72-17

Portland General Electric Company
Trojan Nuclear Plant
71760 Columbia River Hwy.
Rainier, Oregon 97048
(503) 556-3713

June 27, 1996

CPY-032-96

Mr. David Stewart-Smith
Oregon Department of Energy
625 Marion Street NE
Salem, OR 97310

Dear Mr. Stewart-Smith:

Response to Request for Additional Information

On May 29, 1996, Mr. Adam Bless sent a Request for Additional Information to Trojan that contained questions about the Trojan Independent Spent Fuel Storage Installation Safety Analysis Report. Attachment I of this letter provides the responses to those questions, with the exception of Questions 25, 40 and 95. Responses to these questions will be provided by August 29, 1996.

The responses provided in Attachment I are intended to answer the questions as completely and clearly as possible without divulging information that the cask vendor considers proprietary. Proprietary information has not been included in any of the responses.

If you have any questions concerning these responses, please contact me or M. H. Megehee of my staff at 503-556-7334.

Sincerely,

for C. P. Yundt
General Manager Plant
Support and Technical Functions

Attachments/Enclosures

c: A. Bless, ODOE
L. J. Callan, NRC, Region IV, w/o Enclosures
L. E. Kokajko, NRC, NMSS, w/o Enclosures
M. T. Masnik, NRC, NRR, w/o Enclosures
R. A. Scarano, NRC, Region IV, w/o Enclosures
J. Woessner, TAC

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Attachments to CPY-032-96

I. Responses to Request for Additional Information

II. Trojan Nuclear Plant FSAR, Figure 2.5-6

III. Storage Cask Dose Models (proprietary information deleted)

IV. PNL-6364, "Control of Degradation of Spent LWR Fuel During Dry Storage in an Inert Atmosphere," Figure 4.2

V. PNL-4835, "Technical Basis for Storage of Zircaloy-Clad Spent Fuel in Inert Gases." Abstract and Executive Summary

VI. "Characteristics of Potential Repository Waste," DOE/RW-0184, Selected printouts

VII. "Meteorology and Atomic Energy 1968," Figures 3.10 and 3.11

VIII. SNC Air Outlet Drawing (proprietary information deleted)

IX. PNL-7839, "Performance Testing and Analyses of the VSC-17 Ventilated Concrete Cask," Figure 4-9

June 27, 1996

Enclosures to CPY-032-96

1. Cases of ASME Boiler and Pressure Vessel Code, N-71-15

2. ASME Boiler and Pressure Vessel Code, Section II, Materials, Part D - Properties

3. ASME Boiler and Pressure Vessel Code, Section III, Rules for Construction of Nuclear Power Plant Components, Division 1 - Subsection NF

Question: 1

Does the Trojan ISFSI SAR incorporate by reference or otherwise accept or endorse SNC's SAR for the Transtor Shipping Cask System?

Response:

The Trojan ISFSI SAR is intended to be a stand-alone document, and does not incorporate by reference or otherwise endorse SNC's TranStor™ Shipping Cask System SAR. Section 1.5 of the ISFSI SAR will be changed to reflect this in a forthcoming revision.

The Baskets and Transfer Cask are the same components as described in SNC's TranStor™ Shipping Cask System SAR. PGE intends to register as a user of the TranStor™ Shipping Cask once a certification is issued.

Question : 2

10 CFR 72 requires that the cask system be certified for at least 20 years. Since it is likely that the ISFSI will be required for much longer than 20 years, are there steps that PGE must do to certify the cask system for a longer period?

Response:

10 CFR 72.42(a), (b), and (c) address the requirements and process for License renewal beyond the original term of certification. PGE evaluated proposed ISFSI designs against the 10 CFR 72 license renewal requirements before selecting a vendor to design and fabricate the Trojan ISFSI components. Sierra Nuclear Corporation has warranted their design for 50 years.

Question: 3

Section 1.2 references the Transtor Storage System by SNC. There are other references in the text to SNC's VSC-24 SAR. Which are the appropriate vendor SARs and reference documents for each of the major ISFSI components, including the equipment and components listed in Section 3.3.3.1?

Response:

PGE has retained Sierra Nuclear Corporation to provide design services for the Trojan ISFSI dry cask storage system. The major ISFSI components, including those listed in Section 3.3.3.1, are analyzed in the Trojan ISFSI SAR independently from any other Sierra Nuclear product. The Trojan ISFSI SAR identifies specific sections of references which are applicable.

Question: 4

Section 1.5 states that PGE incorporates SNC's Quality Assurance Program by reference. Has the implementation of this program, including aspects of all 18 criteria, been reviewed and audited by PGE's Nuclear Oversight Department? If so, please furnish a copy of the audit report.

Response:

In accordance with procedure QP 17-12, on January 3, 1996, the Sierra Nuclear Corporation (SNC) QA Manual, Revision 4, was reviewed against the 18 criteria and found to be satisfactory.

PGE Nuclear Oversight completed an audit (Audit Report 95-05) to assess the adequacy of SNC's QA Program and to determine the implementation effectiveness of their quality related controls. Certain criteria related to fabrication activities (VIII-XIV) were not specifically addressed during this audit because they are subcontracted by SNC. However, the audit did specify a requirement for PGE QA approval of quality-related subcontractors prior to use.

A copy of this Audit Report was previously made available for ODOE review in May 1996.

Question: 5

Section 2.1.2, p.2-3 defines the Controlled Area as the area that "immediately surrounds the ISFSI and extends out to 325 meters from the edge of the storage pad (Figure 2.1-2)." Since the boundary of this area is used for defining where the 10 CFR 72.104 limit of 25 mrem/year will not be exceeded, will this area be further identified by administrative or physical barriers?

According to Section 8.2.1.3, the selection of 325m as the radius for the Controlled Area Boundary is not a design basis accident. How is this a conservative boundary, since a radiation dose could be considerably greater (perhaps 10 times greater) than the limit at the 10 CFR 72 minimum distance of 100m and be within the limit at 325m?

Response:

10 CFR 72.104 specifies the permissible dose equivalent (25 mrem whole body) to a real individual located beyond the controlled area due to normal effluents and direct radiation. 10 CFR 72.106 specifies the permissible dose equivalent (5 rem whole body or any organ) from any design basis accident for an individual located at the controlled area boundary.

The Controlled Area will not be identified by physical barriers. Per the definition in 10 CFR 20, the Controlled Area is an area over which access may be limited. Limiting access by physical barriers is not required. The Controlled Area is shown in the proposed revision to the Trojan Nuclear Plant Permanently Defueled Emergency Plan (PGE-1060) submitted concurrent with the ISFSI SAR.

The Controlled Area boundary for the ISFSI was selected as 325 meters based on the radiological consequences of a failure of 100% of the fuel rods in a basket with a subsequent breach of the basket (ISFSI SAR Section 8.2.1). This accident is not credible because it cannot be caused by any known initiating event (e.g., drops, earthquakes, explosions, etc.). The other postulated off-normal events and accidents that are described in ISFSI SAR Chapter 8 have either small or no radiological consequences. If the Controlled Area were selected based on these other postulated off-normal events or accidents, and the 25 mrem/yr dose from normal operation, the boundary could be established at the 100 meter minimum required by federal regulations, although the radiological consequences at 100 meters from an off-normal event or accident would be much less than the limits stated in the federal regulations. Therefore, establishing the Controlled Area at 325 meters by using a non-credible accident as the basis is considered very conservative.

Question: 6

Section 2.1.2.2 describes evacuation of offsite facilities within the controlled areas. How quickly could such areas be evacuated, if necessary? How was this time limit verified?

Response:

An evaluation that supports the Trojan Nuclear Plant Permanently Defueled Emergency Plan estimates that members of the general public could be excluded or removed from the Exclusion Area, which is larger than the ISFSI Controlled Area, in 2 ½ hours. This capability has not been demonstrated because analyses of design basis and other accidents show that there is no need to exclude or remove members of the general public from the Exclusion Area. Similarly, no credible off-normal ISFSI events or accidents would require excluding or removing the general public from the controlled area.

Question: 7

Section 2.1.2 describes administrative limits that will apply to offsite facilities such as the Trojan North Building, the Training Building, etc. What are these "limits"?

Response:

The standard lease issued to commercial users of PGE facilities includes a section that specifies the allowed use of the leased facility, e.g., operation of mainframe backup computers. In that way, PGE knows what activities are taking place at the leased facility and can limit the lessees to those with activities that do not affect ISFSI operation.

Question: 8

Section 2.2.3.1 deals with explosions. In the event of a toxic gas release, would the ISFSI Specialists abandon the site?

Response:

The proposed revision to the Trojan Nuclear Plant Permanently Defueled Emergency Plan, which will cover the ISFSI, contains Emergency Action Levels for a toxic material release and general protective measures. Specific actions, such as ISFSI Specialists leaving the site, would be determined by the Emergency Coordinator based on the specific circumstances.

Question: 9

Section 2.3.2 describes Local Meteorology using data taken between 1971 and 1974. Please explain the relevance of this meteorological data, since such data has presumably been taken during the entire period of constructing and operating the Trojan Plant. The SAR states that the meteorological data compares "favorably" with data from Portland. What does "favorably" mean in this context?

Response:

The meteorological data collected at the Trojan site during construction and operation does not cover a long enough period of time to assure that meteorology extremes are accurately determined for use in the design bases of the Trojan ISFSI. Therefore, similarity with Portland, Oregon is established in order that the meteorological data from Portland, which covers a much longer period of time, may be used to determine the most limiting meteorological extremes.

Similarity between the Trojan site and Portland was established for licensing of the Trojan Nuclear Plant using meteorological data from 1971-1974. As the regional climate would not be expected to change in a 25 year period of time, the meteorology at the ISFSI site and Portland would still be similar and the meteorological extremes from Portland may be used for the design bases of the ISFSI. The meteorological parameters used in the design bases of the ISFSI are shown in ISFSI SAR Table 2.7-1.

Compares "favorably" as used in this context is intended to mean similar.

Question: 10

Section 2.3.4 gives Diffusion Estimates. What are the sigmas [horizontal and vertical dispersion coefficients] for the example given at 325m?

Response:

Separate horizontal and vertical dispersion coefficients need not be determined. The diffusion factor can be read directly from Figure 1 of Nuclear Regulatory Commission Regulatory Guide 1.25 (Safety Guide 25) "Assumptions Used for Evaluating the Potential Radiological Consequences of a Fuel Handling Accident in the Fuel Handling and Storage Facility for Boiling and Pressurized Water Reactors".

A copy of this regulatory guide was previously sent with other reference documents on June 11, 1996 (letter CPY-030-96).

Question: 11

Section 2.6.1 references a geophysical survey performed in 1972. Have studies been performed since then that change the assumptions or results of the 1972 study relative to the geology of the Trojan site?

Response:

No additional geophysical surveys or studies have been performed of the Trojan site geology since the work in the early 1970's. Other regional work may have been performed by others, but not specific to the Trojan site. Seismological studies, with respect to the Seismic Margin Earthquake, are discussed in ISFSI SAR 2.6.2.4.

Question: 12

Section 2.6.1.2 refers to drill holes used for geologic mapping. Please provide maps showing the location of these holes, particularly holes DH4,5,7, and 9.

Response:

Drill hole locations are shown in Figure 2.5-6 of the Trojan Nuclear Plant Final Safety Analysis Report. A copy of this figure is shown on Attachment II.

Question: 13

Section 2.6.3, item 6, states that "There is no fault within 5 miles of the site which has experienced movement since Pleistocene time." Was this intended to state "since prior to the Pleistocene time"? The Pleistocene is generally believed to have existed from 65 million years ago to approximately 10 thousand years ago. Earlier in the Section it was stated that "No evidence of post-Pleistocene surface displacement has been found in the area."

Response:

The two referenced statements are consistent and correct as stated.

References define the Pleistocene as the first epoch of the Quaternary Period of the Cenozoic Era, beginning about 600,000 to 1.5 million years ago and extending to about 10,000-12,000 years ago.

Question: 14

Section 2.6.6 states that "If Mount St. Helens were to have another tephra eruption similar to the May 18, 1980 eruption, only directed towards the ISFSI ... then the expected ash fall accumulation would be about 1.8 inches. Without reference immediately available, it seemed that the ashfall from that eruption was much greater than 1.8 inches. This passage does not appear conservative. Please respond.

Although total blockage of air inlets is addressed in Chapter 8 of the SAR, an additional issue if the effect of the ash on the air channels of the cask systems. How would the ash be removed from the air channels? If rain followed an eruption, would the resulting combination of ash and rain be a concern in terms of cleaning the air channels?

Response:

The ash accumulation of 1.8 inches at the Trojan ISFSI is based on the heaviest ashfall near Packwood, Washington of about 0.51 inches/hour. Packwood is about 34 miles northeast of Mount St. Helens and the Trojan ISFSI is about 34 miles west south west of Mount St. Helens.

If the ash accumulation covered the air inlets, the ash could be vacuumed, blown with air, washed with water, or swept out of the air inlets. If the ash mixed with rain, then washing with water or sweeping could be used.

Question: 15

Section 3.1.1.3 states that there are two limits on the contents of the Fuel debris Cans: 10 kg per basket and 20 Ci of Plutonium. Presumably the 20 Ci Plutonium is per basket rather than per Fuel Debris Can. 10 CFR 71.63 specified "20 Ci per package." Section 4.2.6 states that "A fuel mass limit of 10 kg per PWR Basket will be administratively controlled." And Section 5.1.1 states that debris will be "visually inspected as it is loaded to verify that each ... item conforms to the established classification criteria." How and at what time in the loading process will these limits be determined, weighed, or otherwise controlled?

Response:

Fuel/Debris/GTCC waste classification criteria, loading sequence and individual basket/cask inventory is required to be proceduralized by Trojan ISFSI Technical Specification 5.7.1.1 (h). This procedure has not been developed.

PGE has determined that the entire amount of fuel debris will be well below the limit of 20 Ci plutonium and 10 kg U, even if all debris were packaged in one can.

Question: 16

Section 3.2.3.2.2 gives a formula and no results. Please provide the source of the formula, input data, assumptions and results.

Response:

The formula is a standard representation of the fundamental frequency for a simple cantilever beam deflecting under its own weight:

$$f = \frac{1}{2\pi}\sqrt{\frac{g}{\delta_{st}}}$$

where δ_{st} is the maximum static deflection produced by the weight of the beam.

This formula is also further discussed in Section 8.2.5.2.

The specific form of this equation comes from Rourke and Young, Formulas for Stress and Strain, Fifth Edition, 1975, Table 36, Case 3b, page 576.

Thus:

$$f_n = \frac{K_n}{2\pi}\left(EI \cdot \frac{g}{wL^4}\right)^{.5}$$

where:

K_n is a constant equal to 3.52 for the first mode of vibration

E is the modulus of elasticity for normal weight concrete equal to (57,000)(compressive strength)$^{.5}$ = 3,604,996 psi [f_c = 4,000 psi]

I is the moment of inertia of the cask about its central axis and is equal to $\pi/64\ (D_o^4 - D_i^4)$ = 1.49×10^7 in^4 [D_o = 136", D_i = 78"]

L is the height of the cask and is equal to 211.5 inches

w is the uniform weight per unit length of the cantilever and is equal to the cask weight divided by the height, 289,000 lbs/211.5 in = 1366.4 lb/in

g is 386.4 in/sec/sec

This results in a natural frequency for the cask of 48.8 cycles per second. This is well beyond the zero period acceleration threshold of the site spectra. Therefore, the dynamic amplification factor can be taken as 1, and the seismic loads can be treated as static.

Question: 17

Section 3.2.3.2.5 describes the overturning moments and states that horizontal components are applied simultaneously with the vertical components. This would be a conservative application of loads; however, Section 8.2.5.2 applies the three axis loads at a ratio of 100%, 40%, and 40%. Explain why the treatment in the SAR is conservative.

Response:

The derivation of the Seismic Margin Earthquake (SME) was based on an analysis of empirical data and numerical modeling studies. The 0.38g peak acceleration value represents the maximum expected horizontal ground motion acceleration at the site, in any direction, based on those studies. The discussion in 8.2.5.2 derives a vectored peak acceleration using 100% (0.38g) and 40% (0.15g) simultaneously acting on the cask horizontally. Since the cask is round, the vectored acceleration represents the maximum. However, the derived horizontal acceleration of 0.41g is greater than 0.38g SME. Therefore, the approach is conservative. See the response to question 99 for further information.

Question: 18

Section 3.3.2.1 states that the "Failed Fuel Cans and GTCC Cans do not provide a confinement boundary." Section 4.2.2 states that the "Failed Fuel Cans provide a containment boundary for failed fuel assemblies." What is the difference between the use of confinement and containment?

Response:

Use of the term "confinement boundary" refers to features or characteristics included in the design to ensure the confinement of radioactivity. This is the basket shell including the bottom and both the shield and structural lids. Use of the term "containment boundary" is intended to refer to features or characteristics included in the design to restrain, control, or bound spent fuel or debris within the basket. To some degree, the basket internals (including the Failed Fuel, GTCC waste, and Fuel Debris Cans) provide containment boundaries.

Question: 19

Why are the Fuel Debris Cans sealed and the Failed Fuel Cans open to the basket atmosphere? Both fuel debris and failed fuel may contain exposed fuel pellets.

Response:

The Failed Fuel Can contains partial or complete fuel assemblies with failed or suspect rods (SAR Section 4.2.3.2.5). By definition, the possibility exists for pellets to be present which have been previously exposed to the contents of the spent fuel pool. The Trojan ISFSI design requires that moisture be removed from the basket (and basket contents) before helium is sealed inside. The Failed Fuel Can is open to the basket atmosphere to facilitate this process. Exposing fuel pellets to the basket atmosphere during this process or during long term storage has been shown to have no detrimental effect (Report PNL-6364).

PGE has determined that the most effective method for decontaminating the Trojan Spent Fuel Pool and removing fuel debris will necessitate redesign of the Fuel Debris Can. The Fuel Debris Can design currently described in the Trojan ISFSI SAR will be revised as soon as vendor contract negotiations are concluded and a final design is approved.

Question: 20

Section 3.3.1 lists the equipment/components that have been identified as important to safety for the ISFSI. Not listed is the air pad system, although it appears that this system meets the second of the three criteria listed in Section 3.4: "To prevent damage to the spent fuel or the high-level radioactive waste container during handling and storage..." Please respond.

The Fuel Building crane also appears to meet the referenced criteria. Is it considered important to safety or quality related? Will modifications to the crane be reviewed in accordance with the QA program?

Response:

The air pad system in itself does not prevent damage to the spent fuel or the high-level radioactive waste container but is equipment utilized to transport the loaded storage casks from the Trojan Fuel Building to the ISFSI pad or from the pad to the Transfer Station. The Basket is the component which prevents damage to the spent fuel and the Concrete Cask is the component which prevents damage to the high-level radioactive waste container (the Basket). Both of these components are classified as important to safety. Air pad system failure during movement of a loaded cask was addressed in PGE Part 50 License Change Application (LCA) 237-Spent Fuel Cask Loading in the Fuel Building submitted to the NRC on May 29, 1996. The conclusions found that air pad system failure would not result in a radiological release or intact spent fuel damage that results in a k_{eff} of more than .95.

The classification criteria established in Trojan ISFSI SAR Section 3.4 is applicable to ISFSI structures, systems, and components. Activities associated with the Fuel Building crane will be subject to controls developed per the requirements of LCA 237. The Fuel Building crane is classified as non-quality related, seismic category II/I. Per the Trojan design change procedure, modifications to the Fuel Building crane would be reviewed by engineering but not the QA organization.

Question: 21

The ISFSI Technical Specification surveillance requirement is intended to verify that the Concrete Cask air outlet temperature is less than 220°F. There are four temperature monitoring devices per cask. What is the allowable difference in readings between the four devices?

Response:

Any difference in temperature readings between the four devices will be addressed in the surveillance procedure. When a technical specification operating limit is established, a surveillance limit for the recorded readings will be conservatively set such that there is enough margin between the two limits to include the instrument uncertainty. Section B 3.1.1 of the proposed Trojan Independent Spent Fuel Storage Technical Specifications states that for surveillance purposes, the recorded air outlet temperature must consider measurement instrument tolerances.

It should be noted that the four temperature devices are not the sole means of detecting the air outlet temperatures. Any hand held device can be used as a backup if the installed device readings are suspect or when a more accurate temperature measurement is desired (for instance, when the temperature readings of the installed devices approach the surveillance limit).

Question: 22

Section 3.3.4.3 states that the "criticality analysis was performed using NRC accepted codes. These codes were validated in accordance with SNC Quality Assurance Program." What codes were used? Were the input and assumptions validated independently by PGE personnel?

Response:

The code used for the criticality analysis of the fuel during storage is "SCALE-PC Modular Code for Performing Criticality Safety Analysis for Licensing Evaluation, Version 4.1", Oak Ridge National Laboratory, CCC-619. Assumptions, design inputs, results and conclusions were independently reviewed by PGE personnel to ensure that they were reasonable, and that acceptance criteria were met.

Question: 23

Section 3.3.5.3 and Tables 7.4-1 and 7.4-2 provide expected dose rates. What is the source for these numbers?

Response:

The source for these numbers are shielding and dose calculations, which are mainly discussed in ISFSI SAR Section 7.3.2.2. See the response to question 78 for more information.

Question: 24

Section 3.3.6 refers to Section 8.2.15. Our copy of the ISFSI SAR does not contain this section.

Response:

Section 3.3.6 incorrectly referenced Section 8.2.15. The correct reference is 8.2.14. The next revision to the Trojan ISFSI SAR will contain this correction.

Question: 25

Section 3.3.7.1 states that the "ISFSI shall have a minimum design of 40 years." How is this supported by the vendor's SAR and supporting documents? Which components are limiting in terms of lifetime and how is it shown that they will last 40 years?

Response:

PGE will provide a response to this question by August 29, 1996.

Question: 26

Section 3.3.7.1 states that the fuel clad temperature limit for TNP fuel is 730°F. How was this determined? Also, how was the off-normal limit of 1058°F (given in Table 4.2-12) determined?

Response:

The 730°F limit is calculated using Figure 4.2 (Attachment IV) in Pacific Northwest Laboratory, PNL-6364, "Control of Degradation of Spent LWR Fuel During Dry Storage in an Inert Atmosphere". Based on a 42,000 MWd burnup of Westinghouse fuel (more conservative than B&W fuel), the cladding stress at 300 and 380°C are about 78 and 69 MPa, respectively. A line drawn between these points on Figure 4.2 results in a long term storage limit of 388°C, which is 730°F.

The 1058°F limit comes from PNL-4835, "Technical Basis for Storage of Zircaloy-Clad Spent Fuel in Inert Gases" (Attachment V). Dry storage tests and demonstrations were performed in inert gases at temperatures from 50 to 570°C (112 to 1058°F). The tests and demonstrations involved about 15,000 fuel rods and there was no evidence that any rods exposed to inert gases failed. Therefore, the upper temperature limit of these tests and demonstrations is used as a short term temperature limit.

Question: 27

Table 3.1-2 describes TNP fuel. Please verify that there is no 5 year old fuel with burnup greater than 40,000 MWd/MTU. Note that Table 3.1-3 describes 5 year cooled assemblies with 42,000 MWd/MTU.

Response:

The burnups and cooling times listed in Tables 3.1-2 and 3.1-3 were used for the design calculations. These values bound the Trojan fuel because any fuel that has more than 40,000 MWd/MTU burnup will be cooled longer than 6 years.

Question: 28

Please provide the technical bases for the three footnotes on Table 3.1-2. Explain how burnup, enrichment, and cooling time affect the spent fuel as a gamma and neutron source. How do the three variables taken together affect gammas and neutrons?

Response:

The spent fuel gamma and neutron source strengths were determined using the OCRWM computer database, "Characteristics of Potential Repository Wastes", DOE/RW-0184, Office of Civilian Radioactive Waste Management, U.S. Department of Energy, July 1992. This database gives the gamma and neutron source strengths as a function of fuel burnup level, cooling time, and initial enrichment. The Trojan spent fuel gamma and neutron source strengths were determined for both 40,000 MWd/MTU 5 year cooled fuel with an initial enrichment of 3.02% and 45,000 MWd/MTU 6 year cooled fuel with an initial enrichment of 3.30%. These two cases bound the entire Trojan spent fuel inventory. The 5 year cooled fuel results in a higher gamma source strength and the 6 year cooled fuel results in a higher neutron source strength.

Neutron and gamma source strengths decrease as cooling time increases due to radioactive decay. If two fuel assemblies have the same initial enrichment and burnup level, the assembly with the longer cooling time will have a lower neutron and gamma source strength.

Neutron and gamma source strengths increase as burnup level increases. If two fuel assemblies have the same initial enrichment and cooling time, the assembly with the higher burnup level will have a higher neutron and gamma source strength.

Neutron and gamma source strengths increase as fuel enrichment decreases. Attachment VI gives the neutron and gamma source strengths for both 3.02% and 4.42% enriched fuel, when cooling time and burnup level are constant. (This information was obtained from the OCRWM computer database.)

Gamma energy lines above 3.5 MeV and below 0.575 MeV do not contribute significantly to cask external dose rates. Therefore, these photons are excluded when shielding calculations are performed. Attachment VI indicates that both the neutron and significant gamma energy line source strengths increase as fuel enrichment decreases.

Question: 29

What are the critical dimensions of the ISFSI equipment deemed important to safety? Include the heights, diameter, thicknesses, and other parameters sufficient to evaluate the normal mechanical and thermal stresses and various accident scenarios.

Response:

SAR Section 3.3.3.1 classifies the following ISFSI equipment as important to safety:

1. Concrete Cask
2. PWR Basket and GTCC Basket
3. Basket Overpack
4. Fuel Debris Can and Failed Fuel Can
5. Transfer Cask
6. Transfer Station

The Concrete Cask, PWR Basket, and GTCC Basket dimensions are provided in the responses to questions 30, 31, 33, 59, 75, and 104.

The Basket Overpack is 183.5 inches tall (outside dimension), 181.5 inches tall (inside dimension), 68 inches outside diameter, and fabricated from stainless steel. The shell wall is .500 inches thick. The top lid and bottom plate are both 1 inch thick.

The fuel debris can will be redesigned.

The Failed Fuel Can is 168.3 inches tall (outside dimension) not including the lid and 9.05 inches square outside dimension. The shell wall is .125 inches thick and fabricated from carbon steel. Each side of the shell wall has two screen covered holes. Two lid mounting bars are located on the outside of each shell wall which are drilled and tapped to secure the lid assembly. The bottom plate is >.250 inches thick, 9.05 inches square and fabricated from carbon steel. The bottom plate has two screen covered holes near the opposite edges. Four stand off plates are welded under each corner of the bottom plate. The lid assembly is fabricated from carbon steel. The lid assembly also contains two screen covered holes near the opposite edges. Eight bolts are used to secure the complete lid assembly to the Failed Fuel Can top.

The Transfer Cask is 192.1 inches overall in height, 86 inches in outside diameter, and 67 inches inside diameter. The walls are 9.5 inches thick. The inside shell wall and outside shell wall are fabricated from carbon steel. The area between the shells is filled with lead and a neutron absorber shield. The trunnions are approximately 14.5 inches long, 12 inches in diameter and fabricated from high strength carbon steel. A carbon steel cover plate is bolted over the end of the trunnion. Each rail mounted shield door is 9 inches thick and fabricated from carbon steel material.

The Transfer Station is 168.5 inches wide (cl), 308 inches tall (cl), and 420 inches long (cl). The material used for fabrication is W shape (I beam) and angle. The material is A 36 carbon steel.

X bracing connected to the main beams by bolted connections provides uniform support.

Question: 30

Please describe the concrete cask lid. Include information on material, dimensions, fasteners, amount of radiation shielding provided, and weather seal. Is the cask lid designed to prevent rain from entering the cask? If gaskets are provided, are they designed to last 40 years?

Response:

The concrete cask lid is .750 thick, 86 inches in diameter, and fabricated from a carbon steel plate. The top is coated with Carbozinc 11 paint. Six holes are drilled through the lid to be utilized when bolting the lid to the cask liner flange. A 2 inch wide gasket is installed between the lid and the cask liner flange. Because the cask annulus is open to the atmosphere via the inlet and outlet vents, the cask lid is not intended to preclude moisture intrusion. As discussed in the Trojan ISFSI SAR (Section 4.2.3.2.4) the cask lid provides a cover and seal to protect the basket from the environment and postulated tornado missiles.

Question: 31

Please describe the basket shield lid. Include information on material, dimensions, fasteners, and amount of radiation shielding provided.

Response:

The shield lid is fabricated from two separate pieces of stainless steel plates welded together to form a laminated component 8 inches thick. The outside diameter is approximately 64 inches. A hole extends through the shield lid to allow for placement of the swagelok quick connect vent plug. Another hole extends through the shield lid (between the vent plug and lid edge) to allow for placement of a vent pipe. This additional hole is located near the shield lid edge. Both holes are covered by welded vent plugs installed in the structural lid. As the shield lid is attached to the basket by welding, fasteners are not utilized. The shield lid provides 8 inches of radiation shielding.

Question: 32

By how much is the air flow reduced with the Basket Overpack? Is a reduction in cooling capacity a concern? Please provide the analysis and assumptions.

Response:

The natural circulation air flow through the concrete cask varies with ambient temperature and basket heat loading. The reduction in air velocity at the discharge of the outlet screens when an overpack basket is in the concrete cask is less than 2% for 75°F ambient temperature and a 26 kW heat load. This reduction in cooling capacity is not a concern because the concrete and fuel cladding temperature are below the short and long term limits for the short and long term storage conditions as shown in ISFSI SAR Table 4.2-12.

The methodology and assumptions for the air flow analysis are provided in detail in ISFSI SAR section 4.2.5.3 except that the value used for the specific heat of air is assumed to be constant at 0.241 and the average air density (ρ bar) is calculated as (2.7)(14.7 psia)(1.0)/[(air inlet temperature + air outlet temperature)/2 + 460].

Question: 33

Please provide a description of the concrete cask shield ring.

Response:

The shield ring is fabricated from A-36 carbon steel and may be made from as many as 4 or as few as 2 individual pieces of material. The outside diameter is 72 inches. Inside diameter is 64 inches. Height is 6 inches. Thickness at the top is 4 inches. Thickness at the bottom is 3 inches. Approximately 2 inches from the top and on the outside surface, a 45 degree bevel extends down and inward until the thickness is reduced from 4 inches to 3 inches. The outside beveled piece is fabricated individually and welded to the 3 inch by 6 inch piece.

Question: 34

Section 4.2.1 states that the ISFSI Storage Pad provides a smooth level surface to allow operation of the air pad system. Does the air pad system operate off of the storage pad; and if so, is a smooth surface provided for its operation between the Fuel Building and the storage pad?

Response:

The air pad system is intended to be operated on the storage pad and between the Fuel Building and the ISFSI storage pad. It is anticipated that temporary surfacing (smooth metal sheets or concrete slabs) will be placed on the ground between the Fuel Building and the storage pad to facilitate movement of the air pad, as necessary.

Question: 35

Section 4.2.1 states that the GTCC Cans are not important to safety. Why are the Failed Fuel and Fuel Debris Cans important to safety but the GTCC Cans are not?

Response:

The failed fuel and fuel debris cans maintain the geometry of the spent fuel in the basket and the geometry ensures subcriticality. Maintaining the geometry of the GTCC waste is not an important to safety parameter because criticality is not a concern with GTCC waste.

Question: 36

Section 4.2.2.1 references a Figure 4.1-1. There is no such figure in our copy.

Response:

Section 4.2.2.1 incorrectly referenced Figure 4.1-1. The correct reference is Figure 2.1-3. The next revision to the Trojan ISFSI SAR will contain this correction.

Question: 37

Section 4.2.3 states that "The Concrete Casks are arranged on the Storage Pad as discussed in Section 4.1.1." Section 4.1.1 refers to Figure 2.1-3, which does not provide any dimensions. The lack of specificity is a concern. Please provide a layout of the ISFSI that contains the critical dimensions.

Response:

The separation between casks on ISFSI storage pad is not a critical dimension. The calculation that estimated the distance at which the direct radiation dose would be 25 mrem/year (10 CFR 72.104 limit) used a cask center-to-center spacing of 15 feet to account for self-shielding. This distance will be established by actual radiation measurements. Therefore, the separation between casks is not critical.

Question: 38

Section 4.2.3.2.1 describes "neutron absorbing poison sheets are also used in the construction of the PWR Basket internal assembly." Please provide a description of these sheets, including material, dimensions, where attached in the basket assembly, neutron absorption capability, and other pertinent data. NRC Information Notices have described problems with Boral degradation in certain spent fuel pool rack designs. Is this a concern for the PWR Baskets?

Response:

Poison sheets made of boral, which is nominally 69% aluminum, 24% boron, and 6% carbon, are held in the basket internal assembly by metal clips that are welded to the storage cell walls. The poison sheets are clipped to the 4 inside walls of the 12 inner-most normal-sized storage cells. The poison sheets are clipped to the 3 inside walls that are closest to the center of basket assembly of the 8 outermost normal-sized cells. The poison sheets are clipped to the 2 inside walls that are closest to the center of basket assembly of the 4 oversized storage cells.

The poison sheets are 8.25 or 8.6 inches wide and 154 inches high with the wider sheets being for the oversized storage cells. The neutron absorption capability is governed by minimum areal density of Boron-10, which is 0.035 g/cm^2, for the poison sheets.

NRC Information Notice 83-29, "Fuel Binding Caused by Fuel Rack Deformation", and a guidance letter dated April 14, 1978, "Review and Acceptance of Spent Fuel Storage and Handling Applications", both mention boral, but do not discuss degradation. NRC Information Notices 87-43, "Gaps in Neutron-Absorbing Material in High Density Spent Fuel Storage Racks", 93-70, "Degradation of Boraflex Neutron Absorber Coupons", and 95-38, "Degradation of Boraflex Neutron Absorber in Spent Fuel Storage Racks", describe problems with boraflex. Boraflex is about 50% boron carbide, 25% silica, and 25% polydimethyl siloxane polymer, which is a considerably different material composition than boral.

Question: 39

Section 4.2.3.2.3 describes NRC approval of design, fabrication, and testing of the Basket Overpack. What additional information will be provided to the NRC, and when will this be provided?

Response:

At this time, PGE does not anticipate submitting additional information to the NRC regarding the Basket Overpack. The statement referenced in Section 4.2.3.2.3 refers to information contained within the Trojan ISFSI SAR which is currently being reviewed by the NRC and must be approved before PGE commences ISFSI operations.

Question: 40

Section 4.2.3.12.4 discusses the deviation from ACI-349 generic limits for the temperature of concrete; however, it is not clear from the text whether the concrete mix is different from that prescribed in ACI-349. The statement at the bottom of p. 4-9 is that "The concrete mix used to fabricate the Concrete Casks is intended to allow satisfactory long term concrete temperatures as high as 250°F." ACI-349 limits the temperature of bulk concrete in "Extreme Environmental" conditions to just 200°F. Please respond.

Also, has the deviation from ACI-349 generic limits been accepted at other utilities and by the NRC? Is there actual performance data to empirically support a deviation from the standard?

In addition to the temperature limits on concrete, are there limits on the size or range of thermal gradient of the concrete? Table 4.2-12 indicates thermal gradients in excess of 100°F.

A related concern is the affect of high temperature (in excess of 200°F) on the concrete for up to 40 years. Table 4.2-12 lists the steady-state temperature for the concrete as 204°F, and "steady state severe hot" conditions as 239°F, both of which are in excess of ACI-349 limits. Have studies been done to show that ACI-349 limits can be exceeded with no detrimental effect for up to 40 years?

Do these steady state concrete temperatures assume that the basket is centered and the air channel is evenly spaced around the basket? During the loading of the basket or movement of the cask to the ISFSI pad, what precludes the basket from shifting to one side and contacting or being in close proximity to the concrete? There is no mention in the SAR of shims for this purpose. Such movement could increase the temperature of a portion the "inner concrete" wall closer to the 393°F of the basket shell. How would this affect the integrity of the concrete in the long-term? Table 4.2-12 and Section 4.2.3.2.4 list the long term limit as 250°F.

Response:

PGE will provide a response to these questions by August 29, 1996.

Question: 41

Section 4.2.3.2.5 describes radiation resistant, high temperature, hard surface zinc coatings. How will PGE ensure that the coatings will provide protection for 40 years?

Response:

This type of zinc coating is used in the nuclear industry on the interior of containment buildings and has many years of proven service. Following loading of the spent fuel, the basket will be moved to the cask wash pit, the water drained, shield and structural lids installed, and the interior of the basket inerted with Helium. The zinc coating's proven history and the inert environment will ensure protection of the basket internals for 40 years.

Use of the zinc coating will be evaluated with respect to NRC Information Notice 96-34, "Hydrogen Gas Ignition During Closure Welding of a VSC-24 Multi-Assembly Sealed Basket," and any further information as it becomes available.

Question: 42

Section 4.2.3.2.6 states that "Internal filters are designed to retain the fuel debris." Please provide a description of these internal filters.

Response:

The fuel debris can will be redesigned.

Question: 43

Sections 4.2.3.2.5 and 4.2.3.2.6 describe the Failed Fuel Cans and Fuel Debris Cans. What criteria is used to determine what goes into each?

Response:

Fuel/Debris/GTCC waste classification criteria, loading sequence and individual basket/cask inventory is required to be proceduralized by Trojan ISFSI Technical Specification 5.7.1.1 (h). This procedure has not yet been developed.

Question: 44

Section 4.2.5.1 states that the "PWR Basket drying operations could possibly expose the spent fuel cladding to stresses in excess of those established for the 40 year dry storage period. The vacuum drying time will be administratively controlled to minimize the strain that fuel cladding will be subjected to during this operation." Section 5.1.1.2 states that a vacuum of 3 mm Hg will be held for a minimum of 30 minutes on two occasions. Chapter 8 does not describe an off-normal event involved with the vacuum drying operation. What accident analysis has been performed regarding a fuel cladding rupture during the vacuum drying operation? Since this activity is part of the loading operation in the Fuel Building, will this information be furnished with the additional submittal referenced on p. 8-1?

Response:

Postulated off-normal events and accidents related to the vacuum drying process are provided in PGE's letter to the NRC, "License Change Application (LCA) 237 - Spent Fuel Cask Loading in the Fuel Building", dated May 29, 1996 (VPN-031-96). Analysis of a fuel cladding rupture during the vacuum drying process is not included because a credible initiator of such an accident is not known. In addition, a time limit will be established for completing the vacuum drying process to ensure that the fuel cladding strain accumulated during the vacuum drying process does not cause long term degradation of the fuel cladding during the dry storage period.

Question: 45

Section 4.2.3.2.7 states "Up to 29,000 lbs contained within 28 GTCC Cans can be placed in a GTCC Basket." How and in what part of the loading process will these components be weighed?

Response:

Segmentation and GTCC waste can loading would take place in the reactor cavity. Each individual can would be physically weighed before being moved to the Spent Fuel Pool for loading into a GTCC Basket.

Question: 46

Sections 4.2.4.3.2 through 4.2.4.3.5 are presented without references or background information. Please provide more detailed information to support the conclusions given and to interpret the data presented in Tables 4.2-8 through 4.2-11.

Please explain the negative gage pressures on Table 4.2-9. In several places in the SAR it is stated that the normal operating pressure for the basket is 14.7 psia (0 psig). Also, please explain the relationship between pressure and temperature for the storage system.

Response:

Sections 4.2.4.3.2 through 4.2.4.3.5 discuss dead weight loads and load combinations for the Basket. The dead weight loads, thermal and other stresses, were calculated by Sierra Nuclear Corporation. Stresses were conservatively added using absolute values without sign consideration. ASME Section III, NC allowable stresses are calculated per Section NC-3217. ASME Section III, NG allowable stresses are calculated per Section NG-3222 (Level A), NG-3224 (Level B), and NG-3225 (Level D). Allowable stresses are calculated using values from the ASME Code, Section III, Part D, for applicable material. Per ASME III, Section NC-3264.6, a coefficient of 0.75 is applied to the allowable stresses for partial penetration welds and full allowables are used for full penetration welds.

The basket pressure is related to the vacuum drying and backfilling process. As described in Section 5.1.1.2, the sealed basket is backfilled with 99% pure Helium at atmospheric pressure (14.7 psia). The pressure calculations in Table 9.2-1 use a backfill pressure of 14.5 psia. The basket average gas temperature at backfilling is calculated to be 600°F. The Basket gas temperature under study case conditions is taken as the average between the fuel and basket shell temperatures for each case. The ideal gas law is used in the basket pressure analysis.

Question: 47

Section 4.2.4.4.3 states that the thermal stress analysis employs classical hand calculations. Please elaborate.

Response:

Section 4.2.4.4.3 discusses the concrete cask thermal stresses. The calculation uses the ANSYS computer model to calculate the concrete cask temperatures for each of the study case conditions. The thermal gradient across the concrete cask wall is used as input for the thermal stress analysis. The concrete and rebar stress calculation is performed using a balanced tension-compression across the concrete. First, the thermal stresses are determined based on the uncracked section. The section of the concrete is assumed to crack, shifting the neutral axis, which results in the reduction of concrete compressive stress. Assuming that stress distribution across the section remains linear, forces are presented as functions of this reduction in compressive stress. Since the total force on a section must remain 0, a force balance equation is written which allows stresses in all components to be calculated. Formulas in Roark, "Formulas for Stress and Strain", Fourth Edition, are used for the liner shell, liner bottom plate and cask cover plate.

Question: 48

Section 4.2.4.4.2 is presented without dimensions or background information. Please provide the area of contact for the transfer cask.

Response:

For snow/ice loads, the diameter for the top of the concrete cask is 136 inches; surface area is 14,634 in^2 (101.6 ft^2). The contact area for the transfer cask is estimated based on transfer cask rails in contact with the top of concrete cask to be 1956 in^2 (13.6 ft^2).

Question: 49

Section 4.2.5 presents the "two off-normal severe environmental temperature conditions" as -40°F and 100°F. Minus 40 appears conservative, but 100 appears less so. How is "steady-state" defined? What would the maximum clad temperature and inner concrete temperature reach if the steady state severe hot temperature was 105 or 107? Are the data in Table 4.2-12 backed up by actual tests?

Response:

Steady state is defined as the constant air temperature over several days. The maximum clad temperature and inner concrete temperature for environmental temperature conditions above 100°F are bounded by the accident condition of 125°F ambient, with solar insolation. The ANSYS Model used to perform the analysis includes several conservatisms. Even under this severe environmental condition, there is still adequate margin below the fuel clad and concrete temperature limits. A startup test to confirm proper operation of the storage system will be performed once the concrete casks are in place on the storage pad.

Question: 50

Section 4.2.5.3 describes an iterative solution to the air flow and temperature equations, utilizing a spreadsheet program. Please describe how this calculation was checked, and how the program was validated.

Response:

The air flow calculation was prepared and independently checked in accordance with written procedures. The calculation was prepared by the vendor under a PGE approved Quality Assurance Program. The calculation was prepared and independently checked per vendor procedures. The calculation was also reviewed by PGE per procedure TPP 18-9, "Trojan Calculations". Assumptions, design inputs, results and conclusions were independently reviewed by PGE personnel to ensure that they were reasonable.

Question: 51

Section 4.2.5.4.3 states that the natural convection heat transfer coefficients were taken as 2.0 BTU/hr-ft^2-°F. What is the source of this constant?

Response:

The value of 2.0 BTU/hr-ft^2-°F is the lowest (most conservative) convective heat transfer coefficient calculated from storage cask tests of the Transnuclear TN-24P PWR cask, the Castor V/21 PWR cask, and a BWR cask.

Question: 52

Section 4.2.5.5.1 states that "fuel heat generation rates were calculated by assuming 1.08 kWt of heat generation per assembly (26 kWt/24 assemblies)." What is the basis of this assumption?

Response:

At the time of contract issuance for design services, PGE was aware of fuel assemblies which could potentially generate 1.08 kWt per assembly. Designing the Trojan ISFSI for heat loads of 26 kWt is a conservative action which bounds Trojan fuel.

Question: 53

Section 4.2.6 states that the analysis "conservatively assumes an initial fuel enrichment of 4.2 wt% U235 with no credit taken for burnup. Fuel enrichment assumptions conservatively bound the fuel to be stored which has a maximum initial enrichment of 3.56 wt% U235 and burnup." However, a footnote on Table 3.1-2 states that "Low initial enrichments will yield higher gamma and neutron source terms for a given burnup." Therefore, is the assumption of a higher initial enrichment conservative in terms of criticality analysis but not in terms of dose rate?

Response:

Yes, the assumption of higher initial enrichment is conservative for the criticality analysis, but not for the shielding analysis. See also the response to question 28.

Question: 54

Section 4.7.4.1.1 gives the stresses on the trunnions. Please provide background information on the calculations used, including source of equations, assumptions, and constants.

Response:

Stresses for the trunnions are determined using classical structural analysis formulas. Only one of the two trunnions is considered because of symmetry. The formulas are taken from Roark's Formulas for Stress and Strain, Fourth Edition.

For the trunnion shear stress, the following assumptions and constants were used:

weight of the basket and transfer cask = 215,000 lbs (divided by 2, since there are two trunnions)

dynamic load factor = 1.1

A_T, cross sectional area of the trunnion for shear stress = $\pi r^2 = 113.1 \text{ in}^2$
($r = 6$ in)

4/3 is the shape factor used for a solid circular section since the trunnion has a circular cross section.

For the maximum bending stress in the trunnion:

L, distance between the transfer cask outer wall and mid-point of the load application = 2.75 inches

S, trunnion section modulus = $\pi d^3/32 = 169.6$ cubic inches ($d = 12$ in)

Question: 55

Table 4.2-13 gives the emissivity of various materials. Emissivity is normally defined as the ratio of the total radiating power of a real surface to that of a black surface at the same temperature. In standard handbook tables, steel has emissivity rates of from 0.2 to 0.7 in the temperature ranges given. If calculations are based on the numbers in Table 4.2-13, please provide references.

Response:

The emissivity for steel shown in this table was used for the Concrete Cask Thermal Analysis calculation.

The Marks Standard Handbook for Mechanical Engineers and Kreith/Principles of Heat Transfer show emissivity for steel in .8-.9 range for the temperatures of interest (32 - 800°F).

Question: 56

Figures 4.2-5 and 4.2-7 describe several differences between Failed Fuel Cans and the GTCC Cans. What are the purposes of the key differences?

Response:

The Failed Fuel Can lid is configured differently from that of the GTCC Can lid in order to effectively utilize the fuel handling tool available at the Trojan Spent Fuel pool. The fuel handling tool is not available for use during GTCC Can loading in the reactor cavity (see the response to question 45), therefore the GTCC Can has a tool engagement slot to be utilized during loading operations in the reactor cavity.

Question: 57

Please provide the applicable portions of ASME referenced documents to interpret information given in the tables and calculations, including Service Levels, pressure limits, and definitions of cylinder stresses (P, P_l, P_m, P_b, Q)

Response:

PGE provided copies of ASME Section III, Division 1, Subsections NG & NC in transmittal number CPY-030-96 dated June 11, 1996. Additional documents being furnished relating to this response are ASME Section III, Division 1, Subsection NF (1992), ASME Section II, Part D (1992) and ASME Code Case N-71 (enclosed).

Question: 58

The ISFSI SAR quotes codes and standards used in the vendor's SAR, SNC-95-71, Rev. 0. Since the NRC has not written an SER on SNC-95-71, has PGE verified the appropriate application of these codes and standards?

Response:

PGE has verified the appropriate application of the codes and standards of the vendor's SAR (SNC-95-71, Rev. 0) only as it directly applies to the Trojan ISFSI SAR. Please see the response to question 1 for additional information.

Question: 59

Please provide the dimensions of the PWR Basket shield and structural lids.

Response:

The dimensions of the PWR basket shield lid are provided in the response to question 31.

The PWR basket structural lid is 3 inches thick, 64 inches in diameter and fabricated from stainless steel. A hole through the lid is located near the edge. This hole is filled with two stainless steel plugs which cover the fittings in the shield lid. The plugs are welded in place after pressure testing and helium backfilling are complete.

Question: 60

Section 5.1.1.1 states that after a Fuel Debris Can is loaded, it is "moved to a fuel cell location." Describe these fuel cell locations. Are these standard fuel rack locations for a fuel assembly? Are the Fuel Debris Cans and Failed fuel Cans smaller than the spent fuel racks?

Response:

The text of Trojan ISFSI SAR Section 5.1.1.1 regarding fuel cell location is intended to mean standard fuel rack locations. The Failed Fuel Can outside dimension is 9.05 inches square, which is larger than the standard fuel rack location. The forthcoming revision to the Trojan ISFSI SAR will contain a revision to Section 1.3.1 to clarify the loading sequence of the Failed Fuel Can. As stated in Trojan ISFSI SAR Section 5.1.1.2, operations will be conducted in accordance with approved Trojan Nuclear Plant fuel handling procedures.

The fuel debris can will be redesigned.

Question: 61

Since the Fuel Debris Cans are considered a "confinement boundary," are they leak tested?

Response:

The fuel debris can will be redesigned.

Question: 62

Section 5.1.1.2 states that after the basket is loaded into the Transfer Cask, radiation shielding shims are placed in the gap between the Transfer Cask and basket. Please provide a description of these shims. When are they removed?

Response:

The radiation shielding shims are fabricated from A-240 Type 304 stainless steel. They are utilized as needed to close the top gap between the transfer cask and basket. Because the annulus between the basket and transfer cask varies around the circumference (fabrication tolerances), shims are provided in assorted thicknesses ranging from .060 to .750 inches. Height is approximately 4.7 inches and length varies per application. After the proper shim configuration is established, the shims are welded to a round flange approximately .06 inches thick to prevent the shims from dropping into the annulus space. As described in Section 5.1.1.2 (5th paragraph) the shims are removed after the structural lid is welded in place.

Question: 63

Section 5.1.1.2 states that the "basket is then filled with borated water." What is the minimum boron concentration and what is the justification for this concentration?

Response:

The basket will be filled with water from the Spent Fuel Pool, which has a boron concentration greater than or equal to 2,000 ppm. During basket loading in the Cask Loading Pit, the gate to the Spent Fuel Pool will be open. Filling the basket with water from the Spent Fuel Pool eliminates the possibility that the water in the basket could dilute the boron concentration in Spent Fuel Pool.

Boron is not required in the basket for subcriticality because the criticality analysis that envelopes basket loading assumes pure water in the basket, i.e., it conservatively does not credit the boron in the basket water.

Question: 64

Section 5.1.1.2 states that the "water level is lowered in the basket by approximately 16 inches to ensure that the shield lid weld is not affected by percolation." How is this 16 inches measured? By volume?

Response:

A volume calculation will be used to establish the water level in the basket below the shield lid. Trojan ISFSI SAR Section 5.1.1.2 will be revised to address the volume reduction (in gallons) needed to establish an air space between the underside of the shield lid and the surface of the water. The next revision to the Trojan ISFSI SAR will contain this correction.

Question: 65

Section 5.1.1.2 states that the basket is "hydrostatically tested to 7.3 psig which is 1.25 times the normal operating pressure. Depending on the interpretation, this could mean that the normal operating pressure is 17.6 psia [(7.3+14.7)/1.25]. the text states later in the same section that after backfilling and testing with He, the "pressure is then released back to atmospheric pressure (approximately 14.7 psia)." Section 4.2.4.3.6 also states that the normal operating pressure for the basket is 14.7 psia. The first passage referenced appears to conflict with the other two. Is there a conflict in these Sections of the SAR? What is the planned normal operating pressure for the PWR and GTCC Basket? (A similar conflict appears in SNC's VSC-24 SAR. It states that the PWR Basket is pressurized to 17.4 psia (p. 8-5). However, the specification given for the He backfill pressure in both the C of C and SER associated with the VSC-24 is 14.5 $\pm$/- 0.5 psia.)

Response:

The Trojan ISFSI text was not clear in stating that the basket is "hydrostatically tested to 7.3 psig which is 1.25 times the normal operating pressure." The text should have stated that the basket is "hydrostatically tested to 7.3 psig which exceeds 1.25 times the normal operating pressure." Normal operating pressure is approximately atmospheric or 14.7 psia. This correction is being included in the forthcoming revision to the Trojan ISFSI SAR.

Question: 66

Section 5.1.1.2. What are the criteria and limits for measuring the leak tightness of the sealed basket with a helium sniffer? (The limit is given as 1 x 10-4 scc/sec for the VSC-24.)

Response:

The accepted weld will be leak tight to the point that no helium is detectable with a helium sniffer with a known accuracy of at least 1 x 10-4 scc/sec. If any helium is detected, the weld area identified will be ground out, repaired, and retested

Question: 67

Section 5.1.1.2 states that the "lid sealing, hydrostatic testing and draining must be completed within 36 hours." Was the 36 hour time limit calculated using site specific data and assumptions? What temperature of the cladding is reached at 36 hours? Table 4.2-12 lists 851°F as the maximum cladding temperature for a basket in the transfer cask. How long does it take in the transfer cask to reach this temperature?

Response:

The time limit was recalculated and is now 37 hours rather than 36 hours. This value will be changed in a forthcoming revision to the ISFSI SAR.

The 37 hour time limit has been conservatively calculated assuming site specific materials, a 26 kW heat load, a starting temperature of 100°F, and no heat transfer from the basket to the transfer cask or environment. The calculation assumed a simple adiabatic heat up for determining the amount of time for the water in the basket to boil and was not intended to determine the fuel cladding temperature. However, the fuel cladding will be well below any temperature limits during the 37 hours because the fuel cladding will be in direct contact with water that is at temperatures of 212°F or less.

A conservative calculation (i.e., no heat transfer from the basket to the transfer cask or environment) estimates that it would take 23.57 hours for the cladding to reach 851°F in a vacuum. The amount of time that the basket may be in a vacuum will be administratively controlled to minimize fuel cladding strain as described in "License Change Application (LCA) 237 - Spent Fuel Cask Loading in the Fuel Building", dated May 29, 1996 (VPN-031-96).

Question: 68

Section 5.1.1.5.a describes finding a le[illegible] by use of a helium sniffer. If the helium was stored at atmospheric pressure (or if, as suggested by Table 4.2-9, there was a slight negative pressure in the basket), how would a helium sniffer detect a leak?

Response:

Diffusion and convection provide mechanisms which would allow helium migration to the atmosphere.

Question: 69

Section 5.1.1.5.b describes the installation of the Basket Overpack. It states that two concrete casks are moved into the Transfer Station. After the sentence, "Relocate the loaded Transfer Cask over the adjacent Concrete Cask," there is no further mention of this adjacent cask. A Basket Overpack is inserted in the original cask and the leaking PWR or GTCC Basket is inserted into the Basket Overpack. What is the purpose of the second concrete cask? Is the leaking basket put into this cask temporarily?

Response:

The purpose of the adjacent cask in the Transfer Station is to support the loaded Transfer Cask while the overpack is installed in the empty cask. The leaking basket is not temporarily placed in the adjacent concrete cask.

Question: 70

How are the concrete casks brought into the Fuel Building? How is the air pad system installed and removed from under the concrete casks?

Response:

The empty concrete cask is brought into the Fuel Building utilizing the air pad system. Four individual air pads are installed under each concrete cask. A pad is slipped into each 48.5 inch wide air inlet vent opening on both sides of the concrete cask and then pressurized. After positioning the concrete cask in the desired position, the air pads are depressurized and removed. Insertion and removal of the pads is done by hand and requires no specialized equipment.

Question: 71

How are the baskets centered in the casks to maintain a constant and even air flow space?

Response:

The transfer cask provides for centering of the basket in the concrete cask. The basket outside diameter is 66 inches and the transfer cask inside diameter is 67 inches. The minimal difference in diameters serves to guide the basket while it is being lowered into the concrete cask, which has a larger inside diameter of 74 inches.

Centering in this manner is acceptable because the natural circulation air flow is relatively insensitive to the basket being exactly centered. This insensitivity is evidenced by the air flow calculation that shows that air flow in the concrete cask annulus is decreased by less than 2% when the basket overpack, which is 2 inches larger in diameter than the basket, is placed in the concrete cask. (See response to question 32)

Question: 72

How will the loading patterns be designed for each cask? What factors must be considered in selecting a loading pattern?

Response:

The loading patterns for the casks has not been specifically determined. The loading patterns will be based on minimizing the amount of radiation emitted from the casks and ensuring that the decay heat generated by any one cask does not exceed the 24 kW Technical Specifications limit. The loading pattern involves the arrangement of the casks on the ISFSI storage pad and the arrangement of the fuel assemblies within each cask.

The casks with fuel assemblies that have been cooled longer will be on the periphery of the array on the ISFSI storage pad. These casks, which will have lower radiation source terms, will serve as shielding for the casks with fuel assemblies that have been less cooled, which will have higher radiation source terms, that will be placed in the center of the array on the storage pad.

Inside the baskets, the fuel assemblies will be similarly arranged. The fuel assemblies with higher radiation source terms will be placed at the center of the basket and the fuel assemblies that have lower source terms or inserts that would provide additional shielding will be placed in the outer storage cells in the basket internal assembly. The radiation source term for a fuel assembly is based on initial fuel enrichment, fuel burnup, and cooling time.

Question: 73

Figure 5.1-1 gives the estimated times for performing various ISFSI operations. Since some of these times are limited or critical in terms of decay heat, have they been compared to actual times used at other facilities that have used dry cask storage systems?

Response:

PGE has contacted other utilities regarding the Sierra Nuclear VSC-24 system to compare Trojan ISFSI estimated completion times with those actually experienced during system use. Plant configuration differences, fuel handling equipment differences, and procedural differences between utilities preclude an exact comparison to Trojan. The referenced figure is based upon PGE's best estimate of anticipated completion times for the Trojan Plant.

Question: 74

The SAR provides estimates of dose rate for the fully loaded transfer cask, based on site specific calculations. During loading operations, actual dose will vary somewhat from these estimates. Is there a certain measured dose rate at which your procedures would require you to stop and investigate for possible anomalies?

Response:

Upper limits for radiation dose rates will be established during the ALARA job reviews for the spent fuel loading and handling activities. These upper limits have not been established because the loading procedures and loading pattern have not been formalized. The ALARA job reviews cannot be effectively performed until the loading procedures and loading pattern are available.

Question: 75

Section 7.1.2 describes additional features of the GTCC basket design to ensure dose rates similar to the PWR baskets. Please submit the dimensions of these features in order to independently evaluate the effect on the high gamma dose rates anticipated from the GTCC waste.

Response:

The GTCC Basket provides an extra thickness of steel inside the basket shell. This extra thickness of steel is a round cylindrical tube 1 inch thick, 167 inches tall, 62 inch inside diameter and fabricated from a carbon steel plate. An extra steel bottom plate is installed which is 2 inches thick and fabricated from carbon steel.

The GTCC shield lid assembly is 8 inches thick, 64 inches in diameter, and fabricated with multiple layers of steel and lead. The top section is stainless steel. The middle layer is commercial grade lead. The bottom layer is carbon steel. A 0.500 inch thick rolled side plate band is welded to the top and bottom sections to contain the lead center. Two holes for the Swagelok Quick Disconnect fittings are placed through the lid which are similar to those described for the shield lid in question number 31. The extra steel shielding/support plate is fabricated from carbon steel, is 163.3 inches long and is formed to surround GTCC waste cans inside the basket.

Question: 76

Is the Fuel Building sealed from the outside (controlled environment) during PWR basket loading and handling operations prior to the basket being seal welded when airborne contamination or gaseous waste could escape the basket?

Response:

PGE's letter to the NRC, "License Change Application (LCA) 237 - Spent Fuel Cask Loading in the Fuel Building", dated May 29, 1996 (VPN-031-96) states that the Fuel Building ventilation will be operating during spent fuel loading and handling to minimize the amounts of particulate radioactivity released from the Fuel Building to the environment.

Question: 77

Have the environmental issues been evaluated relative to the radiation that birds or small mammals would receive if they nested on or around the casks or in the ventilation openings of the casks in the ISFSI enclosure? The Technical Specifications mention wire mesh screens on the air inlets and outlets of the casks. Are these put in place after the air pad system has been removed? Are these screens designed or adequate to prevent entry by birds or other animals?

Response:

The potential effects of radiation on individual birds or small mammals that might venture into the ISFSI were not addressed in the ISFSI environmental report. Such considerations are typically beyond the scope of the environmental assessments performed for ISFSIs or other nuclear facilities. For example, such considerations were not addressed in the original environmental report for the Trojan Nuclear Plant or the supplement to that report for the post-operations stage.

The ventilation openings in the concrete casks will be screened with a stainless steel mesh to prevent debris from entering the cask and potentially blocking the air flow path. The openings will be inspected on a regular basis (once per 24 hours) to ensure that the air passages have not become blocked. These inspections will ensure that no birds or small mammals are nesting in the ventilation openings.

Other than the ventilation openings, the cask storage area of the ISFSI will not present a particularly attractive environment that would provide nesting sites for birds or small mammals. The cask storage area will consist of a smooth concrete slab with the concrete storage casks resting flush on the slab and arranged in a well spaced manner. The outer surface of the casks (with the exception of the ventilation openings discussed above) are likewise smooth concrete without enclosures or crevices that might provide shelter or nesting sites for birds or small mammals.

Therefore, the operation of the Trojan ISFSI will not result in any significant increase in the radiation exposure to individual birds or small mammals in the vicinity of the Trojan site.

Question: 78

Section 3.3.5.3, p. 3-21, states that "the estimated working dose rates for the PWR cask (maximum fuel burnup) and GTCC cask are 10 mrem/hr and 31 mrem/hr, respectively, and the dose at 100 meters (based on a 2000 hour occupancy) is 24 mrem/yr as described in Chapter 7." Please provide the assumptions and calculations for these results. Also, if the estimate is 24 mrem/yr at 100 meters, what is the estimate at the "Restricted Area" fence, which is less than 100 meters from the ISFSI pad?

Response:

The estimated working dose rates, which are provided in ISFSI SAR Tables 7.4-1 and 7.4-2 and Figures 7.3-9 through 7.3-12, were calculated using various methods and assumptions because the dose rates are a combination of gamma radiation, neutron radiation (for fuel only), and scattered radiation, and the configuration of the cask is not uniform (e.g., air inlets and air outlets). The methodology and assumptions used to calculate the estimated dose rates for the casks are described in detail in ISFSI SAR section 7.3.2.2. The one-dimensional models used to calculate the direct gamma dose rates are shown in ISFSI SAR Figures 7.3-1 through 7.3-8. The model for calculating the air inlet dose rate and the 2-dimensional models used for calculating neutron dose rates are attached. (Attachment III)

The 24 mrem/yr dose at 100 meters is based on 2000 hour occupancy and was calculated by assuming that the concrete casks would be spaced 15 feet center-to-center in a 6 by 6 array on the ISFSI storage pad. The calculation credited cooling times greater than 5 years, but assumed 40,000 MWd burnup for the fuel casks. The calculation includes direct and scattered dose rate contributions. Using these same methods and assumptions, the dose rate at the Restricted Area boundary, which is about 40 feet from the casks, is estimated as 0.26 mrem/hr.

Question: 79

Please provide an explanation or rationale for the conflict in radiation limits between 10 CFR 72 and the Decommissioning Plan: 25 mrem/yr in direct radiation from the ISFSI beyond the boundary of the Controlled Area, 5 μr/hr above background (per Oregon Administrative Rules) and 15 mrem/yr (from NUREG 1500). How will this conflict be resolved? In areas governed by the 15 mr/ yr and 5μr/hr limits, how will masking of these limits by the 25 mr/yr be avoided?

Response:

PGE will perform surveys that show that the limits of 10 CFR 72, NUREG-1500, and the Oregon Administrative Rules are satisfied. These limits do not conflict because the limits have different bases (i.e., direct radiation exposure per year based on occupancy rates, exposure per year to an "average individual" from the soil, and direct radiation dose rate per hour).

In addition, the survey for the ISFSI site (for termination of the 10 CFR 50 license) will be performed prior to radioactive material being stored at the ISFSI (see PGE letter VPN-018-96 dated May 23, 1996). Therefore, the final survey for 10 CFR 50 license termination will not be affected by radiation from the ISFSI.

Question: 80

Section 7.3 states that the gamma and neutron dose rates for the Trojan casks are based on scaled versions from the VSC-24 calculations. Did PGE personnel do a thorough analysis of the VSC-24 calculations? Is there operational data on the VSC-24 calculations that substantiate their reliability?

Response:

PGE personnel did not perform a thorough analysis of the VSC-24 calculations. However, the results of these calculations were reviewed to determine if the results were reasonable. This was accomplished by comparing the calculated dose rates for the VSC-24 system to actual field dose rate measurements taken at Palisades where a VSC-24 system is currently in use. The calculated VSC-24 dose rates compare favorably with the field measurements from Palisades.

Question: 81

Section 7.2.5.1 states that the radiation protection instrumentation and personnel used to monitor the ISFSI will be provided by an offsite facility, presumably a contracted service. Will such contracted equipment and personnel be used to verify that radiation limits are not exceeded? Will this contract and its administration be considered a function that is "important to safety"?

Response:

PGE will maintain responsibility for ensuring that radiation limits are not exceeded even if radiation protection services are provided by a contract service.

Radiation protection is a quality-related activity, but is not "important to safety" as defined in ISFSI SAR Section 3.4. As a quality-related activity, the contract would be reviewed by Nuclear Oversight and Nuclear Oversight would audit or surveil the activities performed under the contract in accordance with the Quality Assurance Program.

Question: 82

Section 7.5.3.2.4 states that "surveys for contamination at the storage cask air inlets and outlets are routinely performed to confirm that contamination is not present." Additionally, other contamination surveys and external radiation measurements are taken in appropriate areas. Since such ISFSI radiation measurements are not complex in terms of geometry or variability, has remote surveillance equipment been considered or will it be used to meet ALARA goals?

Response:

Use of remote radiation monitoring instrumentation is not planned. Any dose saved by using remote instrumentation would be offset by maintenance and calibration of the instrumentation. It is anticipated that radiation measurements will be obtained at the same time as the contamination surveys or other required surveillances, hence, would add very little dose.

Question: 83

Section 7.6.2 states that the calculated dose at 100 meters from the ISFSI is 24 mrem/yr. Please provide the calculation and assumptions.

Response:

The 24 mrem/yr dose at 100 meters is based on 2000 hour occupancy and was calculated by assuming that the concrete casks would be spaced 15 feet center-to-center in a 6 by 6 array on the ISFSI storage pad. The calculation is based on Trojan fuel and credited cooling times greater than 5 years, but assumed 40,000 MWd burnup for the fuel casks. The calculation includes direct and scattered dose rate contributions.

Question: 84

Section 2.1.2, p. 2-3, states that "the doses that could be anticipated at the Controlled Area boundary from an off-normal event or accident are below the limits of 10 CFR 72.106 and Oregon Administrative Rule 345-26-390." the OAR limit of 0.5 rem TEDE is based on the EPA PAG and is 10 times less than the 10 CFR 72.106 limit of 5 rem. All current ISFSI installations in the US are at operating sites, which have a higher standard for an accidental release of radioactivity than for a plant undergoing decommissioning. Please confirm that calculated accident doses at the Controlled Area Boundary for the ISFSI conform to lower limit of less than 0.5 rem.

Response:

The non-credible accident postulated in ISFSI SAR Section 8.2.1 results in the highest radiological consequences for ISFSI operation. The whole body dose for this accident at 325 meters, as shown in ISFSI SAR Table 8.2-2, is 0.05 rem which is well below the OAR limit of 0.5 rem.

Question: 85

Table 7.4-1 lists the "Maximum Expected Dose Rates for the Storage Cask System (Fuel)." The maximum expected surface dose rate for the concrete cask top and side is 98.7 mrem/hr and 18.9 mrem/hr, respectively. The NRC's Certificate of Compliance, Docket Number 1007, for the VSC-24 states that the "external surface dose rate from all types of radiation will be less than 20 mrem/hr on the sides and 50 mrem/hr of the top." The maximum expected on the top of the Transtor cask is double the limit given in the C of C for the VSC-24 cask. What accounts for this difference?

Response:

The basket shield lid design for the VSC-24 has neutron shielding material that is sandwiched between steel plates. The basket shield lid designed for the Trojan ISFSI does not have the neutron shielding material. In addition, the dose rates for the Trojan storage system are calculated using Trojan's fuel.

Question: 86

Table 7.4-1 demonstrates that shielding is increased by a factor of 13.4 between the transfer cask and concrete cask surfaces for the PWR Basket (252.9/18.9 = 13.4). Table 7.4-2 shows shielding to be increased by a factor of 2.6 between the transfer cask and concrete cask surfaces for the GTCC Basket (133.8/52.2 = 2.6). Is this difference due to the kind of radiation emitted, that is, neutron versus gamma, or other factors not described?

Response:

The difference is due to the type of radiation, neutron or gamma, and energy levels of the gamma radiation. The concrete cask shielding and transfer cask shielding are different and attenuate the different types of radiation and different energy levels of gamma radiation differently.

To illustrate the difference due to the type of radiation, the dose rate from the PWR basket needs to be broken down into neutron and gamma radiation. For the transfer cask, the 252.9 mrem/hr dose rate is composed of 163.2 mrem/hr neutron and 89.7 mrem gamma radiation. For the concrete cask, the 18.9 mrem/hr dose rate is composed of 0.7 mrem/hr neutron and 18.2 mrem gamma radiation. Using these numbers, the ratios of neutron-to-gamma dose rates for the PWR basket in the concrete cask and in the transfer cask are 163.2/0.7 and 89.7/18.2. These ratios show that the concrete cask is more efficient at attenuating neutrons than gammas when compared to the transfer cask.

To illustrate the difference due to the gamma energy level, the ratios for the gamma dose outside the concrete cask and transfer cask for the PWR basket and GTCC basket are 89.7/18.2 versus 133.8/52.2. The difference in these ratios is due to the gamma radiation flux for the spent fuel peaking at a lower energy than the GTCC waste gamma. The lower energy gammas from the spent fuel are more effectively attenuated than the higher energy gammas from the GTCC waste by both the concrete cask and the transfer cask.

Question: 87

Table 7.4-3 gives the Estimated Personnel Exposure Doses while Operating the Fuel Cask System. The numbers on this table account for 50.5 manhours from loading a transfer cask to moving a concrete cask to the ISFSI pad. This table, however, does not account for supervisory personnel, and it accounts for only 1 hour exposure for welding both the shield and structural lids. Please provide justification for the manhour estimates. How does this estimate compare to industry experience? What measures are planned based on lessons learned to achieve ALARA goals?

Response:

The man-hour estimates for the loading and movement to storage activities in Tables 7.4-3 and 7.4-4 were provided by the vendor and are based on industry experience. The estimated cumulative exposure per cask is consistent with industry experience.

These estimates were not intended to be exact and account for every person who may be in the work area. Sur. . isory personnel are not included because they do not need to be as close to the work area as t' workers and would receive a much smaller dose than the workers. AL.^?A job reviews, which will be more accurate and provide limits, will be performed prior to loading activities.

The 1 hour exposure time for welding is 1 hour of exposure for 3 different people for a cumulative total of 3 person-hours of exposure. This exposure time considers setting up the automatic welding machine and NDE, but does not include the time for welding because the workers will move away from the basket while welding is in progress.

Trojan personnel have contacted counterparts at other utilities that are using casks from the same vendor to learn from the other utilities' experience. As loading and handling procedures for the ISFSI are developed, it is anticipated that the practice of learning from the other utilities experience will continue. In addition, the ALARA job reviews will incorporate lessons learned as the loading of the casks progresses.

Question: 88

Tables 7.4-3 and 7.4-4 give the exposures in terms of "working dose rates," which are defined on Tables 7.4-1 and 7.4-2 as the "calculated dose rate one meter from the surface." Are these estimates sufficiently conservative given that many personnel, such as welders and inspectors, will be working closer than 1 meter from the surface of a cask?

Response:

The estimates in Tables 7.4-3 and 7.4-4 are considered reasonable because the estimated cumulative exposure per cask is consistent with industry experience. The exposure times and dose rates for the different work activities will vary. Radiation protection personnel will be available to advise workers on minimizing exposure.

Question: 89

What is the total anticipated exposure for the ISFSI, including ISFSI decommissioning? Our review of the Decommissioning Plan indicated 58 person-rem. Adding the estimates on Tables 7.4-3 and 7.4-4 and including surveillance dose indicates approximately 266.4 person-rem, not counting decommissioning.

Response:

ISFSI SAR Section 7.4 provides the required information for calculating a total anticipated exposure for the ISFSI. The 58 person-rem in the Decommissioning Plan only accounts for fuel transfer from the Spent Fuel Pool to the ISFSI.

From ISFSI SAR Section 7.4, if 34 fuel casks and 2 GTCC waste casks are assumed to be in storage from January 1998 until January 2018 and radioactive decay over the 20 year storage period is ignored, then the total dose would be 61.41 rem plus 20 years times 5 rem/year, which is about 161 rem. No person-rem are added for decommissioning the ISFSI because no radioactive contamination of the ISFSI is anticipated.

Question: 90

Is remote reading provided for the temperature monitoring devices described in Section 3.3.3.2 for ALARA considerations? If not, please explain.

Response:

Use of remote temperature monitoring instrumentation is not planned. Similar to the answer to question #82, any dose saved by using remote instrumentation would be offset by maintenance and calibration of the instrumentation. It is anticipated that temperature measurements will be obtained at the same time as visual inspections, hence, will not add an appreciable dose.

Question: 91

Is there a credible possibility that there is failed fuel in the SFP that is not currently designated as such or known to be failed? If so, how would failed fuel be detected? If not, how would this be verified or why would the scenario not be credible?

Response:

An inspection of spent fuel will occur prior to loading spent fuel in the ISFSI. The inspection will determine the current condition of the spent fuel and will identify failed fuel to ensure proper storage within the basket.

Question: 92

Describe the program for doing the accident analyses in terms of how calculations are checked and independently verified.

Response:

Accident analysis calculations are prepared and independently checked in accordance with written procedures. If the calculation is prepared by PGE, then the calculation is prepared and independently checked per procedure TPP 18-9, "Trojan Calculations". If the calculation is prepared by the vendor under a PGE approved Quality Assurance program, then the calculation is prepared and independently checked per vendor procedures and is also reviewed by PGE per procedure TPP 18-9, "Trojan Calculations".

Question: 93

Section 8.0 states that "events that could occur during loading of the Concrete Cask in the Fuel Building and transport to the pad will be addressed in a separate submittal." Please submit that separate submittal for review as soon as it is available. This additional submittal will be considered in ODOE's evaluation of the ISFSI under OAR 345-26-390.

Response:

Postulated off-normal events and accidents related to spent fuel loading and handling are provided in PGE's letter to the NRC, "License Change Application (LCA) 237 - Spent Fuel Cask Loading in the Fuel Building", dated May 29, 1996 (VPN-031-96). A copy of that letter was provided to the Oregon Department of Energy.

Question: 94

Section 8.0 states that the accident analysis is based on the heavier, higher temperature and pressure PWR Basket. However, since the radiation emitted from the two kinds of baskets is markedly different, are there accident scenarios in which the more conservative approach in the analysis would be to use the GTCC Basket?

Response:

Where the results of an off-normal event or accident could cause an increase in direct radiation from the cask, i.e., tornado missile (ISFSI SAR Section 8.2.4.3) and turbine missile (ISFSI SAR Section 8.2.14.3), the GTCC waste is addressed.

Question: 95

Section 8.1.1.1 describes a lateral impact of the PWR Basket against the inside of the Concrete Cask and concludes that the "PWR Basket is designed to withstand acceleration loads which bound this handling load. No corrective actions are required." What happens to the ventilation cooling (thermal loads) when the PWR Basket is against the Concrete Cask and one side of the vertical ventilation channel is effectively blocked? Is there corrective action to re-center the PWR Basket? Is the centering of a basket in a concrete cask a concern?

Response:

PGE will provide a response to these questions by August 29, 1996.

Question: 96

Section 8.1.3 describes the analysis for the airborne release of contamination. Please provide a copy of the assumptions and calculations for this analysis. What are the deep dose equivalent and TEDE results for this release? Why was 2.4 mrem chosen? How do the assumptions and results in this analysis compare with the calculation for decommissioning which showed that a release of 2.07 Ci would result in a dose of 0.5 rem TEDE at the Exclusion Area Boundary? [The concern here is due in part to the example given in the VSC-24 SAR (p.11-9) using the same methodology. In the VSC-24 example a person 200 meters from a 2.32 Ci release receives a total of 1 mrem dose. This result is considerably different from the results presented in the Decommissioning Plan for a similar airborne release, and it makes the method used in the ISFSI and VSC-24 SARs appear unconservative by comparison.]

Response:

The assumptions and calculation of the small release of radioactive particulates from the exterior of the basket are provided in detail ISFSI SAR section 8.1.3.1.3.

The external dose (DDE) from submersion in a cloud of Co-60 particles is less than one percent of the internal dose (CEDE) due to inhalation of the Co-60 particles in the same cloud. Therefore, the DDE is considered insignificant. The TEDE is equal to the DDE plus the CEDE, and in this case since the DDE is insignificant, the TEDE is equal to the CEDE.
The 2.4 mrem dose was determined by calculating the dose at 100 meters based on basket surface contamination levels of 1 x 10-4 μCi/cm2 beta-gamma. The 100 meters is the minimum distance for the Controlled Area boundary, and 1 x 10-4 μCi/cm2 beta-gamma is the surface contamination limit used for the VSC-24 storage system.

The 2.07 Ci release described in the Trojan Decommissioning Plan is calculated to result in a committed dose equivalent (CDE) of 0.5 rem to bone surfaces. The TEDE from this release was calculated to be 57 mrem to an individual standing at the site boundary (662 meters from the point of release). The same calculation methods were used to calculate the 2.4 mrem TEDE to an individual standing 100 meters from the point of release of 946.1 μCi of Co-60 from the external surfaces of the basket. If the release from the basket is increased to 2.07 Ci and calculated at 662 meters for comparison with the postulated Trojan Decommissioning Plan release, the TEDE is about 162 mrem. The difference between the 57 mrem and 162 mrem is due to the difference in radionuclides postulated to be released, with the Co-60 release having more conservative, i.e., higher, results.

PGE did not review the calculation from the VSC-24 SAR where an individual 200 meters from a release of 2.32 Ci of Co-60 receives a total dose of 1 mrem. However, this dose appears to be based on an external (skin and body) dose. The Trojan ISFSI SAR dose and the 0.5 rem dose in the Trojan Decommissioning Plan are based on an inhalation dose, which is a different methodology and the results would not be comparable to the VSC-24 results.

Question: 97

Section 8.1.3.1.3 gives an atmospheric dispersion factor of 0.035 sec/m^3, whereas in Section 2.3.4 the atmospheric dispersion factor, based on meteorology, was given as 0.0043 sec/m^3. Please explain the derivation of these numbers.

Response:

The atmospheric dispersion factor at 325 meters is determined to be 0.0043 sec/m^3 by reading the value directly from Figure 1 of Regulatory Guide 1.25 (Safety Guide 25).

The atmospheric dispersion factor at 100 meters is calculated to be 0.035 sec/m^3 by the formula in section C.2.a of Nuclear Regulatory Commission Regulatory Guide 1.25 (Safety Guide 25) "Assumptions Used for Evaluating the Potential Radiological Consequences of a Fuel Handling Accident in the Fuel Handling and Storage Facility for Boiling and Pressurized Water Reactors" using lateral and vertical diffusion coefficients of 4 meters and 2.3 meters, respectively, from Figures 3.10 and 3.11 of "Meteorology and Atomic Energy 1968" (Attachment VII). This method is used instead of Figure 1 in Regulatory Guide 1.25 because Figure 1 is not plotted at distances less than 200 meters.

Question: 98

Section 8.2.3.2 derives the potential energy of a loaded cask using heights for the center of gravity as 126.9 in and 68 in. How were these derived? The vertical center of gravity of a cask is between 108.4 and 109.3 (Table 4.2-4) or 109.5 (Figure 8.2-2) plus the additional height of 17.8 in (Figure 8.2-2) to overturn the cask, which results in a figure between 126.2 and 127.3 in. Because of the air flow channel around the basket, the horizontal center of gravity would be less than 65 in. (Figure 8.2-2), perhaps 61 in using the figures in the VSC-24 SAR. Assuming a cask loaded with a dry PWR basket that shifts to one side as it falls, the potential energy of the cask would be dE= 290,000 (126.2 - 61) = 1.9 x10+10^7 lb-in.

Response:

The maximum height of the vertical center of gravity is derived from the radius of the cask bottom being 130 in/2 = 65 in (this takes into account the 3 inch chamfer at the outside diameter of the cask bottom). The vertical center of gravity is taken as 109 inches and was chosen to bound the values shown in Table 4.2-4 for a loaded, dry basket in a cask. Thus, the maximum height is

$$h_v = (109^2 + 65^2)^{.5} = 126.9 \text{ in}$$

The height of the cask center of gravity for the overturned cask is taken as the cask horizontal center of gravity, 136 in/2 = 68 in (i.e., cask diameter).

As noted in the text, the tipover analysis is considered a beyond design basis event since the analyses of credible events did not predict cask tipover. The method used in this section coupled the masses at the center of the concrete cask. Uncoupling the masses and shifting the basket 4 inches to the side would shift the overall center of gravity about one inch. Using the shifted center of gravity, the total energy becomes

$$290{,}000\ (126.9 - 67) = 1.74 \times 10^7 \text{ lb-in}$$

The equivalent drop height would then be

$$1.74 \times 10^7 / 290{,}000 = 59.9 \text{ inches}$$

Since 60 inches was used as the drop height, there is no change in the results.

Question: 99

Section 8.2.5.2 refers to NUREG/CR-0098 to justify using one peak acceleration and two at 40% of peak values. Please justify using this as a conservative approach. If as stated, the peak horizontal ground acceleration was developed from the geometric mean of the two peak horizontal ground accelerations, than the horizontal accelerations would be 0.27g in both the x and y directions. Using the standard method of assuming all peak accelerations, the ratio of Restoring Moment to Overturning Moment would still be > 1.

$$[(1.0-.25)(65)]/[.38(109.5)] = 1.17$$

Another concern is the effect of a shifting center of gravity due to the basket sliding to one side. Please address this.

Response:

There is no "standard method" that uses peak accelerations in all directions simultaneously. The accepted method for nuclear plants is to use the square root of the sum of the squares of the orthogonal directions.

The Standard Review Plan for Nuclear Plants (NUREG 0800) in 3.7.2.6 states "When the response spectra method is adopted for seismic analysis, the maximum structural responses due to each of the three components of earthquake motion should be combined by taking the square root of the sum of the squares of the maximum co-directional responses caused by each of the three components of earthquake motion at a particular point of the structure or of the mathematical model."

The approach using 100%, 40%, 40% is described in NUREG/CR-0098 and explains the validity of the approach. EPRI Report NP-6041, Revision 1, "A Methodology for Assessment of Nuclear Power Plant Seismic Margin" states in section 4 regarding the 100,40,40 method "This method is equivalent to, but slightly more conservative than, the SRSS method but retains the correlation between displacements, loads, and stresses which is normally lost in the SRSS method."

The methodology used to determine the SME did not use a simple vector addition of the two orthogonal directions. The peak ground acceleration is based on analysis of empirical data and numerical modeling studies. The value used represents the results of a regression analysis of geometric means of measured responses to earthquakes worldwide. The value includes a standard deviation for further conservatism.

A shift of the basket 4 inches would result in a shift of the center of gravity of the loaded concrete cask of about 1 inch.

A further conservatism in the SAR as presented is that the safety factors for overturning as listed are in reality the factors of safety against uplift of one edge of the cask. The actual forces to tip the cask over are significantly higher, since the center of gravity of the cask must be raised as well as offset the center to edge distance.

Question: 100

Please provide the origin of the 19.5 value for the Moment calculation on p. 8-25.

Response:

The side of the concrete cask is assumed to be cantilevered at the bottom of the inside liner.

The 19.5 inches is the distance from the bottom of the outside of the concrete cask to the bottom of the liner on the inside. There are 21.5 inches from the bottom of the cask to the top of the liner bottom plate and the bottom plate is 2 inches thick.

Question: 101

Section 8.2.5.2 states that the seismic shear and moment are included in the structural evaluation in Table 4.2-13. However, the referenced table lists thermal properties. (Section 8.2.4.2.5 on missile impacts also references Table 4.2-13).

Response:

Section 8.2.5.2 and Section 8.2.4.2.5 incorrectly referenced Table 4.2-13. The correct reference is Table 4.2-10. This correction is being included in the forthcoming revision to the Trojan ISFSI SAR.

Question: 102

Section 8.2.7.2 mentions "two lower outlets which become inlets." Please provide a drawing showing these outlets. None of the cask descriptions indicated that the four outlets were at different heights.

Response:

The air outlet assembly is shown on drawing ISFSI-NQI81106-65 (PCC-004 Sheet 1/1). Item 7 (Attachment VII) shows the dimensional differences between the air outlets that are higher and the air outlets that are lower.

Question: 103

Section 4.2.5 gives a thermal analysis of the cask system and Section 8.7.2 describes an accident scenario involving the full blockage of air inlets. These are based on engineering calculations and theoretical performance. Since similar cask designs have been used at other utilities, can you provide empirical data to support these calculated results?

Response:

Full blockage of the air inlets is considered an accident condition. PGE is not aware of tests conducted by other utilities that would involve blocking the concrete cask air inlets for a prolonged period of time while fuel is being stored in the concrete cask.

However, thermal, shielding, and operational performance of vertical, ventilated concrete casks was demonstrated by full scale testing as described in Pacific Northwest Laboratory report, PNL-7839, "Performance Testing and Analyses of the VSC-17 Ventilated Concrete Cask". Figure 4-9 in PNL-7839 (Attachment IX), shows the temperature profiles for different conditions, including full blockage of the air inlets. The performance of the Trojan ISFSI storage system would be generally similar because the VSC-17 cask is designed by the same vendor and has similar operating principles, i.e., natural circulation cooling, as the Trojan ISFSI storage system.

Question: 104

In order to evaluate or verify any of the accident analyses presented in Chapter 8 (or the calculations presented in Chapter 4), dimensions of the baskets and concrete casks are required. Simple mechanical and thermal stresses, as well as other basic engineering equations to evaluate basket and cask performance, require these dimensions. Please provide them.

Response:

Both the PWR and GTCC Basket shells are 181.3 inches tall, 66 inches in outside diameter, .750 inches thick and fabricated from stainless steel. Descriptions of both lids have been provided in response to question(s) 31, 59, and 75. The response to question number 75 also provides a description of the differences between the PWR Basket and the GTCC Basket regarding additional shielding.

The concrete cask is 211.5 inches tall, 136 inches in outside diameter, 74 inches inside diameter, 21.5 inches thick at the bottom, and has a 2 inch thick carbon steel liner covering the interior side and bottom areas with the exception of the air inlet and outlet vent openings. The bottom of the cask has two 48.5 inch wide and 3 inch tall air inlet channels parallel to each going across the full diameter of the cask. The bottom of the cask is covered with a .500 inch carbon steel plate. Air inlet and outlet vents penetrate the bottom and top side of the cask to permit air to move through the annulus area formed when the basket is placed inside. The lid dimensions have been provided in the response to question number 30.

Question: 105

Please provide the calculations, assumptions, and applicable references to evaluate and verify the conclusions presented in Table 8.1-2.

Response:

This table relates to the concrete cask thermal analysis. The SNC calculation(s) requested is proprietary and is not included with this response. The ANSYS computer model is used in the thermal analysis. Thermal analysis is performed to calculate the temperature distribution of the TranStor™ Concrete Cask. The temperature distribution of the concrete cask was calculated for various design cases using finite element analysis. These temperatures are used to check against material temperature limits and to check the structural integrity through the concrete cask thermal stress analysis (separate calculation). The calculation also determines the maximum temperature at the TranStor™ Basket shell surface and is used in the basket thermal analysis (separate calculation).

Six steady-state cases were considered for the concrete cask containing a sealed basket and for the concrete cask containing a basket with overpack. The concrete cask heat transfer characteristics are modeled using the ANSYS computer program. The ANSYS model uses a slice of 10 degrees of the concrete cask because of symmetry.

The active fuel region in the interior of the basket is modeled as a heat generation region with an effective thermal conductivity. This effective conductivity was estimated only to accommodate the solution method employed by ANSYS. The only interest in the basket in this model is the surface heat flux and the basket shell temperatures, which are not governed by the thermal conductivity of the fuel region. The top and bottom of the active fuel region are modeled as helium only. The gap between the basket shell and basket overpack is also modeled as helium. No conduction is assumed across the air annulus (between basket and concrete cask liner) and between the top of the basket lid and the concrete cask lid.

Radiation, heat generation rates, ambient temperatures, and solar radiation heat input are also considered in the analysis.

Question: 106

Is an analysis available that demonstrates adequate cooling ventilation after a Basket Overpack has been installed?

Response:

The overpack was included in the cask finite element model used in the thermal analysis discussed in the response to question 105.

Question: 107

Due to the height of the concrete casks, will adequate visual inspections of the air outlets be possible?

Response:

Yes. Each air outlet opening is covered with stainless steel mesh. Because of the passive design, air is continuously flowing in the bottom and out the top vents of the system making it highly unlikely that foreign substances would be drawn into the air outlets. A portable stairway, small lifting device, or hand held extension mirror could be utilized if closer examination of the air outlet interior were necessary.

Question: 108

The only radiation measurements described in the ISFSI SAR are quarterly readings of TLDs and quarterly area surveys. Taking readings four times a year does not seem adequate to detect radiation from accidental release. If, for example, fuel failure did occur shortly after a quarterly survey, how would it be detected? Has PGE considered continuous monitoring? Why is the periodic measurement of radiation not required by the Trojan ISFSI Technical Specifications?

Response:

Fuel cladding failure during storage is not postulated because the fuel is stored in an inert atmosphere and at temperatures that will minimize fuel cladding degradation over the 40 year design life of the storage system. However, in the unlikely event that fuel cladding should fail while in storage, quarterly surveillances are not expected to detect the cladding failure because a radiological release would not occur (the spent fuel is inside the sealed basket) and direct radiation would not increase (radiation shielding, i.e., basket and concrete cask, would not be affected by a cladding failure).

Continuous monitoring is not planned because the anticipated radiological consequences of postulated off-normal events and accidents are very small. In ISFSI SAR Chapter 8, the "Small Release of Radioactive Particulates from Exterior of Baskets" in Section 8.1.3.1, is the only credible event that could result in a radiological release. If the 36 baskets in the ISFSI have surface contamination at the stated level, the exposure from this event is 2.4 mrem at 100 meters, which is very small when compared to the Environmental Protection Agency Protective Action Guide of 1,000 mrem during the early phase of an event.

Periodic radiation measurements are not included in the current Trojan Technical Specifications, but the programs that implement the periodic measurement are specified. Similarly, the ISFSI Technical Specifications, in Section 5.7.2.1 and 5.7.2.2, require implementation of a Radiation Protection Program and Radiological Environmental and Effluent Monitoring Program, which will specify the required measurements and intervals.

Question: 109

What emergency measures will be taken if a loss of power occurs during the basket loading operations?

Response:

Spent fuel will be loaded into the basket by moving fuel assemblies from the spent fuel racks in the Spent Fuel Pool to the basket in the Cask Loading Pit using the Spent Fuel Pool Bridge Crane. The Spent Fuel Pool and Cask Loading Pit will be filled with borated water during basket loading. The Spent Fuel Pool Bridge Crane requires electrical power to move the Bridge north and south and to operate the hoist. Therefore, a loss of electrical power would halt basket loading operations.

If a fuel assembly were being moved at the time of the loss of power, the fuel assembly would be suspended more than 9.5 feet underwater, which limits the dose rate at the water surface to no more 2.5 mrem/hr. The Spent Fuel Pool water will adequately cool the suspended fuel assembly.

Vertical motion of the suspended fuel assembly would be stopped by the brake on the electrical hoist. If the suspended fuel assembly has cleared the spent fuel racks, then it could be moved east or west by using the manually operated trolley. This movement may be desired as a precaution if the suspended fuel assembly is over another fuel assembly in the Spent Fuel Pool racks.

Prompt actions would be taken to restore power to the Spent Fuel Pool Bridge Crane.

Additionally, LCA 237 established that the Spent Fuel Building ventilation will be in operation during spent fuel loading and handling in the Fuel Building.

Question: 110

The PWR Basket may not be built to the exacting standards required for a fuel assembly in the reactor vessel. At Palo Verde, a fuel assembly became stuck. What assurance is there that this would not happen during ISFSI operations at TNP, particularly for a bowed assembly? What procedures are available to deal with such an occurrence?

Response:

A Westinghouse assembly is 8.426 inches square and a Babcock and Wilcox assembly is 8.425 inches square (some may be 8.426 inches square due to a difference in grid strap design). Spent fuel pool storage cells are 8.8125 inches square and the PWR Basket storage cells are 8.8 inches square. The fabrication specification for the PWR Basket requires drag testing each cell with a full length gauge to ensure straightness and squareness (8.6 in x 8.6 in x 14 ft long gauge). The corner storage cells are slightly larger and could accommodate bowed fuel assemblies. Procedures will be written for spent fuel transfer to the ISFSI, including recovery from abnormal events and accident conditions.

Question: 111

What is the procedure if the air pad system fails during transport of the cask to the ISFSI pad?

Response:

There is no time limit associated with movement of the loaded concrete cask from the Fuel Building to the ISFSI pad. If the air pad system fails, the loaded concrete cask will be lowered to the ISFSI pad or surface over which the concrete cask is being moved from the lift height of the air pads, which is about 3 inches. The air pad system could be repaired in-situ (e.g., ruptured air hose) or the air pad system could be removed from the air inlets and repaired (e.g., leaking air pad). The amount of time that the air pad system can be inserted into the air inlet openings will be limited to 12 hours, which corresponds to the normal surveillance interval for measuring the concrete cask air outlet temperature, to ensure that adequate decay heat is being removed from the basket.

Question: 112

Is there lab testing or empirical performance data to support the conclusions regarding damage to the cask from a tornado driven missile?

Response:

PGE has not performed laboratory testing (and thus does not have empirical performance data) to support the conclusions regarding damage to the cask from a tornado driven missile. PGE relied upon data referenced in the Trojan ISFSI SAR Section 8.4 to analyze potential tornado driven missile damage. The empirical formulas used in the missile analysis are based on the tests performed by EPRI (References 8.8 and 8.9 of the ISFSI SAR)

Question: 113

What would be the consequences if all of the He were to gradually leak out of a cask? How would this be detected?

Response:

The consequences of a helium leak have not been formally analyzed because the basket, transfer cask, and storage cask are designed such that the integrity/confinement properties of the basket are maintained for the spectrum of credible off-normal events and accidents.

However, the procedures for responding to the off-normal events and accidents discussed in ISFSI SAR Chapter 8 will include methods to determine if the basket is leaking. These methods include, depending on the event, non-destructive examination of welds and basket lid surface, use of a helium sniffer, sampling for radioactive gases or airborne contamination, and/or monitoring the air outlet temperature of the storage cask for subtle increases or decreases.

Question: 114

Section 9.1.1.2.5 describes an Independent Review and Audit Committee (IRAC) with a minimum of five members whose responsibility is to advise "the Trojan Site Executive and Plant General Manager on matters relating to the safe storage of spent nuclear fuel." Section 9.1.2.5 (and part 5.5.2 of PGE-1071, ISFSI Technical Specifications) describes an ISFSI Safety Review Committee with a minimum of three members and an identical responsibility. Why are there two committees with overlapping functions? Please clarify the responsibilities of each.

Response:

As stated in ISFSI SAR Section 9.1 (Organizational Structure), Section 9.1.1 describes the organization that will be in place during ISFSI design, construction, preoperational testing, fuel loading, startup testing, and initial operation while Section 9.1.2 describes the organization that will be in place during long term operation of the ISFSI. At some point after completion of fuel loading, the IRAC will cease to exist and will be replaced by the ISFSI Safety Review Committee.

Question: 115

Section 9.2.3.1.1 states that the vacuum drying system will be tested to ensure that vacuum drying can be accomplished within the 36 hour time limit. Is the Fuel Building ventilation system operated in conjunction with the vacuum drying and if so, is it also tested at this time?

Response:

Fuel Building ventilation will be operated during spent fuel loading and handling activities in the Fuel Building. Formal testing of the Fuel Building ventilation system is not planned, but periodic, routine maintenance, such as filter replacement, will be performed to adequately assure proper operation.

Note that the 36 hour time limit for draining down the basket has been recalculated as 37 hours and will be changed in a forthcoming revision to the ISFSI SAR. The vacuum drying time limit will be administratively controlled at less than 20 hours.

Question: 116

Section 9.2.3.1.3 mentions that the "air outlet temperature monitoring system will be tested and calibrated on each concrete cask prior to inserting a loaded basket." Please describe this system. Is it a remote sensing system?

\Response:

The air outlet temperature monitoring system consists of 4 resistance temperature detectors (RTDs), one of which will be located in each of the 4 air outlets. The RTDs are connected to a junction box that will be mounted about 5 feet from the bottom of the concrete cask. The current plan is to take local readings for each cask at the junction box. However, a remote system could be wired into this junction box in the future, if desired.

Thermocouples may be used instead of RTDs. Either device will provide acceptable temperature monitoring.

Question: 117

Section 3.3.3.2 states that "The temperature monitoring devices are commercial grade." How will PGE ensure that these devices remain accurate over 40 years? The ISFSI Technical Specifications do not describe calibration requirements for these devices, which are used to monitor the only surveillance requirement listed in the ISFSI Technical Specifications. Please explain.

Response:

Trojan ISFSI SAR Section 5.1.3.4 requires that the temperature monitoring devices be calibrated in accordance with Quality Assurance requirements. PGE will procedurally establish the calibration criteria in accordance with PGE-8010 "Trojan Nuclear Plant Nuclear Quality Assurance Program".

PGE is not limited in the type of temperature monitoring device it utilizes to monitor cask outlet temperature. One method is the use of 4 resistance temperature detectors (RTDs), one of which will be located in each of the 4 air outlets. Another method would be the use of hand held pyrometers. Either method would be in compliance with both SAR and Technical Specification requirements.

Question: 118

Section 9.2.3.2 states that the startup test will measure the decay heat of each cask and determine if the loading is as designed. What are the corrective actions planned if the decay heat exceeds 26 kWt?

Response:

The ISFSI Technical Specifications limit the basket heat load to 24 kW, although the thermal design and analysis considers 26 kW.

It is unlikely that a basket will be loaded over 24 kW thermal because 24 kW thermal is based on low initial enrichment, high burnup, and a short amount of cooling, which is not typical for Trojan fuel. However, if the startup test determines that a basket is loaded above 24 kW thermal, then an engineering evaluation will be performed to confirm that 24 kW thermal has been exceeded and to determine acceptable corrective actions. The corrective actions that would be proposed and/or implemented would be reported to and be reviewed by regulatory agencies because exceeding the thermal limit would be a Technical Specifications violation. Potential corrective actions could include applying additional cooling until the decay heat is within the Technical Specifications limit, or unloading spent fuel from the basket.

Question: 119

Section 9.4.1 references ISFSI Technical Specifications, Section 5.7, which lists written procedures to be established, implemented, and maintained. Item f is the "Conduct of routine ISFSI spent nuclear fuel storage operations and surveillances." Does this item include the inspection, pre-operational testing, startup testing, loading of the baskets, and use of the Transfer Station?

Response:

Pre-operational testing, startup testing, loading of the baskets, and use of the transfer station will be performed in accordance with written procedures. Each of these evolutions are planned to occur only once per cask per lifetime of the ISFSI. Therefore, the evolutions are not "routine", hence, were not intended to be covered by Item f of the ISFSI Technical Specifications.

Question: 120

PGE-1072 describes the ISFSI Training Program. The list of subjects given in academic training and OJT does not specifically include loading the baskets with fuel or GTCC waste, or loading fuel debris or failed fuel into special cans. These are operations unique to the ISFSI. How will training be accomplished for these activities? Will these activities be included in mockup training?

Response:

Basket loading will be performed under the 10 CFR 50 license and PGE-1072 is not applicable et loading will be directly supervised by a Certified Fuel Handler. PGE-1057, "Certified Fuel Handler Training Program", will be modified to include the activities that are different from the current program, i.e., loading fuel assemblies into a basket assembly, loading GTCC waste, loading fuel debris, and loading failed fuel into special cans. This training will be accomplished by classroom instruction with the exception that each person who will move and load fuel assemblies will be required to move and load a dummy fuel assembly.

The "mockup" is a part length basket for testing the welding and cutting equipment as described in ISFSI SAR Section 9.3.2.1.1. The "mockup" will not be used to simulate basket loading.

Question: 121

Since there has been virtually no actual fuel handling since 1993, how will PGE fuel handlers maintain the necessary skills prior to loading fuel into the PWR Baskets?

Response:

PGE fuel handlers remain qualified to move fuel per the requirements of the Certified Fuel Handler Training Program. As additional assurance of proficiency, each person who will move and load fuel assemblies will be required to move and load a dummy fuel assembly.

Question: 122

Will ISFSI Training Instructors include people with experience in ISFSI operations at other utilities or nuclear facilities?

Response:

Current plans do not include using instructors from other utilities at Trojan. However, the training plans used for training Trojan ISFSI personnel will include industry experience.

Question: 123

Will mockup training be performed prior to conduct of ISFSI loading operations? What is the planned scope of mockup training? What areas involving equipment, activities, and operations deemed important to safety are not planned to be included in mockup training? Will the mockup training include using a basket overpack?

Response:

The testing described in ISFSI SAR Section 9.2.3 will be performed prior to actual loading operations and is intended to "dry run" the procedures involved with the loading process prior to actual loading.

The lessons learned from this testing will be incorporated into the training for actual loading. Each person involved in the actual loading process will be appropriately trained for the operations that they will be required to perform, but each person may not be involved in the testing of equipment that they may be required to operate, e.g., testing of shield lid retainers. However, each person who will move and load fuel assemblies will be required to move and load a dummy fuel assembly.

Testing of the overpack is described in the ISFSI SAR Section 9.2.3.1.5.

Question: 124

Has the passive cooling system on the concrete casks been tested with a 26 kWt or higher heat source?

Response:

Thermal, shielding, and operational performance of vertical, ventilated concrete casks was demonstrated by full scale testing as described in Pacific Northwest Laboratory report, PNL-7839, "Performance Testing and Analyses of the VSC-17 Ventilated Concrete Cask". The test was performed with a cask heat load of 14.9 kW and demonstrated that the VSC-17 cask had good heat transfer performance and only a small increase in fuel temperature when the inlet vents were partially blocked. The VSC-17 cask is designed by the same vendor and has similar operating principles, i.e., natural circulation cooling, as the Trojan ISFSI storage system.

Question: 125

Section 9.6 deals with ISFSI decommissioning, but there are many issues that are not addressed. How will decontamination be accomplished if most of the equipment and systems used for that purpose have been removed from the decommissioned plant? What types of equipment and systems are anticipated for decontamination? What is the estimated personnel radiation exposure for decommissioning? Will area surveys be conducted to release the site for unrestricted access?

Response:

As described in ISFSI SAR Section 7.5.3.2.3, contamination of the ISFSI is not anticipated because the basket is checked for loose surface contamination prior to movement to the ISFSI pad and the radioactive materials are inside the sealed basket. Decontamination is not anticipated for ISFSI decommissioning.

However, as stated in ISFSI SAR Section 9.6.1, if decontamination of the ISFSI is required, then decontamination will be accomplished by routine radiation protection practices which could be provided by contracted services as described in ISFSI SAR Section 7.5.2. No special equipment is anticipated.

No exposure is projected for ISFSI decommissioning because contamination of the ISFSI is not anticipated. The exposure associated with transfer and off-site shipment of the loaded baskets is already included in the dose estimate in ISFSI SAR Section 7.4.

The ISFSI site will be surveyed prior to release for unrestricted access.

Question: 126

Will a spare basket or cask be available in the event fuel has to be reloaded from one to another? If a cask needs to be removed from service, how is this accomplished? What happens?

Response:

Baskets are designed for minimum 40 year life, hence, moving fuel from one basket to another is not anticipated. However, a basket may be placed in an overpack as described in ISFSI SAR Section 5.1.1.5 (see also ISFSI SAR Figure 5.1-2) rather than moving the fuel to another basket.

Concrete casks are designed for a minimum 40 year life, hence, concrete casks are not anticipated to be "removed from service" until the loaded basket has been transferred to a shipping cask for transportation to a DOE high level waste repository or interim storage facility. However, a basket could be moved to a newly constructed concrete cask, if desired, by using the sequence described in ISFSI SAR Figure 5.1-3 and by substituting the newly constructed concrete cask for the shipping cask.

Question: 127

The Defueled Technical Specifications, Section 3.1.4, restricts loads carried over the spent fuel pool and the heights at which they can be carried to preclude impact energies over 240,000 in-lbs. How will this requirement be modified prior to loading the PWR and GTCC Baskets?

Response:

The restriction in the Permanently Defueled Technical Specifications will not be changed during spent fuel loading and handling. The current Trojan Technical Specifications only imposes this restriction on racks which store spent fuel. Specifically this specification states that, loads carried over the fuel in the spent fuel pool racks, and the heights at which they may be carried, will be limited in such a way as to preclude impact energies over 240,000 inch-lbs if the loads are dropped.

Question: 128

The Technical Specification surveillance requirement for the air temperature at the outlet of the cask is 220°F. The NRC's Safety Evaluation Report for the SNC Ventilated Storage Cask System (VSC-24) lists as an operational limit 110°F above ambient. Both cask systems are designed to handle similar heat loads, yet the VSC-24 cask has a lower limit for outlet air temperature (as long as the ambient temperature is below 110°F). What is the justification for a higher limit for the Trojan ISFSI?

Response:

An air outlet temperature of 220°F corresponds to a concrete temperature of 250°F which is an established functional and operating limit per proposed Trojan ISFSI Technical Specification 2.1.1 (see B 3.1.1 Bases of Technical Specifications for additional detail). Trojan ISFSI SAR Section 4.2.3.2.4 discusses justification for exposing concrete to long term temperatures as high as 250°F as opposed to the VSC-24 SAR which is limited to 225°F, therefore the Trojan ISFSI design is analyzed for higher temperatures.

Question: 129

Will QA/QC review all ISFSI procedures that are deemed quality-related or important-to-safety? Will QA/QC hold points be put into ISFSI procedures?

Response:

As required by procedure TPP 12-4, QA will review each ISFSI procedure which is quality-related. Per this same procedure, a determination is made of the quality-related status of each ISFSI procedure. ISFSI equipment classified as "important to safety" is quality related.

Additionally, Nuclear Oversight has developed an Integrated Assessment Plan (NOIAP) for the ISFSI, which identified current plans for assessment activities. The NOIAP states, in part, "as part of the initial audit of SNC, Nuclear Oversight reviewed a number of SNC QA Procedures. A similar process will be used in evaluating sub-tier suppliers' procedures."

In accordance with procedures QP 17-11 and TPP 17-3, certain procedures and work instructions for ISFSI construction, testing, and fuel loading will contain Circle Q Notification points and QC Inspection Hold Points. Additionally, per the NOIAP, Nuclear Oversight plans to review fabrication and construction work packages in order to evaluate the inclusion of inspection hold points by the fabricator/constructor and SNC. Nuclear Oversight may also choose to include their own inspection hold points.

Question: 130

Will the QA organization verify commitments to 10 CFR 72 and other NRC requirements related to the ISFSI?

Response:

PGE Nuclear Oversight will verify compliance with the QA program, applicable regulatory requirements (including 10 CFR 72), and commitments made in ISFSI licensing documents on a sampling basis as part of their audits and surveillances. PGE QA Surveillance of ISFSI Licensing Activities (Report # 96-006-SURV) provides a recent example of this type of verification.

Additionally, as Nuclear Oversight performs its review of Quality-Related procedures, applicable CFR and licensing document commitments are routinely used as references to ensure the procedure content provides implementation steps for the applicable commitments.

Question: 131

In what aspects of the manufacture of the cask systems will TNP's QA personnel be involved, and to what extent? Will there be a TNP QA representative onsite for the manufacturing process? How many vendor audits will be performed and what will be examined? Will QA evaluate the weld records (radiographs, NDE) of important-to-safety equipment fabricated by SNC and its subcontractors? Will the SNC QA organization be evaluated for its independence from cost and scheduling? Will the QA evaluation include examination of the zinc coatings described in Section 4.2.3.2.6?

Response:

In accordance with current limitations placed on SNC in the Trojan Approved Suppliers List, "PGE QA approval of all sub-tier suppliers performing quality-related work is required prior to use of the suppliers" and "PGE review of Fabrication shop travelers for inclusion of hold points and verification of SNC hold points is required prior to issuance."

Per the SNC Project Plan, PGE QA will review and approve Sub-vendor procedures.

Per the NOIAP, PGE QA "may choose to establish specific inspection hold points in the fabrication process that would require specific source inspection. At this time, most fabrication oversight is planned to be performed via source surveillance instead of inspection to allow for a broader look at both programmatic and technical controls utilized by the fabricator and SNC."

A copy of the Nuclear Oversight Integrated Assessment Plan for ISFSI (NOIAP) was made available for ODOE review in May 1996.

Current plans in the NOIAP, call for source surveillance to be performed during the complete fabrication process for the first PWR Basket and Concrete Cask. Additional periodic surveillances are planned during the fabrication of subsequent containers as well as the Transfer and Shipping Casks. The frequency of these surveillances will be dictated by the quality and complexity of the work performed. Total surveillance time is expected to be approximately 50% of the fabrication time.

The NOIAP provides current plans for both Supplier and Internal audits. These plans address audits of SNC, SNC Sub-tier suppliers and material suppliers, and various Internal audits. One of these planned Internal audits is an ISFSI Pre-operational Readiness audit, which will verify compliance with ISFSI licensing requirements and project commitments. NOIAP, Section 4.0 and Attachment A, contain a proposed schedule for ISFSI oversight activities, including audits. This proposed schedule, covering April 1995 through November 1998, shows 3 vendor audits, 6 supplier surveillances, and 1 supplier inspection, related to ISFSI.

Per the NOIAP, "specific Nuclear Oversight Nondestructive Examination (NDE) Level III approval will be required for all quality-related weld radiographs."

As a portion of the 1995 PGE audit of SNC (Audit Report 95-05), PGE evaluated the SNC QA organization for its independence from cost and scheduling.

It is currently planned to use a different Fuel Debris Can from that described in Section 4.2.3.2.6., and it is not known at this time if the final can will have a coating or not. However, it is expected that applicable coatings procedures would be reviewed and the application of any coating would be observed as part of a PGE source surveillance.

Question: 132

Please provide copies of the vendor (external or supplier) audits performed by PGE personnel or performed by others and used by PGE as a basis for verifying the quality and acceptability of the products and services provided by SNC and related subcontractors on the ISFSI.

Response:

PGE performed an audit of SNC in April 1995 and issued PGE Audit Report 95-06, PGE Utility Audit of Sierra Nuclear Corporation (SNC), dated May 18, 1995. This audit was the basis for adding SNC to the Trojan Approved Suppliers List. A copy of this Audit Report was made available for ODOE review as noted below.

In addition, copies of the following documents were previously made available to the onsite ODOE representative during the month of May 1996:

1. Nuclear Oversight Integrated Assessment Plan for ISFSI, Revision 1
2. 1995 - 1996 Integrated Audit/Surveillance Schedule, Revision 4
3. PGE Audit Report 95-05, PGE Utility Audit of Sierra Nuclear Corporation (SNC), dated May 18, 1995
4. PGE Surveillance Report 95-18, Surveillance of SNC in Roswell, GA
5. PGE Surveillance Report 95-22, Surveillance of SNC in Scotts Valley, CA
6. PGE Surveillance Report (96-006-SURV) of ISFSI Licensing Activities

Question: 133

Is a QA/QC verification performed on the purity of the helium used to fill the Baskets?

Response:

PGE will purchase and receive the Helium in accordance with procedure TPP 16-1, "Material/Service Procurement and Control Process." Due to the specific ISFSI SAR requirement for this Helium to be 99% pure, it is expected that the Purchase Order will require a Certification of Purity from the Supplier. This documentation will be reviewed for acceptability by Nuclear Oversight as part of the material receipt.

Question: 134

Provide copies of applicable portions of referenced documents needed to complete ODOE review. These are: ACI-318; ACI-349; ANSI-57.9; ANSI-N14.6; ANSI-A58.1 TABLE 12; ASME, Section III, Subsections NG & NC; NUREG 1.60; NUREG 0612; NUREG 0800; NUREG/CR-0098; RG 1.25; RG 3.61; RG 3.62; RG 8.8; RG-8.10; "Seismic Margin Earthquake Study" letter from PGE to NRC dated May 26, 1993; Bechtel Power Corporation letter to PGE, Document Control No. T040898, March 4, 1996, Concrete Cask Temperature Review, GSA #618.

Response:

Copies of the applicable portions of the reference documents were provided on June 11, 1996 (Reference CPY 030-96).

Attachment II
Trojan Nuclear Plant FSAR, Figure 2.5-6

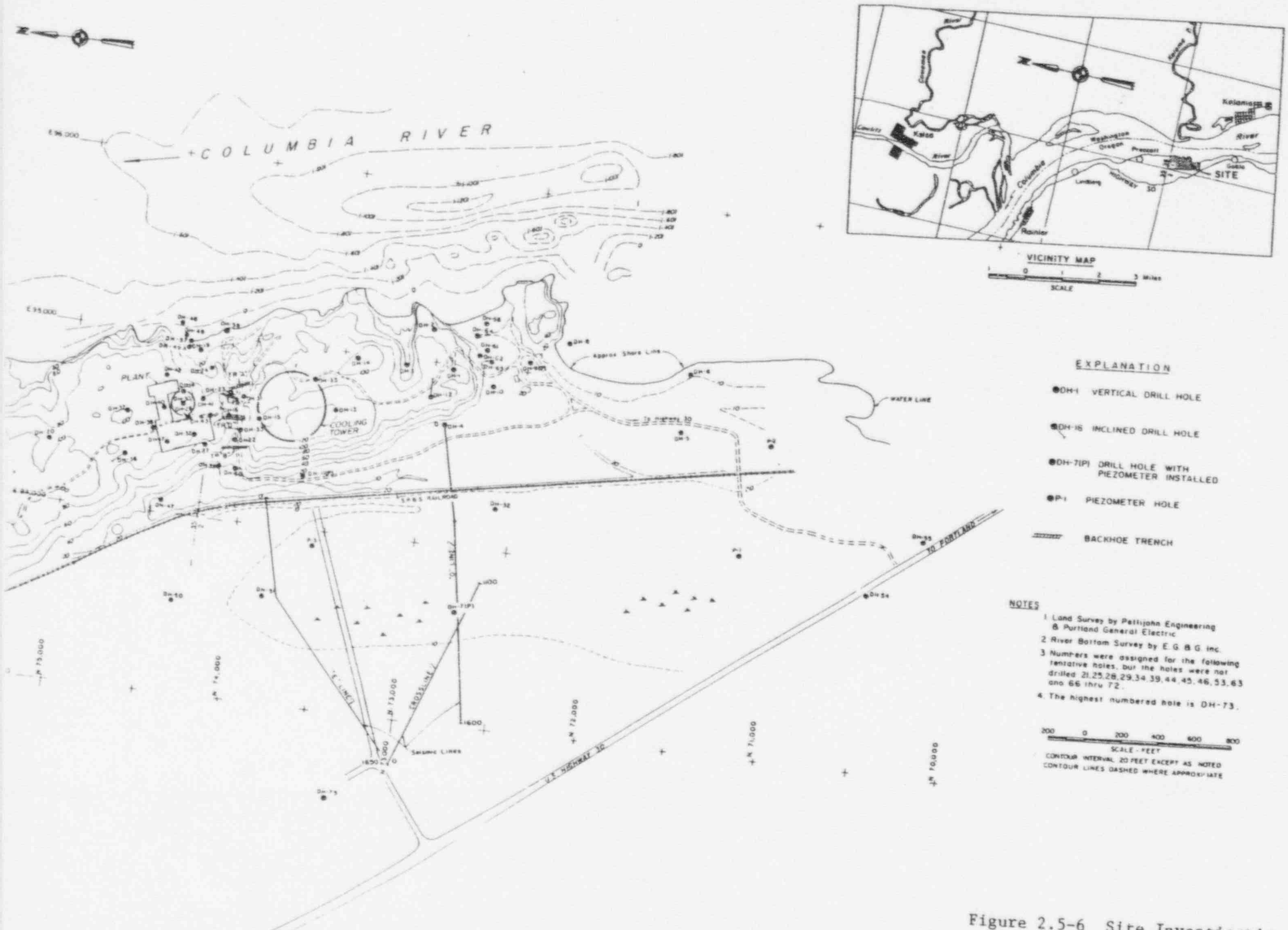


Figure 2.5-6 Site Investigation Map

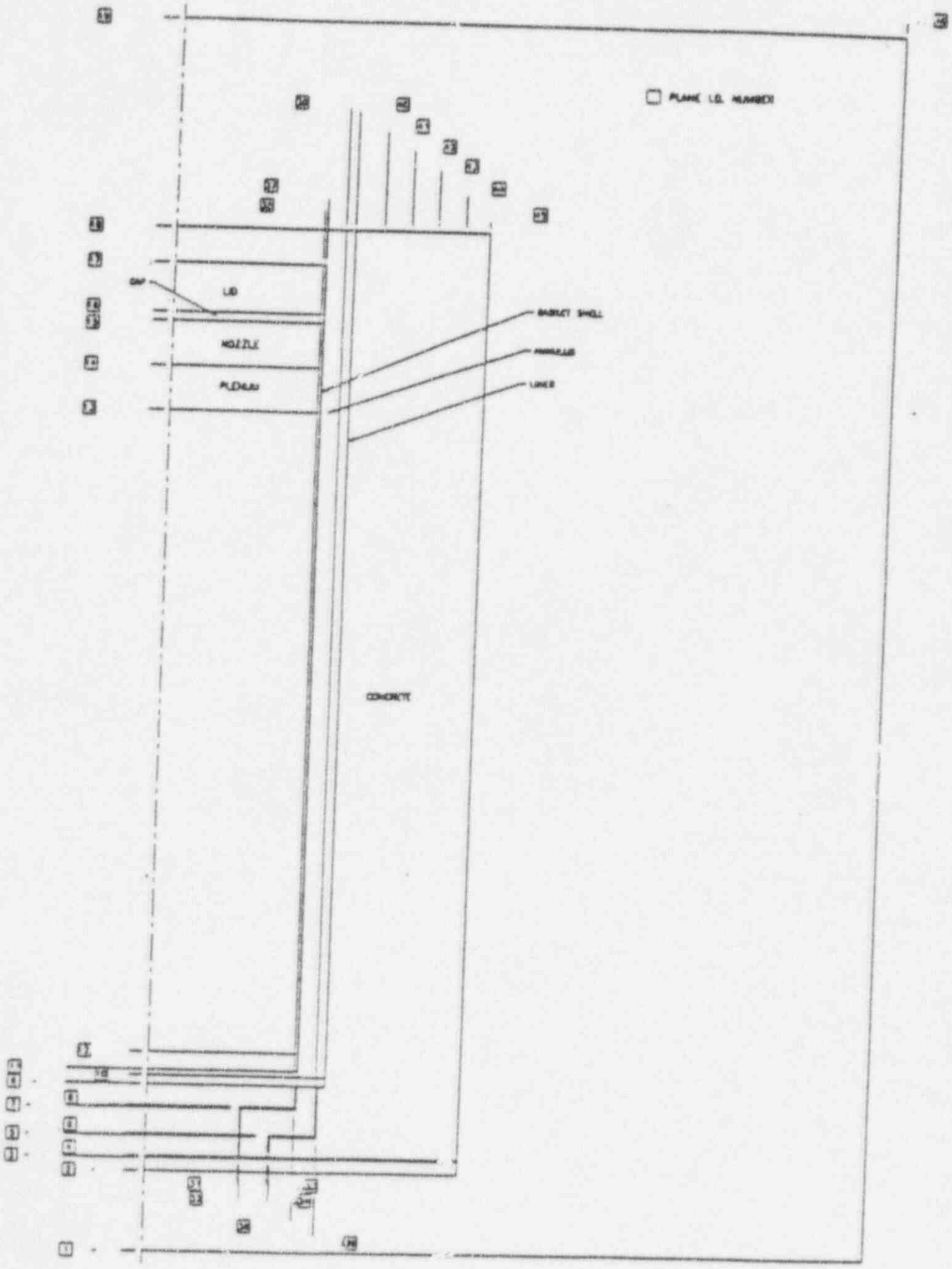
Attachment III
Storage Cask Dose Models (proprietary information deleted)

SNC

Sierra Nuclear Corporation

Figure 6.1

MCNP TranStor™ PWR Storage System
Model Elevations and Plane Identification Numbers

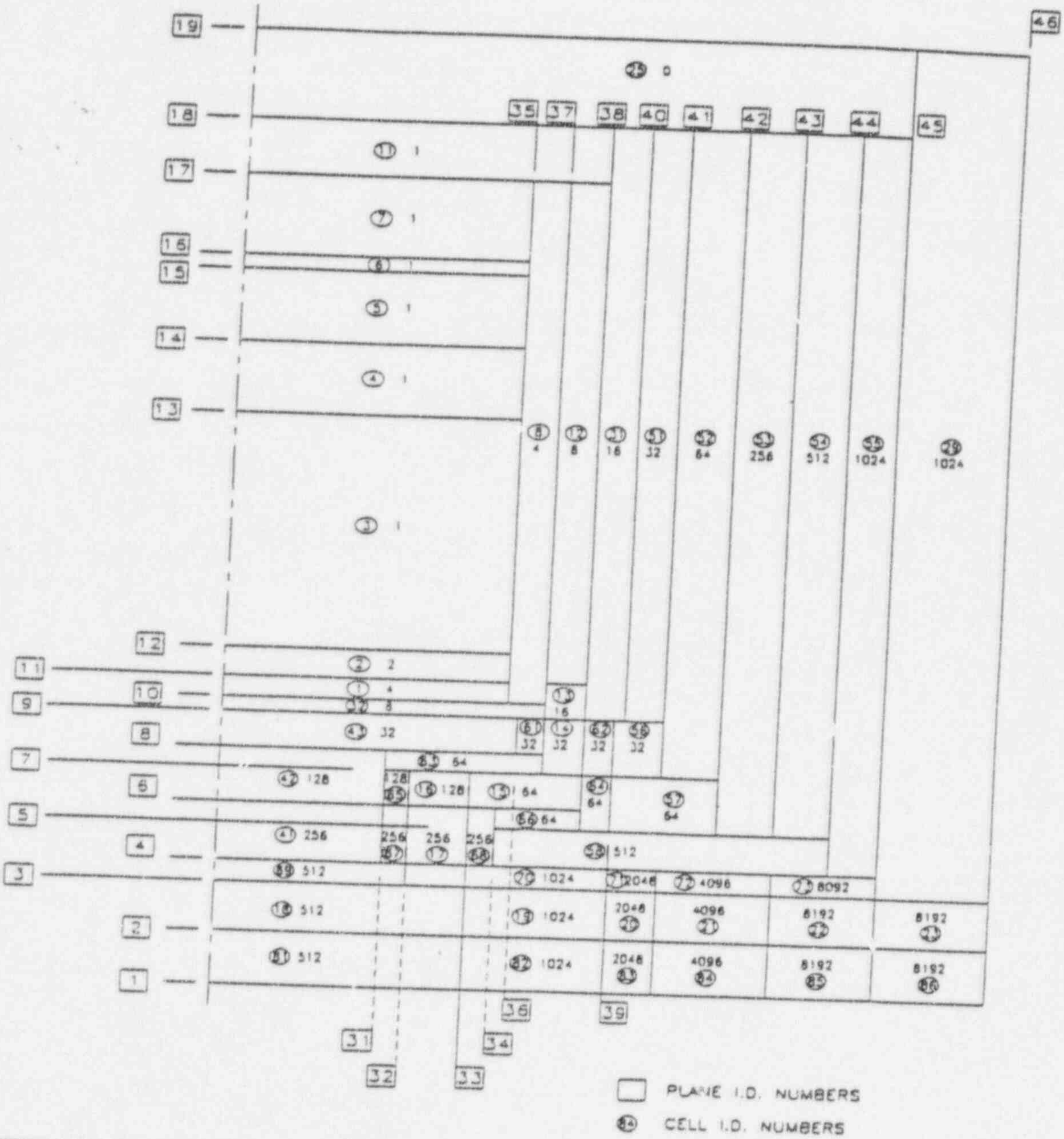


Client/Project	PGE	Revision	Prepared	Date	Checked	Date	Sheet
Subject	TranStor™ Storage Cask Air Inlet and Outlet	0	ll	2-28-96	JEH	2/28/96	12
	Dose Rates for PWR & GTCC Waste Baskets						of
QAP3.0-4 REV 0 (02/95)	Calculation Number PGE01-10.02.01-09						18

SNC
Sierra Nuclear Corporation

Figure 6.2

MCNP TranStor™ PWR Storage System
Importance Regions and Cell Identification Numbers

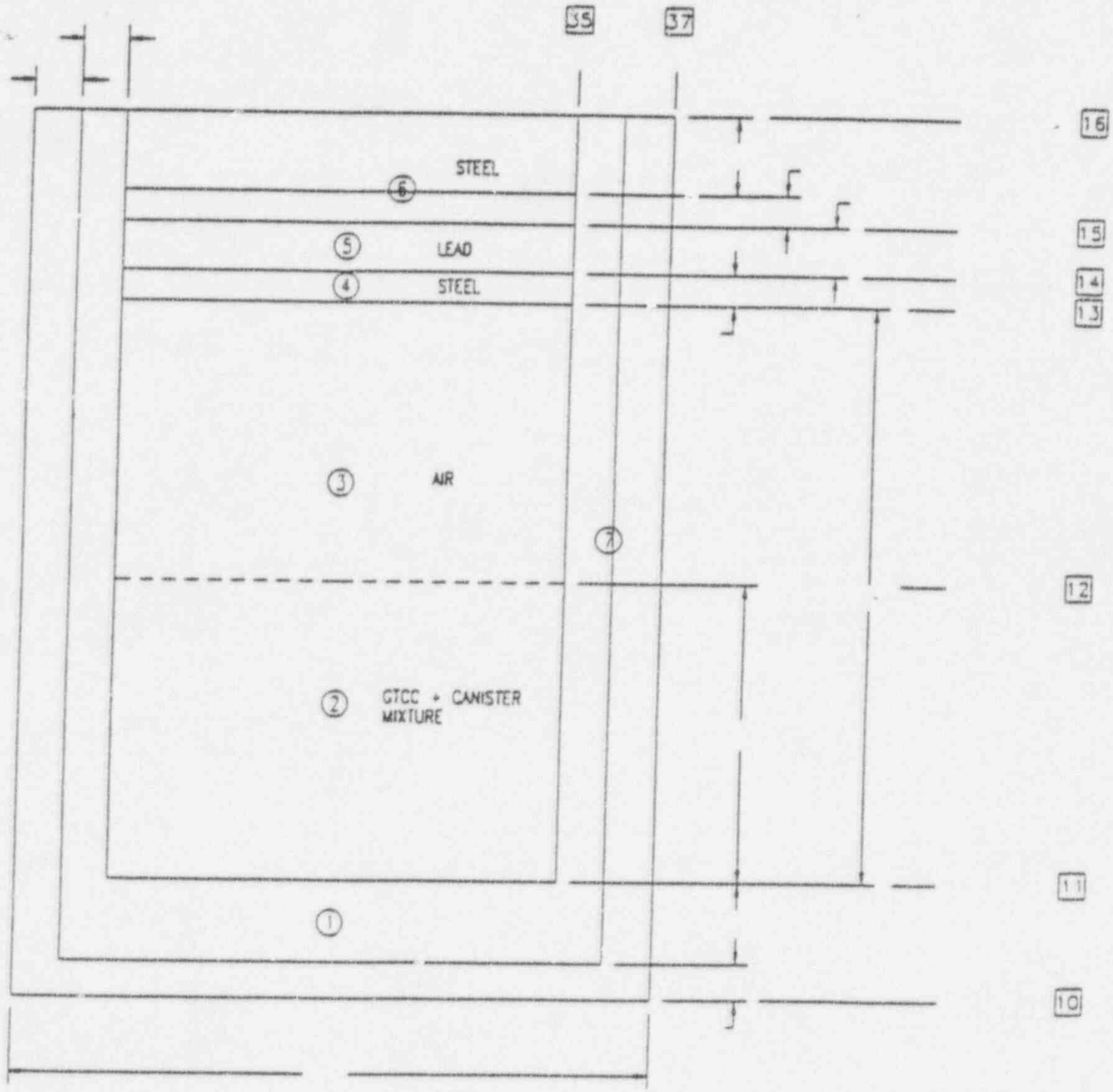


Client/Project	PGE	Revision	Prepared	Date	Checked	Date	Sheet
Subject	TranStor™ Storage Cask Air Inlet and Outlet Dose Rates for PWR & GTCC Waste Baskets	0	JL	2-28-96	JEH	2/28/96	13
QAP3.0-4 REV 0 (02/95)	Calculation Number PGE01-10.02.01-09						18

SNC
Sierra Nuclear Corporation

Figure 6.3

MCNP TranStor™ GTCC Storage System
Importance Regions and
Cell / Plane Identification Numbers



		Revision	Prepared	Date	Checked	Date	Sheet
Client/Project	PGE						
Subject	TranStor™ Storage Cask Air Inlet and Outlet	0	ll	2-28-96	JEH	2/28/96	14
	Dose Rates for PWR & GTCC Waste Baskets						of
QAP3.0-4 REV 0 (02/95)	Calculation Number PGE01-10.02.01-09						18

TranStor™ Transfer Cask

2-D Neutron Shielding Model

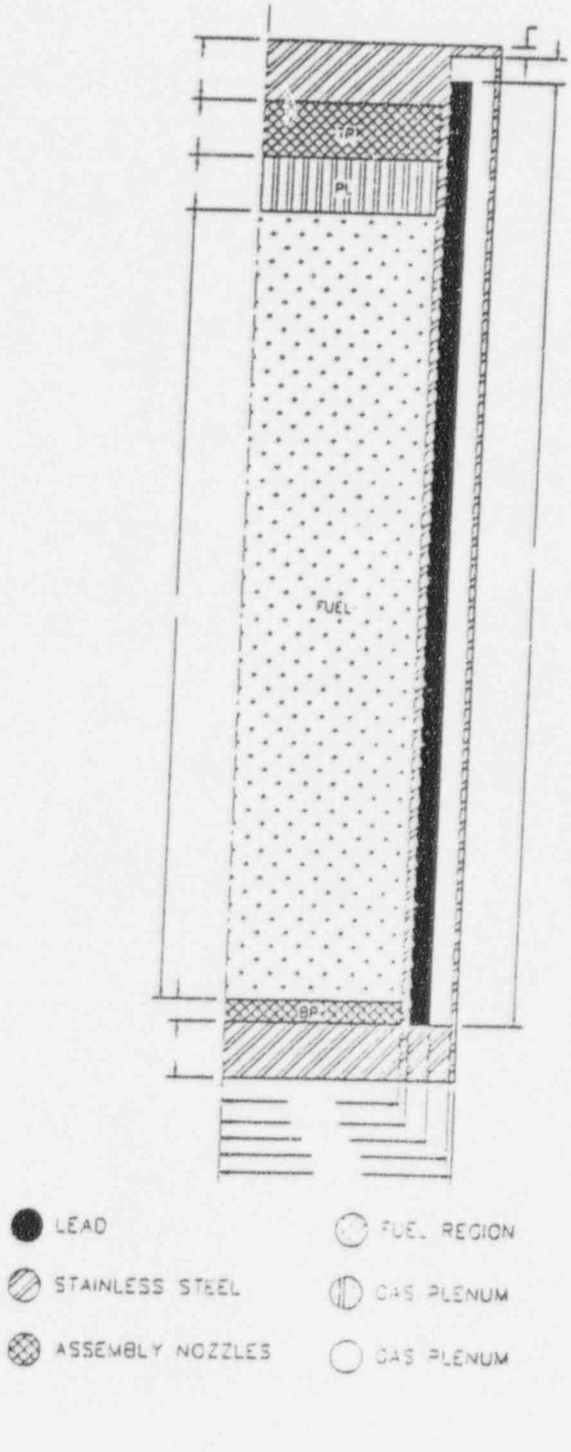


Figure 3.4 - 8

Client Project: Portland General Electric	Revision	Prepared	Date	Checked	Date	Sheet
Subject: TranStor™ Storage Cask Shielding Analysis	0	LR	2-25-96	HP	2-25-96	35
Calculation Number: PGE 01-10.02.01-05						of 97

TranStor™ Storage Cask
2-D Axial MCNP Neutron Shielding Model

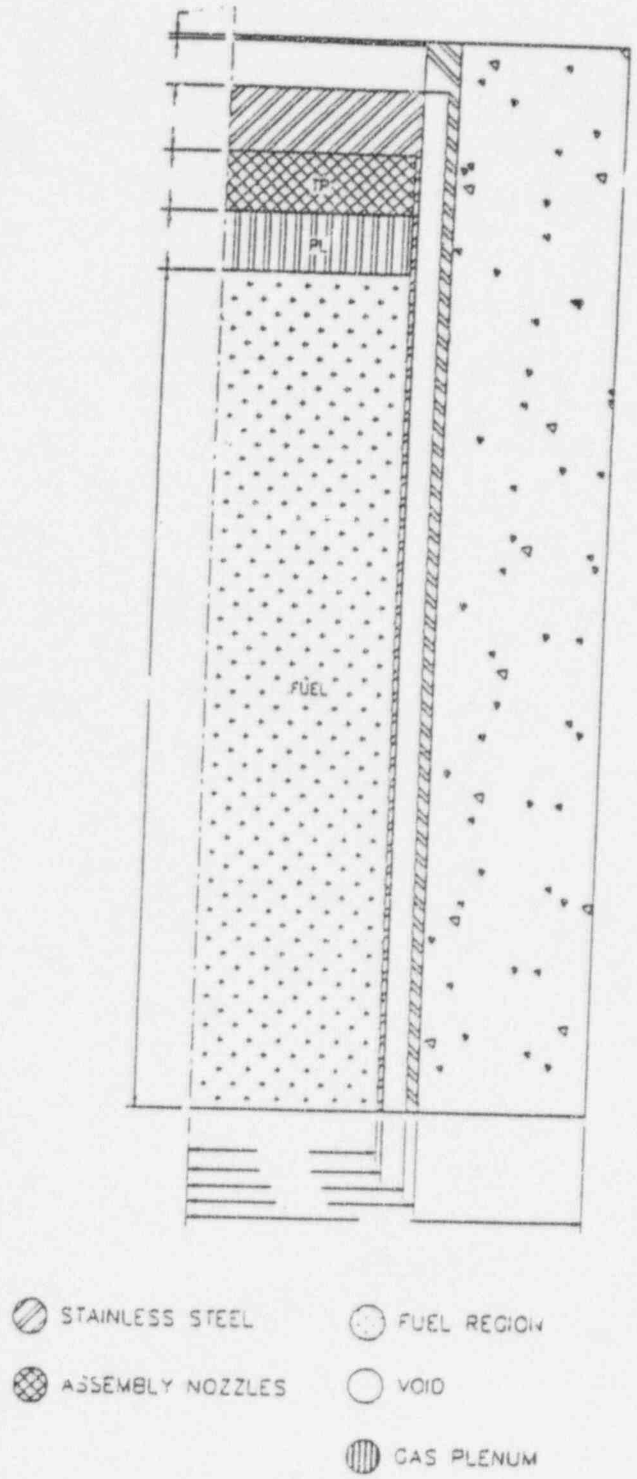


Figure 3.4 - 9

Attachment IV
PNL-6364, "Control of Degradation of Spent LWR Fuel During Dry Storage in an Inert Atmosphere," Figure 4.2

DC6

PNL-6
UC

CONTROL OF DEGRADATION OF SPENT LWR FUEL DURING DRY STORAGE IN AN INERT ATMOSPHERE

M. E. Cunningham
E. P. Simonen
R. T. Allemann
I. S. Levy
R. F. Hazelton
E. R. Gilbert, Project Manager

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the U.S. Department of Energy
under Contract DE-AC06-76RLO 1830

Pacific Northwest Laboratory
Richland, Washington 99352

PNL 4835 1983 Technical Basis for Storage of Zircaloy Clad Spent Fuel in Inert Gases

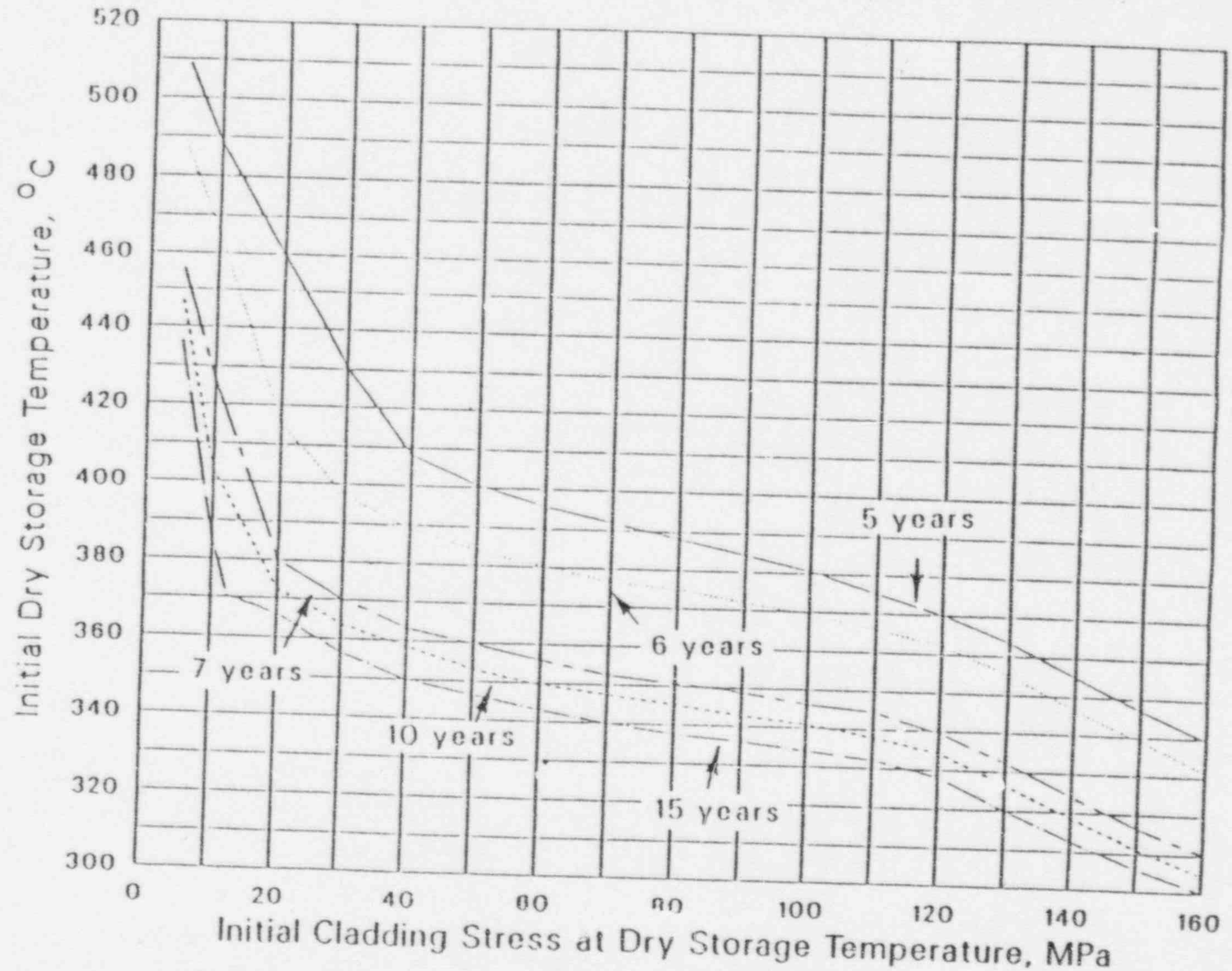


FIGURE 4.2. Comparison of IDS Cladding Temperature Limit Curves for Spent Fuel of Varying Ages

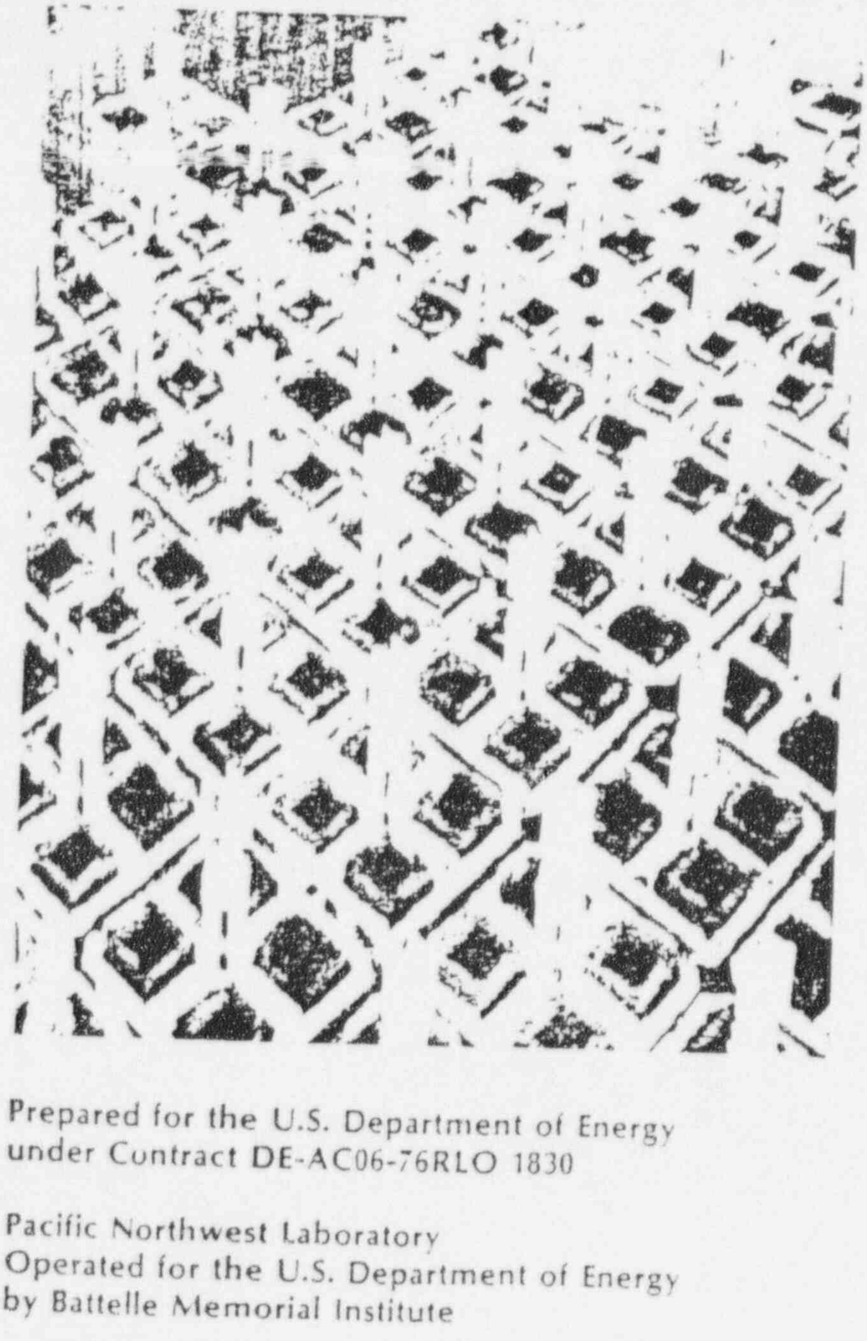
Attachment V

PNL-4835, "Technical Basis for Storage of Zircaloy-Clad Spent Fuel in Inert Gases," Abstract and Executive Summary

PNL-48
UC-

Technical Basis for Storage of Zircaloy-Clad Spent Fuel in Inert Gases

September 1983



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Pacific Northwest Laboratory
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Battelle

PNL-4835

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ABSTRACT

This report summarizes the technical bases to establish safe conditions for dry storage of Zircaloy-clad fuel. Dry storage of fuel with zirconium alloy cladding has been licensed in Canada, the Federal Republic of Germany, and Switzerland. In addition, dry storage demonstrations, hot cell tests, and modeling have been conducted using Zircaloy-clad fuel. The demonstrations have included irradiated boiling water reactor, pressurized heavy-water reactor, and pressurized water reactor (PWR) fuel assemblies. Irradiated fuel has been emplaced in and retrieved from metal casks, dry wells, silos, and a vault. Dry storage tests and demonstrations have involved ~15,000 fuel rods, and ~5600 rods have been monitored during dry storage in inert gases with maximum cladding temperatures ranging from 50 to 570°C. Although some tests and demonstrations are still in progress, there is currently no evidence that any rods exposed to inert gases have failed (one PWR rod exposed to an air cover gas failed at ~270°C).

Based on this favorable experience, it is concluded that there is sufficient information on fuel rod behavior, storage conditions, and potential cladding failure mechanisms to support licensing of dry storage in the United States. This licensing position includes a requirement for inert cover gases and a maximum cladding temperature guideline of 380°C for Zircaloy-clad fuel. Using an inert cover gas assures that even if fuel with cladding defects were placed in dry storage, or if defects develop during storage, the defects would not propagate. Tests and demonstrations involving Zircaloy-clad rods and assemblies with maximum cladding temperatures above 400°C are in progress. When the results from these tests have been evaluated, the viability of higher temperature limits should be examined. Acceptable conditions for storage in air and dry storage of consolidated fuel are issues yet to be resolved.

EXECUTIVE SUMMARY

Wet storage is currently the only licensed interim storage option for spent light-water reactor (LWR) fuel in the United States. Dry storage offers an alternative that appears to be cost effective and technically sound, and it could be implemented on a modular basis to meet near-term storage shortfalls. This report examines the bases to select safe dry storage conditions and emphasizes the evidence for satisfactory cladding integrity.

The world-wide dry storage technology for Zircaloy-clad fuel was assessed, revealing several aspects that are relevant to licensing in the United States:

- Dry storage of Zircaloy-clad fuel in nonoxidizing atmospheres has been demonstrated and licensed in Canada, the Federal Republic of Germany (FRG), and Switzerland. Canadian and Swiss cladding temperatures are relatively low, but FRG dry storage demonstrations have been conducted at maximum temperatures near 400°C. Maximum storage periods range from 1.5 years for FRG demonstrations to 8 years for Canadian fuel. Dry storage tests have been conducted in inert gases up to 570°C without causing rod failures.
- Zircaloy-clad fuel has been emplaced in inert atmospheres and monitored in dry wells, metal casks, concrete silos, and vaults; casks have been emplaced under both wet and dry conditions; fuel retrieval from all of the concepts has also been demonstrated; and both canistered and uncanistered fuel has been emplaced and retrieved. The operations have been conducted with no apparent damage to the fuel.
- A relatively small fraction of U.S. spent fuel assemblies (estimated to be less than 20%) is expected to enter interim dry storage before the federal government begins to accept fuel from utilities (1998). However, that fraction could involve over 20,000 assemblies.
- A large fraction of the U.S. spent fuel inventory has low decay heats, suggesting that fuel can be selected to achieve relatively low storage temperatures, at least for unconsolidated fuel.

- The development of small cladding defects under dry storage conditions cannot be fully ruled out, but dry storage regimes have been identified where cladding defects are not expected to substantially impact fuel storage and retrieval operations.
- Due to thermal gradients in the storage container, only a small fraction of the rods will be at the hottest temperature. In fact, a significant length of each rod will be at temperatures below the maximum temperature. Thus, rod internal pressures that are calculated based on maximum rod temperatures will be conservative.
- Internal pressures will decrease as temperatures decline throughout the storage period, resulting in even more conservative fuel cladding conditions.
- Tests of irradiated fuel with cladding defects in argon at 230°C and 325°C confirmed that storage in inert gas will preclude secondary mechanisms that could propagate cladding defects.

Dry storage demonstrations, hot cell and laboratory tests, Zircaloy materials characterization, and fuel rod modeling have been conducted to assess cladding integrity in dry storage. The following aspects summarize the work that has been completed:

- dry storage demonstrations - Thirty-nine irradiated LWR Zircaloy-clad assemblies[a] containing ~5600 rods have been monitored for up to 5 years. Initial maximum cladding temperatures for these 39 assemblies ranged from 120 to 430°C. Lower temperature demonstrations (120°C) have operated for up to 8 years; demonstrations at higher maximum cladding temperatures (380°C), for up to 1.5 years. No cladding failures have developed in any rods stored in inert cover gases. To date, only one rod has shown evidence of failure; the rod was stored in an air cover gas at ~270°C. Zircaloy-clad pressurized heavy-water reactor fuel rods exposed at up to ~50°C showed no evidence of cladding degradation after 4 years in helium.

(a) 16 boiling water reactor assemblies (BWR) and 23 pressurized water reactor (PWR) assemblies.

- hot cell and laboratory tests - Investigations have been conducted on monitored irradiated Zircaloy-clad rods over a range of dry storage conditions for up to 1 year. Approximately 75 irradiated rods or rod segments have been exposed in inert cover gases from 100 to 570°C; no cladding defects have been detected. In other tests, rods were driven to failure; temperatures of 765 to 800°C were required before the rods ruptured. At slow ramp rates, the cladding failures were pinholes that were not large enough for escape of significant fuel particles. These forced ruptures occurred at temperatures that exceed forseeable dry storage cladding temperatures, even in abnormal storage events.

- Zircaloy materials behavior technology - Much information has been published regarding corrosion, hydriding, radiation damage, creep, and mechanical properties. Assessments of published data provide the basis to model and project cladding behavior in dry storage.

- theoretical treatments and fuel behavior modeling - Peak cladding temperatures up to 430°C (depending on the character of the fuel) are predicted to be acceptable for inert gas storage conditions. Frequently, the temperatures derived from modeling efforts assume isothermal conditions over extended periods (40 to 100 years). Limits derived under this assumption are highly conservative with respect to actual dry storage temperatures, which decrease significantly in a few years.

Fuel integrity results obtained in a given dry storage facility are equally valid for other facilities and concepts operated under similar conditions. Therefore, results reported here are applicable to storage of Zircaloy-clad fuel in metal casks, dry wells, silos, and vaults under equivalent conditions.

Cladding integrity considerations require two principal specifications for dry storage: cover gas selection and maximum cladding temperature.

Cover Gas Selection. Cover gas options include: inert gas (helium or argon); low reactivity gas (nitrogen); or oxidizing gas (air or CO_2). Dry storage in oxidizing cover gases has not been fully defined and requires further investigation. In Germany, two PWR assemblies were stored in moist nitrogen;[a] the assemblies did not show evidence of degradation. The large majority of dry storage experience with Zircaloy-clad fuel involves inert cover gases. Storage in inert gases is the most conservative approach because these gases do not react with the cladding or with exposed fuel if cladding defects are present. Storage conditions in nitrogen appear to deviate only slightly from storage conditions in inert gases.

Maximum Cladding Temperature. Because no cladding failures have occurred in Zircaloy rods stored in inert cover gases, there is no success/failure boundary to serve as a clear selection criterion. Selection of a conservative cladding temperature guideline is therefore based on the following considerations:

- A dry storage demonstration with irradiated fuel assemblies at a maximum cladding temperature of 430°C has operated since February 1983; another demonstration at 385°C has operated since February 1982, and the temperatures have now declined below 250°C.
- Initial modeling assessments have recommended a maximum temperature of 380°C for Zircaloy-clad fuel for storage times up to 100 years.
- The experience base suggests that dry storage at 380°C has ample safety margins. No rods have failed in inert gas exposures up to 570°C, and rods forced to failure required temperatures from 765 to 800°C to produce ruptures.
- Tests and demonstrations are also being conducted at maximum cladding temperatures of 400 to 450°C. After these evaluations are completed, the 380°C guideline should be re-evaluated to determine if it is possible to justify a higher guideline temperature.

(a) One assembly has been stored for ~1 year.

- Storing fuel with cladding defects in inert cover gases is still considered conservative because the defects cannot propagate. However, reasonable fuel screening techniques should be used to minimize the size and number of cladding defects, particularly larger defects where loss of fuel pellets or particles during fuel handling is a possibility.

Due to low heat ratings, older, unconsolidated U.S. fuel is expected to have cladding temperatures considerably below 380°C in dry storage. However, relatively short temperature excursions during cask drying operations suggest a need for temperature margins. Also, allowable temperatures during drying operations may deserve special treatment because the above-normal storage temperature durations are relatively short.

Evidence presented in this report provides a basis to license dry storage of Zircaloy-clad fuel in inert gas at 380°C. Further investigations may be required in the following areas, depending on utility and federal dry storage needs and directions:

- Satisfactory regimes for LWR fuel storage in air could provide a storage option with minimal monitoring requirements.
- Dry storage of consolidated LWR fuel may be demonstrated and evaluated; temperature limits for older consolidated fuel must be considered in the context that temperatures decline less rapidly than for more recently discharged fuel.
- The types of mobile radioactive species and inventories and their impacts during foreseeable fuel recovery and handling operations should be further defined.
- Storage system integrity is a significant consideration but has site-specific and system-specific aspects that are not considered in detail in this report.

- Tests and demonstrations are planned and under way that will augment the statistical data base for 1) longer storage times and 2) temperatures above 400°C over relatively short time spans. Operation at temperatures above 400°C for short periods may be relevant to some cask drying operations or for consolidated or recently discharged fuel.

Attachment VI

"Characteristics of Potential Repository Waste," DOE/RW-0184, Selected printouts

LWR Radiological DATABASE
PHOTONS REPORT
REACTOR TYPE & BURNUP: PWR 40000
ENRICHMENT: 3.02%
DECAY TIME: 5 YEARS
The data is shown in Photons per second/MTIHM

ENERGY (MeV)	PHO/SEC	% TOTAL
1.000E-02	3.573E+15	21.75%
2.500E-02	8.556E+14	5.21%
3.750E-02	9.276E+14	5.65%
5.750E-02	7.107E+14	4.33%
8.500E-02	4.658E+14	2.84%
1.250E-01	4.825E+14	2.94%
2.250E-01	3.878E+14	2.36%
3.750E-01	2.368E+14	1.44%
5.750E-01	6.450E+15	39.26%
8.500E-01	1.605E+15	9.77%
1.250E+00	7.145E+14	4.35%
1.750E+00	1.015E+13	0.06%
2.250E+00	4.436E+12	0.03%
2.750E+00	1.666E+11	0.00%
3.500E+00	2.140E+10	0.00%
5.000E+00	3.519E+07	0.00%
7.000E+00	4.058E+06	0.00%
9.500E+00	4.662E+05	0.00%
TOTAL	1.642E+16	99.96%

Significant Lines

PAGE:

LWR Radiological DATABASE
PHOTONS REPORT
REACTOR TYPE & BURNUP: PWR 40000
ENRICHMENT: 4.42%
DECAY TIME: 5 YEARS
The data is shown in Photons per second/MTIHM

ENERGY (MeV)	PHO/SEC	% TOTAL
1.000E-02	3.645E+15	23.22%
2.500E-02	8.544E+14	5.44%
3.750E-02	9.128E+14	5.81%
5.750E-02	7.215E+14	4.60%
8.500E-02	4.650E+14	2.96%
1.250E-01	4.571E+14	2.91%
2.250E-01	3.879E+14	2.47%
3.750E-01	2.257E+14	1.44%
5.750E-01	6.074E+15	38.69%
8.500E-01	1.364E+15	8.69%
1.250E+00	5.799E+14	3.69%
1.750E+00	8.182E+12	0.05%
2.250E+00	4.390E+12	0.03%
2.750E+00	1.282E+11	0.00%
3.500E+00	1.632E+10	0.00%
5.000E+00	1.284E+07	0.00%
7.000E+00	1.481E+06	0.00%
9.500E+00	1.701E+05	0.00%
TOTAL	1.570E+16	100.00%

Significant Lines

LWR Radiological DATABASE
RADIOLOGICAL TOTALS REPORT
REACTOR TYPE & BURNUP: PWR 40000
ENRICHMENT: 3.02%
DECAY TIME: 5 YEARS

CURIES/MTIHM

ACTIVATION PRODUCTS	8.083E+03
ACTINIDES AND DAUGHTERS	1.469E+05
FISSION PRODUCTS	5.243E+05
TOTAL	6.793E+05

WATTS/MTIHM

ACTIVATION PRODUCTS	8.736E+01
ACTINIDES AND DAUGHTERS	4.482E+02
FISSION PRODUCTS	1.864E+03
TOTAL	2.400E+03

GRAMS/MTIHM

ACTIVATION PRODUCTS	4.403E+05
ACTINIDES AND DAUGHTERS	9.589E+05
FISSION PRODUCTS	4.111E+04
TOTAL	1.440E+06

NEUTRONS/MTIHM

ALPHA,N NEUTRONS	1.342E+07
SPONTANEOUS FISSION NEUTRONS	8.008E+08
TOTAL NEUTRONS	8.142E+08

PHOTONS per Second/MTIHM

TOTAL PHOTONS/SEC	1.643E+16

LWR Radiological DATABASE
RADIOLOGICAL TOTALS REPORT
REACTOR TYPE & BURNUP: PWR 40000
ENRICHMENT: 4.42%
DECAY TIME: 5 YEARS

CURIES/MTIHM	
ACTIVATION PRODUCTS	6.498E+03
ACTINIDES AND DAUGHTERS	1.172E+05
FISSION PRODUCTS	5.328E+05
TOTAL	6.565E+05
WATTS/MTIHM	
ACTIVATION PRODUCTS	7.023E+01
ACTINIDES AND DAUGHTERS	2.750E+02
FISSION PRODUCTS	1.811E+03
TOTAL	2.156E+03
GRAMS/MTIHM	
ACTIVATION PRODUCTS	4.403E+05
ACTINIDES AND DAUGHTERS	9.588E+05
FISSION PRODUCTS	4.120E+04
TOTAL	1.440E+06
NEUTRONS/MTIHM	
ALPHA,N NEUTRONS	7.968E+06
SPONTANEOUS FISSION NEUTRONS	2.916E+08
TOTAL NEUTRONS	2.996E+08
PHOTONS per Second/MTIHM	
TOTAL PHOTONS/SEC	1.570E+16

Attachment VII
"Meteorology and Atomic Energy 1968," Figures 3.10 and 3.11

meteorology and atomic energy 1968

David H. Slade, Editor
Air Resources Laboratories

Prepared by
Air Resources Laboratories
Research Laboratories
Environmental Science Services Administration
United States Department of Commerce

For the
Division of Reactor Development and Technology
United States Atomic Energy Commission

July 1968

U. S. ATOMIC ENERGY COMMISSION Office of Information Services

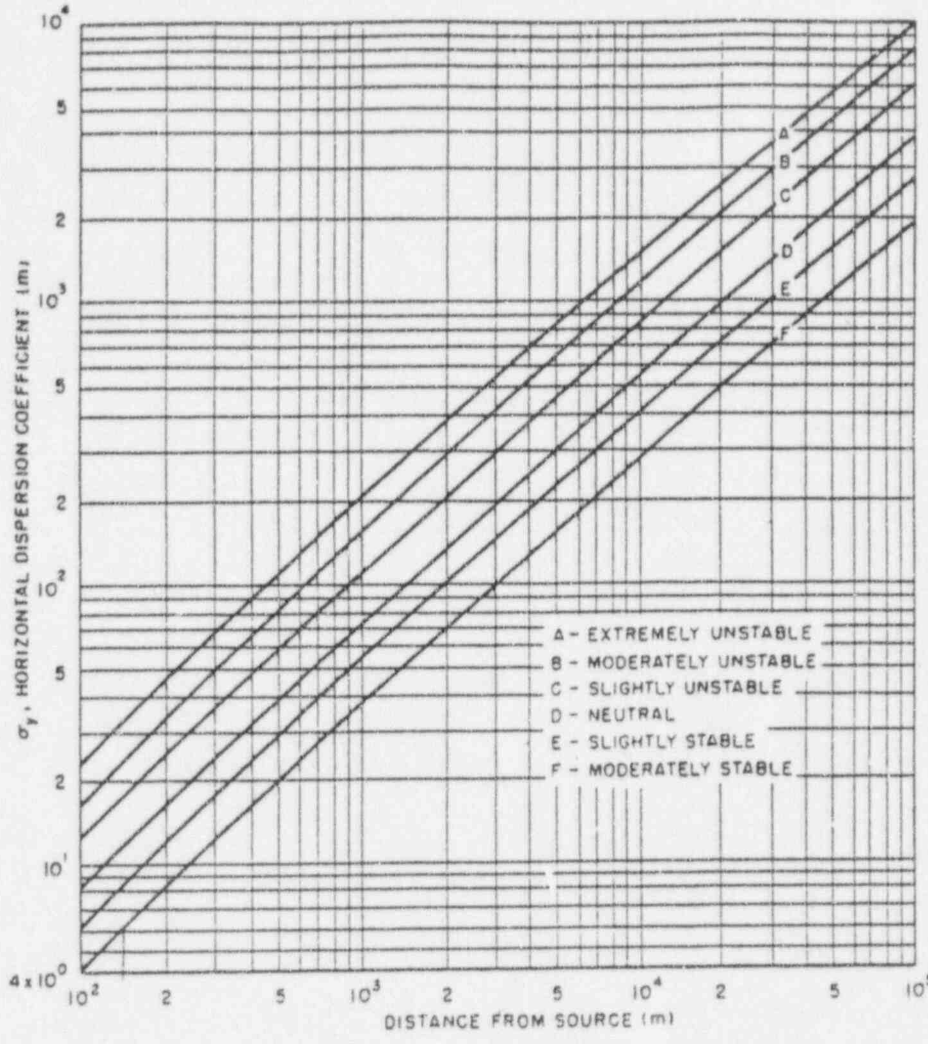


Fig. 3.10—Lateral diffusion, σ_y, vs. downwind distance from source for Pasquill's turbulence types.

and Bryant (1964) have also employed these values of σ in plume diffusion analyses.

Sections 4-4.3 and 4-4.4 of Chap. 4 indicate that Pasquill's curves fit the experimental data collected since the Prairie Grass experiments quite well. Furthermore the experimental data discussed in these sections demonstrate that the standard deviation of the horizontal wind direction, σ_θ, for a short averaging time and for the sampling times used in these experiments (10 min to 60 min) can be related empirically to the measured values of plume width or to normalized average concentration or exposure from continuous sources. On the basis of these data, Pasquill's stability categories can be relabeled approximately in terms of measured values of σ_θ as follows:

Pasquill stability categories	σ_θ
A, extremely unstable	25.0°
B, moderately unstable	20.0°
C, slightly unstable	15.0°
D, neutral	10.0°
E, slightly stable	5.0°
F, moderately stable	2.5°

Pasquill's method of estimating diffusion is well suited to field use because a simple recording wind vane and anemometer erected at a proposed site can, when used with the wind-direction range theory (Chap. 2, Sec. 2-6.2.3),

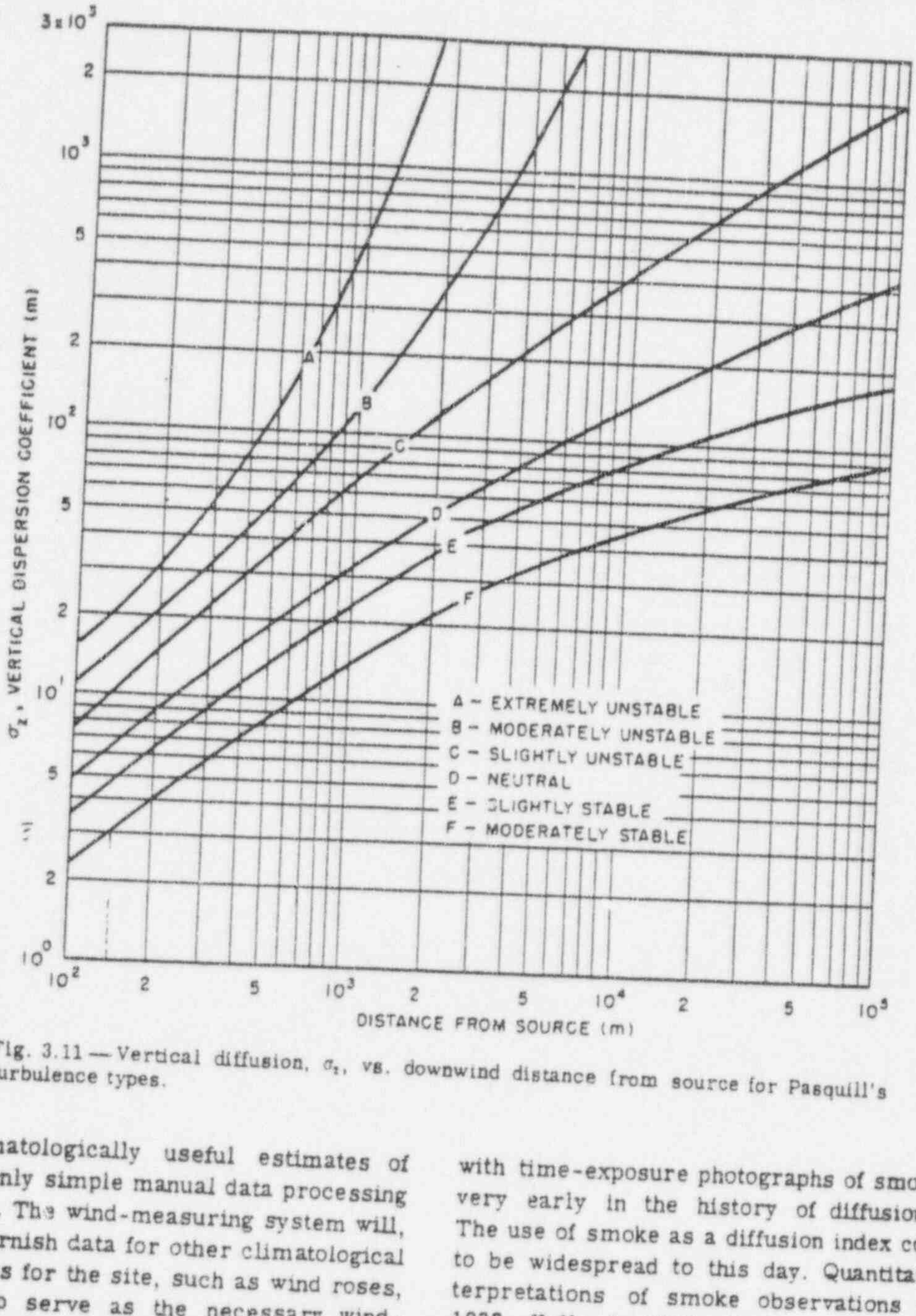


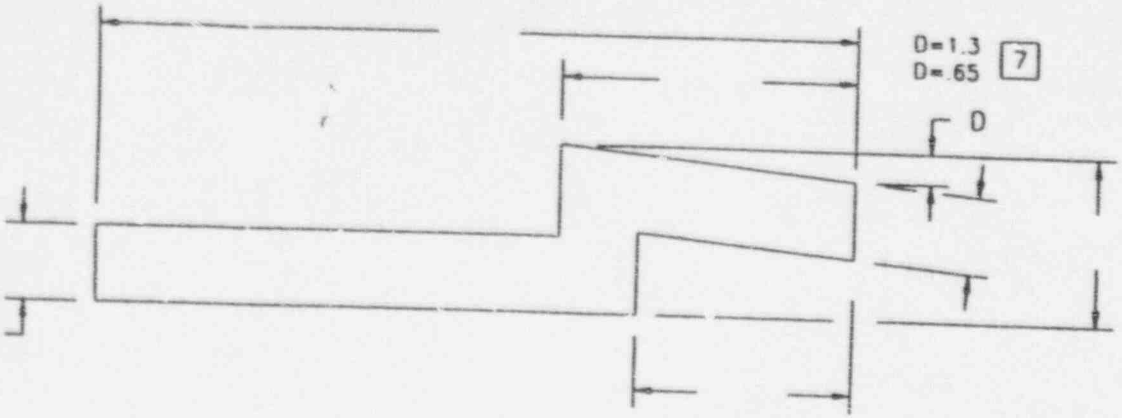
Fig. 3.11 — Vertical diffusion, σ_z, vs. downwind distance from source for Pasquill's turbulence types.

furnish climatologically useful estimates of σ_θ rapidly. Only simple manual data processing is necessary. The wind-measuring system will, moreover, furnish data for other climatological wind statistics for the site, such as wind roses, and will also serve as the necessary wind-velocity monitoring equipment for permanent installation when the reactor or other plant is in operation.

3-3.4.2 Quantitative Use of Smoke Observations to Determine Diffusion Coefficients. Visual and photographic observations of smoke plumes and puffs have always appealed to workers in atmospheric diffusion as a useful research tool. Characteristically Richardson (1920) worked with time-exposure photographs of smoke puffs very early in the history of diffusion study. The use of smoke as a diffusion index continues to be widespread to this day. Quantitative interpretations of smoke observations (Sutton, 1932, Holland, 1953, Kellogg, 1956, Frenkiel and Katz, 1956, Gifford, 1957, 1959, Saissac, 1958, Inoue, 1960, and Högström, 1964) have usually exploited Roberts' (1923) opacity theory in which the visible edge of the smoke plume or puff is supposed to represent a constant threshold density of smoke particles along the line of sight.

The total density of smoke particles is obtained, according to the opacity idea, by integration of the concentration-distribution equa-

Attachment VIII
SNC Air Outlet Drawing (proprietary information deleted)



ITEM 7

NOTES:

1.

2.

[3]

[4]

[5]

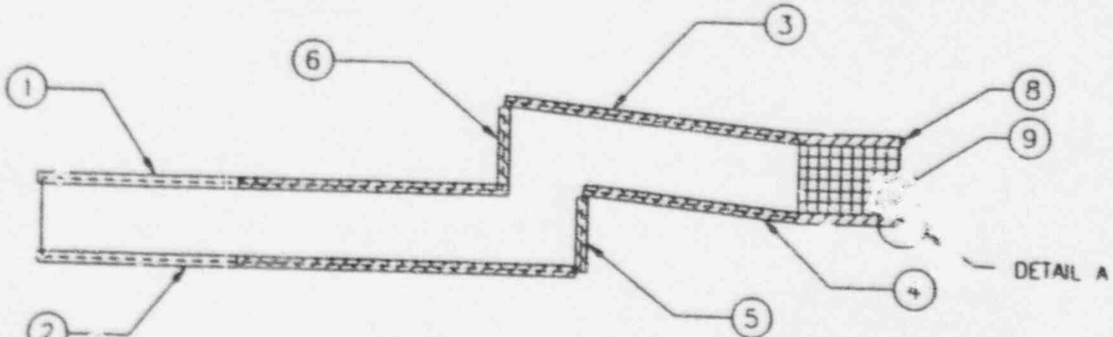
6.

[7] MARK EACH AIR OUTLET ASSEMBLY (P/N PCC-XX-Y)WITH AN SNC APPROVED METAL MARKER: "XX" SHALL BE THE CASK UNIT NUMBER AND "Y" SHALL BE THE UNIQUE AIR OUTLET ASSEMBLY LETTER (A, B, C, OR D) DIMENSION D=1.3 FOR ASSEMBLIES A AND C WITH D=.65 FOR ASSEMBLIES B AND D.

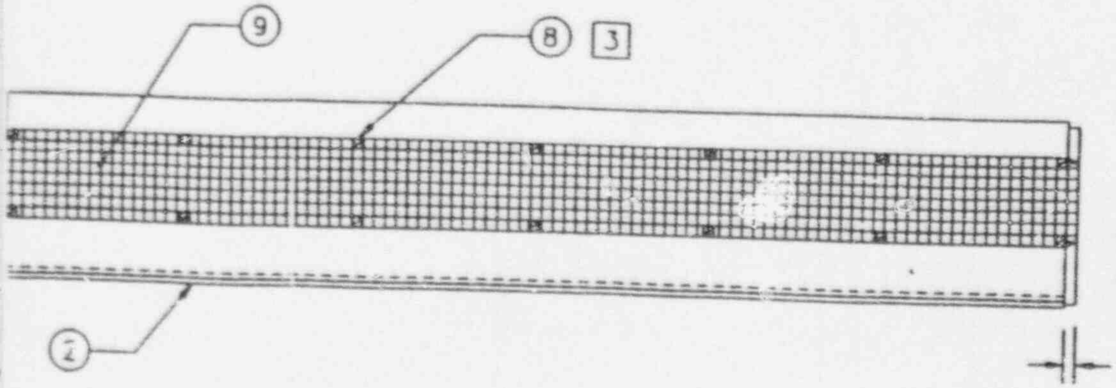
[8]

9.

[10]



SECTION A-A



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REVISION NO.	DESCRIPTION
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UNLESS OTHERWISE SPECIFIED DIMENSIONS ARE IN INCHES TOLERANCES ARE:	WEIGHT	JOB NO. PCC-01	SIERRA NUCLEAR CORP. SNC SCOTTS VALLEY CALIFORNIA
DECIMALS X ±.25 X ±.125 ANGLES ±1°	CAD FILE NAME PCC0401	SIZE	
	DESIGNER	DATE 7/9/96	SAFETY CLASSIFICATION IMPORTANT TO SAFETY

Attachment IX

"PNL-7839, "Performance Testing and Analyses of the VSC-17 Ventilated Concrete Cask," Figure 4-9

U.S. Department
of Energy
Office of Civilian
Radioactive Waste
Management
and
Electric Power
Research Institute

Keywords:
Spent-fuel storage
Thermal-hydraulic models
Heat transfer
Shielding

EPRI TR-100305
Project 3073-1
PNL-7839
UC-85
Final Report
May 1992

Performance Testing and Analyses of the VSC-17 Ventilated Concrete Cask

Prepared by
Pacific Northwest Laboratory
Richland, Washington
and
EG&G Idaho, Idaho National Engineering Laboratory
Idaho Falls, Idaho

REPORT SUMMARY

Performance Testing and Analyses o᷊ the VSC-17 Ventilated Concrete Cask

Full-scale testing of the VSC-17 spent-fuel storage cask confirms that this cask offers a technically sound, practical method for meeting growing utility on-site storage needs. Testing was performed at conditions near the cask's thermal design limits, and the COBRA-SFS code used to predict the cask's thermal performance demonstrated very good agreement with actual test data.

INTEREST CATEGORIES

Light water reactor fuel
Radioactive waste management

KEYWORDS

Spent-fuel storage
Thermal-hydraulic models
Heat transfer
Shielding

BACKGROUND An earlier cooperative program demonstrated dry spent-fuel storage (using intact or consolidated spent fuel) in large metal casks (EPRI reports NP-4887, NP-5128, and NP-5268). Through another cooperative program a horizontal modular storage (NUHOMS) system, which is a horizontal ventilated concrete system, was demonstrated using intact PWR spent fuel (reports NP-6940 and NP-6941). The current report documents the latest storage demonstration program, which tested a concrete ventilated storage cask (VSC) at Idaho National Engineering Laboratory. The cooperative program was supported by DOE, Wisconsin Electric Power Company, EPRI, and Sierra Nuclear Corporation (the cask vendor).

OBJECTIVES To demonstrate the thermal, shielding, and operational performance of the VSC-17 vertical ventilated concrete cask when loaded with consolidated spent nuclear fuel; to demonstrate use of DOE's COBRA-SFS code to model the cask system and accurately predict thermal performance.

APPROACH The project team divided the test program into two general activities: pretest analyses and actual cask testing. Actual testing involved 17 canisters of consolidated PWR assemblies in the VSC-17 cask. After loading, the cask and consolidated fuel were instrumented and tested with three different internal storage environments (nitrogen, helium, and vacuum) and with four cooling vent blockage conditions (all cooling vents open, half the inlets closed, all inlets closed, and all vents closed). The team used the COBRA-SFS code to predict the thermal performance of the cask before the test. Finally, they compared those predictions with actual test data.

RESULTS The test demonstrated that the VSC-17 cask is well suited to store consolidated spent fuel. Its heat transfer performance was good; peak cladding temperatures were under 320°C with a helium backfill, open vents, and a cask heat load of 14.9 kW. Partial blockage of the inlet vents resulted in only a small increase in fuel temperature. Further, the COBRA-SFS code performed very well in predicting both the shapes of the temperature profiles and the actual temperatures. Pretest predictions agreed within 15°C of actual test data. The cask shielding performance met design expectations except for two localized radiation peaks, one on the top of the cask at the edge of the multiassembly sealed basket lid and the other in the outlet vents. Minor modifications to the cask design could remove the peaks.

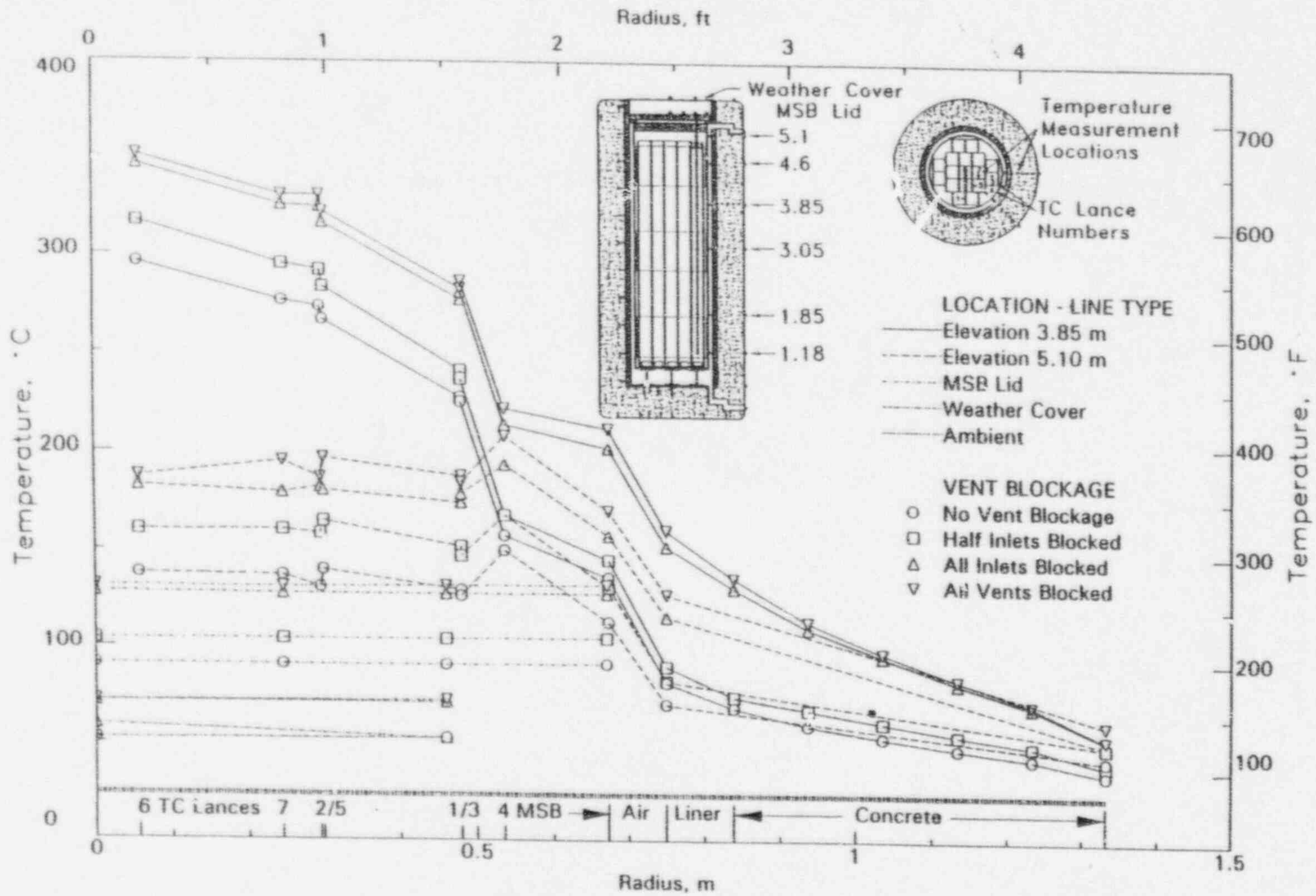

Figure 4-9. Radial Temperature Profiles in Upper Regions of VSC-17 Cask for Four Vent Blockage Conditions