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SHOREHAM NUCLEAR POWER STATION

RADIOLOGICAL SAFETY ANALYSIS FOR SPENT FUEL STORAGE AND HANDLING

NUCLEAR ENGINEERING DEPARTMENT
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RADIOLOGICAL SAFETY ANALYSIS FOR SPENT FUEL STORAGE AND HANDLING

I. EXECUTIVE SUMMARY

The purpose of this report is to provide a radiological safety analysis for the storage and handling of Shoreham's low burnup first cycle spent fuel.

This analysis report is based on the fact that the 560 fuel bundles comprising the Shoreham core are stored under water in the Shoreham spent fuel pool. The fuel bundles are held in a Seismic Category I spent fuel rack within the stainless steel lined spent fuel pool. The spent fuel pool is located in the secondary containment of the Shoreham reactor building. The structures are designed to withstand seismic loads.

It is important to understand that the Shoreham spent fuel is in a low burnup condition. The Shoreham Nuclear Power Station operated during low power testing at power levels not exceeding 5% of rated power. The effective burnup of the fuel is approximately 2 full power days. This results in an estimated total core-wide heat generation rate of approximately 550 watts as of June 1989. The estimated fuel heat load will reduce to approximately 250 watts by June 1991. Figure I-1 depicts the fuel heat load versus time. Based on this low heat generation rate systems for active cooling are not required, and only minimal capacity systems are required for pool makeup to handle evaporation.

The Shoreham spent fuel contains limited quantities of radioactive materials that are available for release. It has been calculated that approximately 176,000 curies of radioactivity reside in the 560 fuel assemblies. The radioactive inventory estimation is based on a two year decay after the last burnup period. The total gaseous activity is primarily Krypton-85 (a noble gas with a half life 10.7 years) and consists of approximately 1560 curies. Krypton-85 is the only isotope in the fuel that exists in significant quantities and is available for release in gaseous form during postulated accidents. Other sources of radioactivity outside the core are minor.

A spectrum of accidents were identified for radiological analysis. The accidents were identified by reviewing the Shoreham USAR for those events that apply to the storage and handling of spent fuel. Based on this review, the following events were identified for analysis:

1. Fuel Handling Accident (Fuel Bundle Drop)
2. Radwaste Tank Rupture

In addition, a worst case radiological event was postulated in which the entire gaseous activity of the whole core is released to the reactor building. This event was postulated to conservatively bound any possible situation involving large-scale mechanical damage to the fuel.

The results of the radiological analysis indicate that integrated doses are very small in comparison with 10CFR100 limits. The results of the radiological analysis indicate that integrated doses are very small in comparison with 10CFR100 limits. For the fuel handling accident and the worst case scenario, a spectrum of cases was analyzed, as follows: operation of the standby ventilation system, operation of the normal ventilation system, and no ventilation (modelled as a puff release). The results of the fuel handling accident analyses indicate that the integrated offsite whole body and skin doses, with the Reactor Building Normal Ventilation System operational, are approximately 0.00005% or less of 10CFR100 limits. For the worst case scenario, under the same HVAC conditions, the doses are approximately 0.03% or less of 10CFR100 limits. The results of the radiological analyses are depicted graphically in Figures I-2A and I-2B, for the fuel handling accident and worst case scenario, respectively. In particular, it was demonstrated that the reactor building standby ventilation system operation does not provide an important filtering or ventilation safety function and is therefore no longer required now that fuel is located in the pool.

Based on this analysis, it has been found that the spent fuel pool provides a high degree of passive safety protection for Shoreham spent fuel. Active safety systems are not required to mitigate postulated accidents; however, support systems are required to meet the intent of the requirements of 10CFR50 Appendix A, General Design Criteria; and Regulatory Guide 1.13. Supporting systems are required to provide for radiation monitoring, fuel pool makeup, fuel pool cleanup, radwaste, and normal support systems to maintain building services.

FIGURE I – 1

SNPS Spent Fuel Decay Heat Load

FIGURE I-2A

Design Basis Fuel Handling Accident
Exclusion Area Boundary Results
RBNVS HVAC System in Operation

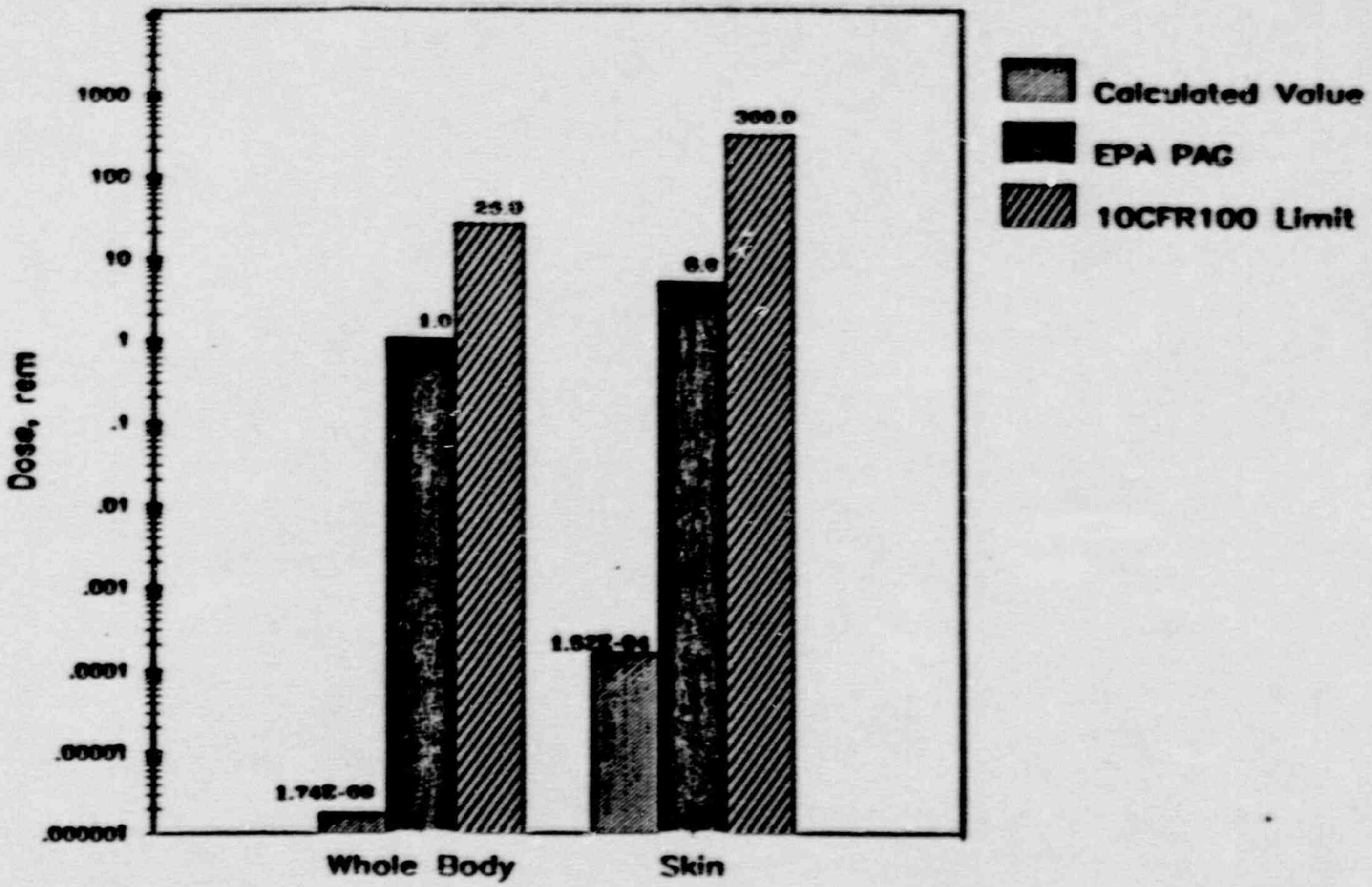
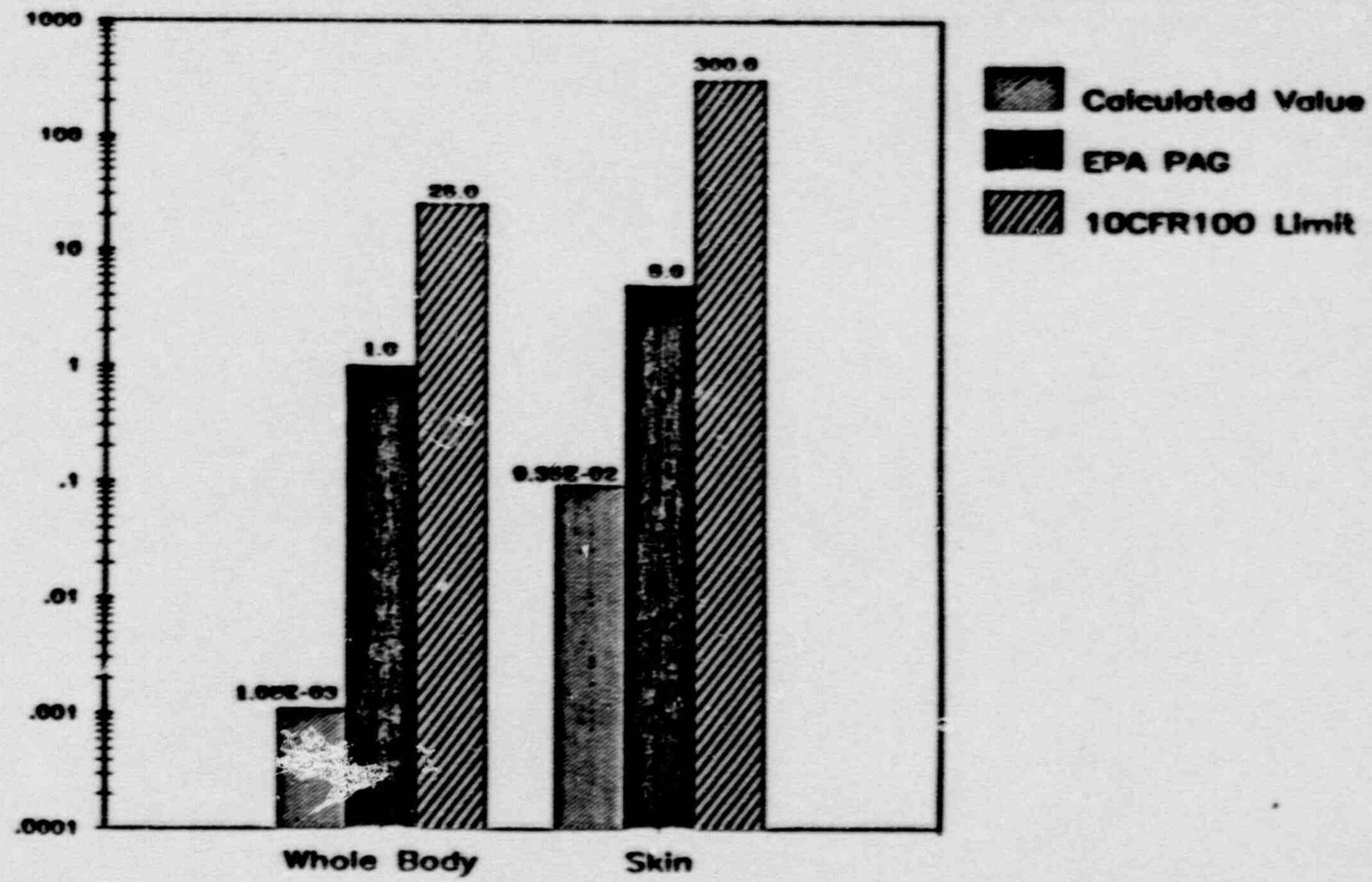


FIGURE I-2B

Worst Case Fuel Damage Accident
Exclusion Area Boundary Results
RBNVS HVAC System in Operation



II. <u>PLANT CONFIGURATION SUMMARY</u>

This analysis is based on the fact that the Shoreham initial core spent fuel will be stored for some interim period in the spent fuel pool contained within the SNPS reactor building.

The configuration of the plant is summarized as follows:

1. All 560 fuel bundles have been removed from the reactor and are being stored in the spent fuel storage pool. The total decay heat power of the entire core has been determined to be approximately 550 watts as of June 1989.

2. The spent fuel storage pool water level will be maintained at its normal water level. Makeup will be furnished from the condensate transfer system or the demineralized and makeup water system. The fuel pool cooling system is not in service due to the low heat load in the pool. Water quality will be maintained by the fuel pool cleanup system. The spent fuel pool transfer canal gates will remain installed.

3. The steam separator and dryer has been placed back in the reactor and the reactor vessel head has been placed on the reactor flange, but the studs have not been tensioned.

4. The drywell head has been re-installed and the reactor cavity and dryer/separator pools have been drained.

5. All reactor protection, nuclear steam supply shutoff and emergency core cooling systems are to be de-energized and isolated.

6. The reactor building normal ventilation system will be operated to provide suitable environmental conditions and to allow for radiological monitoring of building releases.

7. Radwaste systems will be maintained as required by the above.

III. SAFETY ANALYSIS

A. Radioactive Inventory

1. Fuel Sources

The Shoreham reactor core has undergone three periods of low power (0-5%) testing over the past four years. The low power tests are summarized below:

Test Period	Duration	Specific Burnup MWD/MT	Power Range %
7/7-10/7/85	93 days	27.8	0.0 - 3.3
8/5-8/30/86	26 days	13.8	0.0 - 4.0
5/26-6/6/87	12 days	6.7	0.0 - 3.5
	Total	48.3	

The detailed profiles of the above three low power test periods have been input to the ORIGEN2 (Reference A) burnup code, along with the physical characteristics of the reactor fuel and bundle structural elements. Results of this analysis (Reference B) are given in Table III.A - 1. The activities correspond to two years decay after the last burnup period, and reflect total core inventories for those isotopes with greater than 10 curies.

As can be seen from the Table III.A-1 only long-lived isotopes remain from the original actinides and fission/activation products created, along with their equilibrium daughters. By far the most radiologically significant, from a gamma dose rate standpoint, are the Cs-137/Ba-137m pair; about 80% of the whole body dose rate from a spent fuel bundle is due to the Ba-137m photon (Reference C). For dose assessment of accidental gaseous releases (e.g., a postulated fuel handling accident), only Kr-85 is meaningful (Reference E).

2. Non-Fuel Sources

a. Liquid Sources

With the possible exception of liquid radwaste streams, reactor water would be expected to have the highest concentration of radionuclides of any liquid stream in the plant. At SNPS as of June 1989, the concentration of all radionuclides in reactor water is less than the lower limit of detection (LLD), per Reference F. It is concluded that the liquid streams outside the radwaste system are insignificant sources of radioactivity.

b. Gaseous Sources

There are no detectable gaseous sources at SNPS, either present or anticipated. This statement is supported by the fact that the most recent Semi-Annual Radiological Effluent Release Report (Reference D) indicates there were no detectable releases during the six-month period, either from the offgas system or the various building exhaust systems.

c. Radwaste Sources

With the exception of the low burnup fuel, radwaste is the only area of the plant with measurable activity. The maximum whole body gamma dose rate in the plant is about 3.5 mrem/hr, near the Spent Resin Tank.

Current (6/30/89) isotopic concentrations above LLD levels in the Radwaste System are indicated in Table III.A - 2, from Reference G, H and I.

d. Activated Materials Sources

There are no significant out-of-core radioactive materials sources activated at SNPS. While the low power testing program may have activated some materials inside the RPV, these are not considered significant compared to spent fuel sources.

TABLE III.A - 1

Fuel Source Terms

ISOTOPE	CURIES	HALF-LIFE
H-3	1.77E+02	1.23E+01 years
Mn-54	3.36E+01	3.13E+02 days
Fe-55	8.06E+02	2.70E+00 years
Co-60	5.64E+02	5.27E+00 years
Ni-63	4.28E+01	1.00E+02 years
Kr-85	1.56E+03	1.07E+01 years
Sr-89	1.54E+01	5.05E+01 days
Sr-90	1.37E+04	2.86E+01 years
Y-90	1.37E+04	6.41E+01 hours
Y-91	6.81E+01	5.85E+01 days
Zr-95	1.48E+02	6.40E+01 days
Nb-95	3.49E+02	3.51E+01 days
Ru-106	5.98E+03	3.68E+02 days
Rh-106	5.98E+03	2.99E+01 seconds
Sn-119m	3.30E+02	2.93E+02 days
Sb-125	1.45E+03	2.77E+00 years
Te-125m	3.53E+02	5.80E+01 days
Te-127	1.49E+01	9.35E+00 hours
Te-127m	1.52E+01	1.09E+02 days
Cs-134	1.33E+02	2.06E+00 years
Cs-137	1.48E+04	3.02E+01 years
Ba-137m	1.40E+04	2.55E+00 minutes
Ce-144	3.55E+04	2.84E+02 days
Pr-144	3.55E+04	1.73E+01 minutes
Pr-144m	4.26E+02	7.20E+00 minutes
Pm-147	2.95E+04	2.62E+00 years
Sm-151	3.60E+02	9.00E+01 years
Eu-154	1.18E+01	8.80E+00 years
Eu-155	4.47E+01	4.96E+00 years
U-234	1.02E+02	2.45E+05 years
Th-234	3.38E+01	2.41E+01 days
Pa-234m	3.38E+01	1.17E+00 minutes
U-238	3.38E+01	4.47E+09 years
Pu-239	2.77E+02	2.41E+04 years
Pu-241	5.58E+01	1.44E+01 years
Total	1.76E+05	

Note: Only isotopes with activity greater than 10 curies are listed.

TABLE III.A - 2

Radwaste Sources Greater than LLD

Spent Resin Tank, Radwaste Filter, & Floor Drain Filter

The activity concentration is assumed to equal the maximum in the most recent HIC shipment (Nov-Dec 1988) is (from Reference G):

Isotope	Activity Concentration, uCi/cc	% of Activity
Cr-51	9.84E-04	58.46%
Mn-54	2.17E-05	1.29%
Fe-55	4.19E-04	24.88%
Co-57	7.92E-07	0.05%
Co-58	6.43E-06	0.38%
Co-60	1.09E-04	6.51%
Fe-59	4.57E-05	2.71%
Ni-63	6.41E-06	0.38%
Sb-124	3.25E-06	0.19%
Zn-65	1.89E-05	1.12%
H-3	6.21E-06	0.37%
C-14	3.94E-07	0.02%
Sr-90	1.69E-07	0.01%
Zr-95	1.52E-05	0.91%
Nb-95	2.55E-05	1.51%
Tc-99	4.79E-09	0.00%
I-129	7.32E-10	0.00%
Cs-137	1.34E-06	0.08%
Ce-144	2.95E-06	0.18%
Pu-241	1.59E-05	0.95%

Discharge Supply Tanks

The activity concentration in these tanks is assumed to equal the maximum concentration measured in the past 12 months (from Ref. H):

Isotope	Activity Concentration, uCi/cc	% of Activity
Co-60	7.83E-08	100.0%

Note: The remaining radwaste tanks (floor drain collector tanks, waste collector tanks, and recovery sample tanks) were all determined in Reference I to have isotopic concentrations less than LLD.

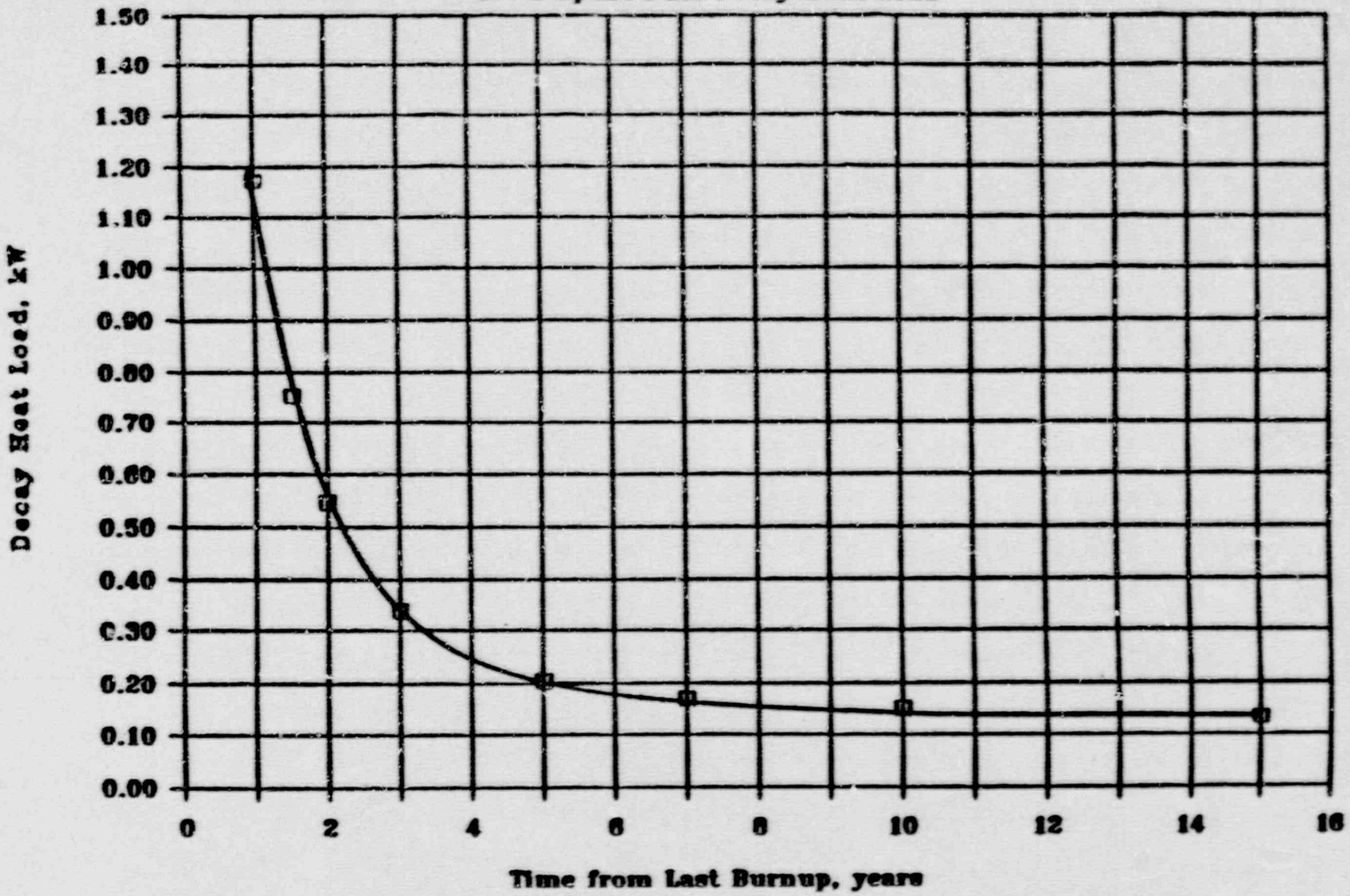
III.B HEAT GENERATION ANALYSIS

One result from the ORIGEN2 calculation is a tabulation of decay heat or thermal power (in watts), as a function of time. Results of this analysis are presented in Figure III.B-1. The calculated decay heat load as of June 1989 is approximately 0.55 kw.

It must be recognized that there are some limitations in the ORIGEN2 model, and potential inaccuracies in the calculational processes of the code and its supporting data sets. For instance, ORIGEN2 is a "point reactor" model, and cannot deal conveniently with the spatial variations in fuel enrichment and burnup. In addition, there are uncertainities associated with averaging of nuclear cross-section data within the thermal, resonance, and fission neutron energy ranges.

FIGURE III.B- 1

SNPS Spent Fuel Decay Heat Load



III.C EVALUATION OF SPENT FUEL POOL COOLING REQUIREMENTS

An analysis (Reference K) was performed to determine the rate of water loss from the spent fuel pool through evaporation under the worst case scenario described below. The time it took to uncover the spent fuel based on the calculated evaporation rate was then determined. The following assumptions are used in the analysis to maximize the calculated pool evaporation rate and hence minimize the time required to uncover the Shoreham low burnup spent fuel:

1) The spent fuel pool temperature is conservatively kept at 110°F.

2) The ambient temperature above the spent fuel pool is conservatively assumed with zero relative humidity.

3) The reactor building air flow exists due to normal ventilation system operation to maximize evaporation.

The result of the calculation shows that the maximum evaporation rate from the pool is approximately 0.6 gpm which translates to a pool level depletion rate of one foot per eleven days. Technical Specifications require that the water level above the spent fuel be at least twenty-one feet. In addition, it should be noted that pool water level is alarmed in the control room and alarm response procedures exist to provide appropriate operator action.

III.D FUEL CRITICALITY ANALYSIS

The Shoreham Spent Fuel Rack (SFR) is of a stainless steel and water neutron flux trap design which uses no additional poison. The criticality analysis of this rack design is described in detail in Appendix 9A of the Shoreham USAR. The reactivity results which are summarized in USAR Table 9A-4 of the same document remain valid for the conditions existing at Shoreham after defueling. Furthermore, due to the differences in U-235 enrichment between the designed and the current fuel in the core, a large negative reactivity credit should be taken into account. This is explained as follows:

> The Shoreham SFR design is based on a maximum U-235 enrichment of 3.1 w/o. The resulting basic cell k is calculated to be 0.9129 without any uncertainty and model adjustments (Table 9A-4, Appendix 9A, Shoreham USAR). The Shoreham Cycle 1 fuel loading uses three (3) enrichments. Of the 560 fuel assemblies in the core, 340 bundles have the highest bundle average U-235 enrichment of 2.19 w/o, 144 bundles of 1.76 w/o and 76 remaining bundles use natural uranium.
>
> If the six inch natural uranium segments at the top and bottom of the fuel are excluded, the average enrichment of a 2.19 w/o bundle becomes 2.33 w/o. Using this enrichment and linearly extrapolating the reactivity vs. U-235 enrichment results given in Figure 9A-5 of Appendix 9A, Shoreham USAR, the reactivity difference between the design enrichment of 3.1 w/o and the current maximum loading enrichment of 2.33 w/o is found to be about -6.0% in Δk (Δk = -0.060). This brings the basic cell k_∞ under nominal storage conditions for the current fuel in the core down to ~0.85, which is well below the regulatory acceptance criterion of $k_\infty \leq 0.95$. All the corrective and uncertainty adjustments listed in Table 9A-4 of the Shoreham USAR remain applicable.

During the period from July, 1985 to June, 1987, Shoreham went through three separate stages of low power testing (less than 5% of rated power), which resulted in a total core exposure of approximately 48 MWd/MT as determined by a series of core-follow analyses. The net effect of the core exposure is a slight decrease in reactivity (-0.002 in Δk) mainly due to the offsetting contributions from the formation of Sm-149 and the slight depletion of the burnable Gd poison in the fuel bundles. In light of the large reactivity margin described previously ($k_{\infty} \sim 0.85$), no additional credit will be claimed here.

III.E.1 OVERVIEW OF USAR CHAPTER 15 EVENTS

Introduction

Chapter 15 of SNPS USAR provides the results of analyses of the spectrum of transient and accident events which are postulated to occur with the plant operating initially at maximum power. The purpose of this analysis is to identify USAR transients and accidents that apply to the storage and handling of the low burnup fuel.

The analysis is based on the fact that the fuel is removed from the core and is stored in the spent fuel pool. The total decay heat is approximately 550 watts, which is small enough that it could be removed by passive cooling and would not require the fuel pool cooling system. Normal and emergency makeups are available.

The design basis of the spent fuel storage excludes fuel uncovery under any postulated loss of coolant (Reference Section II.D, Regulatory Position 6).

As the reactor is not operating and the fuel is not in the core, most of the USAR Chapter 15 events cannot occur.

Analysis

The safety parameter which is evaluated for each transient of USAR Chapter 15 is Minimum Critical Power Ratio (MCPR) which is a measure of the fuel cladding integrity. Maximum Average Planar Linear Heat Generation Rate (MAPLHGR) is the safety parameter for the reactor LOCA-related accidents, and indicates whether the cladding temperature and the zirconium-water reaction is below the specified limits. As the decay power level is extremely low during spent fuel storage, MCPR and MAPLHGR are of no concern. Most of the transients and accidents of USAR Chapter 15 occur at operating conditions and are therefore not applicable.

Those transients and accidents of USAR Chapter 15 which pose radiological release outside the primary containment barrier are of primary concern.

The USAR Chapter 15 events are assigned to one of six analytical categories. The next section presents those analytical categories and discusses all events one-by-one in each analytical category.

Decrease in Core Coolant Temperature

This analytical category of USAR Chapter 15 events includes the following events:

- 15.1.5 Pressure Regulator Failure - Open
- 15.1.7 Feedwater Controller Failure - Maximum Demand
- 15.1.8 Loss of Feedwater Heating
- 15.1.9 Shutdown Cooling (RHR) Malfunction - Decreasing Temperature.

In the spent fuel storage condition, the pressure regulator, feedwater controller, feedwater heating system and RHR system are not operating and all four transients are, therefore, not applicable.

Increase in Reactor Pressure

Since generator, turbine, main steam isolation valve, pressure regulator, feedwater system, condenser and RHR systems are not operating, the following transients are not applicable:

- 15.1.1 Generator Load Reduction
- 15.1.2 Turbine Trip
- 15.1.3 Turbine Trip with Failure of Generator Breakers to Open
- 15.1.4 Main Steam Isolation Valve Closure
- 15.1.6 Pressure Regulator Failure - Closed
- 15.1.18 Loss of Feedwater Flow
- 15.1.21 Loss of Condenser Vacuum
- 15.1.26 Core Coolant Temperature Increase

The transient of this category applicable to spent fuel storage is the following:

15.1.19 Loss of AC Power

A loss of AC power condition can be postulated that will affect normal support systems. However, because of the very low heat generation rate (550 watts) and large thermal capacity of the pool, loss of normal cooling and makeup systems will result only in a very slow evaporation of the pool water. This evaporation rate is so slow that ample time exists to restore normal pool makeup sources so that pool level can be quickly restored. Thus, the passive protection provided by the spent fuel pool eliminates the need for active cooling requirements. The rate of evaporation is discussed in Section III.C.

Decrease in Reactor Coolant Flow Rate

Recirculation pump and recirculation flow controller are not operating and therefore all the transients of this category are not applicable:

- 15.1.20 Recirculation Pump Trip
- 15.1.22 Recirculation Pump Seizure
- 15.1.23 Recirculation Flow Control Failure - Decreasing Flow

Reactivity and Power Distribution Anomalies

Events included in this category are those which cause rapid increase in power. Since the reactor is not fueled, the following events are not applicable:

- 15.1.11 Continuous Control Rod Withdrawal during Power Range Operation
- 15.1.12 Continuous Control Rod Withdrawal during Reactor Startup
- 15.1.13 Control Rod Removal Error during Refueling
- 15.1.14 Fuel Assembly Insertion Error during Refueling
- 15.1.15 Off-Design Operational Transient due to Inadvertent Loading of a Fuel Assembly into an Improper Location
- 15.1.16 Inadvertent Loading and Operation of a Fuel Assembly in Improper Location
- 15.1.24 Recirculation Flow Control Failure with Increasing Flow
- 15.1.25 Abnormal Startup of Idle Recirculation Pump
- 15.1.33 Control Rod Drop Accident

Increase in Reactor Coolant Inventory

Since the HPCI is not operating the following transient is not applicable:

- 15.1.10 Inadvertent HPCI Pump Start

Decrease in Reactor Coolant Inventory

Events Not Applicable to Spent Fuel Storage

Safety relief valve and feedwater system are not operating; therefore the following events are not applicable:

- 15.1.17 Inadvertent Opening of a Safety Relief Valve
- 15.1.37 Feedwater System Piping Break

The following event is not a design basis event and is applicable only to power operation:

- 15.1.27 Anticipated transient without Scram (ATWS)

The single failure-proof polar crane design eliminates the following event:

- 15.1.28 Cask Drop Accident

Instrument line, coolant line and steam line breaks present no consequences due to their inoperable status and therefore the following events are not applicable:

- 15.1.30 Off-Design Operational Transient as a Consequence of Instrument Line Failure
- 15.1.34 Pipe Breaks Inside the Primary Containment (Loss-of-Coolant Accident)
- 15.1.35 Pipe Breaks Outside Primary Containment (Steam Line Break Accident)

Events Without Fuel Damage

- 15.1.29 Miscellaneous Small Release Outside Primary Containment

Releases that could result from piping failures outside the primary containment include the pipe breaks in the fuel pool cleanup system. The offsite dose resulting from this will be negligible and is bounded by the Radwaste Tank Rupture accident.

- 15.1.31 Main Condenser Gas Treatment System Failure

As the mai[illegible] cond[illegible] is [illegible]ing, th[illegible] offsit[illegible] dose resulting [illegible] [illegible] will [illegible] negligible.

- 15.[illegible] [illegible]te Tank Rupture

[illegible] radioactivity [illegible] the [illegible] will [illegible] negligible. Refer to [illegible]

- 15[illegible]38 [illegible]re of Air Ejector Lines

As the main condenser is not operating, the offsite dose resulting from this will be negligible.

Events with Fuel Damage

- 15.1.36 Fuel Handling Accident

The fuel handling accident is assumed to occur as a consequence of a failure of the fuel assembly lifting mechanism. This could cause fuel damage and radioactivity release to the secondary containment. This event is analyzed in Section III.E.2.

Other Events

- Seismic Event

Because the spent fuel pool structure and fuel racks meet seismic Category I requirements, a seismic event is not postulated to create a radiological release.

CONCLUSION

The following events were identified for radiological analysis:

Design-Basis Accident Analyses:

1. Fuel Handling Accident (Regulatory Guide 1.25 assumptions)
2. Radwaste Tank Rupture

In addition, a worst case fuel damage accident was analyzed involving the release of the total gaseous inventory af the fuel. The remainder of Section III provides a detailed analysis of the above identified events.

III.E.2 FUEL HANDLING ACCIDENT

III.E.2.1 Identification of Causes

The fuel handling accident is assumed to occur as a consequence of a failure of the fuel assembly lifting mechanism, resulting in the dropping of a raised fuel assembly onto the top of the core or the spent fuel racks.

III.E.2.2 Starting Conditions and Assumptions

Accidents that result in the release of radioactive materials directly to the secondary containment can occur when the fuel is being handled. In this case, radioactive material released as a result of fuel damage is available for transport directly to the secondary containment. Table III.E.2-1 presents the parameters used in this analysis.

III.E.2.3 Accident Description

The most severe fuel handling accident from a radiological viewpoint is the dropping of a fuel assembly. The sequence of events is as follows:

	Event	Approximate Elapsed Time
1.	Fuel assembly is being handled by refueling equipment. The assembly drops.	0+
2.	Some of the fuel rods in both the dropped assembly and another assembly are damaged, resulting in the release of gaseous fission products to the reactor coolant and eventually to the secondary containment atmosphere.	1 min.
3.	The reactor building refueling floor exhaust radiation monitoring system may alarm to alert plant personnel.	1 min.
4.	Operator actions begin.	5 min.

III.E.2.4 Identification of Operator Actions

1. The operator may initiate the evacuation of the secondary containment.

2. The fuel handling foreman may instruct personnel to go immediately to the radiation protection personnel decontamination area.

3. The fuel handling foreman will make the operator aware of the accident.

4. The operator may initiate action to determine the extent of potential radiation doses by measuring the radiation levels in the vicinity of or close to the secondary containment.

5. If RBSVS were to be used, the operator or a delegate would determine whether the RBSVS is performing as designed. (See Section III.E.2.5)

6. The HP technican will post the appropriate radiological control signs at the entrance to the secondary containment.

7. Before entry to the secondary containment is made, a careful study of conditions, radiation levels, etc., will be performed.

III.E.2.5 HVAC Scenarios Considered

As will be seen in Section III.E.2.6.2, the quantity of gaseous fission products in the fuel's gap which is released is not large (2.52 Ci of Kr-85 only). Calculations indicate that the reactor building refueling floor exhaust radiation monitoring system would not alarm and consequently the RBSVS will not be actuated (i.e., the RBNVS continues to operate). As a result, analyses were performed assuming either RBSVS or RBNVS system operation. Secondary containment discharge rates are 167 and 6580 percent/day for the RBSVS and RBNVS cases, respectively. As a comparison case, a "puff" release over a short period of time (2 hours, as suggested by Regulatory Guide 1.25), has been analyzed. Although this is not a design basis case, it is useful to compare it with the two HVAC cases. Results for all three cases (assuming RBSVS, RBNVS, and puff release) are given in the following sections.

III.E.2.6 Analysis of Effects and Consequences

III.E.2.6.1 Evaluation Methods

The analytical methods and associated assumptions used to evaluate the consequences of this accident are consistent with Regulatory Guide 1.25, and are quite conservative. The assumptions and parameters are given in Table III.E.-2.

III.E.2.6.1.1 Methods, Assumptions, and Conditions

The assumptions used in the analysis of this accident are listed below:

1. The fuel assembly is dropped from the maximum height allowed by the fuel handling equipment.

2. The entire amount of potential energy, referenced to the top of the spent fuel racks, is available for application to the fuel assemblies involved in the accident. This assumption neglects the dissipation of some of the mechanical energy of the falling fuel assembly in the water above the racks and requires the complete detachment of the assembly from the fuel hoisting equipment. This is possible if fuel assembly handle, the fuel grapple, or the grapple cable breaks.

3. None of the energy associated with the dropped fuel assembly is absorbed by the fuel material (uranium dioxide).

III.E.2.6.1.2 Results and Consequences

III.E.2.6.1.2.1 Fuel Damage

The analysis of USAR Section 15.1.36.5.1.2.1 applies to this accident. In that section of the USAR, it was assumed that 125 fuel rods would fail as a result of dropping the fuel assembly into the reactor vessel.

III.E.2.6.1.2.2 Fission Product Release From Fuel

Fission product releases for the fuel handling accident are determined from the inventory in Table III.A-1. Specifically, it is seen that only Kr-85 is of any significance with respect to gaseous releases. The only other gaseous isotope in this table is H-3, which would add, at most, 0.1% to the skin dose from Kr-85. Using the above number of failed rods, and the assumptions given in Regulatory Guide 1.25, the quantity of Kr-85 released, from Reference E, is as follows:

$$\text{Release} = 1.56\text{E+03Ci} \times \frac{125 \text{ damaged rods}}{62 \text{ rods/bundle} \times 560 \text{ bundles in core}}$$

$$\times\ 1.5 \text{ peaking factor} \times 30\% \text{ in gap} = \underline{\underline{2.52 \text{ Ci}}}$$

III.E.2.6.1.3 Radiological Effects

Offsite

Radiological exposures have been evaluated for the meteorological conditions, parameters, and assumptions given in Table III.E.2-1. The results are given in Table III.E.2-2.

Control Room

Because the amount of radioactivity released is so small, the control room air intake monitors will not alarm, and the HVAC system will continue to function in its normal operating mode. The resultant whole body and skin 30-day integrated doses are, at most, 9.59E-08 and 2.08E-04 mrem, respectively, well below the 10CFR50 GDC 19 limits (Reference L).

Discussion

It is seen in Table III.E.2-2 that the (0-2 hour) EAB and (0-30 day) LPZ integrated doses are many orders of magnitude below 10CFR100 limits. Results are graphically shown in Figure III.E-1. Furthermore, the maximum (t=0) dose rates (whole body and skin) are very low and, with the exception of the RBNVS case, below Technical Specifications (see Figure III.E.-1A). This indicates that the HVAC system in use in the reactor building has no meaningful effect on radiological consequences to members of the public during a fuel handling accident, with the present fuel source terms.

TABLE III.E.2-1

FUEL HANDLING ACCIDENT - PARAMETERS FOR POSTULATED ACCIDENT ANALYSES

	Conservative (NRC) Assumptions
I. Data and assumptions used to estimate radioactive source from postulated accidents	
A. Power level	See Section III.A
B. Peaking factor	1.5
C. Fuel damaged	125 rods
D. Release of activity from fuel	30% Kr-85
E. Iodine fractions	
(1) Organic	N/A
(2) Elemental	N/A
(3) Particulate	N/A
II. Data and assumptions used to estimate activity released	
A. Secondary containment discharge rate (%/day)	See Section III.E.2.5
B. Adsorption and filtration efficiencies	
(1) Elemental iodine	N/A
C. Recirculation system parameters	
(1) Flow rate	N/A
(2) Mixing efficiency	N/A
III. Dispersion data	
A. EAB and LPZ distances (meters)	311/3,220
B. X/Qs (sec/m^3)	
EAB (0-2 hr)	1.36E-03
LPZ (0-8 hr)	2.50E-05
(8-24 hr)	1.75E-05
(1-4 days)	7.80E-06
(4-30 days)	2.45E-06
IV. Dose data	
A. Method of dose calculation	Regulatory Guide 1.25
B. Dose conversion assumptions	Regulatory Guide 1.25
C. Doses and Dose Rates	Table III.E-.2-2

TABLE III.E.2-2

FUEL HANDLING ACCIDENT RADIOLOGICAL CONSEQUENCES

HVAC Scenario	Whole Body Dose, rem EAB	LPZ	10CFR100 Limit	Skin Dose, rem EAB	LPZ	10CFR100 Limit*
RBSVS Operates	1.14E-07	1.22E-08	2.50E+01	9.90E-06	1.06E-06	3.00E+02

Maximum (t = 0) Dose Rates, mrem/hr

Whole Body Gamma EAB	LPZ	Tech.Spec Limit	Skin EAB	LPZ	Tech.Spec Limit
6.10E-05	1.12E-06	5.70E-02	5.30I-03	9.74E-05	3.42E-01

HVAC Scenario	Whole Body Dose, rem EAB	LPZ	10CFR100 Limit	Skin Dose, rem EAB	LPZ	10CFR100 Limit*
RBNVS Operates	1.74E-06	3.22E-08	2.50E+01	1.52E-04	2.80E-06	3.00E+02

Maximum (t = 0) Dose Rates, mrem/hr

Whole Body Gamma EAB	LPZ	Tech.Spec Limit	Skin EAB	LPZ	Tech.Spec Limit
4.79E-03	8.81E-05	5.70E-02	4.17E-01	7.66E-03	3.42E-01

HVAC Scenario	Whole Body Dose, rem EAB	LPZ	10CFR100 Limit	Skin Dose, rem EAB	LPZ	10CFR100 Limit*
Puff Release	1.75E-06	3.22E-08	2.50E+01	1.52E-04	2.80E-06	3.00E+02

Maximum (t = 0) Dose Rates, mrem/hr

Whole Body Gamma EAB	LPZ	Tech.Spec Limit	Skin EAB	LPZ	Tech.Spec Limit
8.75E-04	1.61E-05	5.70E-02	7.61E-02	1.40E-03	3.42E-01

* The skin dose limit is set equal to the thyroid limit.

FIGURE III.E-1

Design Basis Fuel Handling Accident
Exclusion Area Boundary Results
RBNVS HVAC System in Operation

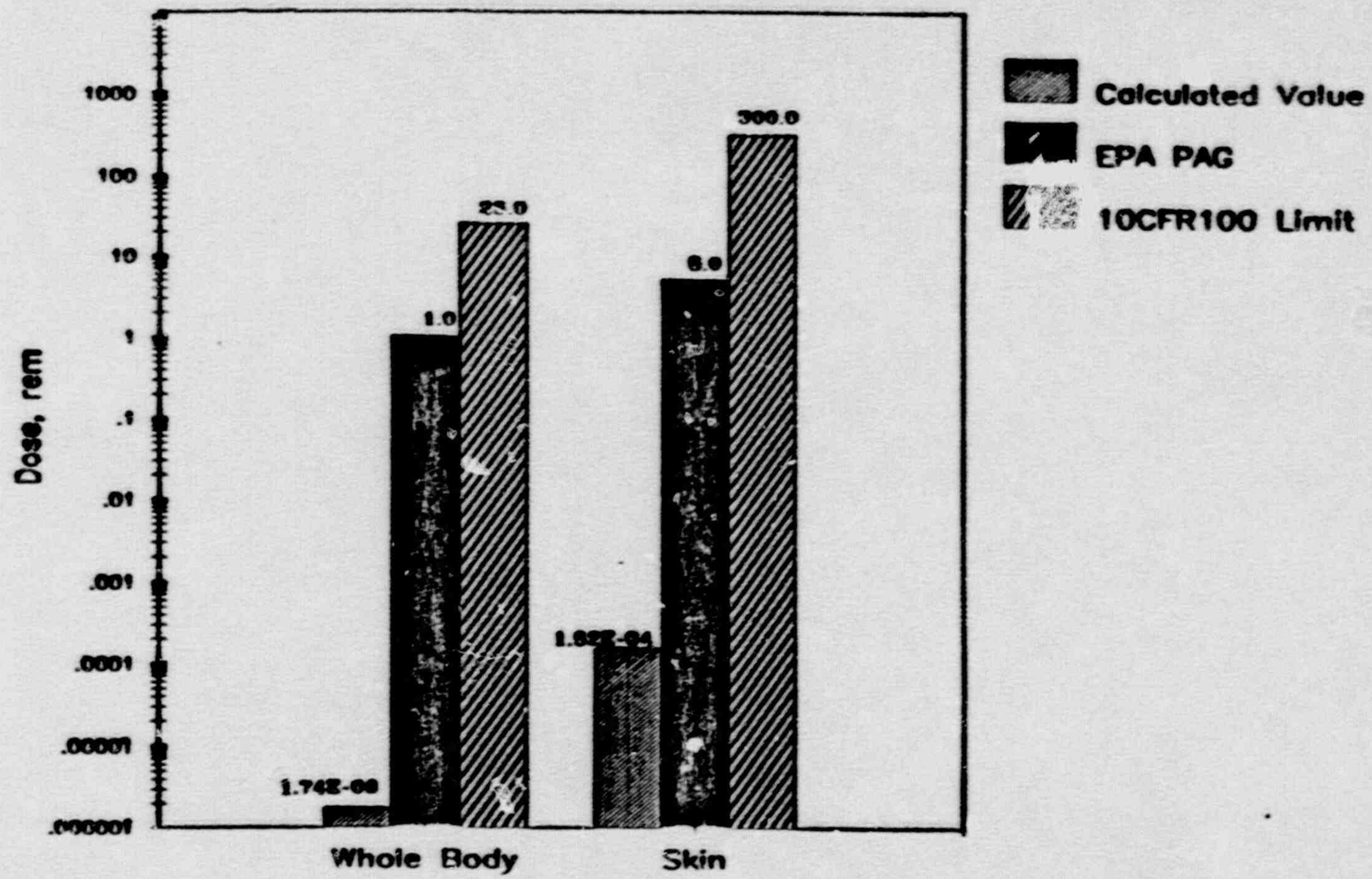
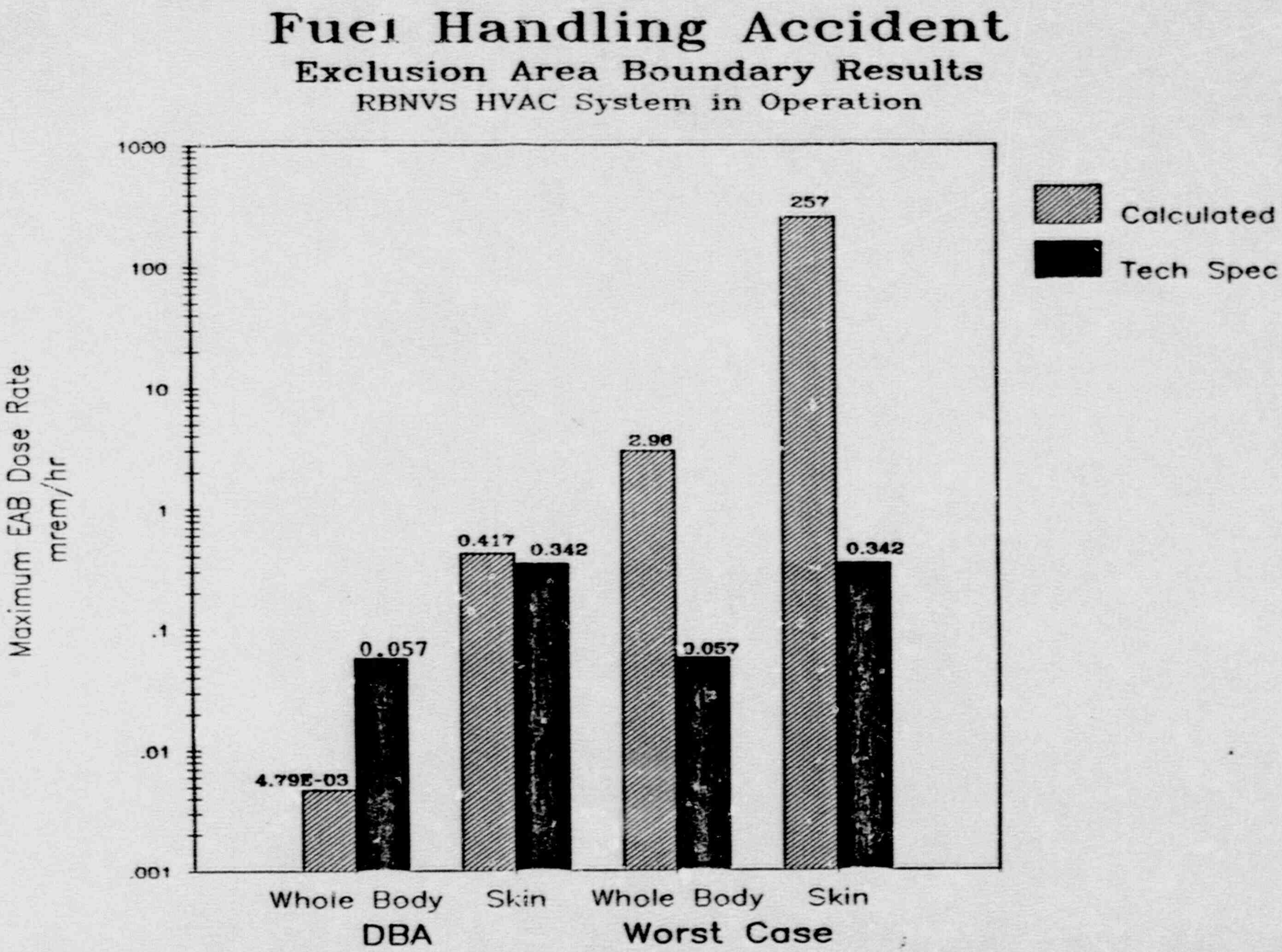


FIGURE III.E-1A



III.E.3 Radwaste Tank Rupture

The accident scenario postulated in the USAR Sections 11.2.3.4.2 through 11.2.3.4.4 is considered here.

1. A conservative partition factor of 1.0E-03 is assumed for all isotopic activities listed in Table III.E.3, with the exception of H-3, for which it is assumed all activity is evolved.

2. A two hour release duration is assumed.

3. Ground release atmospheric dispersion factors are assumed, as given in Table III.E.2-1, for the EAB. Note that the EAB is limiting insofar as 10CFR100 dose limits are concerned, because the release duration is two hours.

4. The breathing rate of persons offsite is assumed to be 3.47E-04 cubic meters per second, consistent with Regulatory Guides 1.3 and 1.25. For other age groups the breathing rate was obtained from the ratio of the maximum age group rates given in Regulatory Guide 1.109 (Reference J).

The doses resulting from the analysis described above are as follows:

	Dose, millirem		
Source	Whole Body Gamma*	Beta Skin	Maximum Organ**
Spent Resin Tank	1.8E-05	2.7E-06	1.3E-03
Radwaste Filters	1.2E-07	1.7E-08	8.3E-06
Discharge Sample Tanks	3.1E-08	1.4E-08	7.7E-06
Totals	1.8E-05	2.8E-06	1.3E-03

The consequences of the above postulated accident are clearly very low. The whole body gamma, skin, and thyroid doses are 7.2E-08, 9.3E-10, and 4.3E-07%, respectively, of the 10CFR100 dose limits. Furthermore, these projected doses are far below those which justify Quality Group D, non-seismic qualification of radwaste equipment (i.e., 500 mrem whole body, or its equivalent to parts of the body).

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* External & internal pathways; child is the limiting age group

** Teen is the limiting age group, and lung is the limiting organ

III.E.4 WORST CASE FUEL DAMAGE EVENT

Scenario

Several "worst case", extremely conservative scenarios were examined. Specifically, for the three reactor building HVAC cases analyzed in Section III.E.2 (RBSVS operating, RBNVS operating, and puff release), instead of assuming the gap activity from 125 fuel rods is released (2.52 Ci Kr-85), it is assumed that all gaseous activity from the entire core in the spent fuel pool is released (1.56E+03 Ci Kr-85). This can only occur if all the fuel is postulated to be mechanically damaged and there is a complete release of gaseous isotopes. The assumption of a complete release of the gaseous inventory is also very conservative with respect to the Regulatory Guide 1.25 assumption of a 30% release fraction given the low burnup condition of Shoreham spent fuel. Doses and dose rates are thus a factor of 617 higher than for the corresponding Regulatory Guide 1.25 cases.

All other conditions and parameters indicated in Table III.E.2-1 apply to these cases. Results are given in Table III.E.4-1.

Discussion

Even with the highly conservative release quantity postulated above, the calculated whole body and skin doses at the EAB and LPZ are very small fractions (< 0.031%) of the 10CFR100 dose limits. Results are graphically shown in Figure III.E-2. Dose rates for the postulated worst case scenario are above current Technical Specification limits (see Figure III.E-1A), but the duration of the high dose rates in the RBNVS and puff release cases is quite short (two hours or less).

TABLE III.E.4-1

"WORST CASE" FUEL DAMAGE ACCIDENT RADIOLOGICAL CONSEQUENCES

HVAC Scenario	Whole Body Dose, rem			Skin Dose, rem		
	EAB	LPZ	10CFR100 Limit	EAB	LPZ	10CFR100 Limit
RBSVS Operates	7.03E-05	7.50E-06	2.50E+01	6.11E-03	6.52E-04	3.00E+02

Maximum (t = 0 Dose Rates, mrem/hr

Whole Body Gamma			Skin		
EAB	LPZ	Tech.Spec Limit	EAB	LPZ	Tech.Spec Limit
3.76E-02	6.92E-04	5.70E-02	3.27E+00	6.01E-2	3.42E-01

RBNVS Operates	Whole Body Dose, rem			Skin Dose, rem		
	EAB	LPZ	10CFR100 Limit	EAB	LPZ	10CFR100 Limit
	1.08E-03	1.99E-05	2.50E+01	9.35E-02	1.73E-03	3.00E+02

Maximum (t = 0) Dose Rates, mrem/hr

Whole Body Gamma			Skin		
EAB	LPZ	Tech.Spec Limit	EAB	LPZ	Tech.Spec Limit
2.96E-00	5.44E-02	5.70E-02	2.57E+02	4.73E+00	3.42E-01

Puff Release	Whole Body Dose, rem			Skin Dose, rem		
	EAB	LPZ	10CFR100 Limit	EAB	LPZ	10CFR100 Limit
	1.08E-03	1.99E-05	2.50E+01	9.39E-02	1.73E-03	3.00E+02

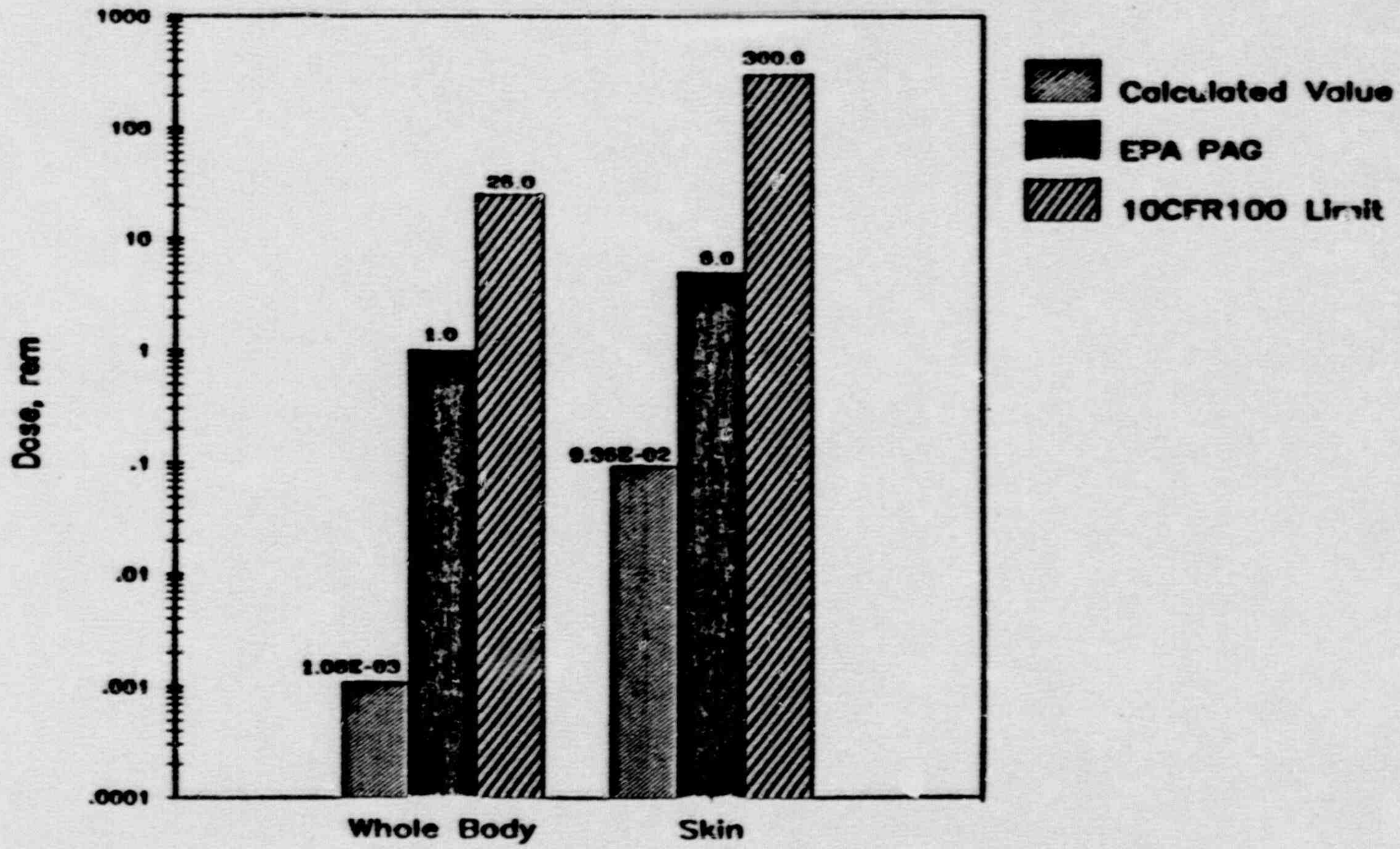
Maximum (t = 0) Dose Rates, mrem/hr

Whole Body Gamma			Skin		
EAB	LPZ	Tech.Spec Limit	EAB	LPZ	Tech.Spec Limit
5.40E-01	9.93E-03	5.70E-02	4.70E+01	8.63E-01	3.42E-01

* Skin dose limit set equal to thyroid limit

FIGURE III.F-2

Worst Case Fuel Damage Accident
Exclusion Area Boundary Results
RBNVS HVAC System in Operation



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