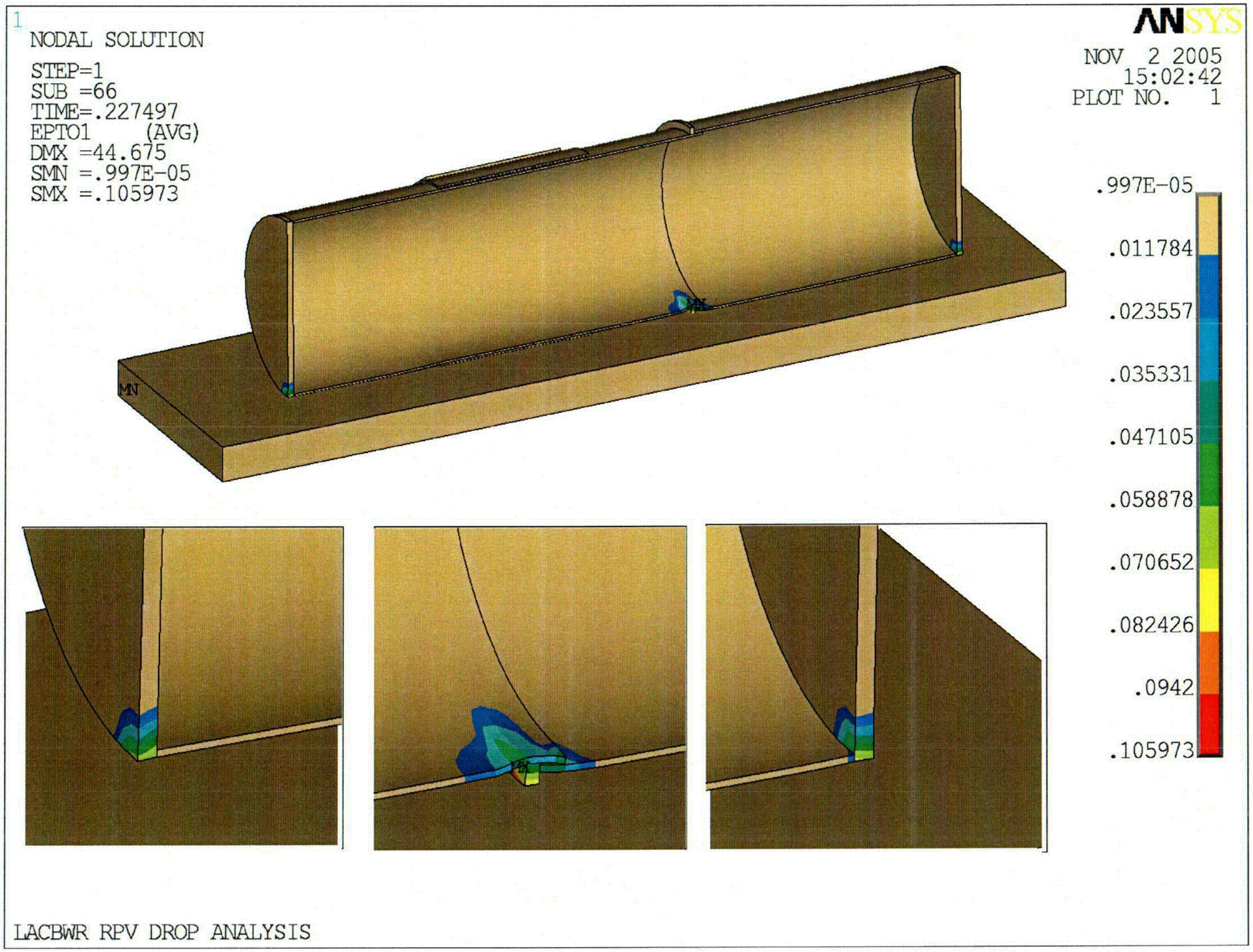


Figure 2-11
Stress Intensity Contour Plot of the Maximum Tensile Strain – 1-ft Inclined Drop

C11

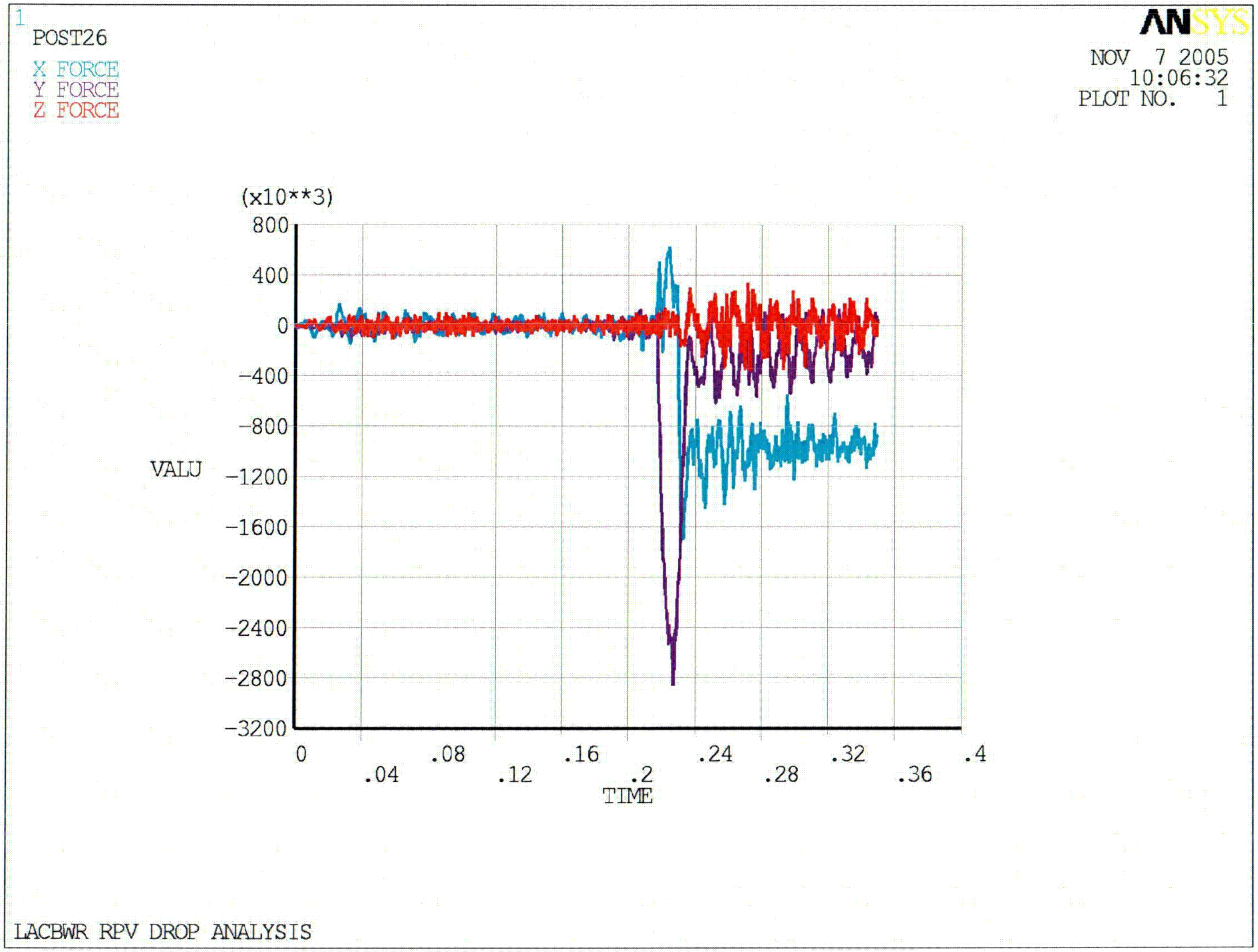


Figure 2-12
Time-History Plot of the Total Forces During 1-ft Inclined Drop in the 1¼ " Supplemental Shield Plates

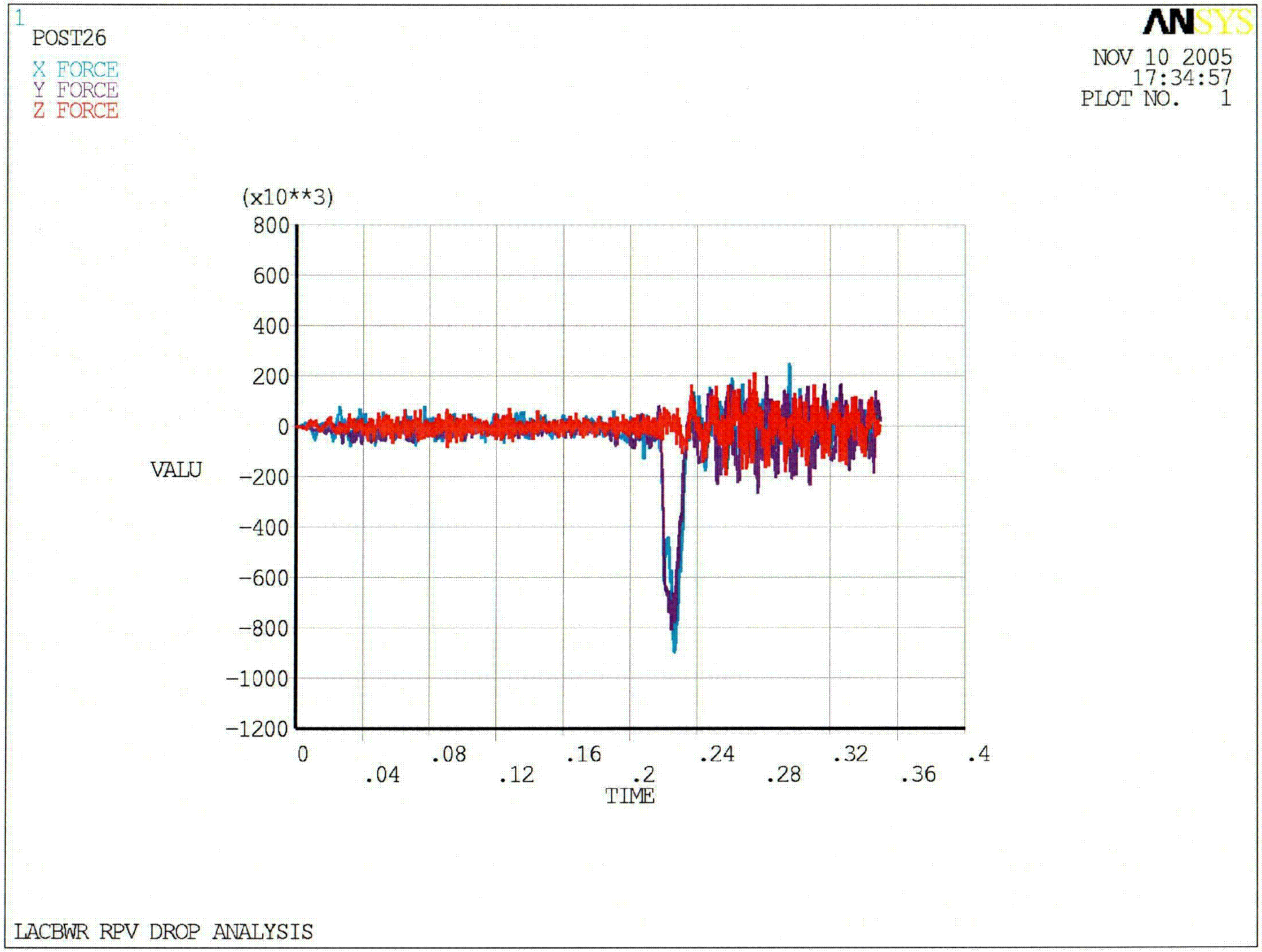


Figure 2-13
Time-History Plot of the Total Forces During 1-ft Inclined Drop in the 1¾ " Supplemental Shield Plates

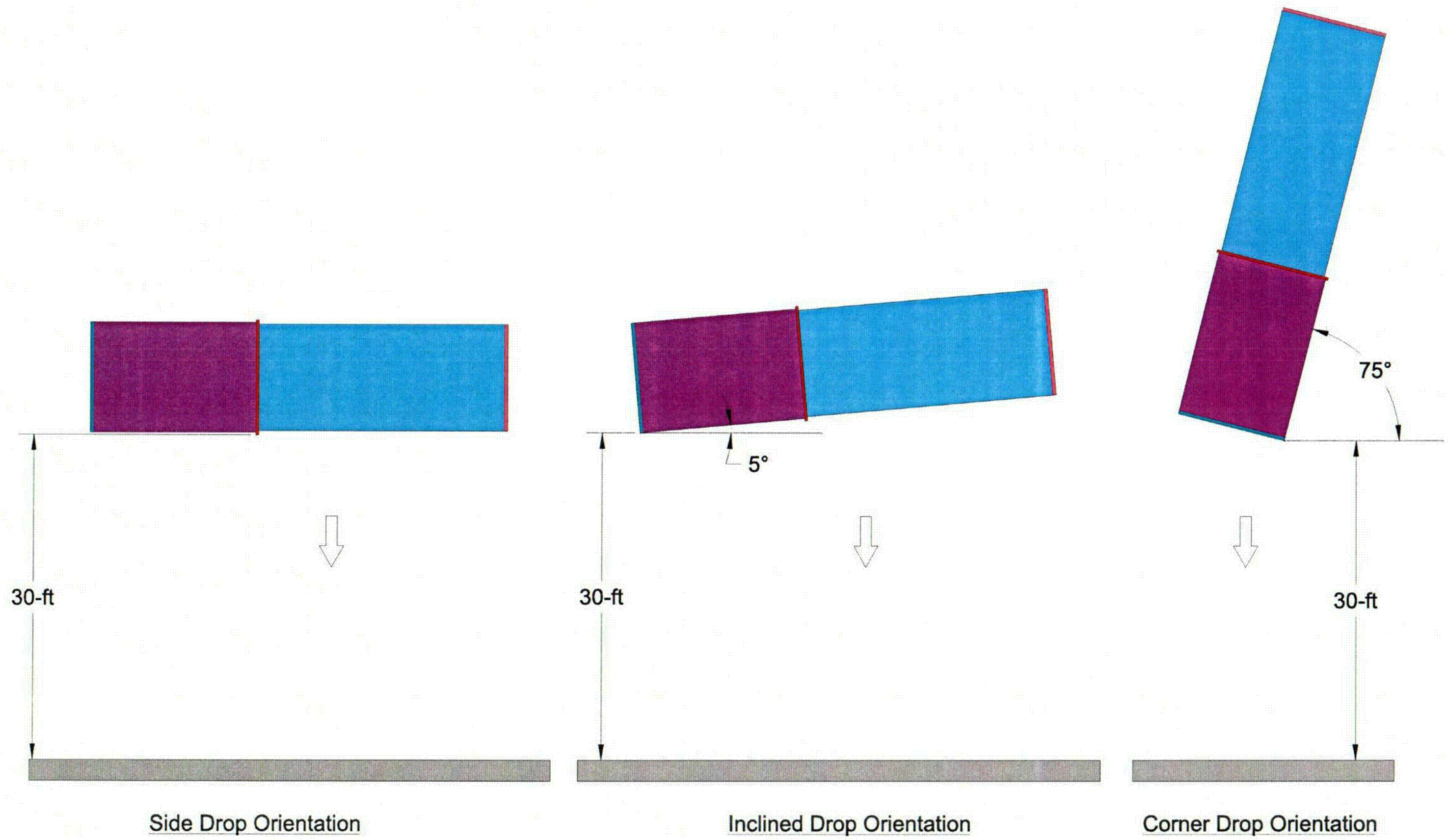


Figure 2-14
Package Orientations Analyzed for the Hypothetical Drop Test Simulation

C14

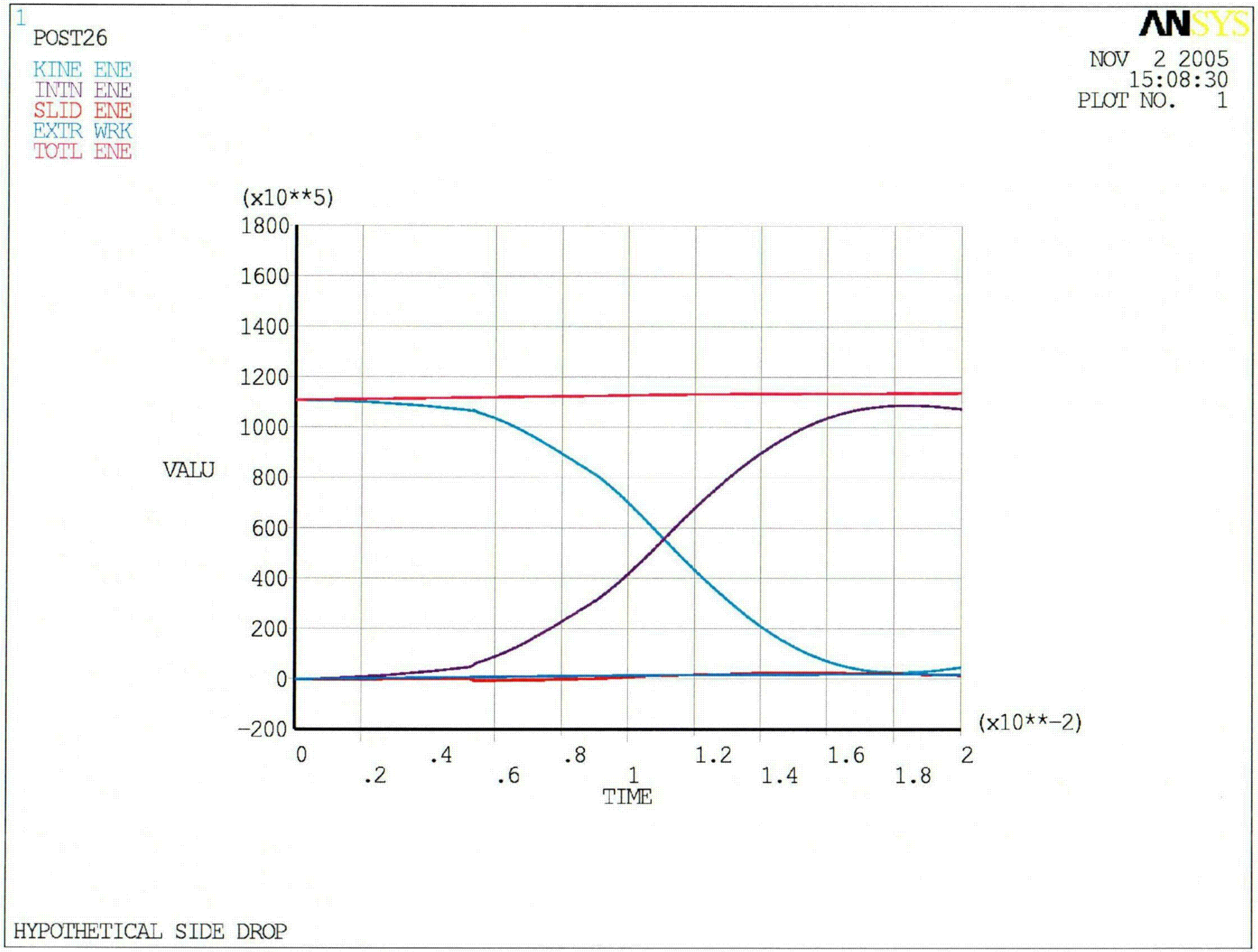


Figure 2-15
Time-History Plot of Various Quantities – 30-ft Side Drop

C15

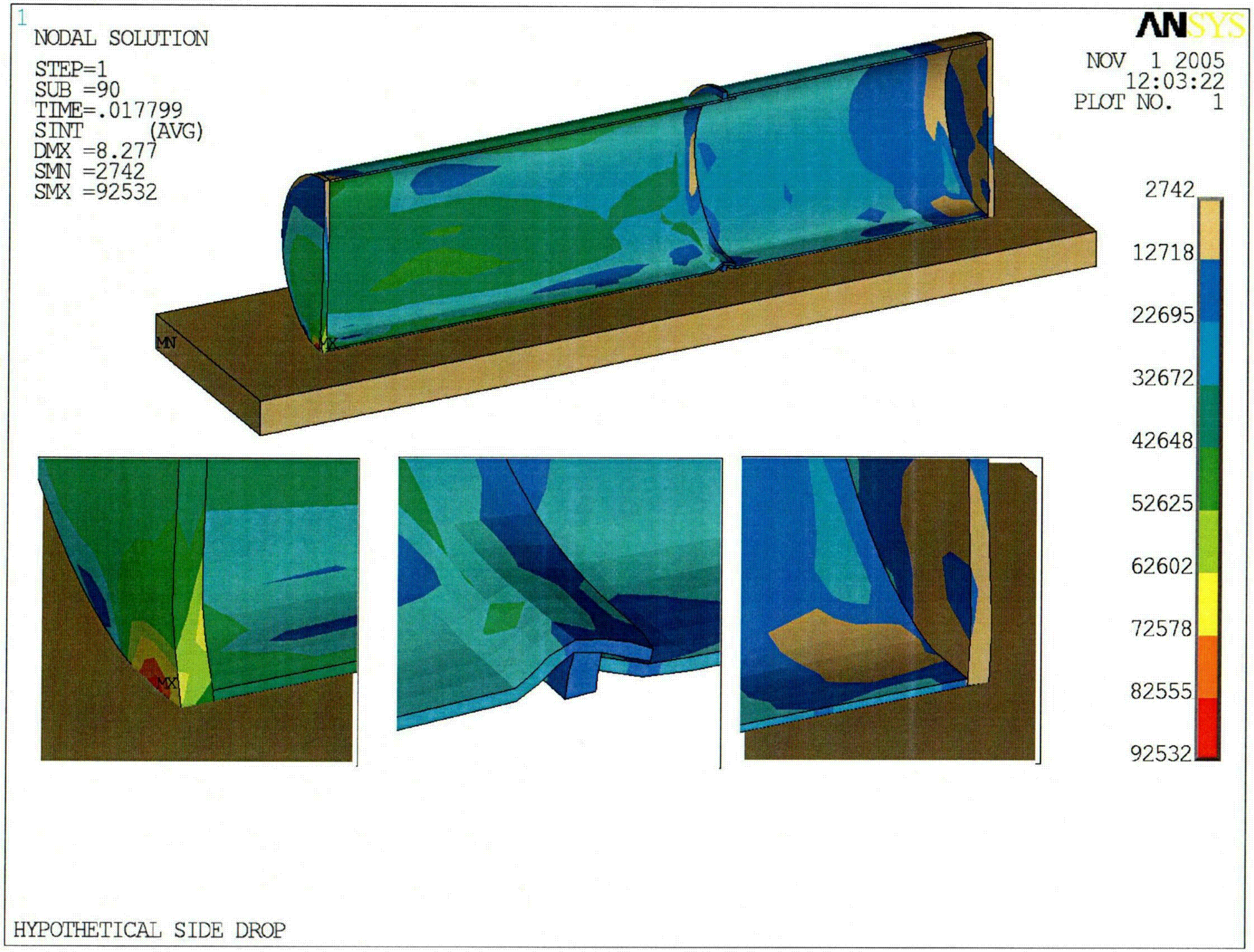


Figure 2-16
Stress Intensity Contour Plot of the Maximum S.I. – 30-ft Side Drop

C16

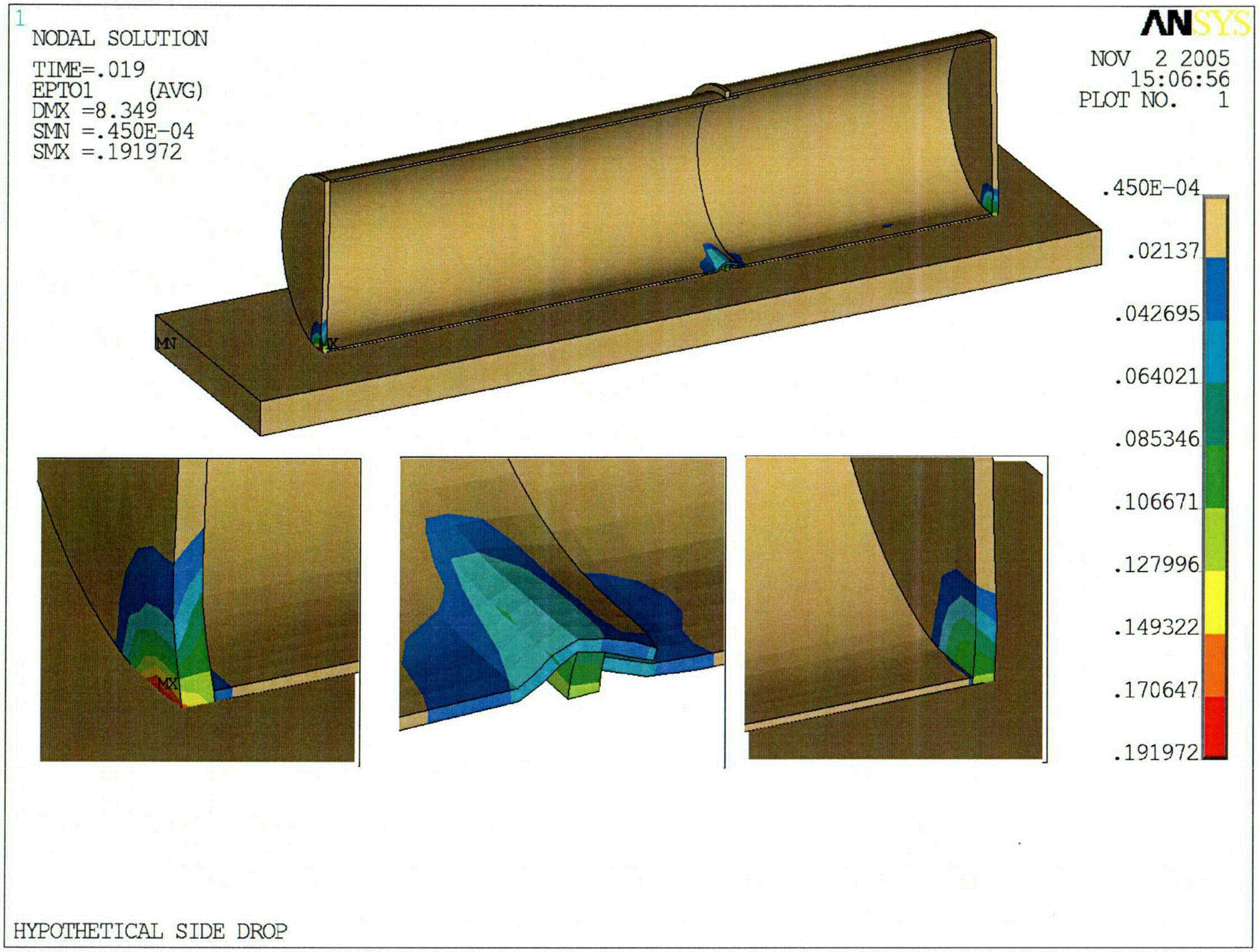


Figure 2-17
Stress Intensity Contour Plot of the Maximum Tensile Strain – 30-ft Side Drop

C17

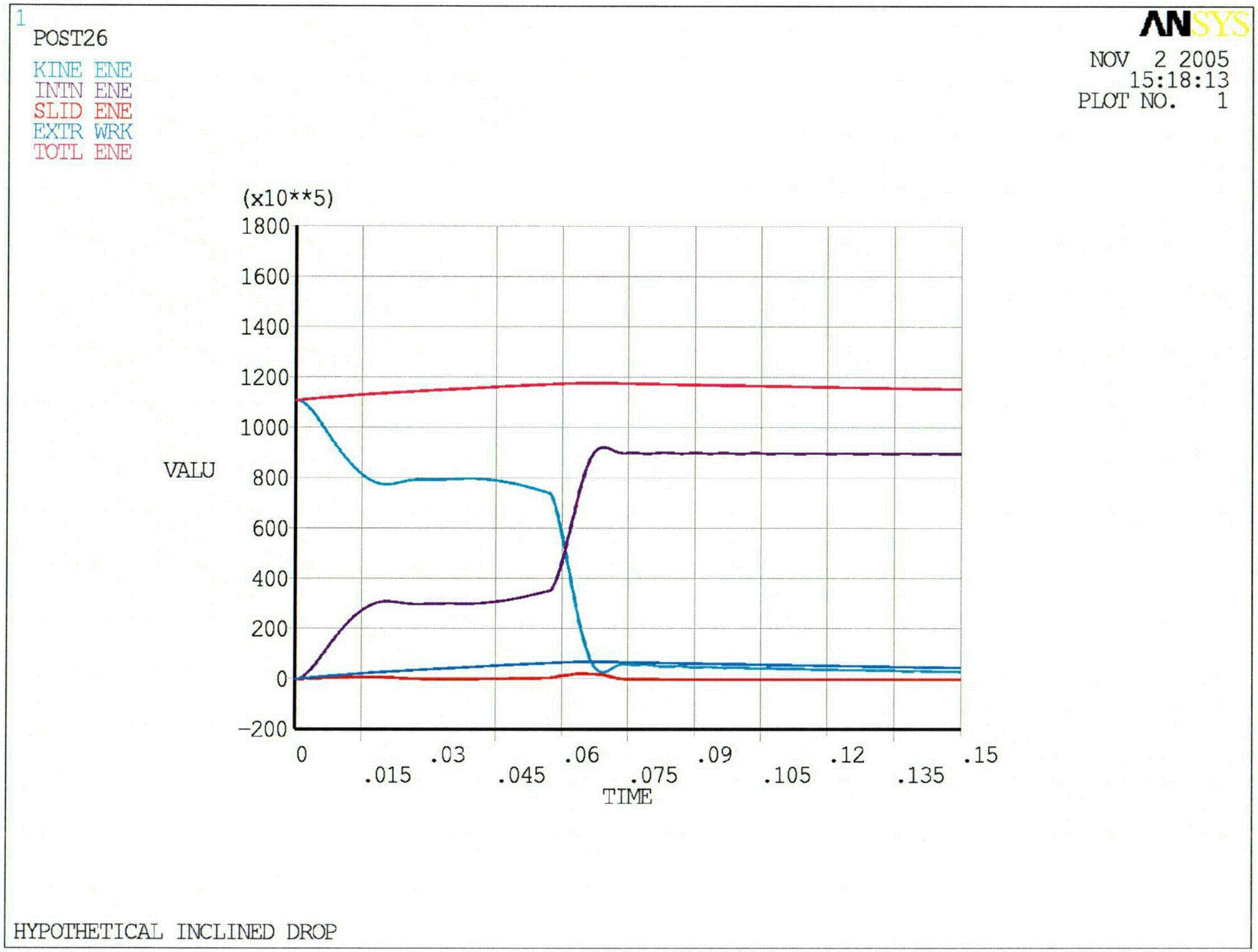


Figure 2-18
Time-History Plot of Various Quantities – 30-ft Inclined Drop

C190

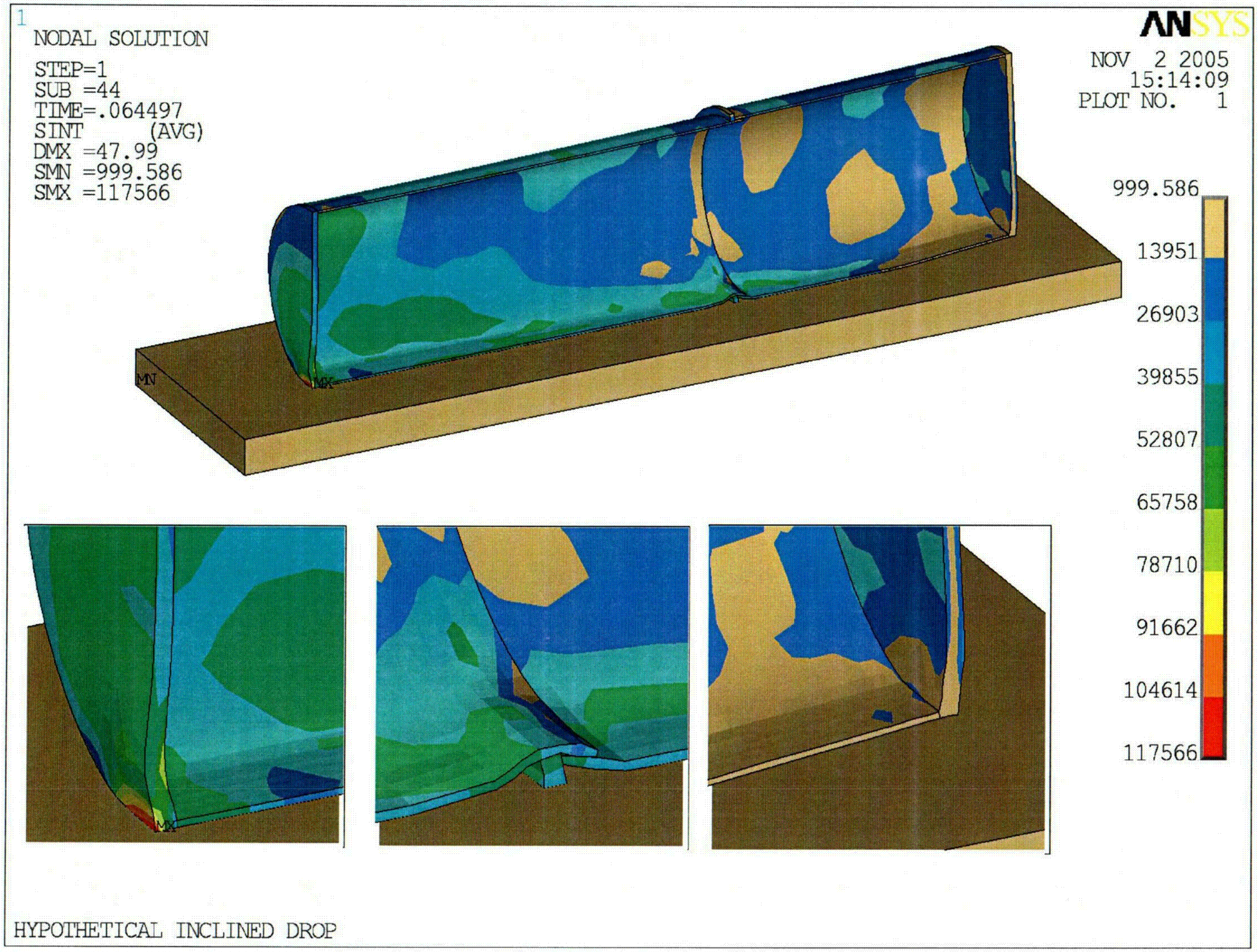


Figure 2-19
Stress Intensity Contour Plot of the Maximum S.I. – 30-ft Inclined Drop

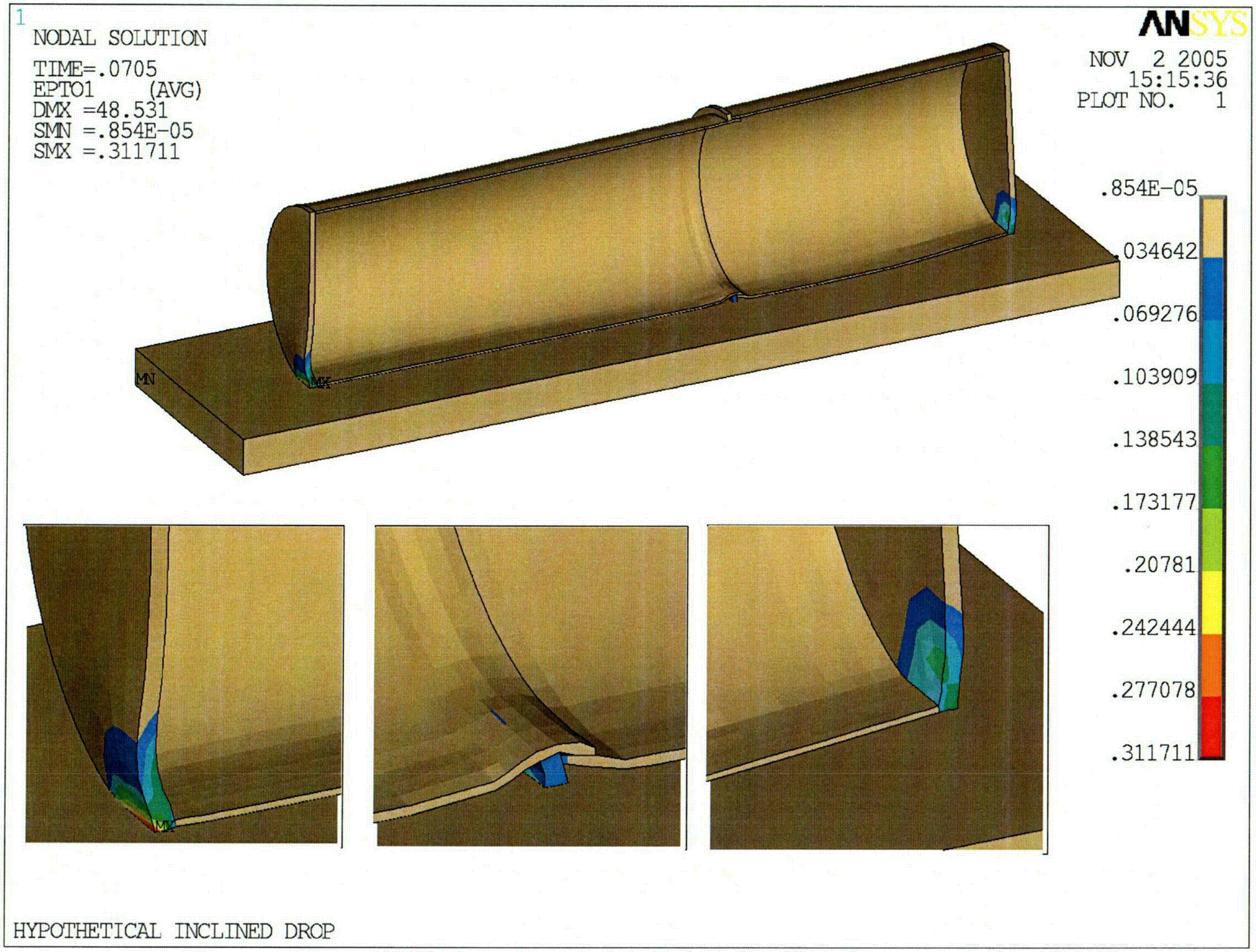


Figure 2-20
Stress Intensity Contour Plot of the Maximum Tensile Strain – 30-ft Inclined Drop

C20

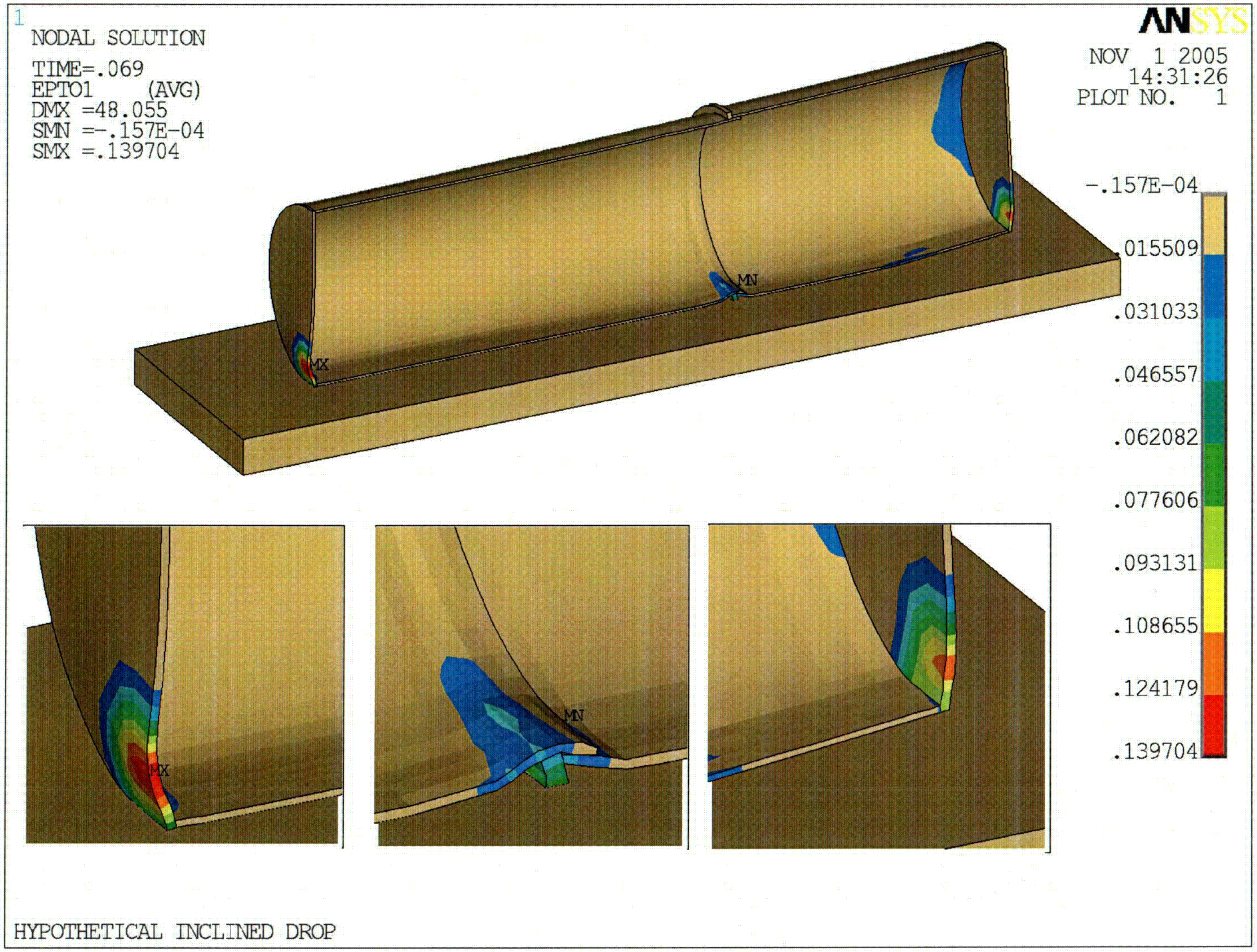


Figure 2-21
Stress Intensity Contour Plot of the Maximum Weld Tensile Strain – 30-ft Inclined Drop

C21

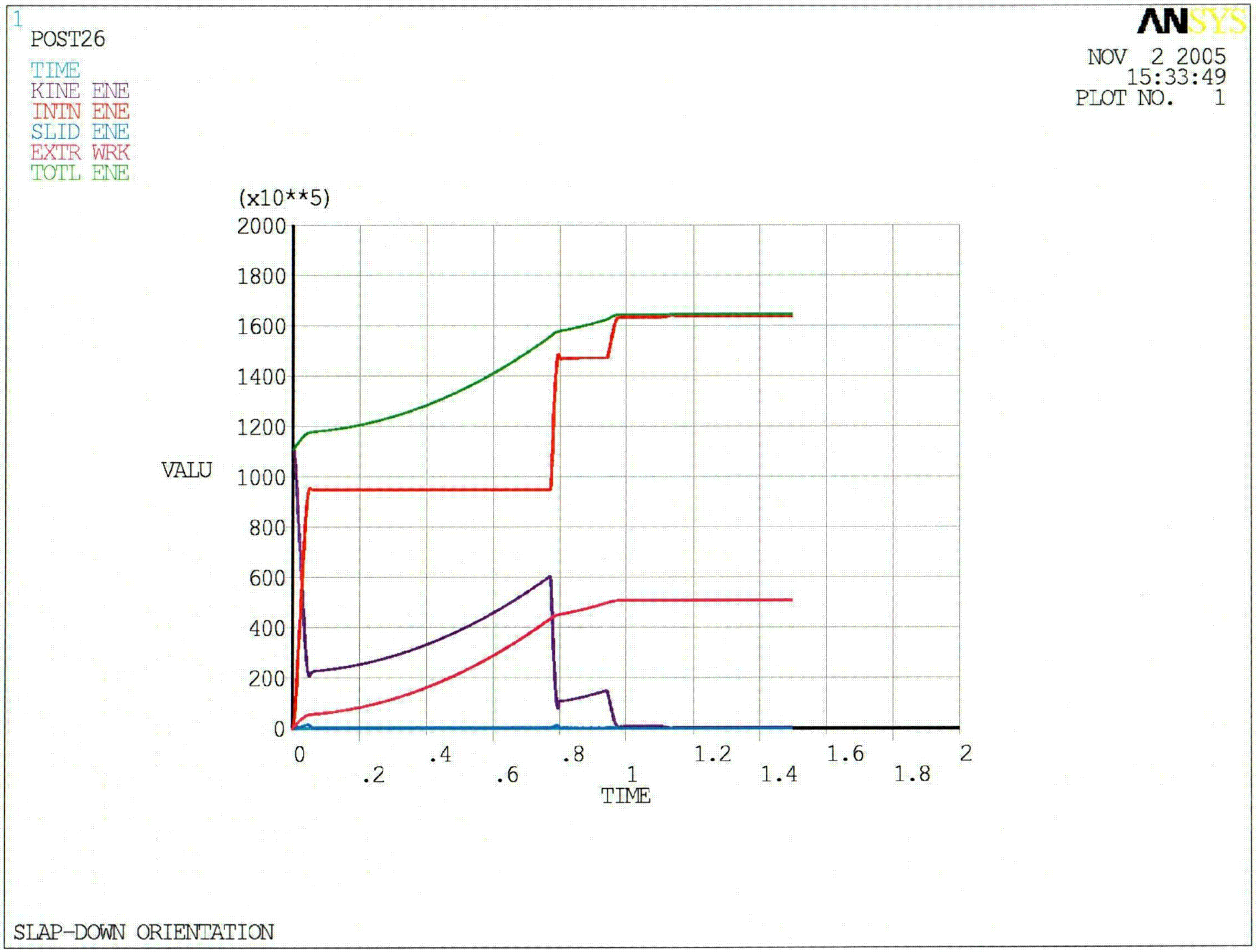


Figure 2-22
Time-History Plot of Various Quantities – 30-ft Corner Drop

C22

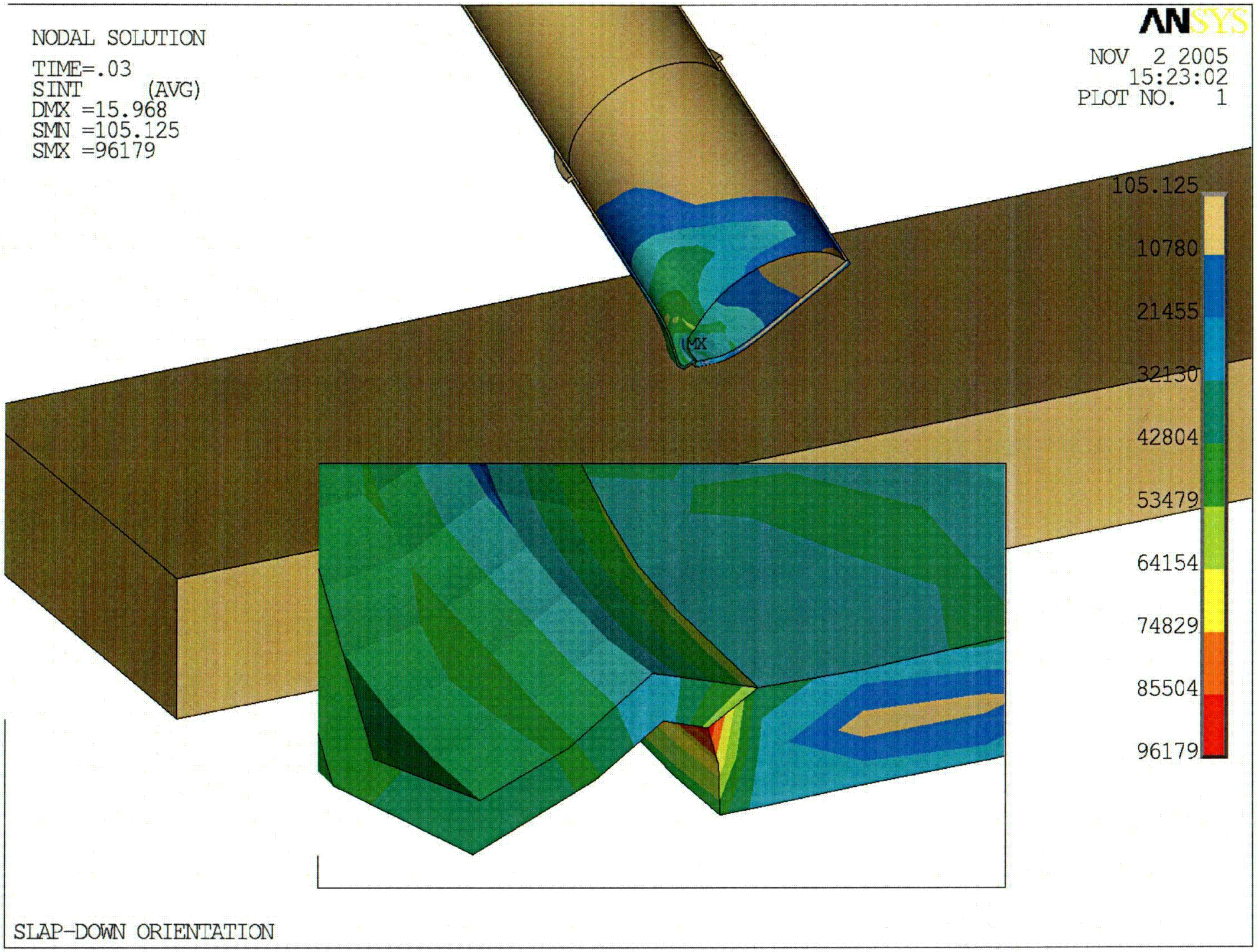


Figure 2-23
Stress Intensity Contour Plot of the Maximum S.I. – 30-ft Corner Drop

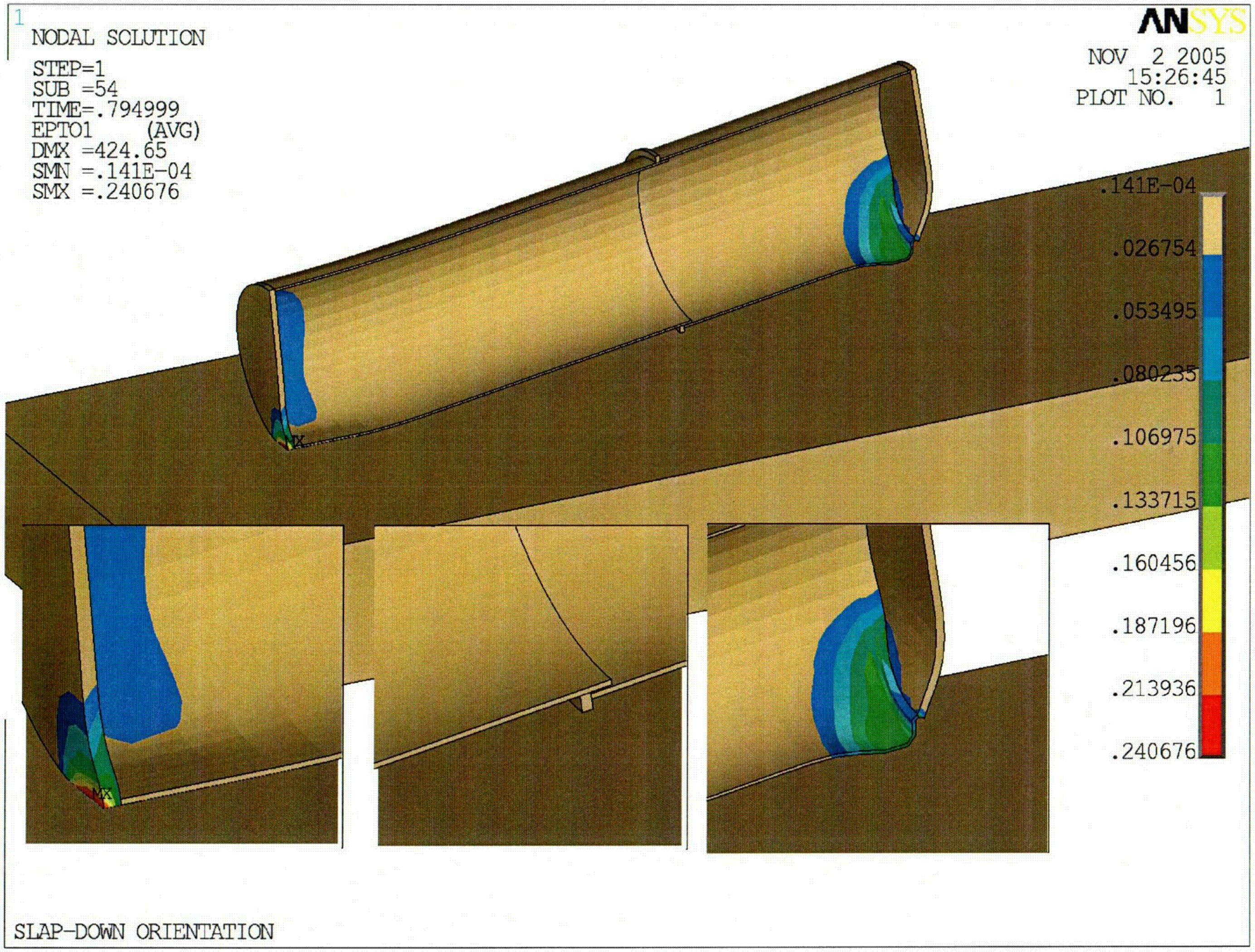


Figure 2-24

Stress Intensity Contour Plot of the Maximum Tensile Strain – 30-ft Corner Drop

C24

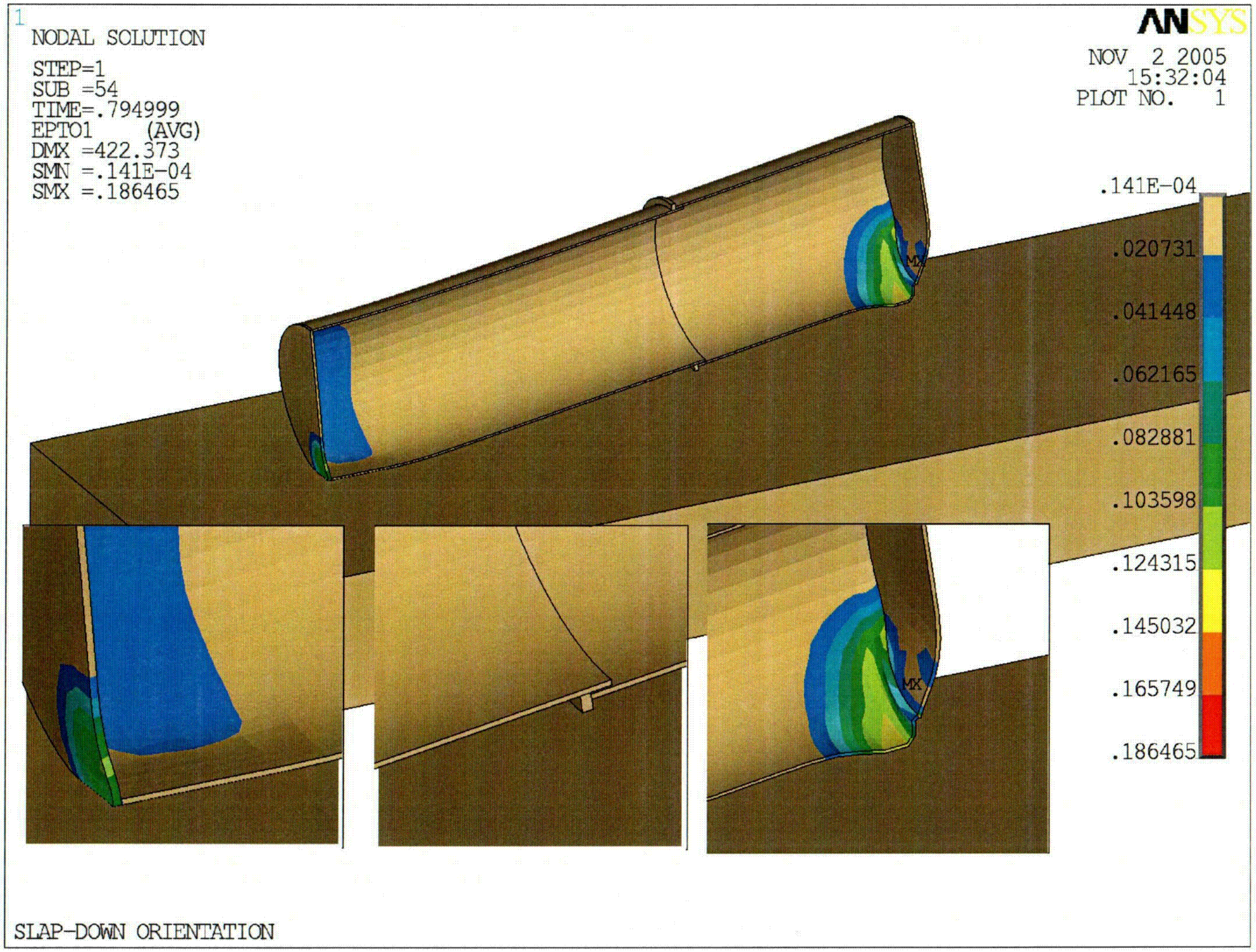


Figure 2-25
Stress Intensity Contour Plot of the Maximum Weld Tensile Strain – 30-ft Corner Drop

C25

Attachment 5

Chapter 4, rev.1

4.0 CONTAINMENT

This Chapter identifies the containment requirements for the La Cross Boiling Water Reactor (LACBWR) RPV package (RPVP). The package is shown on drawings in Appendix 1.3. Several factors enhance the capability of the package to minimize dispersal of the radioactive contents to less than the limits for both the Normal Conditions of Transport and Hypothetical Accident Conditions specified in 10 CFR 71. These are discussed in the sections below.

The LACBWR RPV package has no valves, pressure relief valves, or closure devices that could be operated intentionally or unintentionally.

There are several aspects of the LACBWR RPV package that enhance its containment capabilities:

- The total source term in the radioactive contents is 1.01×10^4 Ci (approximately 410 A_2 values), but only 1.6 Ci (3.43 A_2 values) is loose contamination and therefore dispersible.
- The containment boundary for the LACBWR RPV package consists of an all-welded containment shell described in 4.1.1 below. In addition, there are multiple barriers between the radioactive contents and the environment that provide structural integrity for the package and minimize potential releases of the radioactive contents:
  - The cavity of the RPV will be filled with a low density cellular concrete. This concrete will bind with the loose contents and minimize the likelihood for their dispersal in the event of a breach in containment.
  - The former RPV shell provides structural protection for the low density concrete against damage in potential drop conditions, which minimizes the dispersible contents that might otherwise be created. The RPV will also act as a barrier to minimize low density concrete and radioactive contents from escaping through breaches in the containment shell. In addition, the RPV provides backing for the annulus concrete and for the containment shell, and reduces potential damage to them in drop scenarios.
  - Medium density cellular concrete is poured into the annulus between the RPV shell and the containment shell. This annulus concrete will act as another barrier to dispersal of low density concrete and radioactive contents in the event of a breach in the containment shell.

- The containment shell itself. The containment shell is a unitized structure that is completely welded closed. There are no mechanical closures, gaskets, valves, or other similar types of penetrations.

It is shown in this Chapter that the containment requirements in 10 CFR 71.51(a)(1) for Normal Conditions of Transport, and in 10 CFR 71.51(a)(2) for Hypothetical Accident Conditions are met by the LACBWR RPV package. Periodic and pre-shipment leak test criteria are not applicable as a measure of containment integrity, since there are no seams or closures to test. Instead, it is shown that, for Normal Conditions of Transport, the maximum permitted leak rate in 10 CFR 71.51(a)(1) cannot be exceeded because the welded-closed structure of the containment shell remains intact and there are no leak paths for radioactive contents. For Hypothetical Accident Conditions, it is shown that the maximum permitted leak rate in 10 CFR 71.51(a)(2) cannot be exceeded because of the small quantity of dispersible contents and the multiple barriers to dispersal. See Figure 4-1.

The package contains no explosive mixtures or potential aerosol particulates that could be considered radiological hazards.

4.1 DESCRIPTION OF CONTAINMENT SYSTEM

4.1.1 Containment Boundary

The containment boundary for the LACBWR RPV package is shown on drawings in Appendix 1.3. The containment boundary is called the "RPV canister" on drawings in Appendix 1.3. The containment shell is a completely welded-shut enclosure covering the RPV. It consists of two subassemblies, the lower and the upper, and is constructed of ASTM A516 Gr 70 steel plate rolled to required dimensions and joined with full-penetration welds. The lower subassembly is a cylindrical 1 ½" thick shell that surrounds the lower two-thirds of the RPV. The bottom end of the lower subassembly is closed with a 4" thick plate welded to the barrel portion. A welding ring is welded to the outside circumference approximately 6" from the open end.

The upper subassembly also consists of 1 ½"-thick ASTM A516 Gr 70 steel plate rolled to required dimensions and joined with full-penetration welds, and a 4" thick end plate. It encloses the upper one-third of the RPV. The containment shell is assembled by placing the RPV in the lower subassembly, then placing the upper subassembly over the open end and welding it to the welding ring on the lower subassembly on the lower subassembly.

Containment Penetrations

The only penetrations in the containment boundary consist of ports used for injection of the concrete into the annulus. There will be at least four injection ports in the upper assembly; two on the end and two on the sides. (More fill ports may be added if needed at the time of filling the annulus, up to a maximum of six.) Each port will be a 3 ½" diameter opening in the shell. After the annulus has been filled with concrete, and prior to transport, the openings will be closed and welded shut using a plug assembly in each opening as shown on drawings in Appendix 1.3 in each opening. Each plug assembly will be welded to the shell with a fillet weld as shown on drawing in Appendix 1.3.

Closure

There is no access to the containment cavity, and consequently no closure device on the LACBWR RPVP. The outer containment shell is sealed by being completely welded closed.

The LACBWR RPV package is not vented.

4.1.2 Special Requirements for Plutonium

Not applicable to the LACBWR RPVP.

4.2 GENERAL CONSIDERATIONS

4.2.1 Type A Fissile Packages

Not applicable to the LACBWR RPVP.

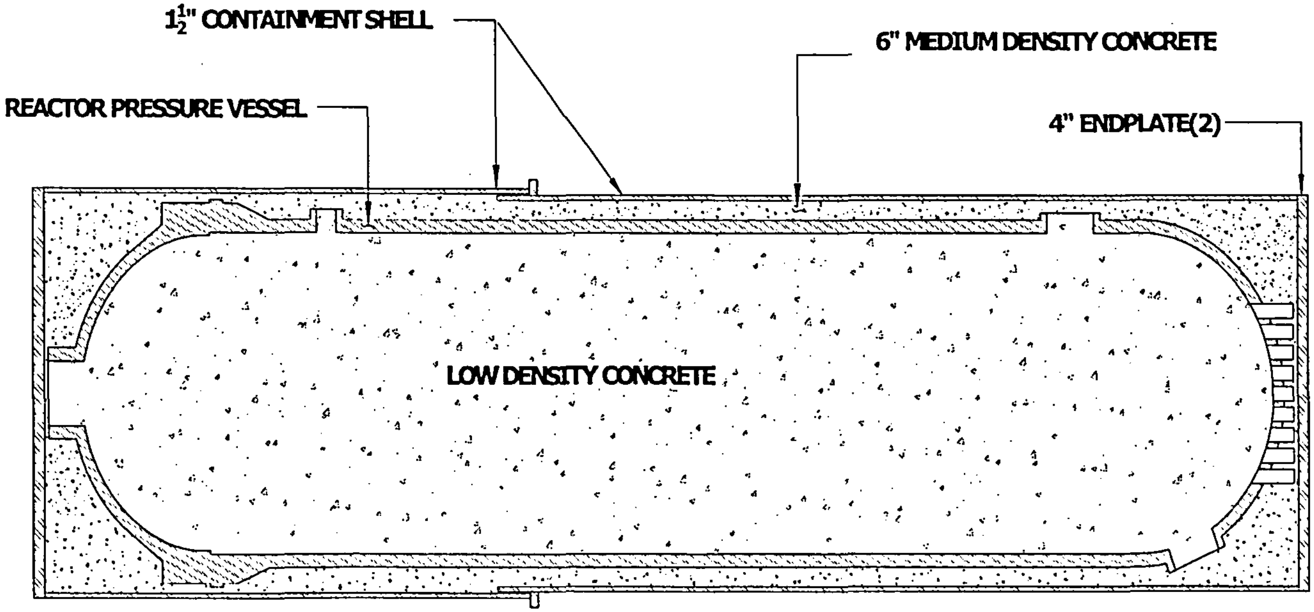
4.2.2 Type B Packages

The contents of the LACBWR RPV package contain approximately 10,100 Ci of radioactivity, of which only 1.61 Ci (3.43 x A_2) will be loose, potentially dispersible materials. The remainder of the radioactive contents is activated materials in the former core region. In sections 4.3 and 4.4 below, it is shown that the rugged construction of the package, the multiple layers of protection, and immobilization by low density concrete added to the internals of the RPV, preclude leakage of the small quantity of dispersible materials from containment during both Normal Condition of Transport and Hypothetical Accident Conditions. The package remains intact and will perform its intended safety function under the tests and conditions in 10 CFR 71.

4.2.3 Hydrogen Generation

The materials in the RPV package are the reactor vessel and internals, the low and medium density cellular concrete, and the steel packaging (canister). The reactor vessel is ferritic steel with an internal stainless steel cladding. The reactor internals are primarily stainless steel. The canister is ferritic steel. The alkaline concrete passivates ferritic steel and does not react with the protective oxide layer of the stainless steel. Thus, there are no materials interactions producing hydrogen in the RPVP.

The RADCALC computer program was used to calculate hydrogen generation in the LACBWR RPVP. RADCALC calculates the radiolytic generation of hydrogen in a waste matrix in a radioactive material package. In the LACBWR RPVP RADCALC calculation, all metal, i.e., the RPV, internals, and canister are considered the container since hydrogen is not produced by radiolysis from these materials. The LDCC and MDCC are considered the waste containing the entire LACBWR RPVP radionuclide inventory. The licensee shall ensure that the water content of the LDCC and MDCC are comparable to the water content of the concretes used to determine the G values that were used in the RADCALC program to calculate hydrogen generation in the LACBWR RPVC. The program option was selected to determine the time after sealing the package that the hydrogen concentration in any void will reach the 5% limit. The results show a 5% hydrogen concentration is reached approximately 1.3 years after the canister is sealed. Thus, the requirements of NRC Information Notice No. 84-72: CLARIFICATION OF CONDITIONS FOR WASTE SHIPMENTS SUBJECT TO HYDROGEN GAS GENERATION are met. The licensee shall ensure that transportation is completed less than one year after the canister is sealed.



Multiple Layers of Protection from Dispersal of Contents for the LACBWR RPV

Figure 4-1

4.3 CONTAINMENT UNDER NORMAL CONDITIONS OF TRANSPORT (TYPE B PACKAGES)

The maximum permitted leakage rate under Normal Conditions of Transport cannot be exceeded during transport of the LACBWR RPV package. 10 CFR 71.51(a)(1) states that the containment requirements for Normal Conditions of Transport are:

> *"... there would be no loss or dispersal of radioactive contents--as demonstrated to a sensitivity of 10^{-6} A_2 per hour, no significant increase in external surface radiation levels, and no substantial reduction in the effectiveness of the packaging"*

As discussed in 4.1.1, the LACBWR RPV package containment is a totally welded-shut enclosure, without mechanical seals or a bolted closure. Integrity of containment welds are verified by non-destructive examination during fabrication as described in Chapter 8. Containment is shown in Chapter 2 of this application to remain intact during Normal Conditions of Transport, precluding release of the radioactive contents. Therefore, there are no events that can breach the containment boundary and lead to the dispersal of radioactive contents or a significant reduction in effectiveness of the packaging. Thus, the LACBWR RPV package satisfies the containment requirements of 10 CFR 71.51 under Normal Conditions of Transport.

4.4 CONTAINMENT UNDER HYPOTHETICAL ACCIDENT CONDITIONS

In this section it is shown that the maximum permitted leakage rate under Hypothetical Accident Conditions cannot be exceeded during transport of the LACBWR RPV package. 10 CFR 71.51(a)(2) states that the containment requirements for Hypothetical Accident Conditions are:

> *"...there would be no escape of krypton-85 exceeding 10 A_2 in 1 week, no escape of other radioactive material exceeding a total amount A_2 in 1 week, and no external radiation dose rate exceeding 10 mSv/h (1 rem/h) at 1 m (40 in) from the external surface of the package".*

As discussed in Chapter 1, the LACBWR RPV package contains only 3.4 A_2 quantities of dispersible radioactive contents. The remaining quantity of radioactive contents consist of activated hardware primarily in the former core region of the RPV. In this section it is shown that the low quantity of dispersible radioactive materials, the structural strength of the LACBWR RPV package, and the multiple barriers that protect the contents, prevent dispersal of contents greater than the maximum permitted in 10 CFR 71.51(a)(2). As discussed in 4.0, for the purposes of containment considerations, features of the LACBWR RPV package that are barriers to excessive dispersal of the radioactive contents include:

- the low density cellular concrete inside the RPV
- the RPV shell
- the medium density cellular concrete in the annulus between the RPV shell and the containment shell
- the containment shell itself

The analysis of the LACBWR RPV package in Chapter 2 has shown that the 30 ft drop Hypothetical Accident Condition could result in a partial breach in a containment shell weld and crushing of the annulus concrete in the area of impact. Also, a localized breach of the containment shell and annulus could occur due to the puncture accident. However, with either scenario, the breach of the containment shell and annulus concrete would be small and localized to the immediate area of impact, rather than being a generalized or extensive loss of the containment shell. Consequently, the quantity of radioactive contents dispersed subsequent to the Hypothetical Accident Condition would be limited to the volume that could pass through the crushed region of the annulus concrete and leak out of the localized opening in the containment shell. In addition, the analysis in Chapter 2 shows that neither the 30 ft drop nor the puncture test would cause damage to the RPV shell or the low density concrete inside the RPV, and that these would remain intact as barriers to dispersal of radioactive contents during Hypothetical Accident Conditions.

Since the quantity of dispersible activity in the RPV is small, the specific activity in the low density concrete will be small. Therefore, a large volume of low density concrete would have to be dispersed to cause a large release of radioactive materials. However, since the low density concrete in the RPV remains monolithic under Hypothetical Accident conditions, it is unlikely that a substantial volume of concrete can escape the RPV shell and through the small breaches in the annulus concrete and containment shell. The following bounding calculation illustrates this:

Given:

Volume of low density cellular concrete: 1690 ft^3 (Ref. 4-1)

Dispersible radioactive contents: 1.61 Ci (Chapter 1)

1.61 Ci = 3.43 A_2 → 0.46 Ci/A_2 (Chapter 1)

Assume that dispersible contents mixes with 25% of the low density concrete inside the RPV shell. [The assumed volume of contaminated LDCC would result from a layer approximately 0.45" in depth over the contaminated surface. The LDCC will be a very fluid mixture as injected so a mixing layer of only ½" is conservative.]

then:

1690 ft^3 x 25% = 422.5 ft^3

$$\frac{1.61\ Ci}{422.5\ ft^3} = 0.0038\frac{Ci}{ft^3}$$

The structural analysis in Chapter 2 shows there will be high stresses in the weld between side wall and the endplate at the impact point but that no failure of the containment is predicted. Nonetheless, we conservatively assume that there is a failure extending 10° circumferentially in both directions from the impact point; this is approximately 5% of the circumference of the endplate. The structural analysis also shows that the stresses on the RPV and the LDCC will be low, thus there will be no failure of the RPV and the LDCC will remain intact. However, if some of the LDCC were to disperse, a mixture of contaminated and uncontaminated LDCC is most likely. In spite of this, for the bounding calculation, assume that 5% of the low density cellular concrete in the RPV is dispersed during Hypothetical Accident Conditions, and that all of the concrete that is dispersed is from the 25% that contains the concrete/radioactive contents mixture:

Thus,

1690 ft^3 x 5% = 84.5 ft^3

$$84.5\ ft^3\ x\ 0.0038\frac{Ci}{ft^3} = 0.32\ Ci \qquad \frac{0.32\ Ci}{0.46\dfrac{Ci}{A_2}} = 0.7\ A_2 \quad \rightarrow$$

The bounding calculation shows that there would be a dispersal of only 0.7 A_2 quantity of radioactive materials. This is less than the limit of 10 times an A_2 quantity per week for Kr-85 and 1 A_2 per week for all other isotopes, as specified in 10 CFR 71.51(a)(2).

While we expect that none of the LDCC would disperse during Hypothetical Accident Conditions, there could be a localized breach of the containment boundary that would allow some LDCC to disperse. However, the amount of radioactive material leaked would be limited by the multiple barriers to dispersal, and the amount that could pass through small breaches in the medium density concrete and containment shell. As shown by the bounding calculation, under very conservative assumptions that 5% of the volume of low density concrete in the RPV (84.5 ft^3) is dispersed through a localized breach in the containment shell, the package meets the containment requirements of 10 CFR 71.

4.5 REFERENCES

4-1 ST-520. Calculation of internal volume of RPV shell.

Attachment 6
Chapter 5, rev.1

5.0 SHIELDING EVALUATION

5.1 DESCRIPTION OF DESIGN FEATURES

The LACBWR RPVP consists of a steel containment vessel (canister) and annular concrete layer that provides the necessary shielding for it to be shipped as a single use package. (Refer to Section 1.2.2 for package contents.) Tests and analysis performed under Chapters 2.0 and 3.0 have demonstrated the ability of the containment vessel to maintain its integrity under normal conditions of transport. Prior to the shipment, radiation readings will be taken to assure compliance with 10 CFR 71.47.

The package shielding is sufficient to satisfy the dose rate limit of 10 CFR 71.51(a) (2) which states that any shielding loss resulting from the hypothetical accident will not increase the external dose rate to more than 1000 mrem/hr at one meter from the external surface of the package.

5.1.1 Shielding Design Features

The LACBWR reactor vessel packaging consists of a 1.5" steel canister surrounding the reactor pressure vessel, end plates of 4" steel plate, and the annulus between the vessel and the canister filled with 120 lbs/ft^3 concrete, as shown in the drawings in Appendix 1.3. At the region of highest dose rate, i.e., the 10 foot section of the RPV centered on the core midplane, the annulus is 5.75". In addition, a cylindrical shield of 1 ¼ " steel is welded to the canister, extending 4.5' on each side of the core midplane. Further, two additional (cylindrical and supplemental) shields, 1 ¾" steel, are welded to each side of the cylindrical shield extending 4 feet on each side of the core midplane and covering a 120° arc of each side of the shield. Under HAC, these shields are assumed to detach from the canister.

5.1.2 Maximum Radiation Levels

Table 5-1 gives Normal Conditions of Transport (NCT) and Hypothetical Accident Conditions (HAC) dose rates. Maximum allowable dose rates given in 10 CFR 71 are shown for comparison. The LACBWR RPVP is shipped exclusive use.

Table 5-1
Summary of Maximum Radiation Levels

	Total Dose Rate (mrem/hr)					
	Package Surface		1 m from Surface		2 m from Package	
Condition	Side	Top/Bottom	Side	Top/Bottom	Side	Top/Bottom
NCT						
Calculated	126	15.8	N.A.	N.A.	7.3	3.6
Allowable	200	200	N.A.	N.A.	10.0	10.0
HAC						
Calculated	N.A.	N.A.	541	6.8	N.A.	N.A.
Allowable	N.A.	N.A.	1000	1000	N.A.	N.A.

5.2 SOURCE SPECIFICATION

The LACBWR RPVP will transport the intact LACBWR reactor vessel with all internals. The radionuclide contents are described in Section 1.2.2. To determine NCT and HAC dose rates, results of gamma surveys on the exterior of the vessel were used as the basis for the source term in the shielding models. Surveys were performed using a shielded probe inserted into the annulus between the reactor vessel and the external thermal shield. Measurements of the vessel were made with the unshielded face of the probe facing the vessel ("Detector Toward RX Vessel") and facing the thermal shield ("Detector Toward Thermal Shield"). Surveys results are provided in Appendix 5.5.2. The maximum contact dose rate on the exterior of the vessel, near the core midplane was 13 rem/hr on 6/1/2005. The maximum contact dose rate on the bottom of the vessel was 1.2 rem/hr on 11/9/2005. From the radionuclides in the vessel, as described in 1.2.2., the external dose rate is due to Co-60. Using the half-life of Co-60, the expected maximum vessel contact dose rate at the time of shipping (6/1/2007) is 10 rem/hr at the core midplane and 0.978 rem/hr at the bottom. For determination of NCT and HAC dose rates, the determination of the maximum external dose rates assumes a uniform source in the core region of the vessel such that the contact dose rate on the side of the vessel at the core midplane is 10 rem/hr and 0.978 rem/hr on the bottom of the vessel..

5.2.1 Gamma Source

The assumed gamma source to give a core midplane reading of 10 rem/hr was used, i.e., 10,653 Ci of ^{60}Co. The gamma decay source strength, in photons/sec and MeV/sec, as a function of gamma energy is shown below. This activity was uniformly distributed over the internals in the core region.

Photon Energy	Activity	Activity
MeV	Photons/sec	MeV/sec
0.6938	6.430E+10	4.46E+10
1.1732	3.942E+14	4.62E+14
1.3325	3.942E+14	5.25E+14
Totals	**7.884E+14**	**9.87E+14**

The assumed gamma source to give a bottom dose rate of 0.978 rem/hr was used in calculating the package axial dose rates with the same gamma energies as shown above but equivalent to ~ 666 Ci of Co-60.

5.2.2 Neutron Sources

There are no sources of neutron radiation in the radioactive materials to be carried in the LACBWR RPVP.

5.3 MODEL SPECIFICATION

External dose rates are modeled using MicroShield.

5.3.1 Configuration of Source and Shielding

5.3.1.1 Source

The source used in calculating radial dose rates is modeled as a right circular cylinder with a diameter of the I.D. of the reactor vessel (99") and a height equal to that of the core region, i.e., 10'. The source is modeled as iron with a density of 2 g/cc. The activity of the source is assumed to be uniformly distributed over the core and of a magnitude such that the calculated vessel contact dose rate at the core midplane (axial value of 5') is equal to the measured value (decayed to the shipping date) of 10 rem/hr. The measurements of the vessel show that nearly all the activity in the vessel is found in the 10' core region and, as expected, is symmetrical about the core midplane. This source configuration is conservative in that the calculated vessel contact dose rate at the bottom of the core region (axial value of 0') is 5 rem/hr while the measured value is less than 1 rem/hr.

Regions of the vessel above and below the core are not considered when calculating maximum radial dose rates since measured vessel contact dose rates drop off rapidly, e.g., at 3 feet below the bottom of the core measured dose rates are 200-300 mrem/hr and at 10' above the top of the core measured values are less than 50 mrem/hr. See Appendix 5.5.2 for the gamma surveys and vessel drawing.

The vessel internals are intact. No dismantlement has taken place with the exception of removal of the used fuel. Prior to moving the vessel, the vessel interior is filled with low density concrete. This concrete will hold the vessel internals in place but is not included in the shielding models. The source configuration does not change under the NCT or HAC free drop nor does the position of the source within the reactor vessel change.

The source used in calculating axial dose rates is assumed to be a sphere with a diameter of the I.D. of the reactor vessel (99") to represent the hemispherical bottom of the vessel. From radiation surveys, the bottom of the vessel is known to have higher dose rates than the vessel head (bottom 1.2 rem/hr, head 60 mrem/hr), so that in determining maximum axial dose rates, only the vessel bottom was modeled. The source is modeled as iron with a density of 2 g/cc. The activity of the source is assumed to be uniformly distributed over the internals and of a magnitude such that the calculated vessel contact dose rate at the bottom of the vessel is equal to the measured value (decayed to the shipping date) of 0.978 rem/hr.

5.3.1.2 Radial Shielding

The radial shields around the source are modeled as cylindrical shells. The shields are identified in Table 5.2 and shown on Drawing in Appendix 1.3).

Table 5-2
Radial Shields

Shield	Material	Thickness (in)	Density (g/cc)
Vessel	Iron	4	7.86
Annular concrete	Concrete	5.75	1.9
Canister	Iron	1.5	7.86
Cylindrical Shield	Iron	1.25	7.86
Supplementary Shield	Iron	1.75	7.86

As described in Chapter 2, the shields remain as shown in the drawing (C-068-163041-002) after all the NCT tests. The cylindrical and supplemental shields are assumed to disengage as a result of the HAC free drop. In addition, there is some deformation of the canister at the corner when the free drop orientation is CG over corner but none in the region surrounding the reactor core. There may be some minor cracking of the weld at the cylinder/end plate intersection but this does not affect the shielding effectiveness. The HAC puncture test is assumed to result in a 6" diameter hole perpendicular to the vessel axis through the canister and annular concrete ending at the vessel wall thus removing canister and concrete shields over the 6" hole. The center of the 6" hole is assumed to be positioned on the core midplane. The other HAC tests do not result in further changes to the shielding configuration.

5.3.1.3 Axial Shielding

The axial shields are the vessel and the end plates of the packaging, as shown in Table 5-3. No credit is taken for any concrete between the end of the vessel and the end plates. Under both the NCT and HAC, the end plates stay in position as described in Chapter 2.

Table 5-3
Axial Shields

Shield	Material	Thickness (in)	Density (g/cc)
Vessel	Iron	4	7.86
End plates	Iron	4	7.86

5.3.1.4 Radial Dose Points

Since the maximum contact dose rate on the vessel is at the core midplane, dose points were located on a line perpendicular to the vessel axis intersecting the core midplane. For the NCT tests, the dose points are located at contact with the cylindrical shielding and at 2m from the supplemental shielding to demonstrate compliance with 10 CFR 71.47(b)(1) and 10 CFR 71.47(b)(3), respectively. After the HAC free drop, a dose point is selected at 1 meter from the surface of the canister and, after the puncture test, at 1 meter from the vessel wall in the center of the 6" diameter hole.

5.3.1.5 Axial Dose Points

The maximum contact dose rate at the ends of the vessel is at the center of the bottom hemisphere. Axial dose points were located on the vessel axis in three locations: at contact with the package, at 1m , and at 2m.

5.3.2 Material Properties

Three materials are used as shields in the shielding model: steel, concrete, and air. Steel is modeled as elemental iron with a density of 7.86 g/cc. Concrete, using the MicroShield default composition, is included with a density of 1.9 g/cc (120 lbs/ft^3) (medium density). Air, using the MicroShield default composition, is included with a density of 0.00122 g/cc.

5.4 SHIELDING EVALUATION

5.4.1 Methods

The basic method of evaluating the external dose rates on the package is to create a gamma source, discussed in Sections 5.2 and 5.3, that results in a calculated dose rate equivalent to the measured dose rates on the reactor vessel (decayed to the shipping date). The shielding afforded by the packaging is added and external dose rates are calculated at the locations of the expected maximums. Modifications to the shielding as a result of the NCT and HAC tests are applied and final external dose rates are determined.

The calculational technique is a point kernel integration using the MicroShield computer program (Ref. 5-1) . The basic inputs are the geometry and gamma source. The attenuation and buildup factors built into the program are taken from ANSI/ANS-6.4.3-1991, Gamma-Ray Attenuation Coefficients and Buildup Factors for Engineering Materials.

5.4.2 Input and Output Data

The input and output data are shown in the MicroShield case outputs for each of the calculations, included in Appendix 5.5.3.

5.4.3 Flux-to-Dose-Rate Conversion

MicroShield calculates results in terms of a photon fluence rate at the dose points. The photon fluence is expressed in units of photons/cm^2/second. For conversion to exposure rate, energy absorbsion in air, and dose equivalent, MicroShield uses tabulated values in ICRP 51, Data for Use in Protection Against External Radiation, and interpolates as required. These conversion factors are in Table 5-4.

Table 5-4
Dose Rate and Exposure Rate In Air Per Unit Monoenergenic Fluence Rate (photons/cm²/s)

Photon Energy	Conversion Coefficient		
(MeV)	10^{-12} Gy cm²	mrad/hr	mR/hr
0.01	7.43	2.68E-03	3.06E-03
0.015	3.12	1.12E-03	1.29E-03
0.02	1.68	6.05E-04	6.93E-04
0.03	0.721	2.60E-04	2.97E-04
0.04	0.429	1.54E-04	1.77E-04
0.05	0.323	1.16E-04	1.33E-04
0.06	0.289	1.04E-04	1.19E-04
0.08	0.307	1.11E-04	1.27E-04
0.1	0.371	1.34E-04	1.53E-04
0.15	0.599	2.16E-04	2.47E-04
0.2	0.856	3.08E-04	3.53E-04
0.3	1.38	4.97E-04	5.69E-04
0.4	1.89	6.80E-04	7.79E-04
0.5	2.38	8.57E-04	9.81E-04
0.6	2.84	1.02E-03	1.17E-03
0.8	3.69	1.33E-03	1.52E-03
1	4.47	1.61E-03	1.84E-03
1.5	6.12	2.20E-03	2.52E-03
2	7.5	2.70E-03	3.09E-03
3	9.87	3.55E-03	4.07E-03
4	12	4.32E-03	4.95E-03
5	13.9	5.00E-03	5.73E-03
6	15.8	5.69E-03	6.52E-03
8	19.5	7.02E-03	8.04E-03
10	23.1	8.32E-03	9.53E-03

5.4.4 External Radiation Levels

5.4.4.1 Radial Radiation Levels

Radial Source

As described in 5.3.1, the determination of the radial radiation source, based on the activation isotopic and the external radiation measurements, is completed first (see Appendix 5.5.3 - DOS file: LACBWR radial source.ms6). This model assumes a uniformly distributed cylindrical source surrounded by the 4" steel vessel. The modeled source , at the core midplane on contact with the vessel, has a dose rate of 10 R/hr (decayed to the date of shipment), which matches the radiation survey, and at the bottom of the core the dose rate of 5 R/hr, which conservatively overestimates the decayed measured value of 0.922 R/hr. This source is used in all the subsequent radial dose rate models, both NCT and HAC.

Puncture Test Source

A special source configuration must be modeled in order to evaluate the conditions resulting from the HAC Puncture Test. As a result of the puncture test, there will be 6" diameter hole through the packaging ending at the vessel wall and, since this test follows the free drop, the cylindrical and supplementary shields will not be present. To model this configuration, a 6" diameter cylinder with a height equal to the vessel diameter and a 4" steel shield is assumed to represent the exposed 6" diameter surface of the vessel wall. The activity of the source is adjusted until the contact dose rate is 10 R/hr, the measured contact dose rate. With this activity, the dose rate at 1 meter from the vessel wall is 297 mR/hr (See Appendix 5.5.3 - DOS file: LACBWR puncture.ms6).

Canister and Concrete

The source is shielded by the canister and the annular concrete giving a maximum (at the core midplane) 1 meter dose rate of 226 mR/hr and a dose rate at 1 meter from the vessel wall of 244 mR/hr (See Appendix 5.5.3 - DOS file: LACBWR 1m.ms6).

Cylindrical Shield

The 1.25" cylindrical shield is added over the core region giving a contact (1cm) dose rate of 126 mR/hr (See Appendix 5.5.3 - DOS file: LACBWR NCT contact.ms6).

Supplementary Shields

The 1.75" supplementary shields are adding to the sides of the cylindrical shields giving dose rates 2m from the vertical plane projected from the sides of the 10'8" wide railcar of 7.3 mR/hr (See Appendix 5.5.3 - DOS file: LACBWR NCT 2m.ms6).

5.4.4.2 Axial Radiation Levels

Source

As described in 5.3.1, the determination of the axial radiation source, based on the activation isotopic and the external radiation measurements, is completed followed by development of the axial shielding models (see Appendix 5.5.3 - DOS file: LACBWRbtm.ms6). The source model assumes a uniformly distributed spherical source surrounded by the 4" steel vessel and shows that, at the centerline of the bottom of the vessel on contact with the vessel, the dose rate is 0.978 R/hr, which matches the radiation survey. The dose rate on the vessel head is only 60 mR/hr so using the bottom dose rate to model the axial source is conservative. This source is used in all the subsequent axial dose rate models, both NCT and HAC.

End Plates

The 4" steel end plates are added as shielding for the bottom of the package neglecting the concrete in the bottom of the package and the distance between the vessel and the end plates. The dose rate on contact with the end plate is 15.8 mR/hr, 6.8 mR/hr at 1 meter, and ,at 2 meters, is 3.6 mR/hr (see Appendix 5.5.3 – DOS file LACBWRbtmNACHAC.ms6).

5.4.4.3 NCT Results

After the NCT tests, there are no changes to the shielding configuration. As the LACBWR package will be shipped "exclusive use", the applicable dose rates are the maximum on contact with the package and at 2 meters from the vertical plane projected from the outer edges of the 10'8" wide railcar. The package contact maximum is the radial dose rate at the core midplane on contact with the cylindrical shield, i.e., 126 mR/hr; the maximum contact dose rate on the end plates is 15.8 mR/hr. The maximum dose rate 2 meters from the sides of the railcar is 7.3 mR/hr and 2 meters from the ends of the package is 3.6 mR/hr.

5.4.4.4 HAC Results

As described in Chapter 2, under the free drop test, the cylindrical and supplementary shielding is assumed to disengage. Thus, the maximum dose rate at 1 meter from the package after the free drop is that due to shielding the vessel with the canister and concrete, i.e., 226 mR/hr.

After the puncture test, the maximum dose rate will be one meter from the exposed vessel wall. This dose rate is conservatively assumed to be the sum of the dose rate from the exposed 6" diameter surface of the vessel wall, i.e., 297 mR/hr, and the dose rate 1 meter from the vessel wall with the canister and concrete in place (to represent the rest of the package), i.e., 244 mR/hr. The sum is 541 mR/hr.

5.5 APPENDICIES

5.5.1 References

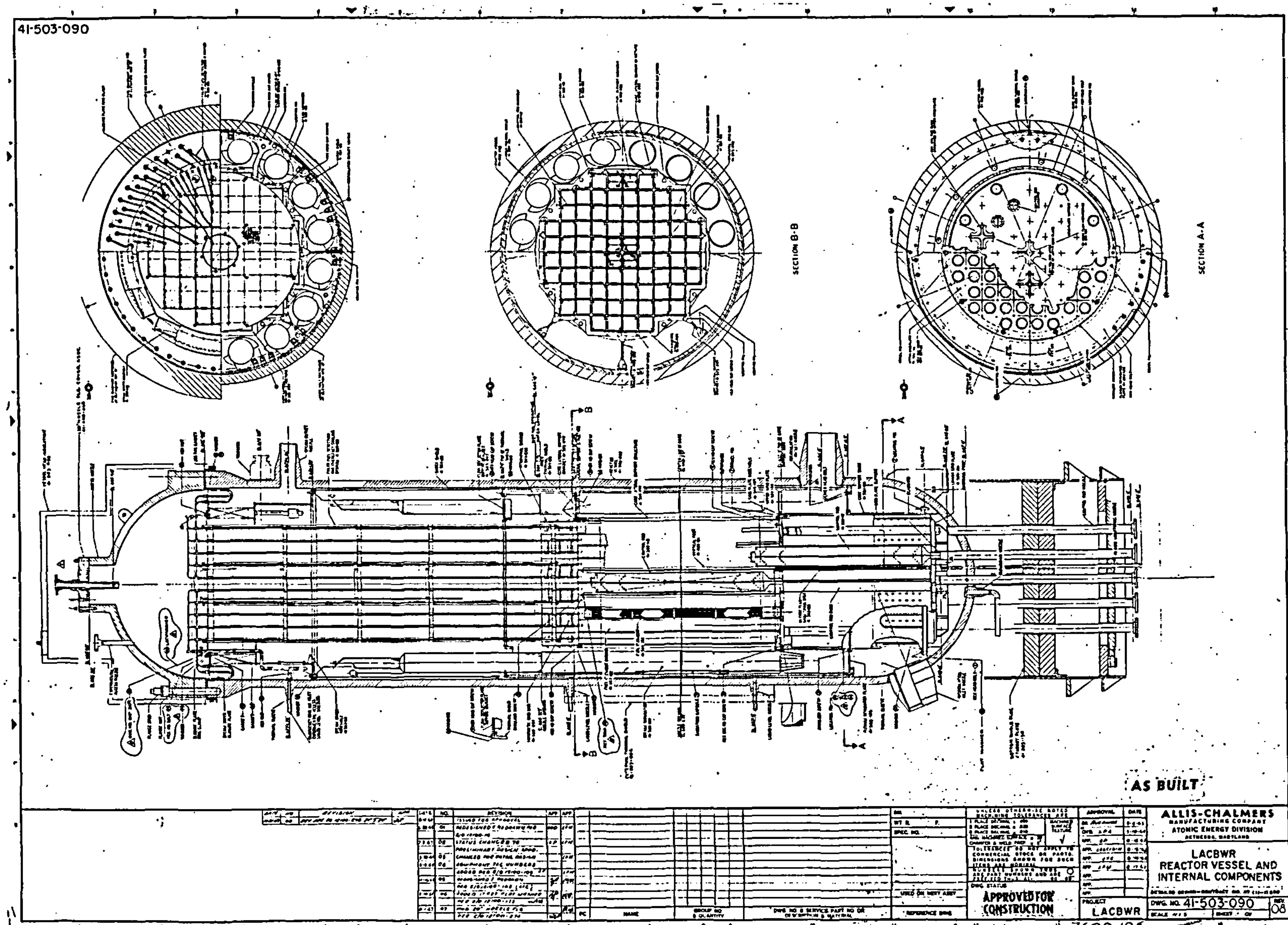
5.5.2 Surveys and Drawing

5.5.3 MicroShield Output

5.5.1 References

5-1 MicroShield Version 6, Grove Engineering Inc., March 2003

5.5.2 Radiation Surveys and LACBWR Vessel Drawing


41-503-090
SECTION B-B
SECTION A-A
AS BUILT
41-503-090
ALLIS-CHALMERS
MANUFACTURING COMPANY
ATOMIC ENERGY DIVISION
BETHESDA, MARYLAND
LACBWR
REACTOR VESSEL AND
INTERNAL COMPONENTS
DWG. NO. 41-503-090
08
APPROVED FOR
CONSTRUCTION
PROJECT
LACBWR
7620-186

Miscellaneous Radiation Survey Record

Date/Time 6/1/5 1300 Surveyor HANSEN
Instrument Type EXTENDER Signature Hansen
Instrument Serial # 15879 SWP # 05-21 (If applicable)
Location Rx BUILDING LOWER CAVITY

ALL readings in mRem/hr unless otherwise noted.

Remarks:

Rx vessel rad survey
6/1/2005
14 degees east of north

	Distance up in feet	Detector towards Rx Vessel	Detector towards Thermal Shield
inside bottom of Thermal Shield	0	800 mR	300 mR
	1	3.2 R	750 mR
	2	6.6 R	1.6 R
	3	9.2 R	2.1 R
	4	12 R	
approximate core mid plan			
	5	11 R	
Met interference		11 R	
Met interference	6		2.6 R
	7		

Miscellaneous Radiation Survey Record

Date/Time 6/2/5 1000 Surveyor KRUEGER

Instrument Type EXTENDER Signature Krueger

Instrument Serial # 15879 SWP # 05-21 (If applicable)

Location LOWER CAVITY - Rx VESSEL SURVEY

ALL readings in mRem/hr unless otherwise noted.

Remarks:

Rx Vessel Rad Survey
6/2/2005
20 degees north of west

	Distance up in feet	Detector towards Rx Vessel	Detector towards Thermal Shield
inside bottom of Thermal Shield	0	1.2 R	550 mR
	1	4.0 R	1.2 R
	2	7.8 R	1.8 R
	3	12.5 R	2.2 R
	4	12 R	2.5 R
approximate core mid plane	5	11 R	2.5 R
	6	7.5 R	2.5 R
	7	6.5 R	2.1 R 2 R
	8		

Rx Vessel Rad Survey
6/2/2005
40 degees north of west

	Distance up in feet	Detector towards Rx Vessel	Detector towards Thermal Shield
inside bottom of Thermal Shield	0	1.2 R	450 mR
	1	4.8 R	850 mR
	2	7.8 R	2 R
	3	11 R	2.2 R
	4	13 R	2.7 R
approximate core mid plane	5	12.5 R	2.6 R
	6	12 R	2.5 R
	7	10R	2.2 R

Miscellaneous Radiation Survey Record

Date/Time 8-9-2005 1000 Surveyor PENDEBECKER

Instrument Type ☆ Signature R.E. Pendebecker

Instrument Serial # ☆ SWP # 05-23 (If applicable)

Location C.B. UPPER & LOWER CAVITY

☆ EXTENDER #15879

RO-3 — #169

ALL readings in mRem/hr unless otherwise noted.

Remarks: SEE ATTACHED SHEETS

COPY

Rx Vessel Rad Survey
8/10/2005

All readings in mRem

Location 1
14 degees east of north

below external thermal shield

	Distance down in feet	Extendor	RO-3	
at bottom of thermal shield el. 655' 2"	0	500	na	
	1	280	310	
	2	200	220	
	3	220	220	
start of RPV lower head	4	130	160	
Rx vessel, upper pipe chase	Distance up in feet			
top shield wall, el 675'	0	40	44	
	1	35	38	
	2	na	na	nozzle
overhead el.678' 3"	3	75	na	

na = not accessible

Extender #15879
RO-3 #169

Rx Vessel Rad Survey
8/10/2005

All readings in mRem

Location 2
20 degees north of west

below external thermal shield

	Distance down in feet	Extendor	RO-3	
At bottom of thermal shield el. 655' 2"	0	580	na	
	1	390	380	
	2	290	300	
	3	300	320	
start of RPV lower head	4	na	na	nozzle

Rx vessel, upper pipe chase	Distance up in feet		
top shield wall, el 675'	0	45	47
	1	45	46
	2	50	50
overhead el.678' 3"	3	80	60

na = not accessible

Extender #15879
RO-3 #169

Rx Vessel Rad Survey
8/10/2005

All readings in mRem

Location 3
40 degees north of west

below external thermal shield

	Distance down in feet	Extendor	
At bottom of thermal shield el. 655' 2"	0	480	na
	1	350	360
	2	230	270
	3	280	270
start of RPV lower head	4	180	220

Rx vessel, upper pipe chase	Distance up in feet		
top shield wall, el 675'	0	45	41
	1	40	40
	2	50	50
overhead el.678' 3"	3	70	60

na = not accessible

Extender #15879
RO-3 #169

Page 9
HSP-02.6
Issue 20
Attachment 1

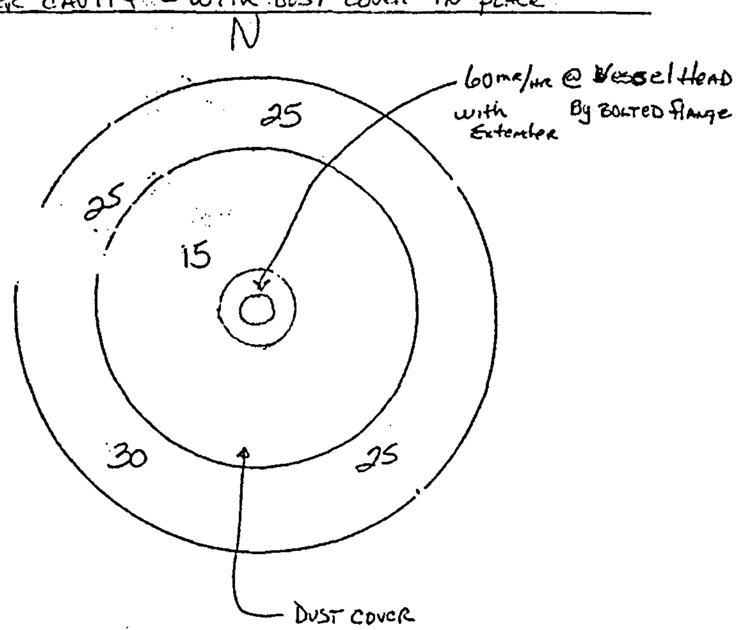
Miscellaneous Radiation Survey Record

Date/Time 11-17-5 / 1050 Surveyor Krueger / Hansen

Instrument Type Extender / RSO-50E Signature Krueger / Hansen

Instrument Serial # 15879 C440J SWP # 05-31 (If applicable)

Location Upper Cavity - with Dust cover in place



ALL readings in mRem/hr unless otherwise noted.

Remarks:

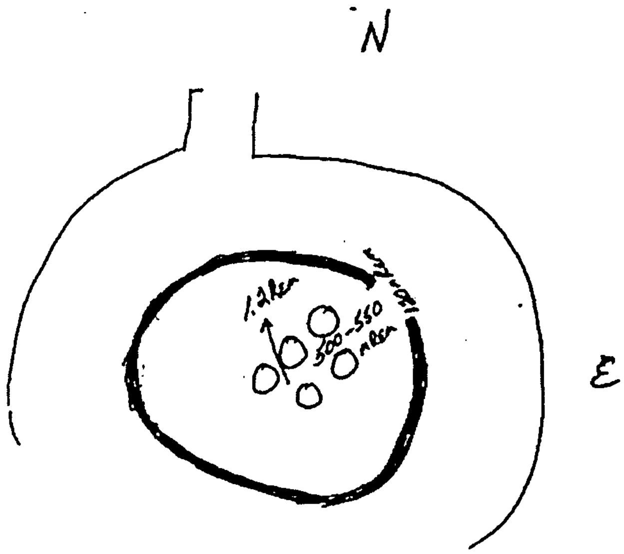
Miscellaneous Radiation Survey Record

Date/Time 11/9/5 1300 Surveyor PENNEBECKER

Instrument Type EXTENDER Signature ____________

Instrument Serial # 15879 SWP # 05-30 (if applicable)

Location LOWER CAVITY - UNDER RX VESSEL - BETWEEN CRD EXTENTION TUBES



ALL readings in mRem/hr unless otherwise noted.

Remarks: ____________

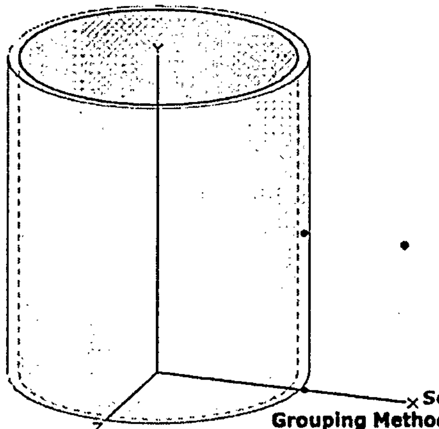
5.5.3 MicroShield Output

MicroShield v6.02 (6.02-00128)

Page : 1
DOS File : LACBWR radial source.ms6
Run Date: November 15, 2005
Run Time: 2:07:24 PM
Duration : 00:00:01

File Ref: ___________
Date: ___________
By: ___________
Checked:

Case Title: RPV
Description: RPV only; match to survey
Geometry: 7 - Cylinder Volume - Side Shields



Source Dimensions

Height	304.8 cm	10 ft 0.0 in
Radius	125.73 cm	4 ft 1.5 in

Dose Points

	X	Y	Z
# 1	235.89 cm	152.4 cm	0 cm
	7 ft 8.9 in	5 ft 0.0 in	0.0 in
# 2	139.7 cm	152.4 cm	0 cm
	4 ft 7.0 in	5 ft 0.0 in	0.0 in
# 3	139.7 cm	0 cm	0 cm
	4 ft 7.0 in	0.0 in	0.0 in

Shields

Shield Name	Dimension	Material	Density
Source	1.51e+07 cm³	Iron	2
Shield 1	10.16 cm	Iron	7.86
Transition		Air	0.00122
Air Gap		Air	0.00122

Source Input
Grouping Method : Actual Photon Energies

Nuclide	curies	becquerels	μCi/cm³	Bq/cm³
Co-60	1.0653e+004	3.9416e+014	7.0377e+002	2.6039e+007

Buildup
The material reference is : Shield 1

Integration Parameters

Radial	10
Circumferential	10
Y Direction (axial)	20

Results - Dose Point # 1 - (235.89,152.4,0) cm

Energy MeV	Activity photons/sec	Fluence Rate MeV/cm²/sec No Buildup	Fluence Rate MeV/cm²/sec With Buildup	Exposure Rate mR/hr No Buildup	Exposure Rate mR/hr With Buildup
0.6938	6.430e+10	2.392e+00	2.970e+01	4.618e-03	5.735e-02
1.1732	3.942e+14	1.413e+05	1.139e+06	2.526e+02	2.036e+03
1.3325	3.942e+14	2.358e+05	1.705e+06	4.090e+02	2.958e+03
TOTALS:	7.884e+14	3.771e+05	2.844e+06	6.616e+02	4.993e+03

Results - Dose Point # 2 - (139.7,152.4,0) cm

Energy MeV	Activity photons/sec	Fluence Rate MeV/cm²/sec No Buildup	Fluence Rate MeV/cm²/sec With Buildup	Exposure Rate mR/hr No Buildup	Exposure Rate mR/hr With Buildup
0.6938	6.430e+10	4.068e+00	5.592e+01	7.855e-03	1.080e-01
1.1732	3.942e+14	2.691e+05	2.273e+06	4.809e+02	4.062e+03
1.3325	3.942e+14	4.561e+05	3.422e+06	7.914e+02	5.938e+03
TOTALS:	7.884e+14	7.253e+05	5.696e+06	1.272e+03	1.000e+04

Results - Dose Point # 3 - (139.7,0,0) cm

Page : 2
DOS File : LACBWR radial source.ms6
Run Date: November 15, 2005
Run Time: 2:07:24 PM
Duration : 00:00:01

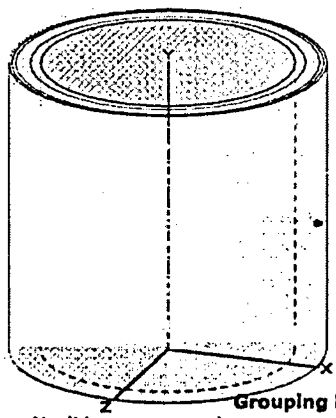
Energy MeV	Activity photons/sec	Fluence Rate MeV/cm²/sec No Buildup	Fluence Rate MeV/cm²/sec With Buildup	Exposure Rate mR/hr No Buildup	Exposure Rate mR/hr With Buildup
0.6938	6.430e+10	2.038e+00	2.796e+01	3.935e-03	5.399e-02
1.1732	3.942e+14	1.346e+05	1.136e+06	2.405e+02	2.030e+03
1.3325	3.942e+14	2.281e+05	1.711e+06	3.957e+02	2.968e+03
TOTALS:	7.884e+14	3.627e+05	2.847e+06	6.362e+02	4.998e+03

MicroShield v6.02 (6.02-00128)

Page : 1
DOS File : LACBWR NCT cont.ms6
Run Date: November 22, 2005
Run Time: 11:03:13 AM
Duration : 00:00:00

File Ref:
Date: ___________
By: ___________
Checked: ___________

Case Title: RPV package
Description: NCT, primary, contact
Geometry: 7 - Cylinder Volume - Side Shields



Source Dimensions

Height	304.8 cm	10 ft 0.0 in
Radius	125.73 cm	4 ft 1.5 in

Dose Points

	X	Y	Z
# 1	158.48 cm	152.4 cm	0 cm
	5 ft 2.4 in	5 ft 0.0 in	0.0 in

Shields

Shield Name	Dimension	Material	Density
Source	1.51e+07 cm³	Iron	2
Shield 1	10.16 cm	Iron	7.86
Shield 2	14.605 cm	Concrete	1.9
Shield 3	3.81 cm	Iron	7.86
Shield 4	3.175 cm	Iron	7.86
Transition		Air	0.00122
Air Gap		Air	0.00122

Source Input
Grouping Method : Actual Photon Energies

Nuclide	curies	becquerels	μCi/cm³	Bq/cm³
Co-60	1.0653e+004	3.9416e+014	7.0377e+002	2.6039e+007

Buildup
The material reference is : Shield 1

Integration Parameters

Radial	10
Circumferential	10
Y Direction (axial)	20

Results

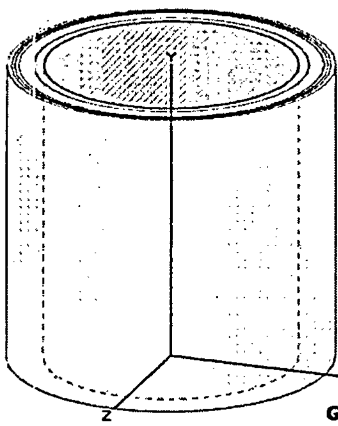
Energy MeV	Activity photons/sec	Fluence Rate MeV/cm²/sec No Buildup	Fluence Rate MeV/cm²/sec With Buildup	Exposure Rate mR/hr No Buildup	Exposure Rate mR/hr With Buildup
0.6938	6.430e+10	6.655e-03	1.922e-01	1.285e-05	3.710e-04
1.1732	3.942e+14	1.544e+03	2.463e+04	2.760e+00	4.402e+01
1.3325	3.942e+14	3.438e+03	4.718e+04	5.965e+00	8.185e+01
TOTALS:	7.884e+14	4.983e+03	7.181e+04	8.725e+00	1.259e+02

MicroShield v6.02 (6.02-00128)

Page : 1
DOS File : LACBWR NCT 2m.ms6
Run Date : November 23, 2005
Run Time : 8:38:14 AM
Duration : 00:00:00

File Ref: __________
Date: __________
By: __________
Checked: __________

Case Title: RPV package
Description: NCT, supplimental shield, 2m
Geometry: 7 - Cylinder Volume - Side Shields



Source Dimensions

Height	304.8 cm	10 ft 0.0 in
Radius	125.73 cm	4 ft 1.5 in

Dose Points

	X	Y	Z
# 1	362.56 cm	152.4 cm	0 cm
	11 ft 10.7 in	5 ft 0.0 in	0.0 in

Shields

Shield Name	Dimension	Material	Density
Source	1.51e+07 cm³	Iron	2
Shield 1	10.16 cm	Iron	7.86
Shield 2	14.605 cm	Concrete	1.9
Shield 3	3.81 cm	Iron	7.86
Shield 4	3.175 cm	Iron	7.86
Shield 5	4.445 cm	Iron	7.86
Transition		Air	0.00122
Air Gap		Air	0.00122

Source Input
Grouping Method : Actual Photon Energies

Nuclide	curies	becquerels	μCi/cm³	Bq/cm³
Co-60	1.0653e+004	3.9416e+014	7.0377e+002	2.6039e+007

Buildup
The material reference is : Shield 1

Integration Parameters

Radial	10
Circumferential	10
Y Direction (axial)	20

Results

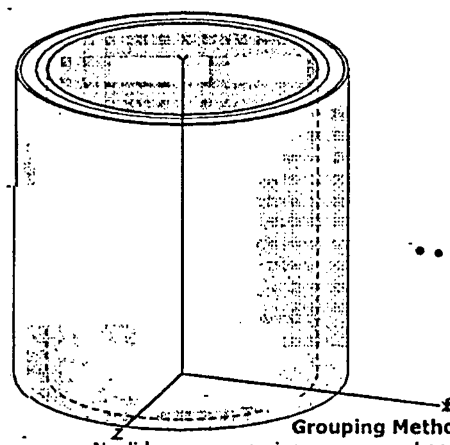
Energy MeV	Activity photons/sec	Fluence Rate MeV/cm²/sec No Buildup	Fluence Rate MeV/cm²/sec With Buildup	Exposure Rate mR/hr No Buildup	Exposure Rate mR/hr With Buildup
0.6938	6.430e+10	1.619e-04	6.010e-03	3.126e-07	1.160e-05
1.1732	3.942e+14	6.651e+01	1.327e+03	1.189e-01	2.371e+00
1.3325	3.942e+14	1.679e+02	2.859e+03	2.914e-01	4.960e+00
TOTALS:	7.884e+14	2.345e+02	4.186e+03	4.102e-01	7.331e+00

MicroShield v6.02 (6.02-00128)

Page : 1
DOS File : LACBWR 1m.ms6
Run Date: November 15, 2005
Run Time: 8:15:37 AM
Duration : 00:00:01

File Ref: __________
Date: __________
By: __________
Checked:

Case Title: RPV package
Description: HAC; no shields; can intact, D2 1m from vessel wall
Geometry: 7 - Cylinder Volume - Side Shields



Source Dimensions

Height	304.8 cm	10 ft 0.0 in
Radius	125.73 cm	4 ft 1.5 in

Dose Points

	X	Y	Z
# 1	254.305 cm	152.4 cm	0 cm
	8 ft 4.1 in	5 ft 0.0 in	0.0 in
# 2	235.89 cm	152.4 cm	0 cm
	7 ft 8.9 in	5 ft 0.0 in	0.0 in

Shields

Shield Name	Dimension	Material	Density
Source	1.51e+07 cm³	Iron	2
Shield 1	10.16 cm	Iron	7.86
Shield 2	14.605 cm	Concrete	1.9
Shield 3	3.81 cm	Iron	7.86
Transition		Air	0.00122
Air Gap		Air	0.00122

Source Input
Grouping Method : Actual Photon Energies

Nuclide	curies	becquerels	µCi/cm³	Bq/cm³
Co-60	1.0653e+004	3.9416e+014	7.0377e+002	2.6039e+007

Buildup
The material reference is : Shield 1

Integration Parameters

Radial	10
Circumferential	10
Y Direction (axial)	20

Results - Dose Point # 1 - (254.305,152.4,0) cm

Energy MeV	Activity photons/sec	Fluence Rate MeV/cm²/sec No Buildup	Fluence Rate MeV/cm²/sec With Buildup	Exposure Rate mR/hr No Buildup	Exposure Rate mR/hr With Buildup
0.6938	6.430e+10	2.205e-02	5.102e-01	4.257e-05	9.850e-04
1.1732	3.942e+14	3.399e+03	4.607e+04	6.074e+00	8.233e+01
1.3325	3.942e+14	6.989e+03	8.270e+04	1.213e+01	1.435e+02
TOTALS:	7.884e+14	1.039e+04	1.288e+05	1.820e+01	2.258e+02

Results - Dose Point # 2 - (235.89,152.4,0) cm

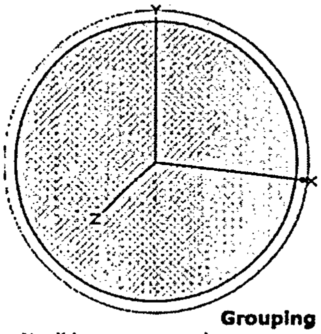
Energy MeV	Activity photons/sec	Fluence Rate MeV/cm²/sec No Buildup	Fluence Rate MeV/cm²/sec With Buildup	Exposure Rate mR/hr No Buildup	Exposure Rate mR/hr With Buildup
0.6938	6.430e+10	2.422e-02	5.568e-01	4.676e-05	1.075e-03
1.1732	3.942e+14	3.691e+03	4.982e+04	6.596e+00	8.904e+01
1.3325	3.942e+14	7.576e+03	8.936e+04	1.314e+01	1.550e+02
TOTALS:	7.884e+14	1.127e+04	1.392e+05	1.974e+01	2.441e+02

MicroShield v6.02 (6.02-00128)

Page : 1
DOS File : LACBWRbtm.ms6
Run Date: November 14, 2005
Run Time: 2:21:34 PM
Duration : 00:00:00

File Ref: ___________
Date: ___________
By: ___________
Checked: ___________

Case Title: LACBWR
Description: bottom
Geometry: 6 - Sphere



Source Dimensions

Radius	125.73 cm	4 ft 1.5 in

Dose Points

	X	Y	Z
# 1	138.43 cm	0 cm	0 cm
	4 ft 6.5 in	0.0 in	0.0 in

Shields

Shield Name	Dimension	Material	Density
Source	8.33e+06 cm³	Iron	2
Shield 1	10.16 cm	Iron	7.86
Transition		Air	0.00122
Air Gap		Air	0.00122

Source Input
Grouping Method : Actual Photon Energies

Nuclide	curies	becquerels	μCi/cm³	Bq/cm³
Co-60	6.6576e+002	2.4633e+013	7.9967e+001	2.9588e+006

Buildup
The material reference is : Shield 1

Integration Parameters

Rho (Radial)	10
Angle	10

Results

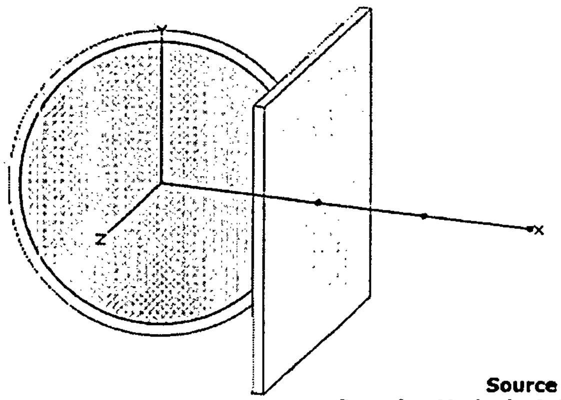
Energy MeV	Activity photons/sec	Fluence Rate MeV/cm²/sec No Buildup	Fluence Rate MeV/cm²/sec With Buildup	Exposure Rate mR/hr No Buildup	Exposure Rate mR/hr With Buildup
0.6938	4.018e+09	4.728e-01	5.870e+00	9.128e-04	1.133e-02
1.1732	2.463e+13	2.784e+04	2.233e+05	4.974e+01	3.990e+02
1.3325	2.463e+13	4.636e+04	3.337e+05	8.043e+01	5.790e+02
TOTALS:	4.927e+13	7.420e+04	5.570e+05	1.302e+02	9.780e+02

MicroShield v6.02 (6.02-00128)

Page : 1
DOS File : LACBWRbtmNCTHAC.ms6
Run Date: November 22, 2005
Run Time: 3:39:16 PM
Duration : 00:00:00

File Ref: ___________
Date: ___________
By: ___________
Checked: ___________

Case Title: LACBWR
Description: bottom
Geometry: 6 - Sphere



Source Dimensions

Radius	125.73 cm	4 ft 1.5 in

Dose Points

	X	Y	Z
# 1	147.05 cm	0 cm	0 cm
	4 ft 9.9 in	0.0 in	0.0 in
# 2	246.05 cm	0 cm	0 cm
	8 ft 0.9 in	0.0 in	0.0 in
# 3	346.05 cm	0 cm	0 cm
	11 ft 4.2 in	0.0 in	0.0 in

Shields

Shield Name	Dimension	Material	Density
Source	8.33e+06 cm³	Iron	2
Shield 1	10.16 cm	Iron	7.86
Transition		Air	0.00122
Shield 3	10.16 cm	Iron	7.86
Air Gap		Air	0.00122

Source Input
Grouping Method : Actual Photon Energies

Nuclide	curies	becquerels	μCi/cm³	Bq/cm³
Co-60	6.6576e+002	2.4633e+013	7.9967e+001	2.9588e+006

Buildup
The material reference is : Shield 1

Integration Parameters

Rho (Radial)	10
Angle	10

Results - Dose Point # 1 - (147.05,0,0) cm

Energy MeV	Activity photons/sec	Fluence Rate MeV/cm²/sec No Buildup	Fluence Rate MeV/cm²/sec With Buildup	Exposure Rate mR/hr No Buildup	Exposure Rate mR/hr With Buildup
0.6938	4.018e+09	9.064e-04	2.511e-02	1.750e-06	4.849e-05
1.1732	2.463e+13	1.982e+02	3.097e+03	3.542e-01	5.535e+00
1.3325	2.463e+13	4.364e+02	5.896e+03	7.571e-01	1.023e+01
TOTALS:	4.927e+13	6.346e+02	8.994e+03	1.111e+00	1.576e+01

Results - Dose Point # 2 - (246.05,0,0) cm

Energy MeV	Activity photons/sec	Fluence Rate MeV/cm²/sec No Buildup	Fluence Rate MeV/cm²/sec With Buildup	Exposure Rate mR/hr No Buildup	Exposure Rate mR/hr With Buildup
0.6938	4.018e+09	4.054e-04	1.114e-02	7.828e-07	2.150e-05
1.1732	2.463e+13	8.683e+01	1.342e+03	1.552e-01	2.398e+00
1.3325	2.463e+13	1.901e+02	2.538e+03	3.298e-01	4.404e+00
TOTALS:	4.927e+13	2.769e+02	3.880e+03	4.849e-01	6.802e+00

Results - Dose Point # 3 - (346.05,0,0) cm

Page : 2
DOS File : LACBWRbtmNCTHAC.ms6
Run Date: November 22, 2005
Run Time: 3:39:16 PM
Duration : 00:00:00

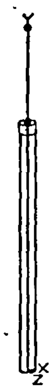
Energy MeV	Activity photons/sec	Fluence Rate MeV/cm²/sec No Buildup	Fluence Rate MeV/cm²/sec With Buildup	Exposure Rate mR/hr No Buildup	Exposure Rate mR/hr With Buildup
0.6938	4.018e+09	2.160e-04	5.917e-03	4.171e-07	1.142e-05
1.1732	2.463e+13	4.595e+01	7.075e+02	8.211e-02	1.264e+00
1.3325	2.463e+13	1.004e+02	1.336e+03	1.742e-01	2.317e+00
TOTALS:	4.927e+13	1.463e+02	2.043e+03	2.563e-01	3.582e+00

MicroShield v6.02 (6.02-00128)

Page : 1
DOS File : LACBWR puncture.ms6
Run Date: November 23, 2005
Run Time: 11:11:44 AM
Duration : 00:00:00

File Ref:__________
Date:__________
By:__________
Checked:__________

Case Title: LACBWR
Description: puncture - cylinder
Geometry: 8 - Cylinder Volume - End Shields



Source Dimensions

Height	251.46 cm	8 ft 3.0 in
Radius	7.62 cm	3.0 in

Dose Points

	X	Y	Z
# 1	0 cm	262.62 cm	0 cm
	0.0 in	8 ft 7.4 in	0.0 in
# 2	0 cm	361.62 cm	0 cm
	0.0 in	11 ft 10.4 in	0.0 in

Shields

Shield Name	Dimension	Material	Density
Source	4.59e+04 cm³	Iron	2
Shield 1	10.16 cm	Iron	7.86
Air Gap		Air	0.00122

Source Input
Grouping Method : Actual Photon Energies

Nuclide	curies	becquerels	μCi/cm³	Bq/cm³
Co-60	9.4607e+001	3.5005e+012	2.0625e+003	7.6313e+007

Buildup
The material reference is : Shield 1

Integration Parameters

Radial	20
Circumferential	10
Y Direction (axial)	10

Results - Dose Point # 1 - (0,262.62,0) cm

Energy MeV	Activity photons/sec	Fluence Rate MeV/cm²/sec No Buildup	Fluence Rate MeV/cm²/sec With Buildup	Exposure Rate mR/hr No Buildup	Exposure Rate mR/hr With Buildup
0.6938	5.710e+08	7.431e+00	7.913e+01	1.435e-02	1.528e-01
1.1732	3.500e+12	3.532e+05	2.367e+06	6.312e+02	4.230e+03
1.3325	3.500e+12	5.557e+05	3.324e+06	9.642e+02	5.767e+03
TOTALS:	7.001e+12	9.090e+05	5.691e+06	1.595e+03	9.998e+03

Results - Dose Point # 2 - (0,361.62,0) cm

Energy MeV	Activity photons/sec	Fluence Rate MeV/cm²/sec No Buildup	Fluence Rate MeV/cm²/sec With Buildup	Exposure Rate mR/hr No Buildup	Exposure Rate mR/hr With Buildup
0.6938	5.710e+08	2.176e-01	2.293e+00	4.201e-04	4.428e-03
1.1732	3.500e+12	1.031e+04	6.990e+04	1.842e+01	1.249e+02
1.3325	3.500e+12	1.628e+04	9.901e+04	2.824e+01	1.718e+02
TOTALS:	7.001e+12	2.658e+04	1.689e+05	4.666e+01	2.967e+02

Attachment 7

Chapter 8, rev.1

8.0 ACCEPTANCE TESTS AND MAINTENANCE PROGRAM

The La Crosse Boiling Water Reactor (LACBWR) Reactor Pressure Vessel Package (RPVP) design requirements and operating activities are discussed in Chapters 1 through 7. The packaging is designed for exclusive use, one-time transportation and disposal of the RPV at the Chem-Nuclear Systems low level radioactive waste disposal facility at Barnwell, South Carolina. This chapter describes the acceptance tests and inspections that will be performed on the RPVP to ensure compliance with its design requirements, and the requirements of Subpart G of 10 CFR71.

8.1 ACCEPTANCE TESTS

Acceptance tests and inspections will be performed prior to the transportation of the package in compliance with 10 CFR 71.85. The sequential order of these inspections and tests will be coordinated with other operations as detailed in Chapter 7. All the tests and inspections on the package described in this chapter will be conducted and documented in accordance with written procedures approved under the licensee's NRC Approved Quality Assurance (QA) Program.

8.1.1 Visual Inspections and Measurements

10 CFR 71.85(a) states:

"The licensee shall ascertain that there are no cracks, pinholes, uncontrolled voids, or other defects that could significantly reduce the effectiveness of the packaging."

10 CFR 71.87(b) requires the licensee to determine (also addressed in Section 7.2.2.b) that:

"The package is in unimpaired physical condition except for superficial defects such as marks or dents."

10 CFR 71.85 (c) states:

"...Before applying the model number, the licensee shall determine that the packaging has been fabricated in accordance with the design approved by the Commission."

The packaging will be fabricated under the licensee's QA Program and in accordance with the design presented in this SAR. Inspections and examinations of the fabricated material and shop welds will be completed prior to shipping the package to LACBWR. The containment shell will be delivered to LACBWR in two sections, a cylindrical lower shell assembly, and an upper shell assembly as discussed in Section 7.1 (see the drawing in Appendix 1.4). Upon arrival on the LACBWR site, the sections will be visually examined to assure no damage has occurred during transport.

The RPV will be placed in the lower shell assembly and grouted. The upper shell assembly will be field welded to the lower shell assembly to form an integral unit. The upper shell to lower shell field weld and grout plug field welds will be visually examined and nondestructively examined as explained in Section 8.1.2. If the examinations reveal any defects, the defects will be evaluated based on the acceptance criteria in 8.1.2 to ascertain whether remedial actions may be warranted. Inspections and repairs, if required, will be appropriately documented.

The above inspections and repairs will be performed using approved written procedures under the licensee's QA Program. Fabrication of the containment shell and other fabricated parts will be performed under the licensee's QA Program, and accomplishment of the inspections described above will satisfy the 10 CFR 71.85 and 10 CFR 71.87 requirements stated in Section 8.1.1. Compliance with the requirements of 10 CFR 71.85(a) is further assured with the weld examinations described in Section 8.1.2.

8.1.2 Weld Examinations

Compliance with the requirements of 10 CFR 71.85(a) is confirmed by the package weld examinations. Examinations of the shop welds will be done at the shop, per the criteria identified in the fabrication specifications and under the licensee's QA Program, prior to shipping the package to LACBWR. All welds will be visually examined, magnetic particle (MT) examined, and volumetrically examined using RT for category A and B welds or UT for category C welds, the acceptance criteria shall be the ASME Code, Section III, Article ND-5300.

The field weld between the Upper Shell and Lower Shell will be visually, MT, and UT examined with the acceptance criteria of ASME Code, Section III, Article ND-5300.

SA-516, Grade 70, plugs mounted on cover plates will be placed into the 3-1/2" diameter holes used for injection of the Medium Density Cellular Concrete (MDCC) into the package. The welds used to install these plugs will be visually examined and MT examined with acceptance criteria the ASME Code, Section III, Article ND-5300.

The above operations will be performed using approved written procedures under the licensee's QA Program. Accomplishment of the inspections and examinations described in Sections 8.1.1 and 8.1.2 will satisfy the requirement of 10 CFR 71.85(a).

8.1.3 Structural and Pressure Tests

The structural integrity of the package is analytically demonstrated in section 2, and based on this analysis showing a safety factor greater than 8, no pressure test will be performed.

8.1.4 Leak Tests

The containment shell consists of two welded cylindrical steel shells plus top and bottom plates welded to the cylinders, the shells are welded together after the RPV is loaded and penetration seal plugs are welded in place after the MDCC is installed. These welds will undergo examinations as stated in Sections 8.1.1 and 8.1.2 to insure that the welds are sound and continuous. There are no mechanical closures, gaskets, valves or other similar types of penetrations.

The package contains primarily solid radioactive material with only a very small percentage of radioactive material as surface contamination in the RPV, which will be fixed in place by the MDCC and low density cellular concrete. There is no gaseous or liquid radioactive material in the package. As concluded in Section 2 and Section 4.3 of this SAR, the package integrity under Normal Conditions of Transport provides assurance that the radioactive materials will remain contained in the package. Therefore, the package meets the requirements of 10CFR71 under Normal Conditions of Transport. The discussion in Section 4.4 of this SAR shows that in the event of a breach of containment under Hypothetical Accident Conditions, the released radioactivity levels are within the limits of 10 CFR 71. Therefore, no leak test is required.

8.1.5 Component and Material Tests

The containment shell is a welded steel enclosure used for the transportation and disposal of the RPV. All plate material shall be provided with certified mechanical and chemical test reports. These tests shall include determination of the nil-ductility transition temperature for materials three inches thick and over. All welding will be performed using procedures qualified for notch toughness requirements to match the requirements of the base materials. Post weld heat treatment (PWHT) will not be performed unless the weld procedure qualification requires PWHT to meet mechanical properties in the weld. Since fabrication of the containment shell will be accomplished in accordance with the licensee's QA Program, verification of the materials of construction against the design requirements is covered under that program.

8.1.6 Shielding Tests

As discussed below, shielding tests prior to final acceptance for shipment are not required for this package. Fabrication of the packaging will be performed in accordance with the licensee's QA Program, which will provide assurance that the package is constructed in compliance with the design requirements described in this SAR. The controlled process for loading the packaging described in Sections 7.1.1 and 7.1.2, the weld examinations described in Sections 8.1.1 and 8.1.2, and the pre-shipment dose rate surveys discussed in Section 7.2.2.(j), will confirm the adequacy of the shielding as required by the package design. Because this is a single shipment package, and the design of the package the dose rate and shielding requirements are applicable only to the final configuration, shielding tests are not applicable to intermediate configurations (i.e., prior to emplacement of the MDCC).

8.1.7 Thermal Acceptance Tests

The analyses performed for thermal evaluation of the package in Chapter 3 have used conservative thermal properties for the materials present in the package. The package materials are capable of withstanding temperatures within its design envelope as shown in Chapter 3. Therefore, thermal acceptance tests are not required.

8.2 MAINTENANCE PROGRAM

The package is a single shipment steel container that will be used for transportation and disposal of the LACBWR RV. This package is a sealed enclosure with no instrumentation or operating control devices that are relied upon for maintaining and monitoring its integrity during the shipment. The initial acceptance tests and inspections described in Section 8.1, and the pre-shipment routine determinations performed in accordance with 10 CFR 71.87 criteria as detailed in Section 7.2.2 will ensure that the package complies with all applicable requirements. The procedures and instructions provided for the transportation operations as discussed in Section 7.3 will ensure safe transportation of the package. Therefore, no maintenance program is required for this package.