72-17 (50-344)

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

September 23, 1996

CPY-035-96

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 August 20, 1996, Mr. Adam Bless sent a Request for Additional Information to Trojan that contained questions about License Change Application 237 (LCA-237), which requests permission to load spent fuel casks in the Trojan Fuel Building. The responses to the questions are provided in Attachment I of this letter.

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 Harold Chernoff of my staff at 503-556-7480.

Sincerely,

C P Yundt

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

Attachments

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

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September 23, 1996

ATTACHMENTS to CPY-035-96

I. Responses to Request for Additional Information

II. Cask Movement Envelope - "Safe Load Path"

III. Memorandum TOM-022-96, "Dose from Four Failed Fuel Assemblies-ISFSI Accidents" and applicable portions of supporting calculation RPC 93-003.

IV. Impact Limiter Properties

V. Pages 14-16 from calculation PGE01-10.02.03-18

VI. Physical Property Data on **LAST-A-FOAM® FR-3700**, seven pages.

VII. Trojan calculation RPC 96-009

VIII. Pages 4-4, 4-5, and 4-16 of "Topical Report Seismic Analyses of Structures and Equipment for Nuclear Power Plants," BC-TOP-4 Rev 2. 1974, Bechtel Power Corporation.

IX. Sections 2.1.1.2 and 2.1.2.1 of the Trojan Defueled Safety Analysis Report.

Question 1

p.9, section 5.1.2: "The thermal analysis considered the basket in the transfer cask with a helium atmosphere and the basket in the transfer cask with a vacuum." Does this bound all scenarios, including the period that the cask is pressurized to 7.3 psig? How is "short term" defined for the temperature limit of 1058°F? The limits of 743°F and 851°F were reached within what time period in the analysis? Although PNL-4835 states that tests were conducted on dry storage in inert gases at temperatures up to 570°C (1058°F), it recommends a maximum temperature of 380°C (716°F), and suggests that evaluations need to be performed to justify a higher guideline temperature. Based on this document, how is a higher temperature limit justified?

Response

The basket in the transfer cask with a vacuum is bounding because it results in the highest fuel cladding temperature. When the basket is pressurized to 7.3 psig, the fuel cladding temperature is lower because the pressurized helium in the basket transfers heat from the fuel to the basket shell more efficiently than a vacuum.

"Short term" is not quantitatively defined but rather used to establish temperature limits for those events which are considered off-normal or infrequent as listed in Table 4.2-12 of the Trojan Independent Spent Fuel Storage Installation Safety Analysis Report (PGE-1069).

The 743°F(helium) and 851°F(vacuum) temperature limits are the maximum steady state temperatures which would occur if the basket remained in the transfer cask indefinitely. The times to reach these steady state temperature conditions were not calculated. The time to reach these temperatures can be conservatively estimated by assuming an adiabatic heat up as explained in the answer to question #2.

380°C is the temperature associated with long term storage. This long term temperature limit will be satisfied for the Trojan ISFSI. The short term temperature limit applies to off-normal or infrequent events as described above. Temperature limits are imposed to minimize accumulated strain which is time and temperature dependent. Shorter durations allow for higher temperature limits.

Question 2

p.9, section 5.1.2; p.35, section 5.3.3: The application states in section 5.3.3 that "The vacuum drying time is conservatively limited to less than 20 hours to ensure that the fuel cladding strain that is accumulated during the vacuum drying process does not exceed 0.1%. The time is limited as recommended by PNL-6364...." PNL-6364 presents a range of strain values as a function of temperature and time: "assuming 70 MPa cladding stress for 24 h [hours], the predicted cladding strain would be 0.1% at 436°C, 0.23% at 450°C, and 3.3% at 500°C.. With a 0.1% strain limit, a temperature of 450°C would be acceptable for an 8-h working shift." What is the basis of the 20

hours? According to section 5.1.2, the results of a thermal analysis were that cladding temperatures reached 851°F (455°C) in a vacuum. No time was given in this analysis for the cladding to reach this temperature; however, if the limit of 0.1% strain were not exceeded, it would imply that the time limit might be considerably less than 20 hours. In addition to the time limit, will temperature be monitored?

Response

Technical papers on cladding strain indicate that 0.1% cladding strain will not occur for a number of days at temperatures below 760°F(404°C). The 20 hour administrative time limit is imposed to prevent fuel clad temperature from exceeding 760°F(404°C). This time limit was derived by conservatively assuming an adiabatic heat up (i.e., heat transfer only takes place between the fuel assembly and cell wall), heat generation of 1KW/fuel assembly and an initial temperature of 212°F.

The response to Question 1 addresses the time required to reach 851°F.

The fuel cladding temperature will not be monitored. The 20 hour time limit has been conservatively developed to maintain fuel cladding temperature below 760°F(404°C). In addition, heat will be removed from the spent fuel during the vacuum drying process when helium is flushed through the basket between pump downs.

Question 3

p.12, section 5.2.1.1: "The design safety factors, load testing requirements, and administrative controls (i.e. procedures, training, maintenance, inspections) for the fuel handling equipment minimize the possibility of a fuel assembly drop actually occurring." When in the sequence of loading events and how often are the two cranes that will be handling the fuel assemblies and transfer cask load tested? Are both cranes and associated tracks and equipment also inspected for wear and fatigue cracking? We observed numerous crane problems during the Large Component Removal Project (LCRP). Were lessons learned from LCRP incorporated into crane preparations for the ISFSI?

Response

The Fuel Building Crane has been load tested at 125% rated load. Unless the cranes are altered or extensively repaired prior to fuel handling, they will not undergo a rated load test. The cranes, rigging and lifting devices will be tested at maximum anticipated loads using pertinent functions during preoperational testing. Cranes and supports are periodically inspected for degradation and wear in accordance with Trojan Procedure MP-1-20, "Cranes, Hoists, and Winches."

The crane problems during the LCRP involved alignment problems associated with the circular rail and bearing design of the Reactor Building Polar Crane. Although similar problems are not anticipated due to the different design of the Fuel Building Crane, lessons learned from the LCRP will be incorporated into crane preparations for the ISFSI.

Question 4

p.13, section 5.2.1.2: The application states that the analysis for the basket shield lid drop was not performed because guidance of NUREG-0612 was implemented, specifically paragraphs 5.1.6(b)(ii) and 5.1.1(5). These criteria are only part of the recommended guidelines in the NUREG. Section 5.1.1 contains specific guidelines on administrative controls, operator training, and equipment testing. Section 5.1.6 additionally specifies upgrades to cranes and interfacing lift points as part of the "single proof failure" criteria. Which of the guidelines in paragraphs 5.1.1, 5.1.2, and 5.1.6 will be followed? Will dual or redundant slings be used, or will other precautions be taken to preclude dropping the shield lid?

Response

PGE has committed to the NRC to implement the guidelines of Section 5.1.1 of NUREG-0612. As documented in NRC Generic Letter 85-11, the NRC determined that implementation of Sections 5.1.2 thru 5.1.6 was not required to reduce the risks associated with the handling of heavy loads. However, PGE has implemented some of the guidance contained within those sections for the fuel building crane.

Section 5.1.2 provides four separate methods for implementing the guidelines for handling heavy loads in the Spent Fuel Pool area. For the Spent Fuel Pool area PGE has selected to implement the method of Section 5.1.2(2) with two exceptions. The first exception is to Section 5.1.2(2)(a) which recommends mechanical or electrical interlocks be provided to maintain a horizontal separation of 15 feet between heavy loads and the Spent Fuel Pool. Due to the design of the Fuel Building this separation had to be reduced to prevent movement of the hook centerline to no closer than 6 ft. from the Spent Fuel Pool (Refer to response for question 8 for additional information). The second exception is to Section 5.1.2(2)(e) which recommends that an analysis of postulated load drops should be performed in accordance with Appendix A. Since the capacity of the crane significantly exceeds the weight of the shield lid, a shield lid load drop was not considered credible and was therefore not analyzed.

Section 5.1.6 is guidance recommended to satisfy Section 5.1.2(1). Section 5.1.2(1) is another of the four methods acceptable to meet Section 5.1.2. Since PGE selected to implement Section 5.1.2(2) rather than 5.1.2(1), the guidance of Section 5.1.6 is not applicable. For added safety, PGE has implemented the guidance provided in Section 5.1.6(1)(b)(ii) by requiring the lifting slings be rated for twice that specified by Section 5.1.1(5).

Question 5

No mention was made in the application about a possible drop of the crane lower load block into a basket loaded with spent fuel. Assuming that it could be postulated that the load block was dropped onto a basket loaded with fuel prior to the shield lid being installed, what would be the radiological consequences, and would they be bounded by one of the existing analyses?

Response

A "drop" of the crane lower load block onto a loaded basket is not considered a credible accident, therefore an analysis was not performed. However if this event were to occur and potentially crush 24 fuel assemblies (maximum allowed in a single basket) the dose consequences would be 0.018 rem (6 times the dose consequences for the failure of 4 fuel assemblies discussed in Section 5.2.1.4), which is only 1.8% of the EPA PAG.

Question 6

Regarding the crane operator training specified in NUREG-0612, how soon prior to load movement will it occur? Does this training adhere to the guidelines in ANSI B30.2-1976? Will the training include movement along the "safe load path"? Will the "safe load path" be defined in procedure and clearly marked on the floor? Please provide a diagram of the safe load path.

Response

Crane operators at Trojan are given training that complies with ASME B30.2. Initial training is provided prior to being qualified to operate the equipment with requalification training required annually. In addition, specifics of crane operation for heavy load handling for ISFSI will be included in prejob briefings and in the preoperational setup and testing phase. Safe load paths for ISFSI will be incorporated in Trojan Procedure TPP 14-9 prior to ISFSI load handling. Safe load paths will be marked on the floor where feasible and diagrams will be placed in the crane operators cab. Sacrificial floor coverings (herculite or equivalent) may be placed on the floor between the Cask Loading Pit and Cask Wash Pit to contain drips from the SFP and rinse down. The floor coverings may cover the floor markings. However, the load path will be designated on those coverings to the extent feasible.

A copy of the safe load path is provided as Attachment II.

Question 7

p.13, section 5.2.1.3: Please provide a copy of the analysis, calculations, and assumptions which show that the shield lid would plastically deform less than 2" and not grossly fail from dropping the lifting yoke onto a basket loaded with fuel.

Response

Sierra Nuclear Corporation has identified this calculation as containing proprietary information. This calculation is available for ODOE review at the Trojan site. If ODOE review concludes that Trade Secret information is required in order to complete their review, then a Confidentiality Agreement would be required.

Question 8

p.14, section 5.2.1.4: "Mechanical stops and electrical interlocks on the crane used to lift the transfer cask will ensure that sufficient distance from the Spent Fuel Pool is maintained." What is this distance and what is the basis for choosing it? [NUREG-0612: 15 feet in section 5.1.2(2) or 25 feet in section 5.1.2(3)]

Response

The distance of 6 feet is based on allowing crane operations in the Cask Loading Pit which is only 9 feet by 12 feet and located adjacent to the Spent Fuel Pool. The 6 foot separation distance is sufficient because the maximum lifting height of 6 inches prevents a dropped load from tipping and entering the Spent Fuel Pool.

The 25 ft referred to in 5.1.2(3) pertains to heavy loads over the SFP and the distance to be maintained from "hot" fuel. Trojan's fuel does not meet the definition of "hot" fuel. Heavy loads are not intended to be handled over the SFP as part of the ISFSI fuel transfer operations.

Question 9

Does the lifting yoke meet the criteria of a "special lifting device" described in NUREG-0612, section 5.1.6(1)(a)?

Response

The lifting yoke is a special lifting device as described in ANSI N14.6. The lifting yoke does not fully comply with the criteria in NUREG-0612 section 5.1.6 (1) (a), which requires doubled safety factors or load paths for single failure proof handling systems. NUREG-0612 provides the alternative of analyzing consequences of potential drops in lieu of single-failure proof load handling. In accordance with NUREG-0612 Sections 5.1.1(3), 5.1.2(4) and 5.1.5(1)(c), the effects of load drops have been analyzed and shown to satisfy the evaluation criteria of Section 5.1 of NUREG-0612.

Question 10

p.14, section 5.2.1.4: Maximum accelerations for intact fuel are given as 82g vertical and 44g horizontal. Maximum accelerations for the closed basket are given as 124g vertical and 44g horizontal. What are the bases for these limits? Were the calculations used to get these values reviewed by the NRC? What acceptance criteria were used to determine that these are the limiting accelerations? For example, what is the unacceptable consequence of 82g being exceeded? Furthermore, after sustaining forces in excess of 82g, would the fuel have enough integrity to be off loaded, if necessary?

Response

The storage basket is designed to withstand accelerations of 124g vertical and 44g horizontal without resulting in loss of confinement integrity. Based on studies performed by Lawrence Livermore Laboratories which were distributed to irradiated fuel licensees by the NRC, intact spent fuel assemblies can withstand 82g vertical and 63 g horizontal.

Since no credible or design basis accidents were identified which would exceeded the bounding limits of 82g vertical (intact fuel) or 44g horizontal (storage basket), the consequences of exceeding these limits were not analyzed.

Question 11

p.15, section 5.2.1.4: Visual examinations on failed fuel assemblies are mentioned. Did these visual inspections look at interior pins? Were other techniques like sipping or eddy current testing used?

Response

Following Cycle 4 fuel failures, Westinghouse performed sipping and sulo probe rod measurements. Failed fuel was then subjected to high and low magnification examinations. In 1988, ultrasonic and visual exams were performed on suspected failed fuel assemblies by PGE/Westinghouse/Brown Boveri in preparation for reconstitution. Visual and UT inspections identified failed fuel pins in the interior. A Westinghouse fuel repair campaign was conducted in 1989, transferred pins were individually inspected, classified, and failed pins were either left in fuel skeletons or removed to a storage container. Past sipping campaigns, RCS Chemistry trending, and refueling visual examinations have provided Trojan with an extensive data base on the fuel assemblies.

Question 12

p.15, section 5.2.1.4: Do the words "criticality concern" in this section mean keff<[>] 0.95?

Response

A criticality concern arises when K_{eff} exceeds 0.95 ($K_{eff} > 0.95$).

Question 13

p.15, section 5.2.1.4: Please provide a copy of the analysis, calculations, and assumptions which show that the dose at the site boundary would be about 0.003 rem if a transfer cask was dropped prior to the shield and structural lids being welded to the basket. Is there an analogous calculation for dose to personnel in the immediate vicinity? Also, does the term "site boundary" have a consistent meaning throughout this LCA? Is it the Owner Controlled Area, the Industrial Area, or some other boundary?

Response

A copy of the calculation and assumptions for the 0.003 rem dose at the site boundary for a transfer cask drop prior to shield and structural lid closure is provided as Attachment III.

There is no analogous calculation for dose to personnel in the immediate vicinity. Local monitoring of radiation levels and airborne activity would be in effect during fuel handling activities. In the event of an accident personnel would be directed to evacuate the affected area.

The site boundary is described in sections 2.1.1.2 and 2.1.2.1 of the Defueled Safety Analysis Report (copy provided as Attachment IX). The site boundary encloses a portion of the Owner Controlled Area (Exclusion Area). The Industrial Area is contained within the site boundary.

Question 14

The consequences of transfer cask drops are diminished by using impact limiters. In order to verify the calculations of the drop a nt scenarios, please describe these devices. Please provide a description of the dimensi material specifications, and energy absorption characteristics of the impact limiters. Are these considered "important to safety"? Are they purchased commercial grade? What procedure or process is used to verify their energy absorption characteristics?

Response

Impact limiter locations, crush strengths, dimensions and densities are summarized on Attachment IV, and the application methodology is described on Attachment V. Note that energy absorptions characteristics are not anticipated to change, however, dimensions may be revised to accommodate installation and placement. Although calculations indicate that an impact limiter for a side impact at the hoistway is not required, LCA-237 committed to providing one. The dimensions for this impact limiter have not yet been determined.

The impact limiters are considered "important to safety." They may be purchased as commercial grade items and dedicated for use, or as basic components under a suitable Quality Assurance program. Vendor certification of material properties will be required to verify the energy absorption characteristics.

Material properties are described on the manufacturer's data sheets provided as Attachment VI.

Question 15

p.15, section 5.2.1.4.1: Please provide a copy of the analysis, calculations, and assumptions for the accident scenario involving the transfer cask drop into the cask loading pit.

Response

Sierra Nuclear Corporation has identified this calculation as containing proprietary information. This calculation is available for ODOE review at the Trojan site. If ODOE review concludes that Trade Secret information is required in order to complete their review, then a Confidentiality Agreement would be required.

Question 16

p.16, section 5.2.1.4.1: The application states that "the concrete at the bottom of the Cask Loading Pit was determined to be the limiting component on which the height of the impact limiter is based." What is that height?

Response

The design height of the impact limiter to be used in the Cask Loading Pit is 34 inches.

Question 17

p.16, section 5.2.1.4.1: If the concrete floor were breached, would water be lost from the pool? Are there any circumstances in which the gate would be open while the cask was being moved into or out of the loading pit? What would be the recovery actions?

Response

An impact limiter will be installed in the bottom of the Cask Loading Pit to preclude damage to the structure and the fuel during a postulated cask drop accident. It is not anticipated for the gate to be open during transfer cask movements. However, breach of the Cask Loading Pit liner, discussed in section 5.2.3 of the License Change Application, addresses the scenario where the gate is damaged, which would result in conditions similar to an open gate.

Question 18

p.16, section 5.2.1.4.2: The analysis assumes a drop from 93'6". What physical controls prevent a higher starting point?

Response

There are no physical controls intended to be used to limit height of the transfer cask above the deck. Administrative controls will be used. Flagging or tape or equivalent may be attached to the load handling equipment to assist the crane operator in maintaining lift heights. Personnel on or near the floor will verify clearance from obstructions and the height of the load.

Question 19

p.17, section 5.2.1.4.3: Please provide a copy of the analysis, calculations, and assumptions for the accident scenario involving the transfer cask drop into Fuel Building Hoist way. Include the determination of the resulting decelerations in both the vertical and horizontal directions. Are the retaining clips used to secure the boral plates designed to remain secure during this drop?

Response

Sierra Nuclear Corporation has identified this calculation as containing proprietary information. This calculation is available for ODOE review at the Trojan site. If ODOE review concludes that Trade Secret information is required in order to complete their review, then a Confidentiality Agreement would be required.

Question 20

p.20, section 5.2.1.8: Please provide a copy of the analysis, calculations, and assumptions for the accident scenario involving the basket drop into the concrete cask. Include a copy of the methodology of EPRI Report NP-7551.

Response

Sierra Nuclear Corporation has identified this calculation as containing proprietary information. This calculation is available for ODOE review at the Trojan site. If ODOE review concludes that Trade Secret information is required in order to complete their review, then a Confidentiality Agreement would be required.

Question 21

p.25, section 5.2.3.1: The application states that water is available from various sources in the event of a breach or tear in the Cask Loading Pit liner. The Trojan DSAR describes several water sources, including Demin Water, Primary Makeup, Service Water, and the Fire Main. However, these systems are not subject to traditional Technical Specification operability requirements, and could be out of service. What steps will PGE take to ensure that makeup is available on a timely basis?

Response

Trojan Procedure, TPP 30-1, "Nuclear Division Defueled Requirements and Defueled Systems List," describes the status and programs for maintenance of certain systems for the defueled plant. Included in TPP 30-1 is the Trojan Defueled Systems List, which lists the plant systems and their status. For those systems identified as Important to Safe Storage of Irradiated Fuel, which includes those systems identified as makeup sources in the DSAR, administrative controls are imposed.

Question 22

p.26, section 5.2.3.2: Please provide a copy of the analysis, calculations, and assumptions for the accident scenario involving the crane mishandling operation for a horizontal impact.

Response

The crane mishandling operation event is presented in the ISFSI Safety Analysis Report, Section 8.1.1.1.

Question 23

p.28, section 5.2.4.2: How fast do the transfer cask bottom doors move? Do they fail open, closed, or as-is?

Response

The Transfer Cask bottom doors open and close in approximately 20-30 seconds. These doors use a three position (open, closed, neutral) valve with a spring return to neutral position. The Transfer Cask bottom doors will fail as-is on loss of hydraulics.

Question 24

p.30, section 5.2.5.1: How does PGE propose to eliminate the possibility of an occurrence similar to the one reported in NRC Information Notice 96-34 where a hydrogen gas ignition occurred during the welding of the shield lid on a spent fuel storage cask at the Point Beach Nuclear Plant? Do any of the ISFSI components involved in the spent fuel loading have a zinc-based coating similar to the ones mentioned in the Information Notice? Is there any concern of degradation of zircaloy cladding due to the presence of hydrogen?

Response

Currently, Carbo-Zinc 11, a zinc-based coating, is applied to the Basket internals and to the exposed carbon steel components on the Concrete Cask. Sierra Nuclear Corporation is re-evaluating the use of this coating in light of the chemical reaction problems at Point Beach. The Trojan design may be modified based on the outcome of this evaluation. Additional information will be provided in a later submittal.

Question 25

p.30, section 5.2.5.1: The application states that "about 37 hours of heat-up are required for the water in the basket to reach boiling." How much time is required for welding and drying, assuming these evolutions are completed without problems? What are the consequences in the event that boiling is reached? Will the water temperature be monitored if there is a delay in welding the shield or structural lids or in draining and vacuum drying the basket?

Response

The time estimated for welding and drying is about 16.5 hours. This estimate is based on data from other facilities that have performed similar operations.

The consequences of the water boiling have not been analyzed because corrective actions will be implemented to prevent boiling of the water if the welding and drying process is not completed within 37 hours.

The elapsed time for the welding and drying process will be monitored for each basket. A linear heat up rate is estimated for the water, hence, the temperature of the water is directly proportional to the elapsed time and direct measurement of the water temperature is not required.

Question 26

p.32, section 5.2.5.2.2: Where does the 12 $\mu Ci/cm^2$ come from, and how does this translate into a dose at the site boundary?

Response

A copy of the calculation is provided as Attachment VII.

Question 27

p.33, section 5.2.6.1.3: The analysis for an earthquake determined the kinetic energy input to the transfer cask using the "square root of the sum of the squares" combination of the peak horizontal and peak vertical ground velocities. Why was this method used here but the 100/40/40 method used in the SAR for the ISFSI?

Response

The reference used as a bases for the calculations specified (see Attachment VIII) the "square root of the sum of the squares" combination of peak horizontal and vertical ground velocities.

Question 28

Throughout the application, the use of procedures are a key method of assuring safety. Since it is appears that many of these procedures are not yet drafted, how will PGE ensure that all of the procedures for which credit is taken in the application are satisfactory and ready for use before fuel moves are started?

Response

PGE uses an administrative procedure, TPP 18-2, "License Change Revision," which is a process used for previous Trojan License changes. This process includes reviewing the requirements in new or revised licensing documents to identify existing procedures that require changes and any new procedures which are needed.

The procedures for preparation, review, and approval of revised and new procedures will provide assurance that the Trojan LCA-237 requirements are met.

Question 29

Will seismic monitors and audible control room alarms be functional? Will there be communication between the control room and the fuel moving crew during loading operations?

Response

Yes, seismic monitors and control room audible alarms will be maintained in accordance with the Trojan Nuclear Plant DSAR. Communications will be established between fuel handlers and the control room. The on shift supervisor will be able to communicate with the certified fuel handler (floor supervisor) located at the spent fuel pool via independent circuits.

Question 30

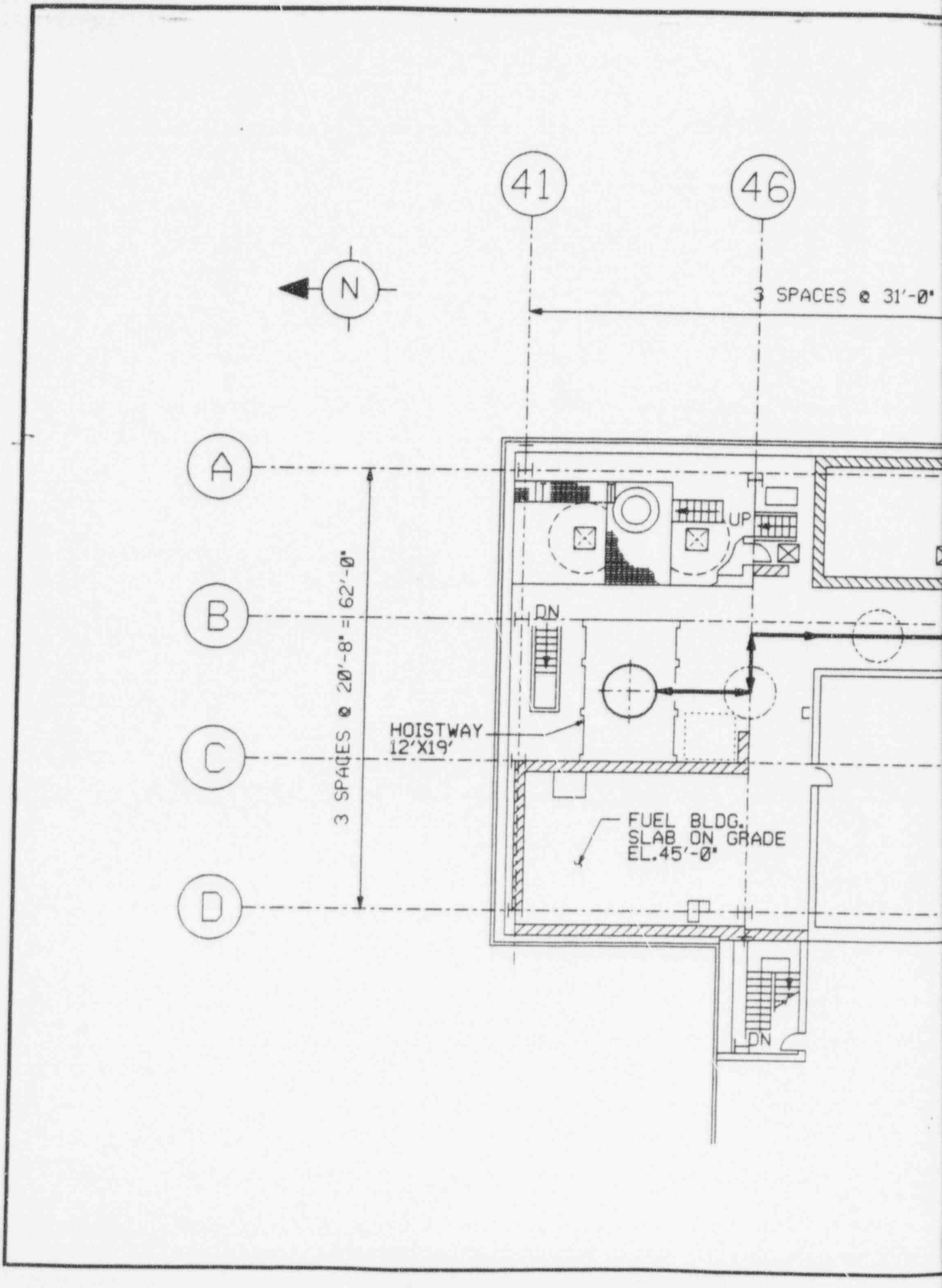
Do procedures preclude fuel handling while the transfer casks is being moved?

Response

Currently they do not. The procedures review process has just begun and will be updated to include operational restrictions as more operating design information becomes available. The Fuel Handling Procedures will administratively control fuel movement and cask movements such that during cask movements near the Spent Fuel Pool, no fuel will be moved except to mitigate hazardous situations.

ATTACHMENT II

Cask Movement Envelope - "Safe Load Path"


41
46
N
3 SPACES @ 31'-0"
A
B
C
D
3 SPACES @ 20'-8" = 62'-0"
UP
DN
HOISTWAY
12'X19'
FUEL BLDG.
SLAB ON GRADE
EL.45'-0"
DN

ANSTEC
APERTURE
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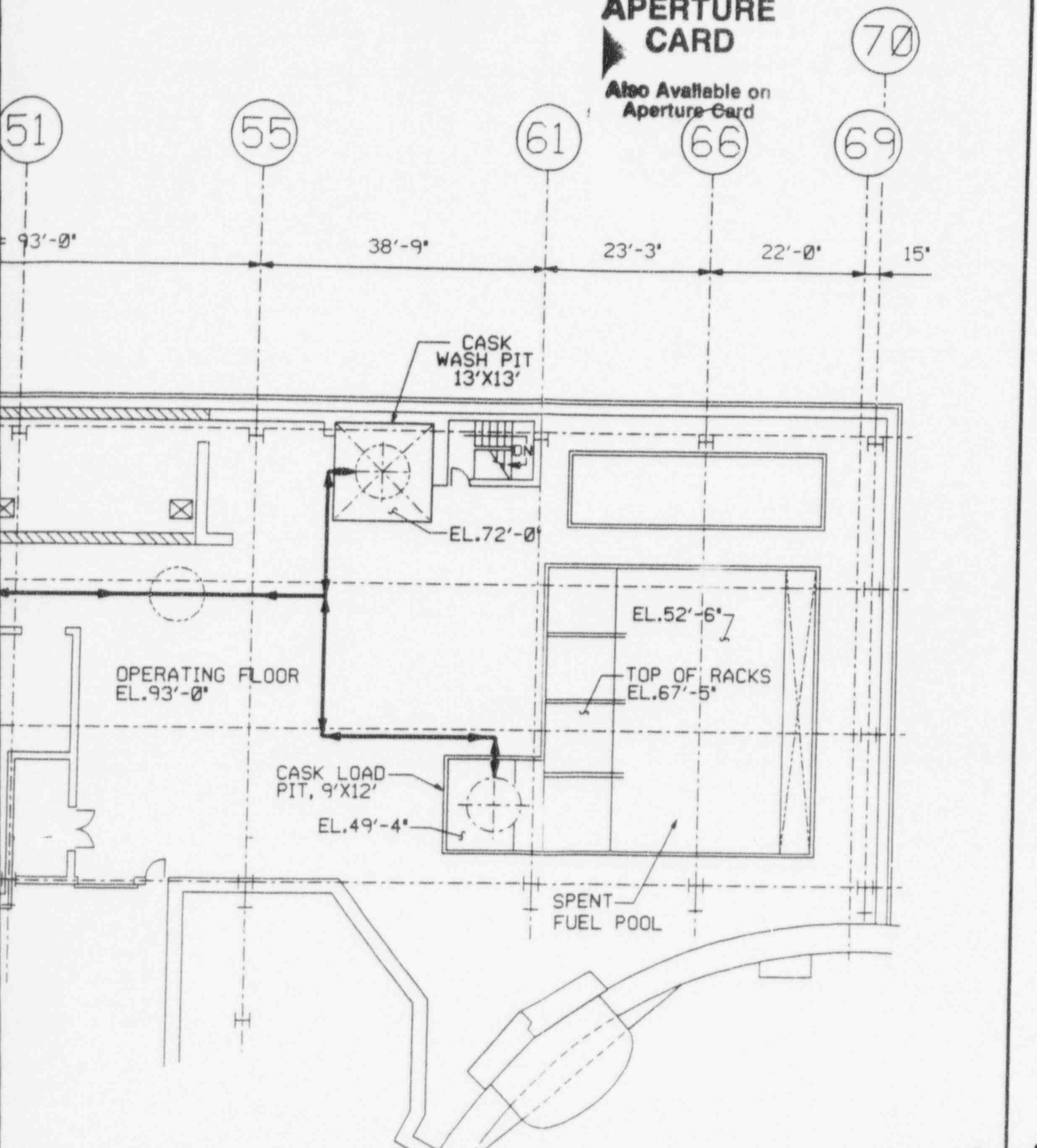
Also Available on
Aperture Card



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ATTACHMENT II
CPY-035-96

ATTACHMENT III

Memorandum TOM-022-96 "Dose from Four Failed Fuel Assemblies - ISFSI Accidents" and applicable portions of supporting calculation RPC 93-003

ATTACHMENT III
CPY-035-96

Trojan Nuclear Plant
Radiation Protection

TOM-022-96

To: Harold Chernoff

From: Tom Meek T

Date: May 23, 1996

Subject: Dose from Four Failed Fuel Assemblies-ISFSI Accidents

I have reviewed previous fuel handling calculations completed at Trojan and have concluded that the results of the previously completed/approved calculations can be used to determine the projected dose from ISFSI handling accidents involving failed fuel. RPC 93-003 'Dose at Existing and Proposed Site Boundary from Fuel Handling Accident after 6 months decay' can be used to determine the dose at the site boundary due to the projected release from four (4) failed assemblies in one MPC basket.

Method From RPC 93-003 the wholebody dose at the site boundary due to the release of Kr-85 is 0.3268 mrem (page 12 of 19).

To correct for the assumption in RPC 93-003 that only 30% of the Kr-85 activity from the fuel assembly is contained I divided the gap dose by 0.3. The dose at the site boundary from the release of all the activity in an assembly is 1.09 mrem.

I next corrected the above dose for the decay of Kr-85 (10.74 year half life) to 5 years post shutdown (t=4.5 years). The decay correction is 0.75 which results in a dose from one assembly with 5 years decay of 0.818 mrem.

Finally I multiplied the dose due to the release of all the Kr-85 activity from one assembly by four (the maximum number of failed assemblies in one basket).

Results The projected dose at the site boundary from the release of all the Kr-85 activity in four fuel assembles with 5 years decay is **3.27 mrem**.

References RPC 93-003 'Dose at existing and proposed site boundary from fuel handling accident after 6 months decay'

Radiological Health Handbook, Bureau of Radiological Health, Revised Edition January 1970.

62-712 11-79

8-40-207

GENERAL COMPUTATION SHEET

Sheet 4 of 10 Sheets
Calc. / Job / File No. RPC 93-003
Project/Job

Subject: Doses at Existing & Proposed Site Boundary From Fuel Handling Accident

By R. M. Sherman Date 2-10-93 Chk. By L E Parks Date 2/10/93 Orig ☑ Rev. ☐

OBJECTIVE

The objective of this calculation is to determine the beta+gamma skin dose, gamma whole body dose and thyroid inhalation dose at the site boundary, and at the intake duct of the control room ventilation system. The calculation includes a 6 month decay time to simulate the decay time from end of power operations to possession only license. The source is a single fuel assembly involved in a fuel handling accident in the Spent Fuel Pool. The results of this calculation are intended to be used to justify (1) the elimination of offsite emergency planning efforts if the dose is less than 1 rem (per EPA 400, "Manual of Protective Action Guides and Protective Actions for Nuclear Incidents"); and (2) to provide the basis for locating the protected area barrier.

In addition, this calculation will estimate the gamma whole body dose at the site boundary, after 6 months decay time, of a shipping cask accident in the spent fuel pool.

RESULTS

The results are provided in Table 1 (attached) and explained below.

Part 1: Existing Site Boundary Dose

Two cases for the site boundary dose were considered using different source terms. Case 1 used the gap inventory shown in FSAR Table 15.0-5 as the best estimate of the source and resulting in a dose of 0.182 mrem. Case 2 used the Reg Guide 1.25 assumptions of 10% of the core inventory of Iodine and noble gases (except Kr-85) and 30% of the core Kr-85 inventory. The FSAR Table 15.0-5 core inventory was used.

Case 2 produced the highest doses compared to Case 1. The Case 2 results were then multiplied by the peaking factor of 1.65 and the results are reported in Table 1.

Therefore, the gamma whole body dose at the site boundary after 6 months decay from the fuel assembly with the highest power density, involved in a fuel handling accident in the spent fuel pool is 0.540 mrem.

Part 2: Dose at Control Room Ventilation System Intake

The dose at the intake of CB-1 was calculated using the same source term and decay time. However, a Chi/Q from an existing calculation (TNP-88-38) was used for the CB-1 intake case. I-129 was included in this calculation as it is a long lived isotope. The dose from the I-129 source is small compared to the thyroid dose from the I-131 that exists at the 6 mo decay time, and therefore are not shown on Table 1 (see page A12). LEP 2/10/93

The results from this calculation are shown on Table 1.

See Appendix C for Revision 1 changes LEP 3/8/93

PGE 1787 (Nov 89)

Subject Doses at Existing / Proposed Site Boundary from Fuel Handling Accident
By R. N. Shannon Date 2-10-93 Chk. By L.E. Porter Date 2/10/93 Orig ☑ Rev. ☐

Part 3: Shipping Cask Accident

The dose from a shipping cask accident was calculated for: A) the existing site boundary and B) the CB-1 intake location. The calculation assumes that 193 fuel assemblies from the recent defueling are damaged and that the radial peaking factor is not included as the all 193 fuel assemblies from the recent core defueling are involved in the accident.

CONCLUSION

This calculation concludes that the beta+gamma skin dose, gamma whole body dose and thyroid dose are fractions of the 10 CFR 100 limits and Protective Action Guides for the existing site boundary and at the CB-1 intake. In addition, the doses are less than the criteria for maintaining an emergency plan.

The gamma whole body dose form a shipping cask accident involving the 193 fuel assemblies from cycle 14 is 63.11 mrem at the existing site boundary and 87.29 mrem at the CB-1 intake.

ACCEPTANCE CRITERIA

EPA 400, Table 2-1 Protective Action Guides for the Early Phase of a Nuclear Incident includes the acceptance criteria of 1 rem.

METHOD

RNS 2/10/93

The method used in this calculation was to use the computer code DOSES which is a documented computer code per NDP 200-5 and is an accepted computer code used in Radiation Protection at Trojan. The code inputs were determined from FSAR data (see references) and included a breathing rate, Chi/Q, decay time and nuclide activity.

ASSUMPTIONS

1. Breathing Rate = 3.47 E-4 M^3/Sec. (FSAR Table 15.0-7).

2. Chi/Q = 4.26 E-4 Sec./[Meter*Meter*Meter] (FSAR Table 15.0-9 for 0-2 Hr release at site boundary).

3. Decay Time = 6 Mo. = 180 Day * 24 Hr/Day = 4320 Hr

4. Assume the fuel assembly involved in the accident had the highest peaking factor in the core. Therefore, use a peaking factor of 1.65 (per Reg. Guide 1.25).

5. Chi/Q = 5.89 E-4 Sec./[Meter*Meter*Meter] (TNP 88-38) for dose at intake of CB-1 case.

See App. C for Rev 1 changes KR 3/8/93

DESIGN INPUTS

1. FSAR Table 15.0-5 Core and Gap inventories were used in this calculation.

REFERENCES

1. FSAR Chapter 15.
2. Reg 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 BWR's and PWR's," March 23, 1972.
3. EPA 400, Manual Of Protective Action Guides And Protective Actions For Nuclear Incidents.

Subject Doses at Existing / Proposed Site Boundary from Fuel Handling Accident

By R. M. Shannon Date 2-10-93 Chk. By L E Rocha Date 2/10/93 Orig ☑ Rev. –

CALCULATION

Part 1: Existing Site Boundary Dose

Source Term Determination:

Case 1 FSAR Table 15.0-5 Gap Inventory

- Use FSAR Table 15.0-5 Gap inventory for 193 assemblies
- Divide Table 15.0-5 values by 193 to obtain Ci in gas gap of one fuel assembly

Case 2 Reg. Guide 1.25 Assumptions (10% core inventory Iodine and Noble Gas except Kr-85 which is 30%)

- Use FSAR Table 15.0-5 Core Inventory values for 193 assemblies
- Divide Table 15.0-5 values by 193 to obtain Ci in the fuel of one fuel assembly
- Assume all gap activity consists of 10% of fuel inventory of Iodine and noble gases except Kr-85
- Assume Kr-85 in gap is 30% of fuel inventory

	CASE 1		Case 2	
Isotope	Ci in Gap (193 ASMBLY)	Ci in Gap (1 ASMBLY)	Ci in Core (193 ASMBLY)	R.G. 1.25 (1 ASMBLY)
I-131	7.24 E5	3.75 E3	8.80 E7	4.56 E4
I-132	1.20 E5	6.22 E5	13.4 E7	6.94 E4
I-133	5.34 E5	2.77 E3	19.7 E7	1.02 E5
I-134	1.28 E5	6.63 E2	23.1 E7	1.19 E5
I-135	2.75 E5	1.42 E3	17.9 E7	9.27 E4
Xe-131m	0.0669 E5	3.47 E1	0.0668 E7	3.46 E2
Xe-133	13.6 E5	7.05 E3	20.3 E7	1.05 E5
Xe-133M	0.225 E5	1.17 E2	0.516 E7	2.67 E3
Xe-135	1.00 E5	5.18 E2	5.55 E7	2.88 E4
Xe-135M	0.165 E5	8.55 E1	5.46 E7	2.83 E4
Xe-138	0.566 E5	2.93 E2	17.9 E7	9.27 E4
Kr-83m	0.135 E5	6.99 E1	1.64 E7	8.50 E3
Kr-85	1.67 E5	8.65 E2	0.0999 E7	1.55 E3
Kr-85m	0.489 E5	2.53 E2	3.95 E7	2.05 E4
Kr-87	0.507 E5	2.63 E2	7.59 E7	3.93 E4
Kr-88	1.07 E5	5.54 E2	10.8 E7	5.60 E4
Kr-89	0.192 E5	9.94 E1	14.0 E7	7.25 E4

See Appendix C for Revision V Changes 3/8/93

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GENERAL COMPUTATION SHEET

Sheet 8 of 10 Sheets

☑ Calc. / ☐ Job / ☐ File } No. RPC 93-003

Project/Job

Subject Doses at Existing & Proposed Site Boundary from Fuel Handling Accident

By R. M. Shannon Date 2-10-93 Chk. By L E Parker Date 2/10/93 Orig ☑ Rev. ☐

CALCULATION continued

Fuel Handling Accident (one fuel assembly)

The Doses code output (attached) result:

	Case #1	Case #2
Skin Dose	1.539 E+1 mrem	2.752 E+1 mrem
Whole Body Dose	1.824 E+1 mrem	3.272 E-1 mrem
Inhalation Thyroid Dose	1.495 E-1 mrem	1.818 E 0 mrem
Total Thyroid Dose	3.319 E-1 mrem	2.145 E 0 mrem

The Case #2 dose estimates from above are multiplied by the radial peaking factor of 1.65 (per Reg. Guide 1.25) to obtain the final dose estimates.

	Final Existing Site Boundary Dose Estimates
Skin Dose	45.4 mrem
Whole Body Dose	0.54 mrem
Inhalation Thyroid Dose	3.00 mrem
Total Thyroid Dose	3.54 mrem

****

Part 2: Dose at the Intake of CB-1

The source term for Part 2 of the calculation is the same as was used in Part 1 (ie. Reg. Guide 1.25 Ci inventories). However, the Chi/Q from TNP 88-38 was used for the new receptor location. The Doses code results are shown below:

Skin Dose	38.12 mrem
Whole Body Dose	0.4523 mrem
Inhalation Thyroid Dose	2.513 mrem
Total Thyroid Dose	2.965 mrem

See Appendix C for Revision 1 changes LEP 3/8/93

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GENERAL COMPUTATION SHEET

Sheet 9 of 10 Sheets

Calc. / Job / File No. RPC 93-003

Project/Job

Subject DOSES AT EXISTING & PROPOSED SITE BOUNDARY FROM FUEL HANDLING ACCIDENT

By R. M. Shannon Date 2-10-93 Chk. By L E Rocha Date 2/10/93 Orig ☑ Rev. ☐

The above dose estimates from above are multiplied by the radial peaking factor of 1.65 (per Reg. Guide 1.25) to obtain the final dose estimates.

	Final CB-1 Intake Dose Estimates
Skin Dose	62.9 mrem
Whole Body Dose	0.746 mrem
Inhalation Thyroid Dose	4.15 mrem
Total Thyroid Dose	4.89 mrem

****

Part 3: Shipping cask accident (using existing site boundary)

Assume 193 fuel assemblies from cycle 14 (last power cycle) are damaged during a shipping accident.

Existing Site boundary Case

Dose = 193 assemblies * Whole body dose
Dose = 193 assemblies * 3.272 E-1 mrem
Dose = 63.11 mrem

CB-1 Intake case

Dose = 193 assemblies * Whole body dose
Dose = 193 assemblies * 4.523 E-1 mrem
Dose = 87.29 mrem

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GENERAL COMPUTATION SHEET

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[X] Calc. / [ ] Job / [ ] File No. RAC-93-003, Rev. 1

Project/Job TROJAN

Subject DOSES FROM A FUEL HANDLING ACCIDENT AFTER 6 MONTHS' DECAY

By PB Chaddy Date 3/8/93 Chk. By L E Rocha Date 3/8/93 Orig [ ] Rev. [X]

APPENDIX C

THIS APPENDIX CONTAINS REV. 1 CALCULATIONS TO CORRECT ERRORS IN IODINE DOSE PROJECTIONS.

ATTACHMENT VI

Physical Property Data on Last-a-Foam® FR-3700, seven pages

$$Q = \left[\frac{d}{(2d-1)}\right] W \left[1 + \sqrt{1 + \left[\frac{(2d-1)}{d^2}\right]\left(\frac{2\pi V}{gT}\right)^2}\right]$$

where:

- d - allowable ductility ratio for the controlling failure mechanism.
- W - static weight of the missile
- T - natural period of the structure
- g - gravity acceleration
- V - impact velocity

This formula is recommended by NRC in NUREG 0800 for missile protection barrier design. It accounts for the dynamic nature of the impacted structure and for the inelastic energy absorption of the structure. This equivalent static force Q can be compared directly to the structure load capacity.

$$W(H-h+\Delta)=\left[f_{cr}\,A+W_w\right]\Delta\,, \qquad \text{thus,} \quad \Delta = W\frac{(H-h)}{f_{cr}\,A-W+W_w}$$

where:
- W - weight of the Transfer Cask
- H - drop total height
- h - ILP height
- f_{cr} - crushing strength of the foam
- A - crushed area
- Δ - depth of crush
- W_w - weight of water (if any) displaced by the Transfer Cask

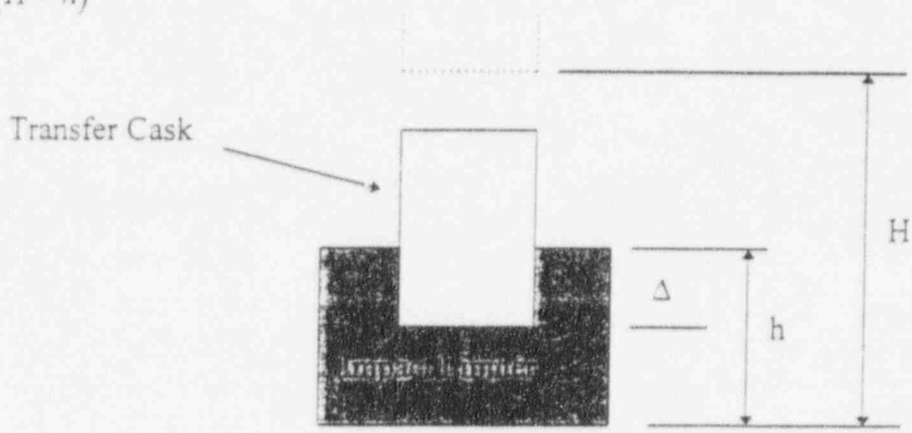
Deceleration is calculated by balancing the forces:

$$\left(\frac{W}{g}\right)a = f_{cr}\,A+W_w\,, \quad \text{thus,} \quad a = g\,\frac{\left(f_{cr}\,A+W_w\right)}{W}$$

where:
- g - gravity acceleration
- a - deceleration of Transfer Cask

The impact velocity is:

$$V_0=\sqrt{2\,g\,(H-h)}$$



b) Methodology for Hard Surface Impacts

For the case where no ILP is used, the Williamson-Alvy formula [Ref. 14] is used to develop the equivalent static force.

where

R_m = plastic resistance force
Δ_y = yield displacement at the location of impact load
μ = ductility of impacted slab
R_o = dead and live load
δ_y = yield displacement at location of concentrated dead and live load or
= 1/2 maximum yield displacement for distributed live and dead load
δ_0 = dead and live load displacement at the location of concentrated load or
= 1/2 maximum dead and live load displacement for distributed live and dead load

For the postulated cask drop into the Cask Loading Pit BC-TOP-9A methodology is used to estimate the velocity at which the cask strikes the submerged impact limiter at the bottom of the pit.

4.4 Estimating Drop Load

Two methodologies are used to determine impact loads. Impact loads are required when checking shear capacities in beams and slabs and when designing an impact limiter.

The first methodology is appropriate for an impact limiter (ILP), and is based on the absorption of the drop energy by ILP foam. The second is for a hard impact without ILP, and uses the equivalent static force. In both cases the inelastic response of the structure is anticipated.

a) Methodology with ILP

This analysis is based on the assumption that the ILP foam crushes under the cask and absorbs the drop energy. The foam dynamic strength multiplied by the crushed area is the drop load to the structure. The force and energy balances allow calculation of the crush depth.

In parallel with the above, the structure load capacity is calculated. The structure capacity must be found for both flexure and shear to determine which controls the design. Based on this result, the allowable ductility ratio is selected.

The required pad height is calculated. Per vendor information, the foam provides constant dynamic strength as long as it is crushed less than 50% of the total thickness. Therefore, the crush depth must be kept under 50% of the ILP height.

Crush depth is determined by balancing the drop energy and work produced by the foam.

ATTACHMENT V

Pages 14-16 from Calculation PGE 01-10.02.03-18

Attachment IV to CPY-035-96
Impact Limiter Properties

Description	Drop Heig.	Impact Limiter Properties				
		P_{cr} (psi)	h (in)	Length (in.)	Width (in.)	Density (lbs/ft^3)
Drop Into Cask Loading Pit	From Elevation 93' 8"	865	34	132	96	17
Tipover At Cask Loading Pit	NA	1454	24	216	18	24
Drop Into Cask Wash Pit	From Elevation 93' 6"	115	125	100	95	5
Tipover At Cask Wash Pit	NA	454	30	108	18	12
Drop Onto Cask Wash Pit Wall	3" above Elevation 93'	NA	NA	NA	NA	NA
Drop onto Shear Wall Between Cask Wash Pit and Hoistway	3" above Elevation 93'	NA	NA	NA	NA	NA
Drop Onto Floor Slab Between Load and Wash Pits	5" above Elevation 93'	48	3.5	108	108	4
Drop Into Hoistway	From Elevation 93' 6"	675	65	216	216	14.7

ATTACHMENT IV

Impact Limiter Properties

SUMMARY OF DOSE RESULTS
SFP FUEL HANDLING ACCIDENT

MAXIMUM SITE BOUNDARY DOSES (mrem)

	WHOLE BODY	SKIN	TOTAL THYROID
All Isotopes Except I-129:	5.398E-1	4.554E+1	5.698E-1
I-129 Dose Contribution :	8.216E-6	2.221E-5	4.901E-2
TOTAL (mrem):	5.398E-1	4.554E+1	6.188E-1

DOSES AT CB-1 INTAKE (MREM)

	WHOLE BODY	SKIN	TOTAL THYROID
All Isotopes Except I-129:	6.512E-1	5.493E+1	6.874E-1
I-129 Dose Contribution :	9.911E-6	2.680E-6	5.913E-2
TOTAL (mrem):	6.512E-1	5.493E+1	7.465E-1

DOSES AT CB-2 INTAKE (MREM)

	WHOLE BODY	SKIN	TOTAL THYROID
All Isotopes Except I-129:	1.060E+0	8.942E+1	1.119E+0
I-129 Dose Contribution :	1.613E-5	4.362E-5	9.624E-2
TOTAL (mrem):	1.060E+0	8.942E+1	1.215E+0

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GENERAL COMPUTATION SHEET

Sheet C2 of C17 Sheets

[X] Calc. [ ] Job [ ] File No. RPC-93-003, Rev. 1

Project/Job TROJAN

Subject DOSES FROM A FHA AFTER 6 MONTHS' DECAY

By JB Chadly Date 3/4/93 Chk. By L E Rocha Date 3/8/93 Orig [ ] Rev. [X]

1. + 2., CONT'D.:

I-131 EQUIVALENT ACTIVITY OF I-129 IN PEAK ASSEMBLY:

$$I\text{-}131_{equiv.} = I\text{-}129\ \text{Activity} * \left(\frac{MPC_{131}}{MPC_{129}}\right) = (1.662E-2\ Ci_{129})\left(\frac{9.0E-9}{2.0E-9}\right) = 7.48E-2\ Ci_{131}$$

$$*(0.3)_{Avail.}^{Ava} = 2.24E-2\ Ci_{131}\ \text{for release}$$

NOTE: REV. Ø OF THIS CALCULATION INVERTED THE ABOVE MPC RATIO, THUS UNDERESTIMATING I-131 EQUIVALENT ACTIVITY BY A FACTOR OF $(4.5)^2$, OR 20.25. ALSO, IN THE DOSES RUN USED TO CALCULATE I-129 DOSE CONTRIBUTION, A DECAY TIME OF 6 MONTHS WAS ERRONEOUSLY APPLIED TO THE I-131 EQUIVALENT ACTIVITY. IN FACT, LONG-LIVED I-129 (AND HENCE THE I-131 EQUIVALENT ACTIVITY) WILL NOT CHANGE APPRECIABLY IN 6 MONTHS.
THE SHORT-LIVED ISOTOPE RUNS OF DOSES WERE ALSO REDONE INCORPORATING AN IODINE POOL DF OF 100, IN ACCORDANCE WITH R.G. 1.25 AND THE FSAR.

A TOTAL OF 6 CASES WERE RUN:

1. SITE BOUNDARY X/Q = ~~8.365 E-4 sec/m³~~ PBC 3/4/93 4.26E-4 sec/m³ (FSAR T. 15.0-9)
   ALL ISOTOPES EXCEPT I-129
   6 MONTHS DECAY TIME (4320 HRS)

2. SITE BOUNDARY X/Q = 4.26E-4 sec/m³ (FSAR T. 15.0-9)
   I-131 equivalent FOR I-129 ACTIVITY = ~~7.48 E-2~~ Ci_{131}
   Ø DECAY TIME — 2.24E-2 LR 3/8/93

3. CB-1 INTAKE X/Q = ~~5.89E-4 sec/m³~~ PBC 3/4/93 5.139 E-4 sec/m³ (Ref: TNP-89-27, App. A, p. 6)
   ALL ISOTOPES EXCEPT I-129
   6 MONTHS DECAY TIME (4320 HRS)

4. CB-1 INTAKE X/Q = ~~5.89E-4 sec/m³~~ PBC 3/4/93 5.139 E-4 sec/m³ (Ref: TNP-89-27, App. A, p. 6)
   I-131 equiv. FOR I-129 ACTIVITY = ~~7.48E-2~~ Ci_{131}
   Ø DECAY TIME 2.24E-2 LR 3/8/93

5. CB-2 INTAKE X/Q = 8.365 E-4 sec/m³ (Ref: TNP-89-27, App. A, Pg. 5)
   ALL ISOTOPES EXCEPT I-129
   6 MONTHS DECAY TIME (4320 HRS)

6. CB-2 INTAKE X/Q = 8.365 E-4 sec/m³ (Ref: TNP-89-27, App. A, pg. 5)
   I-131 equiv. FOR I-129 ACTIVITY = ~~7.48E-2~~ Ci_{131}
   Ø DECAY TIME — 2.24E-2 LR 3/8/93

THE RESULTS ARE SUMMARIZED ON THE FOLLOWING PAGE

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GENERAL COMPUTATION SHEET

[X] Calc. [ ] Job [ ] File No. RPC-93-003, Rev. 1

Project/Job TROJAN

Subject DOSES FROM A FUEL HANDLING ACCIDENT AFTER 6 MONTHS' DECAY

By PB CHADLY Date 3/4/93 Chk. By L E Rocke Date 3/8/93 Orig [ ] Rev. [X]

REV. 1 TO RPC-93-003 CONTAINS THE FOLLOWING:

① CORRECTS ERRORS FOUND IN THE ORIGINAL CALCULATION WITH REGARD TO I-129 DOSES. BOTH THE I-131 EQUIVALENT ACTIVITY AND THE DOSES RESULTING FROM THIS ACTIVITY WERE FOUND TO BE IN ERROR. THE DOSE CALCULATIONS HAVE BEEN REDONE AND A NEW DOSE TABLE HAS BEEN PREPARED. ALSO, AN IODINE POOL DF OF 100 WAS BEEN APPLIED TO THE SHORT-LIVED IODINE RELEASE IN ORDER TO BE CONSISTENT & ACCURATE.

② DETERMINES DOSES AT THE CONTROL ROOM NORMAL VENTILATION INTAKE (CB-2) AS EVIDENCE OF THE NEED (OR LACK OF IT) FOR CB-1 OPERABILITY FOR RADIOLOGICAL CONSEQUENCES.

③ PROVIDES A REVISED SUMMARY STATEMENT.

---

① RECALCULATION OF CASE 2 RELEASE RATES (LIMITING VALUES) TO INCORPORATE A RADIAL PEAKING FACTOR OF 1.65:

	CASE 2 ACTIVITY (AVG. ASSEMBLY) (Ci)	PEAK ASSEMBLY ACTIVITY (Ci) (AVG * 1.65)
I-129*	1.007E-2	1.662E-2 → Fraction released = (0.3 *) = 4.99E-3
I-131	4.56E4	7.52E4
I-132	6.94E4	1.15E5
I-133	1.02E5	1.68E5
I-134	1.19E5	1.96E5
I-135	9.27E4	1.53E5
Xe-131m	3.46E2	5.71E2
Xe-133	1.05E5	1.73E5
Xe-133m	2.67E3	4.41E3
Xe-135	2.88E4	4.75E4
Xe-135m	2.83E4	4.67E4
Xe-138	9.27E4	1.53E5
Kr-83m	8.50E3	1.40E4
Kr-85	1.55E3	2.56E3
Kr-85m	2.05E4	3.38E4
Kr-87	3.93E4	6.48E4
Kr-88	5.60E4	9.24E4
Kr-89	7.25E4	1.20E5

All of these activities were corrected in Case 2 for fraction in gap, i.e. available for release. I-129 is adjusted above by 0.3 factor. (per R.G. 1.25)

* 1.943 Ci/193 = 1.007E-2 Ci average per Assembly.

ATTACHMENT VI
CPY-035-96

GENERAL PLASTICS
MANUFACTURING COMPANY
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(206) 473-5000

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TOLL FREE FROM
Seattle-Renton Area
(206) 623-2795

PHYSICAL PROPERTY DATA

LAST-A-FOAM® FR-3700

General Plastics Manufacturing Company's **LAST-A-FOAM® FR-3700** is a rigid, unicellular polyurethane foam that can be supplied in any density from 3 to 30 lbs./ft.^{3} The standard sheet size for **FR-3700** foam in the 3 to 6 lbs./ft.^{3} density range is 48" ± .25" x 96" ± .50" with a thickness tolerance of ± .015" on sheets up to 2" thick and ± .030" on sheets over 2" thick. The standard sheet size for **FR-3700** foam in the 8 to 30 lbs./ft.^{3} density range is 18" ± .10" x 100" ±.50" with a thickness tolerance of ± .005" on sheets up to 1" thick, ± .015" on sheets 1" to 2" thick, and ± .030" on sheets over 2" thick.

STRENGTH:

LAST-A-FOAM® FR-3700 series rigid polyurethane foams exhibit excellent strength to weight ratio because of the high strength polymer and cellular structure. The strengths shown in the accompanying graphs are nominal ULTIMATE values at which the foam fails to support a higher load. Appropriate safety factors should be incorporated into all designs for structural applications.

DURABILITY:

General Plastics' **LAST-A-FOAM®** rigid polyurethane foam is a very stable material which will not corrode, sustain fungus or attract rodents or insects. **LAST-A-FOAM®** has a high chemical resistance and is unaffected by most chemicals and solvents, except for some of the chlorinated solvents.

USES:

LAST-A-FOAM® FR-3700 series, rigid polyurethane foam in the 10 to 25 lbs./ft.^{3} density range is used extensively as high strength framing, or close-out strips, around the perimeter, and at the fastening points in honeycomb core or other similar structural panels. The **LAST-A-FOAM® FR-3700** foam in the 3 to 8 lbs./ft.^{3} density range is used extensively as the entire core material for many structural insulated panels.

INSTALLATION:

One method for fastening hardware, or other articles to the foam is to pot a threaded receptacle in the foam. Threaded fasteners may then be installed and removed many times without damaging the foam.

CLOSED CELL CONTENT:

Tested per ASTM D-2856, Procedure B.

95% minimum @ 3 lbs./ft.^{3} density
98% minimum @ 25 lbs./ft.^{3} density

LAST-A-FOAM® has low moisture vapor permeability and high resistance to water absorption because of its closed unicellular structure. **LAST-A-FOAM®** will not crack, split or swell when exposed to moisture.

MANUFACTURERS AND MOLDERS OF LAST-A-FOAM® HIGH DENSITY RIGID AND FLEXIBLE POLYURETHANE FOAMS
AND FABRICATORS OF PLASTIC SHEETS FOR AIRCRAFT, INDUSTRIAL, CONSTRUCTION, MARINE, NUCLEAR, SHIPPING AND MODELING

GENERAL PLASTICS
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LAST-A-FOAM® FR-3700

COEFFICIENT OF LINEAR THERMAL EXPANSION:

LAST-A-FOAM® FR-3700 exhibits a thermal expansion or contraction coefficient in the range of 3.5×10^{-5} in/in/°F to 5.0×10^{-5} in/in/°F over the temperature range of -310°F to +200°F.

HEAT DISTORTION:

This test is used to determine the suitability of **LAST-A-FOAM® FR-3700** for use as core material in laminated structural panels, using temperatures up to 250°F and vacuum pressure.

Samples of **FR-3700**, .50" thick are placed on a metal plate under a vacuum bag, with 20 inches of mercury minimum vacuum. These samples are then put into an oven (preheated to 250°F) for 2 hours. The samples are then removed and allowed to cool while under vacuum. The difference in thickness of the samples before and after the test is measured.

LAST-A-FOAM®	THICKNESS CHANGE (%)
FR-3704	- 1.15
FR-3706	- 0.90
FR-3710	- 0.75
FR-3715	- 0.34
FR-3720	- 0.25

THERMAL CONDUCTIVITY:

TESTED PER ASTM C-177 AT A MEAN TEMPERATURE OF 75°F.

LAST-A-FOAM® MATERIAL	k FACTOR BTU/hr-ft.²-°F/in.	R VALUE Hrs.-ft.²-°F/BTU in.
FR-3704	0.184	5.44
FR-3706	0.208	4.81
FR-3710	0.257	3.89
FR-3718	0.355	2.82
FR-3720	0.379	2.64
FR-3725	0.440	2.27

ELECTRICAL PROPERTIES:

LAST-A-FOAM® rigid polyurethane foam offers high electrical resistivity at low weight. The dielectric constant of **LAST-A-FOAM® FR-3700** varies linearly with density from 1.05 at 3.0 lbs./ft.³ to 1.4 at 20.0 lbs./ft.³ when tested at 1.0 MH_z. The change in dielectric constant is negligible within the temperature range of -50°F through 300°F.

MANUFACTURERS AND MOLDERS OF LAST-A-FOAM® HIGH DENSITY RIGID AND FLEXIBLE POLYURETHANE FOAMS
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(206) 623-2795

LAST-A-FOAM® FR-3700

FIRE SAFETY:

LAST-A-FOAM® rigid polyurethane foam is an organic material which will burn in the presence of sufficient heat and oxygen. The ASTM D-1692-74 and Federal Aviation Regulation (FAR) 25.853 flame test methods are comparative tests conducted under specific laboratory conditions. The sole purpose of these tests is to establish the relative burning characteristics of foam plastics. The results of these tests are not to be considered, or used as fire hazard classifications, and correlation with flammability of **LAST-A-FOAM®** under actual use conditions is neither intended nor implied.

In some circumstances, if the **LAST-A-FOAM®** is allowed to remain exposed and unprotected, a fire could spread rapidly under actual fire conditions, creating dense smoke and intense and immediate heat. **LAST-A-FOAM®** rigid polyurethane foams have chemical additives which help reduce their flammability and burning rate, and will produce minimum fire contribution in specific building assemblies when properly protected. It is recommended that adequate automatic sprinklers be incorporated into building construction wherever possible. In order to provide better occupancy protection, and if automatic sprinklers are not feasible, a plaster coating, cement asbestos, gypsum sheetrock paneling, or metal sheeting is necessary to cover exposed foam surfaces.

The ASTM D-1692-74 and FAR 25.853 flame test methods and test results are listed below:

ASTM D-1692-74 FLAME RESISTANCE TEST:

In this test a .5" thick x 2" wide x 6" long foam sample is placed in a horizontal position. The 2" end of the sample is exposed to the flame of a Bunsen burner with a wing tip attachment for 60 seconds. The time to flame extinguishment after start of ignition and the length of the sample which was burned are recorded. Average test values for **LAST-A-FOAM® FR-3700** foams are given below:

LAST-A-FOAM®	EXTINGUISHMENT, AFTER START OF IGNITION, TIME, SECONDS	BURN DISTANCE, INCHES
FR-3704	77.9	1.65
FR-3706	75.0	1.45
FR-3710	69.0	1.40
FR-3718	65.0	1.30
FR-3720	61.0	1.30

MANUFACTURERS AND MOLDERS OF LAST-A-FOAM® HIGH DENSITY RIGID AND FLEXIBLE POLYURETHANE FOAMS AND FABRICATORS OF PLASTIC SHEETS FOR AIRCRAFT, INDUSTRIAL, CONSTRUCTION, MARINE, NUCLEAR, SHIPPING AND MODELING

GENERAL PLASTICS
MANUFACTURING COMPANY
4910 BURLINGTON WAY/P.O. BOX 9097
TACOMA, WASHINGTON 98409
(206) 473-5000

TELEX 989134
FAX (206) 473-5104
TOLL FREE FROM
Seattle-Renton Area
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LAST-A-FOAM FR-3700

FEDERAL AVIATION REGULATION (FAR) 25.853, FLAME RESISTANCE TEST:

In this test a .5" thick x 3" wide x 12" long foam sample is mounted in a vertical position. The lower (.5" x 3") end is exposed to a Bunsen burner having a 1.5" high flame. The time of exposure is 12 or 60 seconds. The time to flame extinguishment after removal of the Bunsen burner flame and length of the sample which is burned are recorded. Average test values for **LAST-A-FOAM® FR-3700** foams are given below:

FAR 25.853 (b)

12 SECOND IGNITION

LAST-A-FOAM®	EXTINGUISHMENT, TIME SECONDS	BURN DISTANCE, INCHES
FR-3704	0.5	5.7
FR-3706	3.0	5.2
FR-3710	2.5	3.8
FR-3718	6.1	2.7
FR-3720	5.5	2.9

FAR 25.853 (a)

60 SECOND IGNITION

LAST-A-FOAM®	EXTINGUISHMENT, TIME SECONDS	BURN DISTANCE, INCHES
FR-3704	0.7	5.6
FR-3706	0.8	5.4
FR-3710	-0-	4.6
FR-3718	-0-	4.5
FR-3720	-0-	4.7

MANUFACTURERS AND MOLDERS OF LAST-A-FOAM® HIGH DENSITY RIGID AND FLEXIBLE POLYURETHANE FOAMS
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LAST-A-FOAM® FR-3700

PHYSICAL PROPERTY TEST DATA

June 1982

PROPERTY TESTED	LINE OF BEST FIT	95% LIMITS OF CONFIDENCE	n*	σ**	TEST METHOD
COMPRESSIVE STRENGTH (psi)					
Parallel to rise					
@ 75° F	$8.745D^{1.6154}$	± 5.75%	6	5.00%	ASTM D-1621
@ 250° F	$4.537D^{1.6050}$	± 11.30%	6	9.82%	
Perpendicular to rise					
@ 75° F	$3.456D^{1.9178}$	± 2.27%	6	1.98%	ASTM D-1621
@ 250° F	$2.966D^{1.7146}$	± 10.40%	6	9.06%	
COMPRESSIVE MODULUS (psi)					
Parallel to rise					
@ 75° F	$276.2D^{1.5413}$	± 5.90%	6	5.13%	ASTM D-1621
@ 250° F	$144.0D^{1.6396}$	± 22.06%	6	19.18%	
Perpendicular to rise					
@ 75° F	$74.66D^{1.9995}$	± 9.73%	6	8.46%	ASTM D-1621
@ 250°,F	$93.03D^{1.7174}$	± 16.82%	6	14.62%	

When D = Foam density in lbs./ft.^{3}

* n = number of densities tested, 1 - 5 specimens each density. **$\sigma = \sqrt{\frac{\Sigma(X-\bar{X})^2}{n}}$

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LAST-A-FOAM® FR-3700

PHYSICAL PROPERTY TEST DATA

June 1982

PROPERTY TESTED	LINE OF BEST FIT	95% LIMITS OF CONFIDENCE	n*	σ**	TEST METHOD
TENSILE STRENGTH (psi)					
Parallel to rise	$29.68D^{1.1496}$	± 3.45%	6	3.00%	ASTM D-1623 Type "A" Specimens
Perpendicular to rise	$14.74D^{1.3852}$	± 6.62%	6	5.76%	
TENSILE MODULUS (psi)					
Parallel to rise	$922.4D^{1.1817}$	± 7.88%	5	5.68%	ASTM D-1623 Type "B" Specimens
Perpendicular to rise	$286.5D^{1.5324}$	± 14.81%	6	12.88%	
SHEAR STRENGTH (psi)					
Parallel to rise	$7.530D^{1.5443}$	± 5.99%	5	4.32%	ASTM C-273 Compression Shear
Perpendicular to rise	$11.60D^{1.3754}$	± 1.80%	3	0.82%	
SHEAR MODULUS (psi)					
Parallel to rise	$40.53D^{1.8555}$	± 8.30%	5	5.98%	ASTM C-273 Compression Shear
Perpendicular to rise	$133.0D^{1.4209}$	± 7.29%	6	6.34%	
FLEXURAL STRENGTH (psi)					
Parallel to rise	$8.287D^{1.6660}$	± 9.56%	6	8.32%	ASTM D-790 Method 1-A
Perpendicular to rise	$21.17D^{1.3614}$	± 3.66%	4	1.99%	
FLEXURAL MODULUS (psi)					
Parallel to rise	$151.2D^{1.8084}$	± 11.48%	6	9.98%	ASTM D-790 Method 1-A
Perpendicular to rise	$545.8D^{1.3628}$	± 19.78%	4	10.56%	

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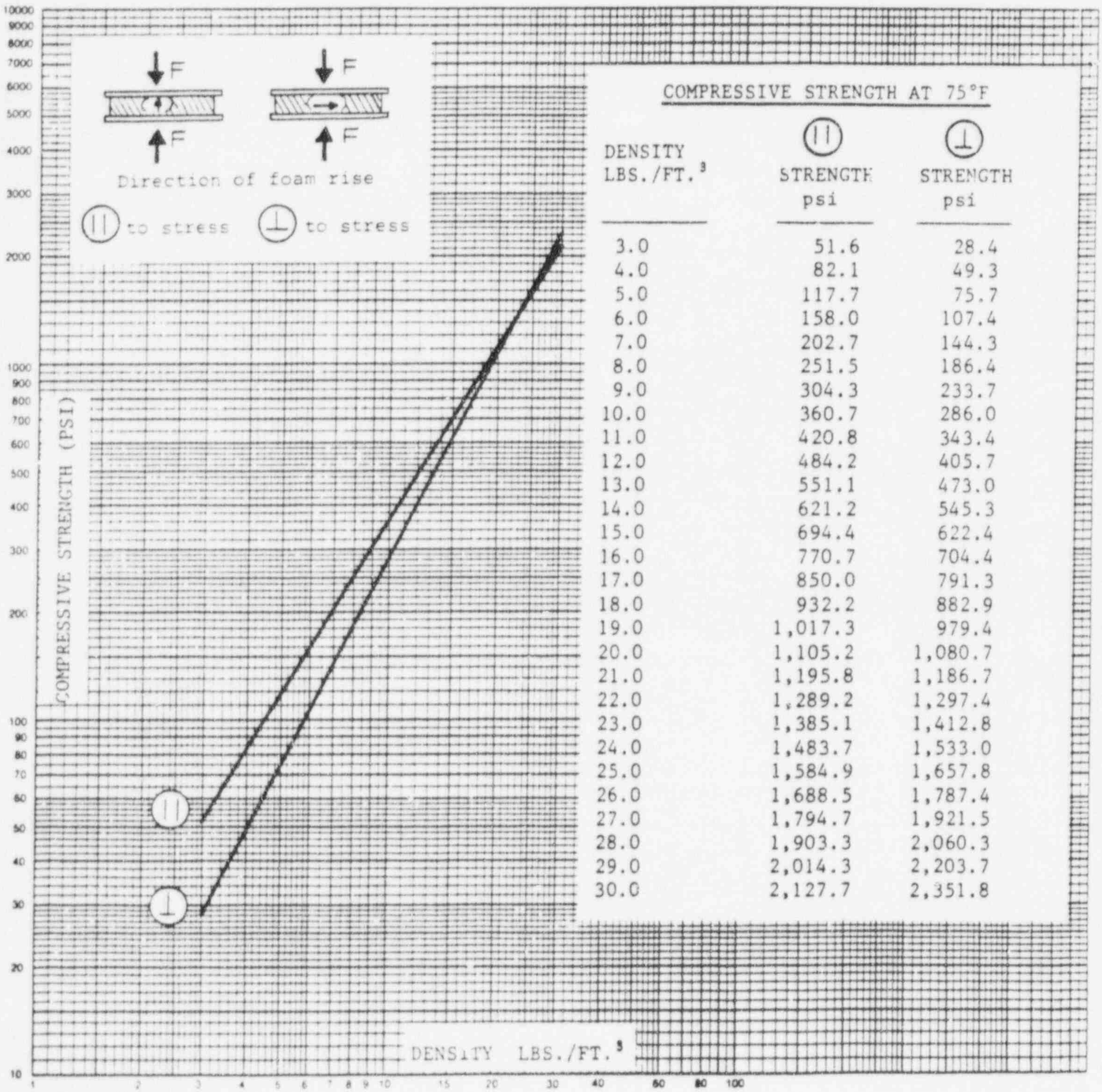
LAST-A-FOAM® FR-3700



COMPRESSIVE STRENGTH AT 75°F

DENSITY LBS./FT.³	(||) STRENGTH psi	(⊥) STRENGTH psi
3.0	51.6	28.4
4.0	82.1	49.3
5.0	117.7	75.7
6.0	158.0	107.4
7.0	202.7	144.3
8.0	251.5	186.4
9.0	304.3	233.7
10.0	360.7	286.0
11.0	420.8	343.4
12.0	484.2	405.7
13.0	551.1	473.0
14.0	621.2	545.3
15.0	694.4	622.4
16.0	770.7	704.4
17.0	850.0	791.3
18.0	932.2	882.9
19.0	1,017.3	979.4
20.0	1,105.2	1,080.7
21.0	1,195.8	1,186.7
22.0	1,289.2	1,297.4
23.0	1,385.1	1,412.8
24.0	1,483.7	1,533.0
25.0	1,584.9	1,657.8
26.0	1,688.5	1,787.4
27.0	1,794.7	1,921.5
28.0	1,903.3	2,060.3
29.0	2,014.3	2,203.7
30.0	2,127.7	2,351.8

(||) Indicates stress is parallel to direction of foam rise.

(⊥) Indicates stress is perpendicular to direction of foam rise

ATTACHMENT VII

Trojan Calculation RPC 96-009

*** QA RECORD WHEN COMPLETED ***

RC 2 4 2
CFP GEN ENGR 7-6
OOR Admin Services
Letter Number NA
System Number NA
Number of Pages 1
Document Date
Calc. Reference #

TROJAN CALCULATION COVER SHEET

Sheet 1 Cont'd on Sheet 2

Title Site Boundary Doses - Rupture ISFSI Basket Helium Supply Line

Trojan Nuclear Plant

Calculation No. RPC 96-009

Structure N/A

Supersedes Calculation No. N/A

System Helium Pressure Test Line

Quality-Related Yes/No

Component ISFSI Basket

Status: ✓ Final ___ Interim

References (PMR/DPMR, SPEER, MR, PSC, etc.)

Affected Document No.	Has Been Changed by Identify Change Vehicle: (MR, DPMR, DCP, PCF, SPEER, PSC, etc.)	Or Revision has been Deferred by (Identify Memo, CTL, etc.)	Responsible Supervisor/Date (Deferrals Only)
—	—	—	—

Calculation Objective — To determine the early-phase dose projections at the site boundary due to the rupture of a helium supply line attached to an ISFSI basket.

Revision Description

Rev. No.	Preparer	Date	Verified By	Date	Approved By	Date
0	Michael Stein	5/28/96	[signature]	5/28/96	Tom Meek	5/28/96

QA RECORD WHEN COMPLETED Sheet 2 of 9

CALCULATION / CMR VERIFICATION CHECKLIST

Calculation No. RPC 96-009 Verified By [signature] Date 5/28/96

Title Site Boundary Doses - Rupture ISFSI Basket Helium Supply Line

		Yes	No	N/A
1.	Are analytical methods appropriate?	☑	☐	☐
2.	Can a person technically qualified in the subject review and understand the calculation without recourse to the originator?	☑	☐	☐
3.	Is the calculation mathematically accurate?	☑	☐	☐
4.	Are the calculation objectives and design inputs (including source) clearly defined?	☑	☐	☐
5.	Do calculational parameters comply with design criteria / dimensions?	☑	☐	☐
6.	Does the calculation reference or list major equation sources?	☑	☐	☐
7.	Is input data appropriate?	☑	☐	☐
8.	Are assumptions properly referenced and correct?	☑	☐	☐
9.	Was am applicable and validated computer program used?	☑	☐	☐
10.	If applicable, were program error notices reviewed?	☐	☐	☑
11.	If an unverified program was used, is an alternate check calculation completed?	☐	☐	☑
12.	Do the output results seem reasonable?	☑	☐	☐

GENERAL COMPUTATION SHEET

Sheet 3 of 9 Sheets
☑ Calc. ☐ Job ☐ File No. RPC 96-009
Project/Job

Subject: Site Boundary Doses - Rupture ISFSI Basket Helium Supply Line
By: M. Stein Date 5/28/96 Chk. By [illegible] Date 5/28/96 Orig ☑ Rev. ☐

1.0 Objective:

To determine the early-phase dose projections at the site boundary due to the rupture of a helium supply line attached to an ISFSI basket.

2.0 Results:

	Aux/Fuel Building Ventilation Isolated	Aux/Fuel Building Ventilation Running
CEDE* (Whole body)	1.43×10^{2} mrem	2.82×10^{-1} mrem
CDE Thyroid	4.17 mrem	8.29×10^{-3} mrem
CDE Red Bone Marrow	1.90×10^{2} mrem	3.76×10^{-1} mrem
CDE Lung	2.89×10^{2} mrem	5.73×10^{-1} mrem
CDE Bone Surfaces	2.31×10^{3} mrem	4.58 mrem
CDE Gonads	3.63×10^{1} mrem	7.19×10^{-2} mrem
CDE Breast	4.73 mrem	9.39×10^{-3} mrem
CDE Remainder Organs	9.10×10^{1} mrem	1.80×10^{-1} mrem

* TEDE = CEDE + DDE. The DDE from submersion in a cloud of particulate airborne radioactivity is insignificant when compared to the CEDE due to inhalation. Therefore, the CEDE (Whole body) calculated above is considered equal to the TEDE.

Subject Site Boundary Doses - Rupture ISFSI Basket Helium Supply Line
By M. Stein Date 5/28/96 Chk. By [illegible] Date 5/28/96 Orig ☑ Rev. ☐

3.0 Method (All method results on Quatro Pro spreadsheets - Appendix A)

3.1 Assume that the nuclide distribution of surface contaminants, on the fuel assemblies and reactor control components, is the same as the nuclide distribution of surface contaminants inside the Steam Generator tubes.
Use TMA/Norcal S/G tube sample analysis results to determine nuclide distribution (RPC 95-001 Attachment D)

3.2 Normalize TMA/Norcal results and decay fractional activity to 1/1/98, which coincides with the earliest date of fuel loading into a basket.

3.3 Normalize decayed fractional Activity. This gives the fraction activity of each nuclide in distribution on 1/1/98

3.4. Determine surface contamination levels on fuel assemblies and reactor control components.
From Sandia Report, SAND88-1358 TTC-0811·UC-71
Estimate of CRUD Contribution to Shipping Cask Containment Requirements,

the surface contamination levels on Trojan fuel, are 220 $\mu Ci/cm^2$ on 5 percent of the fuel rods with the remaining 95 percent being CRUD-free. Contamination levels are at time of fuel discharge.

Assume a 1 $\mu Ci/cm^2$ surface contamination level for the CRUD-free portion of the fuel rods.

The effective surface contamination levels on Trojan fuel are

$$(220 \mu Ci/cm^2 \times 0.05) + (1 \mu Ci/cm^2 \times 0.95) \cong 12 \mu Ci/cm^2$$

For conservatism, the effective contamination levels at time of fuel discharge will be used as the effective contamination levels on all the fuel and reactor control components at the time of the accident on 1/1/98 (No decay corrections).

3.5 Multiply the fractional activity of each nuclide in the distribution on 1/1/98 by 12 $\mu Ci/cm^2$ to determine the surface contamination level of each nuclide in units of $\mu Ci/cm^2$.

Subject: Site Boundary Doses - Rupture ISFSI Basket Helium Supply Line
By M. Stein Date 5/28/96 Chk. By [signature] Date 5/28/96 Orig ☑ Rev. ☐

3.6 Calculate the total Activity of each nuclide in the basket by multiplying each nuclide surface contamination level by the total surface area of the fuel assemblies and reactor control components inside the basket.

From Calculation BNFL 01.10.06.05, TranStor™ Shipping Cask Containment Analysis (PWR), Sheets 15 and 16, the surface area of 24 fuel assemblies is 8.4×10^6 cm² and the surface area of the reactor control components is 4.7×10^4 cm².

Total surface area = $8.4 \times 10^6 \text{ cm}^2 + 4.7 \times 10^4 \text{ cm}^2 = 8.447 \times 10^6 \text{ cm}^2$

3.7 From Sandia Report, SAND88-1358 TTC-0811 · UC-71, Estimate of CRUD Contribution to Shipping Cask Containment Requirements, the maximum spallation of 15% for PWR rods is assumed. Therefore, the release fraction of each nuclide (available for airborne release) is determined by multiplying the total activity of each nuclide in the basket by 0.15.

3.8 The basket free gas volume at 7psig is 359,716 in³. This volume is converted to an equivalent volume at standard conditions of 531,009 in³. (Calculation PGE 01-10.02.05-05, TranStor™ Failure Modes and Effects Analysis, sheet 11)

The airborne activity concentration for each nuclide is equal to the nuclide release fraction divided by 531,009 in³.

3.9 The activity of each nuclide released to the Fuel Building atmosphere is equal to the activity concentration for each nuclide multiplied by the volume of gas released from the basket during the accident. The volume of gas released during the accident is 171,293 in³, from Calculation PGE 01-10.02.05-05, sheet 11.

3.10 The CEDE and the CDE to various organs is calculated for an individual at the site boundary using the methods and parameters from RPC 94-008, Limiting Activity Releases due to Decommissioning Activities.

3.10.1 With the Aux/Fuel Building Ventilation Isolated, the Equations for doses at the site boundary are:

$$CEDE_{i\,(mrem)} = \chi/Q \times BR \times 1.0\times10^{6}\ \mu Ci/Ci \times 7200\ sec \times 1.13\times10^{-1} \times DCF_{WB} \times NQ_i/T$$

$$CDE_{ij\,(mrem)} = \chi/Q \times BR \times 1.0\times10^{6}\ \mu Ci/Ci \times 7200\ sec \times 1.13\times10^{-1} \times DCF_j \times NQ_i/T$$

(Equations from RPC 94-008, Sheet 14).

Where: χ/Q = Atmospheric dispersion coefficient = $5.71\times10^{-4}\ sec/m^3$
= $6.20\times10^{-4}\ sec/m^3$ ^3H

BR = Breathing Rate = $3.33\times10^{-4}\ m^3/sec$

7200 Seconds = 2 hours = Short term accident release duration

1.13×10^{-1} = fraction of Activity released from Aux/Fuel Building in 2 hours with ventilation isolated. (RPC 94-008, sheet 9)

DCF_{WB} = Effective whole body dose conversion factor in units of mrem/μCi for inhalation.

NQ_i/T = Release rate of nuclide i = $\frac{\text{Activity Released (Ci)}}{7200\ sec}$

$CEDE_{i\,(mrem)}$ = committed effective dose equivalent due to inhalation of nuclide i (mrem).

$CDE_{ij\,(mrem)}$ = Committed dose equivalent to organ j due to inhalation of nuclide i (mrem).

DCF_j = inhalation dose conversion factor for organ j in units of mrem/μCi.

62-4712 11-79
8-40-207

GENERAL COMPUTATION SHEET

Calc. [x] Job [ ] File [ ] No. RPC 96-009
Project/Job

Subject: Site Boundary Doses - Rupture ISFSI Basket Helium Supply Line
By: M. Stern Date 5/28/96 Chk. By [illegible] Date 5/28/96 Orig [x] Rev. [ ]

3.10.2 With the Aux/Fuel Building ventilation running the equations for doses at the site boundary are:

$$CEDE_{i\,(mrem)} = \chi/Q \times BR \times 1.0\times10^{6}\ \mu Ci/Ci \times 7200\ sec \times DCF_{WB} \times RQi/T$$

$$CDE_{ij\,(mrem)} = \chi/Q \times BR \times 1.0\times10^{6}\ \mu Ci/Ci \times 7200\ sec \times DCF_{j} \times RQi/T$$

(Equations from RPC 94-008, Sheet 12)

where: χ/Q = Atmospheric dispersion coefficient = 4.26×10^{-4} sec/m³

BR = Breathing Rate = 3.33×10^{-4} m³/sec

7200 sec = 2 hours = Short term accident release duration

DCF_{WB} = Effective whole body dose conversion factor for inhalation in units of mrem/µCi.

$$RQ_i/T = \frac{\text{Activity Released nuclide } i\ (Ci) \times 3.0\times10^{-4}}{7200\ sec}$$

(3.0×10^{-4} is the fraction of particulates remaining after filtration by HEPA units in Aux/Fuel building ventilation for ^3H fraction = 1.0)

$CEDE_{i\,(mrem)}$ = Committed Effective dose Equivalent due to inhalation of nuclide i (mrem).

$CDE_{ij\,(mrem)}$ - Committed dose equivalent to organ j due to inhalation of nuclide i (mrem).

DCF_j = inhalation dose conversion factor for organ j in units of mrem/µCi.

PGE 1787 (Nov 89)

PORTLAND GENERAL ELECTRIC

CALCULATION SHEET

Appendix A

culation No. PPC 96-009 Revision 0

Sheet 1 of 5

Preparer M. Stegm Date 5/28/96

Verifier [illegible] Date 5/28/96

Fuel Assembly Surface Contamination Levels

Nuclide	Half Life (Days)	Sample Results 10/31/94	Fractional Activity Sample	Lambda (Days-1)	Fractional Activity Decayed To 01/01/98	Activity Normalized 01/01/98	Surface Activity (uCi/cm2)	Total Activity Basket (Curies)
H-3	4.51E+03	6.25E-03	4.00E-04	1.54E-04	3.35E-04	5.293E-04	6.352E-03	5.37E-02
C-14	2.09E+06	7.48E-03	4.79E-04	3.32E-07	4.79E-04	7.567E-04	9.081E-03	7.67E-02
Sb-125	1.01E+03	2.31E-01	1.48E-02	6.86E-04	6.70E-03	1.058E-02	1.270E-01	1.07E+00
Ce-144	2.84E+02	4.39E-02	2.81E-03	2.44E-03	1.68E-04	2.648E-04	3.178E-03	2.68E-02
Mn-54	3.13E+02	6.33E-02	4.05E-03	2.21E-03	3.13E-04	4.953E-04	5.943E-03	5.02E-02
Fe-55	9.86E+02	3.90E+00	2.50E-01	7.03E-04	1.11E-01	1.750E-01	2.100E+00	1.77E+01
Co-60	1.93E+03	9.57E+00	6.13E-01	3.59E-04	4.05E-01	6.396E-01	7.675E+00	6.48E+01
Ni-63	3.51E+04	1.33E+00	8.50E-02	1.97E-05	8.30E-02	1.312E-01	1.575E+00	1.33E+01
Sr-90	1.06E+04	6.46E-02	4.14E-03	6.54E-05	3.84E-03	6.062E-03	7.274E-02	6.14E-01
38	3.20E+04	5.57E-03	3.56E-04	2.17E-05	3.48E-04	5.493E-04	6.592E-03	5.57E-02
Pu-239/40	2.39E+06	6.07E-03	3.89E-04	2.90E-07	3.89E-04	6.142E-04	7.371E-03	6.23E-02
Pu-241	5.26E+03	3.81E-01	2.44E-02	1.32E-04	2.09E-02	3.311E-02	3.973E-01	3.36E+00
Cm-242	1.63E+02	1.07E-04	6.88E-06	4.25E-03	5.05E-08	7.974E-08	9.569E-07	8.08E-06
Cm-243	1.17E+04	2.50E-03	1.60E-04	5.93E-05	1.49E-04	2.362E-04	2.834E-03	2.39E-02
Cm-244	6.42E+03	2.50E-03	1.60E-04	1.08E-04	1.41E-04	2.233E-04	2.680E-03	2.26E-02
Am-241	1.58E+05	7.22E-03	4.62E-04	4.39E-06	4.60E-04	7.269E-04	8.723E-03	7.37E-02
Pu-242	1.37E+08	3.06E-05	1.96E-06	5.06E-09	1.96E-06	3.096E-06	3.716E-05	3.14E-04
		15.62 (uCi/sample)	1		0.63	1	12	1.01E+02

PORTLAND GENERAL ELECTRIC CALCULATION SHEET Appendix A

Calculation No RPC 96-009 Revision 0 Sheet 2 of 5

Preparer M. Stag Date 5/28/96

Verifier [illegible] Date 5/28/96

Activity Released to Fuel Building Atmosphere

Nuclide	Total Activity Basket (Curies)	Release Fraction (Curies)	Activity Concentration (Ci/in3)	Activity Released Atmosphere (Curies)
H-3	5.37E-02	8.05E-03	1.52E-08	2.60E-03
C-14	7.67E-02	1.15E-02	2.17E-08	3.71E-03
Sb-125	1.07E+00	1.61E-01	3.03E-07	5.19E-02
Ce-144	2.68E-02	4.03E-03	7.58E-09	1.30E-03
Mn-54	5.02E-02	7.53E-03	1.42E-08	2.43E-03
Fe-55	1.77E+01	2.66E+00	5.01E-06	8.58E-01
Co-60	6.48E+01	9.73E+00	1.83E-05	3.14E+00
Ni-63	1.33E+01	2.00E+00	3.76E-06	6.44E-01
Sr-90	6.14E-01	9.22E-02	1.74E-07	2.97E-02
Pu-238	5.57E-02	8.35E-03	1.57E-08	2.69E-03
Pu-239/40	6.23E-02	9.34E-03	1.76E-08	3.01E-03
Pu-241	3.36E+00	5.03E-01	9.48E-07	1.62E-01
Cm-242	8.08E-06	1.21E-06	2.28E-12	3.91E-07
Cm-243	2.39E-02	3.59E-03	6.76E-09	1.16E-03
n-244	2.26E-02	3.40E-03	6.39E-09	1.10E-03
.m-241	7.37E-02	1.11E-02	2.08E-08	3.57E-03
Pu-242	3.14E-04	4.71E-05	8.87E-11	1.52E-05
	1.01E+02		2.86E-05	4.90E+00

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M. Stein 5/28/96
[illegible] 5/28/96

SITE BOUNDARY DOSE FROM RUPTURE OF HELIUM SUPPLY LINE - VENTILATION ISOLATED

Nuclide	CEDE WB (mrem)	CDE Thy (mrem)	CDE RBM (mrem)	CDE Lung (mrem)	CDE BS (mrem)	CDE Gona (mrem)	CDE Breast (mrem)	CDE Remai (mrem)
H-3	3.88E-06	3.88E-06	3.88E-06	3.88E-06	3.88E-06	3.88E-06	3.88E-06	3.88E-06
C-14	1.67E-04	1.67E-04	1.67E-04	1.67E-04	1.67E-04	1.67E-04	1.67E-04	1.67E-04
Sb-125	1.36E-02	1.34E-03	2.68E-03	8.95E-02	1.13E-02	1.48E-03	1.72E-03	5.99E-03
Ce-144	1.04E-04	1.94E-04	2.76E-03	8.18E-02	4.69E-03	2.47E-05	2.04E-04	1.06E-02
Mn-54	3.50E-04	1.43E-04	3.21E-04	1.28E-03	4.94E-04	1.71E-04	1.76E-04	4.04E-04
Fe-55	4.96E-02	3.71E-02	3.52E-02	7.23E-02	3.50E-02	3.58E-02	3.47E-02	8.26E-02
Co-60	1.48E+01	4.04E+00	4.29E+00	8.64E+01	3.37E+00	1.19E+00	4.59E+00	8.97E+00
Ni-63	8.70E-02	8.70E-02	8.70E-02	1.58E-01	8.70E-02	8.70E-02	8.70E-02	8.70E-02
Sr-90	8.30E-01	6.23E-03	7.91E-01	6.76E+00	1.72E+00	6.23E-03	6.23E-03	1.35E-02
Pu-238	2.27E+01	2.06E-04	3.25E+01	6.82E+01	4.06E+02	6.01E+00	2.14E-04	1.50E+01
Pu-239/240	2.77E+01	2.16E-04	4.04E+01	7.76E+01	5.05E+02	7.63E+00	2.21E-04	1.81E+01
Pu-241	2.87E+01	1.60E-04	4.32E+01	4.11E+01	5.40E+02	8.77E+00	3.93E-04	1.69E+01
Cm-242	1.45E-04	2.92E-08	1.21E-04	4.82E-04	1.51E-03	1.77E-05	2.93E-08	7.62E-05
Cm-243	7.65E+00	3.53E-04	1.09E+01	1.79E+00	1.36E+02	1.91E+00	5.80E-04	5.31E+00
Cm-244	5.86E+00	8.83E-05	8.20E+00	1.69E+00	1.02E+02	1.39E+00	9.09E-05	4.18E+00
Am-241	3.41E+01	4.54E-04	4.94E+01	5.22E+00	[illegible]6E+02	9.22E+00	7.58E-04	2.22E+01
Pu-242	1.34E-01	1.06E-06	1.95E-01	3.71E-01	2.43E+00	3.65E-02	1.14E-06	8.68E-02
	1.43E+02	4.17E+00	1.90E+02	2.89E+02	2.31E+03	3.63E+01	4.73E+00	9.10E+01

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M. Stern 5/28/96
[signature] 5/28/96

SITE BOUNDARY DOSE FROM RUPTURE OF HELIUM SUPPLY LINE - VENTILATION RUNNING

Nuclide	CEDE WB (mrem)	CDE Thy (mrem)	CDE RBM (mrem)	CDE Lung (mrem)	CDE BS (mrem)	CDE Gona (mrem)	CDE Breast (mrem)	CDE Remai (mrem)
H-3	2.36E-05	2.36E-05	2.36E-05	2.36E-05	2.36E-05	2.36E-05	2.36E-05	2.36E-05
C-14	3.30E-07	3.30E-07	3.30E-07	3.30E-07	3.30E-07	3.30E-07	3.30E-07	3.30E-07
Sb-125	2.69E-05	2.65E-06	5.30E-06	1.77E-04	2.23E-05	2.94E-06	3.40E-06	1.19E-05
Ce-144	2.07E-07	3.85E-07	5.47E-06	1.62E-04	9.29E-06	4.89E-08	4.03E-07	2.11E-05
Mn-54	6.93E-07	2.83E-07	6.35E-07	2.54E-06	9.79E-07	3.38E-07	3.50E-07	7.99E-07
Fe-55	9.82E-05	7.34E-05	6.97E-05	1.43E-04	6.94E-05	7.08E-05	6.86E-05	1.64E-04
Co-60	2.93E-02	8.00E-03	8.50E-03	1.71E-01	6.68E-03	2.35E-03	9.10E-03	1.78E-02
Ni-63	1.72E-04	1.72E-04	1.72E-04	3.12E-04	1.72E-04	1.72E-04	1.72E-04	1.72E-04
Sr-90	1.64E-03	1.23E-05	1.57E-03	1.34E-02	3.40E-03	1.23E-05	1.23E-05	2.68E-05
Pu-238	4.49E-02	4.08E-07	6.43E-02	1.35E-01	8.05E-01	1.19E-02	4.24E-07	2.98E-02
Pu-239/240	5.50E-02	4.28E-07	8.01E-02	1.54E-01	1.00E+00	1.51E-02	4.37E-07	3.59E-02
Pu-241	5.69E-02	3.16E-07	8.55E-02	8.14E-02	1.07E+00	1.74E-02	7.79E-07	3.34E-02
Cm-242	2.88E-07	5.79E-11	2.40E-07	9.55E-07	3.00E-06	3.51E-08	5.81E-11	1.51E-07
Cm-243	1.52E-02	7.00E-07	2.16E-02	3.54E-03	2.69E-01	3.78E-03	1.15E-06	1.05E-02
Cm-244	1.16E-02	1.75E-07	1.62E-02	3.34E-03	2.03E-01	2.75E-03	1.80E-07	8.28E-03
Am-241	6.75E-02	8.99E-07	9.78E-02	1.03E-02	1.22E+00	1.83E-02	1.50E-06	4.40E-02
Pu-242	2.66E-04	2.10E-09	3.85E-04	7.35E-04	4.81E-03	7.23E-05	2.26E-09	1.72E-04
	2.82E-01	8.29E-03	3.76E-01	5.73E-01	4.58E+00	7.19E-02	9.39E-03	1.80E-01

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DOSE CONVERSION FACTORS - SITE BOUNDARY DOSE RUPTURE OF HELIUM SUPPLY LINE
(mrem/µCi)

Nuclide	DCF Whole Bod	DCF Thyroid	DCF RBM	DCF Lung	DCF Bone Surf.	DCF Gonad	DCF Breast	DCF Remainder	Activity Released (Curies)
H-3	6.40E-02	6.40E-02	6.40E-02	6.40E-02	6.40E-02	6.40E-02	6.40E-02	6.40E-02	2.60E-03
C-14	2.09E+00	2.09E+00	2.09E+00	2.09E+00	2.09E+00	2.09E+00	2.09E+00	2.09E+00	3.71E-03
Sb-125	1.22E+01	1.20E+00	2.40E+00	8.03E+01	1.01E+01	1.33E+00	1.54E+00	5.37E+00	5.19E-02
Ce-144	3.74E+00	6.96E+00	9.88E+01	2.93E+03	1.68E+02	8.84E-01	7.29E+00	3.81E+02	1.30E-03
Mn-54	6.70E+00	2.[illegible]4E+00	6.14E+00	[illegible].46E+01	9.47E+00	3.27E+00	3.38E+00	7.73E+00	2.43E-03
Fe-55	2.69E+00	2.01E+00	1.91E+00	3.92E+[illegible]	1.90E+00	1.94E+00	1.88E+00	4.48E+00	8.58E-01
Co-60	2.19E+0[illegible]	5.99E+01	6.36E+01	1.28E+03	5.00E+01	1.76E+01	6.81E+01	1.33E+02	3.14E+00
Ni-63	6.29E+[illegible]0	6.29E+00	6.29E+00	1.14E+01	6.29E+00	6.29E+00	6.29E+00	6.29E+00	6.44E-01
Sr-90	1.30E+03	9.77E+00	1.24E+03	1.06E+04	2.69E+03	9.77E+00	9.77E+00	2.12E+01	2.97E-02
Pu-238	3.92[illegible]+05	3.56E+00	5.62E+05	1.18E+06	7.03E+06	1.04E+05	3.70E+00	2.60E+05	2.69E-03
Pu-239/240	4.2[illegible]E+05	3.34E+00	6.25E+05	1.20E+06	7.81E+06	1.18E+05	3.41E+00	2.80E+05	3.01E-03
Pu-241	8.[illegible]5E+03	4.59E-02	1.24E+04	1.18E+04	1.55E+05	2.52E+03	1.13E-01	4.85E+03	1.62E-01
Cm-242	1.73E+04	3.48E+00	1.44E+04	5.74E+04	1.80E+05	2.11E+03	3.49E+00	9.07E+03	3.91E-07
Cm-243	[illegible].07E+05	1.42E+01	4.37E+05	7.18E+04	5.44E+06	7.66E+04	2.33E+01	2.13E+05	1.16E-03
Cm-244	[illegible].48E+05	3.74E+00	3.47E+05	7.14E+04	4.33E+06	5.88E+04	3.85E+00	1.77E+05	1.10E-03
Am-241	4.44E+05	5.92E+00	6.44E+05	6.81E+04	8.03E+06	1.20E+05	9.88E+00	2.89E+05	3.57E-03
Pu-242	4.11E+05	3.25E+00	5.96E+05	1.14E+06	7.44E+06	1.12E+05	3.50E+00	2.66E+05	1.52E-05

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GENERAL COMPUTATION SHEET

Sheet 1 of 11 Sheets

☑ Calc. ☐ Job ☐ File No. RPC 96-009

Project/Job

Subject Site Boundary Doses - Rupture ISFSI Basket Helium Supply Line

By M. Stein Date 5/28/96 Chk. By [signature] Date 5/28/96 Orig ☑ Rev. ☐

Appendix B: Support Documentation

PGE 1787 (Nov 89)

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SANDIA REPORT

SAND88–1358 • TTC–0811 • UC–71
Unlimited Release
Printed January 1991

Estimate of CRUD Contribution to Shipping Cask Containment Requirements

1-408-438-6444
5206

Robert P. Sandoval, Robert E. Einziger, Hans Jordan,
Anthony P. Malinauskas, Walter J. Mings

Prepared by
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Albuquerque, New Mexico 87185 and Livermore, California 94550
for the United States Department of Energy
under Contract DE-AC04-76DP00789

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For shipments of fuel 5 years old or older, the most important radionuclide in CRUD is cobalt-60. A compilation of measured cobalt-60 maximum "spot" activity densities and visual observations versus the number of rods, assemblies, or samples examined, is given in Tables 2 and 3 for PWR and BWR fuel, respectively. The activity density ranges in Tables 2 and 3 are distributed by rod number. The reader is referred to Appendix I, Section I.5 for the details of this distribution. A histogram was constructed for PWR and BWR fuel shown in Tables 2 and 3 to display the percentage of rods in selected activity density intervals. Both types of fuel rods exhibit bimodal distributions that are dominated by an essentially CRUD-free mode (see Figures 3 and 4).

In all likelihood, these distributions are not representative of CRUD deposits on the entire spent-fuel rod population. CRUD measurements are normally made during routine fuel qualification and surveillance programs, or when there is an in-reactor problem which is suspected to be caused by CRUD. The CRUD activity distribution may be skewed to the higher "spot" activities, and may therefore provide a conservative (upper-bound) measure of the activity density associated with CRUD deposits on spent fuel. The number of rods in the first mode of Figures 3 and 4 (i.e., the CRUD-free modes) are expected to increase with time because the control of reactor water chemistry has improved in recent years. Rods recently irradiated and those irradiated in the future will most likely fall in the lower part of the CRUD distribution. The second mode encompasses a much larger CRUD load but represents significantly fewer assemblies.

4.3 CRUD Spallation

CRUD can contribute to the radiological source term during transportation only if it spalls from the rod and becomes airborne. Spallation will depend on the condition of the CRUD at the time of fuel discharge, the effects of pool storage, and subsequent handling. The ability of CRUD to become airborne will depend on the particle size distribution of the spalled CRUD.

There is no published systematic study of CRUD adhesion or cohesion. A flocculent CRUD can easily be removed from the rod with a soft brush. On 108 samples from Brunswick-2 BWR fuel rods, 65% of the CRUD was of the flocculent type [AS78]. Spallation of this reddish flocculent CRUD has been observed during handling in spent-fuel pools; the resulting phenomenon is known as the "tomato soup effect." In contrast, a tenacious CRUD is usually found on PWR fuel rods. Generally some sort of silicon carbide stone or paper is needed to remove this type of CRUD.

The tenacity of the CRUD may be affected by the length of time in the storage pool and the condition of the water in the pool. Examinations of Oconee PWR fuel between 1978 and 1982 indicated that CRUD which appeared rather adherent in 1978 had a flocculent appearance in 1982 [NE86, BA83, JO82]. The data, however, are insufficient to determine if adherence will degenerate in every case.

There are a number of observations of CRUD behavior in stagnant dry air and nitrogen atmospheres. CRUD spallation was observed on a steam-generating heavy-water reactor (SGHWR) fuel rod when the temperature

Table 2

Range of Cobalt-60 Maximum "Spot"
Activity in PWR CRUD Deposit

Reactor	#Rods	#Assemblies	#Samples	Activity Density[(a)] ($\mu Ci/cm^2$)	Visual Inspection
Yankee Rowe	10	2		140	
Oconee 1	300	5	27	10	
Point Beach 1	280	5		0.1-2.0	Very thin
Benzau 1	2100	35		0.1-16	
Zion 1	600	16		37-73	
H.B. Robinson	5	6		19	
Beaver Valley	45	8		4-21[(b)]	
Trojan	3700	24	205	1-220[(b)]	5%-crudded, remainder-CRUD-free
Zorita	Many		12		Thin
Surry	64	1			Not significant
Calvert Cliffs	98				Little
Turkey Point	320				Very thin
Maine Yankee	27	4			Very thin
ANO-1	480				390-no CRUD, 90-minimal
ANO-2	384			0	CRUD-free
BR-3	7	4			Thick
Farley-1	136	2			Light
Point Beach	112	2			Light

(a) At time of fuel discharge.
(b) CRUD levels given in $\mu Ci/gm$ were converted to $\mu Ci/cm^2$ using thickness and density measurements.

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Table 3

Range of Cobalt-60 Maximum "Spot" Activity in BWR CRUD Deposit

Reactor	#Rods	#Assemblies	#Samples	Activity Density[(a)] ($\mu Ci/cm^2$)	Visual Inspection
Peach Bottom	125	5		11	CRUD-free
Monticello	4	5		0-350	Light
Tsuruga	5	1		140-1250	
Millstone 1	4			100-650	
Nine Mile Pt	23	4		200-408[(b)]	
Dresden 1	19	1			Observed
Big Rock Pt			16		Very thin
Oyster Creek	30				Light
AESA-660[(a)]	9	4			Very thick

(a) At time of fuel discharge.
(b) CRUD levels given in $\mu Ci/gm$ were converted to $\mu Ci/cm^2$ using thickness measurements and a density value of of 1.1 gm/cm^3.

reached 100°C in an air atmosphere [GA77]. The rod had failed in the reactor with some indications of in-core overheating, and the observation may not be applicable to light-water-reactor (LWR) fuel rods. Cooper BWR fuel assemblies spent 4 1/2 months in either helium, nitrogen, or vacuum at 241°C, with no noticeable changes in CRUD characteristics [MC86].

Einziger and Cook tested intact and defective H. B. Robinson PWR and Peach Bottom BWR fuel rods in flowing air and in a static argon atmosphere at 230°C for 5960 hr [EC84]. After 5960 hr in flowing air, 5.6% of the CRUD on the Robinson rod had spalled, and between 1.6 and 4.8% of the CRUD on the Peach Bottom rod had spalled. Further testing produced only minimal additional spallation (less than 1%). However, when relating these results to transportation spallation, it must be remembered that the fuel rods were stationary, without vibration.

446°F

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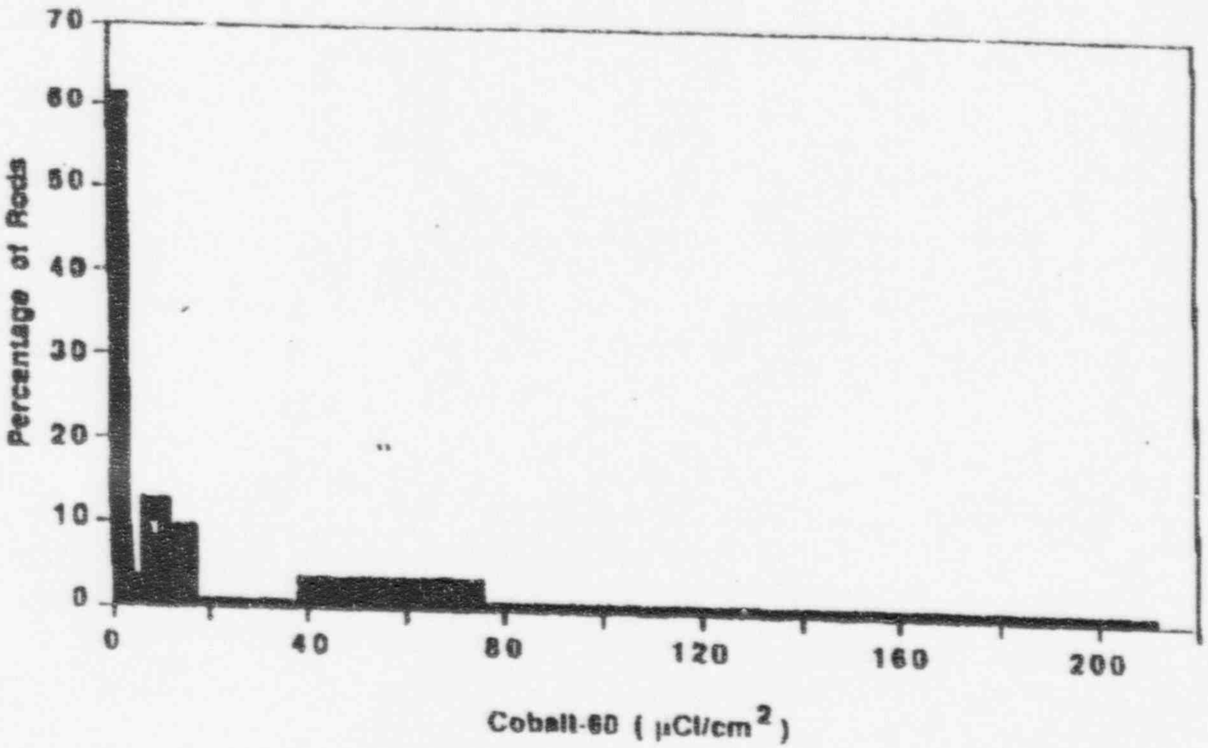

Figure 3. Distribution of PWR Spent-Fuel Rods as a Function of the Maximum "Spot" CRUD Activity at Fuel Discharge

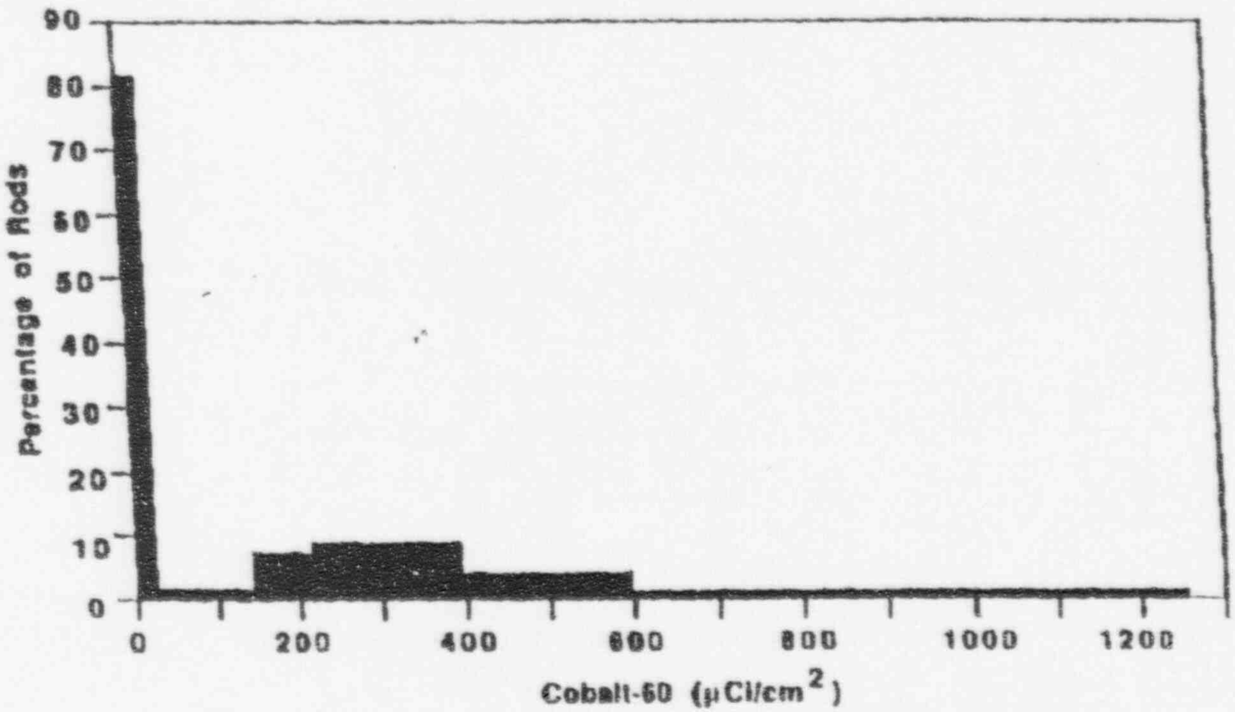

Figure 4. Distribution of BWR Spent-Fuel Rods as a Function of the Maximum "Spot" CRUD Activity at Fuel Discharge

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446°F

Table 4 lists a number of transportation conditions that cover the range of thermal and mechanical stresses prescribed in 10 CFR 71. In the first two thermal cases, estimates of the spallation fractions can be made from existing data [EC84, OL84]. Although the tests by Einziger and Cook [EC84] and Olsen [OL84] were at only 230°C, they should be applicable to a 300°C surface temperature, because surface dryout due to evaporation should have occurred in both cases, and no significant cladding creep will occur during the relatively short period of transportation [EI84b]. Although spallation fractions for these tests are quoted at 6% for PWR rods and 5% for BWR rods, when uncertainties are considered, the upper limit of the spallation fractions in both cases appears to be about 8%. The similarity between spallation fractions observed during these tests for PWR and BWR fuels may be attributable to spallation of the loose flocculent CRUD on the BWR fuel rods during handling in the reactor pool prior to testing.

572°F

842°F

At the higher temperature of 450°C, cladding ballooning might occur at incipient crack sites and cause additional spallation [ST86]. Tests by Stahl et al. indicated that when ballooning does occur, deformation is localized in an approximately 0.15-m-long region [ST86]. If 100% CRUD spallation in the 0.15-m-long region is assumed, and 8% spallation on the remainder of the rod, a total spallation between 12 and 15% would be expected for PWR rods. Less spallation can be expected for BWR rods because their lower internal gas pressure leads to a lower cladding stress, and hence ballooning, than in PWR fuel rods.

The spallation fraction cannot be estimated in the other four cases listed in Table 4 because of insufficient data.

4.4 Particle Size Distribution

The ability of CRUD to transport through the cask fill-gas is a strong function of particle size. CRUD particle size may depend on whether the CRUD is flocculent or adherent, and on whether it is associated with PWR or BWR fuel. For instance, the thick, porous CRUD layer on the rods from the Dodewaard BWR reactor was observed to be quite loose, with poor adherence, and could easily be brushed away as powder [BD83]. On the other hand, CRUD flakes that had been scraped from the Oconee 1 Cycle 1 PWR fuel rods [BA79] were agglomerates of smaller particles and were as large as 2 mm^2. Quantitative data on CRUD particle size are summarized in Table 5.

When CRUD was gathered from underwater sampling, it was deposited on a 0.45-μm filter paper. The particulate size was determined in a few samples by scanning electron microscopic (SEM) examination of the filter paper. The fluffy CRUD on a Brunswick-2 BWR fuel rod was composed of amorphous particles ranging from 0.1 μm to 0.3 μm in diameter, and irregular-shaped particles with well-defined faces ranging from 1 μm to 3 μm in diameter [AN82]. The tenacious CRUD was similar in shape to the fluffy CRUD but consisted of agglomerates of 0.1 μm to 0.3 μm diameter primary particles. CRUD from the Monticello BWR fuel rods contained loose clusters of 0.1 μm to 0.3 μm diameter particles and larger particles ranging from 0.5 μm to 2.0 μm [AN82]. A SEM examination of CRUD from a Nine Mile Point BWR fuel rod indicated agglomerates of 0.1 μm diameter primary particles [ST85]. These CRUD samples were all taken underwater using a scraping stone or brush. This method of obtaining the samples may have affected the

6.0 CALCULATIONS

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6.1 ACTIVITY CALCULATION

Cobalt[60] Surface Contamination

Cobalt[60] is found in a reactors primary system and in its spent fuel pool. As such, it coats the outside of each fuel assembly and control components with a substance called crud. While this crud contains mostly nonradioactive components, it can contain appreciable amounts of Cobalt[60]. From NUREG/CR-3285, crud deposits a maximum 140 $\mu Ci/cm^2$ of Cobalt[60] on fuel assemblies and control components.

*Fuel**

$$A_{Sur}^{Ass} = \pi \times D_{out} \times L_{rod} = \pi \times 0.374\text{inches} \times 160.0\text{inches} = 188\,\frac{\text{inches}^2}{\text{rod}}$$

* Conservatively assume the rods extent the total length of the assembly to to calculate the end fitting surface area.

$$A_{Sur}^{Ass} = 188\,\frac{\text{inches}^2}{\text{rod}} \times (17X17)\frac{\text{rods}}{\text{assy}} \times 24\,\frac{\text{assy}}{\text{Basket}} = 1.30 \times 10^6\,\text{inches}^2 = 8.4 \times 10^6\,\text{cm}^2$$

* It is conservative to assume 17X17 rods compared to the actual number of 264

$$Act_{Sur}^{Ass} = 140\,\frac{\mu Ci_{Co^{60}}}{cm^2}(8.4 \times 10^6\,cm^2) = 1176\,\frac{Ci_{Co^{60}}}{\text{Basket}}$$

Reactor Control Components

$$A_{Sur}^{RCC} = (\pi \times D_{RCC} \times L_{RCC} \times \#rods) + (\pi \times D_{Base} \times L_{Base})$$

$$A_{Sur}^{RCC} = (\pi \times (0.385in) \times 150.574in \times 36) + (\pi \times (1.840in) \times 10.375in) = 6616.31in^2 = 42685.8cm^2$$

$$A_{Sur\ Total}^{RCC} = A_{Sur}^{RCC} \times 1.1 = 42685.8cm^2 \times 1.1 = 4.7 \times 10^4 cm^2$$

* A factor of 10% is conservatively used to estimate the surface area of the RCC's odd geometry's.

$$Act_{Sur\ Total}^{RCC} = 140 \frac{\mu Ci_{Co^{60}}}{cm^2} \times 4.7 \times 10^4 \frac{cm^2}{Basket} \times 24RCC = 158 \frac{Ci_{Co^{60}}}{Basket}$$

The total surface activity.

$$Act_{Sur}^{Total} = Act_{Sur\ Total}^{RCC} + Act_{Sur}^{Ass} = 158 \frac{Ci_{Co^{60}}}{Basket} + 1.176 \frac{Ci_{Co^{60}}}{Basket} = 1.334 \frac{Ci_{Co^{60}}}{Basket}$$

Using the exponential law of decay, shown below, the Cobalt^{60} activity is decayed 5 years.

$$N(t) = N_0 e^{-\lambda t}$$

Where

$T_{1/2}$ is the isotopes half life

λ is a unique decay constant for the isotope

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5.6 Passive Failure of Basket Pressure Test Line

5.6.1 The Basket free gas volume at 7 psig is converted to an equivalent volume at standard conditions:

$V_{free} = 359{,}716\ in^3$

$$V_{stp} = \frac{(359{,}716\ in^3)(14.7psia + 7psi)}{14.7psia} = 531{,}009\ in^3$$

$V_{release} = (V_{free}) - (V_{stp}) = 171{,}293\ in^3 = 99.1\ scf$

5.6.2 Airborne Particulate Concentration:

A_{crud} = 690.2 Ci Co^{60} per basket [Ref 6]

$A_{part.}$ = $(0.15)A_{crud}$ = 103.5 Ci Co^{60} [15% goes airborne]

$C_{part.}$ = $(103.5\ Ci)/(531{,}009\ in^3) = 1.949 \times 10^{-4}\ Ci/in^3$
= 11.89 µCi/cc

5.6.3 Total curies released to fuel building atmosphere:

$A_{release} = (1.949 \times 10^{-4}\ Ci/in^3)(171{,}293\ in^3) = 33.39\ Ci\ Co^{60}$

5.7 Basket Shield Lid Drop onto Basket During Placement

5.7.1 For a flat drop orientation, the analysis is presented in Appendix C.

5.7.2 For the edge drop orientation, the maximum number of impacted fuel assemblies:

Shield Lid OD = 64.10 in [Ref. 7]

Shield Lid thickness = 8 in [Ref. 7]

Distance to top of sleeve assemblies = 161 in [Ref. 7]

Length of fuel assembly with RCCA = 168 in [Ref. 10.11]

Distance from top of Sleeve assemblies to top of RCCA = 168-161 = 7 in

ATTACHMENT VIII

Pages 4-4, 4-5, and 4-16 of "Topical Report Seismic Analyses of Structures and Equipment for Nuclear Power Plants," BC-TOP-4 Rev 2. 1974, Bechtel Power Corporation

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BC-TOP-4
Revision 2
June 1974

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TROJAN NUCLEAR PLANT

TOPICAL REPORT

SEISMIC ANALYSES OF STRUCTURES AND EQUIPMENT FOR NUCLEAR POWER PLANTS

Bechtel Power Corporation
San Francisco, California

BECHTEL

Two consecutive modes are defined as closely spaced if their frequencies differ from each other by less than 10 percent. For modes that are not closely spaced, the criterion of "the square root of the sum of the squares" is used. When modes are closely spaced, they are first divided into groups in such a way that, in each group, the deviation in frequency between the first and the last mode does not exceed 10 percent of the lower frequency. The criterion of "the sum of absolute values" is then applied to each group, and the results from all the groups are then combined according to the criterion of "the square root of the sum of the squares".

Because of the nature of the design spectra and because most structures have a fundamental frequency within the frequency range of maximum spectral response, which is 2 to 7 cps, the effect on the response of these structures due to the possible variation in structural or foundation material properties would be negligible.

b) <u>Time History Analysis</u> - Given the ground motion time history as input, the modal equations, Eq. (4-3), are first solved for each mode, and then the modal responses are superimposed according to Eq. (4-1) to obtain the total response.

4.2.2 <u>Method of Direct Integration</u>

Equation (3-1) is directly integrated by accetable numerical schemes when this equation cannot be decoupled. For this case, the input is the time history motion.

4.3 <u>Total Structural Response From Separate Lateral And Vertical Analyses</u>

The total structural response is predicted by combining the applicable collinear responses, say, R_x, R_y and R_z, 2

calculated respectively from the two lateral and the vertical analyses. The combination is done according to the criterion of "the square root of the sum of the squares" as follows:

$$R_{total} = \sqrt{R_x^2 + R_y^2 + R_z^2} \qquad (4\text{-}7)$$

4.4 Structural Overturning And Soil Pressure

4.4.1 Structural Overturning

When the combined effect of earthquake ground motion and structural response is strong enough, the structure will undergo a rocking motion pivoting about either edge of the base. When the amplitude of rocking motion becomes so large that the center of structural mass reaches a position right above either edge of the base, the structure becomes unstable and may tip over (see structural position (b) as indicated by dotted lines in Fig. 4-1). The mechanism of such rocking motion is that of an inverted pendulum, and its natural period is very long compared with that of the linear, elastic structural response. Hence, so far as overturning evaluation is concerned, the structure can be treated as a rigid body.

If the center of mass of the structure experiences a maximum total lateral velocity v_H and a vertical velocity v_V, the maximum kinetic energy is conservatively estimated to be:

$$E_S = m_o(v_H^2 + v_V^2)/2 \qquad (4\text{-}8)$$

in which m_o is the total mass of the structure and foundation. If a spectral response analysis is done for the structure, the total lateral velocity v_H is computed as follows:

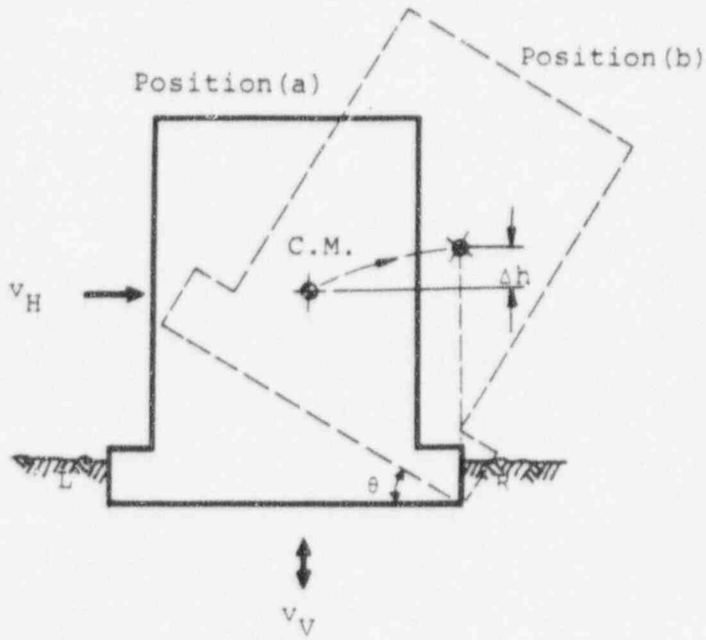

Fig. 4-1 Position of the Structure when Overturning about One Edge

ATTACHMENT IX

Sections 2.1.1.2 and 2.1.2.1 of the Trojan Defueled Safety Analysis Report

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2.1 GEOGRAPHY AND DEMOGRAPHY

The Trojan site was originally selected to minimize hazards to the general public. The site environs have low population densities and minimal usage for such activities as farming and recreation. Some of the site characteristics associated with the Trojan site selection for the operational phase of the facility remain applicable to defueled condition and the storage of irradiated fuel. This chapter provides discussion of those site characteristics applicable to defueled operation.

2.1.1 SITE LOCATION AND DESCRIPTION

2.1.1.1 Specification of Location

The Trojan Plant site is in Columbia County, Oregon, and lies along the bank of the Columbia River at approximately River Mile 72.5, 42 miles north of Portland. The specific geographic location of the site is 46° 02' 25" N latitude and 122° 53' 03" W longitude. In the Universal Transverse Mercator coordinate system, the site location is 5098352 meters N by 509000 meters E, and in the Oregon North Zone Lambert Coordinate, the site location is 874375 N by 1394615 E.

The nearest incorporated communities are Rainier, Oregon, approximately 4-1/2 miles northwest; and across the Columbia River, Kalama, 3 miles southeast, and Longview, approximately 6 miles northeast. Within a 5-mile radius of the site are three small unincorporated communities with a total population of less than 2000: Prescott, Oregon, 1/2-mile north; Goble, Oregon, 1-1/2 miles southeast; and Carrolls, Washington, 2-1/2 miles northeast.

Other than the Columbia River and tributaries, there are no nearby natural geographic features of prominence offsite. The Kalama River joins the Columbia at River Mile 73.1, about 1/2-mile upstream on the bank opposite the site. Similarly, the confluence of

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the Cowlitz and Columbia Rivers is about 4-1/2 miles downstream at River Mile 68. Onsite, however, are the 499 feet natural draft cooling tower, which rises 589 feet above mean sea level (MSL), an approximately 26-acre man-made reflecting lake and an approximately 28-acre recreational lake.

2.1.1.2 Site Area Map

The Trojan Nuclear Plant site is an approximately 635-acre tract of land owned in fee by Portland General Electric Company (PGE) in Sec. 35 and 36, T. 7 N., R. 2 WWM, and in Sec. 1 and 2, T. 6 N., R. 2 WWM, Columbia County, Oregon. The tract is all-inclusive of individual and separate parcels as described in the following deed records on file in Columbia County: BK 168, Pages 13 and 14, 22, 23 to 26 inclusive, 81 to 83 inclusive, 117 to 121 inclusive; BK 171, Pages 935 and 936; and BK 174, Page 436.

The exclusion area, is defined in 10 CFR 100.3(a). The exclusion area boundary coincides with the site boundary on the Oregon side of the river and extends across the Columbia River to the east where the Washington shore of the river forms the eastern boundary.

The major physical facilities are grouped approximately in the geographic center of the exclusion area. The center of the containment lies 2172 feet due south of the nearest point on the exclusion area perimeter, approximately 3175 feet from the nearest point to the west, and approximately 4400 feet from the closest southerly point. Because of its irregular shape, the exclusion area comes within about 4200 feet of the containment at two points approximately southwest of the building. The containment center is located about 2200 feet from the nearest point of approach of the eastern boundary of the exclusion area, and 400 feet from the Oregon bank (mean low water) of the Columbia River.

2.1.1.3 Boundaries for Establishing Effluent Release Limits

As pointed out in the preceding section, the site boundary and exclusion area coincide on land on the Oregon side of the Columbia River.

Programmatic requirements for specifications applying to releases of radioactive material in gaseous effluents are given in the Offsite Dose Calculation Manual. Doses and release limits have been evaluated at the site boundary and at off-site locations of actual exposure.

2.1.2 EXCLUSION AREA AUTHORITY AND CONTROL

2.1.2.1 Authority

The site boundary (owned in fee) extends to mean low water in the southern part of the site and to mean high water in the northern part of the site. By written agreement with the State of Oregon, who is owner of the submerged and remainder of submersible lands in the river at the site, PGE has control of the uses of such areas out to a line at approximately -20 feet MSL. Beyond this line the U. S. Coast Guard has jurisdiction over river operations.

The provisions of the tidelands agreement with the State of Oregon include the following conditions:

(1) RESTRICTED USE: Residential use of and overnight camping in the exclusion area shall be prohibited by the State. Nonresidential activities and uses unrelated to the operation of the Company's adjacent reactor shall be permitted only when no significant hazard to public health and safety exists and then only under appropriate limitations as provided in this Agreement.

(2) USE LIMITATIONS: In managing its lands within the above-described exclusion area and in sales and leases made with respect to such lands, the State will insert in each Deed, Lease, Easement, Permit or other instrument granted a provision to the effect that the lands affected are within an exclusion area as that term is defined by the Nuclear Regulatory Commission; that such lands are subject to safety regulations established by the State, the Nuclear Regulatory Commission, and the Company with which the grantee shall comply; and that the Company has a right to remove or order the removal of all persons and their property therefrom in compliance with said safety regulations.

(3) EMERGENCY PROCEDURES: In the event of an emergency or threat thereof, which may affect public health or safety, the State grants the Company the right to enter upon its lands within the exclusion area and to remove persons and property therefrom. The State also grants the Company the right to proclaim safety regulations affecting persons and property occupying State owned lands within the exclusion area, which regulations upon approval by the State shall be made by the State a part of every Deed, Lease, Easement, Permit or other instrument issued by it as previously provided. Such rules and regulations when so adopted shall also become a part of the State's management policy for the administration of such lands.

Similar conditions have been negotiated with the Burlington Northern Railroad (1978). Mineral rights not part of the original land purchase at Trojan have been subsequently bought in fee by PGE.

2.1.2.2 Exclusion Area Activities Unrelated to Plant Operation

The basic reference for activities and facilities within the exclusion area is "The Trojan Nuclear Power Plant: Master Plan Prepared by Lawrence Halprin & Associates". In this