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Preliminary Design Report
Babcock & Wilcox
BR-100
100-Ton Rail/Barge
Spent Fuel Shipping Cask

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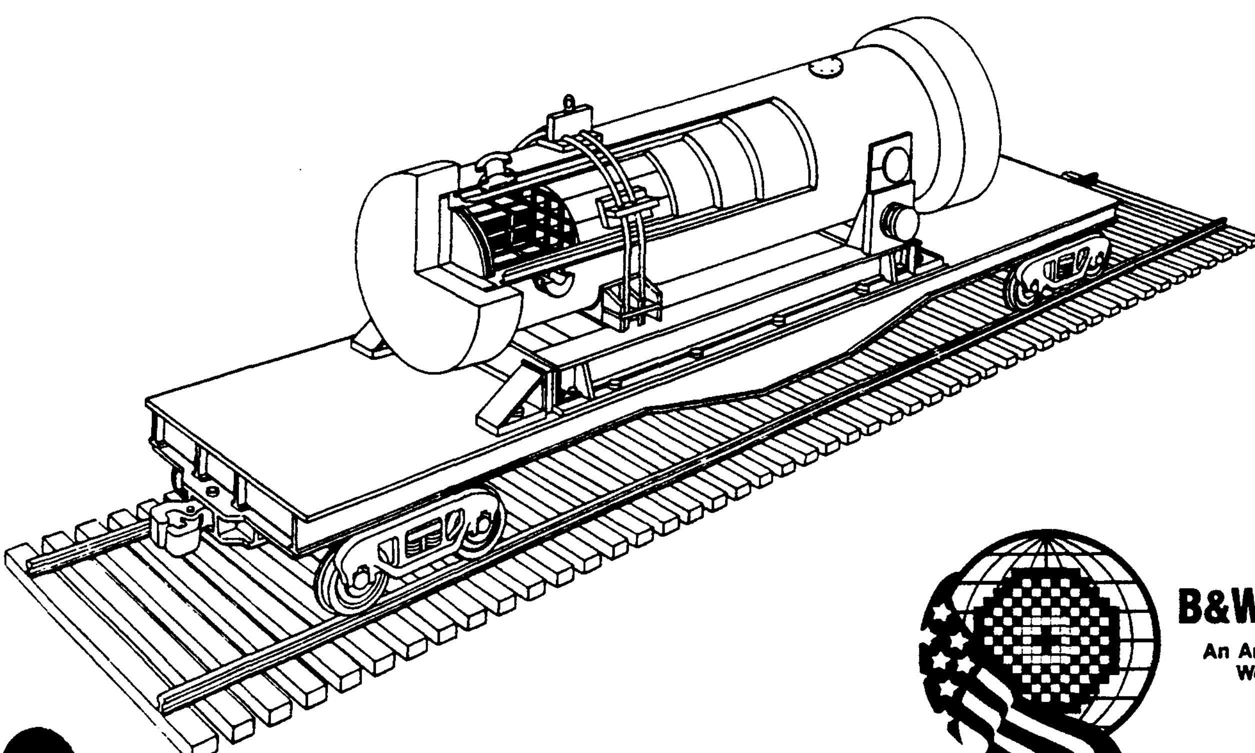

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FROM REACTOR CASK SYSTEM DEVELOPMENT PROGRAM

TRADE-OFF STUDY

B&W BR-100 CASK

EFFECT ON CAPACITY OF BURNUP CREDIT

Document No. 51-1175846-02

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The Effect of Burnup Credit on Cask Capacity

1.0 SUMMARY

This study investigated the effect that burnup credit has on the capacity of the BR-100 spent fuel shipping cask. It was done to satisfy Section 4.10.1 of the Cask Contract Statement of Work(SOW) that states: "The contractor shall conduct trade-off and impact evaluations of the following design considerations on cask payload capacities and costs: ..." For criticality analyses this involved evaluation of the "allowance of fuel burnup credit for criticality evaluations." Starting from the traditional "no-burnup credit" assumption, different levels of burnup credit were analyzed and their benefits documented. The reasonableness of the burnup assumptions and the complications of verifying the assumptions in a certification submittal were also assessed. Burnup credit was found to be worth about 0.006 in k_{eff} per Gigawatt days/metric ton uranium (Gwd/mtu) for the BR-100 configuration. The use of a Pressurized Water Reactor (PWR) assembly average burnup of 18 Gwd/mtu was judged to be reasonable and defendable, if fuel end effects were considered which reduced the analytical credit taken to no more than 16 Gwd/mtu. Use of such credit resulted in a 24% increase in the cask capacity for PWR fuel assemblies. As for costs, burnup credit will have little impact on the material or fabrication cost of the cask. However, due to the reduced number of shipments resulting from the increased capacity, the life cycle costs will decrease about 15%. For these reasons, the 18 Gwd/mtu assembly burnup credit was incorporated into the baseline design of the BR-100 cask for PWR fuel with enrichments greater than 3.2 weight percent (w/0). No judgments were made on any requirements, eg. fuel assembly burnup measurement, that the Nuclear Regulatory Commission (NRC) might impose on shippers of the fuel.

2. INTRODUCTION

The purpose of this document is to provide information on burnup credit as applied to the preliminary design of the BR-100 shipping cask. There is a brief description of the preliminary basket design and the features used to maintain a critically safe system. Following the basket description is a discussion of various criticality analyses used to evaluate burnup credit. The results from these analyses are then reviewed in the perspective of fuel burnups expected to be shipped to either the final repository or a Monitored Retrievable Storage (MRS) facility. The hurdles to employing burnup credit in the certification of any cask are then outlined and reviewed. The last section gives conclusions reached as to burnup credit for the BR-100 cask, based on our analyses and experience.

All information in this study refers to the cask configured to transport PWR fuel. Boiling Water Reactor (BWR) fuel satisfies the criticality requirements so that burnup credit is not needed. All calculations generated in the preparation of this report were based upon the preliminary design which will be optimized during the final design. The results and observations given are to help in evaluation of the feasibility of burnup credit use on spent fuel shipping casks and should not be construed as final and definitive data for the BR-100 cask or any other cask.

3. BASELINE BR-100 BASKET DESCRIPTION

Figure 1 shows the individual extruded aluminum fuel cell that is the basic building block of the BR-100 PWR basket. Twenty-one such fuel cells make up the central, fuel-bearing region of the basket assembly, as shown in Figure 2. Each PWR fuel cell has an square cavity 8.75 inches on a side. This cavity has been sized to easily accommodate loading and unloading of PWR fuel assemblies. Reactivity control is obtained primarily by a combination of poison plates and a water gap between adjacent fuel cells.

The poison plate is an Al/B_4C cermet material with a B^{10} areal density of 50 mg/cm^2. A protective aluminum cover and hard anodizing bring the cermet plate thickness to about 0.100 inch. The plates are 3.625 inches wide and are placed end-to-end to cover the length of the fuel cell in two strips per side.

A nominal water gap of 0.55 inches separates the poison plates of adjacent fuel cells. The water gaps form flux traps that are a necessary part of the criticality control of the basket. The flux traps provide the neutron thermalization required to make the cermet plates an effective absorber.

The synergistic effect of the cermet plates and water gap extends into the formers (cast aluminum cell lateral supports that are assembled with the cells and top/bottom plates to make the basket). As seen in Figure 2, the formers have dished out regions adjacent to the external faces of the fuel cells. Those regions were included to reduce neutron backscattering and add to the total reactivity control of the basket.

4. CRITICALITY ANALYSES

Prior to the adoption of the baseline basket design, a series of scoping calculations was performed to examine different absorber materials, moderators, and geometries. A detailed description of these calculations with specific results is contained in Reference 1 and is not repeated here. Those studies showed that, for a tradeoff between absorber thickness and water gap width, generally a larger gap provided the better method of reactivity control. For the BR-100 cask design, the 50 mg/cm^2 areal density of B^{10} used for the cermet makes it effectively 'black' to thermal neutrons (increased B^{10} density will not appreciably reduce k_{eff} further), while the thinness of the material allows the maximum water gap for a given cell pitch.

Because the BR-100 has both a weight limitation of 100 tons (loaded) and shielding criteria that effectively limits the cavity diameter available for the basket, it was an early objective to determine the minimum pitch without burnup credit. Analyses were performed which showed that, for the most reactive fuel (Westinghouse 17 X 17 Optimized Fuel Assembly (OFA), 4.5 w/0 enrichment), a minimum water gap of 1.3 inches would be required for a k_{eff} less than 0.95 (including bias and uncertainty). The resulting minimum cell pitch of about 10.8 inches was a consequence of the water gap being added to previous fuel handling, structural and thermal requirements.

The basket structure used on the BR-100 requires that cell walls line up in a regular fashion, without offset, as shown in Figure 2. This configuration has good structural and thermal characteristics, but limits the arrays that can be considered for a given cavity diameter. For instance, Figure 3 shows the regular arrays possible for various diameters. The BR-100 evolved into a maximum cavity diameter of 58.5 inches, thus limiting the payload (without burnup credit and with most reactive fuel at 4.5 w/0) to either a 16-array regular basket, a 21-array regular basket with a hollow spacer in four symmetric cell locations, or a difficult-to-fabricate 19-array irregular basket made of welded plates.

Other options were investigated which could have resulted in a smaller cell pitch (compatible with a 21 or 24-assembly regular array) without burnup credit. They included reducing the maximum enrichment to 3.2 w/0, using control rods on high-enrichment fuel, using a cell material of borated aluminum, and using an outer cell coating with a high neutron absorber content. For various reasons, they were all judged to be less acceptable than the benefit to be gained from taking limited credit for fuel burnup.

The k_{eff} benefit available from burnup credit was then investigated. The NULIF[2] code was executed in a depletion mode to generate reactivity values at various burnup points for an infinite array of PWR fuel rods. Depletion steps were obtained for 0, 4, 8, 12, 16, 18, and 20 Gwd/mtu. These values were used to obtain the reactivity differences between fresh fuel and the above burnups. Application of these values to the k_{max} derived from the KENO case for the BR-100 baseline design (21 PWR assemblies) provided the values listed in Table 1. The summation of the KENO k_{eff}, the k_{eff} uncertainty, and the KENO bias provides k_{max}. A plot of the values in Table 1 are shown in Figure 4 and show a k_{eff} benefit of approximately .006 per Gwd/mtu burnup. The same NULIF data was used to relate burnup reactivity credit with the equivalent reactivity of water gaps obtained from KENO fuel cell array models. This relation is illustrated in Figure 5 by the plot of burnup versus gap thickness. The gap thickness corresponding to the BR-100 base design is indicated; however, the data points do not refer to specific basket configurations.

Based on B&W's experience designing and analyzing nuclear fuel over the last thirty-five years, it was decided to use a minimum margin of 12.5% between the burnup claimed for an assembly average and the burnup credit used for criticality calculations. This margin is primarily due to fuel rod end effects which will be explained further in Section 5. This effectively means that an average assembly burnup of at least 18 Gwd/mtu would be required for a credit of 16 Gwd/mtu to be taken.

5. BURNUP CREDIT REASONABLENESS AND EFFECTS

One of the first questions that must be answered about burnup credit is "For the credit claimed, how much of the projected fuel inventory cannot be shipped?" A recent report[3] estimates that about 6% of the PWR fuel to be shipped to the repository will have burnups less than 20 Gwd/mtu. Of that 6%, almost none(if any at all) will have the high enrichment (over 3.2 w/0) that requires burnup credit for criticality control. Even fuel with this unrealistic combination of high enrichment and low burnup could, however, be transported in the standard BR-100 basket by using a spacer in four symmetric inner cells along the diagonal. This would reduce cask capacity to 17 assemblies and result in a maximum k_{eff} of about 0.94, even without burnup credit. The benefit of designing to a more realistic burnup distribution is then a 24% gain in cask capacity with no reduction in flexibility or significant extra equipment needed.

If the benefits of burnup credit are so obvious, then why should the designer stop at 18 Gwd/mtu? The answer to that question has contractual, operational, and analytical factors. Contractually, the Statement of Work between Babcock & Wilcox and the Department of Energy (DOE) requires that the BR-100 accept PWR fuel with a minimum burnup of 18 Gwd/mtu. B&W has interpreted the minimum burnup requirement to mean that the baseline cask should accept that fuel without change in the basket setup or geometry. If the cask were designed to a 24 Gwd/mtu assembly average burnup credit, a hardware change and capacity derating would be required to accommodate the baseline 18 Gwd/mtu fuel.

Operationally, the use of burnup credit depends on the NRC implementing a policy that will either allow administrative controls on the selective loading of casks, require measurement of burnup or k_{eff}, or a combination of the two. The DOE has a program ongoing to develop an acceptable measurement technique, which the NRC has indicated will be necessary before burnup credit is allowed. The use of any measurement technique may involve uncertainty assumptions that could further either reduce the credit allowed or increase the minimum burnup to be measured for credit.

The deliberate progress of these activities indicates that, at least until the measurement technique and is associated accuracy are approved, the credit sought should be the minimum burnup required to meet cask capacity goals.

Analytically, the use of burnup credit presents some formidable challenges. While the concept has been certified for spent fuel pool storage racks in recent years and has been infrequently accepted for spent fuel storage casks, its use in these areas are predicated on inherent excess reactivity margins such as borated water for the spent fuel pools and a lower accident probability for the storage casks. For transportation casks, those inherent margins are reduced, thus making criticality safety more dependent upon the assumptions and methods used for the analysis.

Currently the KENO[4] code, either in versions IV or Va, is accepted by the NRC for criticality analyses of fresh fuel configurations. This is based primarily upon the excellent agreement between actual measurements and KENO predictions for critical experiments which have been conducted almost exclusively with fresh fuel. Such benchmark experiments are not available for irradiated fuel at different burnup levels, thereby raising questions about the accuracy of KENO predictions on spent fuel criticality. BWFC plans to resolve this issue through further discussions with the NRC, although some critical experiments may be necessary.

The effect of axial burnup distribution in a fuel assembly on a criticality calculation is also an important issue. Generally, burnup is quoted as an assembly average value. However, for one-cycle fuel the central portions of the assembly may have burnups about 20% higher than the average, while burnups at the ends may taper off to 20 to 40% less than the average. (Two typical assembly axial burnup profiles[5] are shown in Figure 6). The actual effect may vary due to operational, core position, or geometric reasons. Because of uncertainty as to the impact of these end effects on localized criticality analyses, BWFC used only 16 Gwd/mtu instead of the 18 Gwd/mtu allowed by the DOE. Although this was done to bound the fuel assembly end effects, that value was based upon engineering judgement rather than a comprehensive study. For the final design phase of the BR-100 project, end effects will be quantified by means of KENO axial studies. This quantification will require identifying a bounding axial burnup shape for the burnup credit desired.

6. CONCLUSIONS

The studies performed by BWFC indicate that limited use of burnup credit has sufficient benefits to justify the certification risk taken. The amount of burnup credit taken should reflect fuel inventory realities and provide a real life-cycle-cost savings of about 15%. Contractual and operational issues may complicate the use of burnup credit for any cask designer or user. The hurdles to certifying a cask with burnup credit are not insignificant. It will require substantial analytical effort and perhaps some experimental data on criticality of spent fuel arrays.

7. REFERENCES

1. P. L. Holman and R. L. Fish, Neutron Absorber Material Evaluation For BR-100 Cask Fuel Cells, B&W Doc. 12-1175517-00, June 1989.

2. W. A. Wittkopf, et al., NULIF, Neutron Spectrum Generator, Few Group Constant Calculator and Fuel Depletion Code, BAW-426, Rev. 11, July 1988.

3. N. Barrie McLeod, et al., The Use of Spent Fuel Selection Strategies to Control Waste Stream Radiological Properties, JAI-317, February 1989, Table 3.2.

4. L. A. Hassler,(ed.), KENO4, An Improved Monte Carlo Criticality Program(B&W Version of ORNL Code - KENOIV, B&W Document NPGD-TM-503, Rev G, September 1987.

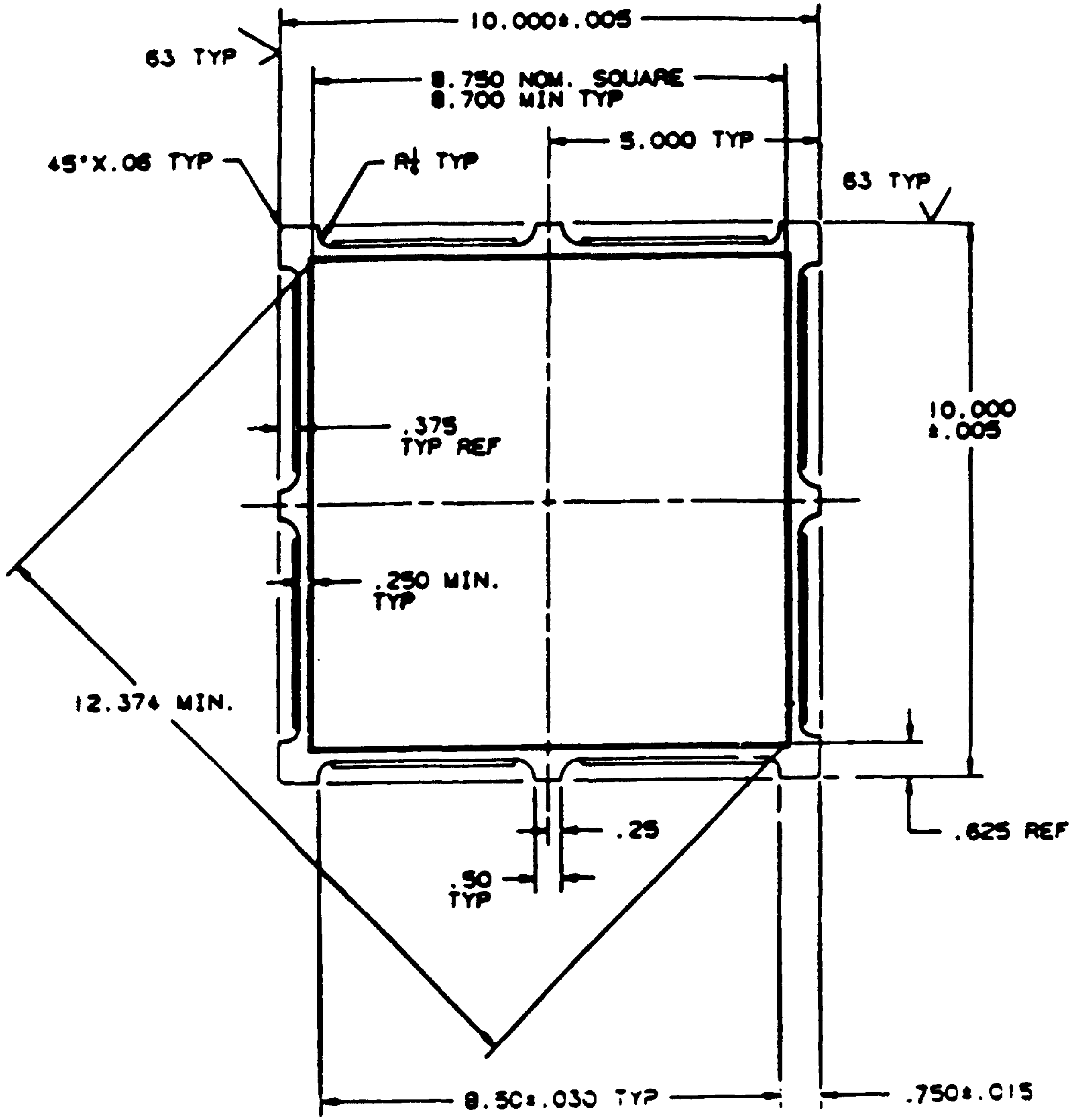
5. C. M. Hove, Axial Shapes For Taco, B&W Doc. 32-1170865-00, January, 1989.

TABLE 1. Burnup Credit Reactivity Effect

Burnup, GWD/mtu	Maximum k_{eff}
0	1.05
4	1.02
8	0.99
12	0.96
16	0.94
18	0.93
20	0.92

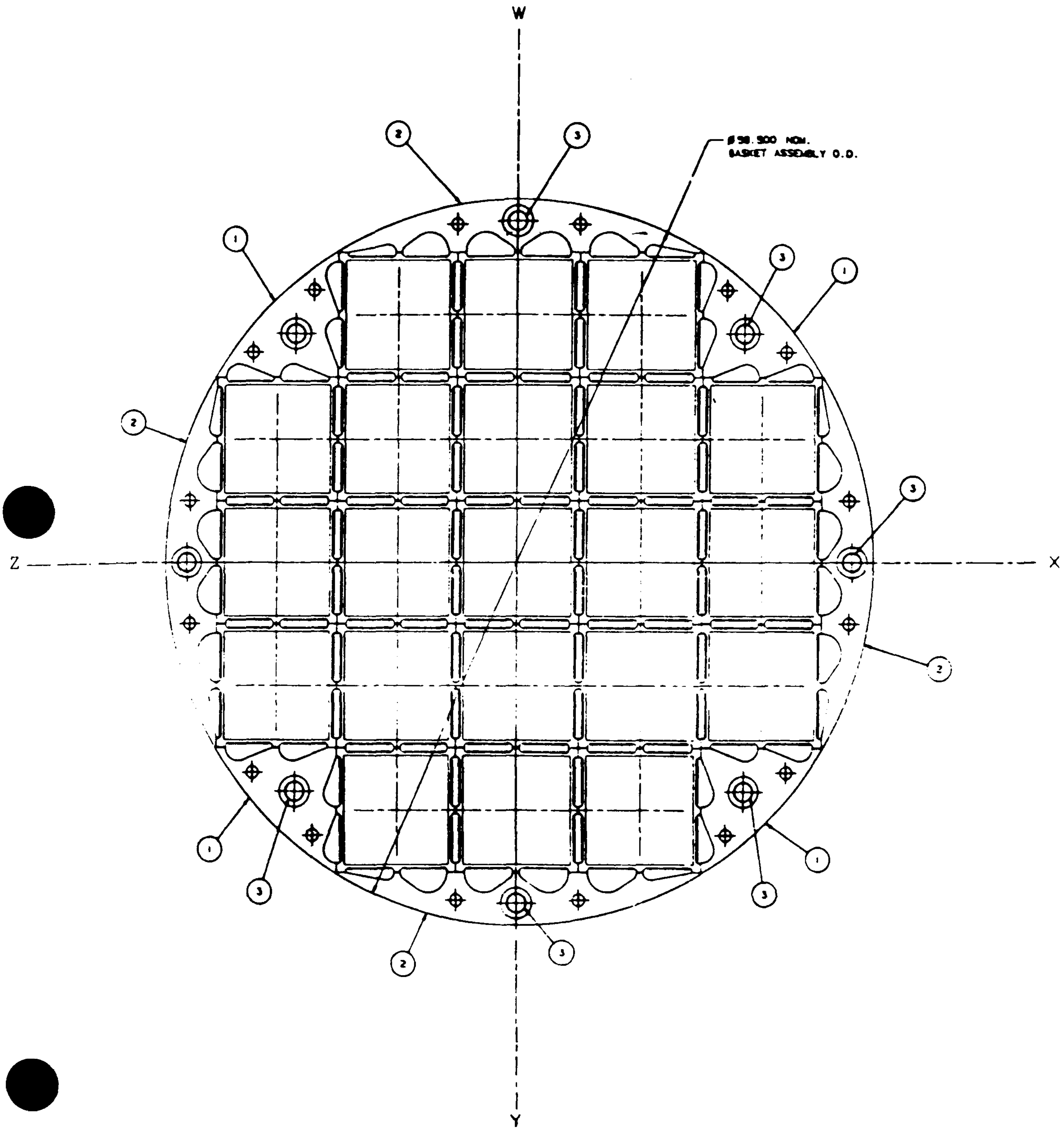
FIGURE 1

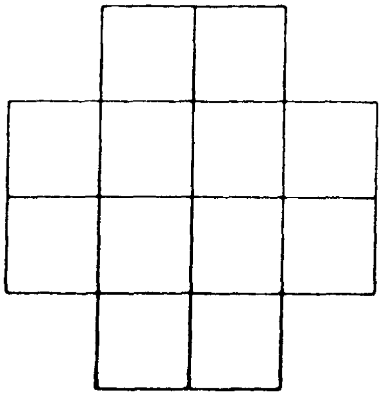
BR-100 PWR FUEL CELL CROSS-SECTION



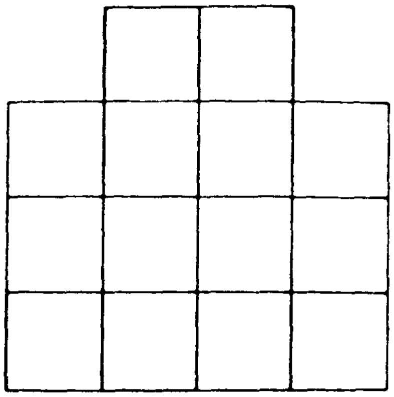
<u>FIGURE 2</u>

BR-100 PWR BASKET CROSS-SECTION

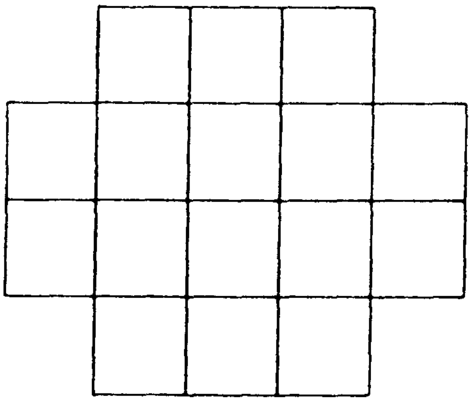




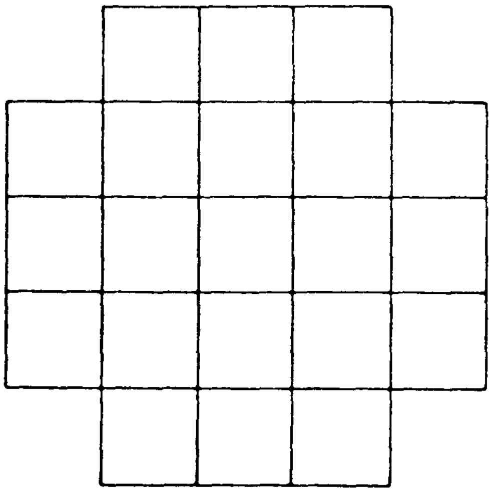
12 Fuel Cells Array
diameter = 4.472 x pitch



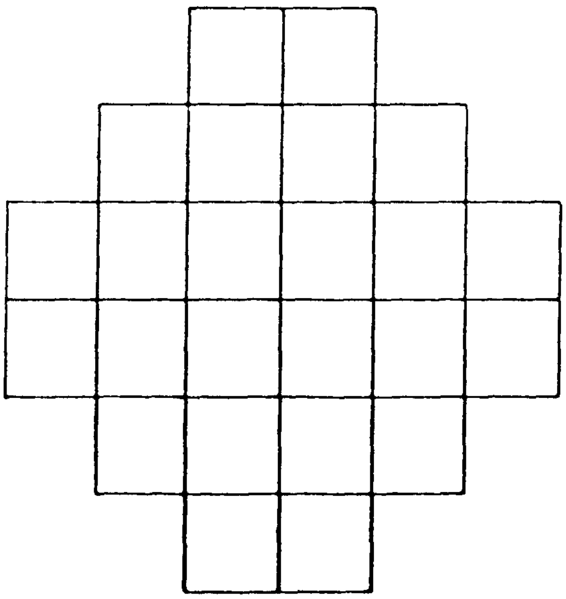
14 Fuel Cells Array
diameter = 5.154 x pitch



16 Fuel Cells Array
diameter = 5.385 x pitch



21 Fuel Cells Array
diameter = 5.831 x pitch



24 Fuel Cells Array
diameter = 6.325 x pitch

<u>FIGURE 3</u>

Diameter function of the pitch for 12, 14, 16, 21 and 24 Fuel Cells Arrays

Figure 4. Preliminary BR-100 Analyses
Burnup Credit vs Cask K-max
(4.5 w/o, 17x17, PWR Fuel, Pure Water)

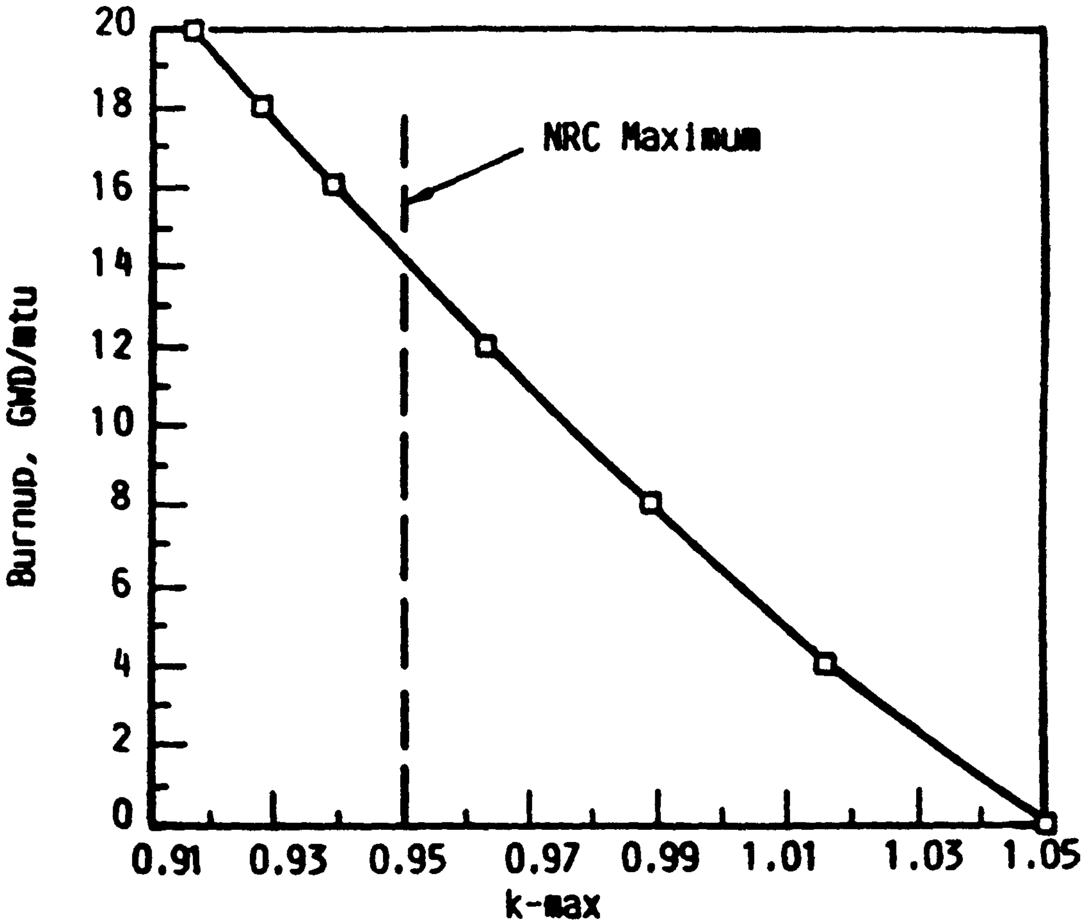


Figure 5. Preliminary BR-100 Analyses
Burnup Credit vs Gap Thickness/Cell Pitch
$Fork_{max}$ = 0.95 (4.5 w/o, 17x17 PWR Fuel, Pure Water)

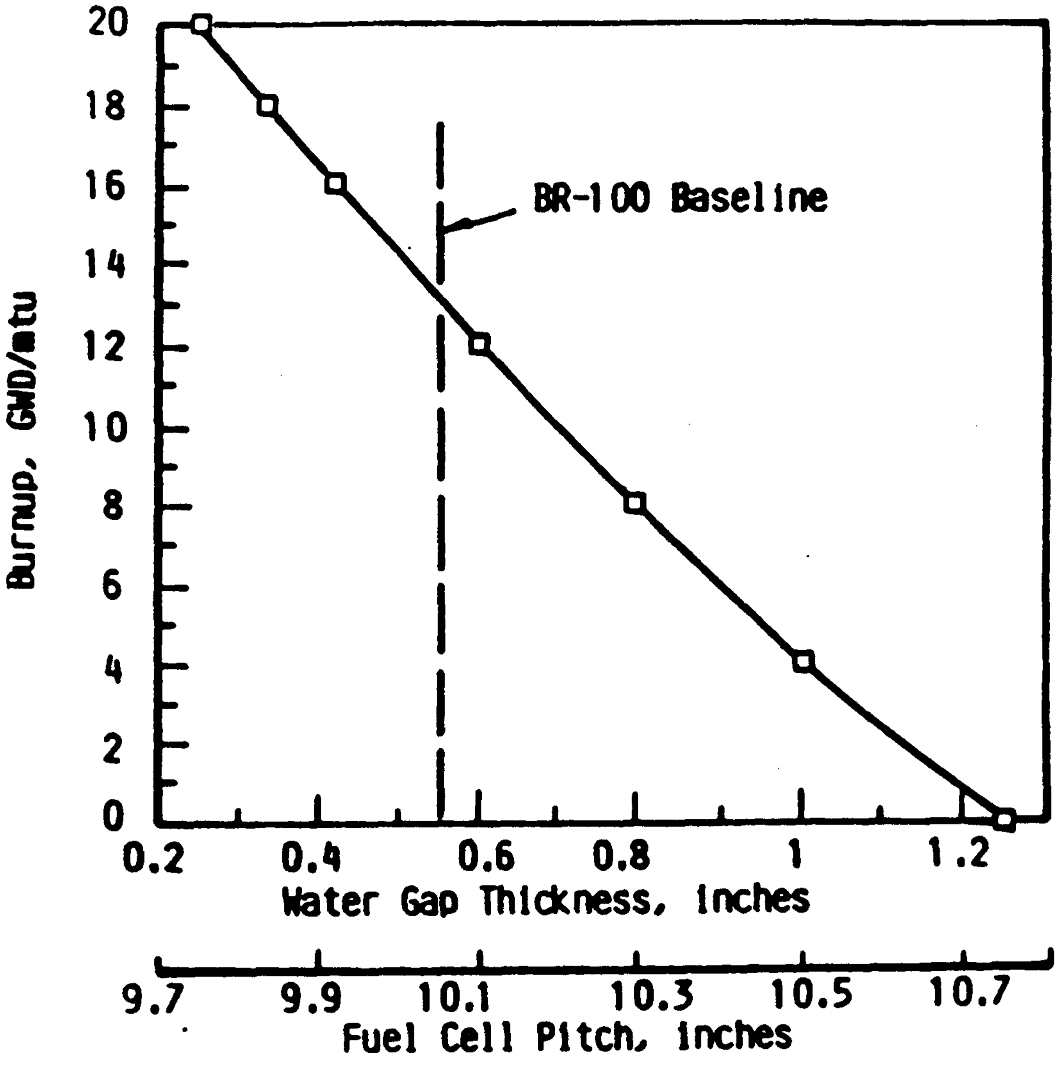
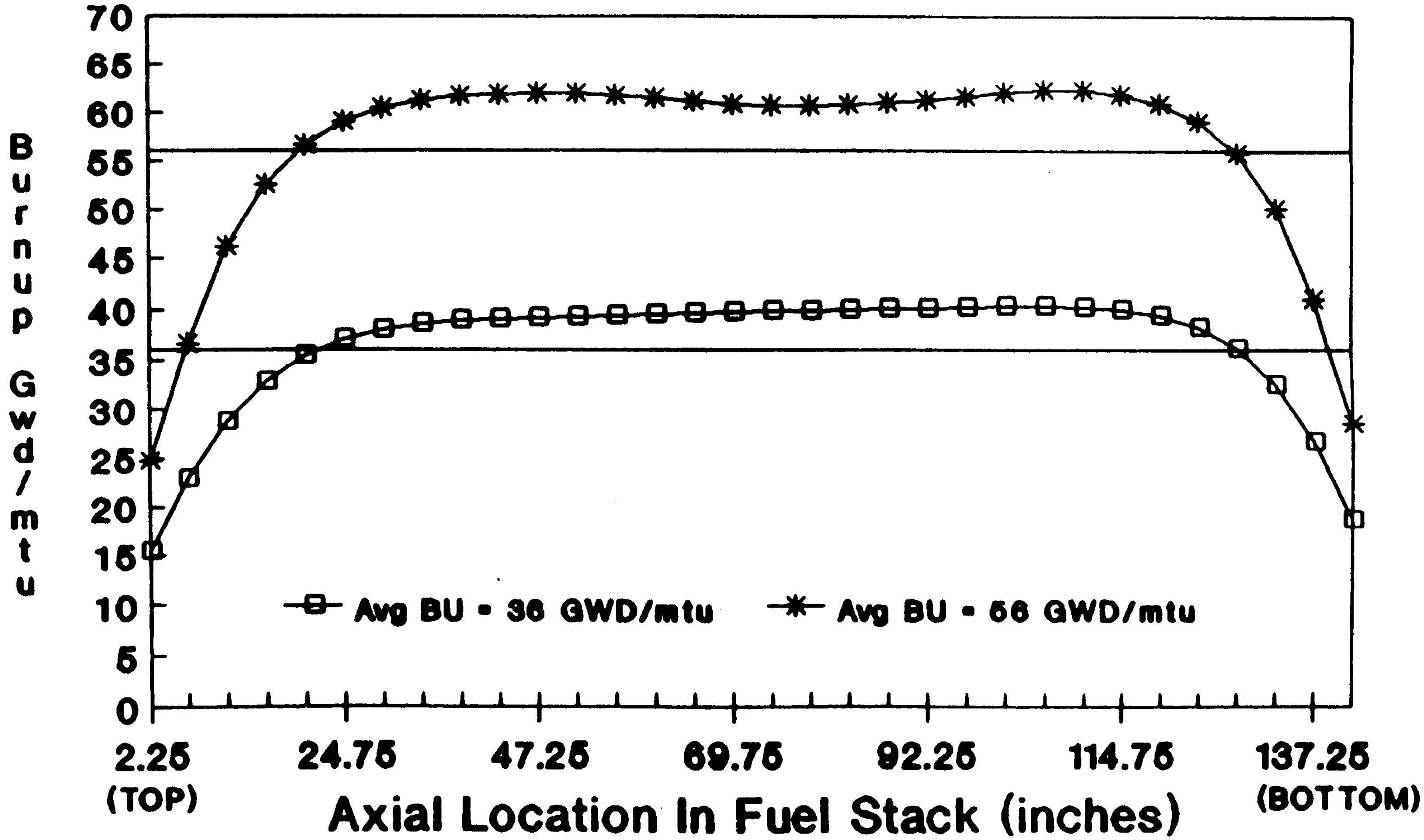


FIGURE 6

Fuel Axial Location vs Burnup



FROM-REACTOR CASK DEVELOPMENT PROGRAM TRADE-OFF STUDY

B&W BR-100 CASK

EFFECT ON CAPACITY OF REDUCING THE 2-METER DOSE RATE TO 2 MR/HR

Document No. 51-1175829-01

October 6, 1989

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EFFECT ON CAPACITY OF REDUCING THE 2-METER DOSE RATE TO 2 MR/HR

1. SUMMARY

This report is a presentation of a study performed by the B&W Fuel Company to satisfy the requirement in Section 4.10.1(2) of the contract Statement-of-Work, i.e., "Reduction of the allowable 2-meter dose rate from 10 mrem/hr to 2 mrem/hr". The current Federal limit (10CFR71) for 2-meters is 10 mr/hr. Supplemental changes to the cask which did not require alterations to the body were one option considered; the other option was to allow the body to be optimized for the payload. For both options, the hook weight limit was kept at 100-tons and the fuel source term considered to be bounding within the contractual limits. Option 1, which required only a basket change and an annular aluminum insert, resulted in a capacity of 12 PWR or 32 BWR fuel assemblies. Option 2, which required thickening the lead gamma shield and the concrete neutron shield, allowed a capacity of 14 PWR or 40 BWR assemblies.

2. DISCUSSION

The initial approach to this study was to define the approximate thickness of shielding required to reduce the dose rate down to the desired level. The baseline BR-100 designs, shown in Figures 1 and 2, have a cask body that has 4.5 inches of lead for a gamma shield and 4.5 inches of concrete for a neutron shield. The source used for this study was the 3 w/o, 35 GWd/mtu, fuel found to be bounding within the contractual guidelines by parametric evaluations based on ORIGEN2 code results. The initial estimates provided by ANISN runs on a 21 fuel assembly loading were that a 5.5/6.5 inch-thickness combination (lead/concrete, respectively) would yield a dose rate of 1.5 mr/hr (0.9 neutron and 0.6 gamma) and a 5.0/6.5 inch-thickness combination would yield 2.3 mr/hr

(1.0 neutron and 1.3 gamma).

Because the number of fuel assemblies was reduced, the total heat loading for the cask was also reduced. Thermal concerns were thus minimized and not an active part of the design iteration process.

The basket structure used on the BR-100 baseline design, individual cell extrusions assembled into a unit structure, was assumed to be continued for any iteration. Because the mass loading was reduced with the lower payload, the structural margins would be greater for any of the study designs. No further discrete structural analysis was performed on the proposed designs.

The lack of change in the cell pitch within the basket structure from the baseline design also made further criticality analyses unnecessary.

3. PROPOSED DESIGNS

3.1 SUPPLEMENTAL CHANGE ONLY (OPTION 1)

The baseline BR-100 has 4.5 inches of lead and 4.5 inches of concrete; comparison to the scoping ANISN runs indicated an additional 2 inches of concrete and 0.75 inches of lead might be required. The initial approach was to take the inner cavity diameter of the BR-100, 58.5 inches, and fit a gamma shield within it. A new basket would then be designed to fit within the internal shield. Reducing the outer diameter of the fuel array also would reduce the total source, so it was decided that supplementary neutron shielding would not be added for the first iteration. The additional shielding required reducing payload both for mass conservation to meet the 100-ton hook limit and for spatial requirements. An internal shield with a thickness of either 6 inches of aluminum, 2.5 inches of steel, 1.5 inches of lead, or 1 inch of depleted uranium would have provided equivalent shielding, but the lead or DU shields would not have increased the payload and would have complicated the basket arrangement. The aluminum was chosen because of its simplicity, weight and heat transfer properties. Borated aluminum was not used because the neutrons penetrating the cask body are fast, not thermal, and boron without hydrogen in that location would have little shielding effectiveness.

Figures 3 and 4 show the proposed "add-on" designs. The BWR configuration uses a similar internal shield, but with a 5.5 inch thickness (later evaluations have indicated that both designs could use a 5.5 inch shield). Supplemental shielding on the bottom and top is provided by a 1.5-inch thick stainless steel plate added to the inside of the impact limiters. Capacity of the design is 12 PWR or 32 BWR assemblies.

3.2 REDESIGNED CASK BODY (OPTION 2)

The cask body was redesigned to take advantage of both the reduction in total source and the excellent gamma/neutron shielding properties of the borated concrete. We initially assumed a lead thickness of 5 inches and a concrete thickness of 7 inches and used the steel shell thickness of the baseline design, 1 inch inner and 1.75 inch outer. Using the 100-ton hook limit as the bounding requirement, the largest cavity ID that could accommodate a payload was then determined. That diameter, 52.75 inches, was set by the 40 BWR configuration and could also accommodate a 14 PWR basket. Additional shielding is provided in the ends by adding 1.0 inches of lead in the shield plug and the bottom. Figures 5 and 6 show the revised BR-100 cask design for the 2 mr/hr limit.

4. RESULTS

The baseline BR-100 with the add-on aluminum inner shield and the heavier impact limiters was discretely analyzed using ANISN for side dose rates. The resulting side dose rate was 1.8 mr/hr (1.0 neutron and 0.8 gamma). The additional end shielding required to obtain an end dose rate of 2 mr/hr was derived by scaling the results of previous baseline QAD and ANISN runs that had established equivalent shielding worths. The hook weight of the limiting PWR configuration was determined to be approximately 199,500 pounds, including water normally drained before lifting. The gross vehicle weight of this package on the railcar was estimated to be 259,000 pounds.

The redesigned BR-100 was similarly analyzed. The side dose rate was 1.7 mr/hr (0.7 neutron and 1.0 gamma) while the end dose rate was scaled to be 2 mr/hr. The hook weight of this design was calculated to be 200,000 pounds in the limiting BWR configuration. The corresponding gross vehicle weight was about 261,000 pounds.

5. CONCLUSIONS

Adaptation of the baseline BR-100 design to add-on hardware was surprisingly easy and should not be an expensive alternative. The hardware chosen could probably be licensed with reasonable care in a relatively short period of time. The reduction in payload from 21 PWR assemblies to 12 assemblies or 52 BWR assemblies to 32 is substantial, but may be offset by the priority of other programmatic issues. Use of an existing cask adaptation could prove attractive for certain applications or special cases. The BR-100 seems to offer an easy way to achieve that flexibility.

The revised BR-100 did not differ greatly from the baseline in construction, but the approximately 6-inch reduction in cavity diameter did reduce the capacity from 21 PWR assemblies to 14

(33%) and 52 BWR assemblies to 40 (23%). The cost of the cask should not be significantly different and total life cycle cost would probably rise only about 20% or less due to decreased turnaround times. The relative importance of these factors is a programmatic issue to be determined by DOE. The BR-100 offers design flexibility to easily accommodate changes in program priority.

TRADE-OFF STUDY: 2 mrem/h limit dose rate at 2 m

BASELINE CASK BODY

21 PWR CONFIGURATION

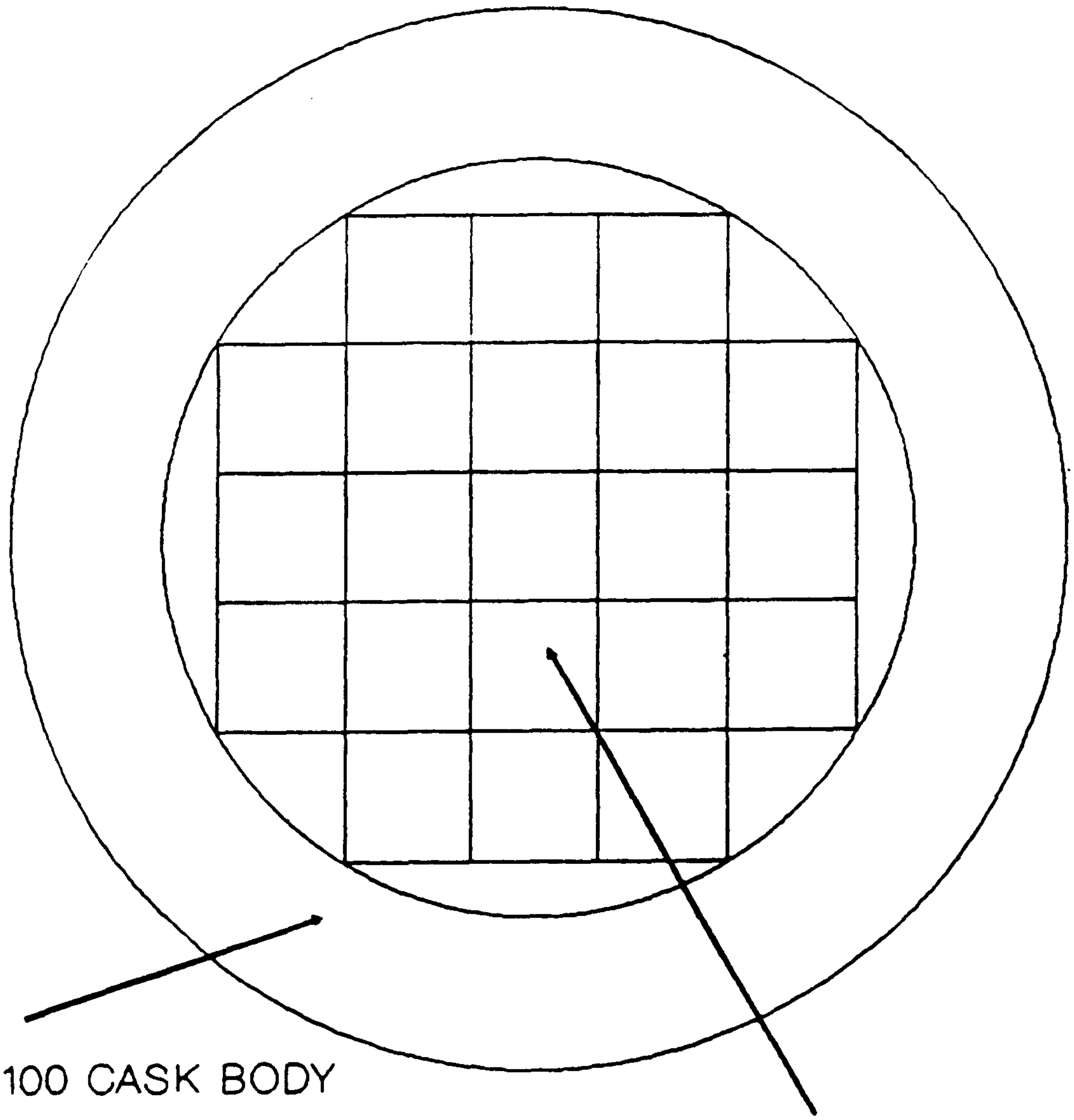

FIGURE 1

TRADE-OFF STUDY: 2 mrem/h limit dose rate at 2 m

BASELINE CASK BODY

52 BWR CONFIGURATION

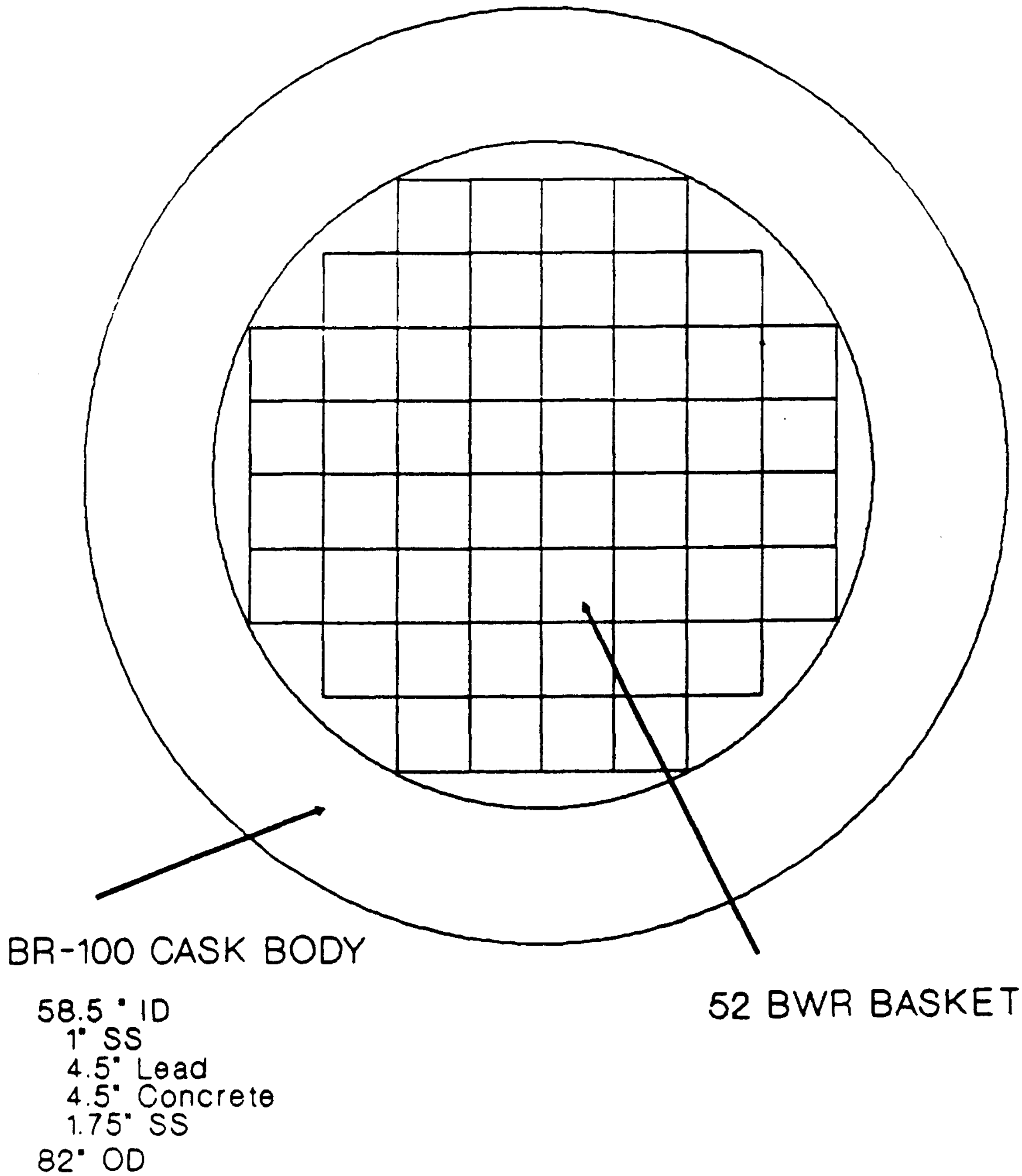

FIGURE 2

TRADE-OFF STUDY: 2 mrem/h limit dose rate at 2 m

MAX. CAPACITY W/O MODIFICATION OF CASK BODY

12 PWR CONFIGURATION

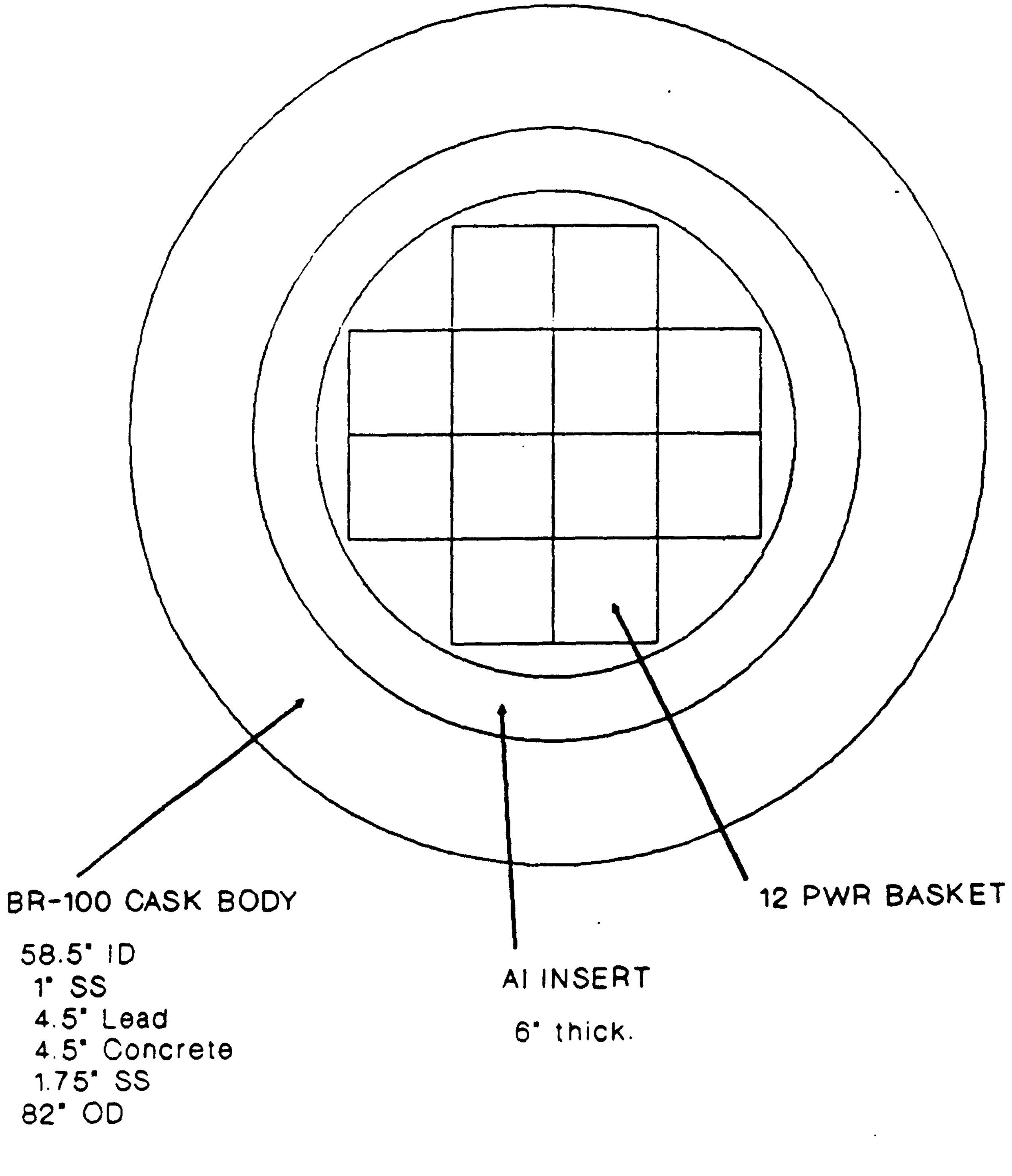

FIGURE 3

TRADE-OFF STUDY: 2 mrem/h limit dose rate at 2 m

MAX. CAPACITY W/O MODIFICATION OF CASK BODY

32 BWR CONFIGURATION

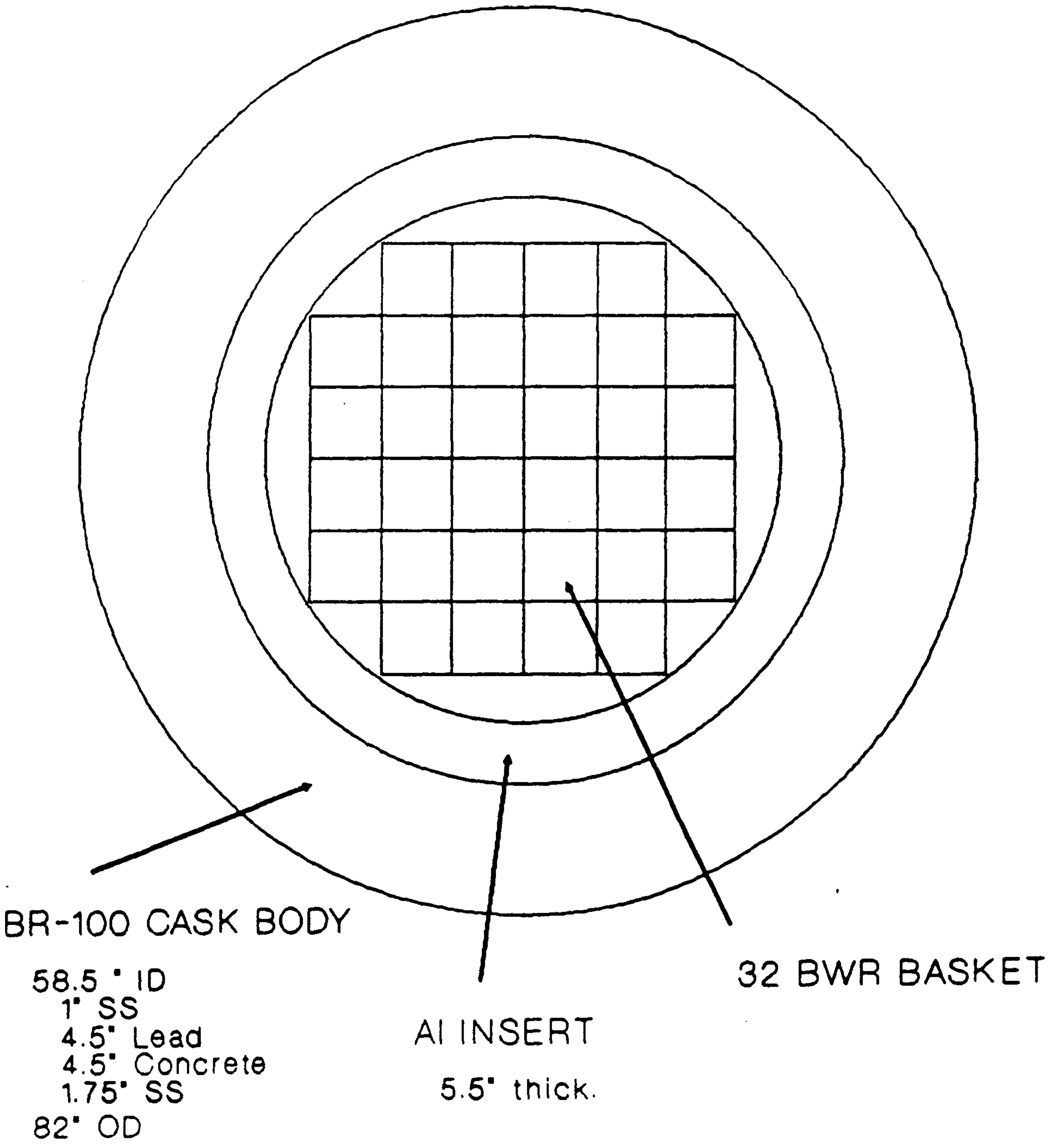

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FIGURE 4

TRADE-OFF STUDY: 2 mrem/h limit dose rate at 2 m

MAX. CAPACITY W/ MODIFICATION OF CASK BODY

14 PWR CONFIGURATION

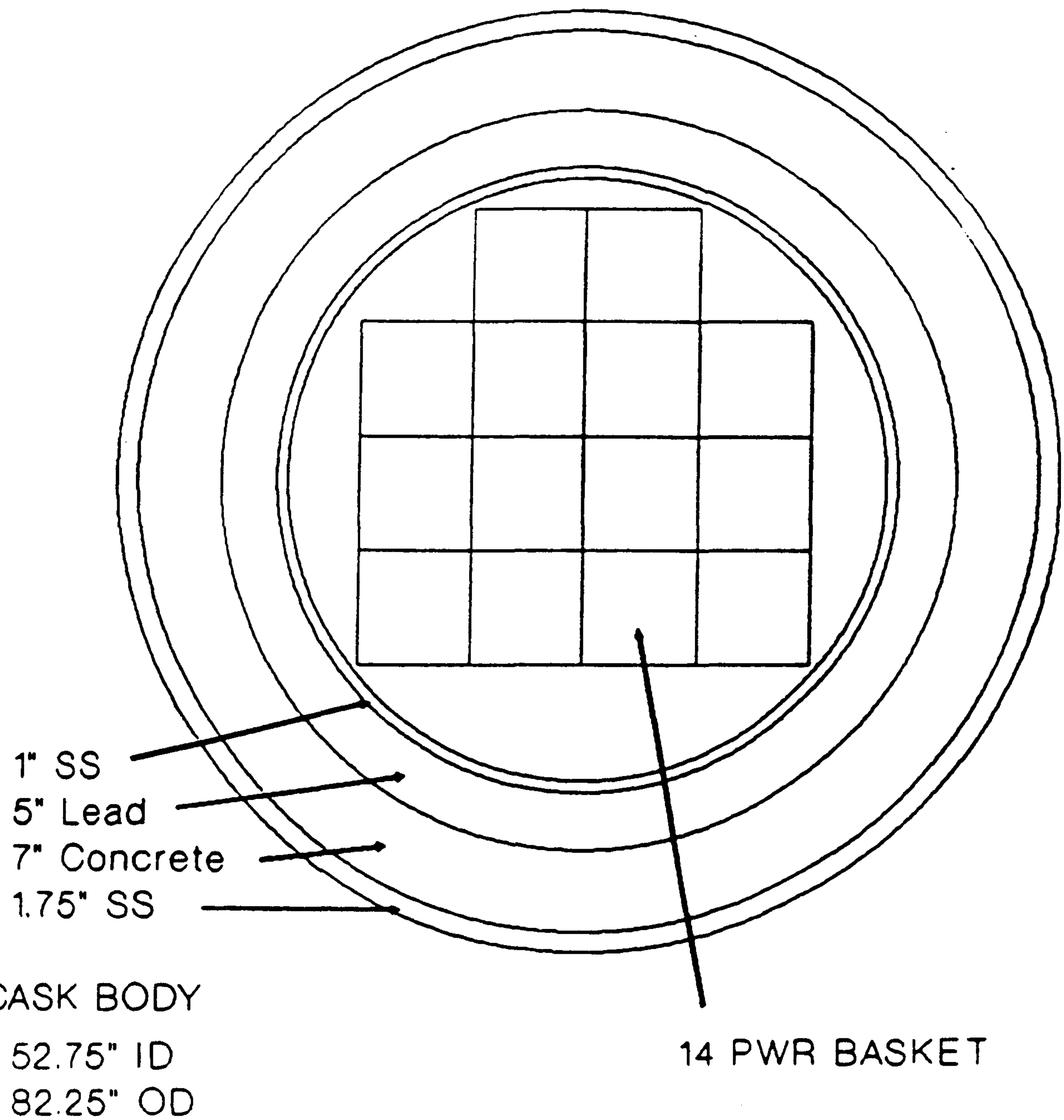

FIGURE 5

TRADE-OFF STUDY: 2 mrem/h limit dose rate at 2 m

MAX. CAPACITY W/ MODIFICATION OF CASK BODY
40 BWR CONFIGURATION

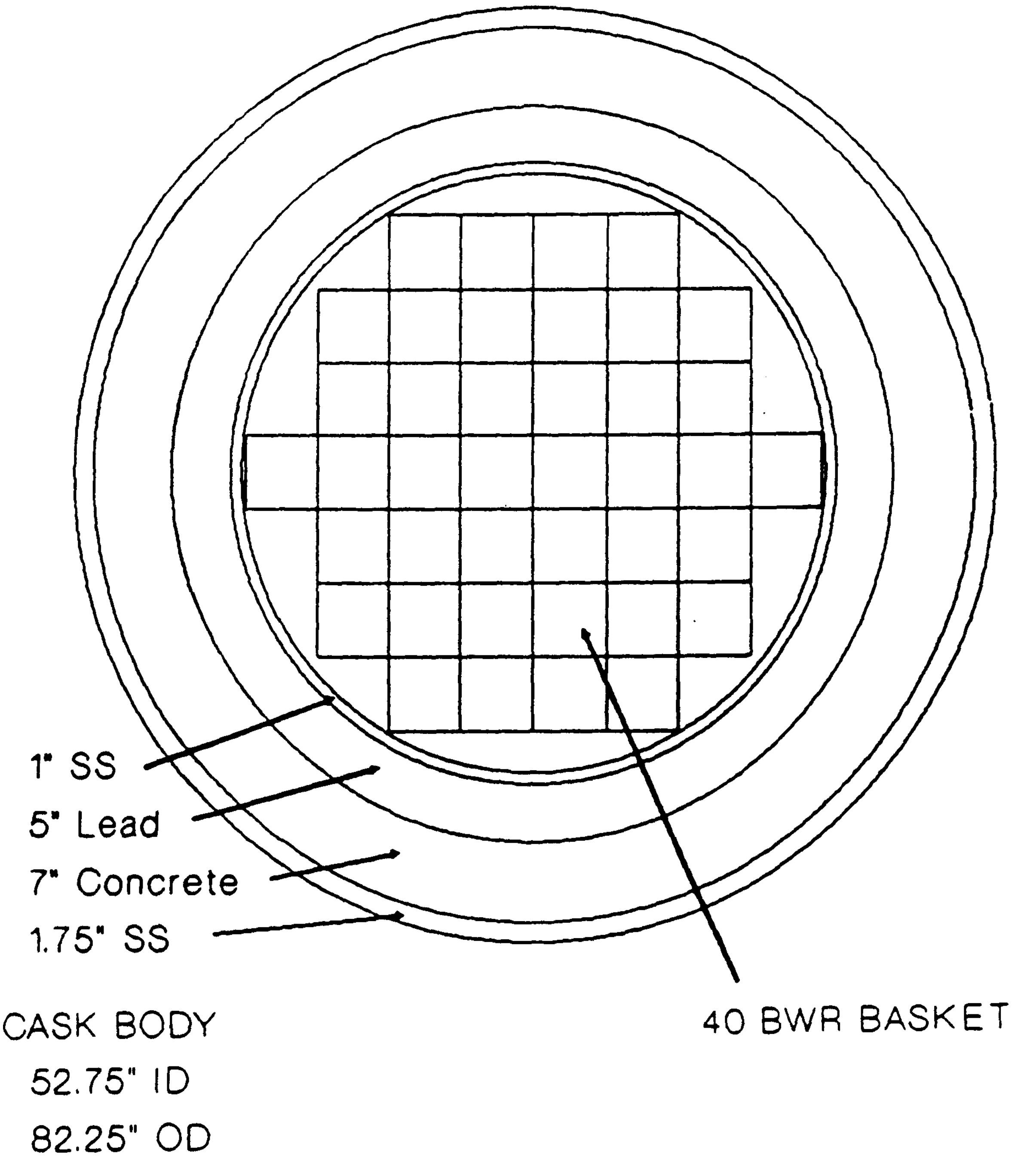

FIGURE 6

FROM-REACTOR CASK DEVELOPMENT PROGRAM TRADE-OFF STUDY

B&W BR-100 CASK

EFFECT ON CAPACITY AND COST OF TRANSPORTING
5-YEAR COOLED FUEL VS 10-YEAR COOLED FUEL

Document No. 51-1176034-00

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EFFECT ON CAPACITY AND COST OF TRANSPORTING

5-YEAR COOLED FUEL VS 10-YEAR COOLED FUEL

1. SUMMARY

This report presents the results of a study performed by the B&W Fuel Company (BWFC) to investigate the capacity and cost impacts of transporting 5-year cooled fuel in the BR-100, a cask originally designed to transport 10-year cooled fuel. Using the same burnup and enrichment criteria as in the contract Statement-of-Work (35 Gwd/mtu (PWR)/30 Gwd/mtu (BWR) and 3.0-4.5%, respectively), three options were evaluated. The first option involved leaving the basket designs unchanged—either PWR or BWR—and determining what payload revisions were required. The second option looked at what basket changes could be made to optimize the BR-100 for 5-year cooled fuel. The third option evaluated if a combination of 5- and 10-year cooled fuel could provide an efficient way of transporting spent fuel.

Shielding requirements were always the limiting factor in determining if a particular combination of fuel would be acceptable. Thermal analysis was also performed on most combinations to confirm adequacy in that area. Structural and criticality analyses were not performed due to the lack of effect that cooling time has on those analyses. The hook weight of each configuration was checked for adherence to the 100-ton limit. No changes were required for the BR-100 to carry a full load of 52 BWR fuel assemblies, but three options were reviewed to accommodate PWR fuel.

Option 1 was fairly straight-forward and involved, replacing all of the fuel in the baseline basket outer row with aluminum shielding inserts. This reduced the BR-100 capacity to 9 PWR fuel assemblies.

In Option 2, an aluminum annular insert 2.25 inches thick was hypothetically placed just inside the cask inner wall, reducing the cavity diameter to 54 inches. A basket was successfully designed for that cavity which contained 16 PWR assemblies and met shielding, thermal, and weight criteria.

For Option 3, it was determined that the basket outer row could be filled with 10-year cooled fuel (12 assemblies) the inner portion filled with 5-year cooled fuel (9 assemblies), and still meet analytical limits.

Overall, there was no reduction in capacity or increase in life-cycle-cost (LCC) for the BR-100 cask to carry 5-year cooled BWR fuel instead of 10-year fuel. There were three options feasible for PWR 5-year cooled fuel; Options 2 and 3 appear to be the most attractive, with Option 2 having an estimated 20% higher LCC than if handling 10-year fuel and Option 3 having possibly no increase in LCC than if shipping 10-year fuel. Option 1 has an estimated 55% increase in LCC for transporting PWR 5-year fuel rather than 10-year fuel.

2. INTRODUCTION

The purpose of this document is to satisfy the requirement in Section 4.10.1.3 of the contract Statement-of-Work to "... conduct trade-off and impact evaluations of the following design considerations on cask payload capacities and costs: ...Transportation of fuel aged 5 years after discharge vs design basis (10 year old) fuel." BWFC has designed a cask, the BR-100, which efficiently transports 10-year cooled fuel, either PWR or BWR in separate baskets, and has a baseline LCC as determined using an internally developed code. BWFC's goal in performing the investigations described herein was to optimize the payload and LCC of the BR-100 for 5-year fuel without requiring any rework or redesign of the cask body and then to compare those values to the payload/LCC for 10-year fuel.

Section 3 of this report describes the baseline BR-100 cask and documents its performance in transporting design basis spent fuel. It also documents the 5- and 10-year fuel parameter assumptions that will be used for assessing performance adequacy. Section 4 details the design options investigated and gives particulars on their performance. Section 5 examines the rationale and LCC for each option presented. Section 6 is the Conclusion and recounts the results of the investigation.

3. BASELINE ASSUMPTIONS

The baseline BR-100 cask is shown in Figures 1, 2, and 3. It has a cavity 58.5 inches in diameter and 181 inches long and has side walls made of—from inside out—1 inch of stainless steel, 4.5 inches of lead, 4.5 inches of borated concrete, and 1.75 inches of stainless steel. Unique baskets for either PWR or BWR fuel have a capacity of 21 or 52 assemblies, respectively. The baskets are made of aluminum cells and supports which efficiently transfer heat from the fuel to the cask inner wall. The hook weight of the BR-100, including interstitial water and the handling equipment, is 100 tons in either PWR or BWR configuration.

The design basis payload for the BR-100 is fuel that is 10-year cooled and has a burnup of 35 Gwd/mtu (PWR) or 30 Gwd/mtu (BWR) and an enrichment of 3.0-4.5%. Sensitivity studies performed by BWFC have shown that, at the burnup levels selected, the lower enrichment has a significantly higher source term and decay heat rate than the higher enrichment. BWFC has also taken into account the axial profile of fuel burnup. As shown in Figure 4, PWR spent fuel has a substantial axial span where the burnup level is up to 13% greater than the assembly average (the analagous value for BWR fuel is 20%). The decay heat rates for those burnups were calculated using ORIGEN2 and are shown in Tables 1 and 2 for 5-year and 10-year cooled fuel. Because of the preliminary nature of the cask design and the nominal values used for some geometric and material properties, an additional 1.10 conservatism factor was used for shielding calculations in determining source strength at the limiting burnup levels.

4. DISCUSSION

Previous conservative analyses to support the Preliminary Design Baseline have shown that the PWR configuration of the BR-100 did not have a large margin in shielding, while temperature margins and BWR shielding margins were fairly large. Based on those results, BWFC made a discrete ANISN shielding analysis and a thermal evaluation of a more conservative case (366 watts/assembly instead of 348 for 5-year fuel) to see if a full load of 5-year cooled BWR fuel (30 Gwd/mtu, 3.0 w/o) could be accommodated within the standard basket configuration. Both in terms of dose rate (9.4 mr/hr) and temperatures, the loading was acceptable.

The PWR investigation was directed toward three options. The first option was simply to replace the fuel in the basket outer row with aluminum inserts (1-inch wall thickness square tubes) which would provide the correct amount of shielding. The total reduction in the mass of the fuel transported and its concentration in the center of the basket also contributed to a reduction in the dose rate to an acceptable level. The total heat load from this configuration was judged to be so low (8.1 kW) as not to require a discrete analysis to assure adequate thermal performance. The dose rate for this array was also not discretely calculated, but was compared to a model with a larger source and equivilent shielding that produced a dose rate of 10.4 mr/hr. It was concluded that the dose rate for Option 1 would be less than 10 mr/hr. The replacement of 12 fuel assemblies with aluminum inserts reduces the payload, and hence the hook weight, by at least 10,000 pounds. Option 1 consequently has an acceptable performance, but only carries 9 fuel assemblies per load.

The second option for PWR review was to investigate what basket changes could be made to optimize the 5-year fuel payload. A 16-assembly basket of 10-inch square BR-100 PWR fuel cells has an outer diameter of 53.85 inches. Dividing the annular gap between this basket and the cask inner diameter (58.5 inches) and leaving some room for insertion, an aluminum annulus 2.25 inches thick was modeled for ANISN analysis (Figure 5). The resultant dose rate was 10.4 mr/hr (2.5 neutron/7.9 gamma). Because of the conservative modeling and input techniques, and the use of aluminum instead of a heavier metal for the annular insert, it was judged that this configuration was acceptable from a shielding perspective. A discrete thermal analysis was not run on this array; the total heat load of 14.45 kW was less than a previously-run successful 17.58 kW case (21 assemblies), thus it was evaluated to be adequate thermally. Again, the reduction in total fuel payload meant a reduction in hook weight from the 100-ton limit. Option 2 was evaluated to have a achievable capacity of 16 PWR assemblies.

Option 3 was pursued to determine if an efficient way of transporting a combination of 5-year and 10-year cooled PWR fuel could be found. The basket outer row was hypothetically filled with twelve 10-year cooled fuel assemblies to provide shielding to the hotter 5-year cooled fuel in the inner nine locations (Figure 6). The resultant array was similar to a previous thermal analysis run that had slightly hotter fuel in the inner locations (956 kW vs 903 kW for 5-year cooled fuel using assembly average burnups), so the acceptable results from that run were used to justify the thermal adequacy of Option 3. Results from previous ANISN runs with similar configurations have shown that cooler fuel in the outer row shields the hotter inner assemblies and that external dose rates are little, if any, different from a full load of the cooler fuel. Based on those results, BWFC anticipates a 2-m dose rate less than 10 mr/hr for the Option 3 configuration. Because the Option 3 payload is identical to the baseline, the hook weight should not be different and would be acceptable. Option 3, with its full compliment of 21 assemblies, is an efficient alternative for shipping 5-year and 10-year cooled fuel.

The results from evaluation of all PWR and BWR options considered are shown in Table 3.

5. RATIONALE/LIFE-CYCLE-COST

The ability to fully load 5-year cooled BWR fuel into the BR-100 is a significant feature which provides DOE with a flexibility that could be extremely helpful in transporting fuel from reactors recently decommissioned. Although the raw LCC numbers do not reflect any improvement, the value to the system is increased at no additional charge.

The PWR options have differing LCC impacts, but also do not quantify the flexibility values that would benefit DOE. Option 1, which derated the baseline basket and used aluminum inserts to provide shielding, has a increase of over 50% in LCC because of the drop in capacity from 21 to 9 assemblies. The proportion of trips that would use this option to carry fuel as opposed to the baseline 10-year aassumption is a direct factor to the 54% that BWFC cannot assign a value to. An additional increase of less than 1% would accrue because of additional development costs (about $300K) to license the use of inserts as an option. The fabrication cost (about $50K per cask) would also add to LCC.

PWR option 2 has an approximate 20% increase in LCC due to its capacity derating from 21 to 16 assemblies; that would also have to be factored by the proportion of use of the annular insert. An increase of about 1% in LCC would apply because of additional development/licensing costs (about $.75-1.0M if drop testing is required). Fabrication cost would be about $150K per basket in production.

PWR option 3 has no significant increase in LCC, due to the retention of a full load of fuel. Should this method be successfully pursued, it would give DOE a significant amount of flexibility in accepting fuel in an efficient manner from utilities. Although it requires some administrative overcheck during loading and could increase turnaround time at the reactor, the NRC should be consulted to see if this is an acceptable method of shipment.

CONCLUSIONS

BWFC investigated the capacity of the BR-100 to carry 5-year cooled fuel in the baseline basket and in a redesigned basket. The use of "shadow-shielding" was also checked. The results were that a full load of 5-year cooled BWR fuel could be accomodated in the baseline basket. Several options are available for PWR fuel, ranging from carrying only 9 assemblies in the baseline basket to carrying only 16 assemblies in a redesigned basket; another alternative is mixing a full 21-assembly load of 5-year and 10-year cooled fuel in a particular array to allow self-shielding to provide maximum benefits.

This report provides DOE and EG&G with data which indicates that the BR-100 cask can provide significant flexibility in managing the efficient transportation of commercial spent fuel.

Table 1

PWR FUEL PARAMETERS - 465 Kg/ASSEMBLY

BURNUP (Gwd/mtu)	ENRICHMENT (w/o)	COOLING TIME (Years)	DECAY HEAT (Watt Assy)
35	3.0	5	903
35	3.0	10	574
35	4.5	5	812
35	4.5	10	533
35x1.13 = 40	3.0	5	1066
40	3.0	10	682
40	4.5	5	958
40	4.5	10	630

Table 2

BWR FUEL PARAMETERS - 176.8 Kg/ASSEMBLY

BURNUP (Gwd/mtu)	ENRICHMENT (w/o)	COOLING TIME (Years)	DECAY HEAT (Watt Assy)
30	3.0	5	279
30	3.0	10	178
30	4.5	5	256
30	4.5	10	169
30x1.2 = 36	3.0	5	348
36	3.0	10	223
36	4.5	5	315
36	4.5	10	208

Table 3
Key BR-100 Performance Values

	Design Limit	PWR				BWR	
		10-Yr Baseline	Option 1	Option 2	Option 3	10-Yr Baseline	5-Yr Case
Capacity (F.A.)	—	21	9 5-yr	16 5-yr	12 10-yr 9 5-yr	52	52
Dose Rate @2M (mr/hr) (neutron/gamma)	10	9.6 (3.8/5.8)	<10	10.4 (2.5/7.9)	<10	5.0 (2.6/2.4)	9.4 (1.8/7.6)
Hook Weight (tons)	100	99.7	95	99.5	100	100	100
Maximum Outer Surface Temp (°F)	—	195	Not Run	Not Run	214**	184	220**
Maximum Concrete Temp (°F)	250*	210	Not Run	Not Run	233**	198	242**
Maximum Aluminum Temp (°F)	350	274	Not Run	Not Run	327**	239	309**
Maximum Rod Clad Temp (°F)	680	364	Not Run	Not Run	456**	285	372**
Total Heatload (KW)	18	12.1	8.1	14.4	15.0	9.3	14.5

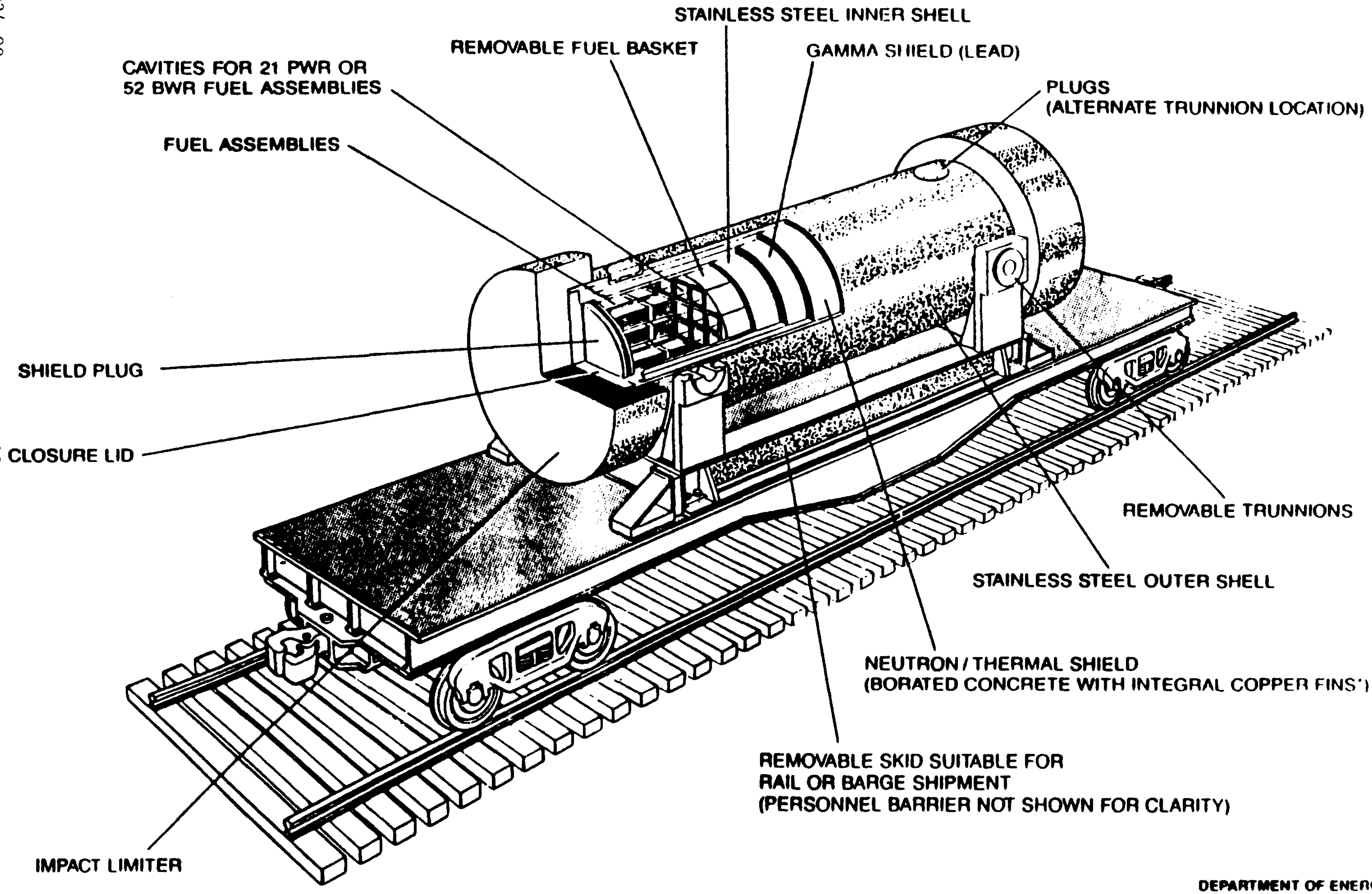
* Steady-state, not applicable for transient events

** Conservative decay heat value used; discrete run would have lower results

B&W Fuel Co

51-1176034-00

BABCOCK & LCOX BR-100
100 TON RAIL / BARGE CASK



DEPARTMENT OF ENERC
CONTRACT NO DE-AC07-88ID12A

* PATENTED BY
ROBATEL SA

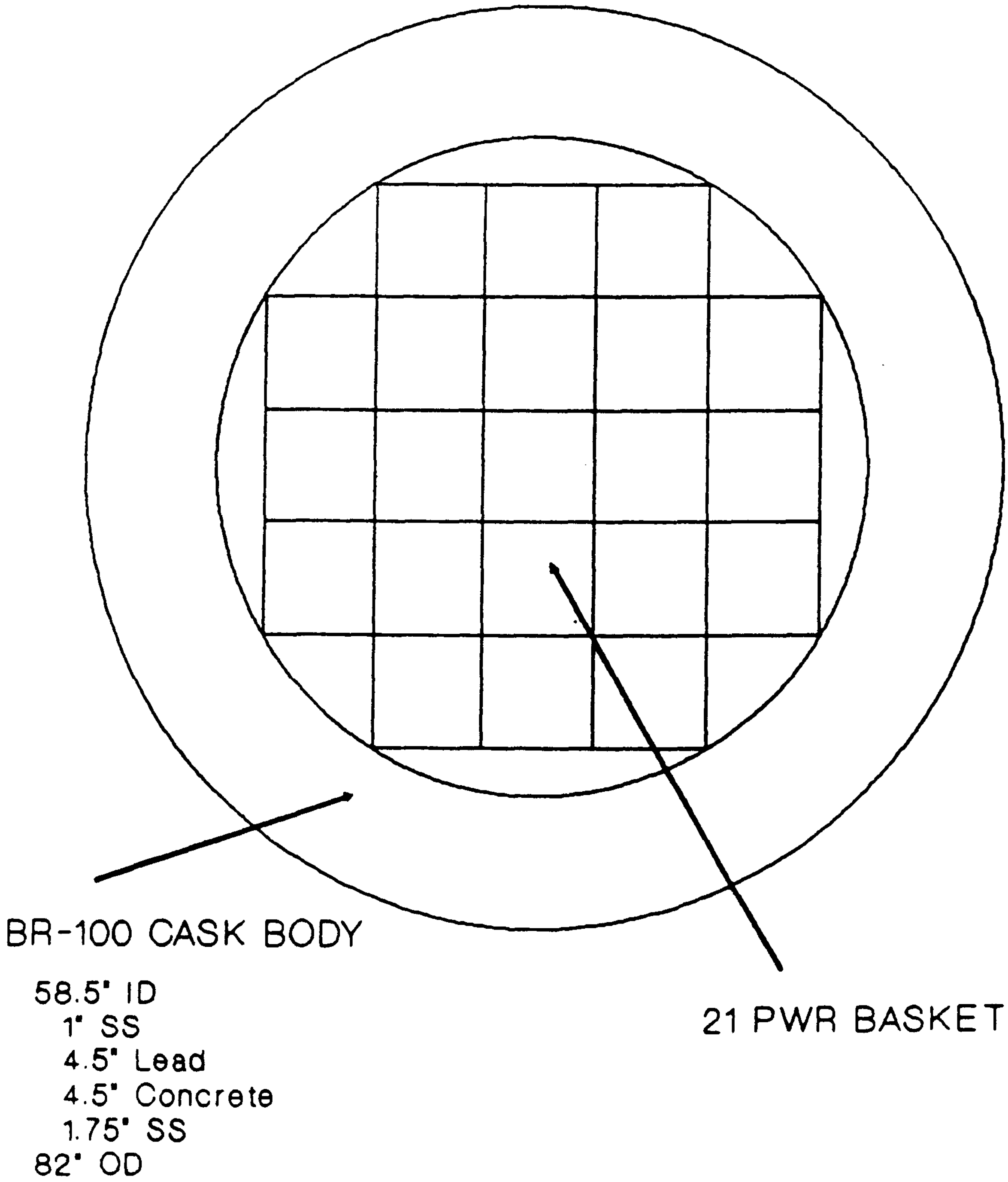
APRIL 19

FIGURE 2

TRADE-OFF STUDY: 5 years cooled fuel

BASELINE CASK BODY

21 PWR CONFIGURATION



TRADE-OFF STUDY: 5 years cooled fuel

BASELINE CASK BODY

52 BWR CONFIGURATION

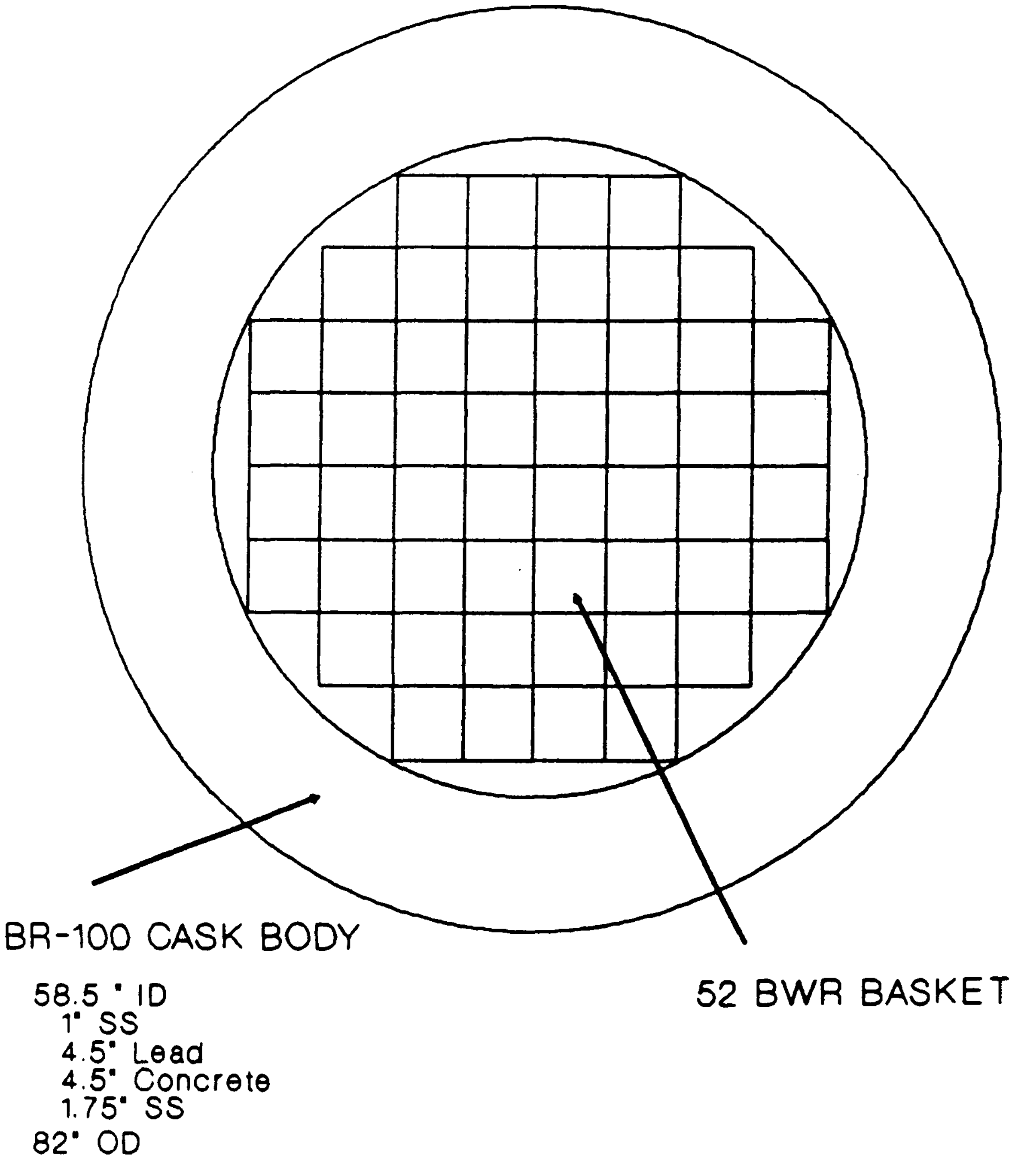


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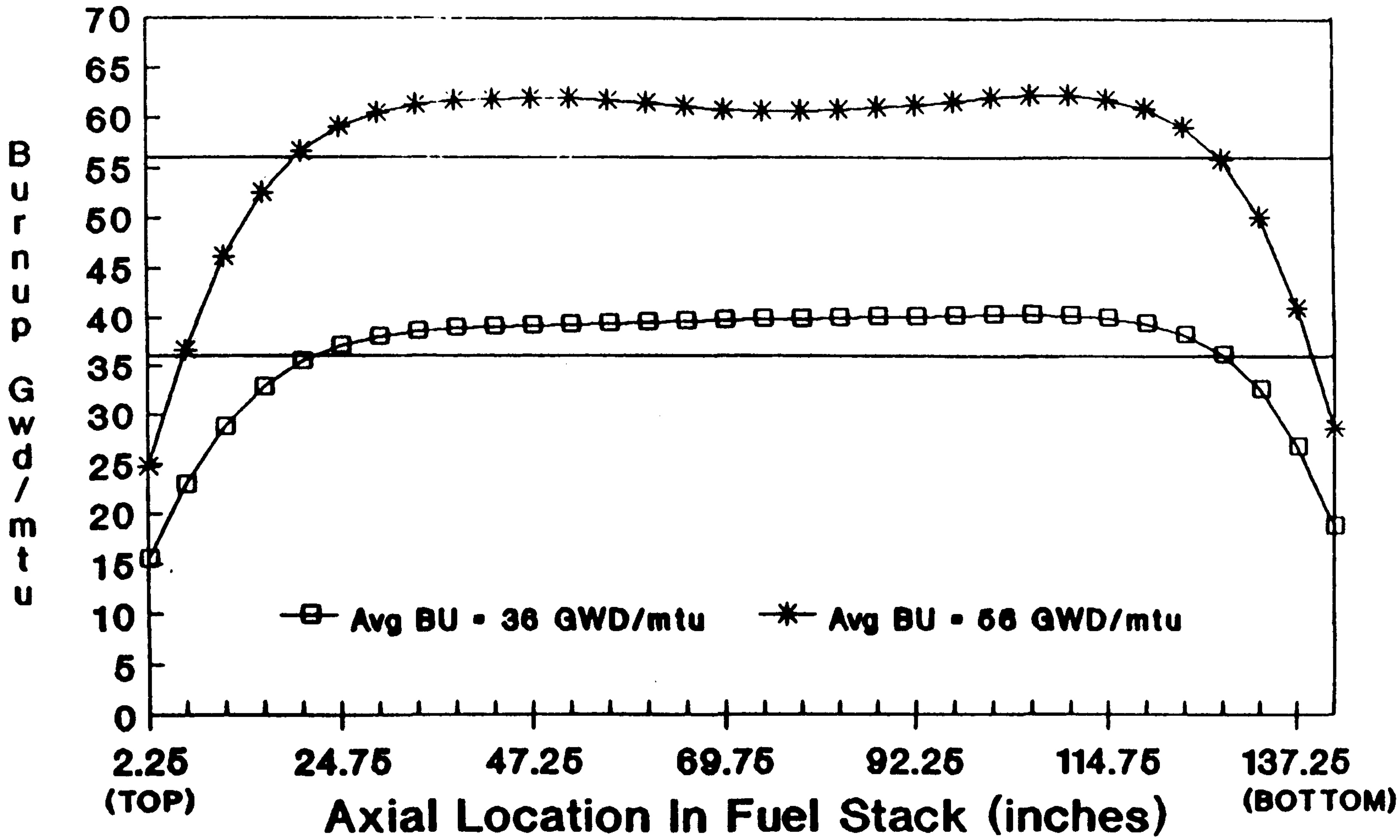
FIGURE 4

Fuel Axial Location vs Burnup



TRADE-OFF STUDY: 5 years cooled fuel

MAX. CAPACITY W/O MODIFICATION OF CASK BODY

16 PWR CONFIGURATION

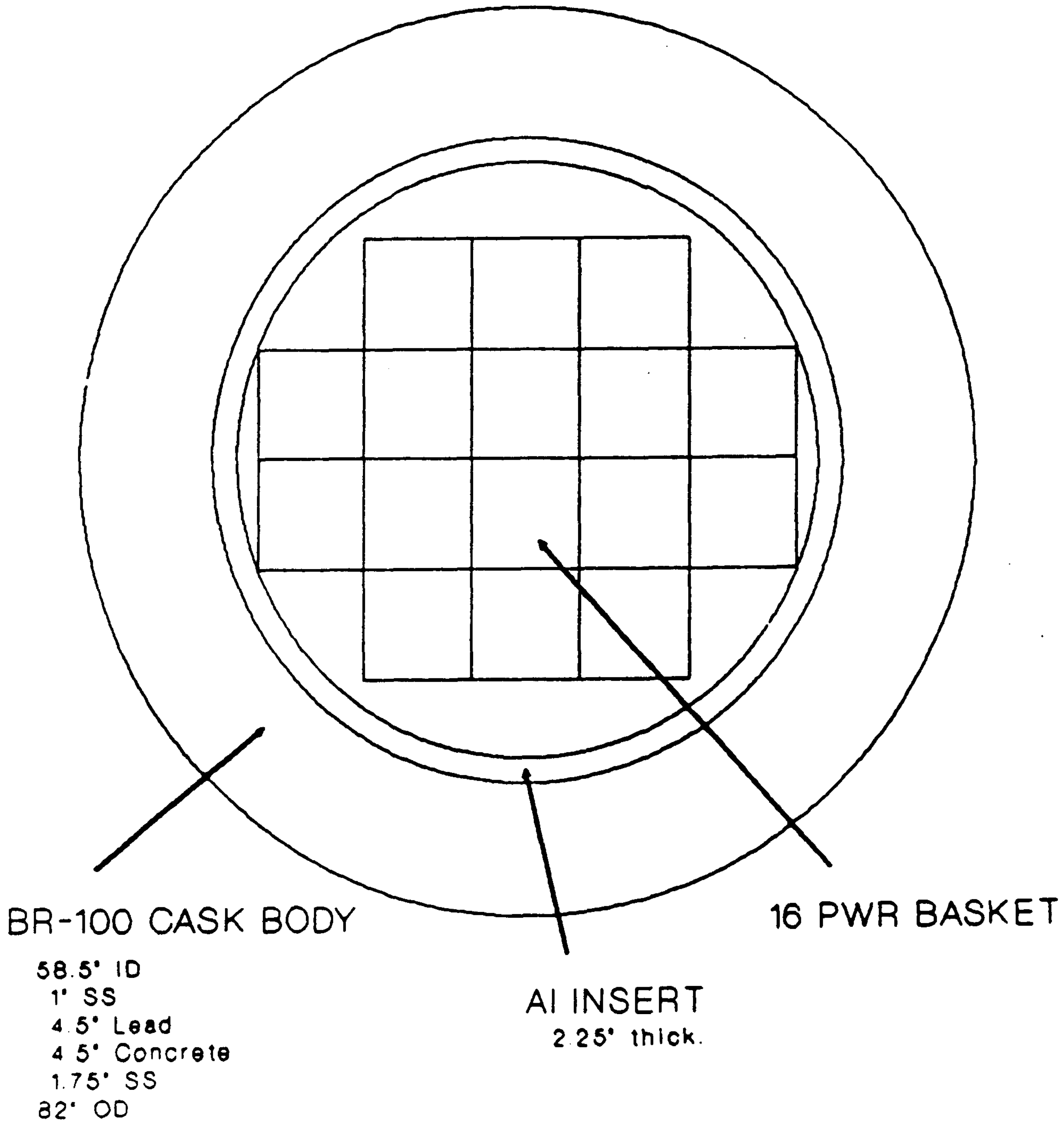
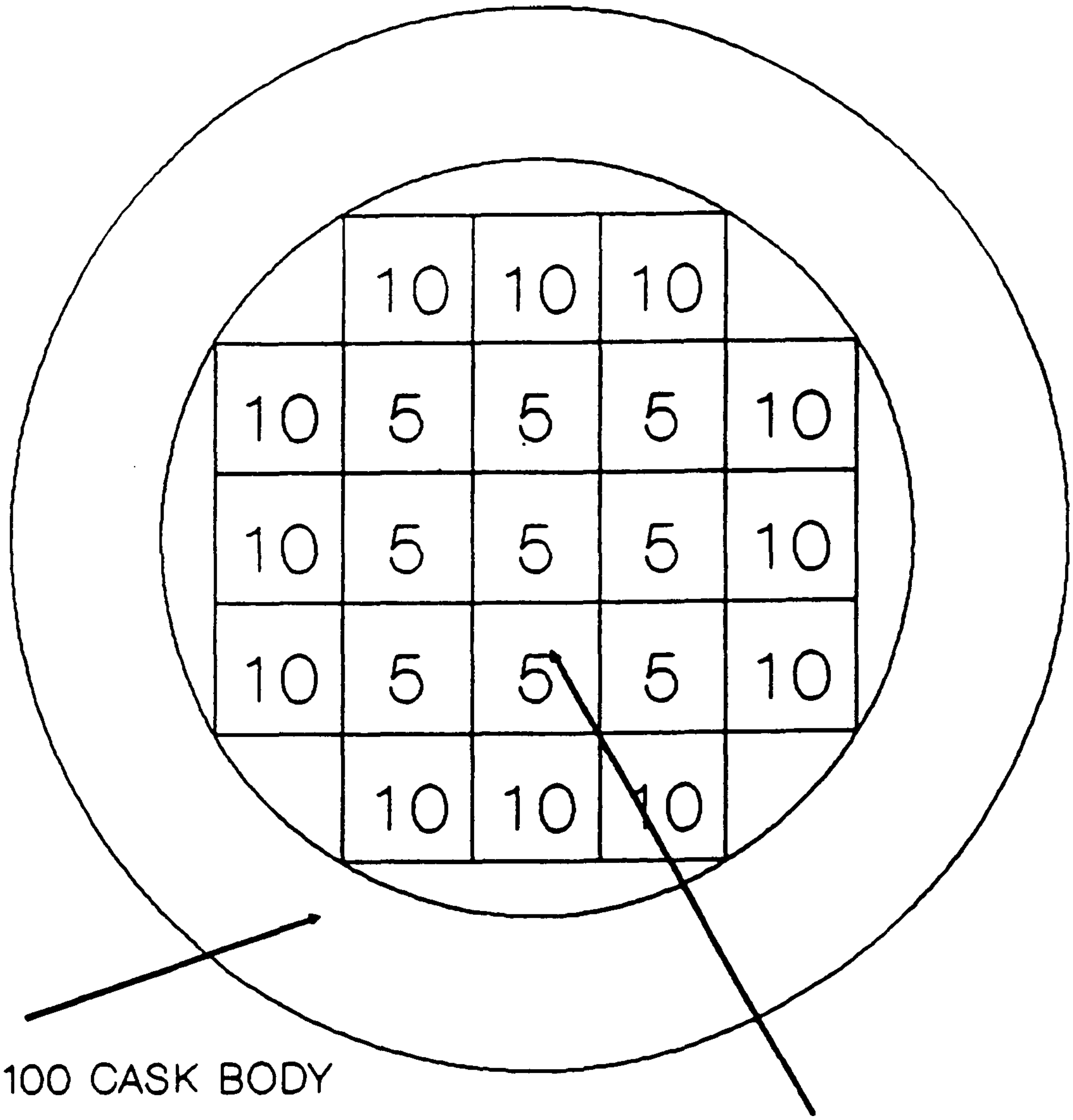


FIGURE 6

TRADE-OFF STUDY: 5 years cooled fuel

BASELINE CASK BODY

21 PWR CONFIGURATION / 12 PWR, 10 years cooled
9 PWR, 5 years cooled



BR-100 CASK BODY

58.5" ID
1" SS
4.5" Lead
4.5" Concrete
1.75" SS
82" OD

21 PWR BASKET

FROM-REACTOR CASK DEVELOPMENT PROGRAM TRADE-OFF STUDY

B&W BR-100 CASK

THE EFFECT ON CAPACITY AND COST OF TRANSPORTING CONSOLIDATED FUEL, NON-FUEL BEARING COMPONENTS, FAILED FUEL, OR NON-STANDARD FUEL

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B&W Fuel Company
An American Company With Worldwide Resources

THE EFFECT ON CAPACITY AND COST OF TRANSPORTING CONSOLIDATED FUEL, NON-FUEL BEARING COMPONENTS, FAILED FUEL, OR NON-STANDARD FUEL

1. SUMMARY

This report presents the results of a study performed by the B&W Fuel Company (BWFC) to investigate capacity and life-cycle-cost (LCC) impacts on the BR-100 100-ton spent fuel shipping cask of transporting payloads other than standard fuel. Payloads reviewed included consolidated fuel at ratios ranging up to 2:1, non-fuel bearing components (NFBC), failed fuel, and non-standard fuel as defined in the contract Statement-of-Work. Package constraints were investigated in hook weight, shielding, structural, and thermal areas. Only baseline BR-100 PWR basket configurations were examined, but the results are judged to be directly extrapolatable to BWR fuel.

The limiting parameter for BR-100 transportation of consolidated fuel was found to be hook weight. When the weight criteria was satisfied, thermal and shielding criteria were also met. If only 2:1 consolidation fuel canisters (each with two baseline fuel assemblies worth of rods at 35 Gwd/mtu burnup, 3 w/o enrichment, 10-year cooled) are shipped in the PWR basket, only 11 canisters can be accommodated due to weight limitations. If the consolidation ratio is 1.2:1, 17 canisters can be accommodated with acceptable performance. A consolidation ratio of 1.8:1 yields an acceptable payload of 12 canisters.

A full compliment of 21 NFBC canisters (each with ten assemblies worth of hardware) has no weight or thermal problems, but has a marginally unacceptable 2-meter dose rate. The weight and thermal margins would allow a gamma blanket to be attached to the outside of the cask, thereby reducing the dose rate to acceptable limits for a full load. Another solution would be the shipment of intact fuel or canisters of consolidated fuel in the basket outer row along with the shipment of NFBC canisters in the inner rows.

The shipment of failed fuel can be handled several ways. Fuel with only

minor defects should be able to be shipped normally. Routine shipments of large quantities of failed fuel may require a double-containment inner cavity which the BR-100 can easily transition to with its existing two-piece closure system. The third method is to canisterize serious "leakers" in a bolted container as Cogema does in France. With either of those options the BR-100 still carries a full payload of fuel.

All non-standard fuel defined by the contract Statement-of-Work except for the extra-long fuel used only at the South Texas Project, can be transported by the BR-100, most at full rated capacity.

2. INTRODUCTION

The purpose of this document is to satisfy the requirement in Sections 4.10.1.5 and 4.10.1.6 of the contract Statement-of-Work to "..conduct trade-off and impact evaluations of the following design considerations on cask payload capacities and costs: ...Transportation of consolidated fuel assuming consolidation ratios ranging from 1.2:1 to 2.0:1, and ...The effects of nonstandard and failed fuel and nonfuel-bearing materials..." The B&W Fuel Company (BWFC) has designed a 100-ton spent fuel shipping cask, the BR-100, which efficiently transports standard fuel in either of two basket configurations, for either PWR or BWR fuel. BWFC's goal in performing the investigations detailed in this report was to determine the capacity and effect on LCC for the BR-100 that payloads other than standard fuel would have.

Section 3 of this report describes the baseline BR-100 and documents its performance in transporting standard design basis fuel. Section 4 defines the performance parameters of consolidated fuel at ratios of 1.2:1, 1.8:1, and 2.0:1 and then looks at their effect on capacity. Section 5 details non-fuel bearing components (NFBC) and how they can be optimized with respect to capacity. Section 6 examines failed fuel and the various options available to transport it. Section 7 examines non-standard fuel parameters and their relationship to BR-100 capacity. Section 8 is the conclusion and reviews both the capacity and cost effects of the proposed alternatives.

3. BR-100 DESIGN DESCRIPTION

The baseline BR-100 cask is shown in Figures 1, 2, and 3. It has a cavity 58.5-inches in diameter and 181-inches long and has side walls make of—from inside out—1-inch of 304L stainless steel, 4.5-inches of lead, 4.5-inches of borated concrete, and 1.75-inches of 304L stainless steel. Unique baskets for either PWR and BWR fuel have a capacity of 21 or 52 assemblies, respectively. The baskets are made of aluminum cells, supports (formers), and end plates which efficiently transfer heat from the fuel to the cask inner wall. The PWR basket has 21 individual fuel cells (Figure 4) which each have a payload cavity of 8.75-inches square by 181-inches long.

The hook weight of the BR-100, including interstitial water (removed before lifting the cask out of the pool) and the handling equipment, is at or just under 100-tons in either PWR or BWR configuration, using the heaviest fuel available. The design basis payload for the BR-100 is 10-year cooled fuel that has a burnup of 35 Gwd/mtu (PWR) or 30 Gwd/mtu (BWR) and an enrichment of 3.0-4.5 w/o.

Standard fuel is defined in the contract Statement-of-Work, Appendix C "Cask Interface Guidelines," Section 1.A, and is shown in Table 1 with several key parameters. The heaviest fuel is used for maximum hook weight and structural calculations, the most reactive fuel is used for criticality calculations, and the highest source term is used for shielding analysis. The BR-100 can carry any 21 of the standard PWR assemblies or any 52 of the standard BWR assemblies.

4. CONSOLIDATED FUEL

BWFC designs and fabricates PWR fuel and has participated in several projects to design consolidation systems for PWR and BWR fuel. BWFC has recently fabricated its own fuel consolidation equipment, Fuel Master™, and is marketing consolidation services to utilities. Some of the criteria for fuel consolidation canisters include insertability back into the same storage cell the fuel came out of, compatibility with existing reactor fuel handling systems, and structural viability under

TABLE 1

STANDARD SPENT FUEL DESCRIPTION

		Transverse Dimension *** (inch)	Maximum Length ** (inch)	Maximum Weight * (pound)
PWR SPENT FUEL				
Westinghouse Electric	17 x 17	8.445	162.0	1510.
Westinghouse Electric	15 x 15	8.445	161.5	1500.
Westinghouse Electric	14 x 14	8.040	161.5	1330.
Babcock & Wilcox	17 x 17	8.546	167.5	1535.
Babcock & Wilcox	15 x 15	8.546	167.5	1540.
Combustion Engineering	16 x 16	8.240	180.0	1465.
Combustion Engineering	14 x 14	8.130	158.5	1295.
Exxon Nuclear	17 x 17	8.435	161.5	1375.
Exxon Nuclear	15 x 15	8.435	163.5	1460.
Exxon Nuclear	14 x 14	8.120	163.5	1320.
BWR SPENT FUEL				
General Electric	8 x 8	5.530	178.0	610.
General Electric	7 x 7	5.530	178.0	610.
Exxon Nuclear	8 x 8	5.260	178.0	600.
Exxon Nuclear	7 x 7	5.260	173.0	630.

* The maximum weight considers a maximum variation of 2% above the nominal value.

** The maximum length considers a maximum 40,000 MWD/MTU burnup and a 400 F temperature of the guide tubes.

*** The transverse dimension includes the fabrication tolerances.

postulated accident conditions. These canisters do <u>not</u> have to provide any containment, and take advantage of that by having openings in their walls to reduce weight and allow cooling water cross-flows. The standard material of construction for canisters is expected to be stainless steel. A typical canister is shown in Figure 5.

BWFC's experience and that of others, including DOE, is that a 2.0:1 fuel consolidation ratio is achievable. This means that all the fuel rods from two fuel assemblies are removed and placed in a canister meeting the criteria described above. Indeed, to be marketable as an efficient fuel storage option, a 2.0:1 consolidation ratio is required. BWFC did look at other ratios for this report—1.2:1 and 1.8:1—but they are not considered realistic scenarios for further investigation. Table 2 gives pertinent data for canisters corresponding to the three ratios of interest. Burnup was assumed to be 35 Gwd/mtu and enrichment 3.0 w/o to obtain conservative decay heat and source term values.

TABLE 2

CONSOLIDATED FUEL CANISTER DATA-PWR

Ratio	1.2:1	1.8:1	2.0:1
Size (inches)	8.5 square	8.5 square	8.5 square
No. of fuel rods	250	370	416
Weight (pounds)	1900	2700	3050
Decay Heat (Watts)	689	1033	1148

When canisters of the three types shown in Table 2 are loaded into the baseline BR-100, the resulting hook weights are such that no more than 11 of the 2.0:1 canisters, 12 of the 1.8:1 canisters, or 17 of the 1.2:1 canisters can be loaded without exceeding the 100-ton hook limit. Lighter fuel than used in the analysis is probable and would result in lighter canisters and a larger carrying capacity, but would vary from reactor to reactor. A discrete ANISN run proved that the maximum 2-meter dose rate of having a full load of 21 canisters—42 fuel assembly's worth of rods—was an acceptable 9.7 mr/hr (3.9 neutron, 5.8 gamma). The thermal performance of placing up to twelve 2.0:1 canisters in the basket outer row or the specified number of 1.2:1 or 1.8:1 canisters in any location was evaluated against other P/THERMAL calculations and judged to be quite acceptable.

Redesign of the BR-100 baskets to optimize capacity of consolidated fuel canisters would increase capacity by up to three of the 2.0:1 canisters, but may not be cost effective if only a few utilities choose consolidation as a storage option.

5. NON-FUEL BEARING COMPONENT CANISTERS

BWFC has, in conjunction with its work in fuel consolidation techniques, investigated methods of compacting the non-fuel bearing components (NFBC) of fuel assemblies into reasonable storage geometries. NFBC, typically stainless steel end fittings, Inconel spacer grids and holddown springs, and zircaloy guide tubes and instrument tubes (some spacer grids also use zircaloy), are left after fuel rods are removed in a consolidation campaign, or, in the case of reactor control devices, after they are discharged from use. These components can have a significant gamma source term associated with them, primarily because of activation of cobalt within the steel and Inconel items. BWFC has found that a 10:1 ratio for fuel assembly NFBC can be obtained using their Fuel Master system, which segregates and stacks the end fittings (actually using one set for the canister itself) and then chops and crushes the cages. A 10:1 ratio means that all non-fuel rod hardware from 10 assemblies can be placed into one canister. Reactor control devices such as control rods or burnable poison rods contain no fissionable materials and can be similarly consolidated into a canister with a much larger ratio.

The resultant canister would weigh about 1000-1200 pounds and would have a negligible decay heat generation. An ANISN run was made to look at the dose rate for a full 21-canister load. The hardware was assumed to have been irradiated for 5 cycles (equivalent to about 60 Gwd/mtu), to have its maximum allowable cobalt content, and to have been cooled for ten years. The canister was assumed to have a stainless steel wall

.125-inches thick. The side dose rate at 2-meters from the personnel barrier was calculated to be 13.5 mr/hr, slightly over the allowable 10 mr/hr. More realistic assumptions could reduce the predicted dose rate to acceptable levels, but a simple flexible gamma shield attached to the outside wall of the cask would also provide ample shielding to achieve acceptable values (only about .25-inches of steel, or equivalent, would be needed). The removable gamma shield is feasible because of the lower payload weight with NFBC as compared to intact fuel (about 10,000 pounds less) and the lack of heat generation with the NFBC.

Another method to ship the NFBC canisters would be to mix them with either intact fuel assemblies or consolidated fuel canisters, using the fuel to provide shadow shielding. To investigate their use with intact fuel, a scenario was devised whereby the inner nine basket locations were loaded with NFBC canisters and the outer twelve were loaded with intact design basis fuel (Figure 6). The predicted 2-meter dose rate from that ANISN case was 8.9 mr/hr (5.8 gamma, 3.1 neutron), quite acceptable and well within the hook weight criteria. A similar scenario was investigated for consolidated fuel (2.0:1) canisters with a resultant 9.5 mr/hr dose rate (5.8 gamma, 3.7 neutron). Weight constraints prevent the 12-fuel canister (2.0:1)/9-NFBC canister from being feasible (overweight by about 12,000 pounds), but a combination of 9 NFBC canisters in the inner cells and 8 fuel canisters (Figure 7) would work from all aspects.

Most reactor control devices that have been discharged (typically control rod assemblies or burnable poison rod assemblies for PWRs or channels for BWRs) are currently being stored within discharged fuel assemblies in a spent fuel pool. The BR-100 offers a unique way of shipping these components, which are normally HLW, to the repository or Federal disposal facility. The fuel cells have been sized in length, width, and weight-carrying capability to carry the fuel and the control devices. Because of hook weight limitations, in rare cases this may require a reduction in total assemblies transported (no more than one less for PWR or seven less for BWR), but usually will be compatible with no reduction in capacity.

Large NFBC such as reactor components, Defense High Level Waste Canisters, or other forms of high level waste can be easily accommodated by the BR-100 with a different basket design. The BR-100 has a stainless steel cavity surface and a cavity 58.5-inches in diameter by 181 inches long (without basket). If dry-loaded, it can carry over 50,000 pounds and still be within the 100-ton hook limit. It also can dissipate up to 18 kW in heat and meet all design temperature criteria. By removing the shield plug, which may not be necessary for shielding 20-year cooled fuel, even the fuel containers being conceptualized for disposal at Yucca Mountain, 26-inches in diameter by 186.5-inches long, can be accommodated three at a time for possible transport from a Monitored Retrievable Storage Facility to the repository.

6. FAILED FUEL

The term "failed fuel" can be used to cover a wide spectrum of fuel conditions. Over the recent past, commercial nuclear fuel in the U.S. has had an average integrity rate of about 99.95%—that is, less than five fuel rods in 10,000 used will develop a cladding defect that will allow a small amount of its fission products to migrate to the environment. These defects are usually either pinhole leaks that sometimes "heal" themselves with corrosion products or slightly larger defects that can be visually spotted with inspection. For the approximately one assembly in 10-25 that may contain one or two rods with such a defect, separate processing is not necessary. Those assemblies have sat in spent fuel pools long enough for any uncombined fission products to have been leached out by the action of pool coolant water within the failed rods. The BR-100 cask, as do all casks in the From-Reactor Cask Program, ships its fuel dry, which avoids the possibility of water leaching out further fission products. Operating procedures require that, before the lid is opened, the cask cavity atmosphere be sampled to check for the presence of fission gas products. Should such products be present, the BR-100 has a vent/purge system which allows the cavity atmosphere to be routed into a filter bank that would remove the radioactive constituents for safe disposal.

There have been rare instances where fuel failures may be severe enough to require separate canisterization (baffle jetting, for instance). The use of canisters to isolate such fuel is common in France, where Cogema uses a bolted canister with a valve at each end to provide containment. The canister is shown in Figure 8 and, because of its tight clearance to the sides of the fuel, fits into a standard storage cell. A similar adaptation could be used for the BR-100, with no reduction in capacity except for fuel over 177-inches in length.

Shipment of large quantities of fuel assemblies that have multiple rod failures could result in a need for a double-containment inner cavity. The BR-100 uses a double-lid closure system that routinely only depends on the bolted top closure for containment. The inner shield plug does not currently provide a containment seal, but could be easily adapted to provide one with no reduction in capacity.

7. NON-STANDARD FUEL

Non-standard fuel is defined in the contract Statement-of-Work and is shown in Table 3 along with some key parameters. The BR-100 has been designed such that it can accommodate every fuel listed, with the single exception of the Extra-Long Westinghouse 17X17 assemblies used only at the South Texas Project. Some BWR fuels with large transverse dimensions may require transport within the PWR basket (double stacked, or 42 per shipment), but the large majority can be shipped at full payload in their appropriately titled basket configuration.

TABLE 3

NON-STANDARD SPENT FUEL DESCRIPTION

		Transverse Dimension * (inch)	Maximum Length * (inch)	Maximum Weight * (pound)
PWR SPENT FUEL				
Westinghouse Electric	16 x 16	7.775	161.5	1335.
Westinghouse Electric	13 x 13	no data in literature		
Babcock & Wilcox	14 x 14	8.435	139.0	1280.
Combustion Engineering	14 x 14 XL	no data in literature		
Combustion Engineering	15 x 15	8.25	149.0	1385.
Gulf United Nuclear	17 x 17	no data in literature		
Westinghouse Electric	17 x 17 XL	8.55	201.0	NA
BWR SPENT FUEL				
General Electric	11 x 11	6.55	83.	NA
General Electric	9 x 9	6.55	83.	NA
General Electric	6 x 6	4.05	136.5	NA
Exxon Nuclear	11 x 11	6.525	85.	465.
Exxon Nuclear	10 x 10	5.625	103.5	385.
Exxon Nuclear	9 x 9	6.55	178.0	585.
Exxon Nuclear	6 x 6	4.285	135.5	335.
Allis Chalmer	10 x 10	no data in literature		
Nuclear Fuel Services	9 x 9	no data in literature		
United Nuclear	6 x 6	no data in literature		
Westinghouse Electric	8 x 8	5.55	177.5	615.

* Determined as for Standard Spent Fuel

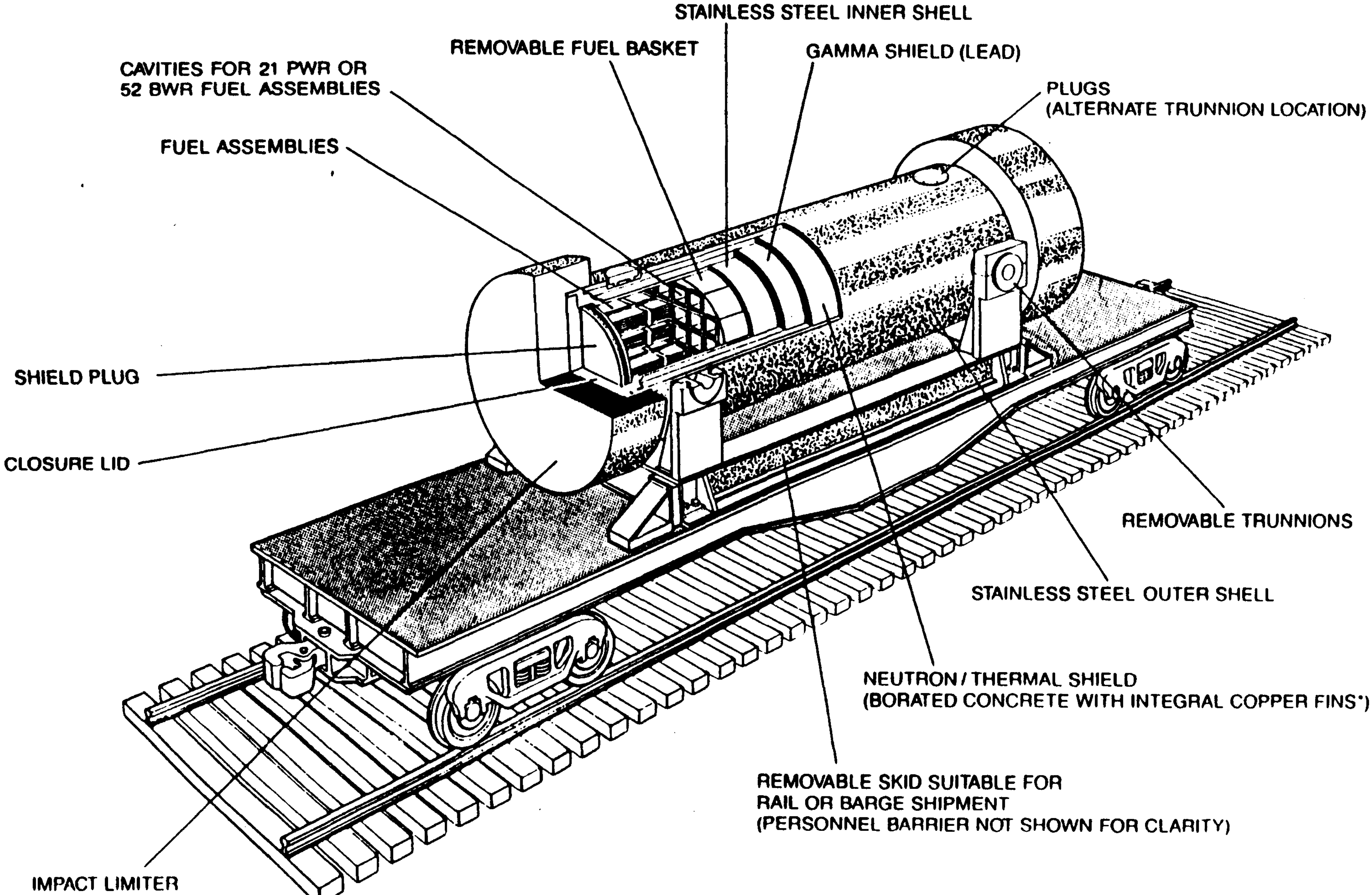
8. CONCLUSION

The BR-100 is well adapted to transport any of the payloads investigated and presented in this report. Although optimized to carry standard intact fuel, the BR-100 has the flexibility to carry attractive quantities of consolidated fuel, failed fuel, or NFBC canisters. Outstanding BR-100 attributes include the ability to provide double containment with its closure system, the ability to transport reactor control devices within intact fuel, and the ability to transport a full payload of practically every type of fuel—standard or non-standard.

The life-cycle-cost effects of this flexibility are difficult to accurately quantify because of the uncertainty of the fundamental assumptions. The BR-100 provides an efficient method to transport any payload investigated in this study. The use of the BR-100 in the applications reviewed above will eliminate the necessity of developing other cask designs. The production of more casks of a single design will also reduce the unit cost, further reducing LCC.

The BR-100 is a viable option to providing optimized LCC with a cask that provides a wide range of payload capabilities.

BABCOCK & WILCOX BR-100
100 TON RAIL / BARGE CASK



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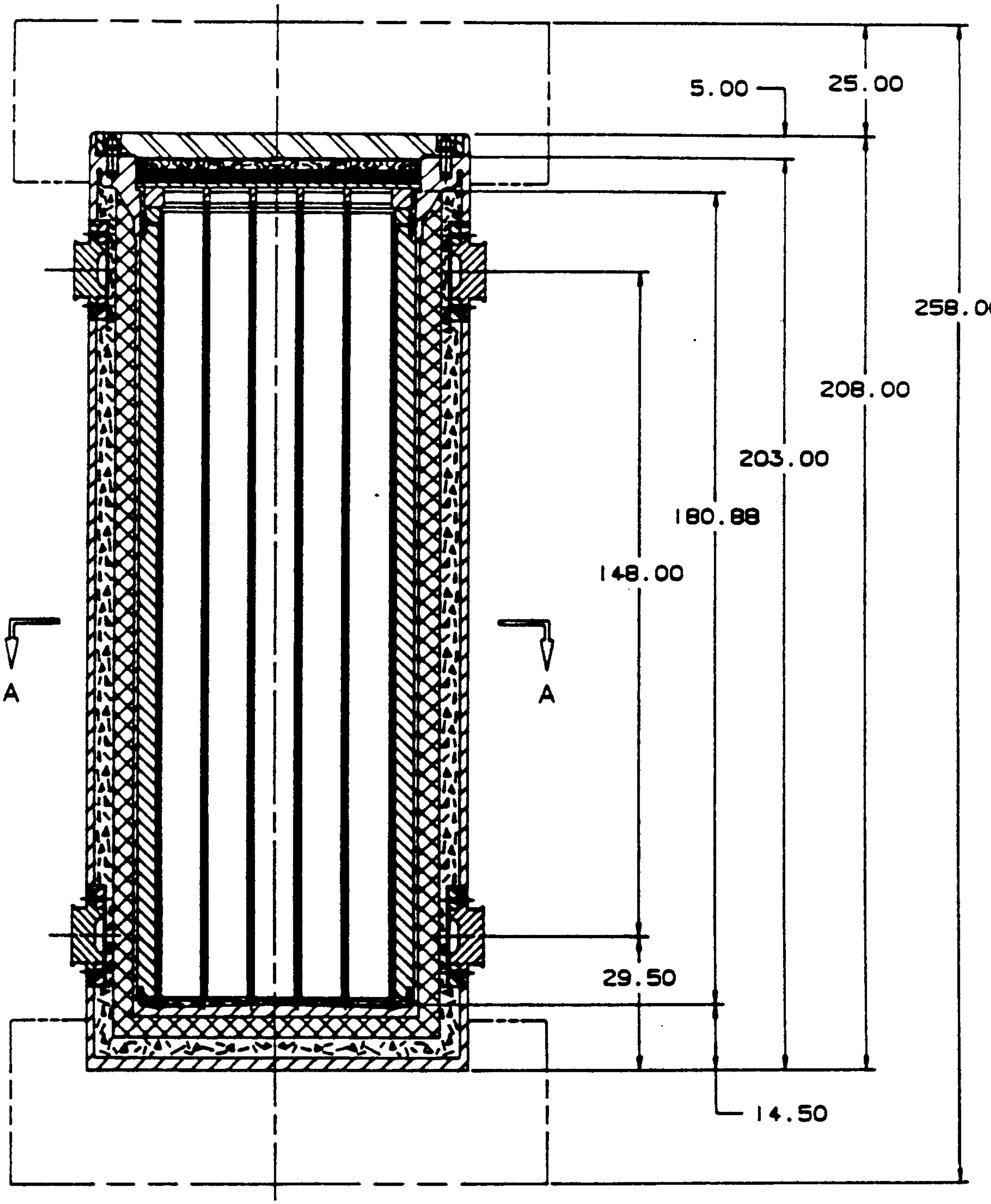
APRIL, 19

FIGURE 2

BR-100 CASK
LONGITUDINAL SECTION

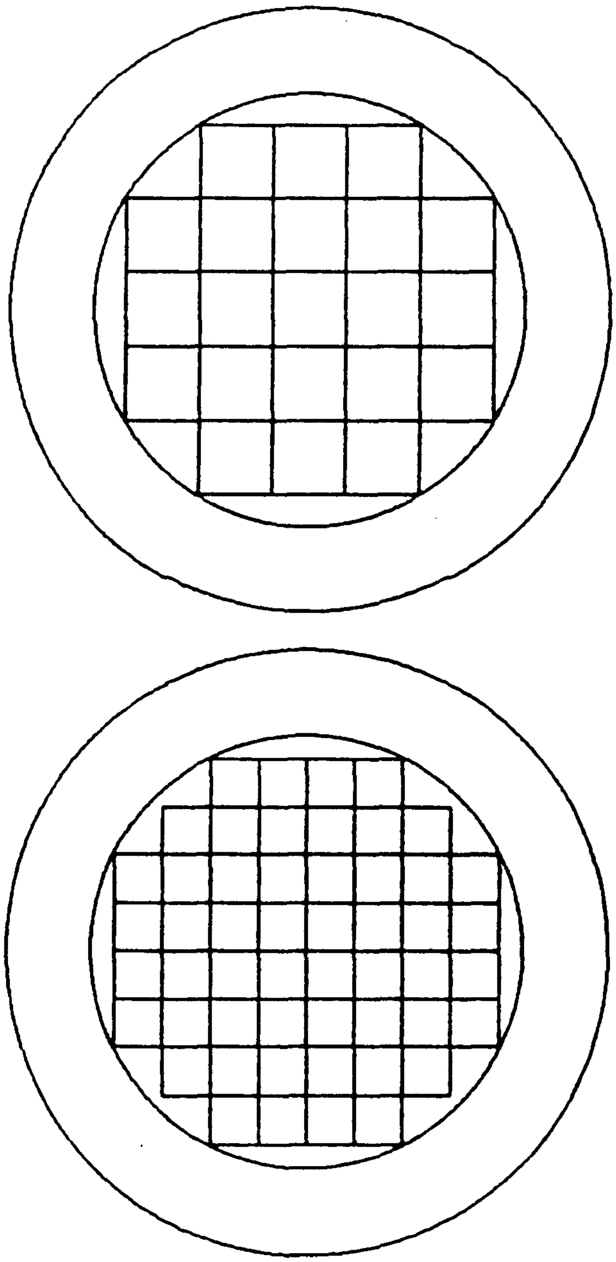


All Dimensions in Inches

FIGURE 3

TRADE-OFF STUDY: effect of non-standard payload on BR-100 capacity

BASELINE CASK BODY



PWR CONFIGURATION
21 FUEL ASSEMBLIES

BWR CONFIGURATION
52 FUEL ASSEMBLIES

FIGURE 4

TRADE-OFF STUDY: effect of non-standard payload on BR-100 capacity

PWR CELL CROSS-SECTION

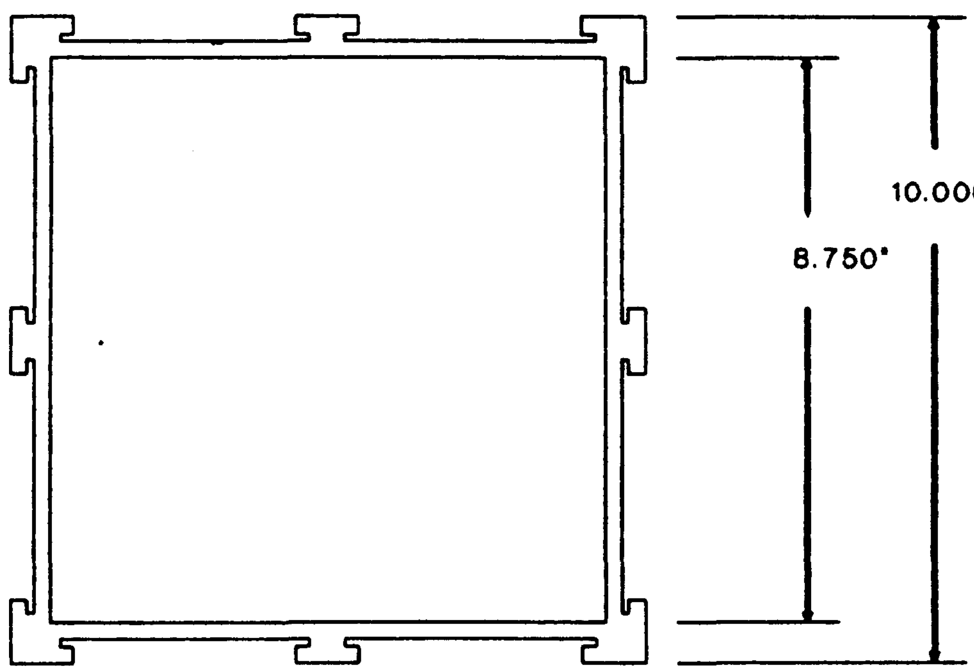


FIGURE 5

B&W CONSOLIDATED FUEL CANISTER
(Westinghouse Fuel)

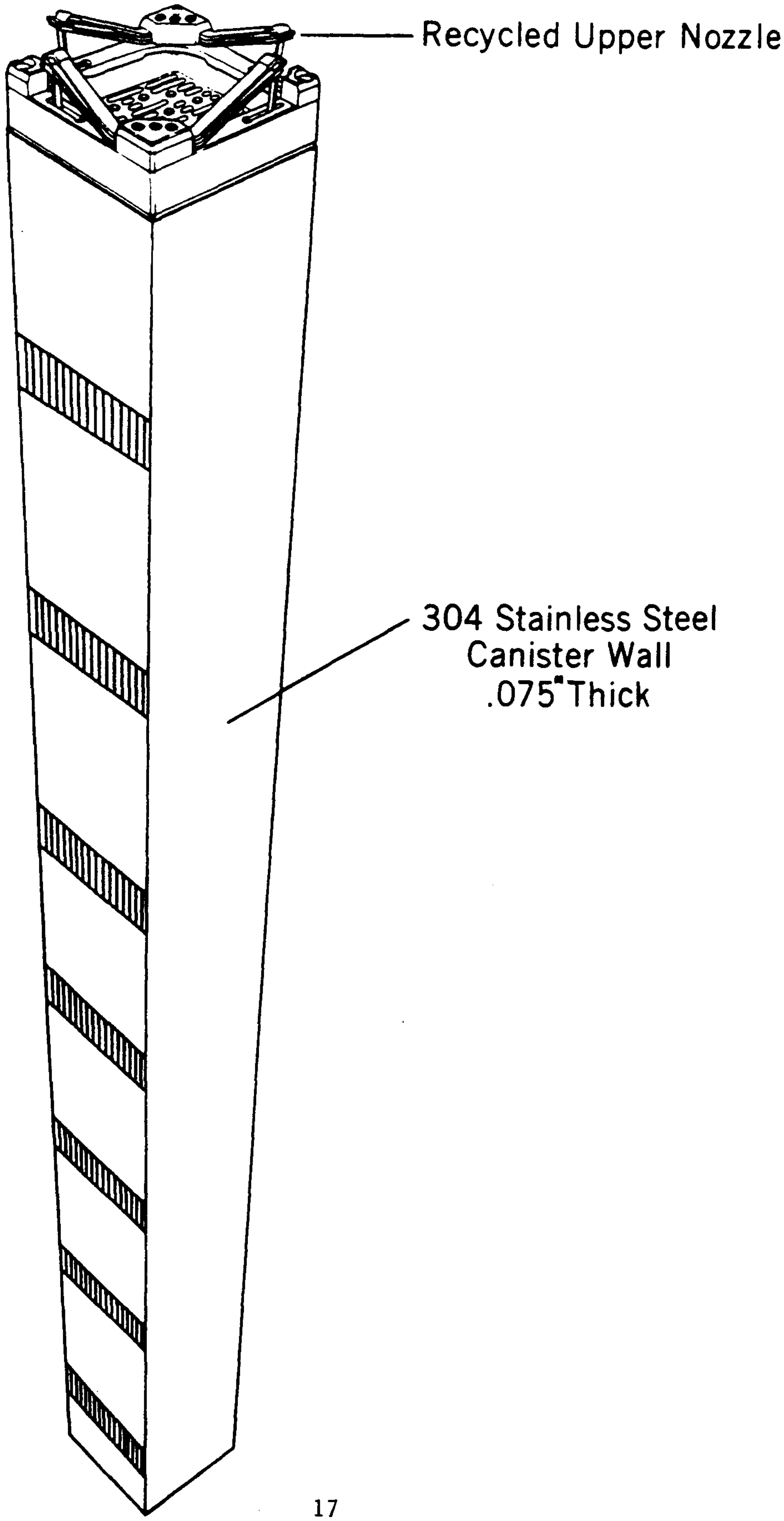


FIGURE 6

TRADE-OFF STUDY: effect of non-standard payload on BR-100 capacity

TRANSPORTATION OF 9 NFBC CANISTERS (NFBC) WITH 12 INTACT PWR FUEL ASSEMBLIES (IFA)

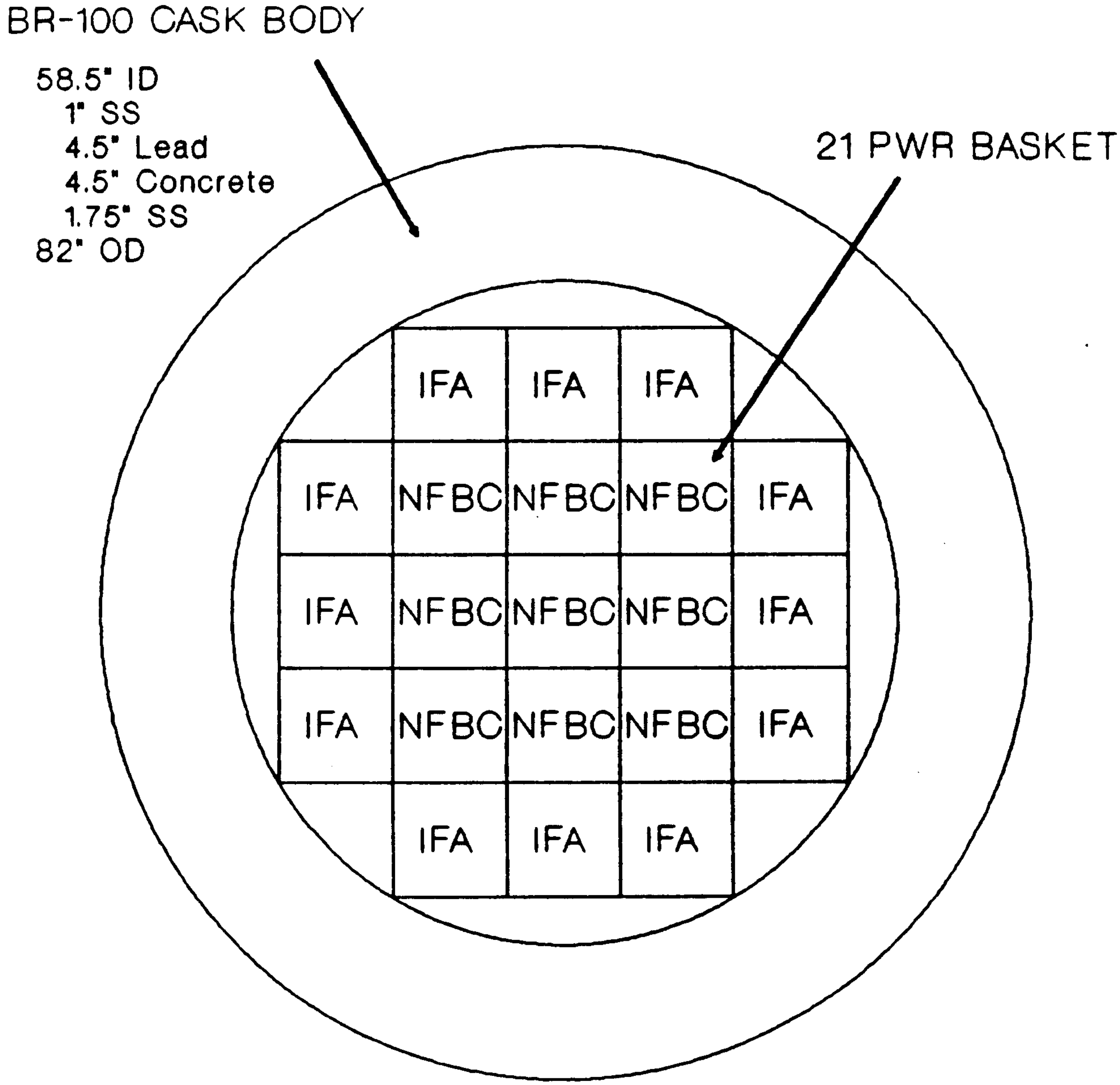


FIGURE 7

TRADE-OFF STUDY: effect of non-standard payload on BR-100 capacity

TRANSPORTATION OF 9 NFBC CANISTERS (NFBC)
WITH 8 CONSOLIDATED FUEL CANISTERS (CF)

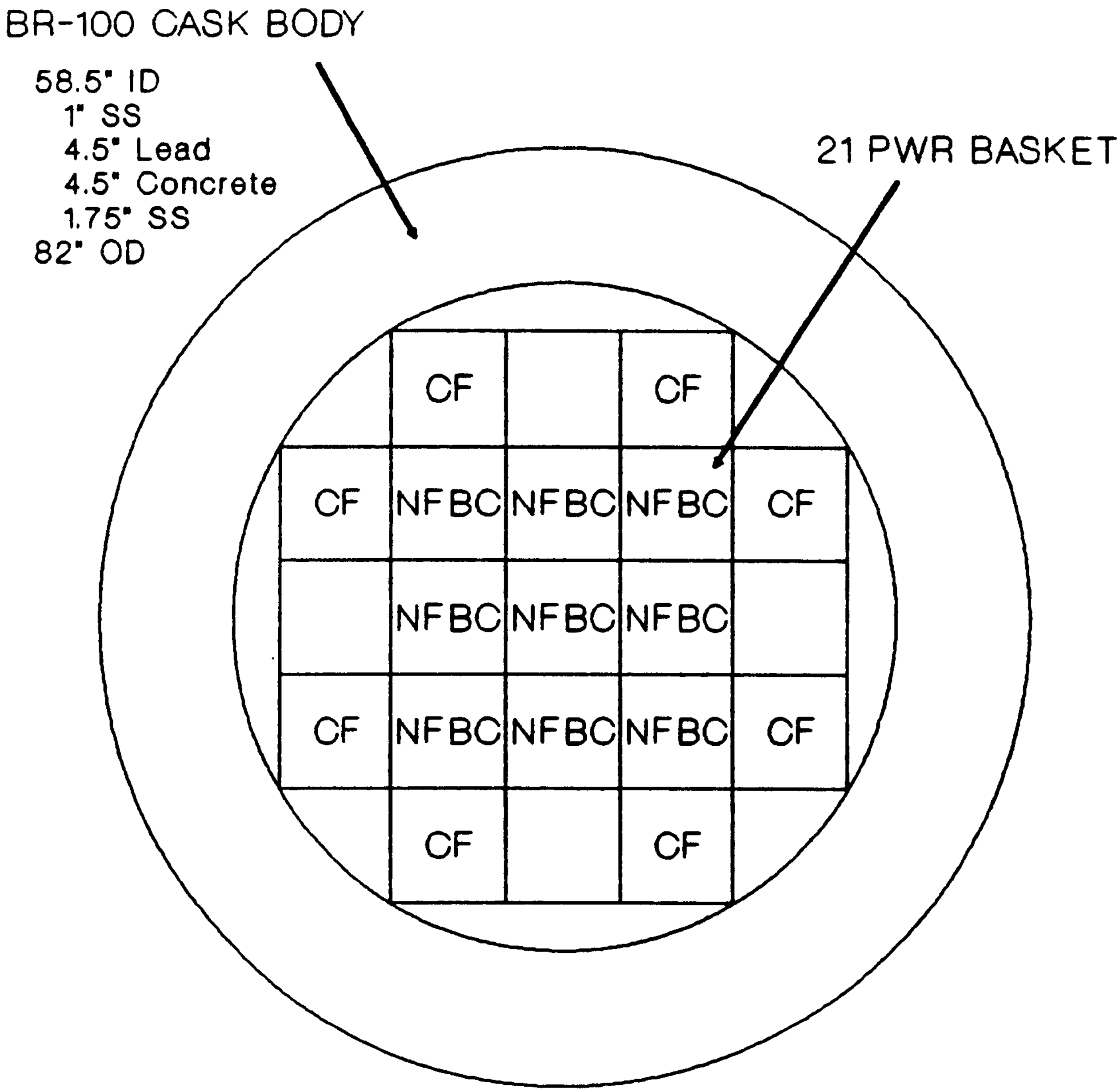
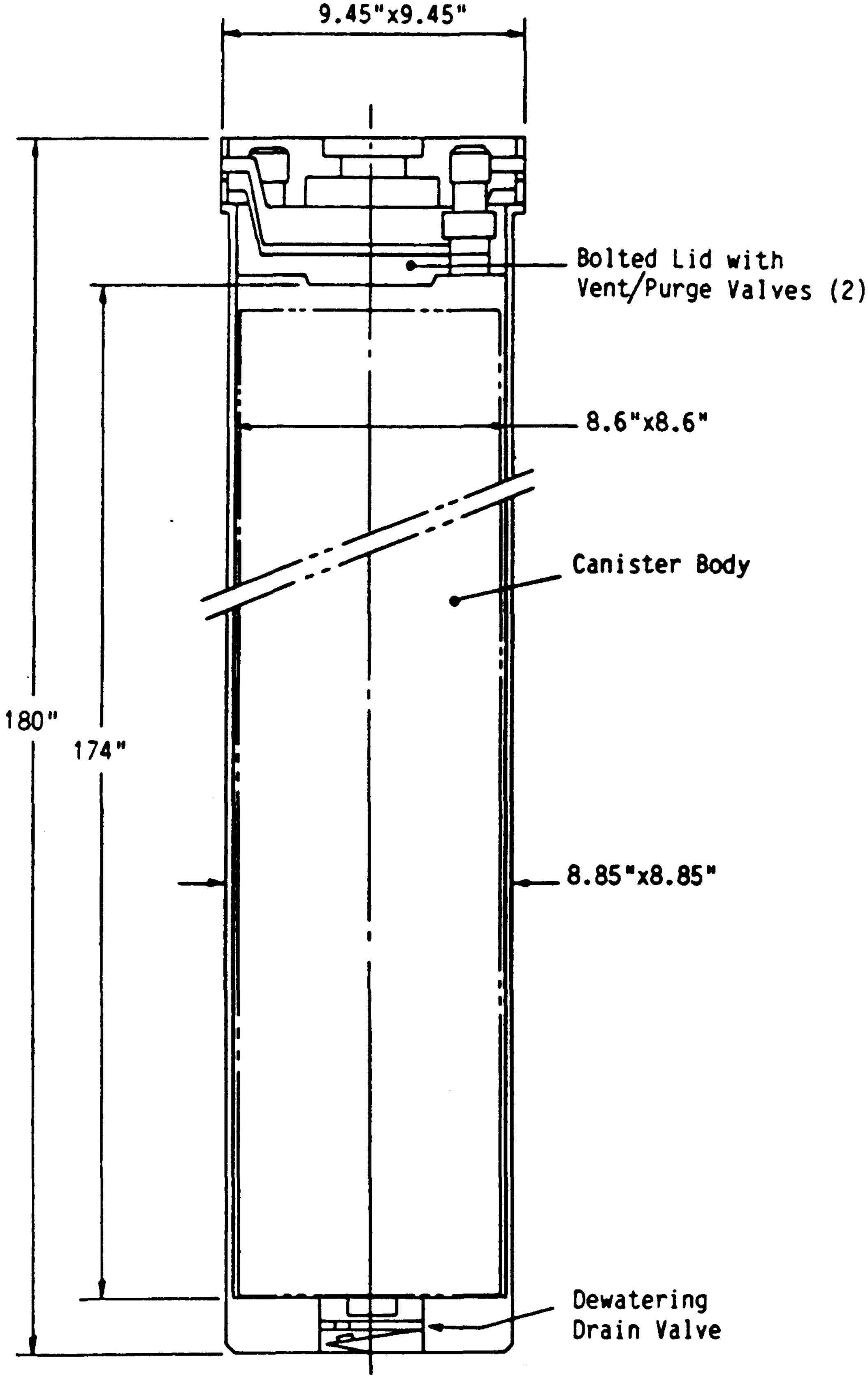


Figure 8
Typical Cogema Failed-Fuel Canister for PWR Fuel



FROM REACTOR CASK SYSTEM DEVELOPMENT PROGRAM

TRADE-OFF STUDY

B&W BR-100 CASK

COMMON-USE vs DEDICATED-USE CASKS

Document No. 51-1175905-00

July 1989

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ADVANTAGES OF COMMON-USE CASKS VS DEDICATED-USE CASKS

1. SUMMARY

The B&W Fuel Company (BWFC) investigated the life-cycle-cost differences between a cask designed to serve both Pressurized Water Reactor (PWR) and Boiling Water Reactor (BWR) fuel and two casks, each designed to serve primarily either PWR or BWR fuel. The base design was the BR-100 cask already designed to carry both types of fuel (21 PWR or 52 BWR assemblies in separate baskets). Design criteria included standard 10CFR71 requirements, a 100-ton hook load, and fuel properties as given in the contract Statement-of-Work (18-35 Gwd/mtu, 3-4.5 w/0 enriched, 10-year cooled PWR; 15-30 Gwd/mtu, 3-4.5 w/0 enriched, 10-year cooled BWR).

The shorter length of a cask designed for PWR fuel was not able to compensate in weight for the increase in diameter required to increase payload capacity. The longer BWR cask could have less shielding and carry 56 BWR assemblies, an increase of four assemblies. The relatively small life-cycle-cost (LCC) benefit from the increased BWR capacity was more than offset for dedicated-use BR-100s by additional development and fabrication costs. The common-use BR-100 cask was found to be the most LCC-effective option.

2. INTRODUCTION

This Trade-Off Study was performed to quantify the LCC of developing two dedicated-use BR-100-type spent fuel shipping casks, one for PWR fuel (with the exception of Combustion Engineering (CE) 16 X 16 and Westinghouse South Texas Project (STP)) and the other for BWR and CE 16 X 16 fuel. Those dedicated-use LCC were then compared to the baseline common-use BR-100 LCC and the differences discussed. A conclusion was reached as to the efficiency of using either dedicated-use or common-use rail casks to minimize LCC.

3. CASK DESCRIPTIONS

3.1 BR-100 COMMON-USE CASK

The baseline common-use BR-100 cask is shown in Figures 1, 2, 3,and 4. It has been sized to accept a payload of either 21 PWR fuel assemblies or 52 BWR fuel assemblies, each in unique basket structures that have about the same diameter, 58.3 inches, and length. The dose rate at midplane using the most conservative fuel has been preliminarily calculated to be 9.6 or 5.0 mr/hr (for PWR or BWR, respectively) at 2 meters from the Personnel Barrier (Federal limit is 10 mr/hr). The baseline BR-100 has a PWR crane hook weight of 199,500 pounds as shown in Table 1 and a BWR crane hook weight of 200,000 pounds as shown in Table 2.

Table 1
PWR Crane Hook Weight (Pounds)

Cask Body	139,000
Shield Plug	5,200
Basket Assembly	9,000
Fuel (without Control Components)	32,300
Stand-offs	500
Water (before dewatering)	10,000
Dewatering Tool	1,000
Handling Equipment	2,500
TOTAL	199,500

Table 2
BWR Crane Hook Weight (Pounds)

Cask Body	139,000
Shield Plug	5,200
Basket Assembly	10,000
Fuel (without Channels)	33,500
Stand-offs	300
Water (before dewatering)	8,500
Dewatering Tool	1,000
Handling Equipment	2,500
TOTAL	200,000

3.2 BR-100P DEDICATED-USE CASK

A cask was then investigated that could be designed similar to the BR-100, but dedicated to serving only PWR fuel with irradiated lengths up to 167 inches. It was designated the BR-100P. The next largest basket diameter that could accommodate an increased number of PWR fuel cell locations (beyond the 21 in the BR-100) is 63.25 inches for a 24-cell array. Using the same cask body thicknesses as the BR-100 for shielding calculations yielded a dose rate for the 24-cell cask of 9.9 mr/hr at 2 meters from the Personnel Barrier. The BR-100P was given a length 13 inches shorter than the BR-100, with the shortening coming out of the middle of the cask. The hook weight of the BR-100P with a 24-PWR capacity was calculated to be 210,000 pounds. That weight is significantly over the 200,000 pound limit and no shielding reduction for that design could be expected based on the small dose rate margin predicted. The conclusion was that the BR-100P would have to stay at the 21-PWR capacity, with the resulting shorter length and smaller weight allowing flexibility for consolidated fuel or control components.

3.3 BR-100B DEDICATED-USE CASK

The BWR version of the BR-100 was then developed. Although the length was kept the same as the BR-100 (to accommodate PWR CE 16 X 16 fuel in a special basket), the shielding thickness was reduced by 0.6 inches (all from the gamma shield) and the basket diameter increased to 61.5 inches (a 56-cell array). The dose rate for this configuration, shown in Figure 5, is 9.8 mr/hr. The hook weight is calculated to be slightly over 200,000 pounds, but is close enough such that the limit can be reached by minor design tweaking.

4. LIFE-CYCLE-COSTS

BWFC has an internal code, generated several years ago, to determine the life-cycle-costs (LCC) of various casks and cask features. This code was used to determine the sensitivity of LCC to ranges of input parameters that could be associated with changes in design or operational philosophy. Factors that would vary in the BR-100 common-use/dedicated-use comparison are shown in Table 3.

Table 3

LCC FACTORS FOR CASK COMPARISON

Factor	BR-100	BR-100P	BR-100B	LCC Difference
PWR Capacity	21	21	0	0% **
BWR Capacity	52	0	56	-5% **
PWR Hook Wt	199.5K #	192K #	N/A	0%
BWR Hook Wt	200K #	N/A	202K #	0%
Fab Cost	$1500K	$1800K	$1800K	+3%
Dev/Cert Cost	$8000K	$6000K*	$6000K*	+4%

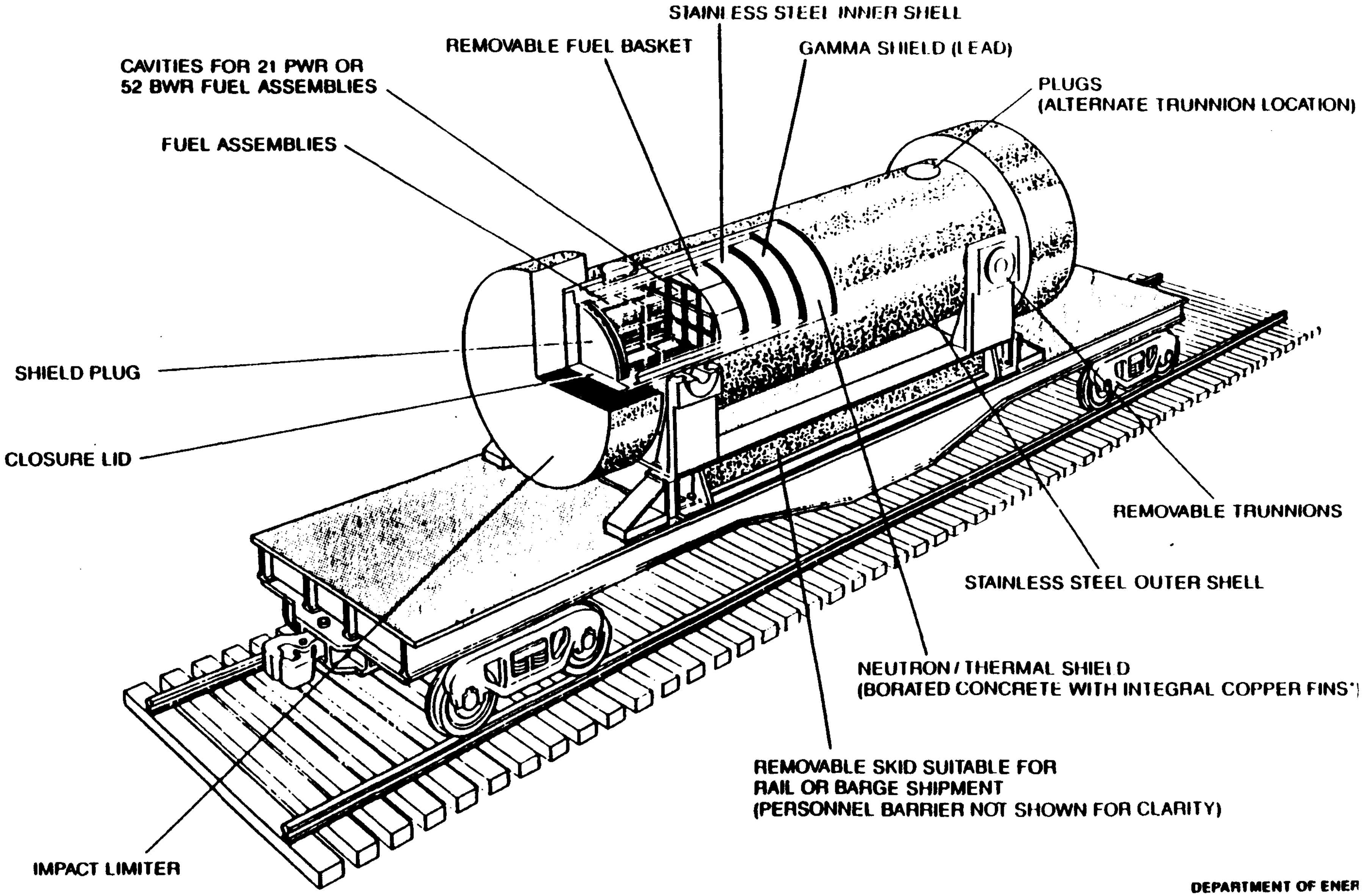
*These costs are valid only if taken together, i.e., $12M total Dev/Cert cost.

**The LCC percentages for PWR and BWR must be averaged together(60% PWR, 40% BWR) to get the correct LCC difference for capacity.
The correct overall value is -2%.

5. CONCLUSIONS

The factors in Section 4 have a cumulative effect of increasing LCC by 5% with a change to dedicated-use casks. This indicates that benefits are not present for BR-100 type casks to go to a dedicated-use system. No benefit has been assigned to the dedicated-use casks, however, for the extra flexibility provided by the lower weight and capability to carry additional payload such as control components. Such programmatic issues should be considered by DOE in its final evaluation.

BABCOCK & WILCOX BR-100
100 TON RAIL / BARGE CASK



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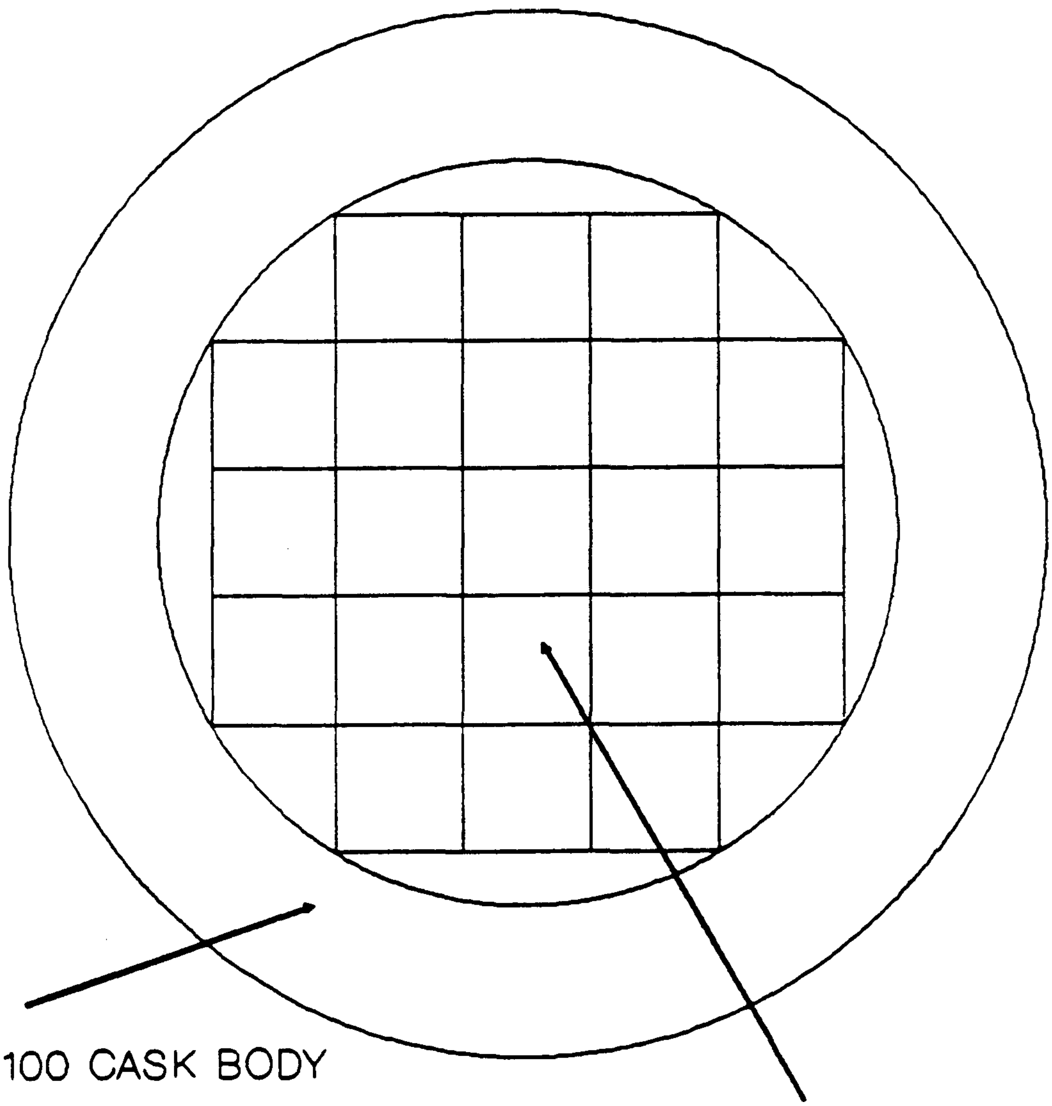
TRADE-OFF STUDY: common use vs dedicated cask

BASELINE CASK BODY

21 PWR CONFIGURATION



BR-100 CASK BODY

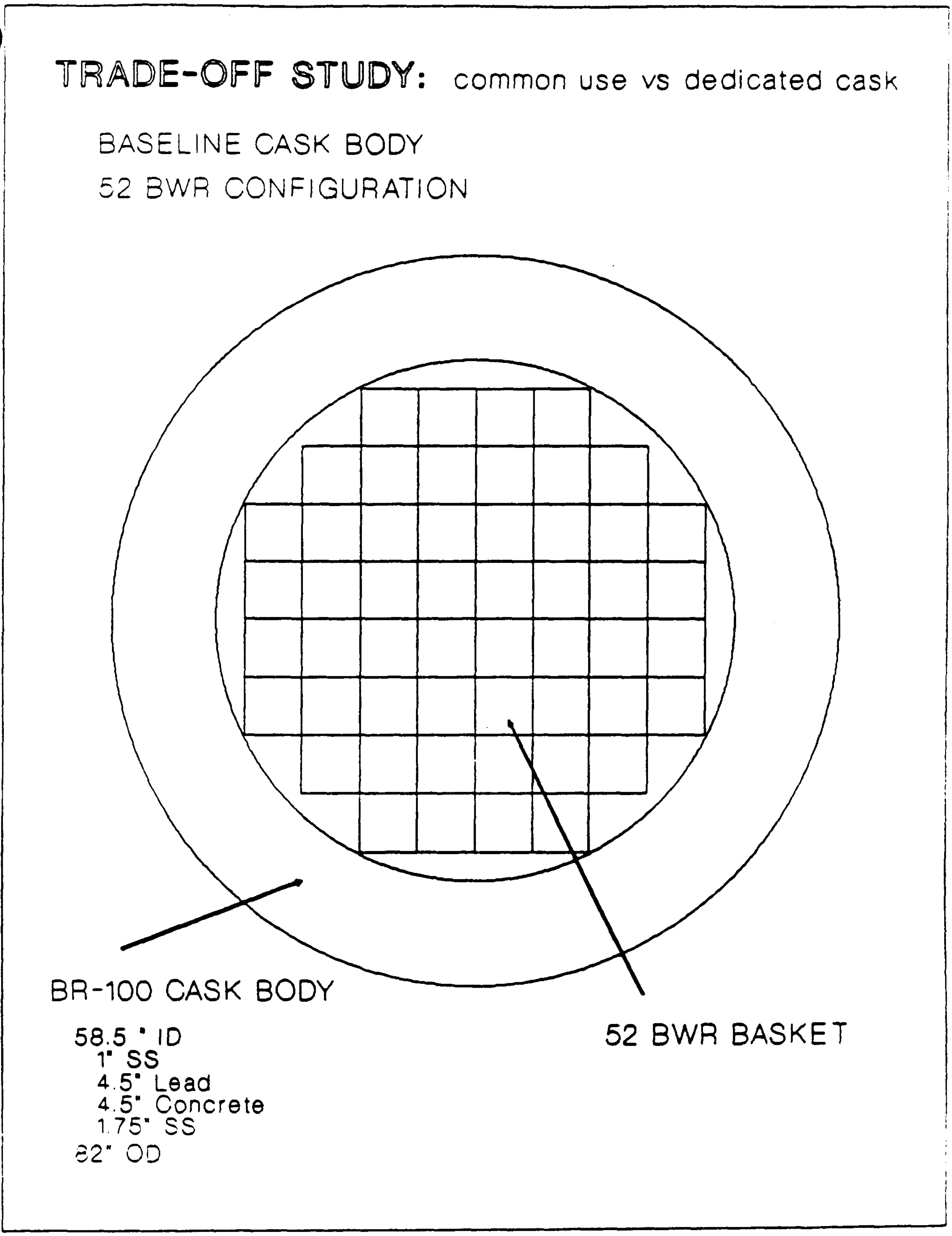
58.5" ID
- 1" SS
- 4.5" Lead
- 4.5" Concrete
- 1.75" SS

82" OD

21 PWR BASKET

FIGURE 3

BR-100 CROSS-SECTION



BR-100 Cask -- Longitudinal Section

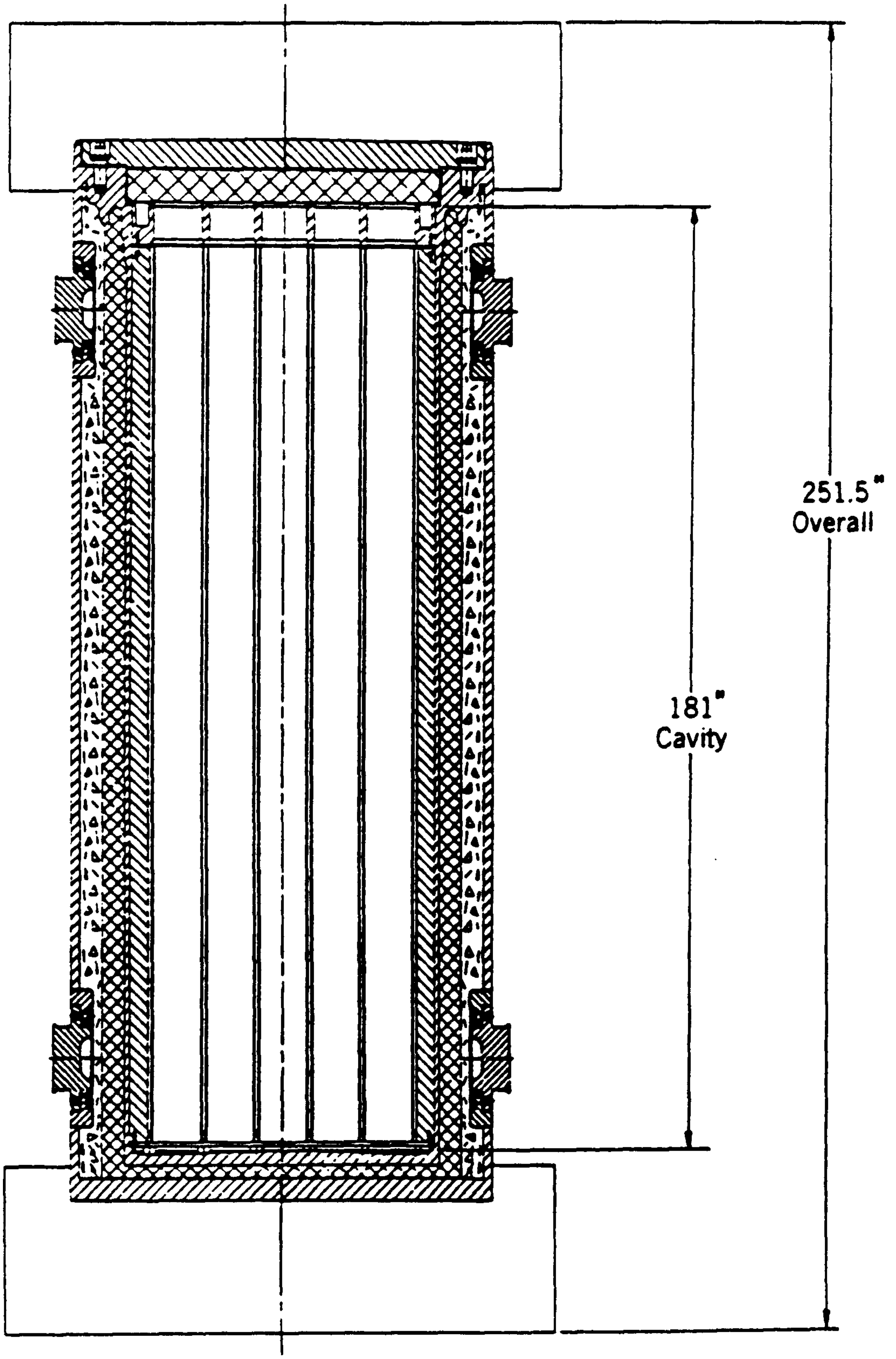
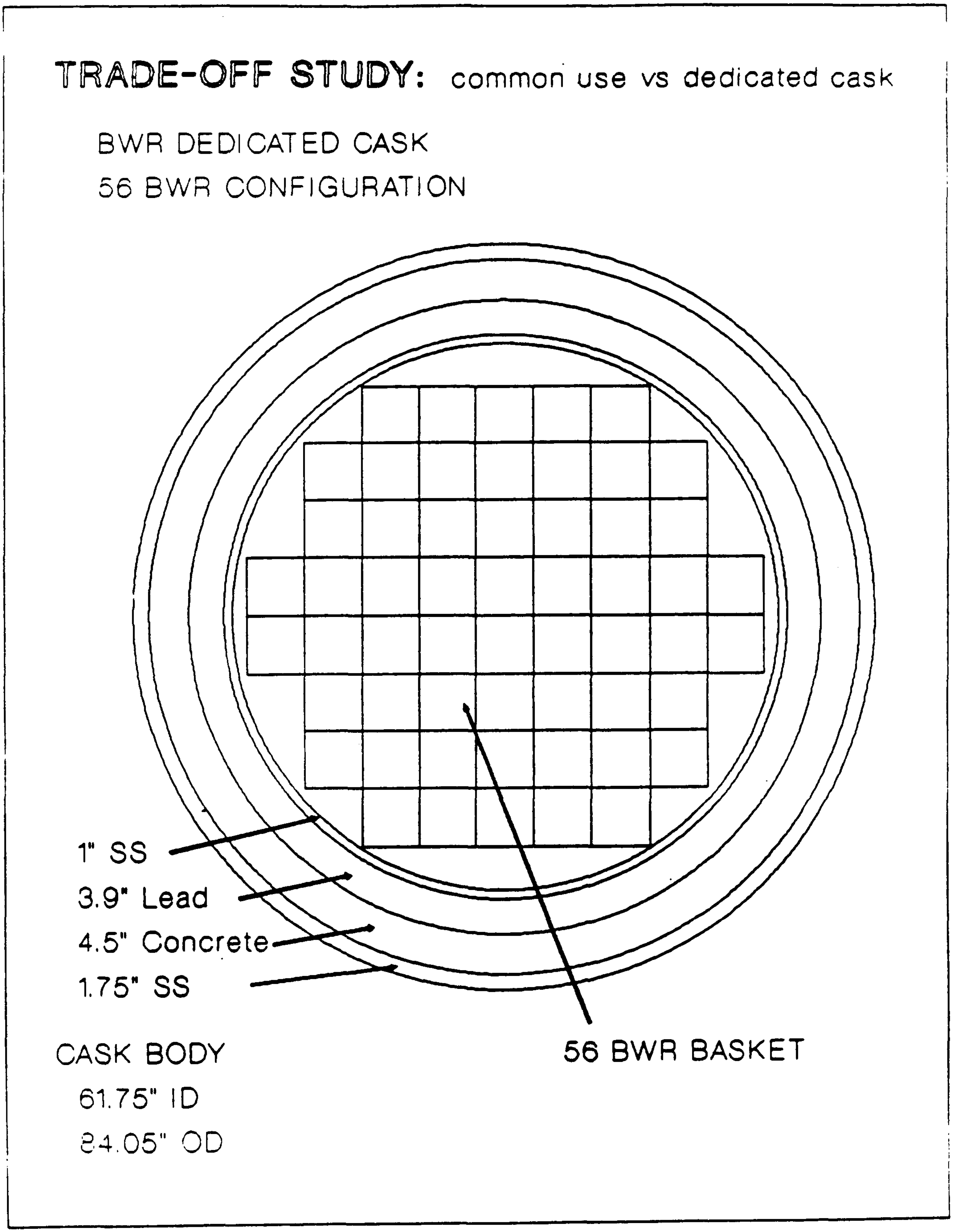


FIGURE 5

PROPOSED BR-100B SPENT FUEL SHIPPING CASK



FROM-REACTOR CASK DEVELOPMENT PROGRAM TRADE-OFF STUDY

B&W BR-100 CASK

EFFECT OF DIFFERENT BURNUP LEVELS ON CAPACITY

Document No. 51-1176106-00

September 1989

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EFFECT OF DIFFERENT BURNUP LEVELS ON CAPACITY

1.0 SUMMARY

This report presents the results of a study performed by the B&W Fuel Company (BWFC) to investigate the capacity for transporting 10-year-cooled, high burnup fuel (up to 60 Gwd/mtu - PWR and up to 50 Gwd/mtu-BWR) in the BR-100 Cask.

The BR-100 cask is designed to transport most efficiently 10-year cooled, 35 Gwd/mtu - PWR and 30 Gwd/mtu - BWR fuel. The options examined for higher burnups were (1) downloading the number of fuel assemblies in the baseline basket, (2) changing the basket design, and (3) selectively loading fuel with different burnup histories and enrichments. Changes to the cask body, increases in environmental dose rates, or lowering of thermal margins were not options investigated.

BWR fuel assemblies examined had burnups of 35, 40, 45, 50 Gwd/mtu, with 3.0, 3.5, 4.0, and 4.5 w/o enrichments, respectively. The studies show that thermal and shielding criteria are maintained for a full complement of 52 assemblies for fuel with those burnups except for the 50 Gwd/mtu fuel. In that case, thermal limits in the concrete region are exceeded unless the cask is downloaded to forty-eight 50-Gwd/mtu fuel assemblies, or unless it is loaded with forty-four 50 Gwd/mtu fuel assemblies and

eight 30 Gwd/mtu design basis fuel assemblies.

PWR fuel assemblies examined had burnups of 40, 45, 50, 55, 60 Gwd/mtu with 3.5, 4.0, 4.5, 4.5, and 4.5 w/o enrichments respectively. For these cases, shielding criteria are limiting for all burnups except the 40 Gwd/mtu fuel. Selective cask loading with nine 45, 50, 55, or 60 Gwd/mtu fuel assemblies surrounded by twelve 35 Gwd/mtu fuel assemblies provides acceptable shielding and thermal results. Alternatively, the cask may be downloaded and an aluminum basket shield provided to accept nineteen 45 Gwd/mtu, sixteen 50 and 55 Gwd/mtu, and fourteen 60 Gwd/mtu fuel assemblies.

The option of mixing fuel of different burnups appears to be attractive since utilization of the cask to its full capacity is achievable. However, availability of fuel with burnups at or below design basis may then dictate the need for cask downloading.

Loaded cask weights for the high burnup fuel are equal to or less than weights with baseline fuel loadings.

The BR-100 cask has the payload flexibility to be attractive for the efficient shipment of high burnup fuel.

2.0 INTRODUCTION

The purpose of this document is to satisfy the requirement in Section 4.10.4 of the Contract Statement-Of-Work to "...conduct trade-off and impact evaluations of the following design considerations on cask payload capacities and costs: Burnup analysis." BWFC has designed a cask, the BR-100 which efficiently transports 10-year cooled fuel, either PWR or BWR in separate baskets. BWFC's goal in performing the investigations described herein was to optimize the payload of the BR-100 cask for extended burnup levels of both PWR and BWR fuel without requiring rework or redesign of the cask body.

Section 3 of this report describes the baseline BR-100 cask and documents its performance in transporting design basis spent fuel. Section 4 describes calculational results, details the design options investigated, and gives particulars on their performance. Section 5 is a conclusion and recounts the results of the investigation.

3.0 BASELINE ASSUMPTIONS

The baseline BR-100 cask is shown in Figures 1, 2, and 3. It has a cavity 58.5 inches in diameter and 181 inches long and has side walls made of—from inside out—1 inch of stainless steel, 4.5 inches of lead, 4.45 inches of borated concrete, and 1.75 inches of stainless steel. Unique baskets for either PWR or BWR fuel have a capacity of 21 or 52 assemblies, respectively. The baskets are made of aluminum cells and supports which efficiently transfer heat from the fuel to the cask inner wall. The hook weight of the BR-100, including interstitial water and the handling equipment, is 100 tons in either PWR or BWR configuration.

The design basis payload for the BR-100 is fuel that is 10-year cooled and has a burnup of 35 Gwd/mtu (PWR) or 30 Gwd/mtu (BWR) and an enrichment of 3.0-4.5%. Sensitivity studies performed by BWFC have shown that, at the burnup levels selected, the lower enrichment has a significantly higher source term and decay heat rate than the higher enrichment. BWFC has also taken into account the axial profile of fuel burnup. PWR spent fuel has a substantial axial span where the burnup level is up to 13% greater than the assembly average (the analogous value for BWR fuel is 20%). The decay heat rates for those burnups were calculated using ORIGEN2 and are shown in Tables 1 and 2 for 10-year cooled fuel. Because of the preliminary nature of the cask design and the nominal values used for some geometric and material properties, an additional 1.10 conservatism factor was used for shielding calculations in determining source strength at the limiting burnup levels.

Table 1

PWR FUEL PARAMETERS - 465 Kg/ASSEMBLY

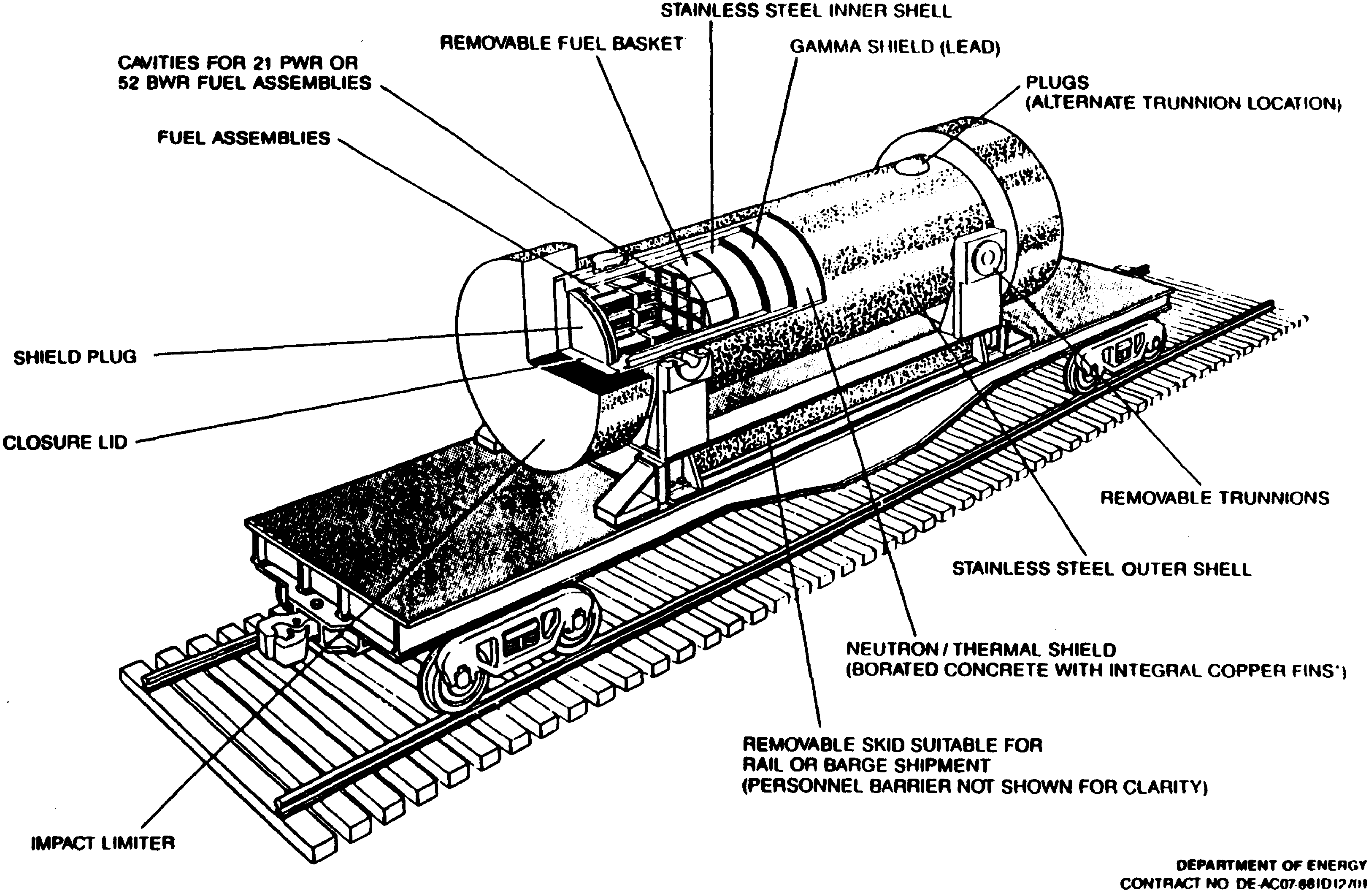
BURNUP (Gwd/mtu)	ENRICHMENT (w/o)	COOLING TIME (Years)	DECAY HEAT (Watt Assy)
35	3.0	10	574
35	4.5	10	533

Table 2

BWR FUEL PARAMETERS - 176.8 Kg/ASSEMBLY

BURNUP (Gwd/mtu)	ENRICHMENT (w/o)	COOLING TIME (Years)	DECAY HEAT (Watt Assy)
30	3.0	10	178
30	4.5	10	169

BABCOCK & WILCOX BR-100
100 TON RAIL / BARGE CASK



DEPARTMENT OF ENERGY
CONTRACT NO DE-AC07-88ID12701

*PATENTED BY
ROBATEL SA

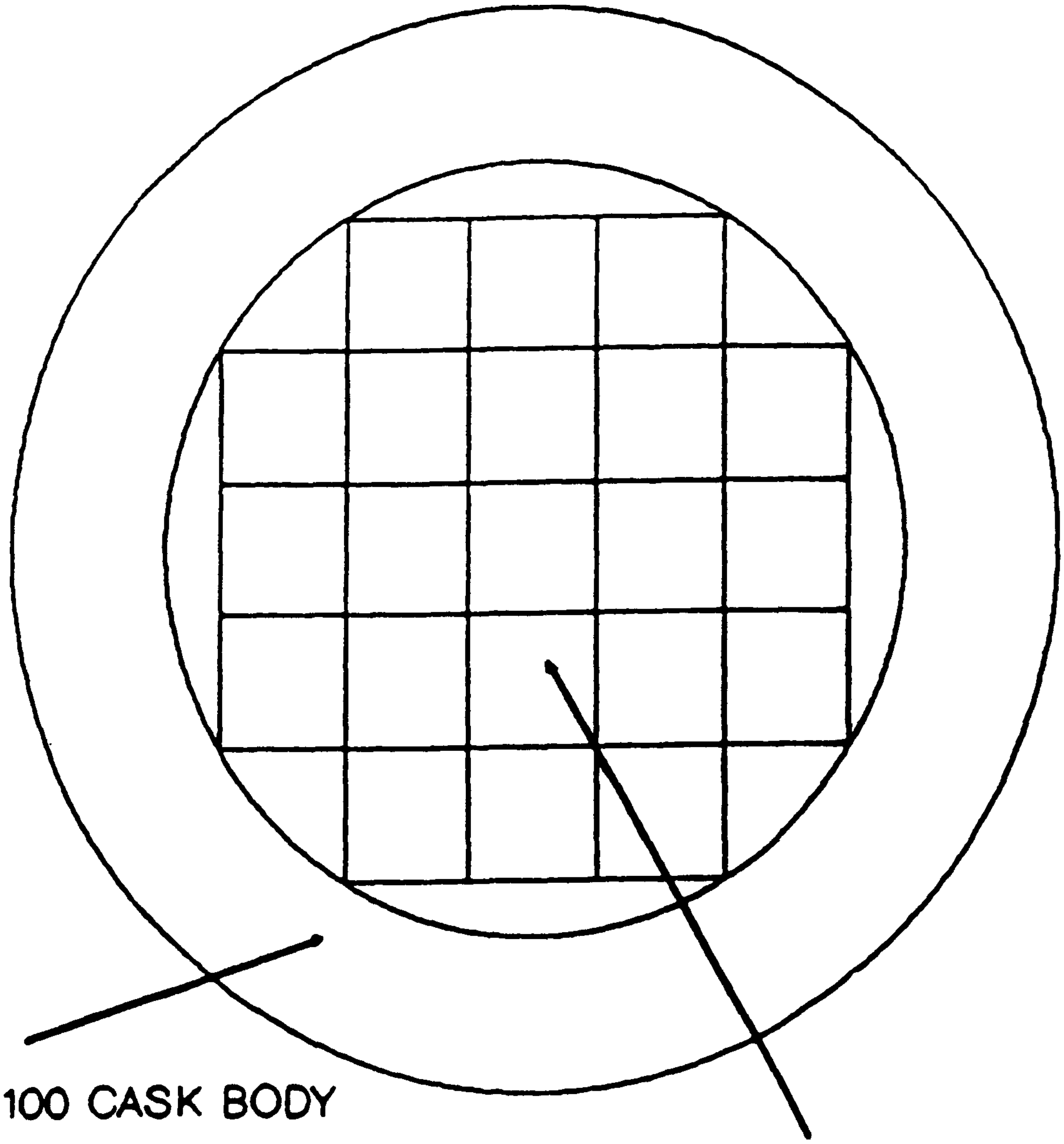
APRIL 1989

51-1176106-00

TRADE-OFF STUDY: 10 years cooled fuel

BASELINE CASK BODY

21 PWR CONFIGURATION



BR-100 CASK BODY

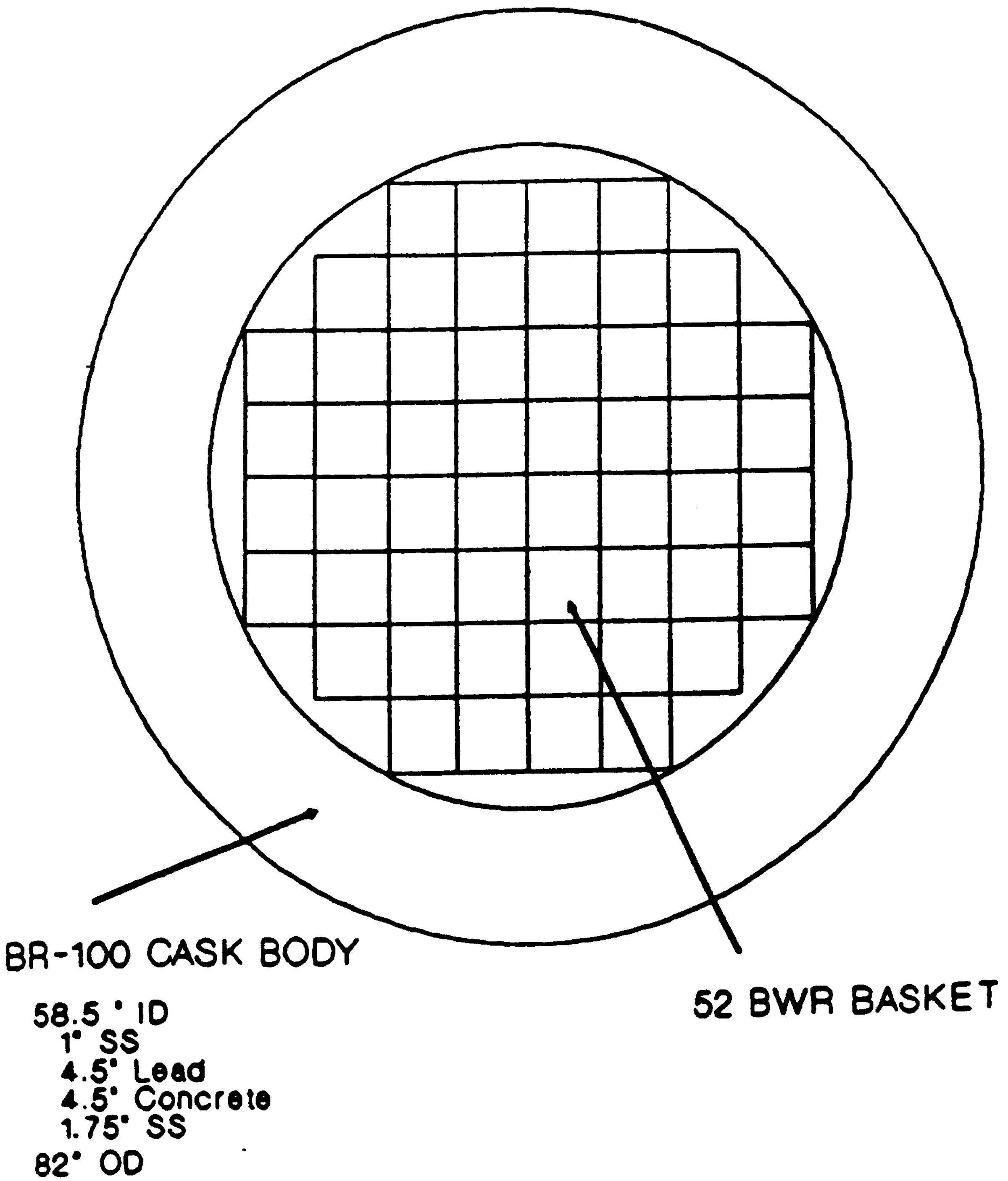
58.5" ID
1" SS
4.5" Lead
4.5" Concrete
1.75" SS
82" OD

21 PWR BASKET

TRADE-OFF STUDY: 10 years cooled fuel

BASELINE CASK BODY

52 BWR CONFIGURATION



4.0 DISCUSSION

Table 3 summarizes the fuel types, burnups, and enrichments considered in this investigation. The study involved assessments of environmental radiation, thermal loading, and total weights.

4.1 Radiation Analysis

From the radiation shielding aspect, the higher fuel burnup effect (which produces greater source terms) is partially negated by the higher enrichments (which result in lower source terms). However, the overall effect is an increase in environmental dose rates requiring, in some cases, either a downloading of the cask or mixed loading of the fuel in order to maintain the dose rate limits specified by 10CFR71. ANISN runs based on ORIGEN2 source terms were made for the discrete geometries described in Section 3 for all fuel types specified in Table 3.

Table 3

BR-100 HIGH-BURNUP CAPACITIES

FUEL TYPE	BURNUP (Gwd/mtu)	ENRICHMENT (w/o)	DECAY HEAT AVERAGE (watts/assy)	QUANTITY/ CASK Fuel Assy's
PWR	35	3.0	574	21
PWR	40	3.5	661	21
PWR	45	4.0	748	19
PWR	50	4.5	837	16
PWR	55	4.5	956	16
PWR	60	4.5	1088	14
BWR	30	3.0	178	52
BWR	35	3.0	215	52
BWR	40	3.5	248	52
BWR	45	4.0	282	52
BWR	50	4.5	316	48

Table 4 shows the 2-meter dose rates that result from a full BWR loading (52 FA's) in the baseline cask. None of the 2-meter dose rates exceed the design dose rate of 10 mrem/hr. Consequently, the baseline cask has adequate shielding for all BWR fuel with the specified burnups and enrichments.

The results of the PWR analyses with a full loading (21 FA's) of high burnup fuel are shown in Table 5. The four highest burnups exceed the design dose rate radiation levels and require cask downloading or mixed loading. The downloading analysis was done in conjunction with inclusion of aluminum inserts on the outer circumference of the fuel basket. The combination of downloading and aluminum shielding reduced the 2-meter dose rates to the design limit of 10 mrem/hr for all cases.

Table 4

BWR 2-Meter Dose Rates With 52 FA's

BURNUP (Gwd/mtu)	ENRICHMENT (w/o)	DOSE RATE (mrem/hr)	ADDITIONAL SHIELDING REQUIRED
30	3.0	5.0	No
35	3.0	5.9	No
40	3.5	7.1	No
45	4.0	8.2	No
50	4.5	9.5	No

Table 5

PWR 2-Meter Dose Rates With 21 FA's

BURNUP (Gwd/mtu)	ENRICHMENT (w/o)	DOSE RATE (mrem/hr)	MIXED FUEL AND/OR DOWNLOADING REQ'D
35	3.0	8.9	No
40	3.5	10.2	No
45	4.0	11.5	Yes
50	4.5	12.9	Yes
55	4.5	17.5	Yes
60	4.5	23.4	Yes

Table 6 shows the acceptable quantity of high burnup assemblies and the corresponding crane hook weights. Figure 4 shows the 19 PWR basket arrangement, Figure 5 shows the 16 PWR basket arrangement, and Figure 6 shows the 14 PWR basket arrangement for downloading the BR-100 baseline cask.

As an alternative to downloading the baseline cask, radiation analyses were performed for selectively loaded fuel arrangements where high burnup fuel assemblies are placed in the center portion of the basket and are surrounded by lower burnup fuel. The results of this study showed that the outer fuel effectively shields the radiation effects of the surrounded higher burnup fuel.

Table 6

PWR HIGH BURNUP FUEL DOWNLOADED CASK QUANTITIES TO REDUCE 2-METER DOSE RATE TO 10 mrem/hr

BURNUP (Gwd/mtu)	NUMBER OF FA's	INTERNAL Al SHIELD THICKNESS (Inches)	DOSE RATE (mrem/hr)	CRANE HOOK LOAD (pounds)
35	21	None	8.9	200,000
40	21	None	10.2	200,000
45	19	0.875	10.2	200,000
50	16	2.25	7.6	198,000
55	16	2.25	10.3	198,000
60	14	3.375	10.2	197,500

Consequently, various combinations of high and lower burnup fuel are possible with the exterior dose rates being mainly a function of the outer-most fuel in the basket. Table 7 illustrates combinations of fuel burnups that can be shipped in the BR-100 baseline cask without downloading.

Table 7

EXAMPLES OF SELECTIVELY LOADED FUEL QUANTITIES ACCEPTABLE IN BR-100 CASK

FUEL TYPE	BURNUP (Gwd/mtu)	NO. FA IN BASKET CENTER		BURNUP (Gwd/mtu)	NO. FA ON PERIPHERY
PWR	45	9	PLUS	35	12
	50	9	PLUS	35	12
	55	9	PLUS	35	12
	60	9	PLUS	35	12
BWR*	50	44	PLUS	30	8

*Location of fuel not crucial in BWR case

Cask top and bottom dose rates are primarily due to Co-60 gamma rays. The calculations for the Co-60 source strengths were based on the assumption that an exposure equivalent to 60 Gwd/mtu had been experienced by the end-fitting hardware. Therefore, inclusion of high burnup fuel in the cask center has been bounded for the dose rates at the cask ends. Dose rates at these locations were below the design limit of 10 mrem/hr.

4.2 Thermal Analysis

The goal of the thermal design of the BR-100 is to maintain the cask, basket, and fuel cladding temperatures within safe limits. The safe thermal design of the BR-100 is achieved by restricting the cask loading to conditions that follow the following project thermal guidelines:

1) The maximum temperature in the concrete neutron/thermal shield will remain below 250°F (121°C), based on conservatively chosen material limitations.

2) The maximum temperature in the aluminum basket will remain below 350°F (177°C), based on allowable ASME material structural properties.

3) The maximum fuel cladding temperature will remain below 680°F (360°C), to meet NRC requirements with margin.

These limits were used to establish various loading limits at the burnups listed in Table 3 for both the BWR and PWR configurations.

The weight percents, or enrichments, were reduced at lower burnups to levels consistent with those burnups and to provide a <u>larger</u> thermal load than provided by high weight percents. Table 3 contains the thermal loads per assembly that are associated with each burnup condition, for both BWRs and PWRs. The peak burnups were obtained by multiplying the assembly average value by the axial peaking factor, 1.20 for BWR and 1.13 for PWR. These peak burnups were used to generate the thermal loads for the trade-off study thermal analyses. All of the thermal loads used in Table 3 are based on 10-year cooled fuel.

The temperature distributions through the cask, from the cask exterior surface to the cask inside radius, and across the cask inside radius-to-basket gap for the various BWR and PWR burnup conditions are shown in Figures 7 and 8 respectively. These two figures indicate that the largest temperature gradients in the cask occur in the concrete only part of the neutron/thermal shield. These figures also demonstrate the effectiveness of the copper fins embedded in the concrete, part of the neutron/thermal shield, for transferring heat. The temperature distributions in Figures 7 and 8 are based on the thermal loads that produce peak concrete temperatures, peak aluminum basket temperatures, and peak cladding temperatures that are within the thermal limits. The BWR 50

Gwd/mtu, PWR 55 Gwd/mtu, and Pwr 60 Gwd/mtu full load conditions exceeded the 250°F (121°C) limit on the concrete temperature and were evaluated using a mixture of lower burnup and the high burnup fuel (see paragraphs that follow).

The BWR temperature distributions are shown in Figure 9 as a function of the assembly peak linear heat rate. Figure 9 shows the cask surface, stainless steel outer shell-to-concrete interface, concrete-to-copper fin interface, and the copper fin-to-lead (gamma shield) interface temperatures. Figure 10 provides that same information for the PWR loadings. These temperatures represent the maximum temperatures in the most limiting region of the cask. None of the thermal limits are exceeded for any of the BWR and PWR conditions (see Table 8).

Table 8

BWR AND PWR FULL-LOAD TEMPERATURE SUMMARY

-Temperatures in the BR-100 for BWR fuel, F (C)-

*Loc, inch (m)	30/3.0 w/o	35/3.0 w/o	40/3.5 w/o	45/4.0 w/o
41.00(1.041)	184.2(84.6)	197.5(91.9)	208.6(98.1)	219.7(104.3)
39.25(0.997)	186.8(86.0)	200.7(93.7)	212.3(100.2)	223.8(106.6)
38.85(0.987)	192.2(89.0)	207.4(97.4)	220.1(104.5)	232.8(111.6)
34.75(0.883)	197.6(92.0)	214.0(101.1)	227.9(108.8)	241.7(116.5)
30.25(0.768)	201.4(94.1)	218.7(103.7)	233.3(111.8)	247.9(119.9)
29.25(0.743)	203.4(95.2)	221.1(105.1)	236.1(113.4)	251.1(121.7)
29.20(0.742)	212.7(100.4)	231.5(110.8)	249.1(120.6)	265.8(129.9)
Peak Basket	239.0(115.0)	263.8(128.8)	286.7(141.5)	308.9(153.8)
Peak Clad	285.4(140.8)	316.8(158.2)	344.9(173.8)	371.9(188.8)

-Temperatures in the BR-100 for PWR fuel, F (C)-

*Loc, inch (m)	35/3.0 w/o	40/3.5 w/o	45/4.0 w/o	50/4.5 w/o
41.00(1.041)	194.9(90.5)	203.6(95.3)	212.2(100.1)	223.9(106.6)
39.25(0.997)	197.9(92.2)	207.1(97.3)	216.0(102.2)	228.3(109.1)
38.85(0.987)	203.7(95.4)	213.6(100.9)	223.3(106.3)	236.7(113.7)
34.75(0.883)	210.1(98.9)	220.8(104.9)	231.4(110.8)	246.0(118.9)
30.25(0.768)	214.6(101.4)	225.9(107.7)	237.1(113.9)	252.6(122.6)
29.25(0.743)	216.9(102.7)	228.5(109.2)	240.0(115.6)	255.9(124.4)
29.20(0.742)	227.9(108.8)	240.8(116.0)	253.6(123.1)	271.3(132.9)
Peak Basket	274.3(134.6)	293.2(145.1)	312.2(155.7)	338.6(170.3)
Peak Clad	364.2(184.6)	390.7(199.3)	416.8(213.8)	452.6(233.7)

Notes:

1. 30/3.0 w/o denotes 30 Gwd/mtu 3.0 w/o fuel.
2. The BWR base case is 30/3.0 w/o. The PWR base case is 35/3.0 w/o.
3. Values in parentheses, (), are metric. The metric units on temperature are degress Celsius. The metric units on location are meters.

* Loc denotes the radial location in the cask, from the cask center. 29.20 is the basket OD, 29.25 is the inner shell ID, 30.25 is the lead ID, 34.75 is the concrete ID, 38.85 is the copper fin OD, 39.25 is the concrete OD, and 41.00 is the OD.

There were several full-load conditions under which the concrete temperature limit of 250°F (121°C) was exceeded, although the basket and fuel cladding thermal limits were not exceeded. These conditions were the BWR 50 Gwd/mtu, PWR 55 Gwd/mtu, and PWR 60 Gwd/mtu full-load burnups. For these high burnup conditions some of the low burnup baseline fuel was added in the exterior cells of the basket, to reduce the thermal load on the cask. The use of low burnup fuel in the outer cells of the basket was dictated by shielding constraints. Any loading using lower burnup fuel in any of the basket cells will produce less limiting temperature conditions. Figure 11 shows the temperature profiles across the cask and the cask inside radius-to-basket outside radius gap for the mixed load BWR and both PWR arrangements. The peak basket and fuel cladding temperatures for the mixed load conditions are summarized in Table 9. The peak basket temperatures occur near the center of the cask. The peak cladding temperature occurred in the center assembly. None of the thermal limits are exceeded for these mixed load BWR and PWR conditions.

Table 9

BWR AND PWR MIXED LOAD TEMPERATURE DISTRIBUTIONS

-Temperatures in the BR-100, F (C)-

*Loc, inch (m)	BWR 50/4.5 w/o, 30/3.0 w/o	PWR 55/4.5 w/o, 35/3.0 w/o	PWR 60/4.5 w/o, 35/3.0 w/o
41.00(1.041)	223.9(106.6)	213.9(101.1)	220.5(104.7)
39.25(0.997)	228.3(109.1)	217.8(103.2)	224.7(107.1)
38.85(0.987)	237.7(114.3)	225.2(107.3)	232.8(111.6)
34.75(0.883)	247.1(119.5)	233.5(111.9)	241.8(116.6)
30.25(0.768)	253.6(123.1)	239.3(115.2)	248.1(120.1)
29.25(0.743)	256.9(124.9)	242.3(116.8)	251.3(121.8)
29.20(0.742)	272.4(133.6)	256.1(124.5)	265.6(129.8)
Peak Basket	319.9(159.9)	327.0(163.9)	345.5(174.2)
Peak Clad	388.3(197.9)	455.7(235.4)	486.5(252.5)

Notes:

1. 30/3.0 w/o denotes 30 Gwd/mtu 3.0 w/o fuel.
2. Values in parentheses, (), are metric. The metric units on temperature are degrees Celsius. The metric units on location are meters.

* Loc denotes the radial location in the cask, form the cask center. 29.20 is the basket OD, 29.25 is the inner shell ID, 30.25 is the lead ID, 34.75 is the concrete ID, 38.85 is the copper fin OD, 39.25 is the concrete OD, and 41.00 is the OD.

TRADE-OFF STUDY: higher burnup levels

MAX. CAPACITY W/O MODIFICATION OF CASK BODY

19 PWR CONFIGURATION for 45 GWD / 4.0 % FUEL

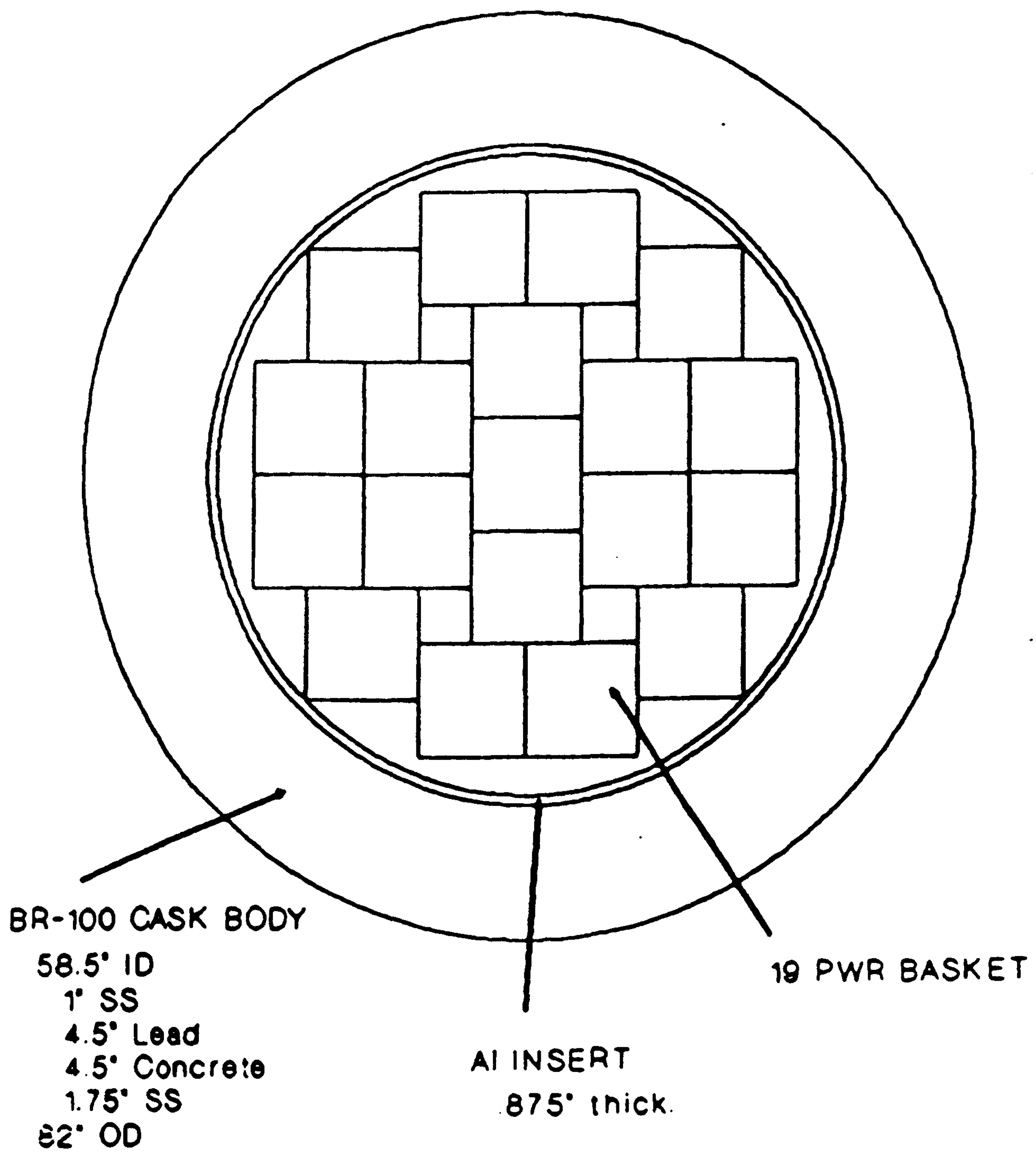


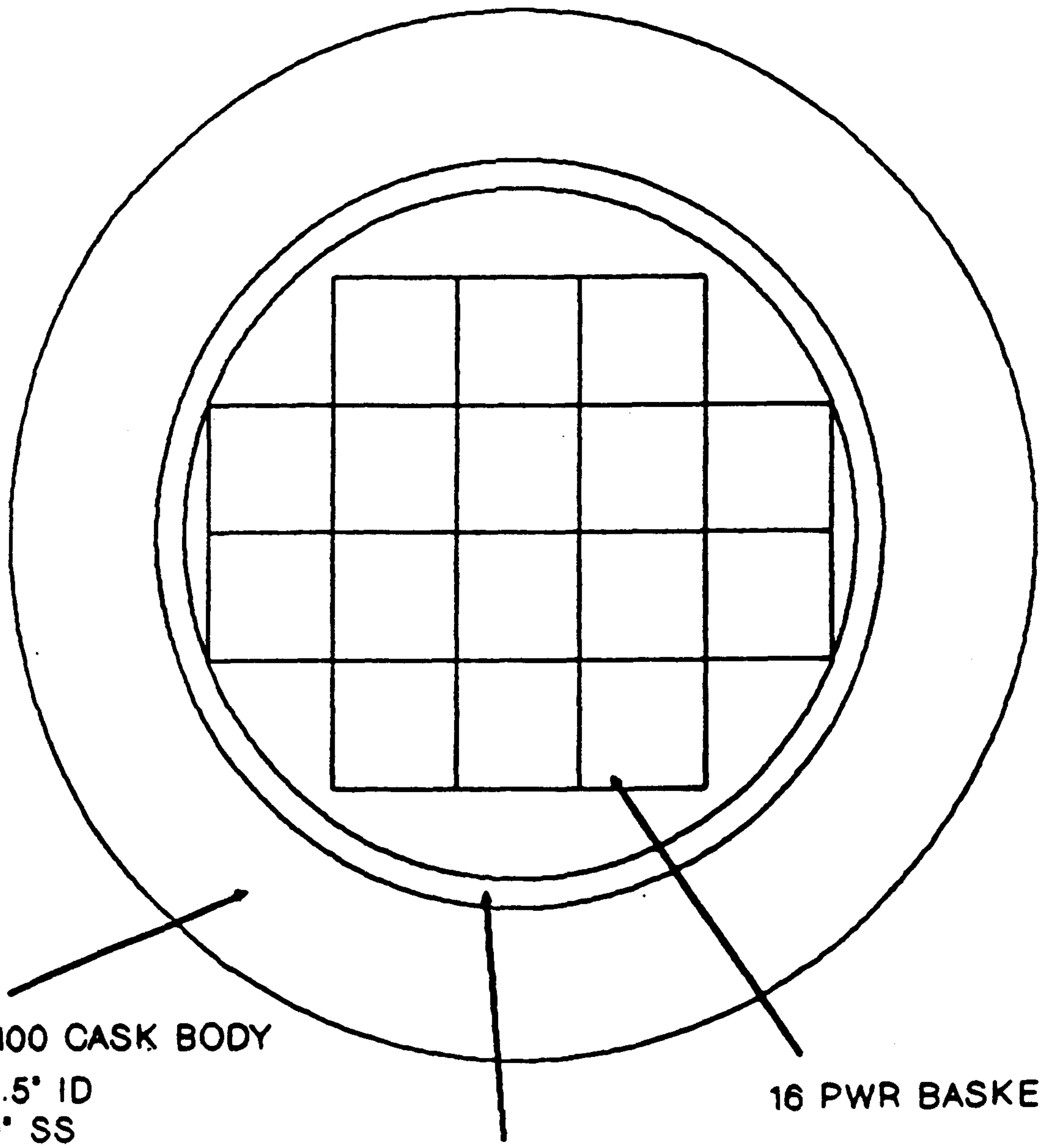
FIGURE 5

TRADE-OFF STUDY: higher burnup levels

MAX. CAPACITY W/O MODIFICATION OF CASK BODY

16 PWR CONFIGURATION for 50 GWd / 4.5 %

and 55 GWd / 4.5 % FUEL



TRADE-OFF STUDY: higher burnup levels

MAX. CAPACITY W/O MODIFICATION OF CASK BODY
14 PWR CONFIGURATION for 60 GWd / 4.5 % FUEL

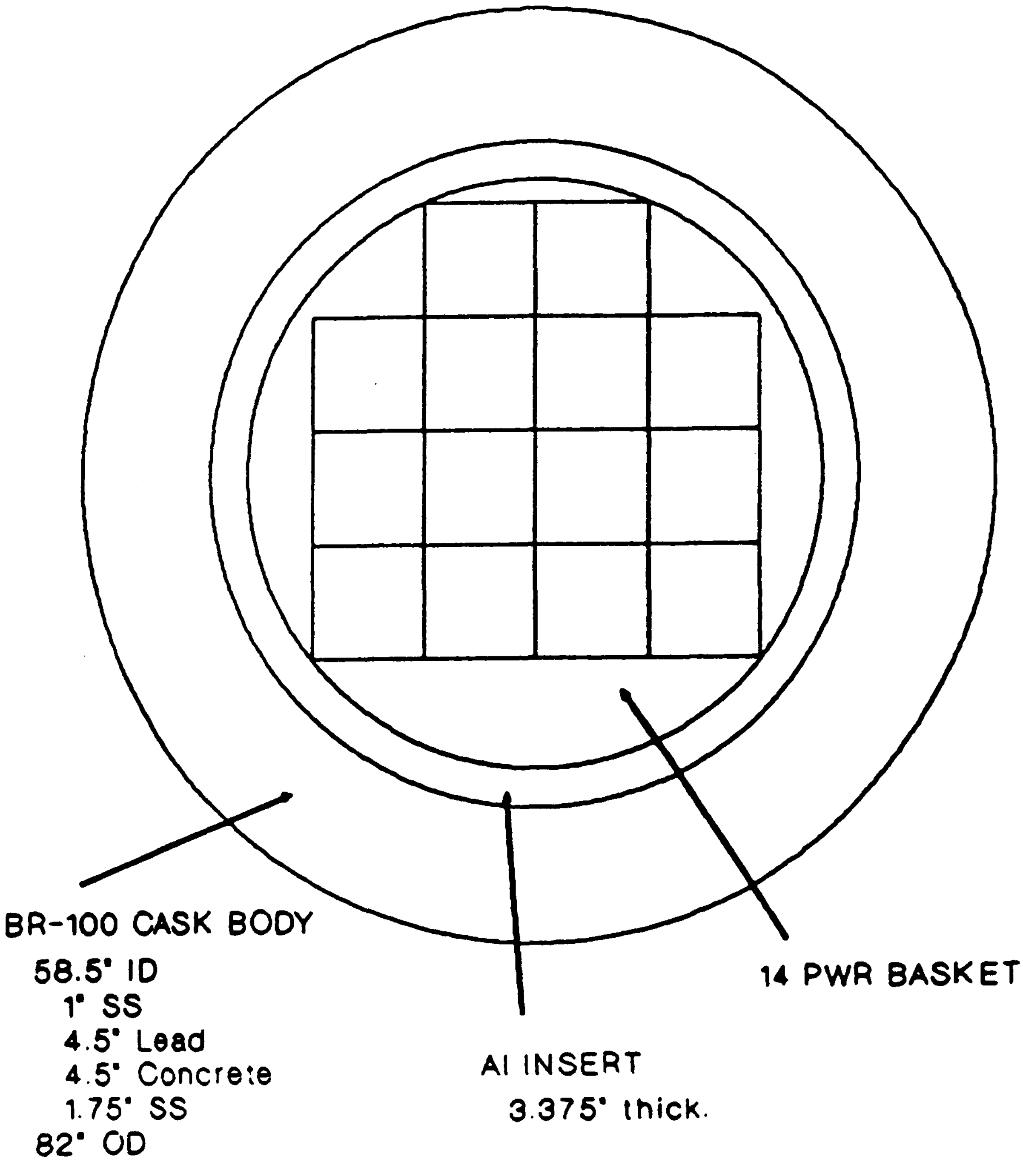


Figure 7 – BWR cask temperature distribution.

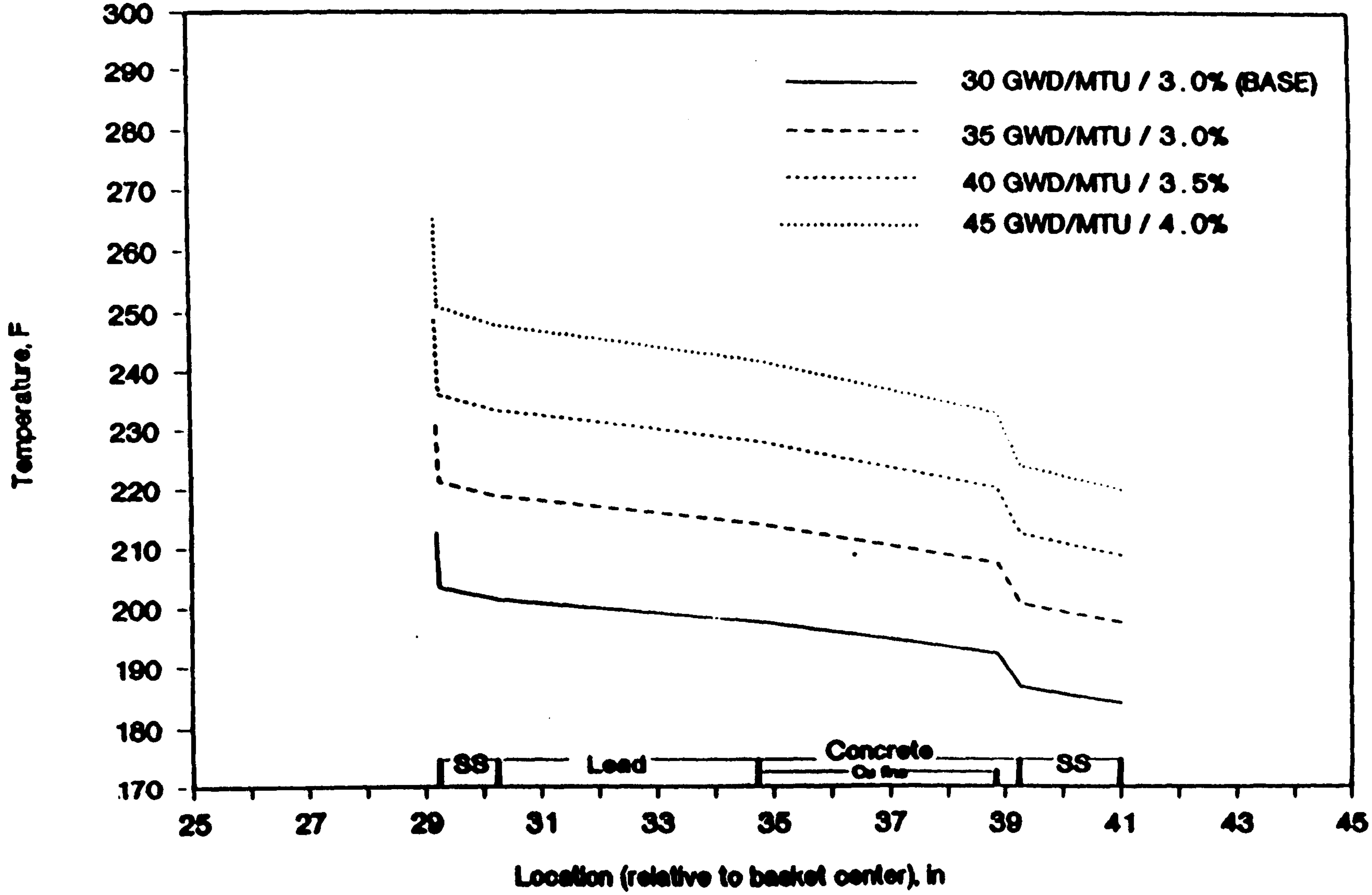


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Figure 8 – PWR cask temperature distribution.

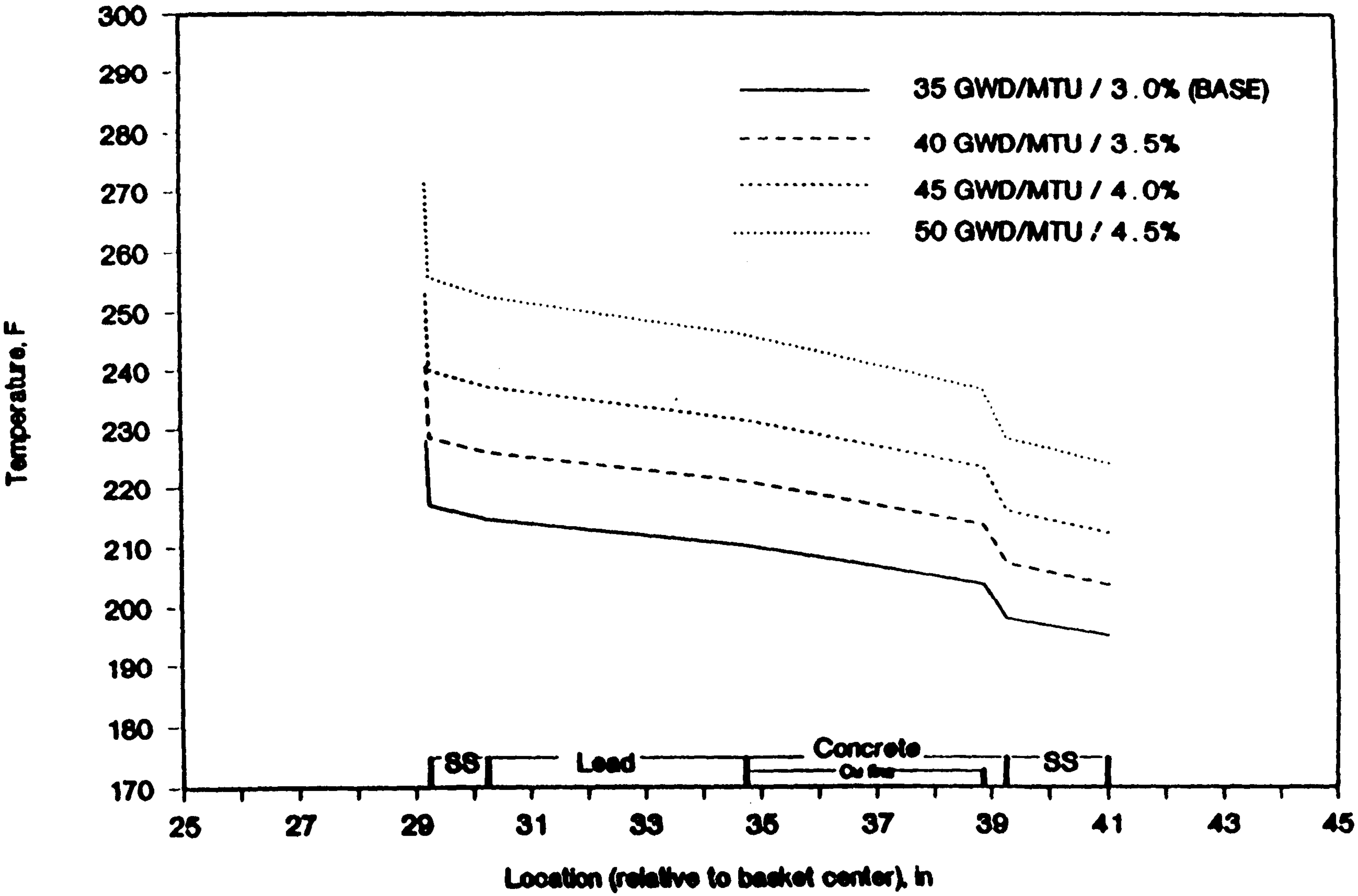


Figure 9 - BWR cask interface temperatures.

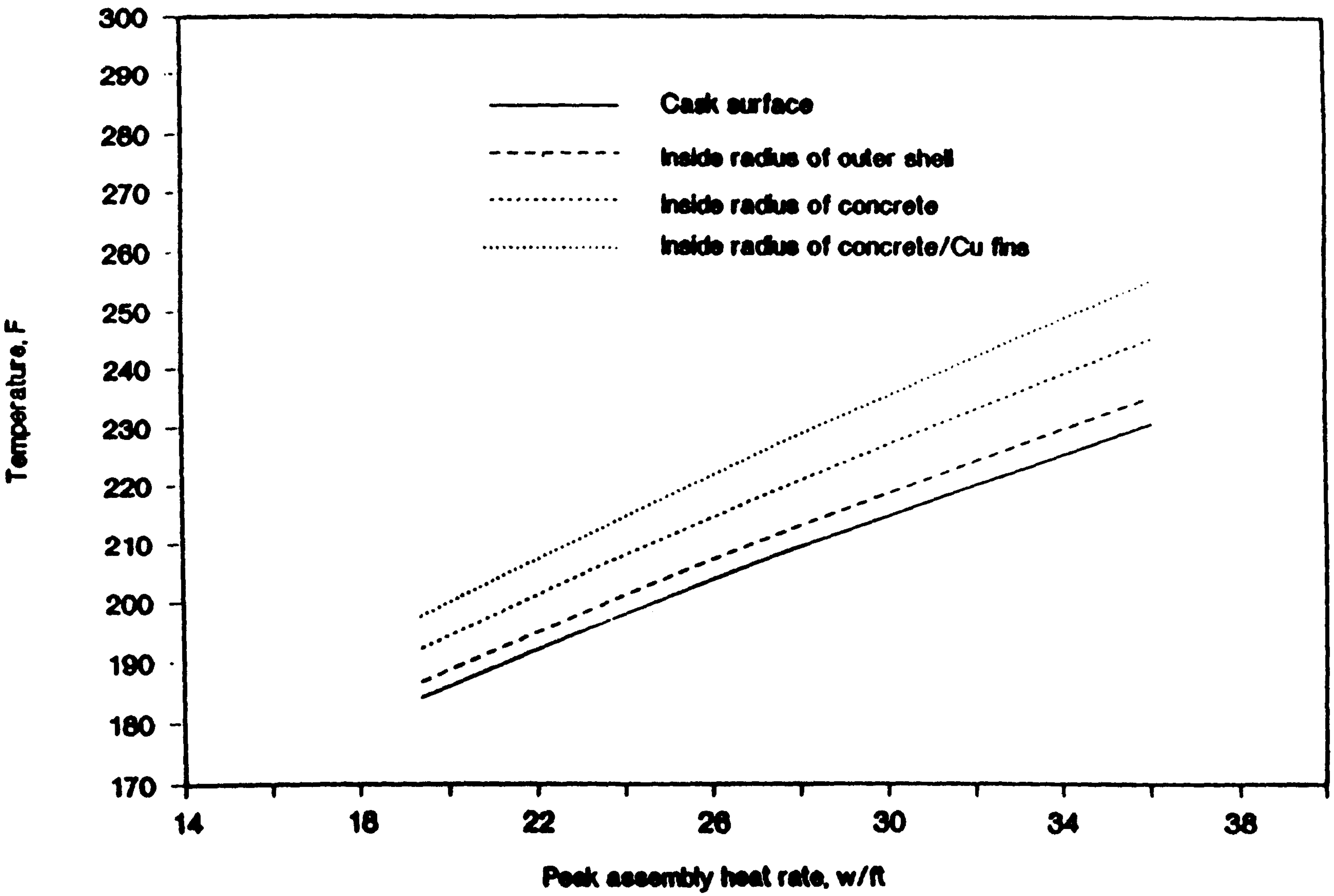


Figure 10 – PWR cask interface temperatures.

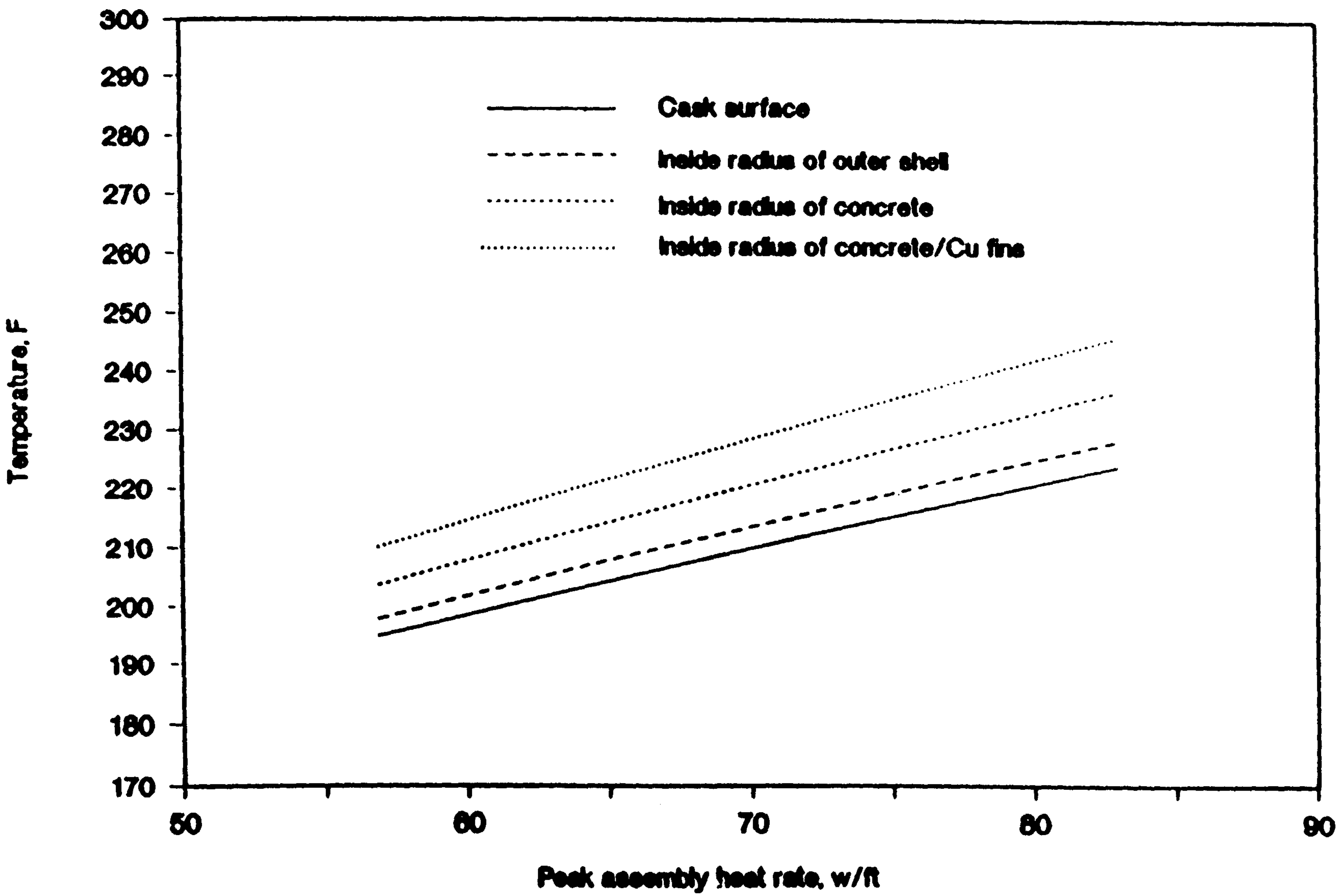
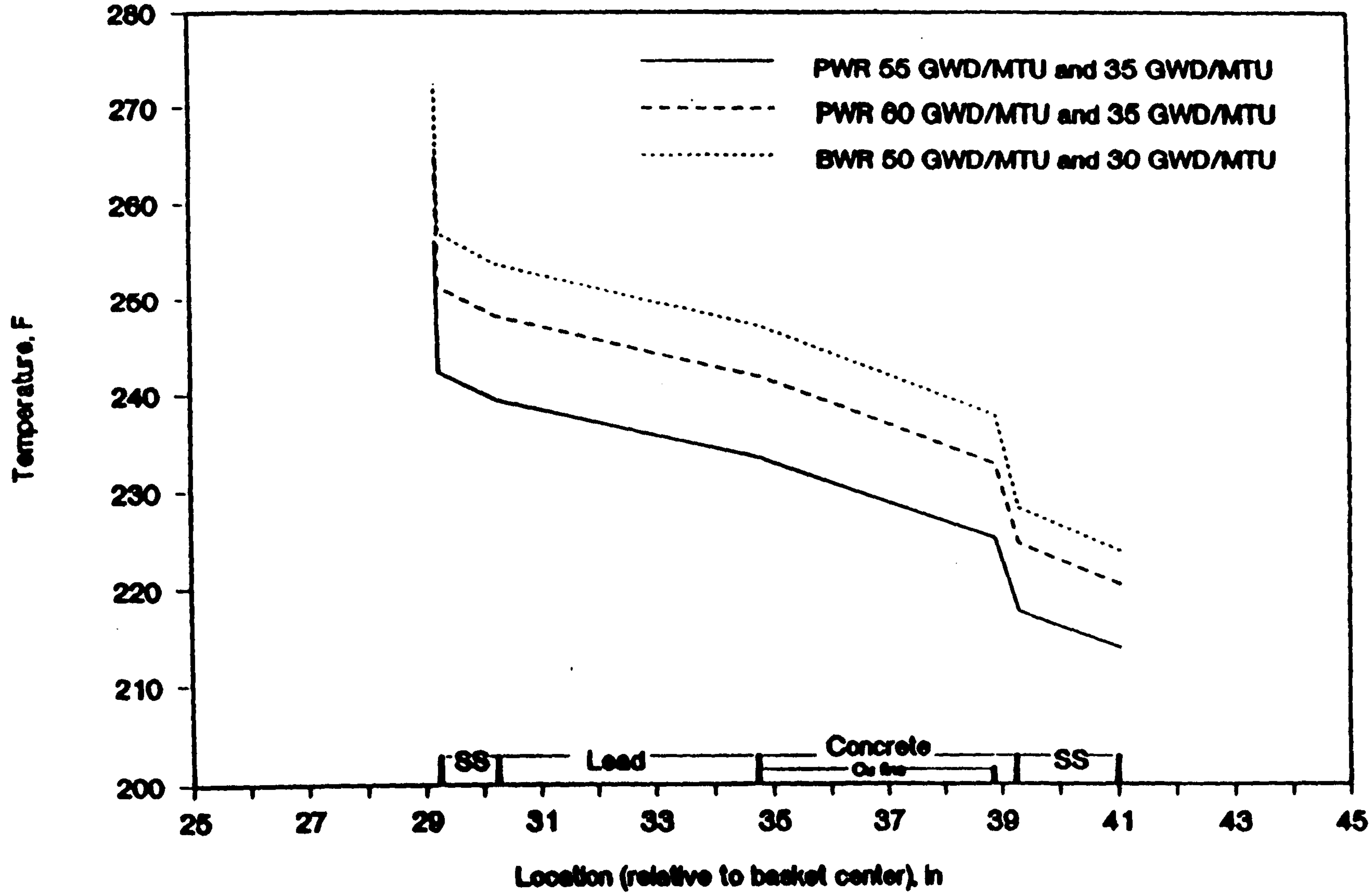


Figure 11 – Mixed load BWR and PWR cask temperature distributions.



5.0 CONCLUSION

Radiation, thermal, and weight studies were completed to determine the acceptable conditions for transporting various burnups and enrichments of both PWR and BWR spent fuel assemblies in the baseline and redesigned basket of the BR-100 cask. The results are that a full loading (52 FA's) of BWR fuel assemblies of all considered burnups and enrichments with 10-year cooling can be transported in the BR-100 cask with no modifications with the exception of 50 Gwd/4.5 w/o fuel. Thermal limitations in the concrete region dictate either a slight downloading or selective fuel loading for this case. Crane hook weight remains at or below 200,000 pounds for all cases.

PWR fuel assembly loadings in the BR-100 cask are shielding limited for the four highest burnups considered. For these fuel assemblies it is acceptable to load the basket centrally with nine high burnup assemblies surrounded on the periphery by twelve design basis fuel assemblies. This produces a full complement of 21 fuel assemblies, maintains dose rates under the 2-meter limit of 10 mrem/hr, and meets thermal loading criteria. Downloading of the cask in conjunction with supplemental aluminum shielding in the outer basket will allow shipment of reduced numbers of the high burnup fuel assemblies. Again, dose rates and thermal margins will meet design criteria. Cask hook weight is equal to or less than the baseline PWR cask loading weight with 21 full assemblies in all cases.

This report shows that the BR-100 cask has the capability to accommodate a wide range of high burnup BWR and PWR fuel with minor fuel loading management or with fuel basket redesign only. It also provides assurance that other spent fuel scenarios which may evolve from future needs can be accommodated.

7.0 THE EFFECT ON CAPACITY OF LOW-BURNUP FUEL

Section 4.10.4 of the contract Statement of Work requests an evaluation of the effect on cask capacity of burnups ranging from 5 Gwd/mtU up to the design baseline (35 Gwd/mtU for PWR fuel, 30 Gwd/mtU for BWR). High enrichments were to be assumed for these studies.

B&W believes that the trade-off studies presented in chapters one through six of this section cover all the relevant information on the issue of burnup and enrichment. The only factor that low burnup would affect is the burnup assumption that B&W uses for high enrichment fuel. If low burnups did occur on high enrichment fuel, Chapter 1 (The Effect on Capacity of Burnup Credit) addresses its impacts. Specifically, the payload on PWR fuel would be reduced from 21 assemblies to 17 assemblies. There would be no effect on BWR fuel.

No discrete report on the effects of low burnup fuel are planned.

CASK SYSTEM DEVELOPMENT PROGRAM TRADE-OFF STUDY

B&W BR-100 CASK

EFFECT OF DIFFERENT BURNUP LEVELS AND AGES ON CASK CAPACITY

Document No. 51-1177245-02

April 1990

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Record of Revisions

Revision #	Description	Date
00	Original Issue	12/89
01	Incorporate DOE Comments Page 31 - Added reference for power peaking factor Page 33 - Corrected figure numbers from 15 & 16 to 18 & 19 Page 52 - Figure 18 added Page 53 - Old Figure 18 is now Figure 19	2/90
02	Page 6 - Changed BWR thermal limitations from one to three design base fuels Page 7 - Changed capacity of cask 2 for BWR 4/5/3% and added notations for BWR 4/5/3%, 50/5/4%, and 50/10/4% in Table 1 Page 19 - Added *** footnote to Table 4 Page 34 - Added 2 paragraphs on BWR Thermal Limitations Page 40A - Added Table 16	4/90

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EFFECT OF DIFFERENT BURNUP LEVELS AND AGES ON CASK CAPACITY

1.0 SUMMARY

This report presents the results of a study performed by the B&W Fuel Company (BWFC) to investigate the effect of transporting high burnup fuel with cooling ages ranging from 5 to 15 years in the BR-100 cask or one very similar.

The baseline BR-100 cask was designed by BWFC to efficiently transport 10-year cooled, 35 Gwd/mtU - PWR and 30 Gwd/mtU - BWR fuel. The options examined in this study were (1) downloading the number of fuel assemblies in the baseline cask and, (2) changing the basket and cask body design. Increases in environmental dose rates beyond Federal limits, or exceeding design thermal limits were not options investigated.

The downloading studies involved the reduction in payload of the baseline cask (also called cask 1) with supplemental annular aluminum shielding inside the cask cavity (see Section 4.1.1). The other two cask designs employ more shielding in the form of lead and concrete and have a reduced design capacity to accommodate the more active fuel. This is discussed in Section 4.1.3.

All capacities were limited by dose rates rather than thermal

considerations except for two PWR design base fuels (55/5/4% and 55/10/4%) and three BWR design base fuel (40/5/3%, 50/5/4% and 50/10/4%).

The design base fuels and the three casks' capacities for both downloading and redesign are summarized in Table 1. Loaded cask weights for the higher burnup fuel are equal to or less than weights with baseline fuel loadings.

The payload flexibility of the baseline BR-100 cask can be seen from the comparison on Table 1. It has the ability to transport, or be downloaded to transport, high burnup and short-aged fuel, in many cases nearly as well as redesigned casks. Although a rigorous life-cycle-cost analysis was not made of using only the baseline cask for all fuel versus using two or three casks specifically designed for different burnup/age combinations, the baseline BR-100 appears to provide an efficient shipping cask for all the fuel projected to be in inventory by the year 2020.

Table 1

Summary of Payload Capacity of Three Casks for all Considered Design Base Fuels

Cask 1: 21 PWR/52 BWR (BR-100)
Cask 2: 18 PWR/45 BWR
Cask 3: 16 PWR/40 BWR

PWR Fuel (burnup/age/enrichment)	Cask 1*	Cask 2	Cask 3
35/5/3%	(16)	18	16
45/5/4%	(14)	(14)***	16
55/5/4%	(12)	(12)***	12**
35/10/3%	21	18	16
45/10/4%	(18)	18	16
55/10/4%	(14)	16**	16
45/15/4%	21	18	16
55/15/4%	(16)	18	16

BWR Fuel (burnup/age/enrichment)	Cask 1	Cask 2	Cask 3
30/5/3%	52	45	40
40/5/3%	(37)	(37)***	37**
50/5/4%	(30)	(30)***	30**
30/10/3%	52	45	40
40/10/3%	52	45	40
50/10/4%	(45)**	45	40
40/15/3%	52	45	40
50/15/4%	52	45	40

* Values in parentheses are downloaded quantities where aluminum annular shields are used inside the cask.

** Download capacity due to thermal restrictions. No basket change is necessary.

*** Estimated capacity based on similar thermal and shielding analyses.

2.0 INTRODUCTION

The purpose of this document is to satisfy the requirement in Section 4.10.4 of the Contract Statement-of-Work (SOW) as supplemented by contract modification A005, "Additional Trade-Off Studies", Nov. 1989, to "...conduct trade-off and impact evaluations of design considerations on cask payload capacities and costs".

BWFC has designed a cask, the BR-100, which efficiently transports 10-year cooled fuel, either PWR (35 GWd/mtU) or BWR (30 GWd/mtU) in separate baskets. BWFC's goal in performing the investigations described herein was two-fold: 1) To optimize the payload of the baseline BR-100 cask for higher burnup levels and for different ages of both PWR and BWR fuel without rework or redesign of the cask body, 2) To perform a conceptual redesign of the BR-100 cask body (changing only dimensions not materials) to accommodate the specified higher burnup levels and different ages.

Section 3 of this report describes the baseline BR-100 cask and documents its performance in transporting the original SOW design basis spent fuel (see Tables 1 and 2). Section 4 describes calculational results, details the design options investigated, and gives particulars on their performance.

Section 5 is a conclusion and recounts the results of the investigation.

3.0 BASELINE ASSUMPTIONS

The baseline BR-100 cask is shown in Figures 1, 2, and 3. It has a cavity 58.5 inches in diameter and 181 inches long and has side walls made of--from inside out--1 inch of stainless steel, 4.5 inches of lead, 4.5/inches of borated concrete, and 1.75 inches of stainless steel. Unique baskets for either PWR or BWR fuel have a capacity of 21 or 52 assemblies, respectively. The baskets are made of aluminum cells and supports which efficiently transfer heat from the fuel to the cask inner wall. The hook weight of the BR-100, including interstitial water and the handling equipment, is approximately 100 tons in either PWR or BWR configuration.

The baseline design basis payload for the BR-100 is fuel that is 10-year cooled and has a burnup of 35 GWd/mtU (PWR) or 30 Gwd/mtU (BWR) and an enrichment of 3.0-4.5%. Sensitivity studies performed by BWFC have shown that, at the burnup levels selected, the lower enrichment has a significantly higher source term and decay heat rate than the higher enrichment. BWFC has also taken into account the axial profile of fuel burnup. PWR spent fuel has a substantial axial span where the burnup level is up to 13% greater than the assembly average (the analogous value for BWR fuel is 20%). The decay heat rates for those burnups were calculated using ORIGEN2 and are shown in Tables 2 and 3 for 10-year cooled fuel. Because of the preliminary nature of the cask design and the nominal values used for some geometric and material properties, an additional 1.10 conservatism factor was used for shielding calculations in determining source strength at the limiting burnup levels.

Table 2

BASELINE PWR FUEL PARAMETERS - 465 Kg/ASSEMBLY

BURNUP (Gwd/mtu)	ENRICHMENT (w/o)	COOLING TIME (Years)	DECAY HEAT (Watt/Assy)
35	3.0	10	574
35	4.5	10	533

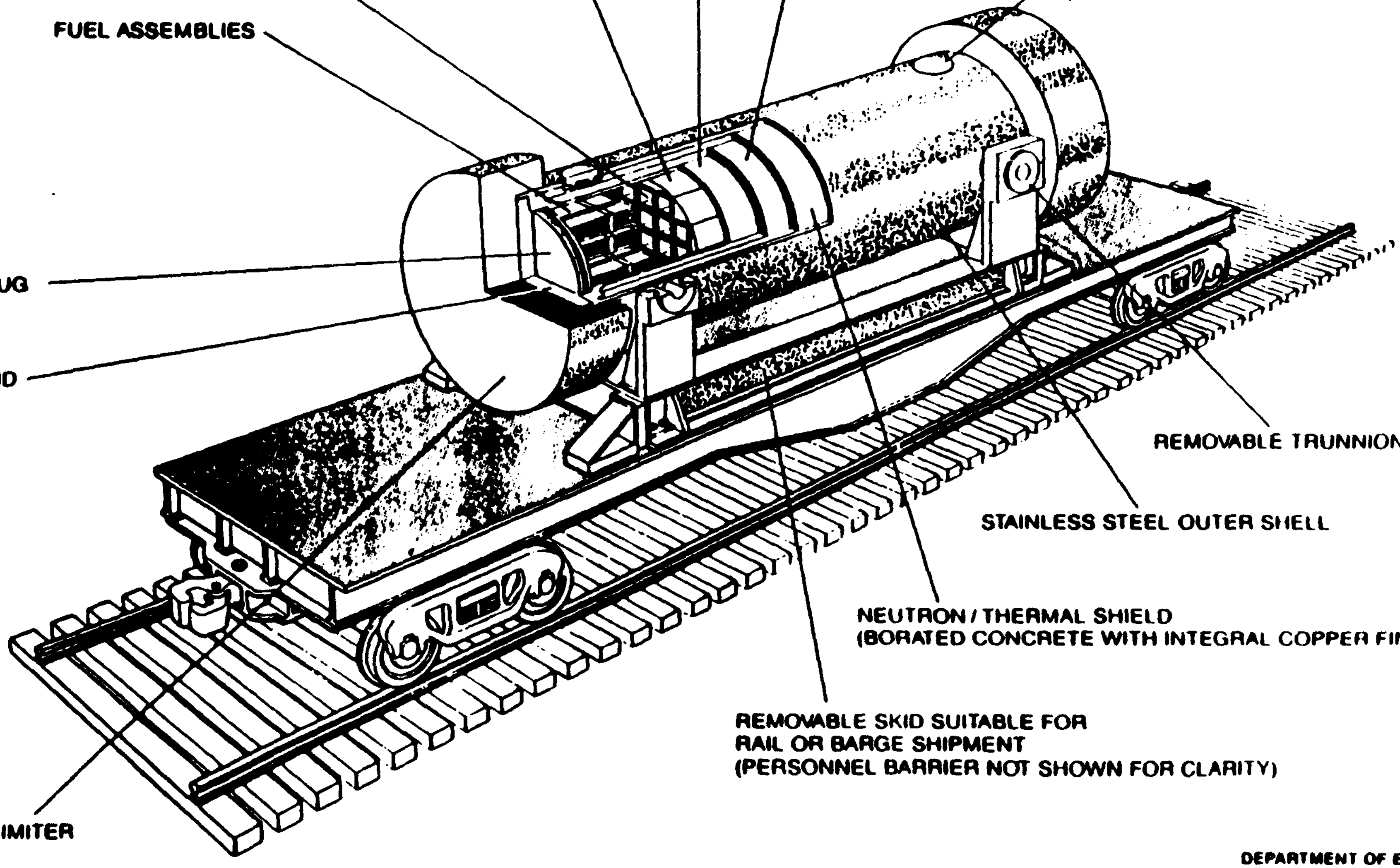
Table 3

BASELINE BWR FUEL PARAMETERS - 176.8 Kg/ASSEMBLY

BURNUP (Gwd/mtu)	ENRICHMENT (w/o)	COOLING TIME (Years)	DECAY HEAT (Watt/Assy)
30	3.0	10	178
30	4.5	10	169

FIG 2

BABCOCK & WILCOX BR-100
100 TON RAIL / BARGE CASK



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51-1177245-02

DEPARTMENT OF ENERGY
CONTRACT NO DE AC07-88ID[illegible]

* PATENTED BY
ROBATEL SA

APRIL [illegible]

FIGURE 2

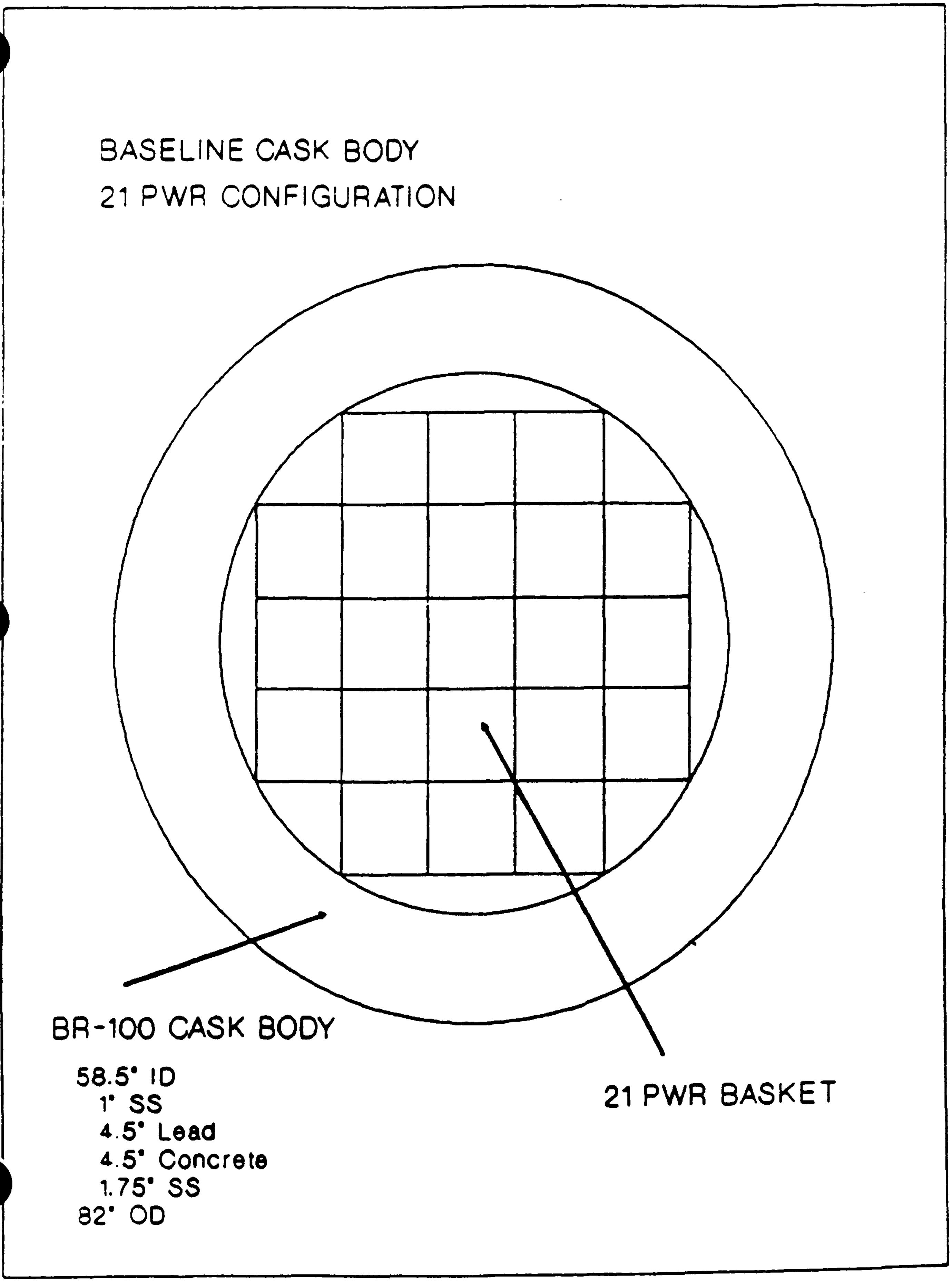
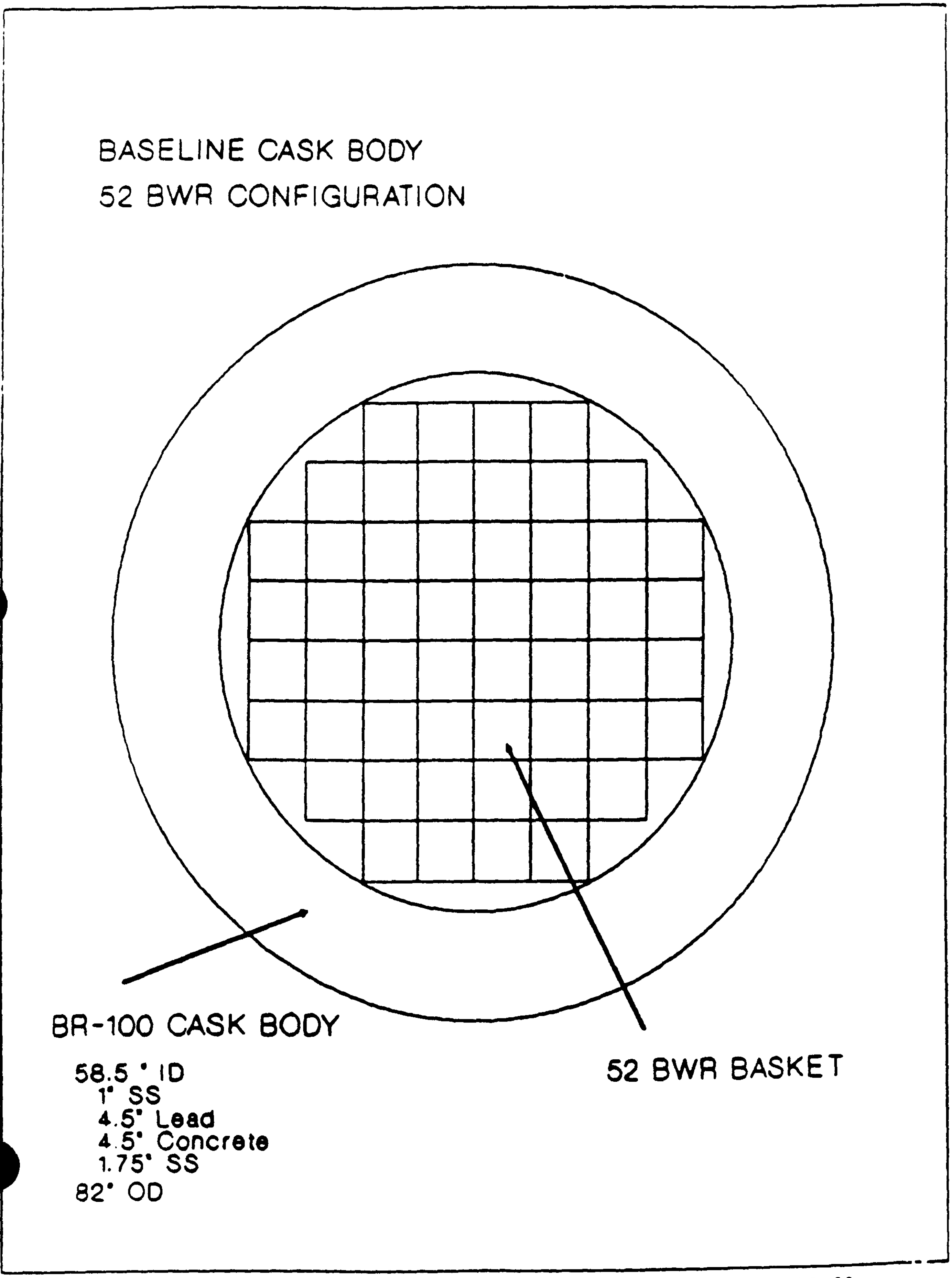

BASELINE CASK BODY
21 PWR CONFIGURATION
BR-100 CASK BODY
58.5" ID
1" SS
4.5" Lead
4.5" Concrete
1.75" SS
82" OD
21 PWR BASKET

FIGURE 3


BASELINE CASK BODY
52 BWR CONFIGURATION
BR-100 CASK BODY
58.5 " ID
1" SS
4.5" Lead
4.5" Concrete
1.75" SS
82" OD
52 BWR BASKET

4.0 DISCUSSION

This investigation provides supplementary information concerning cask capacities for conceptual redesigns of the BR-100 cask and for downloading of the baseline cask to meet overall weight constraints, thermal limits, and environmental dose rate criteria. The fuel used for this study was thermally and radiologically more active than fuel used in previous trade-off studies for the BR-100 cask. This is due to the higher burnup and/or shorter cooling times for the fuel specified for this study as opposed to those for the original contract Statement-of-Work (SOW).

The SOW for this study specifies that cask redesigns are to be variations of the proposed preliminary design, but should be conceptual in nature and not extend to materials substitutions. Also, this study was not to address detailed fuel support basket design for the down-rating scenarios.

4.1 Radiation Analysis

4.1.1 Downloading Studies

The baseline cask described in Section 3 was analyzed with ANISN using source terms from ORIGEN2 for the new study fuels shown in Tables 4 and 5. Discrete geometries were input for each downloading case and the dose rates were calculated to satisfy 10CFR71 criteria at two meters from the personnel barrier. The calculations incorporated the same conservatisms in modelling and power distributions as used throughout the BR-100 preliminary design effort.

Downloading of the baseline cask was done in conjunction with inclusion of aluminum inserts on the outer circumference of the fuel basket. The combination of downloading and aluminum shielding reduced the 2-meter dose rates to within the design limit (10 mrem/hr).

Figure 4 shows the basket configurations for both PWR and BWR fuel, and Figure 5 shows how these baskets were modelled in conjunction with the aluminum shields in the baseline cask.

The quantity of new study BWR fuel assemblies that may be loaded in the baseline cask, and the concurrent amount of aluminum shielding required in the basket to maintain design dose rate limits and cask weight limit are given in Table 4. All the new study fuel types with the exception of the three most active ones may be shipped without downloading.

Only one of the new study PWR fuel types can be shipped at full capacity in the standard baseline basket. All other fuel types require downloading in conjunction with supplemental internal aluminum shielding. Table 5 shows the permissible quantity per cask, the shielding required, and the corresponding dose rate prediction.

Table 4

Downloaded BR-100 Cask Capacities for BWR Fuel

Fuel Design*	Quantity (FA's)	Internal Al Shield Thickness (in.)	Maximum 2-Meter Dose Rate (mrem/hr)
30/5/3%	52	None	9.4
40/5/3%	40***	2.875	8.4
40/10/3%	52	None	9.2
40/15/3%	52	None	<9.2
50/5/4%	40***	2.875	10.5**
50/10/4%	45	1.125	9.7
50/15/4%	52	None	9.0

* (GWd/mtU burnup/years cooling/weight percent U enrichment)

** More discrete analysis expected to reduce dose rate below 10 mrem/hr

*** Max shielding capacity, but thermally limited to 37 (40/5/3%) and 30 (50/5/4%)

Table 5

Downloaded BR-100 Cask Capacities for PWR Fuel

Fuel Design*	Quantity (FA's)	Internal Al Shield Thickness (in.)	Maximum 2-Meter Dose Rate (mrem/hr)
35/5/3%	16	2.25	10.4**
45/5/4%	14	3.375	9.1
45/10/4%	18	0.875	10.2**
45/15/4%	21	None	7.9
55/5/4%	12	6.75	5.9
55/10/4%	14	3.375	9.3
55/15/4%	16	2.25	9.5

* (GWd/mtU burnup/years cooling/weight percent U enrichment)

** More discrete analysis expected to reduce dose rate below 10 mrem/hr

4.1.2 Cask Redesign Shielding Studies

The trade-off studies for the new study fuel scenarios were limited to evaluations of three cask designs: the base BR-100 and two iterations with heavier walls (greater intrinsic shielding). The three cask/basket designs are depicted in Figure 6.

The casks were designed for common-use application (PWR and BWR payloads in separate baskets). For this reason, the design approach was the same as for the baseline cask design: first determine the optimum PWR payload, and then determine the maximum BWR payload allowed by the PWR optimization. Weight and shielding calculations show that, with optimal usage, the maximum PWR payload capacities are: 21 for 45/15 and 35/10 design basis fuel, 18 for 55/15, 45/10, 55/10 and 35/5 fuel, and 16 for 45/5 and 55/5 fuel.

The purpose of the new cask designs was to make these maximum payloads possible. Three cask designs were selected based on the PWR optimization. The BWR baskets were selected based on the resultant internal diameter of the cask.

Cask 1 is the BR-100 baseline design described in Section

3. Casks 2 and 3 have a design similar to the baseline with the same material compositions, but with different thicknesses of gamma and neutron shielding. The new cask diameters and shielding thicknesses are shown on Figure 6 and in Table 6.

Each cask design was analyzed with ANISN for the new design base fuel to determine the quantity and type of fuel suited for the cask. Gamma and neutron source terms used in the ANISN runs were obtained from ORIGEN2 calculations. Tables 7, 8 and 9 show the results of the analyses for the three casks. Dose rates indicated as "less than" (<) were inferred by comparison rather than being calculated (for example, 30/10/3% fuel has radioactively decayed more than 30/5/3% fuel, therefore, the dose rate of the former must be less than the latter).

Conservative models and power distribution factors were used in all calculations. Hence, the dose rates should be on the conservative side in all cases.

Table 6

Redesigned Cask Parameters

	Capacity PWR/BWR (FA's)	Cask I.D. (in)	Radial CL Thickness Lead/Conc (in)	Radial Top & Bot. Lead/Conc. (in)	Shield Plug Thickness Lead (in)	Cask Bottom Thickness Lead (in)
Cask 1	21/52	58.5	4.5/4.5	5.0/4.0	3.5	3.0
Cask 2	18/45	56.75	4.75/5.5	5.25/5.0	3.5	3.0
Cask 3	16/40	54.0	5.0/6.25	6.0/5.25	4.0	3.5

Table 7

Cask 1 (BR-100)

Fuel Type	Fuel Design*	Decay Heat Average (Watts/Assy)	Maximum 2-Meter Dose Rate (mrem/hr)	Quantity/Cask (Fuel Assy's)
BWR	30/5/3%	279	9.4	52
	30/10/3%	178	<9.4	52
	40/10/3%	256	9.2	52
	40/15/3%	219	<9.2	52
	50/15/4%	278	9.0	52
PWR	35/10/3%	574	9.6	21
	45/15/4%	637	7.9	21

*(GWd/mtU/years cooling/weight percent U enrichment)

Table 8

Cask 2 (18 PWR/45 BWR Capacity)

Fuel Type	Fuel Design*	Decay Heat Average (Watts/Assy)	Maximum 2-Meter Dose Rate (mrem/hr)	Quantity/Cask (Fuel Assy's)
BWR	40/5/3%	399	8.2	45
	50/5/4%	497	10.1	45
	50/10/4%	326	<10.1	45
PWR	35/5/3%	903	8.9	18
	45/10/4%	748	5.4	18
	55/10/4%	991	10.2	16**
	55/15/4%	837	7.5	18

* (GWd/mtU/years cooling/weight percent U enrichment)

** Thermally limited. Dose rate corresponds to 18 FA loading.

Table 9

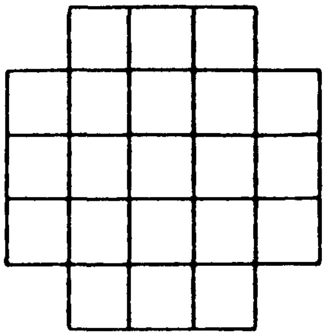
Cask 3 (16 PWR/40 BWR Capacity)

Fuel Type	Fuel Design*	Decay Heat Average (Watts/Assy)	Maximum 2-Meter Dose Rate (mrem/hr)	Quantity/Cask (Fuel Assy's)
PWR	45/5/4%	1147	5.7	16
	55/5/4%	1507	9.5	12**

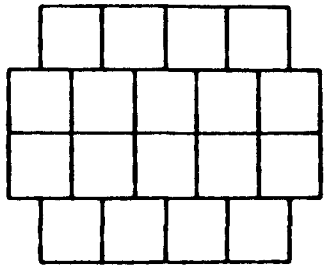
* (GWd/mtU/years cooling/weight percent U enrichment)
** Thermally limited. Dose rate corresponds to 16 FA loading.

FIGURE 4

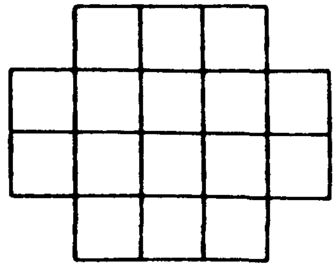
PWR BASKETS



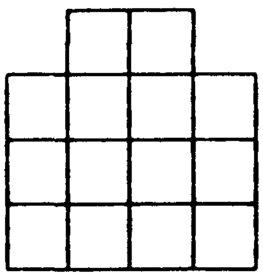
21 PWR BASKET
basket OD = 66.5 inch



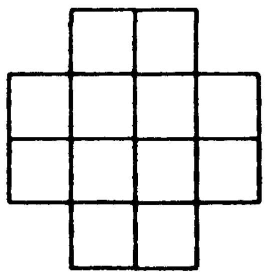
18 PWR BASKET
basket OD = 66.75 inch



16 PWR BASKET
basket OD = 64 inch

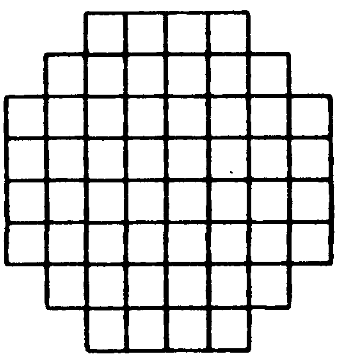


14 PWR BASKET
basket OD = 51.75 inch

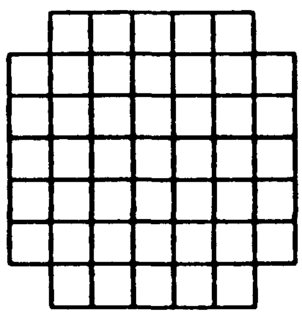


12 PWR BASKET
basket OD = 45 inch

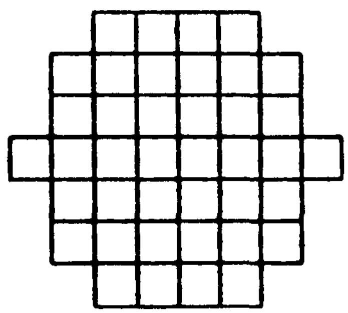
BWR BASKETS



52 BWR BASKET
basket OD = 66.5 inch



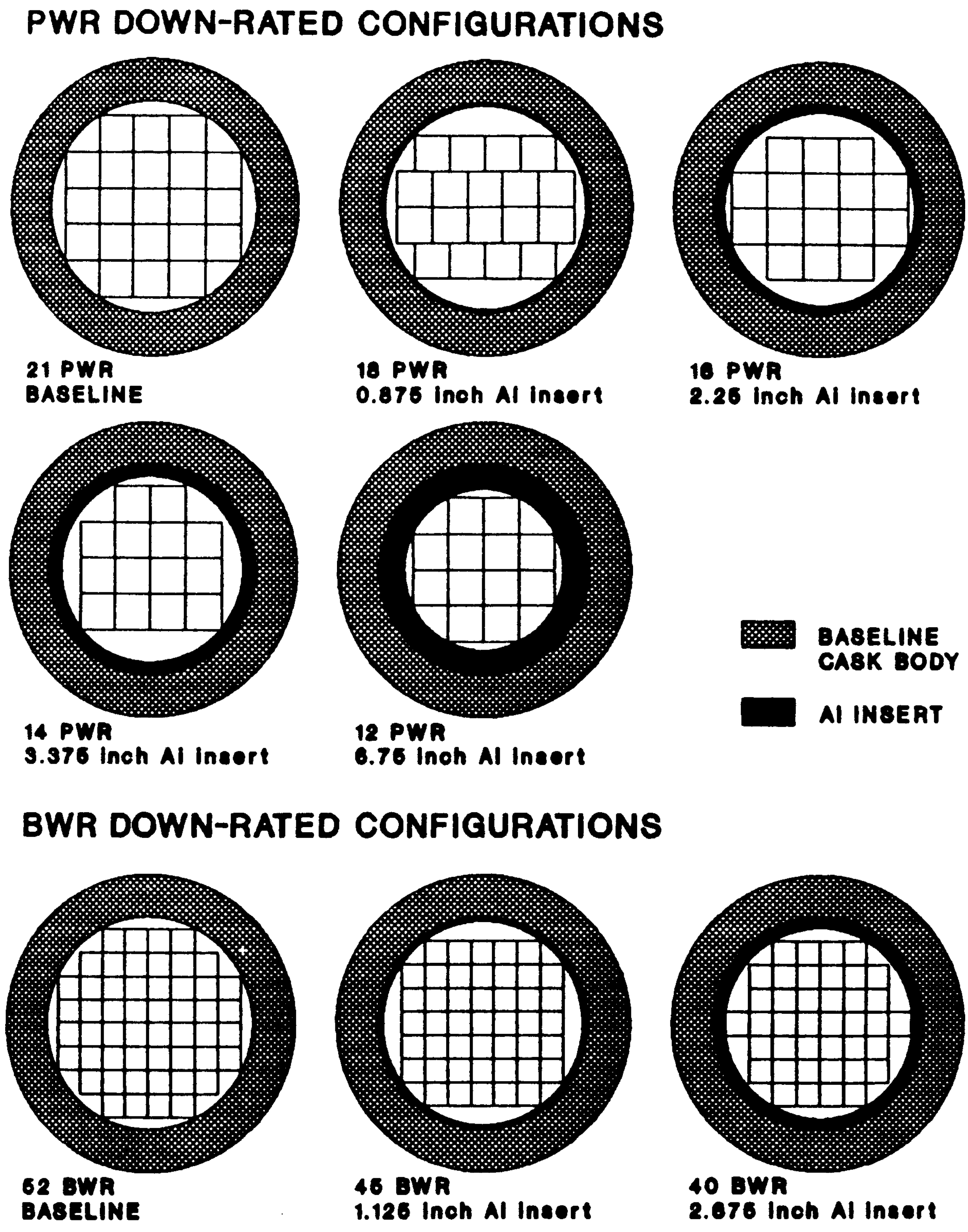
45 BWR BASKET
basket OD = 56.25 inch



40 BWR BASKET
basket OD = 52.75 inch

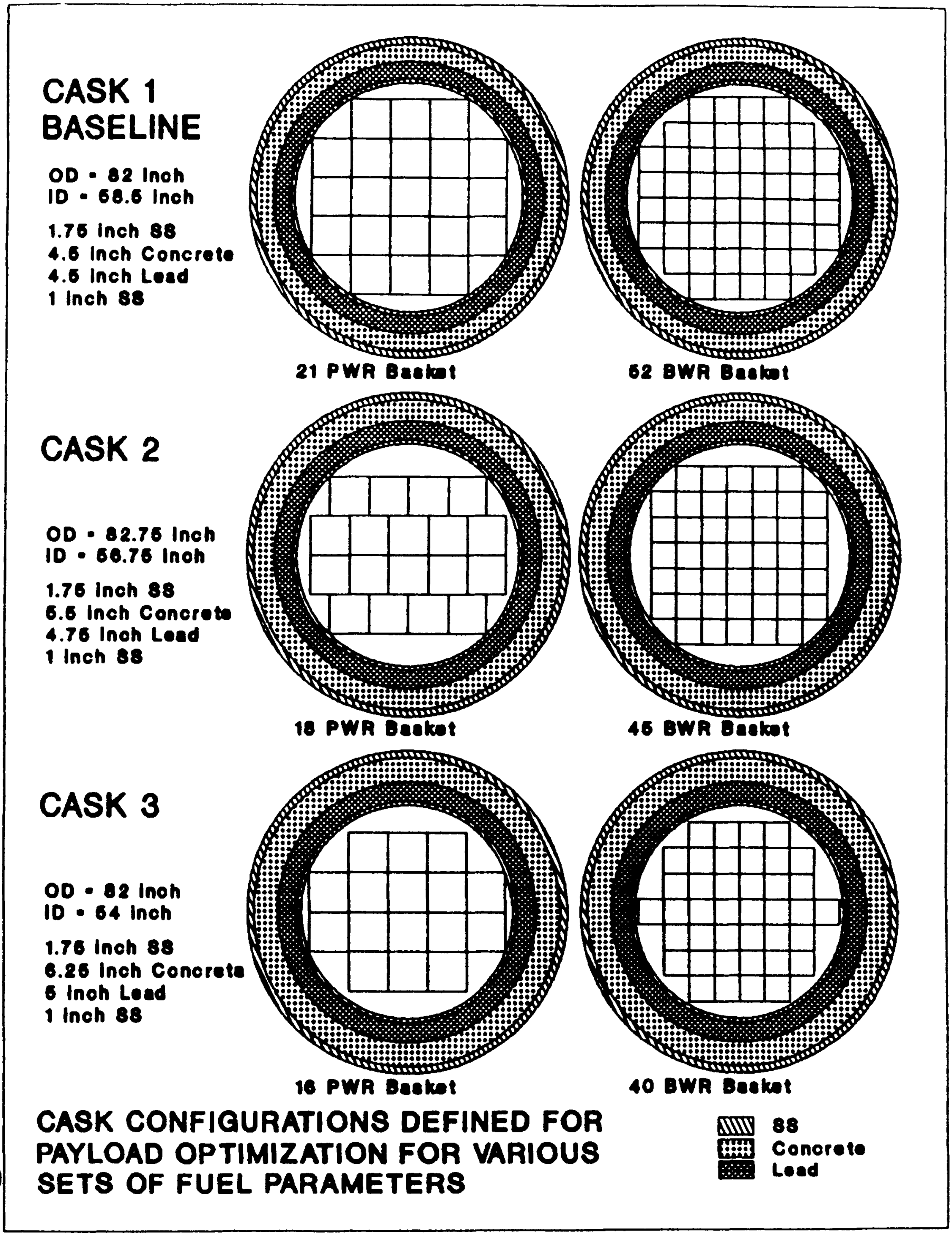
BASKET CONFIGURATIONS FOR TRADE-OFF STUDIES

FIGURE 5



CONFIGURATIONS FOR DOWN-RATING STUDIES

FIGURE 6



CASK CONFIGURATIONS DEFINED FOR PAYLOAD OPTIMIZATION FOR VARIOUS SETS OF FUEL PARAMETERS

4.2 Thermal Analysis

The goal of the thermal design of the BR-100 shipping cask is to maintain the cask wall components, the basket, and the spent fuel cladding temperatures within acceptable limits. These acceptable limits are attained by restricting the cask loading to conditions that do not exceed the following thermal guidelines:

1) The maximum temperature in the Robatel borated concrete-copper layer will remain below 250°F (121°C), based on project-chosen material limitations.
2) The maximum temperature in the aluminum basket will remain below 350°F (177°C), based on desired ASME structural properties.
3) The maximum spent fuel cladding temperature will remain below 680°F (360°C), to provide significant margin against cladding failure.

These limits were used to establish various loading limits for several different PWR basket designs. The different basket designs evaluated for this trade-off study are shown in Figure 4 (page 24):

1) 21 PWR basket design (Baseline cask),
2) 18 PWR basket design (Cask 2) and

3) 16 PWR basket design (Cask 3).

The radial locations of the various cask wall components for the different basket designs are shown in Table 10. The various fuel conditions to be evaluated for each basket design are shown in Table 11. These represent limiting thermal conditions as compared to BWR design base fuel. Therefore, it was not necessary to run discrete thermal analyses for the BWR fuel. Table 12 contains the thermal loads per assembly that are associated with each basket configuration's fuel conditions. The peak burnups were obtained by multiplying the assembly average value by the PWR axial peaking factor of 1.13. The peaking factor was obtained from a FLAME computer run and is documented in B&W Calculational File No. 32-1172659. The peak burnups were used to generate the peak thermal loads at the cask axial midplane for these trade-off studies.

The temperature distributions at the various cask wall component interfaces and former outer surface are shown as a function of the peak assembly linear heat rate in the following figures:

1) 21 PWR (Baseline cask) - Figures 7 and 8,
2) 18 PWR (Cask 2) - Figures 9 and 10, and
3) 16 PWR (Cask 3) - Figures 11 and 12.

These figures assume that a full load of the various assemblies is inserted into the various BR-100 basket designs. The figures indicate the following:

1) The maximum concrete temperature of 250°F is reached at a peak linear heat rate of 86 w/ft for the Baseline cask, 103 w/ft for Cask 2, and 115 w/ft for Cask 3.
2) The total thermal loads for the Baseline cask, Cask 2, and Cask 3 were 18.2 kw, 18.5 kw, and 18.3 kw. These limits are based on the concrete temperature limit of 250°F. For the Baseline cask and Cask 3 the concrete temperature represents the most limiting thermal design condition.

The temperature distributions through the cask wall, from the cask exterior surface to the cask inner radius, and across the cask inner radius-to-former outer radius gap are shown in Figures 13 (Baseline cask), 14 (Cask 2), and 15 (Cask 3). These temperature distributions are also shown in Table 13.

The full load 55 GWd/mtU/10 year-cooled/4.0 wt % (55/10/4.0%) condition in Cask 2 and full load 55/5/4.0% condition in Cask 3 are not shown in figures 3 through 10 since the full load conditions exceeded one of the thermal design limits. These two fuel loading conditions were evaluated using a partial loading pattern. The full load of 18 PWR 55/10/4.0% fuel assemblies in Cask 2 exceeded the aluminum basket temperature limit of 350°F. The loading pattern for the sixteen 55/10/4.0% fuel assemblies in Cask 2 is shown in Figure 16. This loading pattern represents a cask thermal load of 15.9 kw. The full load of sixteen PWR 55/5/4.0% fuel assemblies in Cask 3 exceeded the concrete temperature limit of 250°F. Two different loading patterns, as shown in Figure 17, were evaluated for inserting twelve 55/10/4.0% assemblies in the Cask 3. Both of these loading patterns represent total cask thermal loads of 18.1 kw. The temperature distributions through the cask wall, from the external surface to the former outer radius, are contained in Table 14. The temperature distribution for the Cask 3 partial load condition is used for both of the twelve 55/5/4.0% loading patterns, shown in Figure 17.

Table 15 contains the peak aluminum basket and peak cladding temperatures for the various Baseline cask, Cask 2, and Cask 3 loading conditions. The peak cladding temperatures, predicted using the Wooten-Epstein relationship, for all of the loading conditions were well below the cladding thermal design limit of 680°F. Baseline cask and Cask 2 peak basket and fuel cladding temperatures are shown as a function of the peak linear heat rate in Figures 18 and 19. Figure 19 shows that the eighteen 55/10/4.0% condition, peak linear heat rate of 99.78 w/ft exceeds the project's aluminum basket temperature limit. The limit on the 18 PWR basket due to the aluminum basket restricts the total cask load to approximately 17.3 kw. Some additional fuel could be placed in the two center locations of the 18 PWR basket, but the basket must not exceed the thermal load characteristics of the 45/10/4.0% fuel.

No discrete thermal analyses were performed for the BWR Design Base Fuel. BWR thermally acceptable loadings for the cask were derived by dividing the limiting BWR cask heat load by the assembly decay heat at average burnup. The limiting BWR cask heat load is 14.6 KW as shown in Table 3-1, Section II-3, of the PDR. The assembly decay heat at average burnup is shown on Table 16 of this report.

The full payload of BWR fuel assemblies is within the limiting cask heat load for all the BWR design base fuels except for the 40/5/3%, 50/5/4%, and the 50/10/4% fuel. For these fuels, cask downloading is required to meet the thermal limits. The 50/10/4% fuel limitation of 45 FA in the baseline cask coincides with a shielding limitation for the same number of loaded assemblies.

Table 10

Radial Locations for Various Basket Designs

Description of Interface	Location, inch (m) Baseline cask	Cask 2	Cask 3
Basket maximum capacity	21 PWR	18 PWR	16 PWR
Cask outer surface	41.00(1.041)	41.375(1.051)	41.00(1.041)
Inner radius of outer stainless steel shell	39.25(0.997)	39.625(1.006)	39.25(0.997)
Inner radius of concrete only	38.85(0.987)	39.225(0.996)	38.85(0.987)
Inner radius of concrete-copper fin region	34.75(0.883)	34.125(0.867)	33.00(0.838)
Inner radius of lead gamma shield	30.25(0.768)	29.375(0.746)	28.00(0.711)
Inner radius of the cask wall	29.25(0.743)	28.375(0.721)	27.00(0.686)
Outer radius of the former	29.20(0.742)	28.325(0.719)	26.95(0.685)

Table 11

Fuel Loading Conditions Evaluated

Cask Design	Assembly Avg. Burnup (GWd/mtU)	Cooling Time (years)	Enrichment wt%
Baseline(21 PWR)	35	10	3.0
	45	15	4.0
Cask 2(18 PWR)	35	5	3.0
	45	10	4.0
	55	10	4.0
	55	15	4.0
Cask 3(16 PWR)	45	5	4.0
	55	5	4.0

Table 12

Thermal Loads for Various Basket Designs and Fuel Conditions

Cask Design	Average Burnup Conditions per Assembly	Decay Heat at Average Burnup, (Watts/Assembly)
Baseline(21 PWR)	35/10/3%	574.3
	45/15/4%	637.0
Cask 2(18 PWR)	35/5/3%	902.6
	45/10/4%	748.2
	55/10/4%	990.9
	55/15/4%	837.5
Cask 3(16 PWR)	45/5/4%	1147.6
	55/5/4%	1507.1

Cask Design	Peak Burnup Conditions per Assembly	Decay Heat at Peak Burnup, (Watts/Assembly)	Assembly Peak Linear Heat Rate (Watts/ft)
Baseline	39.55/10/3%	681.7	56.81
	50.85/15/4%	749.1	62.43
Cask 2	39.55/5/3%	1065.8	88.82
	50.85/10/4%	884.0	73.67
	62.15/10/4%	1197.4	99.78
	62.15/15/4%	1008.1	84.01
Cask 3	50.85/5/4%	1350.4	112.53
	62.15/5/4%	1802.8	150.23

Table 13

Baseline, Cask 2, and Cask 3 Full Load Temperature Summary

*Loc, inch (m)	- Baseline Cask(21 PWR) --	
	35/10/3%	45/15/4%
41.00(1.041)	194.9(90.5)	201.4(94.1)
39.25(0.997)	197.9(92.2)	204.7(96.0)
38.85(0.987)	203.7(95.4)	211.7(99.9)
34.75(0.883)	210.1(98.9)	218.1(103.4)
30.25(0.768)	214.6(101.4)	223.0(106.1)
29.25(0.743)	216.9(102.7)	225.5(107.5)
29.20(0.742)	227.9(108.8)	237.5(114.2)

*Loc, inch (m)	--------- Cask 2(18 PWR) ---------		
	45/10/4%	55/15/4%	35/5/3%
41.375(1.051)	201.6(94.2)	211.5(99.7)	216.0(102.2)
39.625(1.006)	204.9(96.1)	215.3(101.8)	220.0(104.4)
39.225(0.996)	212.0(100.0)	223.3(106.3)	228.5(109.1)
34.125(0.867)	217.0(102.7)	229.1(109.5)	234.5(112.5)
29.375(0.746)	222.5(105.9)	235.4(113.0)	241.2(116.2)
28.375(0.721)	225.2(107.3)	238.4(114.7)	244.4(118.0)
28.325(0.719)	237.6(114.2)	252.4(122.4)	259.1(126.2)

*Loc, inch (m)	- Cask 3(16 PWR) -
	45/5/4%
41.00(1.041)	227.0(108.3)
39.25(0.997)	231.5(110.8)
38.85(0.987)	241.1(116.2)
33.00(0.838)	246.9(119.4)
28.00(0.711)	254.9(123.8)
27.00(0.686)	258.6(125.9)
26.95(0.685)	275.8(135.5)

*See Table 11 for definition of locations.

Table 14

Cask 2 and 3 Partial Load Temperature Summary

- Cask 2(18 PWR) -

*Loc inch (m)	Load of sixteen 55/10/4% assemblies
41.375(1.051)	215.9(102.2)
39.625(1.006)	219.9(104.4)
39.225(0.996)	228.3(109.1)
34.125(0.867)	234.4(112.4)
29.375(0.746)	241.1(116.2)
28.375(0.721)	244.3(117.9)
28.325(0.719)	259.0(126.1)

- Cask 3(16 PWR) -

*Loc, inch (m)	Load of twelve 55/5/4% assemblies
41.00(1.041)	227.1(108.4)
39.25(0.997)	231.6(110.9)
38.85(0.987)	241.2(116.2)
33.00(0.838)	247.0(119.5)
28.00(0.711)	255.1(123.9)
27.00(0.686)	258.8(126.0)
26.95(0.685)	276.0(135.6)

*See Table 11 for definition of locations.

Table 15

Peak Basket and Spent Fuel Cladding Temperatures

Cask Design	Fuel Condition	Peak Temperature, F(C) Basket	Cladding
	--------- Full load conditions	---------	
Baseline(21 PWR)	35/10/3%	274.3(134.6)	364.2(184.6)
	45/15/4%	288.4(142.5)	384.0(195.6)
Cask 2(18 PWR)	45/10/4%	300.1(148.9)	407.8(208.8)
	55/15/4%	323.5(161.9)	440.3(226.8)
	35/5/5%	334.3(167.9)	455.0(235.0)
Cask 3(16 PWR)	45/5/4%	344.1(173.4)	487.7(253.2)
	--------- Mixed load conditions	-------	
Cask 2	16 of 55/10/4%	318.2(159.0)	452.1(233.4)
Cask 3	12 55/5/4%(Patt 1)	328.0(164.4)	509.2(265.1)
	12 55/5/4%(Patt 2)	333.8(167.7)	514.1(267.8)

Table 16

Thermal Loads for Various BWR Design Base Fuel

Cask Design	Average Burnup Conditions per Assembly	Decay Heat at Average Burnup (Watts/Assembly)
Baseline (52 BWR)	30/5/3%	278.62
	30/10/3%	177.67
	40/10/3%	256.17
	40/15/3%	219.04
	50/15/4%	278.44
Cask 2 (45 BWR)	40/5/3%	398.66
	50/5/4%	496.96
	50/10/4%	326.35

Figure 7 - 21 PWR Cask Wall Temperature Distribution

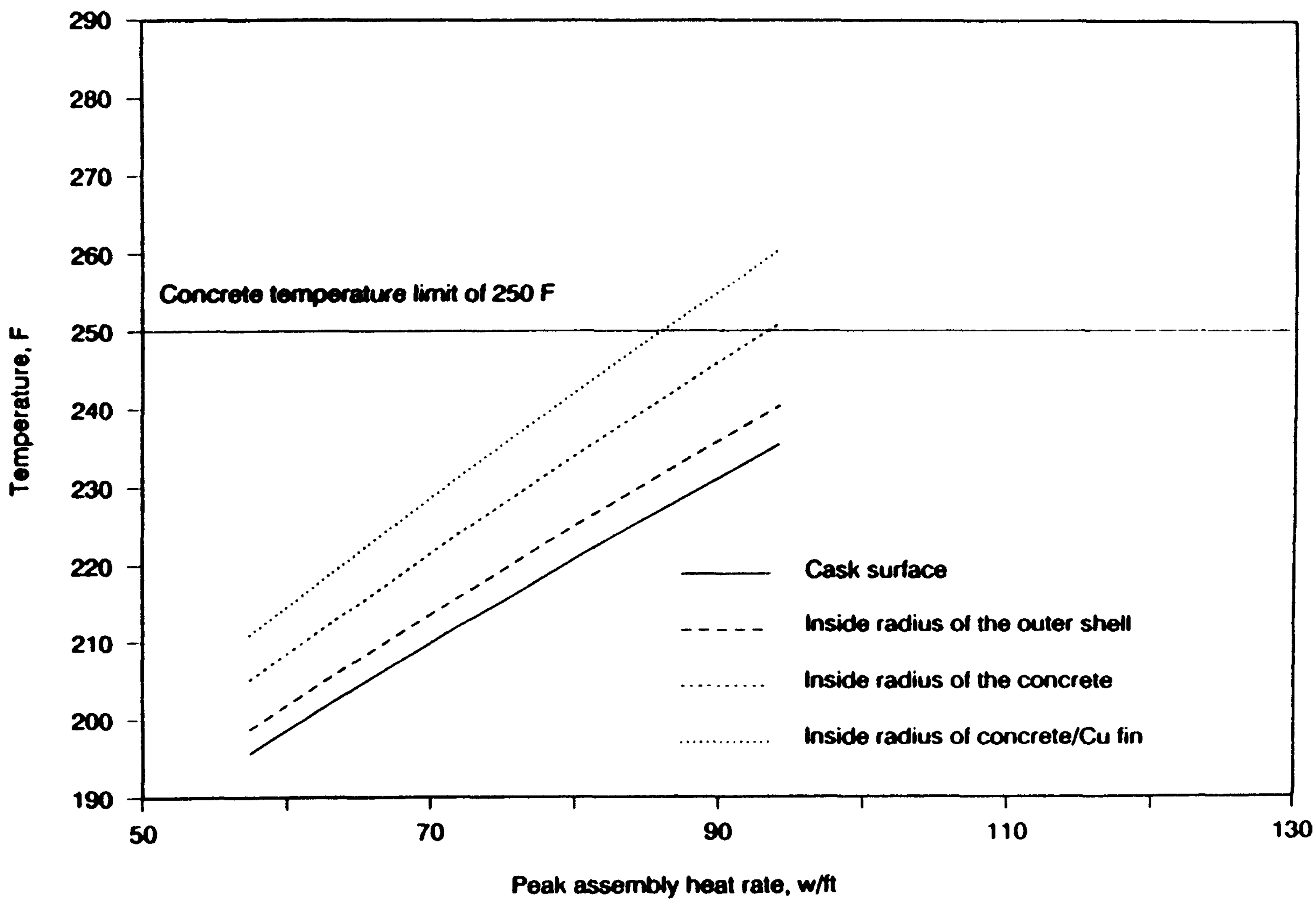


Figure 8 - 21 PWR Cask Wall Temperature Distribution

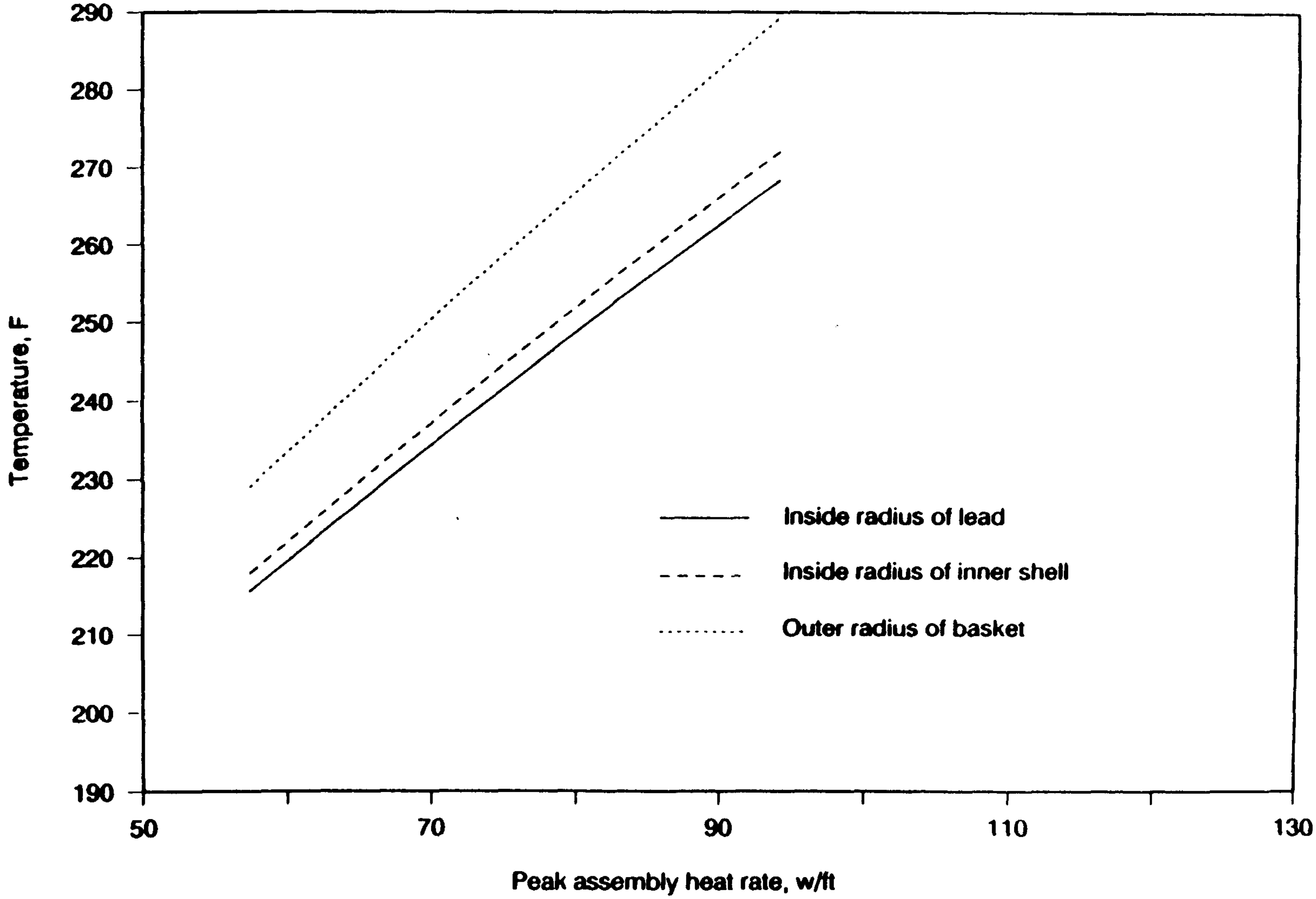


Figure 9 - 18 PWR Cask Wall Temperature Distribution

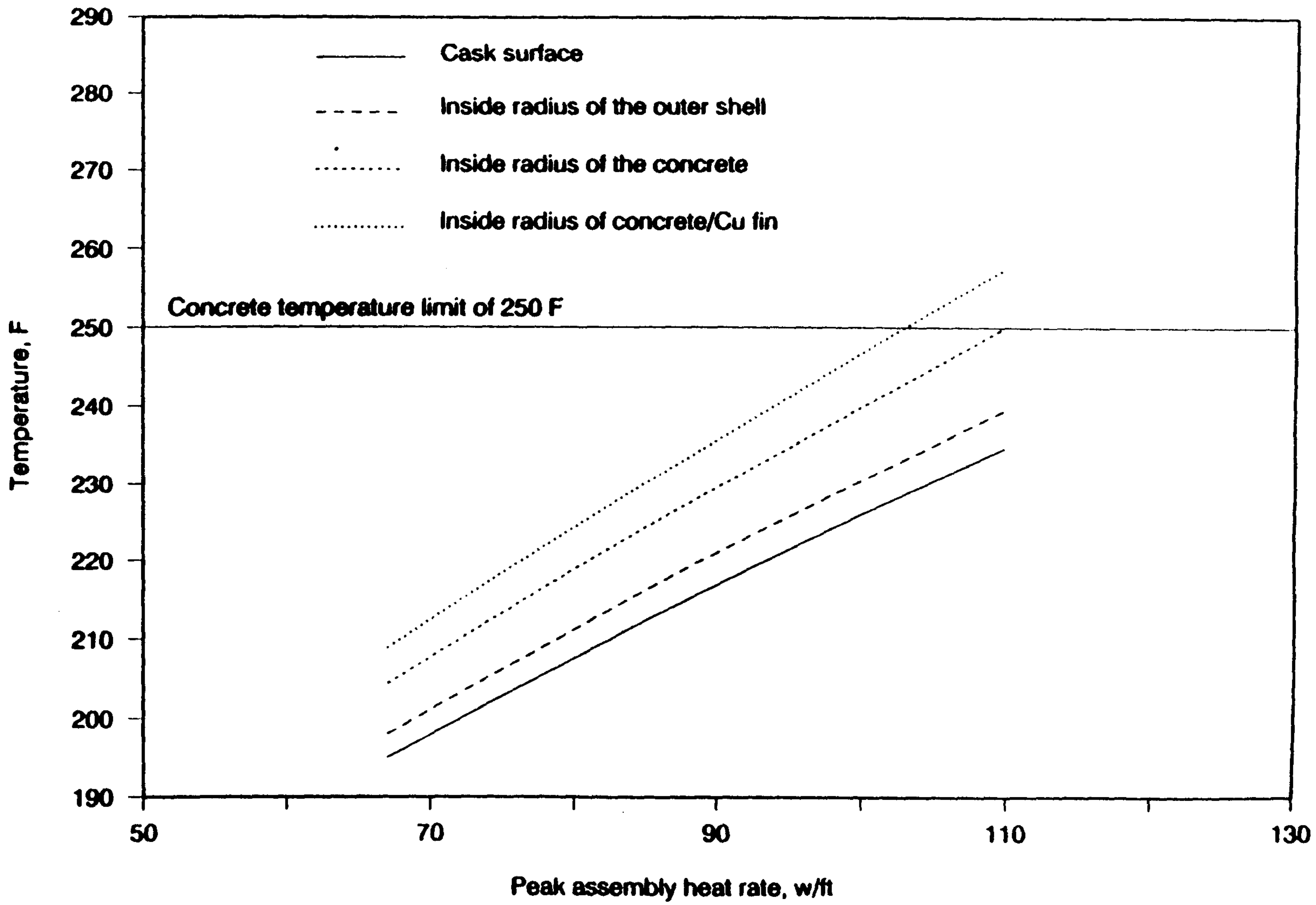


Figure 10 - 18 PWR Cask Wall Temperature Distribution

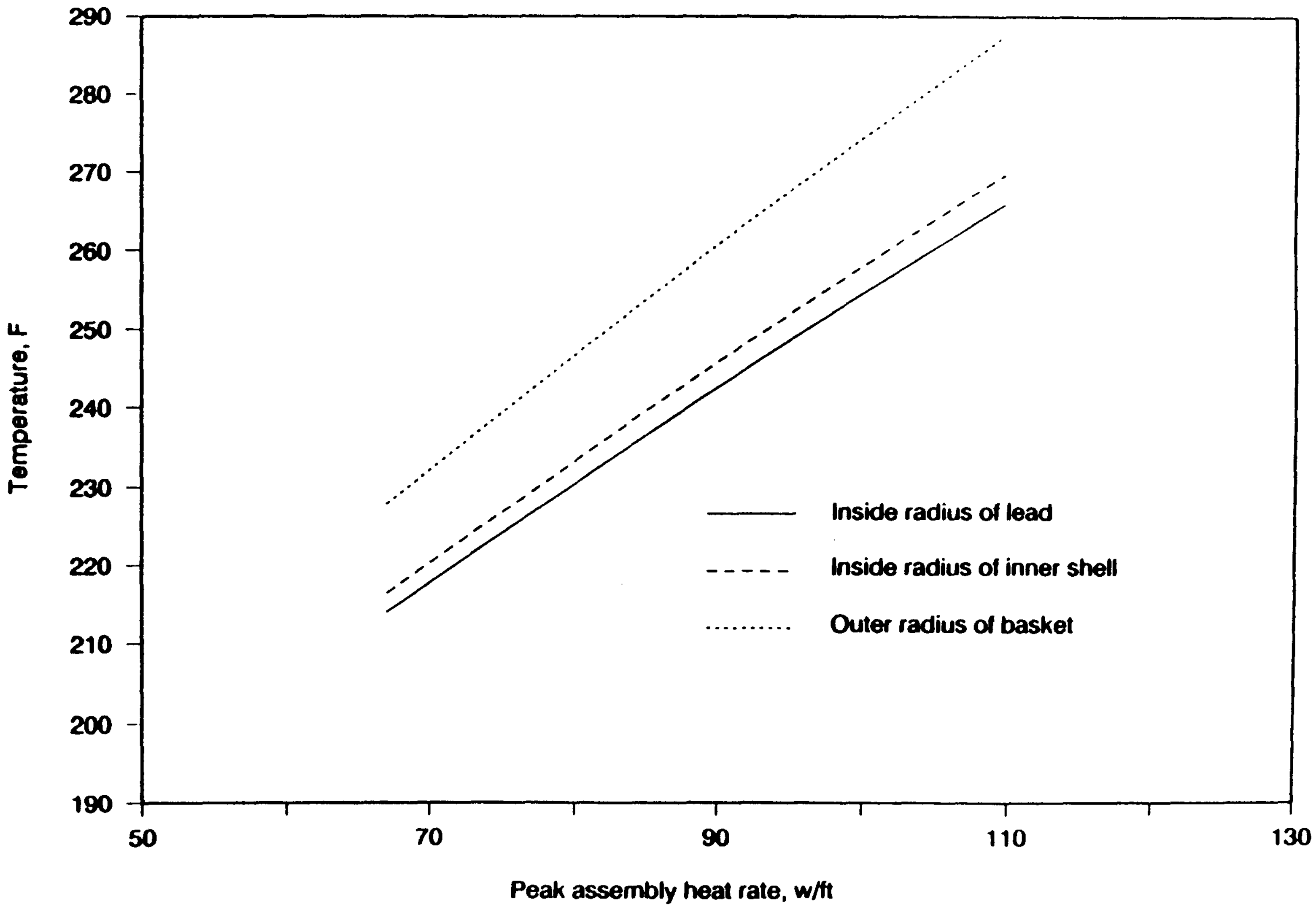


Figure 11 - 16 PWR Cask Wall Temperature Distributions

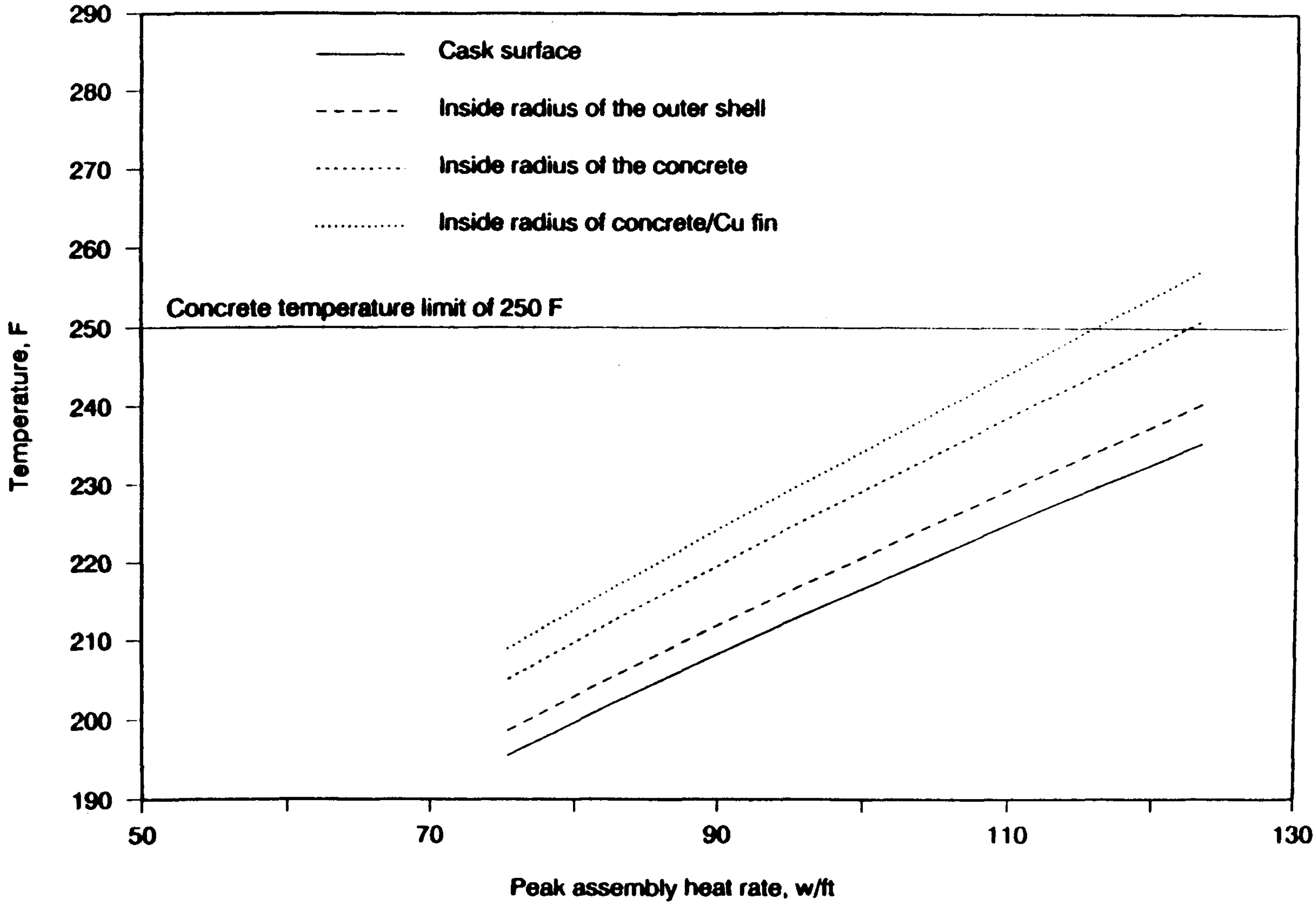


Figure 12 - 16 PWR Cask Wall Temperature Distributions

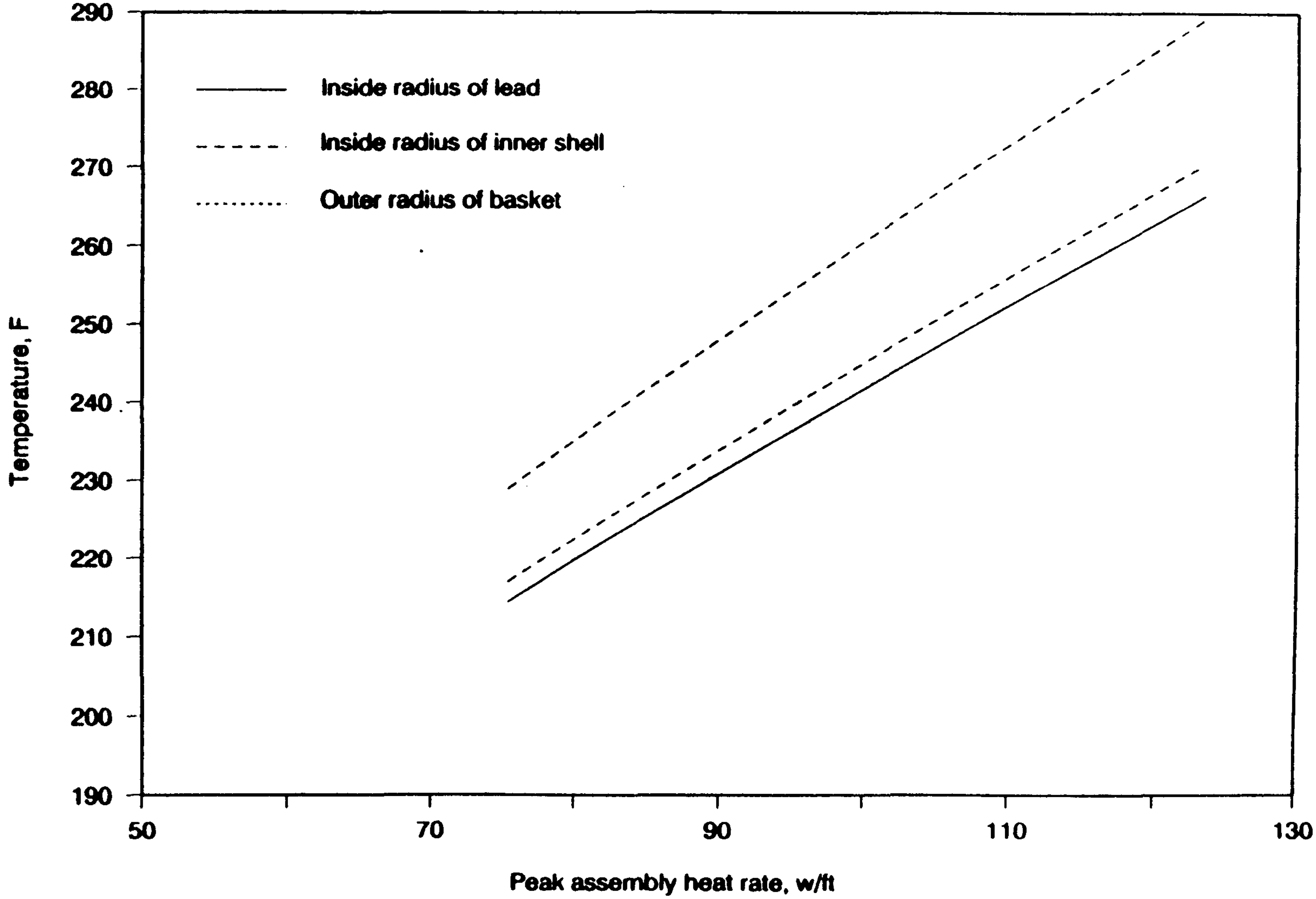


Figure 13 - 21 PWR Cask Temperature Distributions

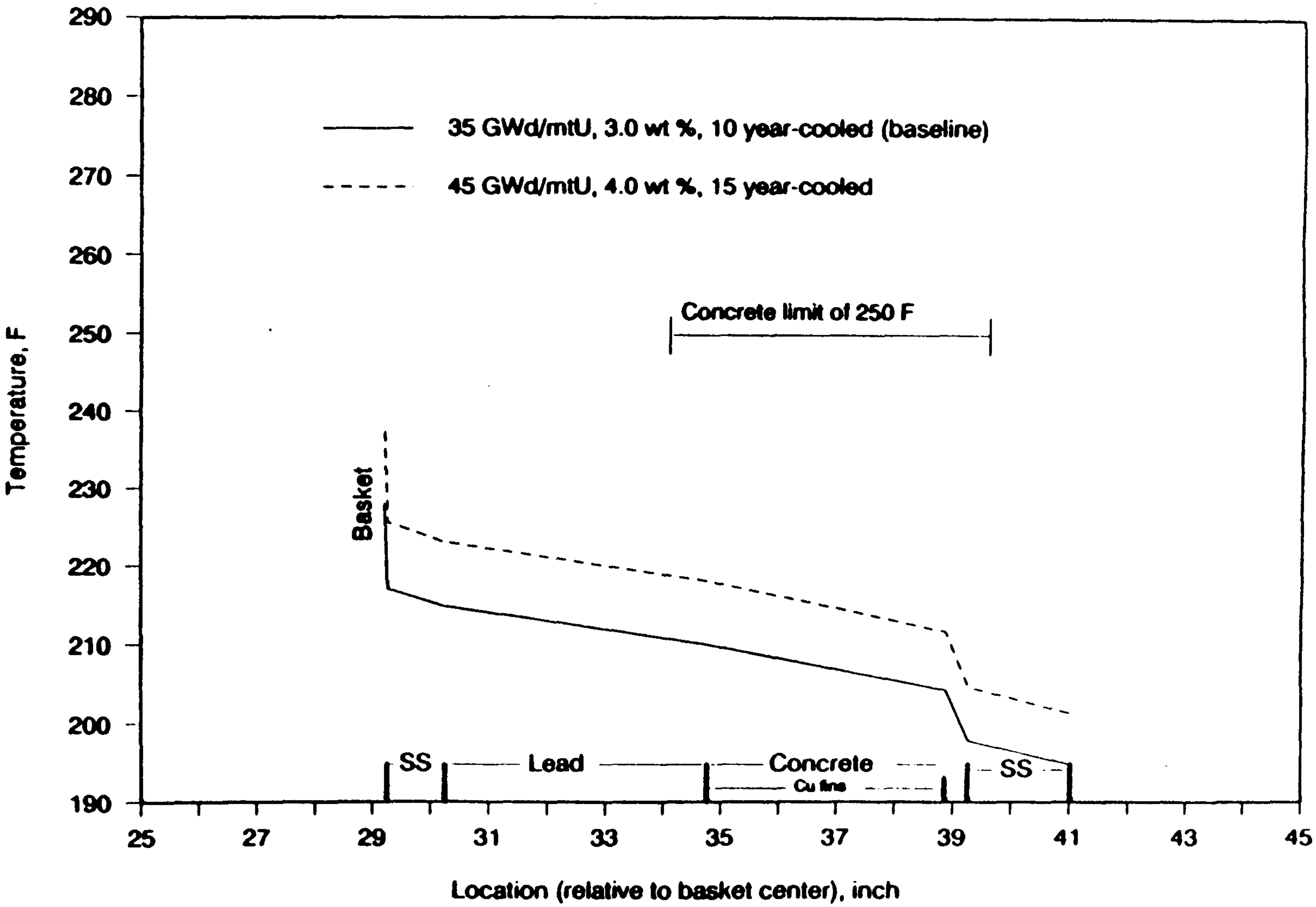


Figure 14 - 18 PWR Cask Temperature Distributions

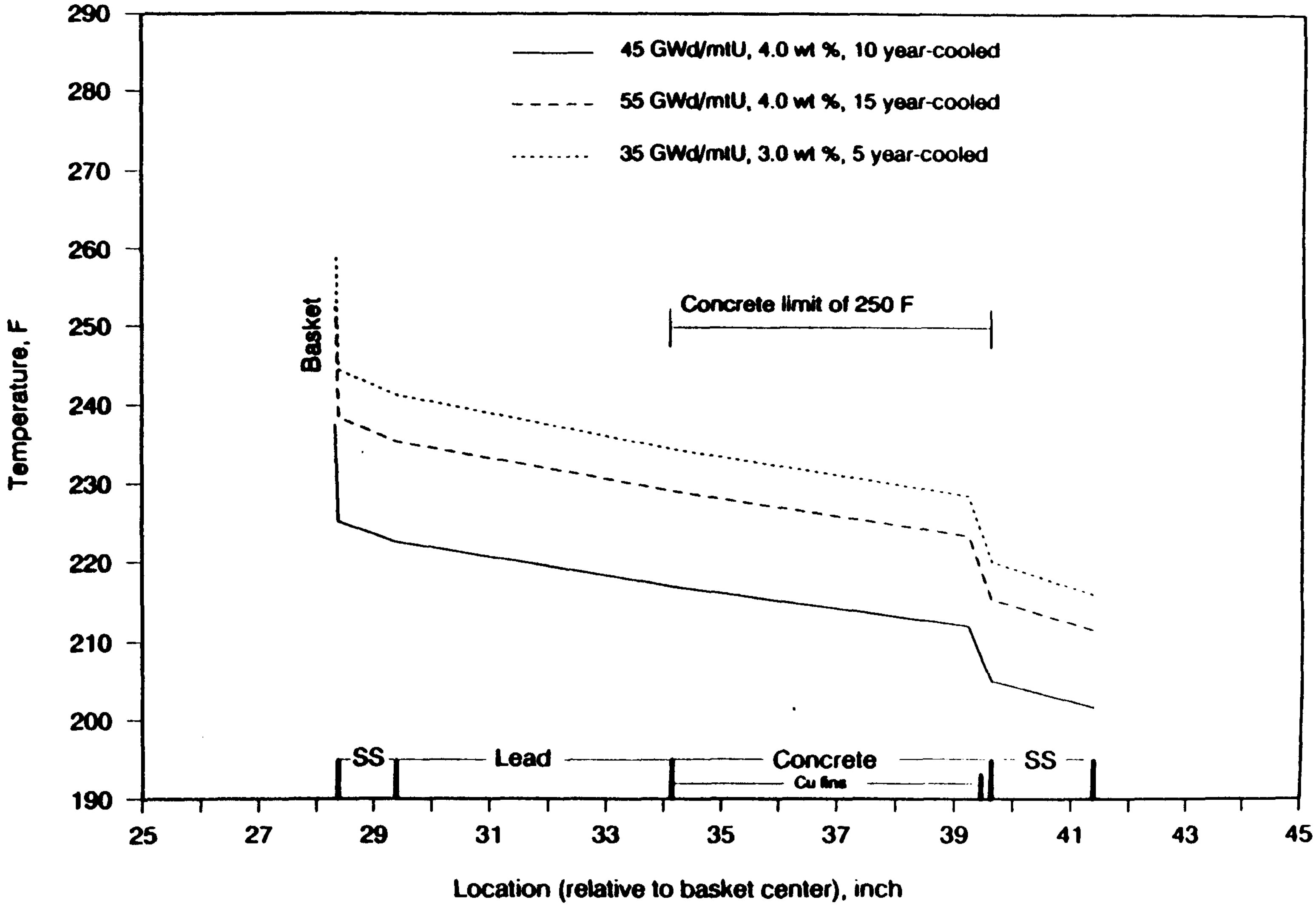


Figure 15 - 16 PWR Cask Temperature Distributions

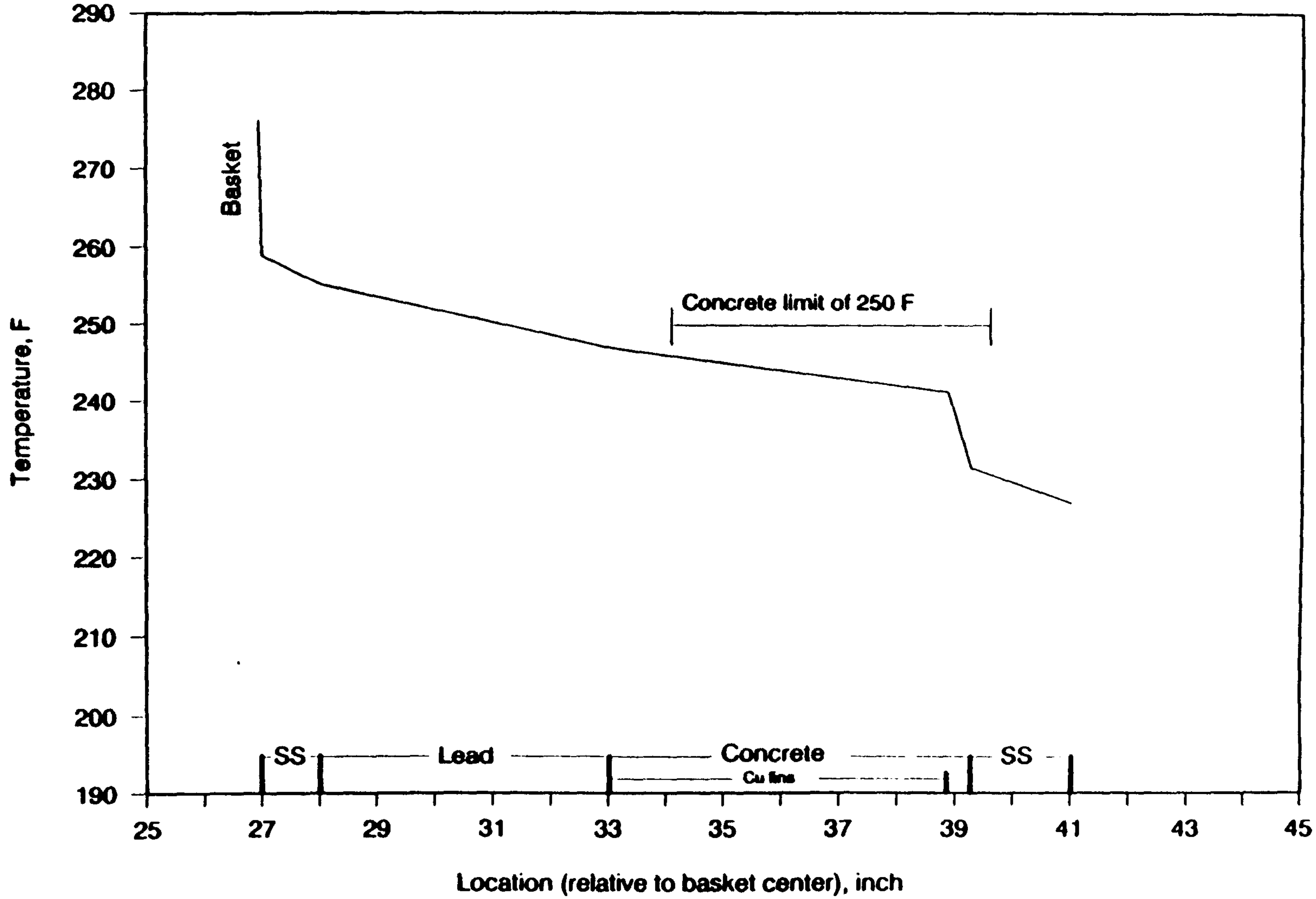


Figure 16 - 18 PWR Partial Loading for 55/10/4%

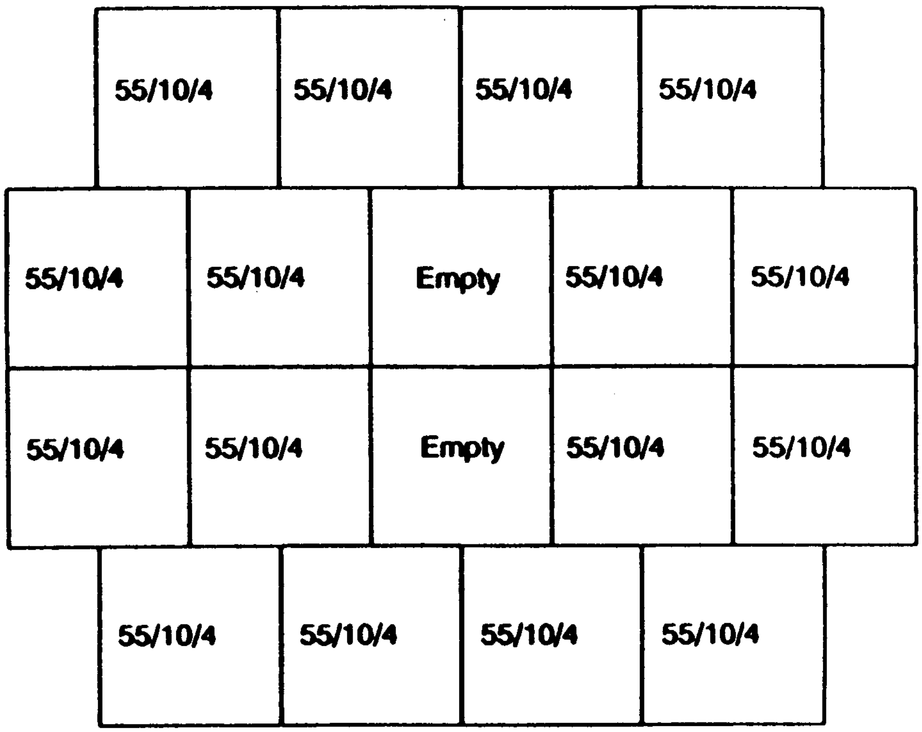


Figure 17 - 16 PWR Partial Loading for 55/5/4%

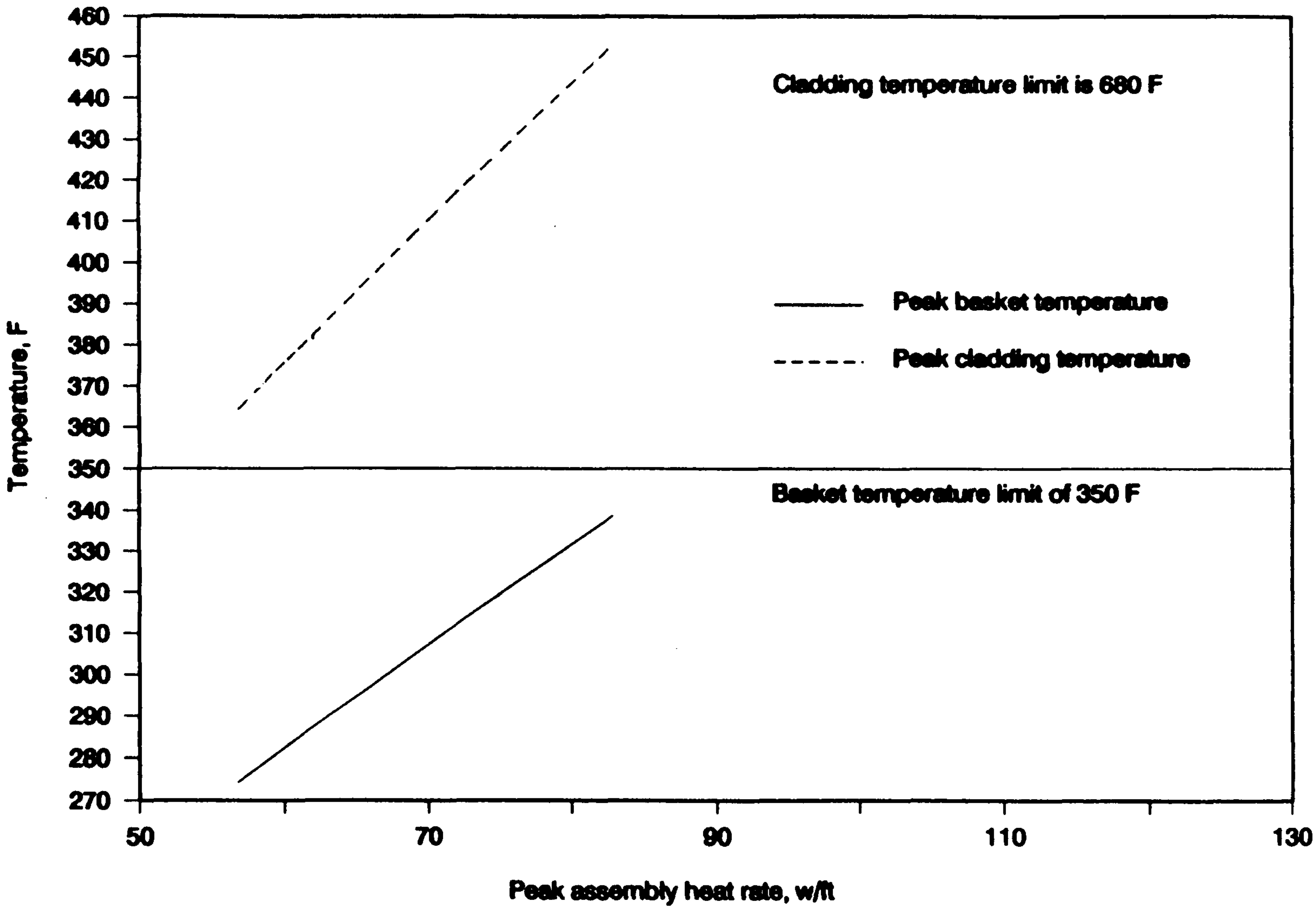
	55/4.0/5	Empty	55/4.0/5	
55/4.0/5	55/4.0/5	Empty	55/4.0/5	55/4.0/5
55/4.0/5	55/4.0/5	Empty	55/4.0/5	55/4.0/5
	55/4.0/5	Empty	55/4.0/5	

Loading pattern 1

	55/4.0/5	55/4.0/5	55/4.0/5	
55/4.0/5	Empty	55/4.0/5	Empty	55/4.0/5
55/4.0/5	Empty	55/4.0/5	Empty	55/4.0/5
	55/4.0/5	55/4.0/5	55/4.0/5	

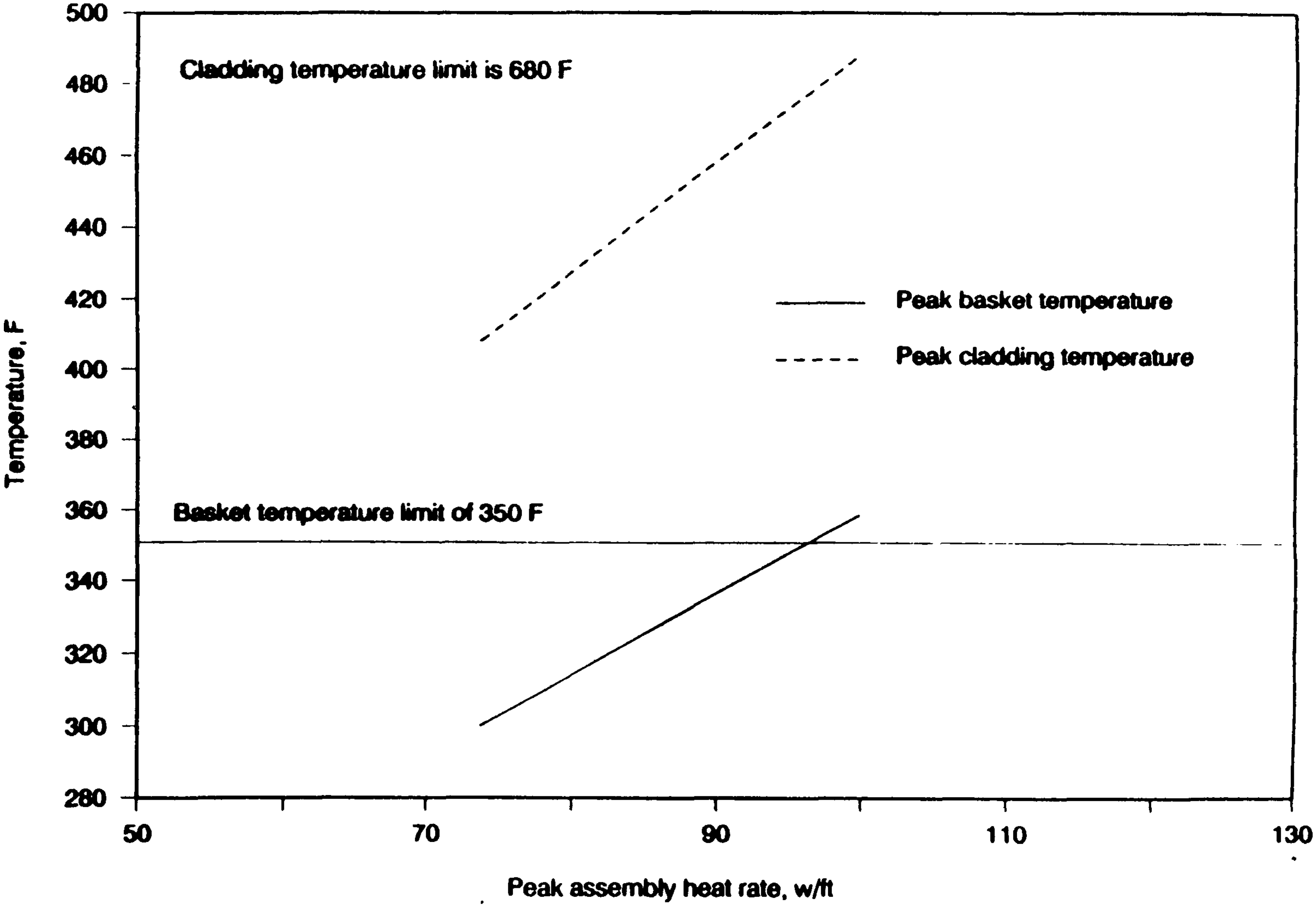
Loading pattern 2

Figure 18 - 21 PWR Peak Basket and Cladding Temperature



51-1177245-02

Figure 19 - 18 PWR Peak Basket and Cladding Temperature



5.0 CONCLUSION

Radiation, thermal, and weight studies were completed to determine the acceptable conditions for transporting various burnups, ages, and enrichments of both PWR and BWR spent fuel assemblies in the baseline BR-100 cask and two similar variations of that cask. The results, shown in Table 1, show that the baseline BR-100 cask with redesigned basket and internal aluminum shielding is capable of transporting the entire spectrum of design base fuel specified by the new SOW with diminishment of payload in some cases.

An optional method of retaining full payload capacity of the baseline cask is to selectively load high burnup fuel centrally in the basket with low burnup fuel placed peripherally. This action will maintain dose rates under the 2-meter limit of 10 mrem/hr and will meet thermal criteria.

The two redesigned casks (cask 2, 18 PWR/45 BWR capacity and cask 3, 16 PWR/40 BWR capacity) are capable of transporting fuel which would require downloading in the baseline cask. They are capable of transporting all the design base PWR and BWR fuels in the optimum quantities shown on Table 1 without reliance on the basket modifications or additional shielding required in the baseline cask. Dose rates and thermal margins meet design criteria.

Cask hook weights meet the design criteria for the three cask designs with all loading conditions and fuel types specified.

This report shows that the baseline BR-100 cask has the capability to accommodate a wide range of high burnup BWR and PWR fuel with minor fuel loading management or with fuel basket redesign only. It also shows that a three cask combination (including the baseline cask) is necessary to transport all the design base fuel scenarios in optimum payload quantities.

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Resume

HUBERT H. DAVIS, JR.

EDUCATION

Bachelor of Science Metallurgical Engineering, 1964
North Carolina State University
Master of Science Metallurgical Engineering, 1966
North Carolina State University
Ph.D. Ceramic Engineering, 1969, North Carolina State University

EXPERIENCE

Dr. Davis is Manager of the Lynchburg Technical Operations and Advanced Materials Laboratory. As Manager of the Advanced Materials Laboratory, he is responsible for three technical sections (Ceramics, Nuclear Materials, and Nondestructive Methods and Diagnostics), and is functionally responsible for Quality Assurance, Personnel, and the Lynchburg Research Center (LRC).

Prior to this, he was Manager of the Nuclear Materials Section at the Lynchburg Research Center, providing technical and administrative direction to research groups within the section who performed programs evaluating all aspects of nuclear and irradiated materials.

Dr. Davis joined Babcock & Wilcox in 1972 as a Research Engineer involved in the development of UO_2 fuel for nuclear power reactors. He performed thermal resinter studies on production fuels for correlation with in-reactor behavior in order to derive models to describe the fuels densification effect. In addition, he directed programs of fabrication development for dimensionally stable fuel and UO_2-Gd_2O_3 fuel. Other past duties and experience include the characterization and fabrication of reactive powdered materials, fuels plant manufacturing support, and nondestructive post-irradiation examination of nuclear fuel components at the reactor sites.

Prior to joining B&W Dr. Davis was a Research Metallurgical Engineer at the Aerospace Research Laboratories, Wright-Patterson AFB, Ohio where he specialized for four years in the area of high temperature reactions, particularly oxidation, hot corrosion, and related mechanisms. During this time he coauthored more than 20 publications and received the Air Force Systems Command Award for Technical Achievement in both 1971 and 1972.

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Resume

HERBERT FEINROTH
B&W CONSULTANT

EDUCATION	Bachelor of Science, Chemical Engineering, University of Pennsylvania, 1957 Bettis School of Reactor Engineering, 1960
PROFESSIONAL EXPERIENCE	
1982-PRESENT	GAMMA ENGINEERING CORPORATION, PRINCIPAL In his present position, Mr. Feinroth provides consultation services to the nuclear industry in the design, development, manufacture, construction, operation and maintenance of nuclear power plants for ship propulsion and electric power generation. He specializes in ensuring incorporation of reliability and human factors engineering into mechanical systems and in development of management and QA systems to effectively control and direct projects.
1976-1982	UNITED STATES DEPARTMENT OF ENERGY, OFFICE OF NUCLEAR ENERGY, DIVISION DIRECTOR In this position, Mr. Feinroth served as technical advisor to DOE management regarding the causes and implications of the TMI accident. He assisted the staff of the Kemeny Commission investigating the accident and assisted the Carter Administration in developing follow-up actions to the Kemeny recommendations. He initiated a new DOE program to learn from the accident by conducting on-site examinations, established a cooperative program among DOE, EPRI, NRC, and GPU to conduct this program at TMI, negotiated an agreement with the Institute of Nuclear Power Operations to conduct joint programs with DOE, and implemented joint programs with the Commonwealth of Pennsylvania to allow independent radiation monitoring by the citizens near Three Mile Island.

1976-1976 ENERGY RESEARCH DEVELOPMENT ADMINISTRATION, OFFICE OF NUCLEAR ENERGY, ASSISTANT DIRECTOR

In this assignment, Mr. Feinroth directed and managed ERDA's sodium breeder development programs in the fields of nuclear fuel, reactor physics, materials, chemistry, core design, and fuel fabrication. His major task was to shift the focus of LMFBR technology toward direct engineering support of design, construction, and operation of breeder reactor plants such as FFTF and the Clinch River Breeder Reactor.

1974-1975 FEDERAL ENERGY ADMINISTRATION, DIRECTOR OF POWER PLANTS DIVISION, ASSOCIATE DIRECTOR OFFICE OF NUCLEAR AFFAIRS

In this capacity, Mr. Feinroth served as a technical advisor on the Price Anderson and uranium enrichment legislation being sponsored by the FEA and initiated actions on fuel reprocessing, waste, disposal, and uranium production aimed at removing constraints to the further expansion of nuclear power. He sponsored new programs to assist in improving the availability of existing coal and nuclear power plants.

1957-1973 UNITED STATES ATOMIC ENERGY COMMISSION

During this period, Mr. Feinroth assisted in the startup and first refueling of the Shippingport Atomic Power Station, and subsequently served as senior staff assistant to Admiral Rickover for core manufacture and production of pressure vessels and steam generators. He was directly responsible for the procurement, manufacture, and quality control of production cores for surface ships and submarines, and for trouble-shooting the manufacturing problems that were delaying delivery for major components to Naval Ships. He was project manager for the development of the core design for the USS Nimitz class aircraft carriers, and led a special heavy pressure vessel forging task force to advance the state of the art for producing high strength steel forgings for reactor plant applications. He also served as project manager for development of a new, ceramic wafer type fuel and core design for the second core of the Shippingport Atomic Power Station. In 1972, Mr. Feinroth joined the AEC's Division of Reactor Development as Chief of the Facilities Branch. There he managed programs to

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HERBERT FEINROTH
B&W CONSULTANT

develop and test components for the LMFBR program, including test facilities for sodium pumps and steam generators.

TECHNICAL PUBLICATIONS

Mr. Feinroth is the author and contributor of numerous papers and books on nuclear science, engineering, and energy development.

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Resume

DAVID FRANKLIN FRECH
Senior Engineer, Duke Power Company

<u>EDUCATION</u>:	U.S. Military Academy: B.S. Military Engineering, 1949 MIT: M.S. Nuclear Engineering, 1968
<u>WORK EXPERIENCE</u>:	
8/69 - Present	<u>Duke Power Company, Nuclear Engineer/Senior Engineer</u> Helps manage shipment and receipt of new and spent fuel to and from three nuclear stations (seven reactors), including interface with shipping casks. Responsible for evaluation of storage rack designs and subsequent installation and use at station spent fuel pools. Responsibilities for design and installation of independent spent fuel storage installation (ISFSI) at Oconee. Represents Duke Power on joint utility Transportation Working Group.
6/61 - 8/69	<u>MIT, Shift Supervisor/Superintendent</u> Responsible for operation, maintenance, and refueling of MIT Research Reactor. Evaluated experiments and irradiations for suitability.
4/56 - 6/61	<u>Union Carbide Nuclear Co., Development Engineer</u> Exercised responsibility on eight-hour shift for testing, startup, operation, and refueling of homogeneous reactor test.
6/49 - 6/55	<u>U.S. Army, 1st Lieutenant</u> Constructed roads and airfields.
<u>OFESSIONAL OLVEMENT:</u>	Registered Professional Engineer - NC 6947 Member, ANS

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Resume

DR. W. J. HARRIS
B&W CONSULTANT

Dr. Harris was [redacted] raised in South Bend, Indiana. He graduated from Purdue University in 1940 with the degrees of Bachelor of Science in Chemical Engineering and Master of Science, major in metallurgy. In October, 1940, he entered the Massachusetts Institute of Technology as a candidate for the degree of Doctor of Science.

From June, 1941 to December, 1945, he served as head of the Aircraft Armor Branch of the Bureau of Aeronautics, Navy Department. He reentered M.I.T. in February, 1946, and graduated with the degree of Doctor of Science in June, 1948.

He began work as head of the Ferrous Alloys branch of the Metallurgy Division of the Naval Research Laboratory in December, 1947.

In January, 1951, he was named Executive Secretary, Metallurgical Advisory Board of the National Academy of Sciences - National Research Council. He left the Board in 1954, but in May, 1957, he rejoined the Metallurgical Advisory Board as Executive Director. In January, 1960, he became Assistant Secretary, Division of Engineering, of the National Academy of Sciences - National Research Council.

He joined Battelle Memorial Institute as Assistant to the Director in January, 1954. He left Battelle in 1957, but in January, 1962, he rejoined Battelle as Assistant Director, Columbus Laboratories, of Battelle Memorial Institute and Head of the Washington Office of Battelle.

In January, 1970, he was named Vice President, Research and Test Department, Association of American Railroads. In that post he established a program that contributed to improvements in a wide variety of railroad components and operations including rail, track, wheels, railroad castings, car and locomotive design, train handling and operations, car distribution, terminal operations and methods of analysis.

In July, 1985, Dr. Harris was named Snead Professor of Transportation Engineering in the Civil Engineering Department of Texas A&M University in College Station, Texas. In April, 1987, he was named Associate Director of the Texas Transportation Institute and continued to hold the Snead Chair.

In 1985, he established W. J. Harris, Inc., a consulting company, and continues as its President. He consults primarily in the areas of product development, testing and analysis of mechanical structures, and application of human factors to system operation and maintenance.

Dr. Harris is a Fellow of the Metallurgical Society, the American Society of Metals, and the American Society of Mechanical Engineers. He has served as President of the Metallurgical Society of AIME, as President of Engineers Joint Council, and as Chairman of the National Materials Advisory Board. He has been an officer or the chairman of senior committees of several professional societies and organizations including the American Institute of Mining, Metallurgical, and Petroleum Engineers, the National Materials Advisory Board, the American Ordnance Association, the National Security Industrial Association, and the Transportation Research Board. In 1969, he served as Secretary and Staff Director of a Presidential Task Force on Highway Safety.

He has had extensive international experience, serving on committees of the International Institute of Welding and the Advisory Group for Aeronautical Research and Development. He is now serving as Chairman and President of the International Heavy Haul Association, an association of counties with major rail bulk commodity operations. He is also serving as Chairman of the Committee on International Trade and Transportation of the Transportation Research Board, National Research Council. He has organized cooperative studies or symposia with England, Germany, France, the USSR, Canada, Mexico, and the People's Republic of China.

He received the AIME Mathewson award for outstanding research in 1950. In 1976, he was named Railroad Man of the Year by Modern Railroads. He was elected to the National Academy of Engineering in 1977. He received the Distinguished Alumnus Award from Purdue

DR. W. J. HARRIS
B&W CONSULTANT

in 1965 and an Honorary Doctorate of Engineering from Purdue in 1978. In 1978, he was awarded the Distinguished Service Award by the Transportation Research Board of the National Academy of Sciences. In 1986, he was named Research Man of the Year by the Transportation Research Forum. He is listed in Who's Who in America and other compendia of leaders in science and engineering.

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Resume

HASSAN AH HASSAN

EDUCATION	B. Sc. Electronic Engineering, Cairo University (1957). Diploma, Reactor Control & Instrumentation, Moscow University (1958). M.S., Nuclear Engineering, University of Illinois (1961). Ph. D., Nuclear Engineering, University of Illinois (1968).
EXPERIENCE	Babcock & Wilcox Company (B&W), Lynchburg, Virginia
1984 - Present	Manager, Fuel Engineering Section, Nuclear Power Division. Major responsibilities of Fuel Engineering Section are: design and development of competitive fuel products; development of technology required for reload analyses; activities to enhance the cost posture of fuel related hardware and services.
1982 - 1984	Manager, Fuel Management Analysis & Services, Utility Power Generation Division. Position entailed management of 16-20 engineers engaged in contract work related to Incore Fuel Management and Fuel Cycle Design, on-line computers, core power distribution measurement, and reload licensing coordination. In addition, contract and internally funded tasks were conducted for advanced fuel management scheme and improved fuel cycle designs.
1980 - 1982	Manager, Reactor Technology, Nuclear Power Generation Division. Responsible for management of 12-16 advanced degree engineers engaged in contract and internally funded development tasks. Functions included models and methods development related to fuel management, reactor physics, radiation transport, and thermal hydraulics.
1976 - 1980	Manager, Reactor Physics and Radiation Transport, Power Generation Division. Directed the work of six advanced degree physicists and engineers in research and development activities in the area of physics and radiation transport.
1968 - 1976	Principal Engineer, Technical Staff, Power Generation Division. Work involved development and implementation of the multidimensional computer codes (PDQ07 & PDQ05) for the

core physics calculations and their associated processing and linking computer codes. Technical Manager in charge of FUPAC (IBM system) implementation at VP, TVA, and GPU. Development of calculational models, methods, and techniques for core power distributions and fuel cycle analysis. Reactor core stability and xenon oscillations. Verification of the multidimensional codes capability using operating data from B&W reactors on line.

1964 - 1967

Part-time Research Assistant, University of Illinois. Work involved the design of complete system for D_2O moderated natural uranium subcritical assembly. Wrote a safety analysis report for the operation of the subcritical as coupled to a TRIGA reactor and was approved by the AEC division of Licensing. Developed a new detection system for the study of turbulent flow. Worked with the Fast Burst Reactor at ORNL, conducting research on neutron thermalization through graphite. Obtained AEC Reactor Operator's License.

1961 - 1963

Senior Engineer, AEC of Egypt. In charge of reactor control and instrumentation groups. The work included the testing of reactor control systems. Initial loading and operation of low-power and high-power experiments. Training of new engineers and technicians. Writing specifications for power reactors as a member of a special presidential committee.

1957 - 1960

Engineer, AEC of Egypt. Participated in the manufacture and assembly of metallic reactor parts and the construction of the test reactor.

PUBLICATIONS

Principal author of twenty five (25) technical and topical reports in support of fuel reload analyses and reactor core design. Published eight technical papers related to reactor physics model and methods development.

PROFESSIONAL AFFILIATIONS

American Nuclear Society. Member of the Fuel Cycle and Waste Management Program Committee; member of the National Program Committee;

member of the Quality and Standard Committee.

HONORARY SOCIETY	Member, Phi Kappa Phi, Sigma Xi Elected to American Men of Science

B&W Fuel Company

Resume

JOHN E. McALLISTER, JR.

EDUCATION	Bachelor of Science Mechanical Engineering, 1975 Virginia Polytechnic Institute and State University.
	Master of Science Mechanical Engineering, 1976 Virginia Polytechnic Institute and State University.
	Ph.D., 1979 Virginia Polytechnic Institute and State University.
EXPERIENCE	BABCOCK & WILCOX NUCLEAR POWER DIVISION
1988 - Present	Senior Principal Engineer - Space Power and Propulsion Department
	Activities include the QA review of thermal-hydraulic computer codes, calculations of gas flow through porous media, rocket thrust calculations, and fuel particle modeling for gas reactors. Activities outside the department were assisting in preparation of work proposals, a natural convection study of a multi-tube nuclear assembly, consulting on design aspects of the new production reactor, and marketing activities.
	DUPONT - SAVANNAH RIVER PLANT/LABORATORY
1987 - 1988	Technical Supervisor - Process Systems Group - Reactor Technology Department
	Supervised a group composed of seven to eight engineers. The group was responsible for review of changes to critical reactor equipment, review of emergency operating procedures, ASME code interpretations, coolant piping integrity, calculations for design basis accidents, reactor seismic policy, and troubleshooting of daily problems for three operating nuclear rectors. The group provided technical support for new projects in two- and

three-dimensional heat transfer studies of reactor assemblies and transient analysis of reactor assembly hydraulics.

1985 - 1987

Technical Supervisor - Reactor Core Engineering Group-Reactor Technology Department

Supervised a group composed of four to seven engineers. The group was responsible for core neutronics and thermal-hydraulic analyses. Personal technical activities included uncertainty analysis of full scale emergency cooling system flow experiments, transient fluid analysis of a major safety system, and heat transfer studies of nuclear reactor components.

1979 - 1984

Research Engineer

Duties included evaluation of energy conservation projects, thermal and hydraulic considerations in nuclear reactor design, development of nuclear assembly operating limits, providing input to a nuclear reactor environmental impact statement, and technical support/troubleshooting problems while assigned to L Reactor. Also, two- and three-dimensional heat transfer studies of reactor components using general purpose finite element (ABAQUS) and finite difference (HEATING6) computer codes were completed.

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Resume

A. D. MCKIM

EDUCATION	Bachelor of Science Business Administration, 1975 Southern Indiana University. Associate of Science Mechanical Engineering, 1973 University of Evansville. Associate of Science Tool Engineering Technology, 1967 ITT - Indianapolis, Indiana.
EXPERIENCE	<u>BABCOCK & WILCOX COMPANY</u>
1985 - Present	<u>Unit Manager - Materials and Structural Analysis - Lynchburg, Virginia</u> Managed engineering unit responsible for stress, seismic, flow-induced vibration, fracture mechanics, and materials analysis associated with major NSS components and systems. Successfully managed the development of B&W Owners Group Comprehensive Life Extension Plan. Developed detailed logic of economic evaluation, technical, and licensing aspects of life extension.
1984 - 1985	<u>Unit Manager - Structural Analysis Unit - Lynchburg, Virginia</u> Managed engineering unit responsible for stress, seismic, flow-induced vibration, and fracture mechanics analysis. Successfully managed B&W Owners Group leak-before-break analysis program. Program covered loads, stresses, fracture mechanics, and leak rate analysis.
1982 - 1984	<u>Unit Manager - Component Structural Unit - Lynchburg, Virginia</u> Managed engineering unit responsible for stress analysis of major components in nuclear steam system - reactor vessel, internals, steam generators, core flooding tanks, pressurizers, piping, control rod housings, reactor coolant pumps, and service support structures. Successfully managed reactor vessel internals bolts replacement program (Boltz II project) and STEEM project - failed internal steam generator auxiliary feedwater header, to support operating plants field fixes and licensing support.

A. D. MCKIM (CONT'D)

1979 - 1982

Supervisor - Component Structural Unit - Lynchburg, Virginia

Technical supervisor for stress analysis of steam generators, piping, and reactor vessels. Successfully supported operating plant field fixes - make-up nozzle and auxiliary feedwater nozzle thermal sleeve failures. This included both stress analysis of fixes and licensing support for restart.

1967 - 1979

Engineer - Mt. Vernon, Indiana

Progression of engineering positions to Senior Engineer. Performed design and stress analysis of reactor vessels and primary piping. Analysis performed to criteria of ASME Section III USAS B31.7, and USAS B31.1. Analysis technique ranged from hand calculations to 3-D finite element analyses.

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Resume

MICHEL ROBATEL
CEO, ROBATEL, SA
PRESIDENT, ROBATEL INC.

EDUCATION:

B.S. - Engineering, Ecole Centrale Des Arts & Manufactures, France, 1950

M.S. - Chemical Engineering, University of Kansas, 1951

PROFESSIONAL EXPERIENCE:

1979-Present

Robatel Inc., President
Robatel Inc is a wholly-owned subsidiary of Robatel SA and markets the parent company's technology in the U.S. Nuclear fuel reprocessing equipment for SRP and West Valley are among their products.

1963-Present

Robatel SA, Chief Executive Officer, Chairman of the Board
Robatel SA is a French company which specializes in the design and fabrication of chemical processing equipment, hot cell equipment, and storage/shipping casks for LLW, HLW, and spent fuel.

1960-1983

Transnucleaire, Co-Founder, Member of the Board

1956-1963

Robatel & Mulatier, General Manager

1953-1956

Robatel & Mulatier, Research & Development Engineer
In this position, Mr. Robatel worked on the development of solid/liquid extraction processes for the nuclear fuel cycle, including the development of the PUREX process for plutonium separation from spent nuclear fuel. He was responsible for starting the nuclear activities of the company.

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TECHNICAL PAPERS & PUBLICATIONS:

Mr. Robatel has written several papers on technical, economic and social topics.

PROFESSIONAL AFFLIATIONS:

Mr. Robatel is a member of the American Nuclear Society, the Societe Francaise d'Energie Nucleaire, the Groupement Intersyndical de l'Energie Nucleaire, and the Alpha Chi Sigma Chemical Engineering Honorary Fraternity.

B&W FUEL COMPANY

Contract No. DE-AC07-88ID12701

BR-100

CASK SYSTEM

ENGINEERING TEST PLAN

51-1173352-03

November 1989

PREPARED BY R. L. Fish 11/20/89
R. L. Fish, BR-100 Materials

REVIEWED BY E. J. McGuinn 11/20/89
E. J. McGuinn, BR-100 Licensing

APPROVED BY C. A. Armontrout 11/20/89
C. A. Armontrout, BR-100 Quality Assurance

APPROVED BY P. C. Childress 11/21/89
P. C. Childress, BR-100 Project Manager

REVISION PAGE

Revision #	Description	Date
01	Incorporation of additional detail in description of testing program and other comments received on Rev. 0 of document	June 1989
02	Deletion of Appendices	August 1989
03	Addition and deletion of tests	November 1989

TABLE OF CONTENTS

1.0 SUMMARY

The purpose of this document is to present, in summary form, the engineering test plan that the B&W Fuel Company (BWFC) will use in support of the BR-100 cask design development program. DOE approval of this document is required before the start of the engineering tests. This plan addresses only the testing proposed for the cask itself and the impact limiters. Testing of other system components such as the handling hoist, rail car, and de-watering system will be addressed in other documents.

Engineering testing is used to characterize material performance and to benchmark analytical codes and methods. Engineering testing will also be used to demonstrate component performance. BWFC plans to perform most of the engineering testing at its manufacturing plant in Lynchburg, Virginia or at its parent company's, Babcock and Wilcox, facilities in Alliance, Ohio and Lynchburg, Virginia. Where judged beneficial to the program, testing will be subcontracted to independent test labs. All testing will be performed in accordance with the controls imposed by the cask project QA plan.

Section 2.0 discusses the scope of the overall BR-100 cask testing program. Section 3.0 presents a general discussion of the methodology used to define and control testing, the selection of a testing facility, and procedures for reporting the test results.

An overview and schedule for the proposed test program are presented in Section 4.0. Additional details of the planned testing are provided in the appendices.

2.0 SCOPE OF TESTING

Three categories of testing will be utilized in the development and evaluation of the BR-100 cask design over the life of the program.

1. Engineering: to determine or confirm material properties or design performance and to benchmark analytical inputs using material specimens and component mock-ups. The engineering testing has been subdivided into two basic types: 1) material testing, and 2) component evaluations.
2. Verification: to demonstrate compliance with 10CFR71 performance requirements for accident conditions using scale or full size models of the final design configuration.
3. Acceptance: to demonstrate functional and operational aspects of the design or contractual requirements using full-scale prototypes.

This document describes the general methodology that will be used in planning and controlling testing activities and summarizes the tests that will be performed as part of the engineering testing task. Separate test plans will be developed for the verification and acceptance testing.

A number of test facility options are available. BWFC plans to conduct engineering tests at B&W's Alliance Research Center (ARC), the Lynchburg Research Center (LRC), or the Commercial Nuclear Fuel Plant (CNFP). Alternatively, BWFC may elect to subcontract the testing to an independent or government laboratory.

3.0 TESTING DEFINITION AND CONTROL

This section presents an overview of the way the testing will be defined, performed, controlled, and reported. Because testing may be performed at any of several test facilities, the following discussion also addresses the site-specific differences in the approaches used in conducting the testing.

In general, the philosophy used in coordinating the design and testing functions of an engineering project is that testing supports the design effort. The design organization is responsible for the adequacy of the design. For safety related testing, designer generates an Engineering Requirements Document (ERD) describing the information required from the test to support the design. In response, a test technical plan, detailed test procedures, and a test QA plan are developed defining the testing and data requirements necessary to satisfy the needs described in the ERD. After each test is completed, a final test report will be issued describing the test and containing the test data. The test report will be submitted in accordance with Section 4.4.7 of the cask contract statement of work.

For non-safety scoping tests, the designer generates the requirements for the tests to be performed, the parameters to be controlled, and the measurements to be taken both during and after the tests. This information is transmitted by procurement authorization to guide the work, but verbal communications may adjust or modify the work as it proceeds. A final report is generated to document the work that was actually performed and to report the test results.

3.1 Test Definition

This section addresses the process used to define and control testing.

Testing at ARC, LRC, and Independent Contractors

The responsible design engineer will specify to the testing organization the test requirements in an Engineering Requirements Document (ERD). The ERD, which is approved by the Project Engineer and Project Manager, defines the test purpose; appropriate requirements to ensure conformance with applicable codes and design specifications; appropriate quality assurance requirements; additional requirements on instrument accuracy, acceptance criteria, environmental test conditions, test specimen condition, special test equipment and/or calibration requirements; and provisions for acquisition, collection, retention, and reporting data. The ERD will be submitted to DOE for information purposes. The testing organization produces a detailed test technical plan based on the ERD for approval by the BWFC Project Manager. Any proposed test or hardware modifications must be approved by the

Project Engineer. The test results, along with all testing procedural deviations, will be documented in the final test report by the testing organization.

Testing at CNFP

Testing at CNFP is performed in much the same manner as in other testing facilities. The principle difference is that the BWFC design organization or other qualified testing organization (e.g. LRC), retains full control of the testing and is responsible for conducting the test and reporting the results. The testing is conducted under the direct guidance of the responsible engineer with trained technicians from CNFP performing the actual test operations, making measurements, and recording the data. All test results will be incorporated into a written report by the design engineer.

3.2 Quality Assurance

BWFC is responsible for the overall Quality Assurance Program (QAP)[1]. The objective of the Spent Fuel High Level Waste Services (SFHLWS) Quality Plan[2] is to ensure the safe, reliable performance of its products and services. The QAP establishes the prerequisites for achieving and maintaining quality by requiring specialized equipment and skills, employing appropriate administrative and process controls, training personnel who will perform activities that affect data quality, and verifying the quality aspects of the procedures that will be used in the testing. The SFHLWS Quality Assurance Program [2] will be implemented by the BR-100 Quality Assurance Plan [1] for the work performed on this project.

Quality Classification

The SFHLWS QA Plan categorizes items or services as either safety or non-safety related. A QA list defining the cask components and activities as safety or non-safety related is contained in the BR-100 Quality Assurance List document.[3] The classifications are as follows:

1. **Safety Related (DOE Quality Levels 1 and 2):**
   Items or services that have a significant potential impact on public and/or occupational radiological health and safety per 10CFR20.

2. **Non-Safety Related (DOE Quality Level 3):**
   Items or services that have minor or no potential impact on public and/or occupational radiological health and safety. These items and services will be controlled per good management, engineering, or laboratory practices.

3.3 Testing Facilities

This section presents a summary of the testing capabilities of ARC, LRC, and CNFP facilities.

Testing Capabilities of ARC

Alliance Research Center, located in Alliance, Ohio, is a multi-disciplinary facility which employs state-of-the-art measurement, diagnostic, and analysis equipment designed specifically for nuclear technology. ARC's Material Engineering Laboratory provides support services in metallurgy, severe-environment corrosion, aqueous corrosion, materials processing, joining processes and materials development, analytical and physical metallurgy, nondestructive methods and diagnostic systems, metallurgical analysis of nuclear materials, and failure analysis. ARC's Mechanical Engineering Laboratory provides services in heat transfer and fluid mechanics systems analyses with advanced numerical applications and fundamental model development. In structural mechanics evaluations, the areas of expertise include analytical mechanics, experimental mechanics, flow-induced vibration analysis, and applied measurement technologies.

The ARC Metallurgy Lab has an array of state-of-the-art equipment that performs standard material tests at temperatures ranging from -320°F to 2200°F. Tests include creep, stress-rupture and relaxation properties in addition to the standard tension tests.

Hardness tests can be conducted for standard hardness ranges as well as for superficial and micro-hardness ranges. Charpy V-notch impact tests are also routinely performed as part of the thick-wall vessel program. The lab has the capability of performing fully automated, computer-controlled testing for around-the-clock operation. This capability permits the applied load, stroke or strain and the resulting changes to the test parameters to be monitored and recorded automatically.

The ARC Structural Mechanics Lab will be used for much of the component structural testing. With its extensive handling and test equipment, the lab is capable of handling test samples weighing from a few ounces to as much as 15 tons. Temperature environments can range from sub-freezing to 1000°F. This lab is routinely used for shock and vibration research and has extensive expertise in impact testing, especially in the area of impact data acquisition, data reduction, and interpretation. They also have experience with various types of impact load/deflection instrumentation, a key element in obtaining quality data from impact tests.

ARC's Electrochemical Lab can perform a wide variety of tests including general corrosion rate, pitting, and stress corrosion cracking tests. Lab capabilities include highly sophisticated, fully automatic constant potential tests, galvanic corrosion measurements, and impedance measurements. A special autoclave facility is equipped to perform constant strain or extension rate tests that can quickly screen materials susceptible to stress corrosion or other problems.

Testing Capabilities of LRC

The Lynchburg Research Center has many of the same general capabilities as ARC but with a traditional emphasis on testing associated with nuclear fuels, irradiated materials, and the development of ceramic materials. It is a multi-disciplinary facility engaged in a wide variety of research activities covering the range of nuclear development work, as well as studies in areas such as process control, metallographic analysis, ceramics, nondestructive testing, chemistry, waste systems, and decontamination. LRC maintains a trained technical staff to perform sophisticated material property testing with a well-equipped metallurgical laboratory. The center's facilities include hot cells that have been used to perform a wide range of investigations and inspections related to radioactive fuel, fuel components, and related hardware. Associated with the hot cells is a cask handling area. Cask maintenance tasks are routinely performed at LRC. This experience is useful in evaluating cask operational features.

Specific LRC facilities of interest for the cask development program include the hot cells and cask handling area with their robotic handling equipment, material testing labs, and the ceramic research and testing laboratory. Previous experience, gained in performing experiments in the hot cells with robotic equipment and with handling, operation, and maintenance of spent fuel shipping casks, will be useful in evaluation of cask design options. More directly related to the engineering testing program is the use of the material testing labs to investigate material behavior including determination of structural properties at room temperature as well as at elevated temperatures. The ceramic research unit will be used for the testing and evaluation of the concrete used as a thermal/neutron shield material.

Testing Capabilities of CNFP

CNFP, primarily a manufacturing facility, maintains a staff of technicians highly trained in the areas of inspection and quality control. They routinely perform testing activities associated with manufacturing. The cask designer is able to take advantage of CNFP's expertise in these areas to perform some of the engineering testing, especially the testing related to the demonstration of component mechanical performance. This type of testing is very similar to typical incoming inspection testing that is a routine part of the CNFP manufacturing effort.

In general, only component functional testing will be performed at CNFP. During testing, technical support will be provided by BWFC, with CNFP-trained technicians from the inspection department operating the test equipment and recording instrument readings. The cask design organization is responsible for both the ERD and the test procedure (including data sheets) and will remain fully involved in all aspects of the testing at CNFP.

3.4 Data Control and Reporting

To ensure the validity of the test results, the testing performance, data acquisition and retention, and disposition of applicable test reports will conform to both the test technical plan and QA plan. The QA plan will include requirements for technical review of all data by the cognizant test engineer and a QA review of the final report. All test instrumentation will be calibrated in accordance with written procedures. Calibration will involve the use of standards traceable to the National Institute of Standards and Technology (NIST).

For Testing at ARC, LRC, and Independent Contractors

Based on BWFC's Engineering Requirements Document, a test technical plan along with specific test procedures will be prepared by the research or testing facility prior to the start of testing. Documents affecting the quality or validity of the test data or results, such as test equipment drawings, inspection checklists, technical procedures, and the final reports, will be reviewed for accuracy and conformance to test requirements. All instrumentation will be calibrated and certified in accordance with procedures. During testing, a log or laboratory notebook will be maintained for the purpose of documenting pertinent activity and data entries. All significant entries, such as records of test parameters or test data, will be signed and dated on the date of occurrence by the testing engineer or his representative. The data collected will be retained by the testing facility for a minimum of four (4) years and eventually transferred to a long-term data retention center for at least forty (40) years, depending upon the nature of the data. All test equipment descriptions, testing procedures, data acquisition, calculations performed during the test or on the final data, and final test conclusions will be documented in the test report. If any non-conformances occur, they will be reported using the standard Contract Variance Report (CVAR) system. The CVAR's will be reviewed and approved by the Project Engineer and quality assurance representative with the final disposition being properly documented. Any corrective action that needs to be taken will be documented and monitored to verify implementation. All final test reports will be reviewed and approved by a QA Representative for conformance to test procedures and the test technical plan.

For Testing at CNFP

Specifications for data collection and retention control for CNFP are similar to those for the other testing facilities. All data and test results will be recorded on data sheets. These data sheets will be signed and dated by the technicians performing the tests and will be reviewed by CNFP's QA organization for conformance to test procedures and the test technical plan. Copies of all data taken during the test are entered into the record center at CNFP and then transmitted to the B&W engineering record center at NPD. The responsible engineer will observe all testing performed at CNFP and incorporate the test results into the final report.

4.0 PROPOSED ENGINEERING TESTING PLAN

This section presents an outline of the engineering testing planned for the BR-100 cask project. For each test, the test purpose is presented along with a brief discussion of the scope of the testing and type of data or results to be reported. The engineering tests are divided into two types: Material Testing and Component Evaluations. More detailed descriptions of the tests will be provided in Engineering Requirements Documents.

As the cask design evolves, the need for the deletion of a test or additional testing may be identified, in which case this document will be updated and the revisions sent to the DOE for review and approval.

4.1 Material Tests

4.1.1 Concrete Compressive and Bending Strengths

Purpose - Determine the compressive and bending strengths of the concrete mixture used in the Thermal/Neutron Shield. The results will be used as input for structural analyses. This testing will be QA Safety Related.

Testing - Perform compressive loading and bending tests on concrete samples cast into airtight containers simulating the air-tightness of the actual concrete/steel cask configuration. The concrete shall be tested in the as-cured, dried, and aged conditions. Also, determine relative changes in concrete integrity due to thermal cycling (-40°F to 200°F) using relative dynamic modulus of elasticity measurements.

4.1.2 Concrete Neutron Shielding Effectiveness

Purpose - Determine the range of total water and boron content and homogeneity in the concrete mixture so that the neutron shielding effectiveness can be calculated. This testing will be QA Safety Related.

Testing - Several samples of concrete will be poured and cured in air-tight vessels. Some of these vessels will be vented and the samples heated to drive off essentially all of the free water (the dried state). Samples will also be subjected to thermal cycling. Samples of concrete from each of these conditioning treatments will be sectioned and analyzed for water and boron content as a function of location within the samples. These data will provide the basis for evaluating the shielding effectiveness under operational conditions.

4.1.3 Concrete Thermal Properties

Purpose - Measure the thermal conductivity and thermal expansion as a function of temperature for as-cured concrete, dried concrete (essentially all of the free water driven off), and fired concrete (essentially all of the chemically-bound water driven off). Also, measure the specific heat of transformation (heat required to essentially eliminate the chemically-bound water) for typical concrete samples for use as input data in analytical models. This testing will be QA Safety Related.

Testing - The thermal conductivity test specimens will be disc-shaped and designed to hold a constant temperature difference across the specimen of about 50°F. The thermal expansion measurements will utilize quartz rods attached to the top of the concrete specimens and a NIST standard. These quartz rods will extend out through the furnace to allow cold measurement of the relative expansion of the standard material and the concrete specimen. Both thermal conductivity and thermal expansion will be measured over the expected operating temperature range of the concrete.

4.1.4 Concrete Volumetric Changes

Purpose - Determine the volumetric changes (if any) that occur during curing of the concrete or due to freezing for use in analytical calculations. This testing will be QA Safety Related.

Testing - Measure the change in external volume of samples of as-cured concrete as the concrete temperature changes ($-40^{\circ}F$ to $200^{\circ}F$).

4.1.5 **Balsa and Redwood Compressive Strength**

Purpose - Determine the compressive strength of the balsa wood and redwood (including variation of moisture content, grain orientation and density) and their energy absorption characteristics during dynamic loadings. This testing will be QA Safety Related.

Testing - Perform comprehensive load versus deflection tests on a series of balsa wood and redwood samples (confined) to determine the structural properties (force vs deflection) of various wood types and the effects of adhesives at various temperatures.

4.1.6 **Kevlar Tension Behavior**

Purpose - Determine the tension behavior of the Kevlar composite to be used in impact limiters for use in analytical calculations. This testing will be QA Safety Related.

Testing - Perform a series of tension tests on representative samples of the Kevlar composite. The samples shall be large enough for adequate load redistribution. The tests shall investigate the effects of test temperature, strain rate and water absorption.

4.1.7 **Anodized Aluminum Wear Properties**

Purpose - Determine wear resistance as a function of load and time of the anodized aluminum fuel basket material for use in assessment of durability. This testing will be QA Safety Related.

Testing - Perform contact wear and scratch testing on representative samples of the anodized aluminum to be used for the fuel basket. Determine contact wear rate for anticipated fuel assembly materials and determine the load required for scratch-through using anticipated fuel assembly materials.

4.1.8 **Corrosion Behavior**

Purpose - Determine the corrosion resistance of anodized aluminum under alternating pool water and elevated temperature inert gas environments. Use information for determination of any detrimental effects. This testing will be QA Safety Related.

Testing - Perform corrosion tests on samples of the aluminum fuel basket material, bare, scratched-anodized and unscratched-anodized, using alternating simulated spent fuel pool water and elevated temperature inert gas environments. The tests will be conducted over a sufficient period of time to allow determination of corrosion behavior.

4.1.9 Emissivity Measurements

Purpose - Determine the thermal emissivity/absorptivity characteristics of the hard anodized aluminum and the outer shell paint. Use information to help perform cask thermal analyses. This testing will be QA Safety Related.

Testing - Measure the emissivity/absorptivity of both the hard anodized aluminum and the painted stainless steel outer shell in both the solar and infrared radiation bands.

4.2 Component Evaluations

4.2.1 Shield Conductivity Tests

Purpose - Measure the effective thermal conductivity of the Thermal/Neutron Shield and provide thermal boundary conditions for the inner shell for use in analytical calculations. This testing will be QA Safety Related.

Testing - Perform heat transfer tests on a cross-section model with actual wall thickness dimensions to measure heat flow through the vessel wall under normal operating conditions.

4.2.2 **Shield Effectiveness (Fire Test)**

Purpose - Investigate the effectiveness of the thermal shield under Regulatory Fire Conditions, and the thermal behavior after return to steady-state conditions. This testing will be QA Safety Related.

Testing - Subject the same model or a similar one to that described in paragraph 4.2.1 to simulated, external fire conditions (consistent with 10CFR71.73) to determine the effect of the regulatory fire accident environment on component performance and thermal conductivity.

4.2.3 **Shield Thermal Cycle Test**

Purpose - Determine the effect of a large number of thermal cycles (representative of the loading/unloading cycles and/or environmental conditions) on the integrity of the Thermal Shield. This testing will be QA Safety Related.

Testing - A model, not necessarily representative of the section but with real thicknesses, will be thermally cycled. Detailed inspections will be performed to document any adverse effects.

4.2.4 **Shield Configuration Impact Test**

Purpose - Determine the effect of the borated concrete on the mechanical behavior of the stainless steel outer shell during drop impact testing. This testing will be QA Safety Related.

Testing - A series of impact tests will be conducted on stainless steel/concrete/lead/stainless steel samples. The test samples will be disassembled and inspected after each test. details.

4.2.5 **Impact Limiter Behavior Tests**

Purpose - Obtain data concerning the ability of the Impact Limiter to absorb the energy of an impact within allowable g limits. Use the information for analytical input and for benchmarking analytical results. Level I and II testing will be scoping in nature and, therefore, will be QA Non-Safety Related. Level III testing which will generate data for qualifying the impact limiter or benchmarking codes will be QA Safety Related.

Testing - Level I: Perform a series of drop tests on scale models of impact limiters; also determine the load/deflection curves for 90° (side), 45° (oblique) and 0° (end) drop orientations by conducting a series of static load-deflection tests. Perform detailed inspections and document damage. Use the test results to identify the best impact limiter design. Level II: Conduct additional dynamic and static tests on the best design to refine the impact limiter for additional design parameters (such as fire and puncture). Level III: Perform static and dynamic tests on final design to qualify impact limiter and benchmark codes.

4.2.6 Impact Limiter Puncture Behavior Tests

Purpose - Determine the ability of the Impact Limiter, with its Kevlar composite skin, to resist puncture. This testing will be QA Safety Related.

Testing - Perform a series of puncture tests representative of the regulatory conditions, but using several different penetrators, by dropping a large section of Impact Limiter (or scale model) onto a penetrator. Measurements will be made of input energy and detailed examinations will be conducted to determine the extent of damage.

4.2.7 Seal Tightness Tests

Purpose - Demonstrate the tightness of the closure seals under operating and accident temperature conditions. This testing will be QA Safety Related.

Testing - Demonstrate the tightness of the closure seals under operating temperatures. Test mechanical and thermal cycle mockup to demonstrate that the seal remains tight after mechanical and high/low temperature cycles.

4.2.8 Fuel Cell Structural Behavior Tests

Purpose - Obtain data to allow structural analyses to be performed on the fuel cell components and to benchmark fuel cell structural performance codes. This testing will be QA Safety Related.

Testing - Tension and compression tests will be performed on specimens sectioned from actual fuel cells. Tests will be conducted as a function of temperature and strain rates. Sections of fuel cells will be tested under simulated drop test conditions.

4.2.9 <u>Boron Carbide/Aluminum Neutron Absorber Evaluation</u>

Purpose - Evaluation to assess the integrity and performance capabilities of a boron carbide/Aluminum Cermet neutron absorber material that is to be attached to the aluminum fuel cell. Scoping tests will be QA Non-Safety Related. Tests which generate data for use in qualifying the material as a neutron absorber will be QA Safety Related.

Testing - The mechanical integrity of a B4C/Aluminum Cermet material will be assessed after exposure to two potentially degrading environments: thermal cycling and vibration. The mechanical integrity shall be evaluated using bend tests and tension tests to simulate the spent fuel assembly loading and consequential deformation of the fuel cell walls and cermet loading during hypothetical cask drop accidents. The bend tests will be conducted in air at nominal as well as rapid loading rate at an elevated temperature. Detailed examination of the cermet after thermal cycling, vibration testing and bend testing will be conducted to characterize the mechanical integrity of the cermet and its ability to remain intact under the hypothetical accident deformations. Neutron attenuation and corrosion testing is also planned.

5.0 REFERENCES

1. 56-1173003-02, "Quality Assurance Plan for BR-100 Cask System," April 5, 1989.

2. 56-1168014-01, "Quality Assurance Program for Spent Fuel and High Level Waste Services (SFHLWS) Product Line," August 30, 1988.

3. 07-1173038-00, "BR-100 Cask System Quality Assurance List," September 13, 1988.

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Resume

ROBERT S. ENZINNA

EDUCATION

Masters of Engineering Degree in Nuclear Engineering from Rensselaer Polytechnic Institute Specializing in Reliability Engineering (1979).

Bachelor of Science Degree in Nuclear Engineering from Rensselaer Polytechnical Institute (1978).

EXPERIENCE

BABCOCK & WILCOX COMPANY (B&W), Lynchburg, Virginia

Application of human factors quantification to Probabilistic Risk Assessment (PRA) of mechanical systems.

PRA of a B&W-designed nuclear power plant to investigate pressurized thermal shock. This study identified potential pressurized thermal shock transients and assessed the frequency and severity of the sequences. Lead engineer for event tree construction and quantification and common cause failure analysis.

Lead engineer for the extension of test intervals for the B&W Reactor Trip System. The project obtained licensing approval for the relaxation of technical specification surveillance requirements. Benefits include increased reliability and safety and reduced manhour expenditure and spurious trips. The effects of surveillance testing on system reliability/availability were determined with a Reliability Block Diagram (RBD) model that included factors such as common mode failure, human error, and wearout (age failure).

Lead engineer for an Advanced Light Water Reactor PRA. The PRA investigated the core melt frequency of the advanced reactor design and evaluated the impact of passive safety systems. The PRA is integrated with the design process to enhance reliability and availability.

IDCOR (Industry Degraded Core Rulemaking). Activities related to the IDCORE Technical Advisory Group included review of the MAP code and a preparation of test cases for the B&W reference plant.

Directed root cause analysis of generator hydrogen seals and reliability centered maintenance analysis of boiler feed pumps for two large oil-fired power plants.

Lead engineer for FMECA analysis of the B&W Integrated Control System. The project involved common mode failure analysis of power supplies and sensor taps and coordination of input from engineers performing simulation on B&W's Advanced Controls Research facility.

Coordinator of reliability analysis for breeder reactor steam, feedwater and condensate systems of DOE conceptual design study LMFBR (Liquid Metal Fast Breeder Reactor).

Milhdbk-217D mean-time-between-failures and RBD analyses of various instrumentation systems. These included an acoustic Loose Parts Monitoring System for the DOE an digital Safety Parameter Display Systems.

Fault Tree analyses with Milhdbk-217D stress calculations for a Bailey Controls Co. "Network-90" digital control system.

Reliability analyses of Engineered Safety Features Actuation Systems and Emergency Feedwater Initiation and Control Systems (EFIC).

Reliability and availability analyses of Auxiliary Feedwater Systems. Studies involved detailed fault tree analysis, human reliability analysis, GO models, common mode failure analysis, recommendation and evaluation of design improvement alternatives, and cost/benefit evaluation.

Development and maintenance of Reliability computer codes for Fault Tree and RBD analyses. Responsible engineer for FTAP, PACRAT, and GO computer codes. Participated in development of IRIS interactive graphics reliability workstation including RBD code interface, validation testing, and conversion to IBM-AT.

PROFESSIONAL

B&W representative to the Atomic Industrial Forum Subcommittee on Probabilistic Risk Assessment and working group on Probabilistic Methodology of the Tech. Spec. Subcommittee. The objective of these subcommittees is to review and attempt to influence NRC policy and regulations related to probabilistic safety goals, degraded core rulemaking, and technical specification relaxation. These committees have subsequently been reorganized under NUMARC.

DOD Secret Clearance.
DOE L Clearance applied for.

PUBLICATIONS

"Probabilistic Analysis for the Babcock & Wilcox Advanced Light Water Reactor", 1988 ANS Topical Meeting, Seattle, Washington. Co-authored with S.H. Levinson.

"A RAM Evaluation Tool: combining abridged-PRA with Root-Cause Analysis," 1987 INTER-RAM conference, Toronto, Canada, May 1987. Co-authored with S.H. Levinson.

"Optimization of Reactor Trip System Test Intervals", 1984 INTER-RAM Reliability Conference for the Electric Power Industry, Las Vegas, Nevada, April 1984. Reprinted in Nuclear Engineering International, September 1985.

"Pressurized Thermal Shock: An Evaluation for B&W-designed NSS Plants", 1984 ANS Annual Meeting, New Orleans, Louisiana, June 1984. Co-authored with D.R. Westcott and S.H. Levinson.

"Computer Automated Fault Tree Construction and Analysis System," ANS Conference on PRA, Port Chester, NY, September 1981. Co-authored with J.E. Lynch.

"Emergency Feedwater System Reliability Improvement Program," 1982 Engineering Conference on Reliability for the Electric Power Industry, Hershey, PA, June 1982. Co-authored with J.E. Lynch.

"Auxiliary Feedwater Reliability Analysis for Plants with B&W Designed NSSs," 1980 ANS Annual Meeting, Las Vegas, Nevada, June 1980. Co-authored with W.W. Weaver.

"Application of System-2000 Data Base Management System to a Nuclear Reactor System Reliability Data BAse," Rensselaer Polytechnic Institute, Troy, New York, August 1979.

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Resume

JOHN F. HENDERSON

EDUCATION

B.S. - Ceramic Engineering, Alfred University, 1968
- Ceramic Manufacturing Technology, The Center for Professional Advancement, E. Brunswick, NY (2.4 CEU's)

EXPERIENCE

1984 - Present

BABCOCK & WILCOX, Lynchburg Research Center, Lynchburg, VA, Senior Research Engineer, Nuclear Materials Section

He is responsible for the day-to-day operations of the LRC hot cells and cask handling facilities, where testing and handling of radioactive materials and components are done. His duties include scheduling, job assignments, and arranging for maintenance and repair of facilities and equipment. He also acts as liaison between the Post-Irradiation Examination Group and the Health Physics, Safety and Licensing functions at the Center and engages in additional activities including product review, cost estimating, proposal preparation and marketing.

1980 - 1984

BABCOCK & WILCOX, Lynchburg Research Center, Lynchburg, VA, Senior Research Engineer, Ceramic Section

As a member of the Ceramics Section's Refractory Technology group, his responsibilities included refractories selection and evaluation, furnace design and graphite electrode evaluation for the industrial furnaces at FPGD and TPG. The group also supplied technical assistance in product development and evaluation for the Insulating Products Division.

1969 - 1980

NATIONAL STEEL CORPORATION RESEARCH AND DEVELOPMENT Metallurgy Division

He was involved with the supervision of the refractory testing laboratory, trouble-shooting in-plant problems related to refractory service, quality control

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JOHN F. HENDERSON

	adherence by refractory suppliers to standards and specifications, new product evaluations, in-plant studies to evaluate performance of refractories under service conditions, and researching new methods of improving refractory life in steelmaking applications.
SECURITY CLEARANCE	DOE-L, 12/16/80
PROPOSED LABOR CATEGORY	Engineer (Sr: Res. Engr., R&D)
MOST SIGNIFICANT TECHNICAL ACCOMPLISHMENT IN LAST FIVE YEARS	Coordination/supervision of rebuild of main hot cell through maximization of remote handling and minimization of worker exposure.
MOST SALIENT SKILL THAT RELATES TO PROPOSED EFFORT	Ability to effectively coordinate/supervise the multi-discipline areas of support necessary for the proper operation of the Hot Cell Facility in conducting nuclear fuel and reactor component examinations. Applies human factors engineering to products and processes proposed for use in cask handling and maintenance or irradiated material handling and examination.

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Resume

DAVID S. HERNANDEZ

EDUCATION

1984	B.S. Engineering Technology, Lamar University, Beaumont, Texas
1977	Diploma - Devry Institute of Technology
1977	Electronics Technology Bell & Howell Correspondence Course
1971	U.S. Navy Nuclear Power School

EXPERIENCE

1985 - Present

POWERSAFETY INTERNATIONAL, Lynchburg, Virginia

Director, Curriculum Development, responsible for managing the development of courseware that involved both classroom and hands-on training. Applied human factors engineering to products ranging from tooling to controls and instrumentation and to the development of maintenance and operational procedures.

9/77 - 9/85

GULF STATES UTILITIES COMPANY, Beaumont, Texas

Director, Operator Development (3/82 - 9/85)

Responsible for developing operating procedures for six power stations with each station having one to five operating units. Responsibilities included training of power plant and power dispatching personnel, and budgeting for and supervising staff of fifteen.

Supervisor, Skills Training (11/79 to 3/82)

Planned, developed, administered, coordinated and supervised the in-depth technical skills training effort for company employees. Was responsible for overseeing budget expenditures, and supervising a seven member training staff. Involved in personnel, training, and industrial relations.

Coordinator, Skills Training (10/78 to 11/79)

Was responsible for planning, developing, administering, and coordinating the technical skills training of employees in power generating and power distributing operations. Duties included overseeing budget expenditures and supervising six member training staff.

Training Representative (9/77 to 10/78)

Responsibilities included developing, instructing, and coordinating technical training programs for fossil and nuclear power plant operations including video tape productions and the use of audio/visual aids.

8/69 - 8/77 UNITED STATES NAVY

Staff Instructor, Nuclear Power Training Unit, Idaho Falls, I.D. Served in Submarine Nuclear Power Field. Honorable Discharge.

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Resume

STANLEY H. LEVINSON

EDUCATION:

Master of Administration - Industrial Management, Lynchburg College, Lynchburg, Virginia, May 1988

PhD - Nuclear Engineering, Rensselaer Polytechnic Institute,Troy, New York, 1982

ME - Nuclear Engineering, Rensselaer Polytechnic Institute, Troy, New York, 1979

BS - Nuclear Engineering, Rensselaer Polytechnic Institute,Troy, New York, 1978

Dissertation - "Methods and Criteria for Evaluation of Nuclear Reactor Fire Protection Alternatives and Modifications". Developed an interactive graphics analysis package to evaluate the effectiveness of fire protection systems in nuclear power plants. Developed models for ignition, detection, suppression, and propagation to be used in the framework of a Monte Carlo simulation.

EXPERIENCE

BABCOCK & WILCOX COMPANY (B&W), Lynchburg, Virginia

Generic Probabilistic Risk Assessment for Pressurized Thermal Shock: This study identified potential pressurized thermal shock transients and assessed the frequency and severity of the sequences. Responsibilities included construction of event trees, quantification of event tree header probabilities, especially events related to human actions, quantification of event tree sequences.

Reliability and availability analyses for various nuclear power plant fluid and electrical systems, including Auxiliary Feedwater System using generic data, and considering operator actions and common mode failure, and digital control system and Safety Parameter Display System (SPDS) using MILHDBK-217D stress calculations coupled with fault tree or reliability block diagram modeling.

Extension of Test Intervals for the B&W Reactor Trip System: The project seeks the relaxation of technical specification surveillance requirements to increase reliability and safety and reduce manhour expenditures and spurious trips. The effects of surveillance testing on system reliability/

availability were determined with an RBD model that included factors such as common mode failure, human error, and wearout.

Probabilistic Risk Assessment for an Advanced Light Water (ALWR): The PRA investigated the core melt frequency for the ALWR conceptual reactor design, and evaluated the impact of the safety systems. The PRA was integrated with the reactor design process to enhance both safety and reliability.

IDCOR (Industry, Degraded Core Rulemaking) Individual Plant Evaluation: Worked with Duke Power Company to evaluate and implement the IPE to the Oconee 3 plant. The IPE is intended to be a "cookbook" core melt/plant evaluation PRA methodology. Attended AIF-sponsored workshop on the IPE.

Development of Engineering Workstation: Conceptual, planning, and implementation activities directed toward the development of the Integrated Reliability Interactive System (IRIS). IRIS consists of pre- and post-processors to analytical codes that quantify fault trees, event trees and reliability block diagrams. Also conceptual, planning and implementation activities directed toward development of generic software to be used by IRIS and other workstations. Other responsibilities include program testing, coordination and documentation, and participation in the marketing effort.

Consultant to Brookhaven National Laboratory: Presented methodology for evaluating fire protection systems in nuclear power plants for possible inclusion in work being performed for the NRC.

Development of interactive menu-driven software to provide an user interface for the initiation, control, and calibration of a real-time Data Acquisition System for the N-Reactor (Hanford, Washington). Software development including writing of requirements specification and design description in accordance with a Verification and Validation Program. Other responsibilities included software testing, interfacing with QA, and preparation of user's and programmer's documentation.

PROFESSIONAL AFFILIATIONS

National Society of Professional Engineers
Society for Risk Analysis
American Nuclear Society
B&W representative on McDermott AI/Expert Systems Committee
Intern Engineer (New York State)

SECURITY CLEARANCE

DOE-Secret Security Clearance.
DOD-L Security Clearance.

PUBLICATIONS

"Feedwater Heater Life Cycle Advisor: An Expert System Application at a Nuclear Power Plant," Author, presented at the EPRI-sponsored Expert System Applications for the Electric Power Industry Conference, Orlando, Florida, June 5-8, 1989.

"Probabilistic Analysis for the Babcock & Wilcox Advanced Light Water Reactor", Co-author, presented at American Nuclear Society International Topical Meeting on Safety of Next Generation Power Reactors, Seattle, Washington, May 1-5, 1988.

"A RAM Evaluation Tool: Combining Abridged - PRA with Root-Cause Analysis," Co-author, to be presented at the 14th INTER-RAM Annual Conference on Reliability of Electric Power Industry, Toronto Canada, May 1987.

"Use of an Engineering Workstation in RAM Analysis," Author presented at the 12th Inter-RAM Annual Conference on Reliability for Electric Power Industry, Baltimore, MD, April 1985. Reprinted, in part, in Nuclear Engineering International, September 1985.

"Pressurized Thermal Shock: An Evaluation for Plants with B&W NSS," Co-author, presented at the ANS 1984 Annual Meeting, New Orleans, LA, June 1984.

"Integrated Reliability Interactive System," Author, presented at the 11th INTER-RAM Annual Conference on Reliability for Electric Power Industry, Las Vegas, Nevada, April 1984.

"Methodology to Evaluate the Effectiveness of Fire Protection Systems in Nuclear Power Plants," Author, Nuclear Engineering and Design, Volume 76, No. 2, November 1983.

"Comparison of the Monte Carlo and Systems Method for Uncertainty Analysis," Co-author, presented at ANS/ENS International Meeting on PRA, Port Chester, NY, September 1981.

"Neutron Spectra Measurement Upon a Spherical Assembly of Thoria," Co-author, Nuclear Cross Sections for Technology, Proceedings of the International Conference on Nuclear Cross Sections for Technology, University of Tennessee, Knoxville, October 22-26, 1979.

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Resume

JOHN T. MAYER

EDUCATION	B.S. - Physics, St. Thomas College, 1962 M.S. - Physics, Case Institute of Technology, 1966 M.E.A. - Engineering Administration, George Washington University, 1980
EXPERIENCE	
1979 - Present	BABCOCK & WILCOX, Nuclear Materials Section, R&D Division, Lynchburg Research Center, Lynchburg, VA Supervisor of the Post Irradiation Examination (PIE) group. Responsible for directing PIE operations at reactor sites as well as in the hot cell facility of the Lynchburg Research Center. Responsibility also includes operation and maintenance of HLW and spent fuel shipping casks and operation and maintenance of several peripheral laboratories and shop areas. The PIE group currently includes four engineers and five technicians. Projects are typically carried through from the proposal preparation stage to the final report.
1973 - 1979	BABCOCK & WILCOX, Principal Engineer, Fuel Engineering Section, Nuclear Power Division, Lynchburg, VA Project leader for all fuel-related PIE programs with the R&D Division. These included poolside and hot cell examination programs, design of tooling to support PIE activities, and assessment of fuel performance based on operating reactor data.
1962 - 1973	NASA LEWIS RESEARCH CENTER, Cleveland, OH Nuclear Engineer - worked on several projects associated with space nuclear power and propulsion systems. These included experimental studies on radiation effects to materials and components, design of irradiation experiments and high-temperature test cells, and analytic studies on fuel behavior.

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JOHN T. MAYER

SECURITY CLEARANCE	DOE-L, 2/18/83
PROPOSED LABOR CATEGORY	Engineer (Group Supervisor, R&D)
MOST SIGNIFICANT TECHNICAL ACCOMPLISHMENT IN LAST FIVE YEARS	Coordination/supervision of efforts related to large DOE nuclear fuel research contracts.
MOST SALIENT SKILL THAT RELATES TO THE PROPOSED EFFORT	Reviews project goals, hardware, and procedures to insure human factors planning has been incorporated to reduce radiation exposure, reduce safety risks, and maximize efficiency.

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Resume

DALLAS T. SCOTT

EXPERIENCE

BABCOCK & WILCOX

7/87 to PRESENT

Sacramento Municipal Utility District (SMUD) DEDICATED TEAM assignment. Control Room console modifications team member responsible for all console labels, equipment data sheets and Master Equipment List (MEL) revisions. SMUD Safety Parameter Display System (SPDS) Engineering Requirements design engineer.

Florida Power Corporation (FPC) DEDICATED TEAM assignment. Responsible for FPC SPDS NRC audit. FPC SPDS engineering requirements design engineer.

Pacific Gas & Electric (PG&E) SPDS engineering requirments design engineer.

Transient Assessment Program Transient Investigation team assignment. Respon-sible for the preparation of a draft report identifying the root cause and plant response to an upset or reactor trip situation.

2/83 to 7/87

Sacramento Municipal Utility District (SMUD) STAFF AUGMENTATION assignment. Human Factors team leader responsible for update of control room design review analysis. I&C Supervisor's assistant responsible for staff baseload assign-ments and tracking. Assistant Principal Investigator SMUD control room design review responsible for overseeing pro-gram plan and activities of design review team. Interfaced with NRC on questions dealing with SPDS, CRDR, R.G. 1.97, and NUREG-0737. Secretary Emergency Response Steering Committee, member CRDR group, R.G. 1.97 group, Computer Display group, Emergency Faci-lities group. Drafted SMUD response to NRC on Supplement 1 to NUREG-0737.

Babcock & Wilcox
a McDermott company

DALLAS SCOTT
(Continued)

8/82 to 1/83

Consumers Power Company STAFF AUGMENTATION assignment. Technical Support Group member. Drafted admini-strative procedures for Midland site.

7/80 to 8/82

Plant Performance Engineering, Transient assessment Program team member. Traveled to sites (Davis-Besse, Crystal River, Oconee, Rancho Seco)to analyze reactor transients for correct response and prepare draft of utility reports. Compiled semi-annual summary reports/ analysis of operating history.

Safety Parameter Display System Design. Drafted the functional specifications for Rancho Seco, Midland I and Midland II. Assisted in the functional specification for Diablo Canyon (Pacific Gas and Electric, a Westinghouse NSS).

NRC Guidance Evaluations. Prepared evaluations of NUREG-0696 and R.G. 1.97 for Tennessee Valley Authority, Florida Power Corporation, Supply System, Public Service Electric and Gas (New Jersey, a Westinghouse NSS).

ARGONNE
NATIONAL
LABRATORY

7/78 to 7/80
Operator

Experimental Breeder Reactor II-Reactor

U.S. NAVY

6/69 to 7/78

Qualifications:

Engineering Watch Supervisor,A1W
Staff Instructor, A1W Leading Training
Instructor, A1W
Shutdown Reactor Operater, USS Long Beach
(CGN 9)
Electrical Operator, USS Long Beach
(CGN 9)

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DALLAS T. SCOTT
(Continued)

EDUCATION

American River College (24 credits)
Central Virginia Community College (45 credits)
Technical Writing Courses, B & W
Statistical Analysis, University of Kentucky

PROFESSIONAL ASSOCIATIONS

Associate Member, Human Factors Society

Babcock & Wilcox
a McDermott company

Resume

ROBERT L. STARKEY
PRINCIPAL ENGINEER

EDUCATION

1965 B.S., Naval Science, United States Naval Academy

1978 M.S., Industrial Engineering-Ergonomics (Human Factors), North Carolina State University

Candidate (in absentia) Ph.D., Industrial Engineering-Ergonomics (Human Factors), North Carolina State University

MAJOR SERVICE SCHOOLS

1967 - Naval Nuclear Propulsion Officer's Course - Mare Island

1968 - S-5-W Reactor Prototype Qualification, NPTU, Idaho Falls

1968 - Submarine School Officer's Course - New London

1971 - Poseidon Strategic Missile Weapons Officer's Course

WORK EXPERIENCE

1980 - Present

Babcock & Wilcox Company, Utility Power Generation Division, Lynchburg, Virginia

Principal Engineer - Plant Engineering Services Ergonomist/Human Factors Engineer

Control Room Design Review (1981 - Present)

Rancho Seco (Sacramento Municipal Utility District (SMUD) (1984 -1985)

Senior Human Factors Engineer on Rancho Seco Control Room Design Review Team. Conducted interviews; task analysis of operator actions and required controls and instrumentation to implement Emergency Operating Procedures. Assessed human engineering suitability of controls and displays; participated in validation of EOPs at simulator and static mockup; conducted component-level human engineering surveys of control room equipment. Evaluated Human Engineering Observations identified from human factors and operational significance viewpoint, recommended solutions.

ROBERT L. STARKEY
PRINCIPAL ENGINEER
(Continued)

WNP-1 (Washington Public Power Supply System) - 1981-1982

Principal investigator, participated in all, lead in five of the nine phases of this Preliminary Control Room Design Assessment. Eighteen month program which included static survey, mock-up construction, structured interviews, task analysis - walk-through/talk-throughs of operating procedures, and preparation of a detailed equipment summary. Principal author of final report submitted to NRC. Developed layout of several panels added or revised as a result of this program.

Midland 1 & 2 (Consumers Power Company) - 1982

Technical Consultant - assisted utility and its human factors consultants while conducting walk-through/talk through-task analysis of procedures at plant control room mock-up.

Pilot Studies (UPGD) - 1982-1983

Proved methodologies for conducting systems analysis/task analysis. Developed hardware needs and associated human factors criteria for controls and displays to implement B&W's Abnormal Transient Operating Guidelines (ATOG) based procedures and a task analysis methodology based upon power plant annunciator reviews.

<u>Safety Parameter Display Systems</u> - 1982

Diablo Canyon 1 & 2 (Pacific Gas & Electric Company) Provided human factors analysis report on display formats.

<u>Systems Design</u> - 1982-1986

Rancho Seco (Sacramento Municipal Utility District) - 1984-1986

Provided human factors input in selection of control room panel-mounted equipment to implement Emergency Feedwater Initiation & Control (EFIC) system. Developed panel layout.

Crystal River 3 (Florida Power Company) - 1982

Provided human factors input into selection and layout of control room panel-mounted equipment to implement Emergency Feedwater Initiation and Control (EFIC) system. Provided layouts, conducted task analysis, and trained operators on system.

Babcock & Wilcox
a McDermott company

ROBERT L. STARKEY
PRINCIPAL ENGINEER
(Continued)

Panel Engineering - 1980 - Present

Rancho Seco (Sacramento Municipal Utility District) - 1986

Participated on engineering team implementing panel modifications developed during Control Room Design Review.

WNP-1 (Washington Public Power Supply System) - 1980 - Present

Assisted in design, manufacture, and shipping of control room panels. Drafted and performed functional system checkout of control room panels and panel mounted equipment.

Procedures Preparation - 1980 - Present

Technical Consultant for B&W Abnormal Transient Operating Guidelines (ATOG) program. Topics included technical feasibility of procedures from human factors engineering/hardware considerations and usability of product from readability, clarity, user-friendly format aspects.

5-1980 North Carolina State University, Raleigh, N.C.

Graduate student in Ergonomics/Human Factors Engineering Program through Industrial Engineering and Psychology Departments. Taught course in Ergonomics to IE senior undergraduates. Consulted with an electronics firm developing a selection program for industrial inspectors. Served on internship with Human Factors/Air Crews Systems Division at Naval Air Development Center, Warminster, PA developing helicopter night vision systems.

1965-1975 United States Navy-Line Officer, Submarine Force

Various operational staffs and on diesel, nuclear fast attack and FBM (POSEIDON) submarines. Department head tours as supply officer, weapons officer, and various engineering division officer billets. One tour with shipyard activity staff. Qualified in submarines, nuclear power Engineering

CASK/HOT CELL SEALING INTERFACE

1. SUMMARY

This document presents recommendations for the sealing interface between the BR-100 cask and a hot cell facility for loading and unloading operations. This is response to the requirements in the Statement of Work (SOW), Appendix C, Section 18.B.

> " Casks should have a surface capable of being sealed to a hot cell enclosure for loading/unloading operations. A study shall be performed and a recommendation made by the Contractor on the location and operation of this surface during the preliminary design phase."

2. DISCUSSION

Each cask contractor is required to make a recommendation for the cask-to-hot cell interface and incorporate provisions in the cask design to accommodate a interface seal.

The recommended method for effectively sealing the BR-100 cask to the hot cell is the use of an expandable seal that mates with the upper flange region of the cask. Attached to the hot cell interface, the seal would expand in the radial direction to close the clearance gap between the hot cell and cask. A sealing arrangement of this type, such as an inflatable O-ring, can accommodate relatively large alignment tolerances between the cask and the hot cell interface. This configuration also permits access to the entire upper surface of the cask from inside the hot cell.

To accommodate this type of seal, the BR-100 has incorporated a machined diameter on the cask upper flange as shown in Figure 1. This provides a smooth sealing surface for the seal.

3. LOADING/UNLOADING OPERATIONAL SEQUENCE

The following is the proposed operational sequence for cask unloading in a dry hot cell facility. The hot cell would be located above the cask receiving and handling area. After the cask is positioned under the hot cell, an extension device could be lowered or the cask lifted to the required elevation to align the cask's sealing surface with the expanding seal. Once the seal between the hot cell and cask has been verified, the closure lid and shield plug can be removed and unloading operation started. Cask loading operations would follow a similar sequence in reverse order.

1. Off-load cask and skid from the railcar and process through in-coming inspection. Key interface dimensions of the cask (with impact limiters attached) are shown in Figure 2.

2. Remove impact limiters.

3. Upright and remove cask from the skid and place the cask on the transporter in a upright orientation. Figure 3.

4. Vent and cool cask interior.

5. Loosen closure bolts. Selected bolts are disengaged from the cask flange.

6. Position cask in a vertical orientation directly below hot cell interface.

7. Align cask vertically with interface seal.

8. Expand ring seal against cask outer diameter in cask upper flange region. Figure 4.

9. Verify seal.

10. Disengage all closure bolts from the cask flange.

11. Remove closure lid and shield plug.

12. Unload fuel.

FIGURE 1

Upper Flange Region

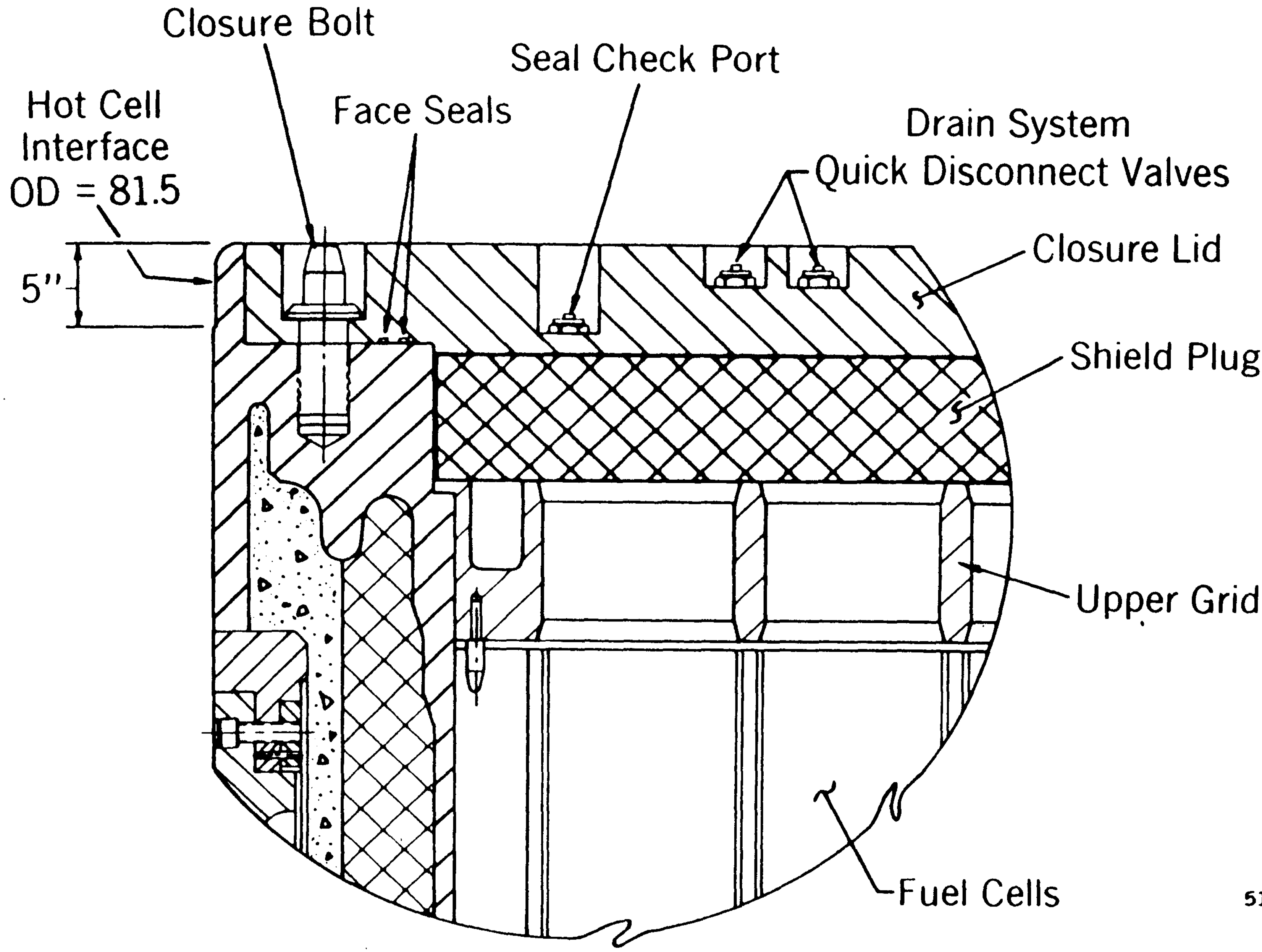


FIGURE 2

BR-100 Cask Package

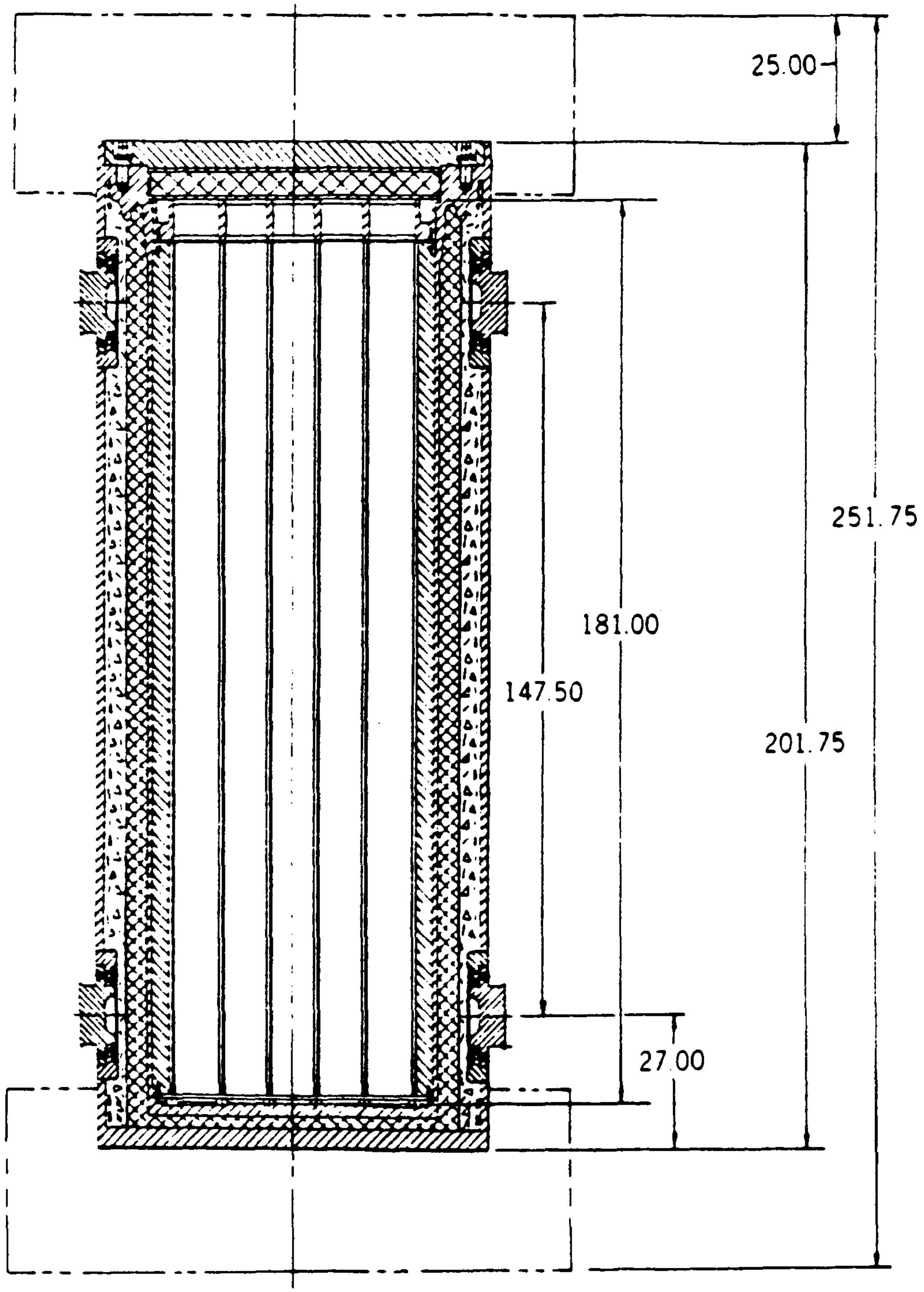


FIGURE 3

BR-100 Cask Longitudinal Section

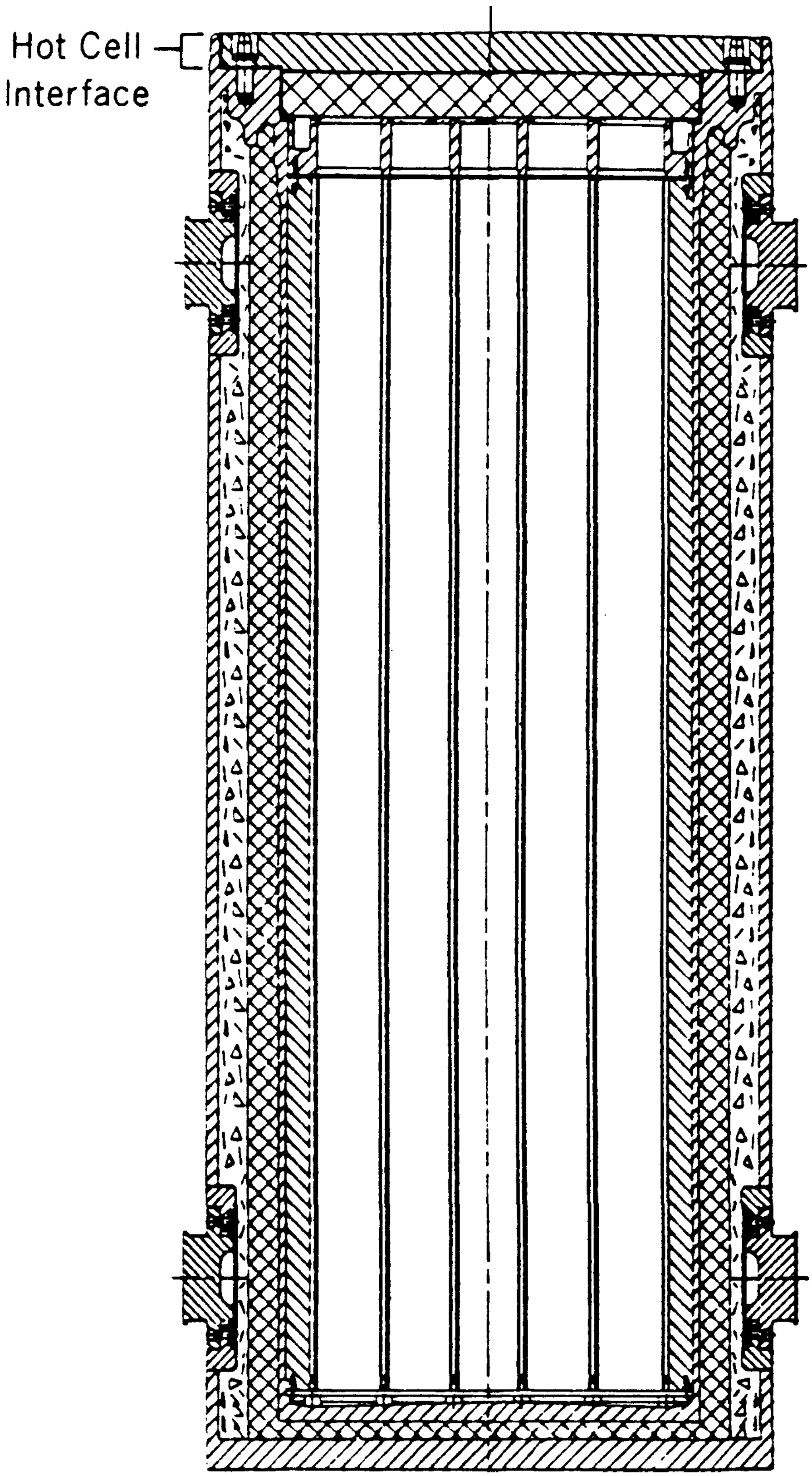
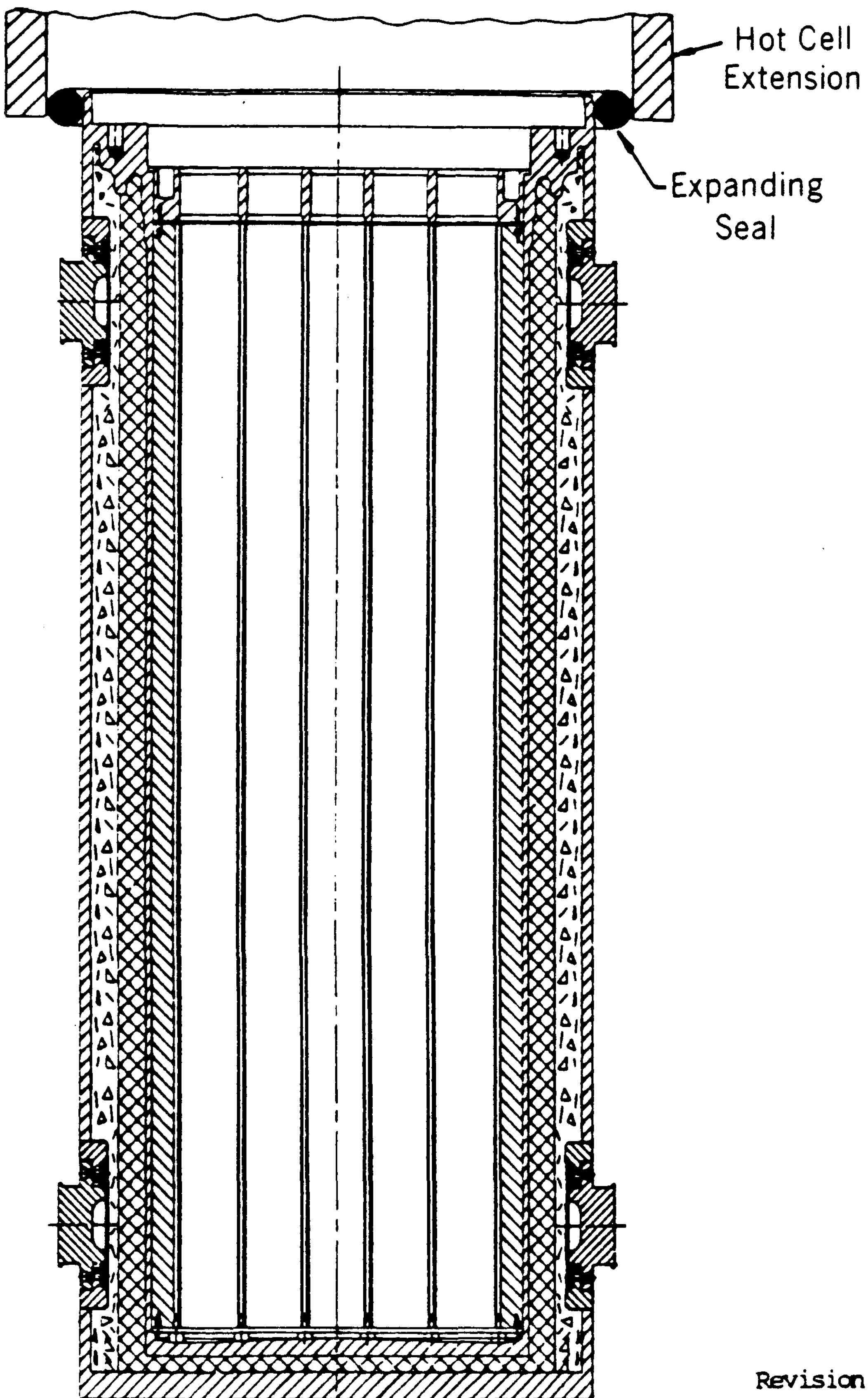


FIGURE 4

BR-100 Cask – Unloading Configuration



Revision 01
51-1176045-01

B&W FUEL COMPANY

CONTRACT NO. DE-AC07-88ID12701

BR-100
CASK SYSTEM

PRELIMINARY DESIGN REVIEW RESOLUTION DOCUMENT

51-1177263-02

APRIL, 1990

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Larry Hassler DATE 04/11/90
LARRY HASSLER, PHYSICS

John Jones DATE 04/11/90
JOHN JONES, THERMAL

REVIEWED BY: Daniel Young DATE 4/11/90
DANIEL YOUNG, PE

APPROVED BY: Danny J. Sanchette DATE 4/11/90
DOMINIQUE SANCHETTE, APM

Record of Revisions

Revision #	Description	Date
00	Original Release	12-08-89
01	Revised to incorporate DOE/EG&G comments	02-20-90
02	Updated Responses A15, E2, E26, E39, E40, E44, E54, E55	04-09-90

This document has been prepared in response to questions and comments posed by the Technical Review Group (TRG) during its review of the BR-100 Preliminary Design held November 14, 1989. The TRG comments/questions and the corresponding BWFC responses are keyed to the individual TRG members as listed below with his/her area of expertise and parent organization:

A - Verbal Comments from TRG Meeting
B - Richard Baehr, Quality Assurance, Sandia National Laboratories
C - Larry Blackburn, Materials and Fabricability, Westinghouse Hanford Company
D - Tom Burr, Structural, Idaho National Engineering Laboratory
E - J. E. Ratledge, Transportation System Interfaces, Oak Ridge National Laboratory; Larry Danese, Cask Handling, SAIC, Oak Ridge, TN (for ORNL)
F - John Friley, Thermal, Battelle Pacific Northwest Laboratory
G - M. M. Madsen, NRC Certification, Sandia National Laboratories
H - Joe Pace, Shielding, Oak Ridge National Laboratory
I - Bryce Rich, Safety/ALARA, Idaho National Engineering Laboratory
J - C. A. Rogers, Criticality, Westinghouse Hanford Company

Sections B through J reflect comments from the TRG that were submitted before the Design Review meeting. Section A contains comments and questions that arose during the meeting.

In the following sections, the question/comments and their corresponding responses are listed in order. Some questions or comments have been paraphrased to improve readability.

A.1 Are thermal or radiation effects most limiting in the design?

Response - The most limiting effect on the BR-100 design, for baseline fuel, is radiation. The shielding required to get the dose rate below Federal requirements for the maximum source strength is the limiting feature in determining the BR-100's capacity. Thermal effects can be bounding for certain burnup/age combinations that are beyond baseline requirements, but usually dose rate is the dominant factor.

A.2 Is the stress created from lead shrinkage accounted for?

Response - The shrinkage of lead after it is poured into the annulus between the inner shell and an artificial outer shell will result in a relatively small compressive residual stress in the inner shell. The BR-100 fabrication technique will allow no gap between the inner shell and the lead, while the later concrete pour will leave no gaps between the lead and the concrete or the concrete and outer shell.

A.3 Is the exterior cask coating a commercial grade paint?

Response - The currently anticipated exterior cask coating is a commercially available paint. White Amercoat 90, available from Ameron, has been selected because of its thermal emissivity characteristics, hardness, durability, and widespread use in nuclear applications.

A.4 What is the fire accident performance of Kevlar?

Response - Kevlar will burn when exposed to a high enough temperature in the presence of an abundant supply of oxygen. The BR-100 impact limiter design will preclude this by providing a non-combustible barrier to oxygen, exterior to the Kevlar.

A.5 Are the aluminum materials properties taken from Class III or Class I of the ASME B&PV Code? What about fatigue effects?

Response - The aluminum 6061 allowable stress levels provided for BR-100 basket design are taken from ASME Code tables designated for Class 3 components. Fatigue effects will be evaluated during final design.

A.6 How will you ensure that there is no galling of the head closure bolts?

Response - To ensure there are no galling problems associated with the lid closure bolts, a lubricant will be used. The lubricant will be identified during final design; a low-friction, polymeric coating is currently being investigated.

A.7 Why are you using ASME Section III in some places and Section VIII, UG-28 in others? What is the rationale for picking and choosing different sections?

<u>Response</u> - The BR-100 cask is not an ASME Code component. The ASME sections are used as a design guide. Originally it was planned that ASME Section III would be used for the cask inner shell and ASME Section VIII for the cask outer shell analyses, but we have now decided to use Section III for both. The buckling requirements in Section III (Subsection NB-3133.3) and Section VIII (Subsection UG-28) are essentially the same, so to prevent any confusion, the reference to Section VIII will be replaced with Section III.

A.8 What is the location of the primary local stresses?

<u>Response</u> - Primary local membrane stresses are defined as stresses caused by mechanical loads at discontinuities. For the cask shell, the local membrane stresses are induced at the intersection of the cylinder and bottom plate, around the trunnions, and at ring flange locations. Those stresses will be calculated in the final design by finite element analyses.

A.9 What is the location of the flange welds?

<u>Response</u> - The ring flange will be welded to the cask at the cylindrical intersection with a full penetration weld. The weld locations will be identified on the drawings in the final design.

A.10 What about welding the dissimilar materials (304L to XM-19)?

<u>Response</u> - We do not anticipate any difficulty welding 304L stainless steel to XM-19 (trade name Nitronic 50) stainless steel. Both metals are austenitic and have very similar properties. E308L filler metal will be used for the welding.

A.11 How will you assess the effects of a "pin drop accident"?

<u>Response</u> - The cask shell and closure lid have been evaluated for the "pin drop accident." The seal check ports and drain system valve ports on the closure lid are protected by the cover plates. The cover plates will be sized in the final design for protection during a hypothetical "pin drop accident."

A.12 What is the torque value of the closure lid bolts? Isn't this quite high? I recommend using a lower torque value when unloaded?

<u>Response</u> - The torque value for closure lid bolts is 80,000 in-lb (or 6667 ft-lb). Our experience shows that this bolt

torque can be achieved with commercially available equipment. This value also is based on very conservative analytical techniques. In the final design, we will remove excess conservatism and probably lower the torque. We agree with the recommendation of using a lower torque value for shipment of the unloaded cask.

A.13 What loads were used (for $2S_m$, for example)?

Response - The 2Sm stress allowable limit applies to closure bolt primary membrane stress under normal conditions of transport. The loading conditions for normal condition of transport are defined in Section 2.6.

A.14 The next question addressed some of Tom Burr's written comments. These were discussed with overhead slides. Their resolution will be addressed in the Written Comments.

Response - This is a statement and no response is required.

A.15 How will you handle material properties generated by testing vs. those tabulated in the literature (specifically addressed to annealed 6061 aluminum)?

Response - The material properties for the stress analysis of aluminum 6061 will be obtained from the ASME Code. Since the fuel cell will be cold-worked and anodized, we will investigate that the minimum strengths reported in ASME codes are still applicable for our design.

A.16 Are the effects of the partial atmosphere of helium gas included in the thermal analysis?

Response - PDR Reference 3.8 serves as the basis for the helium thermal conductivity and thermal diffusivity values used in our analyses. This reference states that these properties are not a strong function of pressure. Thus, the partial atmosphere has only a small impact on these properties.

A.17 What is the basket gap size?

Response - The basket gap size used in the thermal evaluation is 0.050 inch. This represents the maximum expected diametral gap under normal conditions.

A.18 What is the uncertainty of "below limit" temperatures?

Response - The uncertainty on the "below limit" temperatures cannot be easily quantified, although the values are conservative due to the inputs. Among the conservative inputs are the use of the maximum expected diametral gap as a radial gap, neglecting thermal radiation and natural convection within the cells and water gaps in the basket, use of conservative thermal loads based on the criticality calculations, use of uniform heat fluxes on the cell inner surfaces, and using an air-based correlation to predict the peak cladding temperatures in the BR-100 helium-filled environment.

A.19 What is the function of the Inconel sides on the test section?

Response - The Inconel sides serve as steam containment and test section structural members that help hold the test section together. The sides are thin and constructed of Inconel, which has a relatively low thermal conductivity, to maintain the one-dimensional heat flow through the test section and prevent thermal shunting around the test section.

A.20 Will the heaters be left on during the fire test?

Response - The test section heaters will be adjusted during the fire test so that the combined effect of the heat leakage through the test section insulation and the energy generated by the test section heaters simulate the spent fuel heat generation within the cask.

A.21 How have you modeled the convection coefficient as a function of radial position around the cask?

Response - The convection coefficient on the exterior surface of the cask has been assumed to be constant. During normal operation there is expected to be little circumferential variation of the conditions around the cask, due to both convection and thermal radiation. This is not the case for the hypothetical fire accident. This variation will be examined in the final design.

A.22 Have you analyzed a basket model without water ellipsoids?

Response - An explicit analysis using the model developed for the preliminary design has not been executed without the water ellipsoids. However, during the iterations to arrive at the preliminary design, analyses were made both with an aluminum former with water gaps adjacent to the fuel region and with a solid aluminum former. The use of the ellipsoidal regions caused a reduction in the k_{eff} of the cask. Physically these regions provide thermalization for neutrons scattered from the regions external to the basket, i.e. the aluminum former, the stainless steel shell, and the lead shield. Such thermalization enhances the probability of absorption in the outer cermet plates. This reduces the neutrons back-scattered into the fuel region which causes the reduction in $k_{eff.}$ During the early phase of the final design, the effect of these gaps will be explicitly assessed if they remain a part of the design.

A.23 10CFR71 states that the dose rate calculation be taken 2 meters from the vehicle edge.

Response - Dose rate calculations for the limiting design cases were performed for points at two meters from the personnel barrier. This distance is slightly less than the two meters from the vertical planes projected from the outer edges of the conveyance specified in 10CFR71. This assumption produces a slight conservatism in the calculated dose rates.

A.24 Shielding in the area near the closure lid needs to be re-examined. The shielding analysis should be done on the puncture models.

Response - The area near the closure head will be examined in detail in the final design stage with discrete geometry models in both DOT 4.3 and QAD. Any area found to exhibit dose rates beyond the design limit will be modified to bring the dose rates within required limits.

Consideration will be given to performing Co-60 gamma flux attenuation measurements using the puncture models.

A.25 With respect to the 1D analysis, how were the copper fins modeled? Is streaming a concern?

Response - In the 1D analysis the copper fins were homogenized within the concrete region. In reality the copper fins occupy separate layered regions within the concrete. This poses no problem for gamma radiation since the copper attenuates gammas better than the concrete. The question then is whether or not neutrons find a streaming path through the fins. Neutrons do not have as high a probability of a scattering collision in copper as in concrete due to the hydrogen in the latter, but it is unlikely that neutrons will find an uncollided path along the entire length of a fin. More likely, collisions will occur in the copper and the neutrons will be diverted into the surrounding concrete. However, since there is a possibility of an enhancement of neutron dose rates due to the copper fins, DOT 4.3 will be used to model a section of concrete and fins in the final design to determine the magnitude of the effect, if any.

A.26 Will ALARA analyses be performed without the closure lid?

Response - ALARA analyses will be performed without the closure lid when final handling procedures are formalized.

A.27 What is the extent of streaming effects in the region of the closure lid where there is no lead, only steel?

Response - Both the top and bottom heads of the cask contain lead (3.5" top, 3.0" bottom). The lead in the bottom head is continuous with the cask sidewalls. The lead in the top head lacks only a small ligament of meeting the cask sidewall lead. This steel ligament is on the cylindrical corner of the cask where the greater slant penetration of steel is worth as much gamma shielding as the straight-through penetration of lead. Final design analyses will verify that this is the case or shield modifications will be made.

A.28 What the effects of the standoffs (to handle shorter fuel) vs. CE's long fuel rods?

Response - Standoffs were considered in the shielding analyses. They have little effect on the design dose rates. The main effect is on the location of the highest dose rate from the axial positioning of the lower end fittings. This has been accounted for by the length of the additional lead in the lower cask region.

A.29 The next question addressed some of Joe Pace's written comments. Their resolution will be addressed in the Written Comments appendix.

Response - Covered in response to Mr. Pace's comments (see responses in "H").

A.30 What Quality Factor is used?

Response - Nominal quality factors of 10 for fast neutrons and 3 for thermal neutrons were used in all BR-100 shielding calculations.

A.31 I suggest that you quantify the margins of safety (for those values close to 10).

Response - All modelling and assumptions used in the calculation of the BR-100 dose rates were consistently taken on the conservative side, including the use of the highest source strength fuel within the contract statement-of-work. In many instances these conservatisms compound. When computer code uncertainty is coupled with the conservatisms, it is not possible to predict with a degree of accuracy the numerical value of the deviation from absolute that should be assigned to the results derived. It is believed that the dose rate values reported are nearer the maximum than the average values. If an uncertainty were assigned to individual dose rates based on engineering judgement, it would be +15%, - 35%.

To compare the PDR dose rates to those from more normal source strengths, the midplane value for 3.5 wt % U-235 fuel is 7.2 mr/hr (5.1 gamma, 2.1 neutron) instead of the 9.6 shown for 3.0 wt % U-235 fuel.

A.32 Is the margin of uncertainty known?

Response - See response to A.31.

A.33 What the are maximum vs. average values with $\pm$ 2 ?

Response - See response to A.31.

A.34 What are the wear characteristics (with respect to abrasion) of Kevlar?

Response - Kevlar has good abrasive wear characteristics. The actual wear characteristics of the impact limiter with the cask system will be investigated when the final impact limiter design is completed.

A.35 I suggest that you get rid of the 2 attachments on the top and bottom. They are difficult to get to.

Response - The impact limiter must be designed to accommodate high slapdown attachment forces. To remove four attachments (two top and bottom) would significantly degrade the impact limiter's ability to remain on the cask during a slapdown drop. However, provisions will be made in the attachment final design to ensure the easiest installation achievable.

A.36 Have you looked at the time phasing of the crush pressure on the closure lid for the "end-on-drop" accident scenario? The inertial effects may be significantly altered by the dynamic time history.

Response - The dynamic effect of the impact limiter on the closure lid will be investigated after the impact limiter design is finalized.

A.37 What is the coating and color of the Kevlar overcoating? Does this protect the Kevlar from abrasion and UV degradation. What is expected lifetime?

Response - The Kevlar overcoating will be determined during the impact limiter final design phase. Abrasion and UV protection will be implemented in the design of the impact limiter overcoating. The Kevlar composite is expected to exceed the 25-year design lifetime of the cask.

A.38 Are there fabrication difficulties with the various angular ply orientations you are proposing? What testing is planned?

Response - No unusual fabrication difficulties have been encountered with ply orientations. The impact limiter testing program is detailed in Section 2.10.2, page II-2-68.

A.39 How do you plan to compensate for the NRC mandated 75% credit of B^{10} in the cermet neutron absorber?

Response - We anticipate that the NRC will not allow us to take full credit for the B^{10} loading in the Aluminum/B_4C neutron absorber plate. Therefore, design refinements relating to enriched B_4C, plate thickness, attachment mechanism, plate location, or a combination of these will be used to accommodate B^{10} loading requirements.

A.40 The test rate of 1/4 of the expected strain rate seems non-conservative.

Response - The cermet plate test rate of 1/4 of the expected strain rate is not necessarily nonconservative. Typically, straining capacity of a material increases with increasing strain rate (over a specific strain rate range, depending on the material).

A.41 Data derived from materials properties testing needs to have statistically valid numbers of tests to be valid for design purposes, especially with less-than-ductile materials. You also need to look at the combination of temperature and rate dependence effects on the material properties.

Response - B&W agrees with this comment relative to testing of the cermet plate. The focus on statistically significant numbers of tests will be important once all manufacturing parameters and procedures have been established.

A.42 How is impact limiter safety in an accident ensured with respect to skid attachment options?

Response - Several accident scenarios are being reviewed to determine the tie-down strategy that will assure the impact limiter stays on the cask for all conditions. This will be analyzed in the final design.

A.43 Have you looked at "depressed center" railcars in order to reduce the CG height?

Response - We are presently looking at a flat bed railcar with options for a "depressed center" and a "Schnable" railcar. The "Schnable" railcar incorporates the cask skid as part of the railcar center section; the advantage would be weight reduction.

A.44 I suggest that you eliminate the void behind the trunnion.

Response - A silicone seal will be used on the trunnion and on bolt heads to prevent interstitial water accumulation.

A.45 How will you achieve (and verify) the large torques on the various bolts?

Response - Hydraulic wrench systems are commercially available which torque up to 75,000 ft-lbs. A system with a torque range of 570 to 5675 ft-lbs weighs 32 lbs. A similar system with a range of 1845 to 18,450 ft-lbs weighs 64 lbs. These commercially available systems torque to preset values. Torques can be verified with an ultrasonic bolt measurement (elongation) system which works in conjunction with the hydraulic wrench.

A.46 6,000 ft/lbf of torque seem excessive for the closure lid bolts.

Response - This subject is covered in response A.12. The comment is noted and will be investigated in the final design.

A.47 Is the lifting equipment shipped on the railcar with the cask?

Response - It is intended to ship the lifting equipment on the railcar with the cask. This will be dependent on the existing weight restrictions of the GVW and will be determined in the final design.

A.48 Has there been any attempt to feed Failure Modes and Effects Analysis (FMEA) back into the design?

Response - B&W performed a FMEA during the Preliminary Analysis and it is attached to the PDR as Appendix 6. The FMEA will be used early in the Final Design phase and a more rigorous FMEA will be performed near the end of that phase for the cask and its associated tooling.

A.49 What considerations have been taken with respect to personnel platform ease of access?

Response - The personnel platform is not a part of the scope of work. Therefore, no consideration has been given to this equipment.

A.50 This comment referred B&W to the written comments on cask operations already submitted. These comments will be addressed in the Written Comments.

Response - These comments are addressed in responses to E1 to E57.

A.51 How are the containment sealing aspects measured? Leak tightness vs. source term?

Response - The BR-100 has check ports to test the leaktightness of the seals on the closure lid. There are no penetrations into the cavity other than through the lid, so all leak-check operations are accessed from the top of the cask. We anticipate using a helium leak check or a pressure drop method before each shipment to ensure the performance of the seals. We plan to use the source term method to determine leaktightness acceptability.

A.52 A suggestion to have the A2's annual inspection criteria be 10^{-6} to 10^{-7} std cc/sec, while in normal use (shipping) 10^{-3} std cc/sec would suffice.

Response - That comment is consistent with the strategy outlined in the response for A.51 and will be implemented.

B.1 The BR-100 Preliminary Design Report references approximately 22 computer codes. What is the software quality assurance plan?

Response - The B&W Fuel Company has very stringent requirements for certification of software used for licensing analyses; those requirements are listed in QA procedures provided for Mr. Baehr's review during the TRG meeting. Those requirements were not imposed for the analyses performed on the preliminary design, although other QA procedures were followed. Mr. Baehr indicated, after review of the documents supplied to him, that the software QA procedures to be used in the final design phase were satisfactory.

C.1 Provide more specific information on what property data for which alloys is obtained from the ASME Boiler and Pressure Vessel Code or from the Aerospace Structural Metals Handbook, or from some other source (p. II-2-22 and Table 3-2).

Response - The specific references to the data sources for mechanical and physical properties of each cask material can be found in the BR-100 materials handbook referenced in the text, BWFC Document 51-1174181. Data sources for Table 3-2 are now included as part of the table.

C.2 Provide a consistent reference for the spent fuel temperature limit. On p. II-2-61, mention is made of "work performed by Pacific Northwest Laboratory..." without any formal reference, while on p. II-3-4 a formal reference is made to Ref. 3.5, but that reference is not generally available in technical literature.

Response - The text has been modified to reference the spent fuel temperature limit.

The reference used in Chapter 3.0 for the spent fuel temperature limit has been changed to Ref. 3.13. Chapter 2.0 will use this same reference.

C.3 Correct the statement that 6061 aluminum alloy can be upgraded to a Class 1 material (p. II-2-7). Only the ASME Boiler and Pressure Vessel Code Committee can approve materials for Class 1 construction. There are significant differences in design methodology between Class 1 and Class 3 construction which are more important than consideration of nondestructive examination of materials. Design methodology of Section VIII, Division 2, is comparable to that of Section III, Class 1, and allowable stress intensity (S_m) values for 6061-T6 aluminum alloy are given in Section VIII, Division 2 which are higher than the allowable stress (S) values of Section III, Class 3. However, Section VIII, Division 2 does not yet contain design fatigue curves for aluminum alloys. The nature of the design methodology of Section VIII, Division 2 is such that it is important to conduct all the analyses specified, including fatigue.

Response - The BR-100 cask is not an ASME Code component. The ASME Code is used as a design guide since it is an acceptable code for nuclear power plant components. Aluminum alloy 6061 is an ASME-approved component material. ASME Section III currently defines it as a Class 3 component material. There are some differences in design methodology between Class 1 and Class 3 construction; among those differences are inspection requirements and consideration of fatigue. B&W will meet the requirements of ASME Code Class 1 component materials (including fabrication, examination, and testing), and will design aluminum 6061 structural components following Class 1 guidelines. Since the cask is not an ASME Code component, the ASME Code Committee's approval (ASME Section III, Article IV-1000) is not required.

We have identified to the NRC Transportation Branch (July 18, 1989 meeting) that we intend to upgrade a normally Class 3 material to Class 1 criteria for use in fuel cells. The NRC did not voice any special concern for using the alloy 6061 for this application.

C.4 Explain why the stress limits for threaded structural fasteners from NG-3230 should be applicable to closure lid bolts (Table 2-13, p. II-2-45). According to the requirements of NG-3231, Alloy 718 would not be an acceptable bolting material.

Response - The closure lid bolts are designed and analyzed per

ASME Section III, Subsection NB, requirements. The requirements of subsection NB are for pressure vessel components. Closure bolts not only experience pressure loadings, but also loading due to impact events. In some areas, the requirements of subsection NG-3231 are more stringent. B&W tried to meet both requirements. Since this was confusing to some readers, all references to subsection NG were removed from the text.

C.5 Correct the Section of the ASME Boiler and Pressure Vessel Code to be III rather than VIII on p. II-2-56.

Response - The text has been corrected to refer to ASME Section III in Section 2.7, paragraph 3.

C.6 Provide a reference to the fracture toughness of 6061 aluminum alloy (p. II-2-8). Explain whether additional fracture toughness testing and fatigue crack growth testing is required to support a fracture mechanics analysis.

Response - Fracture toughness data for aluminum 6061 are obtained from the Structural Alloys Handbook, Volume 3, 1988 edition, produced and published by Battelle's Columbus Division, Columbus, Ohio. This handbook provides fracture toughness data for 6061-T651 aluminum from -300°F to 70°F but none from the 77°F to 300°F range. We are evaluating whether or not fracture toughness testing will be required to obtain the fracture toughness data in the 77°F to 300°F range. No testing is planned for the fatigue crack growth.

C.7 Explain the reasons for experimental determination of mechanical properties of 6061 aluminum alloy (p. II-2-22) in preference to the use of properties listed in existing codes and handbooks.

Response - The experimental determination of tension and compression mechanical properties of 6061 aluminum alloy is expected to show large margins and may not be performed. Generation of these properties using specimens cut from actual fuel cells, fabricated from extruded tubes, machined and anodized in an identical manner to that planned for production fuel cells, will allow us to establish quantitative margins of safety on the structural behavior of the fuel cell during hypothetical drop accidents.

C.8 Provide additional information to clarify how thermal shields for impact limiters are to function (p. II-2-64).

Response - The PDR will be modified to include the following:

In concept A, the outer surface of the impact limiter is covered with a thermal shield. The shield material will be designed to withstand the hypothetical thermal accident condition. The shield will prevent impact limiter wood ignition by providing an oxygen barrier between the wood and the atmosphere. The impact limiter wood core will act as a thermal insulator to prevent the lid seals from exceeding their maximum operating temperature.

Concept B is similar to A, but the thermal shield is located inside the impact limiter instead of on the outer surface. In Concept B, the wood outside of the thermal shield is allowed to burn. The wood between the shield and closure lid is protected from ignition by the shield. The inside wood will be sized in thickness to ensure seal thermal protection. The advantage of concept B over A is weight savings due to less shield material.

C.9 Provide additional information on structural integrity objectives for aluminum clad B_4C/Al cermet and how original fabrication supports those objectives. Explain whether vibration and wear have been considered as potential damage modes for cladding.

Response - The structural integrity requirements of the B_4C cermet plates are that they remain in place for both normal and hypothetical accident conditions. If there is any permanent deformation, those displacements will be provided to physics calculations to ensure that a subcritical array of the spent fuel is maintained. In terms of allowable stress, we will maintain a limit of yield strength (Sy) for normal condition and limit of ultimate strength (Su) for hypothetical accident conditions.

Vibration and wear tests are in progress for the B_4C cermet plates for evaluation of potential damage for the life of BR-100 cask.

For cermet plate developmental purposes only, a goal was established for the flexural deflection capacity of the plate as measured in a three point bending test. The goal is twice the maximum fuel cell wall deflection anticipated during a 30 ft. hypothetical drop accident, without considering potential additional structural support by the cermet plate itself. The B_4C composition, enrichment, and plate thickness are the primary fabrication variables available to help meet this goal. However, other attachment and location design options are available to assist in developing a successful design. Vibration and thermal cycle effects, as well as wear, will be evaluated as an integral part of cermet plate qualification. However, it is anticipated that these will not be significant. All testing performed on the cermet plate material to qualify its mechanical, and/or chemical interaction performance will be included in the Engineering Test Plan.

C.10 Provide further explanation of Figure 2-21; its meaning is not clear.

Response - The subject Figure has been clarified in the Preliminary Design Report (PDR).

D.1 General - In order to evaluate this document properly, a requirements document is needed. Section III and Section VIII, Division 2 of the ASME Boiler and Pressure Vessel Code require a Design Specification whose contents are given in NCA-3250. A document or section in the Design Package analogous to the Design Specification is suggested. Is B&W required to meet all of the requirements of the referenced government and industry standards? If not, the exceptions should be stated and justified as to why these exceptions will not compromise the safety of the cask.

Response - The BR-100 cask is not a stamped ASME Code component. The ASME Code is used as a design guide to meet NRC requirements. It is our intent to satisfy the ASME Code requirements for materials, fabrication, examination, and testing. Even though these requirements are not presented in the preliminary design report, B&W has an in-house document ("BR-100 Cask System Requirement Document," document #15-1175528) which specifies the material and fabrication requirements. The examination and testing requirements will also be incorporated in this document during final design.

D.2 Introduction - Provide a discussion on the design temperatures for the cask components. Describe how the design temperatures will be determined.

Response - Design temperatures for cask components were determined using the hottest fuel within the fuel contained in the contract Statement of Work and the methods described in Section II-3. A clarifying statement will be added to the PDR. Design temperature information has been added to Section 3 of the PDR.

D.3 Page II-1-2, Packaging, Table 1-1 - The cask materials, where appropriate, should be called out with their ASME specification number and temper.

Response - Table 1-1 has been modified to include ASME and ASTM specifications for cask materials as applicable.

D.4 Page II-1-4 Fuel Support Structure - The aluminum materials are acceptable materials for Section III, Class 3 components only. It is not appropriate to perform a Subsection NB analysis on a component made from a Subsection ND material. Also, see Comments No. 10 and 21.

Response - Refer to response C.3.

D.5 Page 2-2, Note 2 and Paragraph 2.1.1.1 - The third paragraph under 2.1.1.1 states that the cask is able to survive a 30-foot drop and a 40-inch drop onto a punch. Table 2-2 indicates that the results are not yet known for the inner cask.

Response - The results for the puncture analysis for the inner cask are now included in the Sections 2.6 and 2.7.

D.6 Page II-2-1, Structural Evaluation, Tables 2-1 and 2-2 - List the design temperatures that determine the allowable primary stress intensity values. State the loads that contributed to the actual stresses. Margins can be reported a number of ways; this reviewer prefers a simple ratio of Allowable Stress ÷ Calculated Stress where a ratio of 1.0 means you are at the limit.

Response - Tables 2-1 and 2-2 are executive summary tables. The details of temperatures and loading conditions have been added to Tables 2-6 through 2-14. A footnote is added to Tables 2-1 and 2-2 referring to Tables 2-6 through 2-14.

D.7 Page II-2-1, Structural Evaluation - The 1st paragraph indicates that normal and hypothetical events will be evaluated. The reader does not know what these are without searching. There should be a list or table of these events with a brief description, concurrent pressure and temperature loads, and service limits (Level A, B, C, or D). See Comment No. 18 also.

Response - Summary descriptions of normal and hypothetical accident conditions are now included in sections 2.6 and 2.7, respectively. The stress limits for the normal condition of transport relates to Level A service limits of the ASME Code and the hypothetical accident condition relates to Level D service limits of the ASME Code.

D.8 Page II-2-2, Table 2-2 - State which of the three buckling criteria given in Table 2-3 were used in the Table 2-2 "compressive buckling" evaluation. Also, it is not clear why the inner shell was not evaluated for puncture when the outer shell was. Note 2 says it will be evaluated after test results can be incorporated, which is not consistent with the procedure implied by the 1st paragraph of 2.0. The last paragraph beginning on Page II-2-3 gives an apparent reason. This needs to be clarified.

Response - The PDR has been revised to identify use of individual buckling criteria. The puncture stresses for the inner shell are now included in the report.

D.9 Page II-2-6, Design Criteria - Paragraph 2.1.2 provides design criteria. It should also be stated which criteria will be used for materials, fabrication, examination, and testing.

Response - Criteria for materials, fabrication, examination, and testing will be incorporated in the final design. The criteria for concrete and impact limiters will be developed as a part of the final design.

D.10 Page II-2-7, Noncontainment Structure - This reviewer knows of no Section III rules that allow the use of Class 1 allowable stress intensities for a Class 3 material simply because inspection requirements have been upgraded. Use of Subsection NB construction rules for aluminum requires further justification.

The last two sentences of this paragraph (2.1.2.2) are confusing. It must be assumed that you have not or expect not to meet the Section III limits for noncontainment structures. If this assumption is correct, then the "higher allowables" should be discussed and justified.

Response - For Class 3 versus Class 1 component materials for aluminum 6061, refer to response C.3.

It is common practice to use Class 2 and Class 3 component allowables for noncontainment structures. If we use other allowables in the final design, it will be justified and NRC approval to use them will be obtained.

D.11 Page II-2-8, Impact Limiter - It should be stated how the impact load, reaction forces, and inertia load from the tests will be measured. The basis for the factor of safety of 2.0 on ultimate strength should be given. It appears to be more than adequate compared to the Section III allowable of 0.7 S_u for Level D events.

Response - Dynamic test measurements will include cask "g" loadings only. Since "g" load limits are the dynamic design criteria for the cask and basket, accelerations will be measured to assure the impact limiter does not allow the cask to exceed those limits.

A safety factor of 0.7 S_u will be used vs. the 2.0 currently used on ultimate strength.

D.12 Page II-2-8, Other Structure Failure Modes - Linear elastic fracture mechanics is not meant for brittle failure prediction. K_{1c} from a fracture toughness test, primary load, and maximum possible flaw size are used to determine susceptibility to brittle fracture. Linear elastic fracture mechanics can be used in a ductile material to determine if a flaw will grow under cyclic loading to a critical crack size. This comment also applies to Paragraph 2.6.2 on Pg II-2-39.

Response - There are no NRC approved criteria for brittle fracture predictions. Reference 2.11 provides the proposed brittle fracture acceptance criteria. Our analysis is based on the proposed criteria, and it specifies linear elastic fracture mechanics analysis using fracture toughness properties. The results of the fracture mechanics analysis are provided in Section 2.7.3.

D.13 Page II-2-29, Trunnion - The trunnion is treated as a cantilever beam, but it is also a short beam with great depth. That is, it is a beam with a small length to thickness ratio. Roark (6th Ed. Page 201) indicates that bending stresses and shear stresses need to be amplified when the length/depth ratio is less than 3.0. This should be investigated.

Response - Although the trunnion is a short beam of relatively great depth, it also is of relatively great thickness. Subsequent comparative analysis has been performed on the trunnion as a very short cantilever beam using the referenced Roark equation. The results show no significant difference in previously determined trunnion stresses (2% difference in margin of safety).

D.14 Page II-2-30, Reinforcement Ring - Male and female thread shear stresses are different for the same axial load (see ANSI B1.1). Explain what is meant by "classic analytical techniques." It is not clear why there is reference to Subsection NF. The reinforcement ring weld is within the jurisdiction of Subsection NB for the vessel.

Response - Female thread shear stresses are less than for male threads for the same axial load (due to greater shear stress area in the female thread). The final analysis will consider unique male and female thread stresses. The PDR will reflect that the classic analytical techniques for linear elastic analysis used the simple flexure formula (MC/I). The reinforcement ring weld will be redefined under the jurisdiction of subsection NB.

D.15 Page II-2-30, Outer Shell - Explain and justify the assumptions used to adapt the Bijlaard method to the reinforcement ring shell intersection. It would seem that it would be more appropriate to use the finite element model mentioned in 2.4.4 (Tiedown Devices) to determine stresses in the reinforcement ring and weld.

Response - The Bijlard method is used to determine the stresses at the intersection of two cylinders. The reinforcement ring weld is located at the intersection of the cask body and the ring.

As an alternate stress analysis method, the cask body and reinforcement ring were modelled as intersecting cylinders with the weld at the intersection. The weld will be analyzed using the finite element method in the final design phase.

D.16 Page II-2-30, Tiedown Devices, 4th Para - Explain why the cap screws on the opposite trunnion are not in tension.

Response - The transverse force was considered reacted by one side only. The tie down bracket on the opposite side was not considered to be toleranced to allow transverse load carrying capabilities on both sides. This is a conservative approach due to transverse loads not being reacted by tie down devices on both sides of the cask. However, the final design will be developed to allow equal distribution of transverse loads on both sides of the cask. The final design will then reduce trunnion attachment stress levels below preliminary design stress levels.

D.17 Pages II-2-38 & II-2-39, Tables 2-7 and 2-8 - There are two bottom plates (forgings), one for the inner shell and one for the outer shell. Which one is represented in Table 2-8? Which bottom plate is being discussed in the bottom portion of Table 2-7? Why aren't the stresses calculated for both plates and the inner shell given in Table 2-8? Differentiate between general membrane and bending stress intensities. State when and how local primary membrane stress intensities will be calculated. Margins may be much less for P_L stresses than P_m stresses.

Responses - The bottom plate referenced in Tables 2-7 and 2-8 is the outer bottom plate. The inner bottom plate and primary localized stresses and margins will be calculated in the final design.

D.18 Section 2.6, General Comment - After examining Table 2 of Reg. Guide 7.8 (attached), it was difficult to sort through Section 2.6 and feel certain that all of the Normal Conditions shown on Table 2 were addressed or would be addressed. It would be helpful to reviewers to have a similar table in the Design Package with, instead of x's, actual values of temperature, insolation, decay heat, pressure, weights and associated paragraphs, and stress table numbers.

In addition, it would be helpful if the stress summary tables (2-8 to 2-13) were more specific as follows:

(1) Type of Stress Intensity (P_m, P_L, P_b, or Combination).

(2) Temperature used to determine S_m or allowable stress intensity.

(3) Allowable displacement (Table 2-9).

Response - In the structural analysis, the worst combination of pressure and temperature is used to envelope the majority of load combinations defined in Table 2 of Regulatory Guide 7.8. A table of results parallel to the Regulatory Guide 7.8 table will be provided in the final design.

Tables 2-8 through 2-13 have been revised to include temperature, allowable stress intensities, and type of stress intensities.

Table 2-9 provides the fuel cell displacements. Rather than calculate allowable displacements, the normal and hypothetical accident condition displacements are provided to physics analysis to verify the subcriticality of the spent fuel array.

D.19 General - Indicate pressures as psig or psia rather than psi.

Response - Text has been revised to identify pressure in psia or psig where applicable.

D.20 Page II-2-46, Table 2-14 - It is not apparent what "Based on ASME Code Section III" means with respect to a margin of 393%, and why Section VIII was used instead of Section III to evaluate the external pressure load. The procedure of UG-28 is also included in Section III.

What is the basis for the allowable bearing stress (378,000 psi)? Table 2-3 indicates an allowable stress of S_u.

Response - The external pressure buckling criteria are essentially the same among ASME Section III, Subsection NB-3133.3, and ASME Section VIII, UG-28. To prevent any confusion, the reference to ASME Section VIII, UG-28, will be replaced by ASME Section III.

Allowable bearing stress for the closure bolts is 2.7Sy per ASME Section III, NG-3231.1 (c). As stated in response C.4, the reference to subsection NG is deleted from the text to prevent confusion.

D.21 Page II-2-56, 3rd Paragraph - A detailed document justifying the use of aluminum in Section III, Class 1 construction should be developed. It should demonstrate clearly what "detailed inspection and testing" will be done and why these are the only things that need to be done to qualify aluminum for Class 1 construction. The document should address both the cast and extruded aluminum and indicate the source of the strength data used to derive S_m and the basis for the derivation of S_m.

Response - In the final design, a justification document will be generated for use of aluminum 6061 as an application of Class 1 component material. This document will also address the basis for derivation of Sm.

D.22 Page II-2-56, Table 2-14 - The recommendations of Comment No. 18 also apply here.

Response - A footnote is added to Table 2-14 to identify primary membrane and (primary membrane + bending) stresses.

D.23 Page II-2-8, Trunnion - The second paragraph indicates that the requirements of NUREG 0612 will be used. In the DOE "Packaging Review Guide for Reviewing Safety Analysis Reports for Packaging," the standard for evaluating lifting devices is ANSI N14.6, "American National Standard for Special Lifting Devices for Shipping Containers Weighing 10,000 lb or More for Nuclear Materials." This standard is recommended.

Also, the DOE Packaging Review Guide recommends the use of ANSI Standard N14.2, "Tiedown for Rail Transport of Radioactive Materials for Tiedown Devices." B&W should review this document for applicability to the BR-100 design.

Response - ANSI N14.6 imposes the same design criteria and "g" loading factors as NUREG 0612 and 10CFR71. However, the final design stress analysis will include ANSI N14.6 as a design standard in addition to NUREG 0612 and 10CFR71. The draft of ANSI N14.2 was withdrawn from consideration in October 1989.

D.24 Page II-2-40, Paragraphs 2.6.4 and 2.6.5 - Don't forget to consider the effects of vibration on the vessel contents. Also consider the effects of the water spray on thermal stress from quenching and water freezing in joints.

Response - Vibration effects due to normal transportation will be evaluated as well as effects of water freezing. Due to the massiveness of the cask and slow thermal response, the water spray condition has been eliminated in Regulatory Guide 7.8, however, it will also be evaluated in the final design.

D.25 Page II-2-74, Computer Codes - Sufficient information should be provided to demonstrate that the in-house computer codes used in the design analysis have been benchmarked, verified, and validated where,

benchmarked = results compared to established codes

verified = results compared to closed form solution established test problems, and
validated = results compared to test results.

Response - All codes used in the final design to generate data to be used for licensing will be certified for verification and validation. Proper information will be included in the Final Design Report and the SARP.

E.1 Operational features of the cask system are not fully presented.

Response - It is true that the majority of the work performed by B&W during the Preliminary Design was related to the design and analysis of the BR-100, rather than to the detailed operational aspects. Our conception of the Preliminary Design is to provide evidence that the BR-100 is a cost-efficient cask which has a high probability of being licensed. That does not mean that we underestimated the operational aspects. On the contrary, B&W took into consideration in the proposal phase, operational aspects, ALARA, maintainability and reliability. We agree that these aspects have not been covered in detail in the Preliminary Design Package (PDP), mainly for two reasons: 1) the format of the Regulatory guide 7.9 does not provide an adequate support for this purpose, and 2) we considered that most operational features need the support of detailed drawings which will be generated during the Final Design phase.

E.2 Closure Lid, Shield Plug, and Penetrations

There was not enough detail to allow a thorough review of (1) alignment of drain pipes, shield plug, and closure lid, (2) sealing of drain pipes, (3) the use of a fixture plate for dewatering, (4) sealing the shield plug to the cask, (5) dose rate for workers on the closure lid/shield plug. Would routing the drain lines up the side of the cask be beneficial?

Response - Alignment of the drain pipes, shield plug and closure lid will be done by classical and proven methods, such as keys and guide pins. This will be part of the final design.

The method of sealing between drain lines in the shield plug and closure lid will also be detailed in the final design. However, for a better understanding of this unique feature, sketches are added in PDR section II-7.4.

The fixture plate is aligned with the shield plug and the protective ring, while the protective ring is aligned with the cask body. This is described in section II-7.4 of the PDR.

There is no sealing of the shield plug to the cask cavity. A paragraph has been added to the PDR section II-7.4 to describe the operations.

Figure 7.1 has been revised to show the "S" shaped penetrations through the shield plug.

We considered early in the proposal phase the possibility of bringing the vent and drain lines to the sides of the cask. We abandoned this design for the current one for the following reasons:

- o side penetrations are less protected than end penetrations,

- o pipes welded to the inner and outer shells, passing through the gamma and neutron shielding, are a "weak" point. They are difficult to analyze, especially the stresses generated by thermal expansion and regulatory tests. On the other hand, penetrations through the lid are very safe. There is no penetration through the cask cavity wall itself.

- o shielding analysis projects a lower dose rate at the surface of the lid than at the upper side corner. There is no benefit from an ALARA point of view to move the penetrations to the side of the cask.

E.3 Basket Design

The basket-to-fuel grapple interface should be thoroughly checked. The bottom of the cask cavity should be designed to assist decontamination during draining operations. Basket lift points should be balanced for careful insertion.

<u>Response</u> - A complete review of the grapple mechanism will be performed before the final design to determine the proper length of the basket.

The bottom plate (dwg # 1192213) provides a nominal gap varying from 3/4" to 1" (dwg # 1192002). This gap ensures satisfactory draining and flushing capabilities.

To lift the basket, we use four of the eight tie rods. This will avoid binding during the installation, and will allow the basket to hang true.

E.4 Yoke Design

The cask yoke should be designed to permit easy operation, including attachment to the crane, alignment to the cask, and decontamination.

Response - The cask yoke(s) will be developed to accommodate attachment to the crane hook. A support structure which holds the cask yoke in vertical position for crane hook attachment will be developed in the final design. The support structure may be two components, one component as a permanent part of the yoke to allow a vertical position and the second component as a tiedown fixture on the railcar for shipping. The yoke arm lifting plates will be developed for aligning with the trunnions in the final design. A lifting plate with tapered lead-in grooves will be considered.

A non-redundant lift system has been designed in accordance with NUREG 0612. B&W will consider a redundant lift system if more explanation is given as to why a redundant system is needed.

The yoke will be designed with decontamination as a consideration.

E.5 Impact Limiters

The impact limiters should be attached to the side of the cask, if possible.

Response - The impact limiter will attach to the side of the cask body. A quick installation and release mechanism will be designed to reduce operator radiation exposure.

Impact limiter lifting mechanisms will be designed with lift points in line with the center of gravity.

E.6 Personnel Barrier

Consider extending the personnel barrier to cover the impact limiters. Can the personnel barrier be removable without a crane and stay on the railcar?

Response - Consideration will be given to extending the personnel barrier to cover the impact limiter. Perhaps the personnel barrier can offer partial UV and weather protection for the impact limiters. Further consideration will include a multi-segment personnel barrier which operates on a trolley system (maybe the same system as the impact limiter trolleys) or a "clam-shell" type arrangement.

E.7 What are the in-service tests that are anticipated on the Kevlar/wood impact limiters?

Response - While Kevlar is unique in this application, its use is widespread. Therefore, Kevlar composite maintenance/repair expertise is not uncommon. In-service inspection requirements will be defined during the Final Design phase.

E.8 Why are left-hand threads used on the closure bolts? Why is use of a special torquing tool a positive control? Have torque limits or lubricants been considered for these bolts? Is a locking mechanism beneficial?

Response - The use of left-hand threads is proposed to provide a higher level of insurance for the positive closure. The operating procedures will make clear to the operator the details of bolting and unbolting operations, and a special tool will be used. We do not expect specific problems because of this characteristic of our design.

Control of the special tool means that it will be available only to the cask operator and only during bolting/unbolting operations, per operating procedures. Also, all the special tools available will be individually identified and their locations will be tracked.

The special tool will be tested during final design and its torque "control" will be characterized through this testing. The necessary margin between bolt torque specification and tool torque capability corresponding to eventual de-torque values higher than the initial torque will be considered.

We do not expect a need for any bolt locking device, which is not required for 10CFR71. However, if final design evaluations show it necessary, a locking mechanism will be designed for locking the closure bolt in place during transportation.

E.9 Has a thorough analysis of the trunnions been performed? Are the hollow trunnions or the trunnion blanks a contamination risk? Should the trunnion bolts be locked in place?

Response - The trunnion analysis details are given in a Babcock & Wilcox calculation document. The size of the document (88 pages) precludes including it in the PDP. Vibration and fatigue will be considered in the final design analysis. The void behind the trunnion is provided to reduce the overall weight.

The trunnion (and trunnion blanks) will be sealed at installation with a silicon sealant to prevent interstitial water entrapment in the trunnion attachment.

A trunnion bolt locking device (perhaps lock wires) will be incorporated in the final design.

E.10 Why was only the cask mid-plane thermal analysis presented? Was a partial atmosphere of helium factored into the analysis? How does the white paint ease the decontamination process? Is the white paint to be a specifically designated item? Clarify the use of Table 3.1. Clarify the need for a personnel barrier. Clarify the use and operation of the fusible plugs in the outer shell. Was the dense population of copper fins in the concrete taken into account for shielding?

Response - The responses to this question are as follows:

a). The temperature predictions presented were obtained for the cask axial midpoint location where the temperatures are maximum. The cask thermal analysis was performed primarily to determine if material (concrete, aluminum basket, or fuel cladding) temperature limits were exceeded. The predicted maximum temperatures provided this information; a detailed three-dimensional understanding of the cask temperatures was not necessary. Valve and seal temperature predictions will be presented in the final design report.

b). It is not necessary to consider the partial atmospheric conditions of helium for the thermal evaluation. Reference 3.8, page 543, states that thermal conductivity and thermal diffusivity of helium are not strong functions of pressure. Thus, the values at atmospheric pressure can be used.

c). Evidence demonstrating the ease of decontamination for the white exterior paint will be presented in the final design.

d). The white exterior paint that will be used on the BR-100 will be specified in the final design report.

e). The placement of Table 3-1 has been changed and Section 3.1 has been modified to more clearly define the term "baseline."

f). The maximum predicted cask surface temperatures for the baseline BWR and PWR spent fuel loads are 184.2°F and 194.9°F, respectively. Both of these predicted temperatures exceed the maximum 180°F allowed by 10CFR71.71 for direct exposure to personnel. Therefore, a personnel barrier was needed to prevent burns on people accidentally contacting the cask. The personnel barrier temperature predictions will be presented in the final design report.

g). The inspection, testing, and performance of the fusible plugs will be addressed in the final design.

h). The potential for neutron streaming and the modeling of the concrete-copper fin region for the criticality calculations is discussed in the response to question

A.25. Although there is a "dense" population of fins, the fins are relatively thin (0.08 inch, 0.20 cm). This characteristic helps reduce the chance of neutron streaming. With respect to the thermal calculations, the population of the copper fins must be dense to achieve desirable heat transfer through the region. The concrete-only section has been modeled, as can be seen in the temperature profiles in Figures 3-2 and 3-3. This concrete-only section has the largest temperature drop in the cask wall.

E.11 What leakage criteria will be used to demonstrate compliance with 10CFR71? What is the curie content and gas inventory that the cask is assumed to contain for design purposes? Have acceptable leak rates been calculated? Is Figure 4-1 correctly pictured?

<u>Response</u> - The seal criteria is discussed in the response to comment A.51/52.

The maximum curie content of the BR-100 is estimated to be approximately 4 x 10^6 when using the largest amount of fuel in its highest source strength state (21 PWR assemblies x 465 Kg U x 412 Ci/Kg U (for 3 w/o, 35 GWd/mtU, 10 year decay fuel)). The total inventory of gas in the 21 PWR fuel assemblies is estimated to be approximately 6000 cubic inches at an initial temperature of about 400 degrees F and an initial pressure of about 1600 psia; elemental constituents per cask would be about 175 g-moles of helium, 57 g-moles of xenon, 10 g-moles of krypton, 5 g-moles of nitrogen, and 1 g-mole of oxygen, using typical end-of-life fuel compositions.

Estimated leak rates were judged to be not essential for a preliminary design report. They will be included in the final design report.

Figure 4-1 is a schematic meant to illustrate containment components, not true physical configuration or scaled sizes. Figure 4-1 is considered acceptable for the use intended.

E.12 How much do the closure bolts weigh and is the indicated torque value correct? Is a locking mechanism being considered?

<u>Response</u> - The design goal for the bolt is a weight of 10 lbs, but the weight of the bolt currently in use is approximately 12.5 lbs. The torque value was calculated with standard equations and leads to acceptable stress levels on the bolt; however the assumptions leading to this torque value will be evaluated during final design to track potential over-conservatism. The special tool is designed and will be tested to the corresponding level of torque.

If deemed necessary, a locking mechanism will be designed for locking the closure bolts in place during transportation (see response to question E8).

E.13 Is the use of a 10^{-3} leak rate acceptable for testing the cask? What is the basis for determining the MNOP? What quantity of residual water is assumed in the cask cavity? Is vacuum drying to be used?

Response - The preliminary design analysis is based on a MNOP (Maximum Normal Operating Pressure as defined in 10CFR71.4) of 100 psi (which is the maximum allowed by NRC for B(U) cask).

The 10^{-3} leak rate will be justified during final design based on ORIGEN runs which are not yet available.

This value is conservative; our calculations show maximum pressures lower than 80 psi in the hypothetical case of all rods rupturing. During final design, the 100 psi will be replaced by the actual value calculated for the MNOP.

Our interpretation of NRC requirements is that all rods must be considered ruptured for normal conditions evaluation. Some SARP's have been submitted to NRC with the assumption of a maximum of 1% of rods ruptured under normal conditions. We intend to clarify this point during our next meeting with the NRC.

The cask cavity will be as dry as reasonably achievable, our intent being to ship the fuel under inert gas atmosphere with an internal pressure of about 10 psi.

Per 10CFR71.85(b), the containment system is required to be tested "at an internal pressure at least 50% higher than the MNOP". During final design this hydro-test will be evaluated separately for an internal pressure 50% higher than the MNOP and the internal pressure during normal conditions (expected to be no more than 80 psia) will be taken as the MNOP. For preliminary analysis, the pressure used was 150 psi for all normal conditions (pressure 50% higher than the 100 psi assumed value of the MNOP) which added some conservatism in the evaluations.

E.14 Should the Fission Gas Products statements in Section 4.3.1 be also in Section 4.2? Does the regulatory fire accident raise the internal pressure of the cask?

Response - This section will be completed during final design.

This section is organized as required in Regulatory Guide 7.9, but we agree that some paragraphs should be presented in a different order to improve the comprehension of the text. We will improve the presentation in the SARP.

The internal pressure considered in the cask cavity will be as discussed in E13 for normal conditions evaluation. Because of the regulatory fire conditions, the temperature increase in the cask cavity will produce a limited increase of the internal pressure with consequences that will be evaluated during final design.

E.15 Is a more detailed dose rate analysis planned? Can the cask be tailored to provide more shielding to workers? Have both PWR and BWR hardware been considered for gamma sources?

<u>Response</u> - A complete radiation mapping of the top head region of the cask with and without the closure lid in place is planned for the final design phase of the project. This mapping is needed to assess any radiation streaming effects through the fit-up between the shield plug, closure lid, and cask body and for ALARA considerations during lid handling operations.

One means of reducing the dose rate to workers near the upper cask is to "extend" the present design slightly, as suggested, by increasing the lead in the upper cask cylinder. Since this encompasses weight, structural, and radiation assessments, it will be addressed in the final design phase when operating and handling details are firmed.

Preliminary Design assessments for the BR-100 indicated that the PWR hardware would be the design limiting NFBC source for analyses. Hence, those were the activation sources used in the analyses. It is intended that both the BWR and PWR activation sources be re-analyzed in the final design phase.

E.16 How are utilities going to assure that the fuel to be shipped has more than the minimum burnup?

<u>Response</u> - Nuclear utilities have had administrative controls on fuel loading in reactor cores since the inception of power reactors. A part of these controls include core follow calculations on the burnup of each assembly in the utility's inventory. These calculations are verified at the beginning of each fuel cycle on a core-wide basis by core physics tests relating criticality with either soluble boron concentration and/or rod withdrawal. These tests verify the calculational methods used for burnup estimates and the administrative controls on fuel loading. In addition to the above loading procedures, many utilities have installed close-packed racks in their spent fuel storage pools that require burnup credit loading patterns for higher enrichment fuels. Thus administrative controls are in place at many sites. While an

additional layer of checking would be prudent, administrative controls have been demonstrated to be an acceptable method to ensure the proper fuel assembly burnup. An alternate method would be use of a "burnup" meter prior to insertion in the cask. The final decision as to which method will be used resides with the NRC, which has not made a final decision.

E.17 Does underwater loading of the cask require borated water?

Response - Loading in a borated pool is not required by the analysis. The statement merely reflects the conditions in spent fuel storage pools at reactor sites. For clarity, the statement has been changed to "... underwater loading in storage pools..."

E.18 Is a particular loading sequence required for criticality control of consolidation canisters? What is the effect on criticality potential if the cask is shipped with less than a full load of fuel?

Response - Based upon the hook weight constraints, only 10 to 15 consolidated fuel canisters can be loaded into the cask. For a fresh fuel assumption for enrichments greater than 3.2 weight percent U-235, a specific loading pattern is expected. This pattern is expected to be a checkerboard configuration, i.e. fuel assemblies only touching along a diagonal. It is further expected that spacers will be positioned to define the allowable fuel locations. Once this configuration is defined, loading can proceed in any sequence without impacting criticality. The cask will be less reactive with fewer than the allowable number of either consolidated or intact fuel assemblies. However, the worth of the missing assemblies is dependent on the location from which it was removed, with the central locations being generally more reactive than those on the periphery.

E.19 Chapter 7. Operating Procedures

Why is Subpart G of 10CFR71 referred to under Operating Procedures? Clarify the use of slightly different procedures.

Response - 10CFR71.87 "Routine Determination" is part of Subpart G "Operating and Control Procedures". That is the reason why we refer to subpart G.

The statement "Procedures will be slightly different..." is not really relevant to the purpose of this section, therefore we have removed it from the PDR.

E.20 Section 7.1.2.1 Moving the Cask to the Preparation Area

Is there space on the railcar to store the barrier or bolts from the barrier and impact limiters? Can the railcar be moved with the impact limiters supported by the tray or should they removed first?

Response - Step a. At this stage of the design, there is no space on the railcar to store the personnel barrier. Consideration will be given during final design to provide space, at least for the bolts of the impact limiters (I/L) and personnel barrier.

Step c. The I/L are stored on the railcar. Consideration will be given during final design to secure them to the trolley, so that it is possible to move the car.

E.21 Section 7.1.2.2 Cask Preparation Before Immersion

Can the cavity between the shield plug and closure lid contain radioactive gases? Is the fixture plate described in the PDR? Can there be a airborne contamination risk if the cask is filled with water before insertion into the spent fuel pool for loading?

Response - Step c. We cannot assume that there is no radioactive gas (or other potential airborne contamination) in the cask cavity because it depends on the extent to which the cask is cleaned and decontaminated after unloading at the repository and/or MRS. Therefore we included this step in the procedure before opening the lid. A description of the fixture plate, as well as the installation of the drain pipes, is provided in the PDR section 7.4.

Step h. In the assumed configuration the cask is empty and therefore there is no significant radiation field. Nevertheless, it is easy to fill the cavity with water during step e., before removing the fixture plate and the shield plug. We can also, for additional safety, connect the vent to the air effluent system of the plant.

E.22 Section 7.1.3 Immersion and Loading of the Cask

Can the cask bottom be protected during the loading sequence? Are any tools needed to assist fuel insertion into the basket?

Response - Step c. In final design we will consider a design feature such as having a protective plate fitted with a gasket placed on the bottom of the cask before immersion. This plate will protect the bottom from any material on the floor of the pool.

Wetting of the cask on its way in the pool will be addressed in the final procedure.

Insertion of fuel is facilitated by the funneled shape of the upper grid of the basket. Consideration will be given during final design to the need for a specific tool.

E.23 Section 7.1.4 Cask Draining and Removal from Pool

Are all tools anticipated for use described in the PDR? Can verification occur at Step 7.1.3.e? Where is the shield plug/fixture plate interface described? The plug lifting tool is not described. How does the plug/fixture plate seal the cask cavity? How does the main drain pressure relief valve operate; is it connected to a filter system? Clarify "pressurization fitting." Is the yoke-to-trunnion operational interface defined? Have "wiping down" operations on the pressure and drain lines been considered? Is the shield plug locked in place during cask lift out of the pool? Clarify Step j?

Response - We did not intend to provide a detailed description of the auxiliary equipment in the PDP. This equipment is commercially available and does not need any specific design effort.

Step a. We agree that this step can be performed during the loading operation itself (7.1.3 e.), as long as it is possible to ascertain the location of the fuel elements within the basket.

Step b. Description is given in the PDR section 7.4.

Step c. This tool is not yet designed.

Step d. Seals are provided between the protective ring and the cask body and between the fixture plate and the protective ring. The fixture plate is locked to the protective ring during operations. Further description is given in section 7.4 of the PDR.

Step e. The pressure relief valve is a commercially available component which will be defined during final design. If it is not viable to vent under water, we will require a connection to the liquid effluent system.

Step f. This step is confusing. We meant to remove the pressurized air inlet. Because all operations are monitored from a central command board, this operation is not relevant to the cask operation itself. Consequently it was removed from the procedure.

Step g. A lead-in device will be added to the lifting yoke arms to facilitate the alignment with the trunnions. The closing system will be defined during the final design.

Step i. Wiping of the pressure and drain lines has been included. The shield plug is secured during all handling operations since the fixture plate is locked to the protective ring (see section 7.4 of the PDR).

Step j. This step has been interchanged with step k.

E.24 Section 7.1.5 Preparation for Shipment

Has dose rate been factored into the port design? Are the operational procedures complete? How is the cavity sealed? How is the closure lid handled; are the seals replaced for each shipment? Clarify bolt installation. Clarify in Step l how the cask cavity is backfilled with a partial atmosphere of helium. How is dewatering and vacuum drying performed? Where are the cask holddown brackets described; is there a general "Control Check" on the entire process?

<u>Response</u> - As indicated during the presentation to the TRG, the sketch showing the drain and vent lines through the shield plug is confusing. For clarity these lines are shown "straight" although they are "S" shaped. There is no direct radiation streaming for the operators at the ports level. However, to enhance ALARA requirements, we will suggest during final design the use of additional portable shielding to protect personnel during these operations.

Steps a, b, f. As stated in the introduction of section 7, we intend to give only the highlights of the operating procedures. We will address these concerns during final design.

Step d. Seals are provided between the protective ring and the cask body, and between the protective ring and the fixture plate (see section 7.4 of the PDR).

Step j. The handling of the lid will be defined during final design. At the moment we intend to replace the gaskets before each shipment. Based on the tests on the gasket materials, this could be changed during final design.

Step k. The installation of bolts according to an installation diagram means only that there is a sequence to follow to ensure the proper closure of such a lid. Bolts are interchangeable. Due to the protective ring, there is no water and no contamination in the bolt holes. The final torquing pattern is not yet available. It will likely be necessary to use a pneumatic or hydraulic torquing tool for such bolts.

Step l. At this stage, the lid is in place (step j) so that the 2" line is no longer connected. The cavity is filled with helium to atmospheric pressure (step f.). To pressurize the cavity to 0.5 - 0.7 atm, it is necessary to use a vacuum pump, through the vent port of the lid.

Steps m, n. The arrangement of the cover plate is given on dwg # 1192011 in section "Preliminary Design Baseline Drawings" of the PDR. Method of testing will be the "pressure rise" test method. There is no gasket between the shield plug and the cask body. Therefore, no water is retained in this area. The vacuum is achieved at the level of the closure lid.

Steps u, v. Will be defined during final design.

E.25 Section 7.2.2.1 Rotating the Cask to the Vertical Position in the Preparation Area

Response - Clarify the Holddown U-bolt function.

Step d. Text has been corrected. The U-bolts capture the cask body, not the trunnions.

E.26 Section 7.2.2.2 Sampling and Cooling Down

Clarify how the sampling is performed.

Response - Step b. An efficient and easy method of sampling is to connect a bottle in which a high vacuum has been made to the vent port. In that case we do not need to pressurize the cask cavity, diluting radioactive gas if any is present.

Step d. Sampling is done through the vent port provided on the cask lid. If circulation is required, this is done through the 1/2" drain line. Both lines are fitted with quick disconnects.

E.27 Section 7.2.3 Unloading

Clarify handling and the need for decontamination at the Federal facility.

Response - Steps d, f. At present there is no definition of the receiving/unloading facilities. We assume that the receiving hot cell is equipped with the necessary handling crane to remotely remove the lid and shield plug. Decontamination of the lid (internal surface) may be necessary due to some contamination occurring during transportation (for instance, from crud particles).

E.28 Section 7.3.2 Decontamination, Draining, and Drying

Clarify decontamination at the Federal facility. How detailed is the "survey" of the cavity? What is the torque value on the lid for shipment of an unloaded cask?

Response - In this section we assume that after dry unloading of the cask, a decontamination/cleaning of the cavity is performed. Therefore there will be water and/or decontamination solutions in the cavity, which has to be drained and dried before shipment. This is the reason for steps d to l. Here again we have given only highlights of the detailed procedure we intend to develop during the final design phase. We made some assumptions for the unloading of the cask in a "dry" facility. Detailed procedure will be based on the information available at that time.

Step a. The "survey" of the cavity is supposed to be a remote visual inspection to check if there is any crud in the basket which may have to be removed before the next shipment. Radiation monitoring can also be performed.

Step b. There is no specific criterion in the preliminary design, and we do not intend to have one during the final design, regarding the decontamination level of the cavity.

Step r. This is a possibility we shall consider during final design.

Chapter 8 Acceptance Tests and Maintenance Program

E.29 Section 8.1.4.3 Miscellaneous

Clarify the meaning of miscellaneous tests.

Response - B&W has been required to follow the format of Regulatory Guide 7.9 in writing the PDP. That is the only reason why these important "operational tests" are within section 8.1.4.3 "Miscellaneous". This section is intended to cover only the "regulatory acceptance" tests, or the contractual "operational" tests. It does not cover the QA inspections and controls which are normally part of a Fabrication and Inspection Program. There is no intention to separate these tests, "form and fit" or "operational" from the other acceptance tests. Weight measurement is included in this section.

E.30 Section 8.1.5 Shielding Integrity Tests

Have the fabrication techniques for lead pour on the BR-100 been demonstrated?

Response - Techniques exist to avoid out-of-round conditions during the lead pouring. Robatel has been designing and manufacturing lead shielded casks for more than 35 years and the fabrication procedure will implement such techniques. There is no need to provide a radial space for lead expansion.

Longitudinal space needs will be evaluated during final design.

E.31 Section 8.2 Maintenance Program

Is the maintenance of the BR-100 tied to calendar year or shipments?

Response - Mixing calendar activities and number of shipment activities may create some more work to track the maintenance operations. On the other hand, it may reduce the cost of maintenance. The final decision should be left to those in charge of the cask fleet maintenance.

E.32 Section 8.2.1 Cask Structural and Pressure Tests

Why is the MNOP set at 150 psi? Why not pressure test the cask cavity on a yearly basis? Is it cheaper to test the used trunnion bolts or replace them routinely?

Response - The test pressure has been determined to be 150% of the maximum normal operating pressure as required in 10 CFR 71.85 (b). The maximum normal operating pressure is defined as "the maximum gauge pressure that would develop in the containment system in a period of one year under the heat test specified in 10 CFR 71.71 (c) (1), in the absence of venting, external cooling by an ancillary system, or operational controls during transport". We believe that 5 years would be acceptable for such a maintenance test, but a one-year period would not increase the life cycle cost significantly.

The Ultrasonic Testing (UT) of the trunnion bolts is supposed to check the soundness of bolts after the structural test of the trunnions. We will consider, during final design, the possibility of replacing bolts without performing UT.

E.33 Section 8.2.2 Leak Tests

Clarify the use of a pressure rise method of testing fusible plug integrity.

Response - The "pressure rise" test method is applicable to the fusible plugs. It requires the use of a "vacuum bell" to cover the plug and make a vacuum. Alternate methods, such as helium leak detection, will be evaluated and ports added if required.

E.34 Section 8.2.3.3 Fuel Basket Assembly

Will procedures for inspection of the basket take potential contamination into account?

Response - This will be included in the detailed procedures during final design.

E.35 Section 8.2.7 Discussion

A requirement to have the painted outer surface free of pits and scratches will be difficult to implement.

Response - Testing and analysis during final design will quantify a more reasonable acceptance criteria.

E.36 The drawing needs more details for evaluation. Can the 2-inch drain line be capped before the closure lid is installed? Does the dewatering tool attach to the plug?

Response - The shield plug is equipped with three tie-down lugs for handling. It is also equipped with threaded holes for attachment to the fixture plate during operation; in this case, the fixture plate being equipped with lifting devices, shield plug and fixture plate will be handled together. There are three penetrations in the shield plug, corresponding to the 2 inch drain line, and the 1/2 inch and 3/8 inch drying, inerting, venting and sampling lines; they will be represented in the final design drawings. A key will be used to locate the shield plug relative to the cask ring flange and to the basket for drain lines connections. All these details will be provided on the final design drawings.

It could be a good idea to cap the 2 inch drain line right after removal of the fixture plate and we will pursue this idea during final design.

All the connections for dewatering will be installed on the fixture plate which is part of the ancillary equipment. The details of the fixture plate will be provided during final design.

E.37 How is the Lid handled and aligned to the cask?

Response - The lid is equipped with three tie-down lugs for handling. Two guide pins (dowels) will be used to locate the lid relative to the cask ring flange and to the shield plug. These details will be provided on the final design drawings.

E.38 Is the O-ring groove cut-back to help retain the seal? Is the vent/drain cover plate designed to protect the fittings in a regulatory pin drop accident? Are the existing drain lines adequate to circulate water, if required? Are details of the quick-disconnects available? Are the connections clearly marked and incompatible to prevent incorrect installation? How does the test port work?

Response - The representation of the seal grooves on the current baseline drawing is stylized. In the groove design there will be a cut-back on each side to have the groove retain the seal under any condition.

The penetration cover plate behavior during accident conditions was not addressed during preliminary design. This analysis of the pin drop accident will be addressed with NRC during final design.

The BR-100 cask is not currently designed to permit true circulation of water. The drain lines (2-inch and 1/2-inch lines) can be use either to fill the cask or to empty it. If final design evaluations show that circulation of water can improve the BR-100 cask operability, we will more likely add another 2-inch penetration in the shield plug.

The details of the quick-disconnects will be provided during final design and we will consider their selection from an easy-operation point of view. The male coupler is installed in the lid.

The different lines will be clearly identified; however, wrong connections will be impossible because of the difference of size of the male and female couplers.

Operation of the test port will be detailed during final design.

E.39 How is the basket handled? How are the drain lines attached? Is the basket keyed in place? How does the drain system work? Do the basket cells "communicate?" Is the basket compatible with all interfacing site fuel grapples?

Response - The basket will be handled in one piece through fittings adapted at the extremity of three or four of the longitudinal tie-rods.

The drain pipes are incorporated in the basket formers and attached to the basket upper grid. Draining of the cask if the basket is not installed is not considered as a normal operating condition. This condition will be considered during final design.

The basket is keyed to the flange of the cask to ensure proper positioning of the draining lines and avoid potential rotation during operations and transportation.

The bottom of the basket is supported by "support posts" above the cask bottom. The distance between bottom of the basket and bottom of the cask is 0.75 inch at the level of the cask centerline and 1 inch at the periphery of the cask cavity (where the drain lines are attached), which creates a slight slope allowing for easier and more complete dewatering.

The basket fuel cells have no "communication" allowing for transfer of water from one to another. However, dewatering and filling of the cask are effected with the cask in vertical position through the drain lines and by the bottom of the cask cavity (under the basket bottom plate). Holes in the basket bottom plate at the level of each fuel cell allow for uniform filling or draining of the fuel cells and of the cask cavity.

See response to question E.3.

E.40 Has the NRC had comments on the unique construction of the inner shell-to-outer shell transition? Is the annulus around the shield plug to be dewatered? Is a contamination barrier between the front impact limiter and the cask/lid beneficial? Is there a method to remove water from the lid bolt holes? Is the potential for water to get behind the trunnions or blanks a contamination risk? Will the cask lip protect the lid during a regulatory side pin drop condition?

Response - The design of the inner shell, welded to the upper flange of the cask but independent of the cask bottom forging, has two principal advantages: 1/ the secondary stresses creating longitudinal compression in the inner shell and due to differential thermal expansion between inner and outer shell of the cask are reduced, and 2/ the length of the cask is reduced by the use of lead gamma shielding enclosed between the inner shell and the outer shell bottom forgings. NRC will review the stress analysis for evaluation of this design.

There are communication grooves between the annulus around the shield plug and the inside of the cask cavity to allow for dewatering of this annulus. The details are not currently represented on the drawings but will be included on the final design drawings.

The protective ring seals prevent the closure bolt circle from contact with contaminated water, and there is therefore no need for a contamination barrier between top of the cask and top impact limiter.

Because during normal operations pool water is prevented from entering the bolt holes, no method was needed for draining and decontamination of these holes. If, after thorough evaluation of the operating procedures, we consider that a bolt hole contamination incident is likely to happen, we will design drain lines providing for communication between the bolt holes and the outside of the cask upper flange. The inconvenience of such a feature will be an increase of the decontamination problem: contamination of the bolt holes and of the cask sealing surfaces.

Silicone sealing is provided at the periphery of the trunnion and at the level of each trunnion bolt. The object of this sealing is to prevent pool water from entering anywhere in the trunnion. The same type of sealing is provided at the level of each penetration in the cask body, to prevent pool water of from entering and to limit the decontamination operation to the outside surfaces only.

The function of the lip is not only to protect the lid during the side pin drop accident but also during a corner drop. We agree that the 0.5-inch thickness is likely insufficient. We expect more elaborate structural analysis to allow us during final design to reduce the closure bolt circle diameter, a reduction of at least 3 inches being possible. In this condition, the lip will be 2 inches thick, which should be sufficient to protect the lid.

E.41 Are the impact limiter attachments defined? Does the impact limiter have any contribution to the regulatory fire condition?

Response - The attachments for the impact limiter will be developed during the final design phase. The limiter will be designed to make no contribution to the fire accident condition.

E.42 Could the personnel barrier extend beyond the impact limiters? Can the barrier be removable without using a crane? Can the barrier damage the limiters during installation? Is the railcar to be designed to industry safety standards?

Response - See response to question E.47.

E.43 Clarify the statement regarding access to two crane hooks.

Response - The statement has been clarified in the PDR.

E.44 Should the cost of the lift fixtures and transporter be included?

Response - The cost of the lift fixtures and transport vehicle cannot be estimated accurately enough at that stage of the design. It is our intent to include these costs in the cask system cost during final design.

E.45 Is there a shielding concern because of the liberal use of copper fins?

Response - This concern is addressed in A.25.

E.46 Will there be an evaluation of acceptable coating loss due to routine handling? Can water get between the coating and the cask outer shell?

Response - Information on the amount of coating that can be removed before stripping and replacing are necessary will be evaluated in the final design phase.

Periodic maintenance examinations of the cask outer coating are planned to assure coating integrity is not significantly compromised and that water entrapment is thereby precluded. See Section 8.2.7 Miscellaneous of the PDR.

E.47 Will industry and Federal safety standards be used on the railcar design? Will any handling equipment be shipped on the railcar with the cask? Will the personnel barrier protect the cask from weather and incidental damage? Are the impact limiter trays described in detail?

Response - The necessary safety appliances as per AAR and FRA will be shown on the final railcar design.

See response to Q A.47

The cask is open to the environment to provide natural air circulation for heat transfer. The final design will determine the type of barrier to be used. The barrier will remain on the railcar.

The trays or fixtures that align and translate the impact limiters will be completed during final design.

E.48 Clarify the difference between the tie-down arrangements in Figures 1 and 2. Are the longitudinal forces reacted by only the rear tie-downs? How is the skid to be moved with the cask on it?

Response - Text and figures will be modified to incorporate comments. The longitudinal forces are reacted by the rear tie downs only. The final skid design will be optimized to allow a more even distribution of longitudinal forces (between saddle and rear tie downs).

E.49 Clarify the design of the yoke as shown in the figure. Clarify if yoke is part of a non-redundant lift system. How is the yoke handled into position? Pneumatic motors would be a preferable choice to control the yoke, although operation after motor failure should also be addressed. How is the system decontaminated after use? Should the U-bolts be shown in Figure 9-2?

Response - The narrative will be modified to address the figure.

The cask yoke has been designed to non-redundant lift system requirements. If a redundant capability is identified as a requirement, Babcock & Wilcox will develop a redundant yoke and modify weight allowance appropriately.

A means of standing the yoke in a vertical position is addressed in comment E.4.

A pneumatic motor will be used for yoke operation; yoke operation after motor failure and yoke decontamination will be addressed in the final design.

Figure 9-2 has been revised.

E.50 The design should minimize opportunity for damage during removal or installation. The barrier should cover the limiters and not require use of a crane. Can the barrier prevent water spray from contacting the cask? Are design details available? Is a clam-shell design feasible?

Response - Proper consideration to personnel barrier handling/impact limiter damage will be given in the final design stage. A "clam-shell" or trolley for personnel barrier handling will eliminate potential for impact limiter damage.

Extending the personnel barrier for impact limiter environmental protection will be considered at final design. However, we anticipate the impact limiter design to provide for protection against environmental elements. The personnel barrier is designed as non-solid to allow convective heat transfer from the cask. A "no-crane" barrier will be considered in final design.

Water spray requirements will be considered during final design.

The details of personnel barrier handling (handles, lift lug access and number of bolts) will be developed in the final design.

E.51 Can air be used for dewatering or inerting? Is the design of the operational control panel too far along to consider features which might benefit site personnel and cask turnaround?

Response - Air can be used for dewatering purposes. The cask is later inerted with helium due to the high thermal conductivity and inerting properties of helium.

The draining, drying, inerting, and testing equipment system will be revisited during final design. Babcock & Wilcox will involve utility representatives in determining the best methods to use existing plant services while minimizing the requirements for special tools.

E.52 What research supports the use of paint to minimize "weeping." Is the specific paint to be used already identified?

Response - A Sandia National Laboratories report on in-service cask weeping experience has been referenced. The paint selected for the exterior surface of the cask is a commercially available item. (See response to question A.3)

E.53 Could a figure be added to show the system as configured for an intermodal lift?

Response - An additional figure, Figure 9-6, has been added.

E.54 Are further features required to make the closure bolts or lid "tamper-indicating"?

Response - Operation of closure lid bolt is described in Section 2.4.2.

E.55 Are other NRC approvals required to permit the use of the BR-100 as a "lag" storage cask?

Response - It is not intended that the BR-100 transportation cask be used as a permanent storage cask at the Federal processing facility, although its use in that role is feasible. Instead, this statement addresses the time waiting for the facility's operational schedule to permit unloading. This mode of operation is within the scope of the design requirements for the BR-100 cask. The DOE could elect to increase the mission of the BR-100 to include use as a long-term storage cask. In that case, BWFC agrees that there may be additional regulatory and licensing issues to be addressed.

E.56 Comment:

a) Is the underwater dewatering step only incorporated to aid in meeting crane hook weight restrictions?

b) What is the basis for stating that the BR-100 will be compatible with Federal facilities?

c) Minimizing the time that the loaded cask is in the poolside work area, as opposed to in the pool or on the railcar, should result in lower worker exposures.

Response:

a) The 100-ton limit is a contractual requirement that is based on a hypothetical 100-ton crane limit at the different reactor sites. The BR-100 is designed such that underwater dewatering is easier and quicker than using conventional techniques, although it also reduces the weight to be lifted out of the pool.

b) The repository/cask interfaces at the French reprocessing facility (La Hague) were studied in determining that the BR-100 has the flexibility to accommodate potential repository or MRS requirements.

c) That observation is the basis for our operational strategy and will be pursued through final design.

E.57 Has the cask/hot cell interface been optimized to reduce any worker dose rate at the Federal facility?

Response - The Figure showing the Cask/Hot Cell Sealing Interface was meant to only illustrate the concept envisioned for the interface between the cask and hot cell. Specific details relating to mechanical and shielding design of the interface are not within the scope of the cask development project. BWFC expects to work with the DOE to ensure compatibility between the BR-100 cask and the hot cell interface as the hot cell design is finalized.

F.1 10CFR71 states that the hypothetical accident conditions are to be applied in the following sequence: free drop, puncture, thermal, immersion. The PDR states that wall section testing will be employed to address the hypothetical thermal issue (Fig. 3-11). If this is the case, then it should be verified that drop testing and puncture testing of the cask will not significantly influence the thermal resistance through the cask wall. Puncture and impact might create crushing of the concrete shield, gaps between the wall layers, or possible fracture of the solder joint between the copper fins and the lead gamma shield.

In Section 3.5, it is mentioned that the test specimen (with all sides insulated except the outside) will be exposed to a preheated furnace. It should be verified that the following items don't invalidate the test:

a. In an actual fire, the cask wall will be at normal operating temperatures before the fire starts and thus the test might be too non-conservative.

b. Neglecting the fuel thermal loading on the wall is also non-conservative.

Response - The thermal fire test will be used to provide boundary conditions for the cask for thermal evaluation of the hypothetical accident and is not meant to be a verification test. This test specimen is of an un-dropped and un-punctured cross section. The quarter scale drop test will be used to demonstrate that there is only localized damage to the cask, which has a minimal effect on the thermal performance. The types of cask damage suggested in the question (crushing of the concrete, opening of gaps between the cask wall components, or fracture of the copper fin-to-lead shield joint) all serve to increase the thermal resistance of the cask, thus decreasing the heat transferred to the spent fuel in a fire accident. The most conservative arrangement for the fire testing is a configuration that will transfer the most heat to the cask spent fuel contents, which is a pre-drop and pre-puncture cross section.

Following are the responses to the two specific questions:

a). In the fire test, the test specimen will be preheated to a steady-state condition that approximately produces a 225°F maximum concrete temperature before the specimen is inserted into the furnace. The test section preheating is accomplished with heaters that are positioned near the test specimen inner wall surface.

b). The thermal loading of the fuel during the fire will be simulated with the test specimen heaters and energy leaking through the test specimen insulation.

Section 3.5 has been revised to include a discussion of the pre- and post-accident test specimen conditions.

F.2 The 12 fusible plugs in the outer cask wall are to allow the escapement of steam in the event of a fire. It should be checked to see if this is enough; i.e., if the concrete is securely bonded to the shell then this might be questionable.

Response - The number and placement of the fusible plugs will be checked during the final design.

F.3 The coefficient of thermal expansion of copper is about 50% greater than that of concrete. The inside copper fins will therefore try to expand much more than the concrete shield during operation. Repeated usage of the cask might cause thermal stress cycling of the soldered joints and potential joint opening. Perhaps Robatel should be contacted to see if this has ever been a problem for them.

Response - Robatel has performed a number of thermal cycling tests on the concrete/copper fin thermal shield and have not observed any failure of the joints between the copper fins and the lead gamma shield. The thermal efficiency of the system is not degraded, however, by cracks at the fin/lead joint.

F.4 On page II-3-13, I think a typo exists: .04 inches should be .4 inches.

Response - The .04 inches should be .4 inches. Section 3.6.1 has been revised accordingly.

F.5 It might be stated that all temperatures in radiation terms are absolute temperatures.

Response - The temperatures in the terms of section 3.6.1 have been modified to denote an absolute temperature scale was used.

F.6 In Table 3-2, thermal properties of the Robatel borated concrete are given. The temperature variation of conductivity and diffusivity seem somewhat contradictory. This should be checked.

Response - There were some typographical errors in Table 3-2, which have been corrected. These typographical errors were in the borated concrete and lead thermal diffusivity. The behavior of the concrete thermal properties is due to the dehydration of the concrete at a temperature of approximately 284°F (140°C).

F.7 In reviewing equation 3-5, I had trouble with the definition of l' and l''. This needs clarifying. Dimensioning Figure 3-11 would help. Here's what I think might be right:

l' -- radial width of the fin attached to the lead
l''-- axial distance between two neighboring attached and unattached fins.

Response - The definitions of l' and l" have been added to Figure 3-11 and section 3.6.1. The definitions of the two terms are as follows:

l' -- radial distance over which heat transfer occurs from the attached to the unattached copper fin, and
l" -- circumferential distance between the attached and unattached copper fin.

F.8 This comment deals with equations 3-1 and 3-3, the one dimensional heat transfer equation between the cask outer wall and the environment (equation 3-2 needs units to be clear, ditto for equation 3-7).

This is a very important equation in arriving at concrete and fuel cladding temperatures. From Table 3-4, it is seen that for the 50/4.5 wt% case, the temperature drop through the wall of the cask is only 48°F. The drop from the outer surface of the cask to ambient is 123°. The concrete temperature is given as 246°, which is very close to the 250° concrete limit temperature. Any surface temperature increase will cause a corresponding temperature increase in the concrete (assuming constant properties) and somewhat less of an increase in the cladding temperature. While equation 3-1 might be the best one dimensional approximation available, it's validity and precision should be looked at.

It was mentioned that radiation is the dominant term in this equation. This term is the classic grey body term for radiation to an infinite far field body. Hence, the emissivity of the infinite body doesn't matter, only the emissivity of the cask outer surface. I assume that the cask wall emissivity used was .9 as mentioned on page II-3-2. If this value is reduced, perhaps by grime picked up during transportation, then outer wall temperature will increase. Additionally, the rail car deck and the personnel barrier will serve somewhat as a radiation shield and tend to increase the temperature of the outer cask wall. This condition should be checked either with hand calculations or possibly with a two dimensional FE model similar to the basket model.

The convection coefficient term used was for a horizontal cylinder suspended in air. The effects of the cask car and the personnel barrier will likely restrict convective flow and 'h' will likely tend to decrease. This will raise the cask wall temperature, especially the bottom side. I'm not sure how much, but some scoping calculations should be performed to see if it's excessive.

<u>Response</u> - The 50/4.5 wt% case presented is a limiting case and was included to demonstrate the large thermal margins that exist with the BR-100 design. The statement of work only requires the BR-100 cask to transport bounding fuel defined for the baseline case (35 GWd/mtU/3.0 wt%/10 year-cooled fuel for the PWR and 30 GWd/mtU/3.0 wt%/10 year-cooled fuel for the BWR). The predicted maximum concrete temperatures for the BWR and PWR baseline cases are 197.6°F and 210.1°F, respectively. These temperatures are significantly less than the 250°F concrete design limit temperature.

The deterioration of the cask surface thermal radiation characteristics is a recognized concern. The deterioration of the surface characteristics would affect the solar energy absorbed as well as the energy emitted to the environment. These effects will be investigated more fully during the final design phase. Preliminary scoping calculations indicated that the thermal radiation and natural convection effects from the rail car bed and the screen-type personnel barrier are small. The circumferential variation of the cask surface temperature is also expected to be small during normal operation, but may vary significantly during the hypothetical fire accident. All of these issues will be addressed in detail during the final design phase.

The units for the equations are provided in the PDR.

F.9 If a small leak should develop in the cladding surrounding the borated concrete, then it seems to me that with continued cask usage, the concrete could become dehydrated and no longer function in the thermal diode fashion as required. If this is so, a periodic inspection technique needs to be developed to check for this condition.

<u>Response</u> - The concrete is poured in a tight enclosure defined by the upper flange, the inner shell, the outer shell, and the forged bottom. The minimum thickness of this enclosure is 1". All welds are x-rayed or helium leak tested during fabrication. Then an annual leak test is performed on each fusible plug to ensure the tightness of this enclosure.

F.10 The impact limiter covering the cask lid is composed of balsa and redwood. During the hypothetical impact event, this wood crushes and dissipates energy. It should be verified that the impact loads generated on the outside of the cask lid by the crushing wood will not yield the lid or flange.

<u>Response</u> - The impact load from the impact limiter on the closure lid will be evaluated in the final design.

G.1 Text needs to specify which normal condition stress margins of safety (heat, cold, pressure, vibration or drop) margins of safety are tabulated in Table 2-1.

<u>Response</u> - The loading conditions for given margins are provided in Tables 2-8 through 2-13. Table 2-1 is the executive summary table. A footnote will be added to Table 2-1 referring the reader to Tables 2-6 through 2-13 for additional information.

G.2 Text needs to specify which accident condition stress margins of safety (drop, pressure, buckling) are tabulated in Table 2-2.

<u>Response</u> - The loading conditions and temperatures for given margins are provided in Table 2-14. Table 2-2 is the executive summary table. A footnote will be added to Table 2-2 referring the reader to Table 2-14 for additional information.

G.3 It is stated that XM-19 is being considered for the cask shell because of small margins for 304L. Suggest also considering XM-19 for the closure lid because of a preliminary margin of safety of 1.

Response - The closure lid is made of XM-19 stainless steel plate. In the final design, excess conservatisms will be reduced to improve the safety margins.

G.4 Suggest consideration of inserts for closure bolts to minimize galling.

Response - We are considering Helicoil type hardened inserts for the closure lid bolts to minimize galling and improve joint integrity as a part of the final design.

G.5 Text needs to explain function of fusible plug in Figure 2-5.

Response - Section 2.1.1.1 has been modified to include the discussion on fusible plug functions.

G.6 Need to specify length and material for impact limiter bolts.

Response - The impact limiter exact attachment method will be defined in the final design.

G.7 Need to explain seal pre-conditioning.

Response - The seal pre-conditioning refers to mechanical and thermal cycling of seal gaskets prior to leak-tightness testing. This will be explained in the text of Section 2.3.4. However, should Sandia National Laboratories demonstrate either that the effect is insignificant for the single-cycle anticipated for BR-100 application, or that the effect is quantifiable, this would help minimize B&W testing efforts on the seal.

G.8 Figures 2-9 and 2-10 do not relate to cell modeling as the text states.

Response - The figure designation has been changed to Figures 2-14 and 2-15 for fuel cell modelling.

G.9 Design drawings show copper fins are .08 inch thick. Fabrication of a 1/4-scale model would involve pouring concrete around .02 inch thick fins. The fins may be too thin to maintain the design geometry.

Scaling of the copper fin welds for 1/4-scale testing should be considered.

Response - The concrete as-poured consistency is nearly like water. Careful concrete pouring during quarter-scale model fabrication will assure maintenance of design geometry. Scaling of copper fin brazements will be attempted.

G.10 Regions II and III are not designated in Figure 2-21.

Response - Figure 21 has been modified to show discrete regions I, II, III.

G.11 Explain integers under balsa and redwood columns in Table 2-16.

Response - Table 2-16 has been revised to explain integers.

G.12 The Kevlar impact limiter attachments are not visible in test photos. Diagrams showing attachment are needed. Three methods have been mentioned - bolts, Kevlar attachment, and Kevlar epoxy.

Response - Figure 21 has been modified to show attachments.

G.13 Explain the difference between limiting and baseline.

Response - The baseline condition in Table 3-1 refers to the base conditions specified by DOE in the statement of work. Limiting refers to the condition that produces the most restrictive of the three thermal design limits (250°F on the concrete, 350°F on the aluminum basket, or 680°F on the spent fuel cladding). In all of the applications in Chapter 3.0, the concrete temperature was the most limiting of the design limits. Section 3.1 has been modified to clarify the differences in the limiting and baseline conditions.

G.14 Low temperature (-40°F ambient) behavior of water in the concrete should be discussed.

Response - The behavior of the various cask components, particularly the concrete, will be evaluated in the final design phase.

G.15 Page II-3-4 states the Reference 3.5 spent fuel cladding limit is 716°F. This section states the Reference 3.5 limit is 680°F.

Response - The references used for the spent fuel cladding limit have been clarified. The 680°F value is used for the BR-100, but NRC allows up to 716°F.

G.16 Clarify whether the test specimen is a wedge-shaped specimen. If the inner wall is insulated from heat, provide explanation of how the heat from the spent fuel is simulated. Explain how the moisture content of the concrete will be controlled and what type of quality assurance will be required.

Response - The test specimen is not a wedge-shaped design, rather a rectangular-shaped design. The flat surface of the rectangular test section and a corresponding section of the full-scale cask have silhouettes that differ by less than 0.4 inches. The heat from the spent fuel will be simulated by heaters that are positioned near the inner surface of the test specimen.

The moisture content of the concrete will be controlled by mixing and curing procedures followed during fabrication. Safety related Quality Assurance is required for the thermal/neutron shield thermal performance testing.

G.17 States "The object of the containment analysis is to demonstrate that the cask is leaktight during normal and accident conditions, per ANSI 14.5 definitions. The tightness criteria we intend to use is a maximum allowable leakage rate of 10^{-3} std cm^3/sec." ANSI 14.5 defines leaktight as a 10^{-7} std cm^3/sec leakage rate. Leaktightness is mentioned in numerous places in this chapter.

Response - The term "leaktight" was improperly used in the section on containment. Per ANSI N14.5, the leaktightness criteria is to be used when the leakage rates corresponding to the regulatory release rates criteria are too small (under 10^{-7}) to be demonstrated by standard tests. In this case, leaktightness is defined as a leakage rate not exceeding 10^{-7} cm^3/s, and satisfaction of the leaktightness criteria is considered to meet the release rates criteria. For BR-100 containment evaluation, we intend to demonstrate a maximum leakage rate criteria of 10^{-3} cm^3/s, this value being below the maximum permissible leakage rates determined per ANSI N14.5 (5.3).

G.18 Recommend testing of the seals at high temperatures (both maximum normal and accident condition). Manufacturer temperature specifications often are not based on ANSI 14.5 leakage criteria.

Response - Seal testing will be conducted over a range of temperatures from -40°F to temperatures anticipated during a fire accident. This is explained in Section 2.3.4 of the text.

G.19 Part 1: AAR specification M-1001 requires specific testing of railcar designs. The specified testing should be discussed.

Response - Part 1: Railcar Testing:

The amount of testing required on the BR-100 railcar will be determined by the AAR Car Construction Committee, which is one of eight standing committees of the Mechanical Division of the AAR. Certification of a new car design by the AAR is not required unless the design is for a car of an "untried type". Since the Car Construction Committee is the judge of whether the car is new, varying amounts of analysis and testing may be required. We will not know which category our design falls under until application for certification is made.

Part 2: Comments and recommendations associated with the aluminum basket design and impact limiter design were documented in Attachment 1. Additional testing is recommended for the thermal/neutron shield. Topics to be addressed include reliability of the fusible plug that allows steam to escape, maintaining the water content during the life of the cask, and non-destructive examination methods that would assure the proper water content is initially poured.

Response - Part 2: Thermal/Neutron Shield Testing:

a) Use of fusible plugs is common with casks licensed by the NRC and is a proven technique. Also, the acceptance test procedure requires leak testing of each fusible plug.
b) The welds used to seal the concrete cavity will be qualified welds, and both fusible plugs and welds will be fabricated under a QA program.
c) A QA program and procedures will control the procurement of concrete constituents and concrete mixing to assure the proper water content.

B&W has developed a very comprehensive testing program about the thermal/neutron shield constituents, which has been discussed with NRC. This testing program should allow a successful licensing of the BR-100.

G.20 Part 1: The proposed aluminum alloy, 6061-T6511, is included in ASME, Sec. III as a Class 3 material. NRC Regulatory Guide 7.6 recommends using Class 1 materials where possible. The principal difference between the two classes of materials is the level of acceptable stresses that the ASME code allows. For Class 1 materials, an allowable stress under accident conditions is 3.6 times the design stress intensity (Sm). For Class 3 materials, the allowable stress under accident conditions is 2.4 times the design allowable stress (S). Although Reg. Guide 7.6 lists the allowable stress under accident conditions as 3.6 x Sm, the limits are predicted on the use of Class 1 materials. The use of a Class 3 Material and the use of a Class 1 material allowable stress multiplier (3.6 Sm) for a Class 3 material are not specifically covered in Reg. Guide 7.6.

Response - Part 1: NRC Regulatory Guide 7.6 is for "Design Criteria for the Structural Analysis of Shipping Cask Containment Vessel." It does not address noncontainment structures. The aluminum fuel cells are noncontainment structures. For Class 1 component material, accident condition allowable stress limits are $3.6S_m$ (S_m is design stress intensity as defined in ASME Section III). Aluminum 6061 is included in ASME Section III as a Class 3 component material. Our intent is to meet the requirement of Class 1 component material for aluminum 6061 such that we can use Class 1 component stress limits. The aluminum 6061 fuel cells are not ASME Section III Code component, and ASME Code is used as a good design practice. We have discussed this subject with the NRC, and they have not identified any concern for using aluminum for fuel basket application.

Part 2: Reg. Guide 7.6 specifically excludes the evaluation of brittle materials. Aluminum can fail in a brittle manner. Valid ASTM E399 measurements have been made which show K_{Ic} = 26.5 ksi in @ 100°F. The NRC has provided limited guidance for brittle fracture acceptance criteria; but only for a select class of ferritic steels.

Note: It is recognized that NRC Reg. Guide 7.6 applies specifically to containment vessels. However, since there is no published criterion for basket design, the NRC philosophy towards structural analysis/design, as provided for in Reg. Guide 7.6, is applied to basket design for these comments.

Response - Part 2: Even though Regulatory Guide 7.6 does not specify brittle fracture evaluation, we have performed the fracture mechanics evaluation for aluminum 6061. The results are favorable and are presented in Section 2.7.3 of the report.

G.21 Part 1: A preliminary equivalent static analysis has been performed for a single fuel cell which resulted in a maximum stress of approximately 400 psi. Preliminary equivalent static stresses are estimated by relating the 400 psi stress at a level of 1g and linearly scaling the stress upward with increasing g loads. Recognize that this analysis was performed without impact limiter data. For example, if it is assumed that the impact limiter design was to limit the cask g loading to 20 g's, the preliminary stress calculations would indicate stresses of approximately 8000 psi.

B&W is well aware of the low ASME design stress allowables for this material at operational temperatures (S = 4,400 psi @ 400F.). For accident conditions, the allowable stress (according to ASME Section III, Subsection ND) is 2.4 x S = 10,560 psi.

Response - Part 1: For fuel cell analysis results, refer to sections 2.6.6 and 2.7.1.

Part 2: B&W's basket design uses individual fuel cells that are stacked within the cask. B&W used a three-dimensional equivalent static analysis of a single fuel cell to estimate the basket fuel cells. The 3-D analysis was used to account for the non-uniform pressure distribution resulting from the fuel cell transmitting its load to the basket.

It is suggested that a non-linear dynamic analysis of the entire basket should be performed to define the assumptions used in the 3-D single fuel cell analysis. (A non-linear analysis will appropriately assess the interaction of the multiple contacts that result from the individual fuel cells contacting each other). A plane strain analysis representing a section of the basket and fuel can be used to verify the assumption in the linear elastic equivalent static analysis. Questions that are difficult to answer using only a static, linear-elastic analysis include:

- How does the stress distribute across the fuel cell contacts?
- How does the relative stiffness of the fuel cells affect the stress distribution?
- How do gaps between the basket cells affect the stress distribution?
- How are stresses distributed in the basket as a function of rotational orientation (e.g. analyze the basket through a number of rotated angles)?
- What is the margin against dynamic instability (buckling) of the fuel cells for different basket orientations?

The equivalent static analysis can be used to design and certify the basket; however, a non-linear transient dynamic analysis will provide the basis that defines the equivalent static analysis.

Response - Part 2: The fuel basket assembly will be analyzed in the final design.

Part 3: It is suggested that buckling be evaluated using the ASME N-284 code case approach.

Response - Part 3: Buckling will be evaluated per ASME Code Case N-284 in the final design.

Part 4: The internal four corners of the individual fuel cell are machined to a very tight radius (0.031 in.). This is a location for a stress riser. A model using a refined finite element mesh at these corners would capture both the stress gradient and the maximum stress value. Also, since it is a sharp notch, a fracture mechanics evaluation would assess the brittle fracture potential.

Response - Part 4: We have kept the option to modify the PWR fuel cell corners if final design analyses identify stress risers in the corners.

G.22 Brittle Fracture - It is recommended that the issue of brittle fracture of aluminum be evaluated using a fracture mechanics approach.

Response - Results of fracture mechanics analysis are provided in Section 2.7.3.

G.23 Testing - Current test plans call for drop testing an individual fuel cell. It is recommended that a drop test include bundled fuel cells. The interaction of fuel cells and payload is very difficult to analyze. A drop test of bundled cells would provide confidence in the analysis and in the 1/4-scale prototype test.

Prototype testing of a 1/4-scale basket model may be difficult since the resulting scale model would require extruding the aluminum basket walls to a thickness of 0.0625 inches.

Response - The quarter-scale design verification test model will include fuel basket with bundled fuel cells.

Interactions with Alcoa Extrusion/Tube Division have confirmed that quarter-scale extrusions can be produced from which fuel cells can be machined to produce true, quarter-scale final dimensions. We have been working with experienced manufacturing personnel within B&W to ensure the manufacturability of all components.

G.24 B&W's impact limiter design uses balsa wood and redwood confined by a composite Kevlar outer skin. The geometry of the wood is very similar to the geometry used in the TN-BRP impact limiter, and also used by ROBATEL, France. The use of a Kevlar-epoxy composite as a confining membrane for impact limiters is new, and B&W intends to design by testing and analysis. The impact limiter analyses considered to date by B&W are very similar to the analysis performed by Transnuclear on the TN-BRP impact limiter.

As a recommendation, there are several issues that the test program or analysis program should evaluate: Larger scale testing or even full scale testing of the impact limiter may be required to answer some of these concerns.

Part 1: The relatively short insertion length for the impact limiter will require large attachment forces for a slapdown drop orientation. The proposed attachment with Kevlar loops is an unproven method, and should be carefully examined.

Response - Part 1: The Kevlar attachment is fabricated of unidirectional material with the fibers orientated parallel to the load. This is an ideal application of Kevlar. The Kevlar attachment will be well characterized in the testing program.

Part 2: If the Kevlar-epoxy composites are less stiff than steel, then less energy will be absorbed by the outer shell. In the side impact, the reduced stiffness of the Kevlar could result in bottoming-out of the impact limiter leading to high accelerations as the result of trunnion impact.

Response - Part 2: The impact limiter is designed with the wood as the sole energy absorbing material. B&W calculations and testing have shown full energy absorption prior to a deflection which would allow trunnion impact.

Part 3: Kevlar-epoxy composites are not as ductile as stainless steel, and the possibility of rupturing the impact limiter casing at a local buckle should be examined.

Response - Part 3: Local buckling has been observed in the impact limiter test specimens. However, local buckling does not degrade the performance of the material. We will continue to investigate the material behavior and consequences in our test program.

Part 4: The performance of the impact limiter in the thermal accident should be examined. The Kevlar may not provide an adequate thermal/oxygen barrier to prevent wood ignition, especially if there is a local rupture.

Response - Part 4: Kevlar will not be used as a thermal shield. Special materials and alternative designs are being investigated to provide thermal protection.

Part 5: Kevlar is sensitive to environmental attack. Exposure to acids, heat, sunlight, and moisture could damage Kevlar. Performance of the Kevlar over a long term should be examined.

Response - Part 5: The BR-100 impact limiter will be designed to prevent material degradation due to environmental attack. Outer coatings are being investigated to protect the Kevlar from environmental damage.

Part 6: Caution should be used in interpreting any static impact limiter test since the dynamic behavior could be different from the static test results.

Response - Part 6: The impact limiter test program includes vigorous dynamic as well as static testing. Correlations of dynamic and static results will be addressed as the testing program progresses.

G.25 Dynamic tests on 1/4- and 1/2-scale impact limiters on mass models are recommended.

Response - B&W is planning to perform several dynamic tests on 1/4-scale impact limiter models. In addition, we are doing dynamic tests on different sizes of material specimens to quantify any scaling effects on the Kevlar/wood combination.

H.1 For better reading, Figure 5-4 should be placed closer to Table 5-4.

Response - Tables and figures were separated due to standard formatting of all sections of the PDP. In this case, the separation of Figure 5-4 and Table 5-4 certainly causes the reader an inconvenience. These pages will be brought together.

H.2 Points 3, 6, 9, and 12 of Figure 3-4 and Table 5-4 are near the limit of 10 mrem/h. An estimate of the calculational uncertainty (due to hand calculations, one-dimensional assumptions, etc.) on these values would be very useful.

Response - Addressed in A.31.

H.3 The doses at points 6 and 9, Figure 5-4 and 5-4 are large compared to points 3 and 6. I assume this is due to the Co-60 concentration in those regions. There appears to be a gap in the lead and concrete shielding at the edge of the top shield plug and the cylinder wall. Therefore, a dose mapping from points 3 to point 6 might reveal a streaming path near the top corner which has a dose larger than the 10 mrem/hr limit.

Response - Concerns addressed in A.24 and E.15.

H.4 Page II-3-11, par. 3, line 5 - I believe the spectra would come from ORIGEN2 and not ANISN.

Response - Page II-5-11, par. 3, line 5 has been corrected. Correct reference number is 4.

H.5 Page II-5-12, par. 2, line 3 - Flux profiles used in the activation calculation should be given.

Response - PDP text on Page II-5-12, par. 2 has been changed to clarify how the flux profiles were used in the hand calculations.

H.6 Page II-5-12, par. 2, line 8 - Homogenization factor should be given and how calculated.

Response - The standard and homogenized densities for the end fittings and spacer grids are given in Table 5-10. A reference to the table and an explanation of "homogenization factor" has been added to the PDP text on Page II-5-12.

H.7 Page II-5-17, par. 6, last sentence - the source-length correction factor should be given and how calculated.

Response - The PDP text on Page II-3-17, par. 6 has been changed to explain the source-length correction applied.

H.8 Page II-5-17, par. 7 - The QAD geometric factors should be given.

Response - The QAD geometric correction factors, as applied to the ANISN infinite slab, are a function of axial position. For the design limiting position 2 meters from the impact limiter, the factor is 0.0115. The QAD geometric factors are not being incorporated into the PDP text since they alone have little meaning without the QAD model, its dimensions, and the ANISN output values. These calculational details are beyond what is normally included in a preliminary design report.

H.9 Page II-5-23, par. 1, lines 5-9 - I do not understand these two sentences. The wording implies that ANISN is 2-D when it states that the peaking factor was applied at the fuel midplane. If ANISN used a source that had a 1.13 peaking factor applied, then it would not underestimate, but overestimate, which would be conservative as indicated.

Response - The 1.13 power peaking factor was applied to all source terms input to ANISN, both neutron and gamma. This does not simulate a 2-D calculation, it only increases the dose rates to the value corresponding to the highest burnup. Wording on Page II-5-23, par. 1 of the PDP has been changed to clarify the effect of decreased burnup at the ends of the fuel assemblies.

H.10 Table 5-12 shows the flux-to-dose conversion factors for the gammas in the QAD calculations. A table of the neutron and gamma (if different from QAD) factors used in ANISN would be useful.

Response - Flux-to-dose conversion factors used in ANISN are based on the ANSI/ANS-6.1.1 (1977) Standard values and have been incorporated in the PDP as Tables 5-13 and 5-14.

I.1 Neutron Quality Factor - Current DOE orders and NRC regulations allow conversion of dose rates in rad to rem using QF factors as listed (Page 16) in NRCP report #38 dated January 1971. A new ICRP report is expected within a year with new QF. Is the cask design now incorporating the new standards?

Response - BWFC recognizes that a change to the neutron quality factor for conversion of dose rates in the future could cause shielding revisions relative to 10CFR71 criteria. However, the NRC has informed BWFC that the Committee Interagency Radiation Research & Policy Coordination (CIRRPC), which is studying the change recommendation, has decided against modifying the thermal and fast neutron quality factors. Arbitrary inclusion of the higher quality factors would translate into a heavier cask or less payload. BWFC has not elected to make such a change during this design phase.

I.2 ALARA - The cask handling procedures at this point are very general and do not indicate a formal concern for personnel dose minimization. In fact, it would be well advised to give some design thought to auxiliary platforms, shadow shielding, etc., to provide for ease and speed of handling and keeping personnel exposures low.

Response - As stated in the comment itself, procedures are very general at that stage of the design. Personnel dose minimization was taken into account in the design of the two-piece closure lid and in the use of the protective ring and fixture plate, for instance. The first goal of the Preliminary Design is to define a cask with a maximum payload within the limits of the applicable contractual regulations. More consideration will be given during final design to the dose rate minimization with the use of proper tooling, additional portable shielding, and limitation of the activities performed on the cask itself (essentially bolting and unbolting of the closure lid).

J.1 The cermet plates are very important for safety, and their presence must be assured. A statement should be made as to the projected lifetime of the Cermet plates and the methods used to insure the presence of these poisons over the lifetime of the package (PRG, Sect. 8.1, p. 8-3).

Response - At present no explicit calculations have been made of the burnout of the boron in the cermet plates. An estimate of the lifetime will be made during the final analysis of the cask. However, the boron density in the cermet is similar to that in Boral shrouds used for criticality control in spent fuel storage pools. The flux density in the cask will also approximate that in the storage pools. Based upon the spent fuel pool experience, the boron lifetime in the cermet should greatly exceed the projected useful life of the cask. The mechanical integrity of the plate is discussed in the response to question C9. The method of attachment of the plate to the fuel cell is under development. Structural analyses to ensure retention of the plate for operation and accident conditions are to be done during the final design.

J.2 Flux Trap - In the computer model the flux trap between cermet plates is filled with aluminum. This is conservative when compared to a water-filled gap. Check calculations show a void gap to have almost the same impact on reactivity as an aluminum-filled gap.

Response - For the flooded case analyses, the gap between cermet plates is filled with water, not aluminum. The following phrase in Section 6.3.1.1, ": 1)replacement of the small water gaps between the cermet plates and the extruded aluminum cell with aluminum, a conservative assumption, ..." has been replaced with ": 1) complete contact between the cermet plate and the walls of the fuel cell, i.e. no gaps included for plate insertion, ..." This should clarify the modeling assumption. The remainder of the comment is correct regarding aluminum replacing water and the void versus aluminum-filled gap.

J.3 Optimal Moderation - The criticality evaluation considers the worst case moderation to be full water flooding within the cask and no water outside. Check calculations show almost no difference in k_{eff} when the volume between casks is fully flooded as when void. This indicates relatively little interaction between casks.

Partial flooding of the cask interior when a cask is drained of its water. This creates a configuration in which the upper layer of assemblies are free of water, while the lower levels are flooded. This configuration is somewhat more complex to model and is not calculated, but it is expected to result in a decrease in k_{eff}. A check calculation is made in which the density of water is uniformly reduced to 0.90 g/cm^3, and k_{eff} is found to decrease. Since the rod lattics in the assemblies are under-moderated, k_{eff} should continue to decrease as the water density is decreased.

Response - The comment concerning "relatively little interaction between casks" is expected. Due to the close spacing of the casks in the model, a void between the casks in an array is only slightly larger than a case with an interspersed moderator. Regarding a partially flooded cask and vapor conditions, the reactivity for the extremes of fully flooded and air filled show the decrease in reactivity as the moderator is removed. An intermediate reactivity higher than the fully flooded case for reduced water level and/or low density(vapor) moderation are not expected. The vapor condition will be examined in the final design, but the poison plates and close-packed finite array generally removes the "second-peak" phenomena.

J.4 The criticality evaluation states that the bounding PWR assembly is the Westinghouse 17 X 17 assembly with 264 fuel rods containing 4.5 wt% enriched UO_2. The bounding BWR assembly is the ANF 8 x 8 JP-4,5 assembly with 62 fuel rods also containing 4.5 wt% enriched UO_2. These choices are not evaluated as a part of this review.

Poison plates are composed of Al/B_4C cermet material with a B^{10} real density of 50 mg/cm^2. The isotopic composition of Robatel concrete (1.61 g/cm^3) is not specified in the analysis.

KENO IV input and output were not available for review. Inclusion of a KENO IV input listing in the final report is recommended.

A comparison calculation for one cell (fuel bundle in a hole in a neutron poison basket) should be presented between a homogenized fuel cell and a discretely modeled fuel cell. This model should include the neutron poison basket modeled discretely (PRG, Sect. 8.2.3.1, p. 8-4). This provides a check on the accuracy of using homogenized cross section sets.

Table 1 shows a comparison of atom densities for materials used in the criticality evaluation and for fresh fuel calculated for this review. Although close, the atom densities do not show precise agreement. The difference between fresh and 16 gWd/mtU PWR fuel is not independently verified.

Table 1

ATOMIC DENSITIES FOR MATERIALS

Mat'l	Density	Volume Fraction	Section 6.0 Het at/b-cm (Fresh Fuel)	My Values Homo at/b-cm (Fresh Fuel)	Section 6.0 Homo at/b-cm (16gWd/mtU)
PWR Fuel					
UO_2	10.31	0.3044			
U-235	0.408		1.0458-03	3.1834-04	2.0106-04
U-238	8.661		2.1914-02	6.6712-03	6.0520-03
O-16	1.220		4.5919-02	1.3978-02	see water
Zirc-2	6.56	0.0970			
Zr	6.448		4.2568-02	4.1278-03	4.1151-03
Sn	0.0918		4.6593-04	4.5181-05	4.0381-05
Fe	0.0138		1.4856-04	1.4406-05	1.2875-05
Cr	0.0066		7.5982-05	7.3680-06	6.5850-06
Water	1.00	0.5863			
H	void =	0.0123	6.6854-02	4.0019-02	4.1045-02
		0.5986	3.3427-02	2.0009-02	3.3029-02
			Section 6.0 Het at/b-cm	My Values Homo at/b-cm	Section 6.0 Homo at/b-cm
Fresh BWR Fuel	(0.641-in.	pitch)			
UO_2	9.98	0.3213			
U-235	0.395		1.0131-03	3.2551-04	3.1536-04
U-238	8.390		2.1229-02	6.8209-03	6.6082-03
O-16	1.182		4.4484-02	1.4293-02	see water
Zirc-2	6.56	0.1081			
Zr	6.448		4.2568-02	4.6016-03	5.2489-05
Sn	0.0918		4.6593-04	5.0367-05	5.7456-05
Fe	0.0138		1.4856-04	1.6059-05	1.8318-05
Cr	0.0066		7.5982-05	8.2136-06	9.3692-06
Water	1.00	0.5706			
H			6.6854-02	3.8173-02	3.6876-02
O			3.3427-02	1.9087-02	3.2285-02

Response - The 17x17 Westinghouse PWR fuel assembly was determined to represent the bounding PWR assembly. The ANF 8x8 was chosen as a representative BWR assembly but an independent assessment to determine a bounding BWR assembly will be done during the final analysis. Due to the relatively low reactivity of the BWR basket configuration, it appears that it is not the limiting configuration and thus a bounding BWR assembly determination will have little impact on the cask reactivity limits.

The isotopic composition of the Robatel concrete has been added to the material list is Section 6.3.2.

A KENO IV input listing will be included in the report for the final design analysis.

Past experience has shown that the homogenization technique provides slightly conservative results. Verification of this will be done during the analysis for the final design.

The difference between the number densities for the PWR fuel assemblies is due to the inclusion of the 24 guide tubes and 1 instrument tube in the homogenization technique. This results in the following volume fractions: fuel, 0.2781; zirc, 0.09668; moderator, 0.6140. For the BWR fuel assemblies the differences are due to inclusion of the capture rod and the inert rod. The resulting volume fractions are 0.3113, 0.1233, and 0.5516 for the fuel, zirc, and moderator respectively.

J.5 A calculation should be provided for a worst case, showing the k_{eff} of the package modeled with unirradiated fuel (PRG, Sect. 8.2.3.2, p. 8-5). Section 6.0 provides a calculation for an air-filled cask containing fresh fuel. However, since this cask is normally loaded and unloaded underwater, a calculation for a water-filled cask should also be provided, or an explanation made as to how water is excluded. The possibility of fuel irradiated less than expected should be considered. If fresh fuel is shipped, it must be assured that the cask would not be improperly placed underwater for unloading.

No check is made of burnup isotopics or of the axial distribution of burnup.

<u>Response</u> - The worst case calculation essentially has been included in the report. The case for the flooded cask loaded with burned PWR fuel assemblies at 4.5 weight percent U-235 enrichment essentially represents this case. The qualifier is added since tolerance and calculated accident deformations have not been factored into the preliminary design analyses. The cask loading currently has the restraint of 18 GWD/mTU average burnup for PWR assemblies. This will be assured either by administrative controls and/or measurement. During the final design analysis, the impact of misloading a lower burnup assembly into the case will be examined. If fresh fuel is shipped, the cask capacity will be downrated to about 17 fuel assemblies. It is expected that blocks will be inserted in prohibited locations to define the required loading pattern. The calculation for the case with fresh fuel in an air-filled cask was included to indicate the reactivity margin for the normal shipping condition of the cask. Fresh fuel was included to provide a maximum reactivity; burned fuel has a k_{eff} about 2 percent less than that for fresh fuel.

J.6 At least one confirming calculation should be made for a worst case in which the components in the package are modeled at temperatures determined from the thermal evaluation. The k_{eff} results from this model should be lower than those results obtained for models at room temperature. If this is not the case, more calculations are required to find the peak k_{eff} for all possible thermal conditions (PRG, Sect. 8.2.4, p. 8-6).

Administrative controls required to insure that checkerboard loading is done properly must be administered with a high level of reliability.

Response - The effect of the moderator temperature will be evaluated in the final design. Experience with spent fuel storage racks indicates a very flat curve of reactivity versus moderator temperature in the region from 50 to 150°F. Generally the 50° value provides the maximum reactivity. This will be confirmed in the final design analysis.

J.7 In the computer model fuel assemblies are homogenized, a process which is non-conservative unless correction is made to the cross section sets. Correction for self-shielding has been made using NITAWL and XSDRNPM, both well-established cross section processing codes. This method should accurately account for the effects of homogenization. No review is made of input to NITAWL and XSDRNPM.

Response - [NITAWL/XSDRNPM cross section weighting] - agreed.

J.8 Fabrication tolerances, off-center fuel assembly placements, accident deformations, and axial details for the basic contents/basket arrangement are not included in the criticality evaluation, but are to be considered in the future. Check calculations indicate that shipment of partially disassembled, or damaged, assemblies potentially might increase the maximum achievable k_{eff}.

Response - As noted, tolerance effects, off-center placement, final accident deformations, and axial details for intact and consolidated fuel assemblies will be examined in the final design analyses. It was understood that the normal configuration of the cask will accommodate either intact fuel or consolidated canisters. Partially disassembled and/or damaged fuel is a special category that will be examined in the future and may require derating of the cask to handle the potential increase in k_{eff}.

J.9 The formula for k_{max} uses the one-standard-deviation uncertainty in the bias. If k_{max} is to be at the 95% confidence level, it should be based on the 95% confidence level uncertainty in the bias and the 95% confidence level in the KENO IV convergence. The correct value of k_{max} appears to be:

sigma = 0.0050 (KENO IV convergence uncertainty)
bias uncertainty (U) = 0.0075 (see comments below, section 6.5.2)

$k_{max} = k_{eff} + 0.0103 + ((2 \times sigma)^2 + (2 \times U)^2)^{1/2}$

$k_{max} = k_{eff} + 0.0103 + (0.0100^2 + 0.0150^2)^{1/2}$

$= k_{eff} + 0.0283$

Response - The formula presented in Section 6.4.3 was incorrectly stated. As noted in the comment, a factor of 2 should have been applied to the KENO-IV bias uncertainty value of 0.0049. This has been corrected in section 6.4.3. The k_{max} values in Table 6-5 have also been corrected using the revised formulation.

J.10 Bias uncertainties shown in Table 6-7 are based solely upon the KENO IV convergence uncertainty. This uncertainty does not include uncertainties due to geometric or material representations or to cross sections. This uncertainty should reflect agreement between calculated values and experimental values. The correct uncertainty in the bias is the root mean square variation in the difference between the calculated and experimental values of k_{eff}. Table 2 below shows a comparison of values from Table 6-7 and root mean square values.

Table 2

UNCERTAINTIES IN BIAS

Spacing in.	Number of Experiments	Table 6-7 bias + Unc.	New Value bias + Unc.
0.000	2	0.0025 ± 0.0051	0.0025 ± 0.0075*
0.644	8	0.0010 ± 0.0049	0.0010 ± 0.0075
1.88	6	-0.0040 ± 0.0050	-0.0040 ± 0.0075
1.932	4	-0.0105 ± 0.0047	-0.0105 ± 0.0065
2.576	1	-0.0160 ± 0.0050	-0.0160 ± 0.0075*
average	21	-0.0033 ± 0.0048	-0.0033 ± 0.0087

* There is insufficient data at 0.000-in. and 2.576-in. spacings for good value. Value of 0.0075 is based on uncertainties at other spacings.

Response - B&W agrees that the root mean square variation method would provide the best estimate of KENO bias uncertainty. During the final design all the benchmark cases will be executed on the Data General machine to provide a suitable data base for the method. Currently only the benchmark cases that most closely approximated the cask configuration have been executed on the Data General. Based upon these cases the current method seems appropriate and will be used in the PDR. However, it appears the bias uncertainty may be underpredicted. This will be checked during the final design analyses.

Preliminary Failure Modes and Effects Analysis

for the BR-100 Spent Fuel Shipping Cask

1. Introduction

A Failure Modes and Effects Analysis (FMEA) will be used to examine reliability and safety considerations of the preliminary design of the BR-100 Spent Fuel Shipping Cask. The objective of this analysis is to provide a systematic, qualitative review of the design to identify the effect of failures on operability and maintainability.

The FMEA method uses inductive reasoning to ensure that the effects of all components and their failures modes are examined. An appropriate level of detail is chosen, and all "components" at that level are enumerated to produce a mutually exclusive and complete rendering of the entire "system" under study. For each component, a complete set of failure modes is specified, and the effect of each failure mode on the system is determined. From this, each failure mode can be judged for criticality, based on the likelihood of the failure mode and the serious of its consequences.

2. Assumptions

Several assumptions were required to perform this analysis during the preliminary design phase. First, it was assumed that all parts of the cask have been manufactured and assembled correctly. Second, it was assumed, due to material testing, that no material defects exist in completed cask. Finally, it was assumed, due to the routine maintenance performed during loading operations, that wear-out of gaskets is not a credible failure mode.

These assumptions will limit the possible causes for a failure mode to human error and random failure. For example, common mode failure, due to a material defects, will not be considered.

3. Analysis

The scope of the analysis is limited to the cask. Since, at this stage of the preliminary design, the ancillary equipment is not detailed enough, it will not be considered in the FMEA. A list of the ancillary equipment is given in Table 1. For similar reasons, the Impact Limiters were not included in this FMEA.

When evaluating the effect of a component's failure mode on the BR-100, the basic concern was whether the cask could successfully perform its function of transporting fuel rods while maintaining the integrity of the cask with no radioactive releases to the environment.

The BR-100 can be broken down into six sub-assembles: Basket, Cask Body, Shield Plug, Lid, Impact Limiters, and ancillary equipment. The latter two sub-assemblies were not considered in this analysis. The first four sub-assemblies were divided into components, as shown in Table 2. These components represent the level of detail for which the FMEA was conducted.

Table 2 details the information obtained during the analysis. The table is divided into eight headings. These are discussed individually below:

Component: the components of the sub-assemblies for which the failure modes were investigated.

Usage: the mode in which the components are being used or are of concern; these are recorded on the table as:

- O - operations, the cask loading, unloading, and transportation operations,
- R - routine maintenance, performed during the cask loading operations,
- M - periodic maintenance, done at intervals of one year or greater when the cask is empty; the Basket is removed during this maintenance.

Failure Mode: the failure mode(s) of consideration for each component.

Likelihood: the qualitative likelihood of occurrence of the failure mode; these are recorded on the table as:

- ϵ - extremely low likelihood of occurrence
- L - low likelihood of occurrence
- M - medium likelihood of occurrence
- H - high likelihood of occurrence

These likelihood condition are roughly equivalent to the condition categories used in ANSI N18.2-1973 (Nuclear Safety Criteria for the Design of Stationary Pressurized Water Reactor Plants). Note that ANSI N18.2-1973 has been superceded by ANSI/ANS-51.1-1983, and that the condition categories have been replaced by plant condition categories. Nonetheless, the condition categories help to provide a frame of reference for these occurrence likelihoods. Note that these condition categories refer specifically to the operation of a nuclear power plant.

Condition I (normal operations, H) - expected to occur frequently or regularly in the course of power operation, refueling, maintenance, or maneuvering of the plant.

Condition II (incidents of moderate frequency, M) - expected to occur during a calendar year for a particular plant.

Condition III (infrequent incidents, L) - expected to occur during the lifetime of a particular plant.

Condition IV (limiting faults, ϵ) - not expected to occur, but are postulated because their consequences would include the potential for the release of significant amounts of radioactive material.

Operational Effect: the effect that the failure mode will have on the operability of the cask.

Is Cask Usable?: the term "usable" is restricted to the regulatory sense; in most cases, the cask will be physically usable, but not accordingly to the existing regulations. When the cask is being subjected to periodic maintenance, and there is no operational effect, then this question is moot.

Postulated Caüse: the most likely reason(s) for the failure mode; this can help focus procedures in the appropriate areas to reduce cask unavailability.

Comments: appropriate comments about any of the entries in the table.

4. Conclusions

There is only one failure mode whose likelihood of occurrence is medium or high: surface scratches on the fuel cell. The impact of this failure mode is not yet known. Tests are being conducted; when complete, they will have to be carefully examined to determine whether it is necessary to develop methods to prevent these surface scratches.

Almost all the failure modes are failures of consequence (i. e., affect the operability of the cask), since there is no redundancy designed into the cask (except maybe for the trunnion bolts). However, the likelihood of occurrence is low or extremely low in all instances. In addition, almost all cases where the likelihood is low, the cause can be attributed to human error (generally poor maintenance practices). If this is understood in advance, maintenance procedures can be written to specifically avoid the situations identified by this FMEA. Further, judicious stocking and availability of spare parts will ameliorate many of the human maintenance errors. The stocking program can also benefit from the results of this FMEA. When the final FMEA, based on the final cask design, is completed, it (the FMEA) will be used in the development of operational procedures and a spare parts list.

Table 1 Ancillary Equipment (Not Included in FMEA)

Lifting Yoke
Skid
Railcar
Protective Ring
Fixture Plate
Dewatering Equipment
Drying Equipment
Inerting Equipment
Personnel Barrier
Contamination Control and Removal Equipment

Table 2 Spent Fuel Shipping Cask BR-100 FMEA Results

Component	Usage O¦R¦M	Failure Mode	Likeli-hood	Operational Effects	Is Cask Usable?	Postulated Cause	Comments
BASKET:							
Fuel cell	O	Surface scratches	M	Unknown	Yes	Sliding of fuel elements in and out of cell	This is an expected operational occurrence. Tests are being conducted to determine how deep of a scratch will impact the cell's functionality.
•	M	Structural deformation	L	None	---	Human error	Fuel cell could be damaged by dropping it, or dropping something on it during maintenance.
Cermet plate	O	Fracture	ϵ	None - the cermet plate's function is to absorb neutrons to prevent criticality	---	Vibration during transport; thermal cycling	No human error is possible since plates are not accessible during operation, and are not handled during maintenance. Can only be discovered during periodic maintenance. The final design will hold the cermet plate in place, even in the event of fracturing.
	O	Peeling	ϵ	None - see comment above	---	Thermal cycling	Testing is currently under way to investigate the effects of thermal cycling.
Former	M	Structural deformation	ϵ	None	---	Human error	Human error might involve the dropping of a piece of heavy equipment on a former.
Bottom plate	M	Structural deformation	ϵ	None	---	Human error	Same comment as Former.
Upper grid	O	Structural deformation	ϵ	Prevent free passage of one or more fuel cells	Yes	Human error or equipment failure	Human error might involve a dropped fuel assembly; equipment failure might be a crane failure.
Tie rod	M	None	-	None	---	-----	Tie rods are stainless steel and hand-loaded.
Steel band	M	None	-	None	---	-----	Steel bands are stainless steel and hand-loaded.

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Component	Usage O¦R¦M	Failure Mode	Likeli-hood	Operational Effects	Is Cask Usable?	Postulated Cause	Comments
CASK BODY:							
Cask		Concrete dry-out	ϵ	Degrade cask performance	Yes	Leaking fusible plug	These plugs are inspected before and after shipment. Thus, no significant amount of water can be lost.

Table 2 (continued) Spent Fuel Shipping Cask BR-100 FMEA Results

Component	Usage O¦R¦M	Failure Mode	Likeli-hood	Operational Effects	Is Cask Usable?	Postulated Cause	Comments
		Copper fin weld failure	ε	Degrade cask thermal performance	Yes	Hypothetical event, no postulated cause.	Copper fin welds have been tested by Robatel. No weld failures have been observed
Seal surface	O R M	Surface scratches	L	Loss of seal integrity	No	Human error	May be difficult to repair because cask is heavy (150,000 lb) and unwieldy.
Outer surface	O R M	Discoloration	ε	Degrade cask performance	Yes	Age	Paint must remain white to maintain thermal emissivity. A percentage of paint which must remain intact will be specified during the final design.
	O R M	Chipping/ scratching	L	Degrade cask performance	Yes	Human error, age	
Trunnions	O	None	ε	None	---	-----	
Trunnion bolt	M	Damage threads on bolt or in bolt hole	ε	None	---	Human error	
	M	Break/shear	ε	None	---	Age, over-torquing	
Fusible plug	O	Loss of seal	L	Water leakage	Yes	Age	Fusible plugs are easily replaced. (See Cask entry.)

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SHIELD PLUG:

Component	Usage O¦R¦M	Failure Mode	Likeli-hood	Operational Effects	Is Cask Usable?	Postulated Cause	Comments
Plug	O	Crack, fracture	ε	Reduction or loss of gamma shielding capability.	No	Human error	Human error might involve dropping the plug; no spares will be available.
Seal Surfaces	O R	Surface scratches	L	Impairs the draining operation	Yes	Human error	Small surface area and less weight (7900 lb) than the cask, so less difficult to repair (by hand or machine) than the cask body.
Threaded hole	O R	Clogged	L-M	Can not get bolt in	No	Human error	Improper maintenance, build-up of lubrication grease.
Vent/drain pipe	O	Clogged	<L	Inhibits draining operation	No	Human error, crud in the cask	Improper maintenance while Plug is off.

Table 2 (continued) Spent Fuel Shipping Cask BR-100 FMEA Results

Component	Usage O¦R¦M	Failure Mode	Likeli-hood	Operational Effects	Is Cask Usable?	Postulated Cause	Comments
Vent/drain pipe gasket	O	Physical deformation (i.e. cut)	L	Inhibits draining operation	No	Human error	Easily replaced.
LID:							
Lid	O	Deformation	ε	Can not close cask	No	Human error: dropped lid	Lid is large stainless steel disk.
Seal surface	O R	Surface scratches	L	Loss of seal integrity	No	Human error	Small surface area and less weight (7600 lb) than the cask, so less difficult to repair (by hand or machine)
Lid gasket	O R	Physical deformation (i.e. cut)	L	Loss of seal integrity	No	Human error: insufficient lube, gasket shelf life exceeded	Error could occur during routine maintenance. Spare gaskets should be available.
Cover plate	O	Unavailability	L	Loss of seal integrity	No	Human error	Only scenario imagined is dropping cover plate into the spent fuel pool. Small size may make it difficult to recover.
	O	Physical deformation	ε	Loss of seal	No	Human error	While cover plates are made of stainless steel, they may be damaged if dropped.
Cover plate gaskets	O R	Physical deformation (i.e., cut)	L	Loss of seal integrity	No	Human error	Error could occur during routine maintenance. Spare gaskets should be available.
Quick connection	O R	Valve fails open	L	Prevents the drying function and the ability to create a vacuum	No	Random failure	Should valve fail open during transport, seal integrity is maintained by cover plate and cover plate gasket. If valve fails during operation, it must be replaced.
	O R	Valve fails closed	L	Prevents complete sampling of the cask atmosphere before opening	No	Random failure	Should valve fail closed during transport, special steps for sampling the cask prior to opening would be required. If the valve failed closed during operation, it must be replaced.
Lid bolt	O R	Damaged threads on bolt or in bolt holes	ε	Loss of seal integrity	No	Human error, random failure, age	Bolt may be dropped and have its threads damaged.

Table 2 (continued) Spent Fuel Shipping Cask BR-100 FMEA Results

Component	Usage O¦R¦M	Failure Mode	Likeli-hood	Operational Effects	Is Cask Usable?	Postulated Cause	Comments
Threaded hole	O R	Clogged	L-M	Can not get bolt in	No	Human error	Improper maintenance while the Lid is off.
Cover plate bolt	O R	Damaged threads ε on bolts or in bolt holes		Loss of seal integrity	No	Human error, random failure, age	Bolt may be dropped and have its threads damaged.
Test port plug	[Currently not designed sufficiently to investigate failure modes.]						