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PRELIMINARY SAFETY ANALYSIS REPORT ADDENDUM
LOFT FACILITY

November 5, 1965

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SUMMARY

SUMMARY

In 1964, the Division of Reactor Licensing and the Advisory Committee on Reactor Safeguards reviewed the Preliminary Safety Analysis Report and approved the LOFT facility for construction at the Test Area North (TAN) of the National Reactor Testing Station. The report analyzed the hazards attendant to operating an experimental pressurized power reactor designed for the specific purpose of investigating the consequences of a loss-of-coolant accident. The proposed test program embodied the use of a pressurized water-cooled and -moderated power reactor system containing a uranium-dioxide stainless-steel clad core of 50 MW(t) capacity.

Subsequent to the approval of the Preliminary Safety Analysis Report, the Atomic Energy Commission requested that Phillips Petroleum Company determine the feasibility of converting the fuel cladding material from stainless steel to Zircaloy, and perform nuclear engineering and physics studies on a Zircaloy-clad core to identify related engineering problems in the facility and experimental systems. This addendum to the Preliminary Safety Analysis Report has been prepared to present the results of these studies, in particular to show the influence of a zirconium-clad system on the overall safety of the plant and on the potential hazards associated with the loss-of-coolant test. Although the design of the experimental and facility systems have progressed since the last review, this report presents only those design changes which are of significance to plant safety.

The potential hazards of conducting a loss-of-coolant test program with a pressurized water reactor system containing UO_2 fuel clad with Zircaloy have been evaluated. The areas of consideration in this evaluation include metal-water reactions with the ensuing hydrogen release and burning, start-up accident, rod withdrawal accident, and an

unplanned loss-of-coolant reference accident which is postulated for the purpose of providing a situation which will call for proper functioning of the engineered safeguards. This is referred to subsequently as the maximum credible accident (MCA) in conformity with common usage.

The safety analysis of the MCA is based upon the postulation of a pipe break in both the primary and secondary coolant systems during adverse meteorological conditions. It is further assumed that as the result of the rupture a missile is generated with sufficient energy to penetrate the containment vessel. The same assumptions used in the analysis of the planned test were used to describe the fission product release and transport to the containment; however, the employment for the MCA of the pressure reduction sprays was included for determining the release to the atmosphere. The results of this analysis indicate that the doses that could be received at the nearest site boundary are below the guide limits set for accidental releases to the atmosphere.

On the basis of this safety analysis, it is concluded that the loss-of-coolant test using a zirconium-clad core can be safely performed in the LOFT facility at its present location. It is further concluded that in the unlikely event of an MCA the consequences will not present undue risk to on-site personnel or the general public.

PRELIMINARY SAFETY ANALYSIS REPORT ADDENDUM
LOFT FACILITY

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CHAPTER I

INTRODUCTION

I. INTRODUCTION

The Preliminary Safety Analysis Report⁽¹⁾ for the LOFT (Loss-of-Fluid-Test) facility was submitted in 1964 to the Division of Reactor Licensing and the Advisory Committee on Reactor Safeguards. The report was reviewed and construction approval granted by the Atomic Energy Commission in August of that year. Since that time, the reactor system design has been modified to include a fuel pin cladding material of Zircaloy-4 in lieu of the previously proposed and approved stainless steel cladding. In the opinion of Phillips Petroleum Company, this modification constitutes a significant change in the facility concept as approved for construction and, as a result, can affect the response of the facility to the planned loss-of-coolant test. For these reasons, this addendum has been prepared to show the influence of this conversion on the overall safety of the plant and on the consequences of the loss-of-coolant test.

Specific recommendations were made by the Advisory Committee⁽²⁾ on Reactor Safeguards at the conclusion of their review of the Preliminary Safety Analysis Report. As a result of these recommendations additional studies were made and the results of these studies are included in this addendum. Specifically, these studies included the incorporation of an in-core emergency cooling system, the increase in the performance of the containment pressure reduction spray system, further investigations into the containment structural integrity under sub-zero climatic temperature conditions, and the selection of specified containment leakage rates. Since issuance of the original report, the detailed design has progressed far from the conceptual

stage and, as would be expected, many other design modifications have resulted; however, the intent of this report is not to cover those detailed design changes which do not have a significant effect on the operational safety of the plant or the conduct of the loss-of-coolant test. As in the original report, this addendum is concerned principally with the analysis of the hazards attendant to carrying out the loss-of-coolant test and the analysis of potential hazards associated with the postulated maximum credible accident.

In the safety analyses, an attempt has been made to arrive at predictions of the test and accident consequences, using realistic assumptions where reliable experimental data exist and conservative assumptions where data are not available. The results of this work demonstrate that the conversion of the LOFT core from stainless steel cladding to Zircaloy-4 cladding can be accommodated without undue risk to site personnel or to the general public.

-
- (1) J. M. Waage, Editor, "Preliminary Safety Analysis Report - LOFT Facility," IDO-16981, April 1964.
 - (2) Herbert Kouts, "Report on LOFT Facility," Letter to Glenn T. Seaborg, August 28, 1964.

CHAPTER II

DESIGN MODIFICATIONS

II. DESIGN MODIFICATIONS

A. Core Design

The core design modification consisted primarily of changing from cold-worked stainless steel to cold-worked Zircaloy cladding material. Retention of the same fuel pin spacing minimized mechanical and nuclear design modifications to the core. The changes in the reactor design features are presented in Table II-1.

1. Mechanical

Zircaloy-4, cold-worked to an H-12 temper (1/4 hard), is proposed for use as the cladding material. Zircaloy-4 was favored over Zircaloy-2 because it is less susceptible to hydriding due to its lower nickel content.

The mechanical changes to the core consist primarily of increasing the cladding thickness and subsequently increasing the fuel pin outside diameter. The fuel pellet diameter and the pellet-to-cladding gap were unchanged. This modification has little effect on the rest of the core dimensions. The fuel-assembly fuel-pin spacers will require a slight modification to accommodate the larger diameter pin. Because of the larger pin the pressure drop across the core will increase slightly but not significantly.

2. Nuclear

Reactor Physics calculations for the LOFT Zircaloy-clad core included the determination of fuel enrichments, core life, peak-to-average power ratios, rod worths, and reactivity coefficients. The Zircaloy-clad core nuclear characteristics are compared with the corresponding stainless-steel clad core characteristics in Table II-1. The Zircaloy-clad core data reflects, in addition to the change in fuel

cladding material, changes in control rod and dummy pin poison content, an increase in metal/water ratio due to increased clad thickness, improvements in nuclear data, and improvement in calculational techniques. Comparisons made of values in Table II-1 must take into consideration these changes and improvements. When the Zircaloy-clad values were compared to a later stage of the stainless-clad design having similar control rod and dummy pin poison contents but not the same calculational techniques, a more meaningful comparison of some of the nuclear parameters resulted. For the same hot, all rods out, excess reactivity, the Zircaloy core needed a lower fuel enrichment combination. With the lower fuel enrichments, the Zircaloy core yielded a higher average core flux and a shorter core life (EFPH). The cold total reactivity worth of the control rods was greater for the Zircaloy core and the ℓ/β ratio increased for the change from stainless to Zircaloy cladding.

a. Fuel Enrichments

Fuel enrichment calculations used a number of PDQ-40⁽¹⁾, two-dimensional, four-group problems with X-Y geometry. A quarter-core PDQ model of the LOFT core is shown in Figure II-1. A number of problems were run to balance the radial peak-to-average power values in the two fuel regions while searching for an eigenvalue which produces a core life greater than 1600 effective full power hours.

Homogeneous thermal group constants were obtained using the TOTEM⁽²⁾ code. An S-8 approximation with a mixed isotropic and mirror reflection boundary condition was applied to the fuel-pin unit-cell. Homogeneous fast group constants were determined from GAM-MOD⁽³⁾.

Except for the control-rod poison section, both fast and thermal group constants for all non-fuel interior regions were obtained

by averaging the cross sections over the GAM-MOD and the TOTEM fuel core spectrums, respectively. Thermal constants for the control rod poison sections were calculated using a TOTEM blackness edit for the absorption and stainless-steel numbers for the diffusion coefficients. Fast group constants for the poison sections were obtained from the GAM-MOD blackness edit. All constants for the regions external to the core were calculated by averaging the cross sections over a water spectrum.

An enrichment pair of 2.44 and 5.33 a/o U-235 has tentatively been chosen for the inner and outer loading zones, respectively. These enrichments are subject to slight adjustments due to refinements in calculational methods and nuclear data.

b. Core Life

The core life was calculated from four-group TURBO⁽⁴⁾ calculations in X-Y geometry using core cross sections for a uranium enrichment pair of 2.38 and 5.20 a/o U-235 which had been determined by preliminary calculations. The model used for the problem was a quarter core with rods out, i.e., all rod positions were occupied by follower sections. The core average power density remained constant with the time and no attempt was made to maintain criticality throughout core life. The problem started with an initial excess reactivity due to the rods being out and the remaining reactivity was calculated after various time steps until criticality was reached, indicating end-of-core life.

A lumped fission-product absorption cross section of 29.2 barns thermal and 34 barns epithermal was used to represent all fission products except Xe-135 and Sm-149. The first value corresponds to a 2200 m/sec cross section of 50 barns which was spectrum hardened for the inner loading zone. These values were found to give results in good agreement with measured reactivity loss rates in Yankee Nuclear Power Station, Reactor Core I.

The TURBO results indicated a core life of greater than 1600 effective full power hours with the final enrichment pair of 2.44% and 5.33%. This calculated core life is subject, however, to change with enrichment adjustment or calculational refinement.

c. Reactivity Coefficients

Reactivity coefficients were determined from the results of four-group one dimensional FOG problems using the radial core model shown in Figure II-2 and four-group two-dimensional PDQ problems. The moderator temperature, Doppler, void and pressure coefficients were determined and are listed in Table II-1 and shown in Figures II-3, II-4, and II-5.

To calculate the temperature coefficient, four FOG problems were run at different reactor temperatures. In computing constants, all materials were assumed to be at the same temperature; i.e., zero power conditions. PDQ problems were run for 68°F and 545°F, and the FOG results were normalized to these points.

The Doppler coefficient was calculated by varying fuel temperatures to the U-238 resonance inputs to the GAM-MOD code. FOG problems were run for both hot, 545°F, and cold, 68°F, moderator. In each set of hot and cold cases, the FOG problems were normalized to one PDQ problem. The resulting reactivities are shown in Figure II-4.

The void coefficient was determined by running FOG problems with the moderator uniformly voided from 0 to 60% and normalized to two PDQ problems. Figure II-5 shows the resulting void coefficient. The pressure coefficient was obtained from the void reactivity curve. Uniform void formation and pressure reduction affect reactivity by exactly the same mechanism; i.e., moderator density change. Therefore, by relating the density of the moderator to the

pressure and to the void fraction, the pressure coefficient was calculated.

d. Axial Calculations

Axial peak-to-average power ratios, power profiles, control rod worths and neutron multiplication were calculated for the Zircaloy-clad core as a function of rod insertion depth for both hot and cold reactor cases. In addition, the reactivity insertion due to the ejection of a single rod from the core was calculated.

The axial calculations were developed by using four-group FOG problems employing a simple window shade model. The constants and radial bucklings used for the fueled core regions were flux weighted values previously calculated by PDQ for a horizontal plane through the "rods out" vertical section of the core. Control rods were represented in the axial model by the addition of a $1/v$ poison to the flux-weighted core constants. The amount added was such that its addition over the whole core length resulted in the correct hold-down reactivity as determined from PDQ "rods in" and "rods out" problems. Figures II-6 through II-11 show neutron multiplication, axial power distribution and rod worths for the hot and cold cases.

A full core PDQ model was used to run two cold stuck-rod problems. Both problems had eleven poison sections and a single follower section. The follower occupied an outer control rod position in one problem and an inner rod position in the other problem. The PDQ results listed in Table II-1 indicate that the reactor is shut down by a slightly greater margin in the inner stuck-rod case.

e. Flux and Power Contour Plots

The PDQCP (PDQ Contour Plot) program was used to obtain the contour plots of four-group fluxes shown in Figures II-12 through II-15. The fluxes used by the program had been stored from a previously run PDQ hot, "rods-out" case using the model shown in Figure II-1. The fluxes were normalized to the 50 MW LOFT power by summing the neutron fission rate over all PDQ regions and energy groups. The power contour plot shown in Figure II-16 was plotted from point-to-average power values derived from the same "rods-out," hot PDQ problem.

B. Modifications Recommended By the Advisory Committee on Reactor Safeguards

At the conclusion of its 57th meeting, August 24-26, 1964, the Advisory Committee on Reactor Safeguards recommended that four specific areas of the LOFT facility and experimental assembly receive special attention during detailed design of the plant⁽⁵⁾. The results of further design studies and modifications to incorporate the committee's recommendations are summarized in this section.

1. Containment Vessel Temperature

Sub-zero climatic temperatures at the NRTS demand careful selection of Materials when pressure vessels are exposed to the outside environment. The nil-ductility transition (NDT) temperatures of reactor containment vessel materials must be carefully specified and analyzed for operational and environmental compatibility to insure adequate structural integrity at all times during reactor operation. The importance of adequate confinement in LOFT is further emphasized because of the proposed experimental violation of the first two passive fission product barriers, i.e., fuel cladding and the primary system. In this regard the LOFT containment building is designed in accordance with Section III of the ASME Nuclear Vessel Code and employs an A-212 Grade B material modified to meet the requirements of A-300 and the

Charpy V-notch Tests specified in paragraph N-330 at -50°F . This will permit operating the containment with the service metal temperature as low as -20°F as per paragraph N-1210 of Section III.

Provisions have been included in the containment design to measure and record the surface temperature of the vessel for purposes of determining containment response during simulated accident conditions. These experimental measurements will be utilized to assist in an administrative decision to temporarily discontinue reactor operation in the unlikely event the surface metal temperature should indeed approach -20°F .

Tests are currently in progress at EBR-II under the direction of the AEC-ID, Health and Safety Division to investigate containment surface temperatures as a function of atmospheric conditions, temperature, wind, solar effects, etc. The preliminary experimental data⁽⁶⁾ indicate that surface temperatures are higher than the ambient temperatures and become progressively more divergent as ambient temperatures decrease. Additional studies on EBR-II and on the LOFT containment during conduct of the non-nuclear test program may conclusively demonstrate that surface temperatures are above -20°F , even with ambient temperatures approaching the lowest recorded winter time temperature at NRTS (-41°F at TAN in January 1963). However, regardless of the outcome of the supporting experimental programs to develop surface temperatures and ambient temperature correlations, it is presently planned that in the unlikely event the containment shell temperature should fall below -20°F an orderly shutdown which includes depressurization of the primary system, will be performed in accordance with established operational procedures.

2. Containment Leakage

The LOFT containment vessel is conservatively designed to insure a minimum leakage of fission products to the atmosphere following

the planned loss-of-coolant test and also during the accident case where the primary and secondary system are inadvertently released to the containment followed by a 100% metal-water reaction of the zirconium cladding. The predicted peak pressure in the first case is 18.7 psig and 27 psig in the latter or accident case.

The containment vessel will be pressure tested to 46 psig (115% of design pressure) at the conclusion of the shell erection with the railroad door in place, penetrations capped, and prior to installation of internal facility and experimental apparatuses. The total leakage is not to exceed 0.2% by wt. of the building free volume per 24 hours at the design pressure of 40 psig at this stage of construction. At the conclusion of the facility construction, with all piping and electrical penetrations completed, the building will again be pressure tested to 46 psig. Leakage from the completed containment is not to exceed 0.2% by wt. of the building free volume in 24 hours at a internal pressure of 24 psig.

Accomplishment of the minimum leakage requirements will be attained by four stage construction of the containment building. Phase I will permit helium leak testing of the basement areas prior to installation of concrete around the lower head. The leak testing with helium will be essentially at atmospheric pressure with the maximum leakage limited to concentrations as measured by the helium detector (10^{-6} cc/liter). Satisfactory completion of leak testing in the basement areas (void of concrete) is required prior to lowering the lower head to the permanent vessel supports, grouting external to the head, and installing concrete inside the shell. Phase II of the containment vessel leak testings will commence upon completion of the fabrication and erection of the cylindrical walls, airlocks, hemispherical dome, railroad door, shielding window and all vessel nozzles. To meet the 0.2% leakage criteria at 40 psig pressure testing will be performed at the conclusion of the vessel erection. Phase III of the leak testing will commence after the installation of the

horizontal steel membrane which separates the test chamber from the basement enclosure. The leak testing with helium will be essentially at atmospheric pressure with the maximum leakage limited to concentrations as measured by the helium detector (10^{-6} cc/liter). Upon satisfactory completion of leak testing the concrete floor above the membrane and the grouting between the membrane and lower sub floor, will be installed. Phase IV of the leak testing will be conducted upon completion of the final installation of services through the penetrations. Phase IV testing is divided into two separate parts. Part one consists of leak testing the entire containment (test chamber and basement). In part 2, only the test chamber will be tested. The leak testing in each part will be conducted at a pressure of 46 psig.

As described in Section V, the 0.2% leakage criteria is sufficient to preclude excessive dose values at the site boundaries when employing reasonable assumptions in the fission product release and transport model. It is fully expected that leakage from the containment can be further reduced during phase I of the experimental program⁽⁷⁾ and with the employment of the pressurization or collection systems on each vessel double penetration. Therefore, if further reduction of leakage through the penetration is desired, the plenum between the double penetrations can be pressurized to above that existing in the containment.

3. Containment Pressure Reduction

The design of the containment pressure reduction spray system has been modified to incorporate the suggestion of the ACRS that this system be improved or supplemented to provide the capability for more rapid pressure reduction in the containment vessel. The pressure reduction spray cooler, secondary cooling flow rates, and spray water delivery rates have been increased substantially. An analysis of this system, using conservative assumptions for the energy release and the

heat removal capabilities of the containment, indicates that the containment pressure will be reduced from the accident peak value of 26.5 psig to 2 psig within one hour. At the end of one hour the containment sump water temperature is approximately 123°F and the containment vapor and concrete surface temperatures are 117°F. With the low temperature differentials available for energy transfer, the efficiencies of the sprays and the cooler are significantly reduced beyond this time.

The performance of the system has been improved by increasing the spray delivery rate from 200 gpm to 1000 gpm and the secondary coolant flow rate from 350 gpm to 1000 gpm. As a result, the time required to reduce the peak pressure to 1 psig following accidental blowdown has been decreased from 4 hours to 1.3 hours.

4. Additional Emergency Cooling

An emergency core cooling system has been incorporated into the LOFT reactor design. The system, as presently conceived, will supply 400 gpm of service water at 100 psig into a reactor vessel access nozzle. The nozzle penetrates the reactor vessel head in such a manner that the water jet will be directed onto the center of the upper core support plate. The coolant control valving includes two normally closed air-operated control valves mounted in series which are opened or closed by either automatic or manual initiation. When operating in the automatic mode the valves open when reactor system pressure drops to 40 psig.

The emergency core cooling system provides 8 gpm per megawatt of thermal reactor operating power as compared with current emergency cooling systems which provide from one to two gpm per megawatt of reactor operating power. The LOFT emergency cooling system then provides ample flow to prevent core melting in the event an accidental loss-of-coolant is experienced.

- (1) "PDQ-40, A Program for the Solution of the Neutron Diffusion Equations in Two Dimensions on the IBM-7040," Phillips Petroleum Company, Program No. 40.0018.
- (2) "TOTEM," Phillips Petroleum Company, Program No. 40.0269.
- (3) "GAM-MOD," Phillips Petroleum Company, Program No. 40.0181.
- (4) O. J. Marlowe, et.al., "Nuclear Reactor Depletion Program for the Philco 2000 computer," WAPD-TM-221, January 1961.
- (5) Herbert Kouts, "Report on LOFT Facility," Letter to Glenn T. Seaborg, August 28, 1964.
- (6) Alden P. Richter, "Experimental Breeder Reactor II Dome Temperatures," Weather Bureau Research Station, NRTS, May 1965, Unofficial Report.
- (7) T. R. Wilson, et.al., "An Engineering Test Program to Investigate a Loss-Of-Coolant Accident," IDO-17049, October 1964.

TABLE II-1

LOFT REACTOR DESIGN FEATURES

	<u>Fuel Cladding</u>	
	<u>Stainless Steel</u>	<u>Zircaloy</u>
Normal Operating Power MW	50	50
Core Lifetime (EFPH)	>1600	>1600
Core Characteristics		
Effective Diameter (in.)	41.5	41.5
Active Height (in.)	36	36
No. of Fuel Assemblies	52	52
No. of Fuel Pins Per Assembly	64*	64*
No. of Fuel Enrichment Zones	2	2
No. of Low Enrichment Fuel Assemblies	32	32
No. of High Enrichment Fuel Assemblies	20	20
No. of Poison Pins	36*	36*
No. of Control Rods	12	12
Fuel Assembly		
Fuel	UO ₂ (94% TD)**	UO ₂ (95% TD)
Outer Zone Fuel Enrichment (a/o Uranium)	5.71	5.33
Inner Zone Fuel Enrichment (a/o Uranium)	2.82	2.44
Fuel Pellet Diameter (in.)	0.357	0.357
Fuel Pellet Length (in.)	0.720	0.720
Cladding Material	SS-304L	Zircaloy-4

*Some assemblies will have one poison pin substituted for a fuel pin.

**Theoretical density

TABLE II-1 (Con't)

LOFT REACTOR DESIGN FEATURES

	<u>Fuel Cladding</u>	
	<u>Stainless Steel</u>	<u>Zircaloy</u>
Cladding Thickness (in.)	0.015	0.025
Gap Between Cladding and Fuel (in.)	0.002	0.002
Fuel Pin Diameter (in.)	0.391	0.411
Fuel Pin Spacing (in.)(ctr. to ctr.)	0.580	0.580
Fuel Pin Metal/Water Ratio (unit cell)	0.54	0.64
Poison Pin Diameter (in.)	0.391	0.411
Poison Pin Material	0.05 w/o B ¹⁰ - SS	0.10 w/o B ¹⁰ - SS
Control Rod		
Type	Cruciform	Cruciform
Width (in.)	8.875	8.875
Thickness (in.)	0.404	0.404
Poison Section Material	1.5 w/o B ¹⁰ - SS	1.75 w/o B ¹⁰ - SS
Effective Poison Width (in.)	8.0	8.25
Poison Thickness (in.)	0.312	0.312
Poison Length (in.)	36.0	36.0
Follower Section Material	Zircaloy-2	Zircaloy-2
Core Volume Composition (%)		
UO ₂ Fuel	23.96	23.96
Water	61.69	58.63
Stainless Steel		
Cladding	4.24	None
Fuel Assembly	2.11	2.11

TABLE II-1 (Con't)

LOFT REACTOR DESIGN FEATURES

	<u>Fuel Cladding</u>	
	<u>Stainless Steel</u>	<u>Zircaloy</u>
Poison Pins	0.31	0.35
Zircaloy		
Cladding	None	7.26
Control Rod Follower and Filler Pieces	7.15	7.15
Void	0.54	0.54
Miscellaneous Reactor Characteristics		
Reflector (in. of water)	3.4	3.4
Lower Core Support Barrel (in. of Stainless Steel)	1	1
Thermal Shield(in. of Stainless Steel)	2	2
Water Gap Between Barrel and Shield (in.)	1	1
Reactor Operating Characteristics (at 50 MW)		
Average Coolant Temperature in Vessel (^o F)	540	540
Average Coolant Temperature Rise in Vessel (^o F)	24	24
Operating Pressure (psia)	2330	2330
Coolant Flow Through Reactor (lbs/hr)	5.6×10^6	5.6×10^6
Coolant Flow Through Core (lbs/hr)	5.0×10^6	5.0×10^6
Pressure Drop Through Vessel (psi)	14.2	14.7
Core Operating Characteristics (at 50 MW)		
Hydraulic		
Core Pressure Drop (psi)	7.5	8.0

TABLE II-1 (Con't)

LOFT REACTOR DESIGN FEATURES

	<u>Fuel Cladding</u>	
	<u>Stainless Steel</u>	<u>Zircaloy</u>
Ave. Coolant Velocity (ft/sec)	5.3	5.7
Coolant Flow Past Fuel Pins (lbs/hr)	4.5×10^6	4.5×10^6
Thermal		
Ave. Core Coolant Temp. ($^{\circ}\text{F}$)	543	543
Ave. Core Coolant Rise ($^{\circ}\text{F}$)	31	31
Ave. Fuel Pin Film Drop ($^{\circ}\text{F}$)	61	60
Ave. Fuel Pin Surface Temp. ($^{\circ}\text{F}$)	601	600
Maximum Fuel Pin Surface Temp. ($^{\circ}\text{F}$)	662	662
Ave. Fuel Pin Fuel Temp. ($^{\circ}\text{F}$)	1400	1400
Per Cent of Ave. Fuel Pin at Nucleate Boiling	25	20
Active Core Surface Area (ft^2)	1022	1073
Ave. Heat Flux in Core (Btu/hr-ft^2)	1.67×10^5	1.59×10^5
Core Power Density (KW/liter)	62	62
Fuel Power Density (KW/liter)	256	256
Peak-to-Average Power		
Axial (initial critical)	2.1	2.09
(all rods out)	1.4	1.37
Radial (all rods out)	1.6	1.64
Hot Channel Factor	3.3	3.2
Nuclear		
Average Thermal Flux ($\text{n/cm}^2\text{-sec}$)	3.2×10^{13}	3.5×10^{13}

TABLE II-1 (Con't)

LOFT REACTOR DESIGN FEATURES

	<u>Fuel Cladding</u>	
	<u>Stainless Steel</u>	<u>Zircaloy</u>
Reactivity Coefficients		
Temperature		
Doppler ($\Delta\rho/^{\circ}\text{F}$)		
Cold	-1.5×10^{-5}	-1.3×10^{-5}
Hot (Zero Power)	-1.3×10^{-5}	-1.1×10^{-5}
Hot (50 MW)	-0.9×10^{-5}	-0.8×10^{-5}
Overall ($\Delta\rho/^{\circ}\text{F}$)		
Cold	-0.57×10^{-4}	-0.3×10^{-4}
Hot (Zero Power)	-2.6×10^{-4}	-3.4×10^{-4}
Average ($68^{\circ}\text{F} - 540^{\circ}\text{F}$)	-1.3×10^{-4}	-1.2×10^{-4}
Void, Hot 50 MW, ($\Delta\rho/\%$ Moderator Void)	-2.2×10^{-3}	-2.0×10^{-3}
Pressure, Hot, 50 MW, ($\Delta\rho/\text{psi}$)	$+2.5 \times 10^{-6}$	$+2.3 \times 10^{-6}$
Control Characteristics		
k_{eff}		
Cold (12 Rods Out)	1.123	1.136
Hot (12 Rods Out)	1.052	1.059
Cold (Outer Rod Stuck)	0.954	0.967
Cold (12 Rods In)	0.890	0.890
Reactivity ($k_{\text{cx}}/k_{\text{eff}} = \rho$)		
Cold (12 Rods Out)	0.110	0.120

TABLE II-1 (Con't)

LOFT REACTOR DESIGN FEATURES

	<u>Fuel Cladding</u>	
	<u>Stainless Steel</u>	<u>Zircaloy</u>
Hot (12 Rods Out)	0.049	0.056
Cold (12 Rods In)	0.124	0.124
Total Inserted Control Rod Worths ($\Delta k/k_1 k_2$)		
Cold	0.233	0.244
Control Rod Worths, 11 Rods, Cold ($\Delta k/k_1 k_2$)		
Outer Rod Stuck	0.158	0.154
Inner Rod Stuck	0.161	0.160

1. THRU 5. LOW ENRICHED FUELED CORE
 6. " 8. HIGH " " "
 9. CONTROL RODS
 10. WATER SLOTS
 11. WATER & CANS
 12. FLOW SKIRT
 13. WATER REFLECTOR
 14. POISON PIN UNIT CELL
 15. ZIRCALOY PLUGS

16. CORE BARREL, THERMAL SHIELD
 & WATER
 17. REACTOR VESSEL
 18. NON-DIFFUSION REGION (FLUX=0 AT
 EXTRAPOLATED VESSEL BOUNDARY)
 19. LOW ENRICHED REMOVABLE SUB-ASS'Y
 20. HIGH " " " "

032-2094

Fig. II-1 LOFT Quarter Core PDQ Model

<u>REGION</u>	<u>COMPOSITION</u>	<u>OUTER RADIUS (cm)</u>
1	2.44 % ENR. FUEL	13.29
2	H ₂ O AND SS CANS	14.69
3	ZIRCALOY	16.25
4	H ₂ O AND SS CANS	17.40
5	2.44 % ENR. FUEL	39.25
6	H ₂ O AND SS CANS	40.56
7	ZIRCALOY	42.14
8	H ₂ O AND SS CANS	43.36
9	5.33 % ENR. FUEL	52.58
10	H ₂ O AND SS CANS	53.10
11	SS	54.06
12	H ₂ O	60.96
13	H ₂ O AND SS	73.66
14	Fe	82.55

Fig. II-2 LOFT Radial FOG Model.

Fig. II-3 Core Reactivity vs. Core Temperature.

Fig. II-4 Reactivity vs. Fuel Temperature.

Fig. II-5 Hot Core (50 Mw) Reactivity vs. Moderator Void Fraction.

Fig. II-6 Axial Power Distribution at the Beginning of Core Life.

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Fig. II-7 Neutron Multiplication vs. Rod Insertion Depth
for the Hot Reactor Case.

Fig. II-8 Worth of all Twelve Rods (Banked) vs. Rod Insertion Depth for the Hot Reactor Case.

Fig. II-9 Peak to Average Power Ratio vs. Rod Insertion Depth for the Hot Reactor Case.

Fig. II-10 Neutron Multiplication vs. Rod Insertion Depth for the Cold Reactor Case.

Fig. II-11 Worth of all Twelve Rods (Banked) vs. Rod Insertion Depth for the Cold Reactor Case.

Fig. II-12 LOFT Zircaloy Core Neutron Flux ($\times 10^{13}$)

Fig. II-13 LOFT Zircaloy Core Neutron Flux ($\times 10^{13}$)

Fig. II-14 LOFT Zircaloy Core Neutron Flux ($\times 10^{13}$)

II-15 LOFT Zircaloy Core Neutron Flux ($\times 10^{13}$)

Fig. II-16 LOFT Zircaloy Clad Core Point to Average Power.

CHAPTER III

ANALYSIS OF REACTOR OPERATION USING ZIRCALOY-4 CLADDING

III. ANALYSIS OF REACTOR OPERATION USING ZIRCALOY-4 CLADDING

As discussed in Section II of this addendum, the conversion to Zircaloy-4 cladding has resulted in a change in the nuclear characteristics of the LOFT reactor. To demonstrate the effect of this change on the excursion behavior of the reactor under accident conditions from that reported in Section IX of the PSAR⁽¹⁾, several postulated accidents have been analyzed. Specifically, these include the startup accident, control rod withdrawal at power accident, and the rod ejection accident.

Since the mechanical design concept of the reactor system has not changed, no further safety analyses have been included concerning the response of the system to mechanical failures such as pipe rupture or steam generator failures. It is recognized, however, that an extensive analysis is necessary of all phases of the plant design and test program to assure the integrity and safe operation of the plant. This analytical effort will continue throughout the design and construction phases of the LOFT facility and experimental assembly. The emphasis in this addendum has been placed upon the two situations which could potentially result in the greatest release of fission products to the environment. These two situations, the loss-of-coolant test and the loss-of-coolant accident which is the maximum credible accident, are discussed in detail in later sections of the addendum.

A. Startup Accident

The establishment of conditions resulting in uncontrolled rod withdrawal during startup requires multiple failures in the nuclear instrumentation and control system with a simultaneous violation of procedure by the operator. Such unrelated and unrealistic sequential accident assumptions are analytically imposed on the LOFT system (with Zircaloy-4 clad fuel pins) to demonstrate the inherent safety included in the design of the system.

Continued withdrawal of the control rods inserts reactivity at a rate determined by the degree of insertion in the core as demonstrated in Figure III-1 and by the rod removal speed. Continuous withdrawal of all 12 rods at the maximum rate of 4 inches per minute and at the most effective position in the core corresponds to a reactivity insertion rate of $0.128 \Delta\rho/\text{min}$.

The time history of the LOFT transient, postulating the "high power" (55 Mw) scram as the only means of terminating rod withdrawal, is shown in Figure III-2. Rod withdrawal is initiated at time zero from the maximum insertion depth. As can be seen, the Doppler effect terminates the power rise before the scram system shuts the reactor down. This is typical of low-enriched UO_2 pressurized-water reactors because of the large prompt negative Doppler coefficient exhibited by this type of reactor.

In order for rod withdrawal to proceed uncontrolled, even the "high power" scram must be assumed to have failed. The effects of uncontrolled rod withdrawal are shown in Figure III-3 with no credit taken for the negative reactivity effect of void formation which will become significant at average fuel temperatures above 2000°F . Under these conditions the reactor will eventually reach a minimum burnout ratio of 1.15 approximately 32 seconds after the reactor power exceeds the 55 Mw "high power" level. Core damage would be expected to occur after this time.

A comparison of the LOFT Zircaloy-4 clad core to the stainless-steel clad core shows the negative Doppler reactivity coefficient to be slightly less for the Zircaloy clad core resulting in a slightly higher power peak and energy release. The Zircaloy clad core has a longer neutron lifetime making the transient period longer, but giving little effect in limiting the total energy output. The combined effects result in a system having only minor differences in kinetic behavior from the stainless-steel clad core.

B. Rod Withdrawal at Power Accident

The factors which determine the reactor transient during control rod withdrawal "at power" (50 MW operation) are the same as for control rod withdrawal during startup, i.e., a smaller Doppler reactivity coefficient and longer neutron lifetime for the Zircaloy-4 clad core. A preliminary analysis indicates no significant difference in dynamic behavior between the Zircaloy-4 clad system and the stainless steel clad system for the "at power" condition.

C. Rod Ejection Accident

The control rod ejection accident is assumed to result from the failure of all the flange bolts on the control rod support nozzle while the reactor is operating at full power. Such an accident could only result from fatigue failure of the flange bolts or gross overpressurization of the primary system. Fatigue failure of the flange bolts is extremely unlikely for the LOFT system due to the relatively short operating history of the reactor and the regular inspection of the flange bolts throughout the experimental program. System pressure is controlled at 2300 psi by the pressurizer under normal operating conditions with extreme overpressurization controlled by a rupture disc and relief valve set to release at 2580 psi. Inasmuch as the entire primary system will be pressure tested at 3700 psi, it is not considered that flange bolt failure due to gross overpressurization can result. Although the occurrence of flange bolt failure is very unlikely the accident has been analyzed for the Zircaloy-4 clad core to ascertain the possible consequences of a rod ejection.

One of the major objectives of this analysis was to determine the pressure transient in the primary system to ascertain if system integrity could credibly be violated as a result of the accident. The most important factors determining the pressure transient are: (1) mass flow from the system decreasing the pressure, (2) heat addition and subsequent steam formation in the core increasing the

system pressure, and (3) flow into the pressurizer resulting from a pressure surge inherently reducing the excessive pressure in the system. If the mass flow out of the system is considered alone, the system pressure decreases by an average of 670 psia/sec until the saturation pressure of 960 psia is attained, after which the pressure decreases more slowly.

Another major objective was to determine the possibility of a zirconium-water reaction taking place if sufficiently high clad temperatures were attained during the rod ejection accident. The related problem of hydrogen burning after its dissociation from water in the zirconium-water reaction was also considered.

The controlling factor in this accident is the reactivity insertion which results from rapid expulsion of a control rod poison section from the operating critical reactor. A FOG-3⁽²⁾ axial window-shade model was employed to determine the worth of one ejected poison section from the rod bank at an insertion depth of 45.5 cm. Simulated control rod poison was represented by flux-weighted constants and radial buckling from two dimensional PDQ⁽³⁾ calculations for "stuck rod" and "rods out" reactor configurations. These calculational results when compared to reactor values corresponding to a rod insertion depth of 45.5 cm reflect the worth of an ejected inner control rod. Calculational comparisons of the reactivity worth of an outer control rod with an inner control rod from a "just critical" rod bank position shows that ejection of an inner rod results in a greater reactivity insertion (by a small margin) than ejection of an outer rod.

The transient analysis was performed using the IREKIN⁽⁴⁾ kinetics code with the desired subroutines programmed in conjunction with the basic code. These program subroutines were designed to represent the Doppler reactivity feedback as a function of fuel temperature, moderator expansion feedback as a function of moderator temperature, coolant flow, and an overall heat transfer coefficient to define heat flow from the system. Consideration was given to void formation and

pressure response which were concluded to result in a net negative reactivity effect. However, this reactivity effect was not incorporated in the IREKIN kinetics analysis which is in effect a conservative approach.

The power, temperature, and pressure history of the reactor core are shown in Figure III-4 with a scram initiated electronically by a "high power" signal and without consideration of the negative reactivity effect of void formation resulting from the coupling of increased power and decreased pressure. The power peaks at 500 MW at 0.1 seconds after the control rod is ejected and then recedes because of the Doppler broadening effect at elevated fuel temperatures. After a lag time of 0.25 seconds resulting from electronic delay in the scram system, the remaining 11 control rods start to insert causing a prompt drop in power. The power then follows the 80 second half-life decay of delayed neutron power. The average fuel temperature has increased 170°F from an initial value of 1300°F and then decreases following scram. It is assumed that coolant flow continues throughout the transient until the decreasing system pressure precludes pump operation. Due to coolant leakage from the vessel, the system pressure continually decreases from the time of rod ejection until a saturation pressure of 960 psi is reached.

The core behavior was also analyzed for the unlikely event of failure of all the flange bolts permitting the rod to eject from the core and a simultaneous complete failure of the scram system. The effects are shown in Figure III-5. The results are identical to the previous case (Figure III-4) until the time that the remaining control rods are normally inserted, but then the Doppler reactivity feedback causes a power reduction to 88 MW which results in an average fuel temperature of approximately 1700°F. The "hot spot" fuel temperature will reach 5000°F with some fuel melting at the pellet centerline and the burnout ratio will reach 1.25 compared to a minimum burnout ratio of 1.15.

Comparison of the rod ejection accident analysis for the Zircaloy-4 clad system with the stainless-steel clad system indicates similar effects as found in the startup analysis. The Doppler reactivity coefficient is found to be less for the Zircaloy-4 clad core resulting in higher power peaking and higher energy content in the fuel to compensate for any added reactivity. The prompt neutron lifetime is longer resulting in a slowing of the system dynamics. Under these conditions the rod ejection accident is not significantly different for this core than for the stainless-steel clad core.

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- (1) J. M. Waage, Editor, "Preliminary Safety Analysis Report - LOFT Facility," IDO-16981, April 1964.
 - (2) NAA-SR-6104, "The FOG One-Dimensional Neutron Diffusion Equation Codes," Atomics International.
 - (3) WAPD-TM-230, "A Program for the Solution of the Neutron-Diffusion Equations in Two Dimensions," Bettis Atomic Power Laboratory, Westinghouse Electric corporation.
 - (4) IDO-16928, "IREKIN Program for the Numerical Solution of the Reactor Kinetics Equations," Phillips Petroleum Company.

Fig. III-1 Differential Rod Worth vs. Rod Insertion Depth

Fig. III-2 LOFT Startup Analysis With Scram

Fig. III-3 LOFT Startup Analysis w/o Scram

Fig. III-4 LOFT Rod Ejection Accident With Scram

Fig. III-5 LOFT Rod Ejection Accident w/o Scram

CHAPTER IV

ENGINEERING SAFETY ANALYSIS OF THE LOSS-OF-COOLANT TEST

IV. ENGINEERING SAFETY ANALYSIS OF THE LOSS-OF-COOLANT ACCIDENT

A. Coolant Blowdown

The mechanical response of the Zircaloy-clad core during blowdown is essentially identical to that discussed for the stainless-steel clad core in the LOFT Preliminary Safety Analysis Report. The pressure drop across the core will increase slightly but not significantly because of the reduction in flow area due to the increased cladding thickness.

One basic difference between the cases analyzed for the two different cladding materials is the location assumed for the break in the primary coolant system. When the safety analysis was performed for the stainless-steel clad fuel, the effects of metal-water reactions were not considered because available data at this time indicated that little, if any, reaction would occur with the stainless steel. Any water remaining in the vessel then represented a potential source of cooling for the core, and consequently, the worst case was chosen as the one whereby the least amount of water would remain in the vessel. For LOFT, this case corresponds to a break in the bottom inlet primary coolant line.

In the case of Zircaloy cladding, metal-water reactions can occur which will contribute significantly to heating the core. The presence of water in the form of steam enhances the rate and extent of these reactions once they have begun, and the energy generated during the metal-water reaction will rapidly override the previous cooling effect of the steam. Thus, the presence of water in the reactor vessel represents a worse case for the Zircaloy-clad core. A top break, one in the primary coolant outlet line, allows the greatest amount of water to remain in the reactor vessel and provides a well defined flow path through the core for the steam. Consequently, the top break was chosen for the worst case analysis of the LOFT Zircaloy-clad core.

The particular effects of blowdown on the ensuing phases of core heatup and fission product release are included as a part of the detailed discussions of these phases in the following sections.

B. Core Heatup

The change to a Zircaloy-4 cladding, 25 mils thick, in place of the 15 mil stainless-steel cladding necessitated a re-examination of the thermal behavior of the LOFT core heatup and meltdown. Basically, the heatup is initiated in the same manner as for the stainless-steel clad core: blowdown occurs and the power generation drops promptly to 6 - 7% of the reactor full power as the fission process is shut down by void formation and control rod scram. Because of the high coolant discharge rate and resulting high coolant velocities, high heat transfer rates from the core will exist during this period of time causing the fuel and cladding temperatures to decrease rapidly.

With the loss of forced convection cooling, the core will undergo a rapid heatup due to the inherent fission product decay energy. When high temperatures are reached, a chemical reaction of the zirconium cladding with available water or steam can occur. The results of this reaction are formation of a layer of zirconium oxide on the surface of the fuel pin and the evolution of gaseous hydrogen and thermal energy. The occurrence of this reaction will provide an additional heat source to the core plus produce an oxide layer which may be a significant barrier to the transport of molten cladding. The heatup will continue until other heat removal mechanisms such as radiation and convection become large enough to balance the total heat generation rate or until the onset of melting enhances the heat removal rate.

Recent experimental evidence⁽¹⁾ is now available which demonstrates that a reaction between stainless steel and steam can and does occur, but not until the melting point of the stainless steel is reached. This reaction is accompanied by a foaming of the metal into a spongy mass which greatly increases the surface area and allows a faster rate of

reaction for the stainless steel than was found for similar experiments with Zircaloy-2. In the case of an actual reaction, however, the foaming metal may clog the flow channels and reduce the rate at which steam can reach the reacting material. Regardless of the cladding material used, once the metal-water reaction begins, the rate of reaction will rapidly become limited by the rate at which steam is supplied for reaction. The zirconium-water reaction, which produces about eight to nine times as much energy per weight of steam reacted as does the stainless steel, then represents the more severe hazard potential. The loss-of-coolant test environment appears to provide the necessary conditions for a zirconium-water reaction to take place; thus, the occurrence of metal-water reactions is a definite possibility. The large energy and hydrogen generation associated with this reaction can alter the temperature and pressure within the containment vessel and can affect the thermal behavior of the reactor core and greatly affect the release of fission products from the fuel.

In the following discussions, the zirconium-steam reaction in general is examined, and the experimental and theoretical data available for analyzing this reaction are presented. The important factors during heatup including metal-water reactions are investigated independently and the limiting mechanisms defined. These factors are then coupled together and applied to a core heat transfer model and the core temperature history including metal-water reaction energy and hydrogen release as functions of time are calculated. From these data, models for fission product release and energy release to the containment are obtained.

1. Zirconium-Steam Reaction

At temperatures above 800 - 900°C zirconium has been observed to undergo reaction with steam resulting in the generation of heat and gaseous hydrogen⁽²⁻¹⁰⁾. Under the conditions likely to exist in a reactor core after a loss-of-coolant accident, it has been found that an external heat source is necessary to initiate and to sustain the

zirconium-water reaction. Experiments have also been performed by dispersing molten zirconium particles underwater and allowing them to react. The extent of this reaction depends upon the energy input and size of the particles. For molten zirconium droplets falling into liquid water, reactions on the order of 10 - 15% completion have been observed. In all cases where the zirconium reacted underwater, it was found that rapid boiling heat transfer soon caused the reaction to be quenched.

2. Available Experimental and Analytical Data

The potential for metal-water reactions to occur during various nuclear reactor accidents has been recognized for some time and several investigators have undertaken analytical and experimental efforts to better understand the hazards associated with these reactions⁽¹¹⁻¹⁹⁾. For zirconium in the presence of excess steam, the rate of reaction has been found to follow a parabolic oxidation law which is characteristic of a reaction controlled by a solid state diffusion process. Several different values of the constants for the zirconium-steam parabolic rate law have been reported. These constants are discussed in report ANL-6548⁽⁹⁾ and the recommended values for the constants given.

The parabolic rate law given in integrated form for isothermal conditions is:

$$W^2 = 33.3 \times 10^6 t \exp \left(\frac{-\Delta E}{RT} \right) \quad (1)$$

where:

W = weight of zirconium reacted per unit surface area, mg/cm²

t = time, sec

ΔE = 45,500 cal/g

R = 1.987, cal/g mole °K

T = absolute zirconium temperature, °K.

This equation can be reduced to a more convenient differential form based on the radius of a fuel pin which permits inclusion of variable temperatures:

$$-\frac{dr}{dt} = \frac{K}{(r_0 - r)} \exp(-\Delta E/RT) \quad (2)$$

where:

r = radius of unreacted metal in fuel rod

r_0 = original radius of fuel rod

K = rate law constant.

It can be readily shown that the reaction rate given by the parabolic oxidation law is a rapidly increasing function of temperature. During heatup, the energy supplied by the metal-water reaction and the decay heating cause a rapid temperature rise which, in turn, results in an accelerating reaction rate predicted by the parabolic rate law. As the reaction rate increases, the rate limiting mechanism will cease to be the solid state diffusion process, and the reaction will then be controlled by the availability of one of the reactants. The reaction continues until the availability of one of the reactants is exhausted or the temperature of the zirconium has been reduced sufficiently to quench the reaction.

At low temperatures, the parabolic rate law severely limits the rate at which metal-water reactions can occur. Although little reaction occurs during the time period when the parabolic rate law governs the reaction, this relationship is essential to determine the time at which another mechanism will limit the reaction. This relationship is then essential in determining the time history of the metal-water reaction heat source and the subsequent effect on core heatup.

3. Factors Affecting Heatup and Metal-Water Reaction

a. Decay Heating

During nuclear operation of a reactor core, radioactive isotopes are built up within the reactor fuel as products of the fission process. In most cases, the concentration of these isotopes will reach

an equilibrium value during nuclear operation; however, after shutdown, these products remain in the fuel and continue to undergo radioactive decay according to their respective decay chains. The energy generated during this process can be absorbed in the core materials and become a substantial heat source. Essentially all the fission product decay energy is emitted as beta particles and gamma radiation. The following assumptions have been developed to describe the decay heat source:

- 1) the spatial distribution of the fission product concentrations within a homogenized core is the same as the thermal neutron flux distribution;
- 2) beta radiation is readily absorbed in heavy materials, thus it is assumed that all the beta energy is absorbed at the point of generation.

The gamma heat source was obtained as follows: (1) the core was assumed to be a homogeneous medium and a gamma emission source was calculated based on thermal neutron flux profiles; (2) a seven-group gamma energy spectrum was obtained using available data⁽²⁰⁾; (3) the radial gamma heating was then determined using the Grace II code⁽²¹⁾ on the IBM-7040 for various decay times. It was found that the spatial variation of the gamma heating did not significantly change with time and that a single radial gamma heating profile could be used to represent the spatial distribution of the gamma heating over the time span of interest. By adding the beta and gamma heat sources, a radial heat source profile and the time variation of the heat source can be found.

Examination of the radial neutron flux profile for the Zircaloy-clad core show that the profile is essentially the same as that for the stainless-steel clad core. The bulk of the gamma absorption occurs in the denser material, i.e., the fuel, and since the amount and configuration of the fuel remains unchanged with the new cladding, the gamma absorption for the Zircaloy-clad core is considered the same as for the stainless-steel clad core. The decay heat source, therefore,

is approximately the same for both cladding materials. The radial heating profile and the variation of the heat source with time are shown in Figures IV-1 and IV-2. The axial heat source variation is assumed to be a "chopped cosine" with an axial peak-to-average value of 1.5 symmetrical about the core vertical center as shown in Figure IV-3.

b. Metal-Water Reactions

The analysis of metal-water reactions following a loss-of-coolant accident requires an accurate knowledge of the physical sequence of events occurring within the reactor vessel. Two factors are important in analyzing the effects of the energy release from metal-water reactions: (1) the extent of reaction, and (2) the rate of reaction. Since the rate of reaction is limited by different mechanisms at different times in the course of the accident, it is possible to analyze the metal-water reaction as occurring in three distinct phases: (1) kinetics-limited phase, (2) steam-limited phase, and (3) quenching phase. A discussion of each of these phases and the limiting phenomena is presented below.

(1) Kinetics-Limited Phase

This phase of the reaction is limited by the solid state diffusion process and is expressed mathematically by the parabolic rate law. Basically the limiting factors are the oxide film thickness and the strong temperature dependence of the rate law. At low temperature, the reaction rate is negligible; however, once temperatures of 800 - 900°C are reached, the zirconium-steam reaction becomes significant and the additional heat source must be considered. For LOFT, the reaction rate given by the parabolic rate law is:

$$-\frac{dr}{dt} = \frac{0.0611}{r_o - r} \exp\left(\frac{-41,220}{T}\right) \quad (3)$$

where:

$\frac{dr}{dt}$ is expressed in inches/sec,

$(r_o - r)$ is expressed in inches, and

T is expressed in °R.

The metal-water reaction heat source for the cladding can then be written in terms of the parabolic rate law. Based on a unit length of fuel pin, the heat generation rate becomes:

$$\frac{dq}{dt} = h_r \rho 2\pi r \left| \frac{dr}{dt} \right| \quad (4)$$

h_r = heat of reaction (assumed constant = 6.11 Btu/gm)

ρ = density of zirconium

r = radius of reacting interface in cladding

$\left| \frac{dr}{dt} \right|$ = magnitude of $\left(-\frac{dr}{dt} \right)$ given by parabolic rate law.

This expression is valid when an excess of reactants is present for reaction. At high temperatures, the reaction rate from the parabolic rate law approaches an infinite value and the availability of one of the reactants will then become the limiting mechanism.

(2) Steam-Limited Phase

Before a zirconium-water reaction can occur, both reactants, zirconium and water or steam, must be present and in contact with one another. The maximum amount of the reaction is limited by the amount of the limiting reactant present in the system. The rate of reaction, when not limited by the rate law, will be governed by the rate of transportation of the reactants (steam) to the surface or the transport of the products (gaseous hydrogen) from the reacting surface.

The maximum amount of zirconium which could react is essentially all that contained within the reactor vessel; however, only part of the zirconium has an associated heat source sufficient to initiate the metal-water reaction. The amount of zirconium actually capable of reacting was assumed to be that contained within the active length of the core after shutdown because of the large decay heat source. This includes the fuel pin cladding and the zirconium filler pieces between the control rod blades. The zirconium not in the core, i.e., the control rod followers, fuel pin end caps, etc., will not achieve temperatures at which significant reaction can occur, and thus, was not assumed to react.

The limiting mechanism in the zirconium-steam reaction process following the kinetics-limited phase, will be the transport of steam to the reacting surface. This portion of the reaction will be referred to as the steam-limiting phase. Steam-limiting means that the rate of metal-steam reactions at all times is limited to a value equal to or less than the corresponding rate at which steam is supplied for reaction. The parabolic rate law only limits the reaction at low temperatures. Thus, if the rate at which steam is supplied for reaction can be found, an upper limit to the rate and extent of the metal-water reaction can be determined. This fact is the primary basis for the LOFT analysis as will be shown later.

(3) Quenching Phase

The large heat sources afforded by decay heating and metal-water reactions combined with the poor heat removal capability will eventually cause melting within the reactor core. When this happens, portions of the core are free to fall into the water pool at the bottom of the reactor vessel, thus providing another means by which the reactants can be brought together to further promote a metal-water reaction. An additional energy release amounting to approximately 10 - 15%⁽²⁻¹⁰⁾ reaction of the zirconium not already consumed can occur before the reaction is quenched. The additional energy release, however, will be very minor compared to the decay and metal-water reaction energy already stored as sensible and latent heat in the molten core material, and which will now be released rapidly by boiling water which will appear as steam in the containment.

(4) Steam Availability Potential for LOFT

The source of steam for metal-water reactions is determined by the amount of water remaining in the primary system after the initial blowdown. This amount depends, to a large extent, on the size and location of the break within the primary coolant system. A break occurring at a level near the top of the reactor vessel would

allow the most water to remain in the reactor vessel. Immediately following blowdown, the temperature of the metallic components in the primary system will be near the operating coolant temperature ($\sim 540^{\circ}\text{F}$), whereas the water temperature will be the saturation temperature corresponding to the containment pressure ($\sim 250^{\circ}\text{F}$). The excess sensible heat contained in the metal will then be dissipated by boiling of the water in contact with the metal surface. If the break location is such that the low-resistance flow path for the steam lies through the reactor core, the steam and zirconium will be brought together and a metal-water reaction may occur. This situation for LOFT is again typical of a top break in the outlet primary coolant line. If the low-resistance flow path is not through the core, i.e., a bottom break, diffusion or natural convection, may provide the only mechanisms for bringing the reactants together; thus, resulting in a relatively insignificant reaction compared to the top break case.

For a top break, the water level immediately after the initial blowdown could be within or above the reactor core. For these conditions, the decay heating in that portion of the core beneath the water level will contribute to the steam generation and, thus, increase the amount of steam available for reaction. However, the steam generation rate for all water levels from the core bottom to about the core centerline causes the water level to drop beneath the core before the reaction rate becomes significant. The metal-water reaction energy release will be approximately the same for all these water levels. The effect of water levels above the center of the core is to delay heatup of the central portions of the core and allow the hot vessel components to dissipate most of their excess sensible heat by generating steam before regions within the core are capable of reacting with this steam. The lower steam availability at the time at which steam-limiting does occur reduces the predicted rate and extent of reaction. Results of preliminary calculations based on a separated-flow coolant expulsion model predict a water level at the bottom of the core support structure

following blowdown through a large break in the vessel outlet line. This level was then chosen for the LOFT analytical studies, since it represents the worst case for LOFT and is also the most likely to occur.

(5) LOFT Steam Availability Calculation

An estimate of the steam availability for the steam-limited portions of the analysis was obtained in the following manner:

- 1) a top break was assumed such that the water remaining following the initial coolant discharge was at a level in the reactor vessel just below the core;
- 2) steam was assumed to be generated by the hot vessel walls and components located below the core and was assumed to flow upward through the core;
- 3) steam flow was assumed to be uniformly distributed radially across the core as the core flow orificing will tend to distribute the steam in this manner (in addition, no flow external to the core, i.e., around the lower support barrel and thermal shield, was assumed to occur);
- 4) horizontal movement of steam while passing axially through the core was not accounted for as the expansion of steam in the hotter portions of the core would tend to force more steam flow through the cooler portions of the core, thus, reducing the rate of metal-water reaction;
- 5) analytical models were formulated for calculating the transient heat transfer from the submerged portions of reactor vessel bottom head, reactor vessel cylindrical walls, and inlet flow distributor;
- 6) the outside of the reactor vessel was assumed to be insulated; and
- 7) the thermal resistance of the water film in contact with the walls was assumed to be negligible compared to the conduction resistance in the walls themselves.

Based on these assumptions, the heat transfer rate to the water was calculated using the Heat I code⁽²²⁾. The corresponding steam generation rate was then calculated for each of the submerged components and summed to give the total steam generation rate. The resultant steam generation rates are shown in Figures IV-4, IV-5, and IV-6.

(6) Steam-Limited Reaction Rate

By calculating the steam generation rate per fuel pin, and assuming all of the steam reacts uniformly over a specified length of the fuel pin, the following expression can be written for the reaction rate in terms of the steam generation rate and the radius of the reacting interface:

$$\frac{\text{Steam generation rate lb/sec}}{\text{No. of pins in core}} \times \frac{\text{lb Zr reacted}}{\text{lb steam}} = \rho_{\text{Zr}} 2\pi r L \left(\frac{-dr}{dt} \right) \approx \rho_{\text{Zr}} 2\pi r_o L \left(\frac{-dr}{dt} \right) \quad (5)$$

where:

$$\rho_{\text{Zr}} = \text{density of zirconium} = 0.235 \text{ lb/in}^3$$

$$r = \text{radius of interface, in.}$$

$$r_o = \text{initial pin radius} = 0.2055 \text{ in.}$$

$$L = \text{length reacting per fuel pin, in/pin}$$

$$\frac{dr}{dt} = \text{rate of change of radius, in/sec}$$

which for a given length reduces to $-\frac{dr}{dt} = \frac{K}{L}$ (steam generation rate), lb/sec

$$K = \text{constant} = 2.506 \times 10^{-3} \frac{\text{in}^2}{\text{pin} - \text{lb steam}} \quad .$$

Thus, two expressions have been written for $\frac{dr}{dt}$, one from the parabolic rate law and the other from the steam generation rate, the smaller of which at anytime will control the reaction. The problem then remains to determine $\frac{dr}{dt}$ as a function of time and to integrate to find the total amount of zirconium reacted and the accompanying energy and hydrogen release.

4. Analytical Solution for Heatup and Metal-Water Reactions

Having outlined the general problem of heatup when accompanied by metal-water reactions and having defined the limiting mechanisms, this information is now applied to obtain quantitative data for predicting meltdown, energy release, hydrogen release, and finally fission product release to the environment.

Once a calculational transient heat transfer model of the core has been established and the time and spatial dependence of the heat sources are known, the temperature history of the various parts of the core can be determined. The metal-water reaction, however, represents one of the heat sources to the core, and that phase of the reaction controlled by the parabolic rate law is highly dependent upon the temperature. The ideal solution would be to obtain the core temperature history and metal-water reaction energy and hydrogen generation simultaneously by use of a computer code and a single core model. However, a computer code with this capability is not available; therefore, a different approach was necessary to solve the problem. The analytical technique used to obtain the needed information required two separate calculations, the first to evaluate metal-water reaction energy and hydrogen generation as functions of time, and the second to obtain the temperature history of the reactor core. The approach used was to determine the metal-water reaction energy and hydrogen generation as functions of time using only a simple heat transfer calculation prior to steam-limiting. The resulting heat source was then used as input to the heat transfer calculation which determines the temperature history within the core. From these data, models for meltdown, release of decay heating, combustion energy, metal-water reaction energy, and release of fission products were formulated.

a. Metal-Water Reaction Calculations

The method used for calculating metal-water reaction energy generation is an extension of the analysis performed by Louis Baker, Jr.,

et. al., for LOFT⁽¹⁹⁾. The extended analysis includes a time varying steam availability to the core. To obtain the reaction rate from the parabolic rate law, the temperature as a function of time must be known. Therefore, a simplified energy balance was made which resulted in a differential equation relating temperature and time which can be solved simultaneously with the reaction rate equation. Integration of the reaction rate then gives the amount of zirconium reacted as a function of time and the corresponding energy and hydrogen generation.

The LOFT core was first modeled as a cylinder and divided radially and axially into regions. Using the space and time dependence of the decay heating given in Figures IV-1, IV-2, and IV-3, a decay heat source was found to represent each region. The other source of heat is from the metal-water reaction itself and is given by equation 4. The simplified energy balance was obtained as follows:

- 1) adiabatic conditions were assumed for each region, i.e., no heat transfer to or from the region;
- 2) the initial temperature of the fuel and cladding after blowdown was 540°F (this is the approximate overall average temperature calculated based on the conditions predicted to occur during blowdown);
- 3) the time rate of temperature rise in each region was assumed to be uniform throughout the material in the region from the decay heat source;
- 4) the change in dimensions and thermal properties of the cladding due to formation of the oxide on the surface were neglected; and
- 5) the chemical energy of the metal-water reaction was distributed across the fuel pin, thus a flat temperature profile was maintained.

The general energy balance then becomes:

$$(\Sigma mc_p) \frac{dT}{dt} = \text{heat generation rate from metal-water reactions} \\ + \text{heat generation rate from decay heating}$$

which reduces to:

$$\frac{dT}{dt} = \frac{2.992 \times 10^7}{\Sigma mc_p} \frac{r}{(r_o - r)} \exp \left[-\frac{41,220}{T} \right] + \frac{P/P_{avg}}{\Sigma mc_p} [B(t)] \quad (6)$$

where:

$\frac{dT}{dt}$ = rate of change of absolute temperature with time, $^{\circ}R/sec$.

Σmc_p = sum of mass x heat capacity for materials in which energy deposited (LOFT fuel and cladding - 384.3 Btu/ $^{\circ}F$)

r = radius of unreacted cladding, inches (an initial value of $r_o - r$ of 0.0001 inches was used)

r_o = outer radius of fuel pin = 0.2055 inches

T = absolute temperature, $^{\circ}R$

P/P_{avg} = time function for decay power for core at radial center, average axial, Btu/sec.

The reaction rate from the parabolic oxidation law is given by equation 3:

$$-\frac{dr}{dt} = \frac{0.0611}{r_o - r} \exp \left(-\frac{41,220}{T} \right)$$

The reaction rate determined by the steam generation rate is given by equation 5:

$$-\frac{dr}{dt} = \frac{2.506 \times 10^{-3}}{L} \times (\text{steam generation rate, lb/sec})$$

A computer program was written for the IBM-7040 to simultaneously solve this energy balance and the parabolic rate law until the reaction rate equals that corresponding to the steam supply. At this time, the steam-limited phase begins and the code continues to calculate energy and hydrogen generation based only on the steam availability until either the steam or zirconium is exhausted. Solution of the energy balance was not necessary for obtaining energy during the steam-limited phase and consequently no temperature calculation was made

using this code for this phase of the reaction. A more detailed description of the models used and the manner in which the calculations were made is presented in IDO-17132⁽²³⁾.

The results of these calculations are shown in Figures IV-7 and IV-8 which give the energy generation rate and the integrated energy generation as functions of time for the overall core. Approximately 70% of the available zirconium reacts over a one hour time period assuming no core disassembly. The potential energy release from combustion of the hydrogen gas is also included in the energy curves assuming the hydrogen burns as it is generated. These curves then show the time variation of the total overall rate and extent of metal-water reaction energy generation. With this information, the second calculation, that of temperature in various regions within the core as a function of time, can be made using a model which allows heat transfer within and from the core.

b. Core Temperature History

The variation of temperature with time and space for the LOFT core was found using the Heat-I code⁽²²⁾. The core was modeled as a cylinder composed of rings of fuel and cladding separated by gaps very similar to the model used for the heatup analysis of the stainless-steel clad core. Two additional rings were included to represent the control rods and filler pieces, and an additional ring was included outside the core to represent the flow skirt. An effective thermal conductivity was then used to approximate radiation heat transfer across the gap regions. Previous calculations have shown that axial heat transfer in the core is not important; thus, it was possible to approximate the axial temperature by running several radial cases corresponding to different axial positions in the core.

From the energy data given by the metal-water reaction calculations, the metal-water reaction heat source was approximated and input to the heat transfer code. This was done in the following manner: (1) the control rod rings will provide a heat sink within the

core and also act as a barrier to radiation heat transfer and thus represent the logical place for separating core regions. Three radial regions were then defined for the metal-water reaction heat source: the first region being from the core center to the first control rod ring, the second between the two control rod rings, and the third between the outer control rod ring and the flow skirt. (2) The time at which each of these regions would begin significant reaction was approximated from the metal-water reaction code output. All the steam passing through this region was then assumed to react until the zirconium contained within the region was completely reacted. (3) The outer region contained only a few rings of fuel, and from the metal-water reaction calculation it was found that the steam availability for this region was low enough at the time at which steam limiting occurred that only partial reaction resulted. Since the outer portion of the core will experience the greatest loss of heat during heatup, the temperature at which steam limiting actually occurs will not be reached until later. Because of this, no metal-water reaction heating was assumed for the outer radial region. (4) Radiation heat loss was allowed from the flow skirt to a heat sink at 540°F. (5) When metal-water reactions begin in the regions, the calculated temperatures rise rapidly and soon exceed the melting and boiling points of all the materials present. Obviously, core disassembly will occur before temperatures of this magnitude can exist. Radiation heat transfer is highly dependent upon the temperature level, thus, the fictitious elevated temperatures that might be calculated assuming no disassembly would destroy the value of the heat transfer calculation. To avoid this, a high heat capacity was given to the core materials when they reached fuel melting (~5070°F) thus limiting the temperature level at this value. Radial temperatures were obtained as functions of time for five axial regions in the core in order to predict the time release of fission products. A schematic representation of the 15-region core model is shown in Figure IV-9. The ratio of fission product concentration in any region to the average concentration for the core was assumed equal to the ratio of power in the region to the average core power. The power ratios, fractions of core material, and times of clad and fuel melting are given in Table IV-1. The approximate temperature vs. time curves for the core regions are shown in Figure IV-10.

TABLE IV-I
CORE HEATUP RESULTS

Region	% of Core in Region	Power Factor P_k/P_{ave}	Time When Clad Begins Melting (sec)	Time When Fuel Begins Melting (sec)
1-A	1.522	0.697	830	916
1-B	12.176	0.594	1500	2020
1-C	6.302	0.443	3600	- - -
2-A	1.522	1.59	375	440
2-B	12.176	1.35	533	651
2-C	6.302	1.01	1845	- - -
3-A	1.522	1.91	325	325
3-B	12.176	1.63	458	528
3-C	6.302	1.21	1552	- - -
4-A	1.522	1.59	375	440
4-B	12.176	1.35	533	651
4-C	6.302	1.01	1845	- - -
5-A	1.522	0.697	830	916
5-B	12.176	0.594	1500	2020
5-C	6.302	0.443	3600	- - -

C. Fission Product Release, Transport, and Plateout Model

The fission product release model employed in the radiological analysis for the loss-of-coolant test is based on the temperature data for the steam-limited metal-water reaction presented in Section IV-B, as summarized in Table IV-1, and on fission product release data obtained from experiments conducted at the Oak Ridge National Laboratory (ORNL)⁽²⁴⁻²⁷⁾. The plateout model is based on experiments performed in the United Kingdom⁽²⁸⁾ to study the deposition of fission products on various surfaces and in various environments.

1. Fission Product Release From Fuel

In order to reduce the time required for computer calculations and facilitate the radiological analysis, the 15 region core model presented in the previous section has been reduced to a 3 zone fission product release model. This analytical model is based upon the times at which cladding and fuel melting start to occur in each of the 15 regions.

a. Three Zone Release Model

The times at which cladding and fuel melting start to occur in each of the 15 regions, as presented in Table IV-1, are shown in physical configuration in Figure IV-9. Since fuel melting does not occur in the outer Annulus C, all regions in this annulus have been chosen to comprise one zone (Zone III) for analytical purposes. The remainder of the core has been divided into two zones (Zones I and II) depending upon the temperature ranges at which melting starts to occur. Since the initial melting times in Regions 1B and 5B are significantly greater than those for the remaining regions in Annuli A and B, these two regions are considered as Zone II. The remaining regions in Annuli A and B are combined to form Zone I.

The physical configuration of the 3 zone release model used in the radiological analysis for the loss-of-coolant test is shown in Figure IV-11. The zones are numbered according to the chronological order in which fission products are released.

b. Time of Release

For each of the three zones, it is assumed that the fission product release from the fuel occurs instantaneously at an effective time based on initial cladding and fuel melting times for each region in the zone. The methods used to obtain the effective release time for each zone are discussed below.

(1) Zone I

As shown in Figure IV-11, Zone I includes eight of the 15 regions described in Table IV-1. The effective time at which fission products are assumed to be released from all these regions is based upon the average of the cladding and fuel melting times for each region and is normalized according to the power factor and core fraction for each region. Thus, for analytical purposes, the fuel and cladding in Zone I are assumed to melt simultaneously releasing the fission products at an effective time $t_{eff} = .544$ seconds where t_{eff} is determined by:

$$t_{eff} = \frac{\sum_i (P_k/P_{ave})_i p_i \left(\frac{t_f + t_c}{2} \right)_i}{\sum_i (P_k/P_{ave})_i p_i}$$

where:

- i = regions of fuel in zone
- $(P_k/P_{ave})_i$ = peak-to-average power density for region i
- p_i = percent of core in region i
- t_f = time of fuel melting
- t_c = time of cladding melting

The values used in the above equation for each region i are presented in Table IV-1.

(2) Zone II

As shown in Figure IV-11, Zone II is comprised of only two regions, 1-B and 5-B. Since these two regions have identical characteristics (see Table IV-1), t_{eff} in the above equation reduces simply to the average of the cladding and fuel melting times for either region. Thus, $t_{eff} = 1760$ seconds for Zone II.

(3) Zone III

Since fuel melting does not occur in Zone III, the effective time for release of fission products is determined by replacing

$$\frac{t_f + t_c}{2}$$

with t_c in the equation presented for Zone I. The t_{eff} for Zone III is 2137 seconds.

c. Distribution of Fission Products in Zones

The distribution of fission products in each zone of the core is determined by the expression

$$\sum_i (P_k/P_{ave})_i P_i$$

It is found that 62.6 percent of the total fission product inventory of the core is produced in Zone I, 14.5 percent in Zone II and 22.9 percent in Zone III.

d. Release Fractions

The release fractions for Zones I and II are based on the experimental data obtained from the fuel meltdown experiments performed in the Oak Ridge Research Reactor (ORR) in which irradiated UO_2 fuel elements were melted in an atmosphere of 87% steam, 12% helium, and 1% hydrogen⁽²⁴⁾. This atmosphere was assumed to typify the atmosphere in a pressurized-water reactor following loss-of-coolant since hydrogen could exist in the system either from hydrogen present in the coolant as a corrosion preventative or from hydrogen released during a metal-water reaction.

For Zone III in which fuel melting does not occur, the fission product release fractions are based on ORNL experimental data in which irradiated UO_2 was heated, but not melted, in a pure helium atmosphere⁽²⁶⁾.

The ORNL experimental data used as a basis for obtaining the fission product release fractions for the three zones are summarized in Table IV-2.

The data presented in Table IV-2 for the case of 100% UO_2 melting was obtained with the fuel held near the melting temperature for a 5 minute period. The data for zero UO_2 melting was obtained with the specimens heated for 5.5 hours. For the LOFT radiological analysis, the conservative assumption is made that the fission product release from each zone occurs instantaneously.

Since the data used to obtain the release fractions for Zones I and II are for trace-level irradiation, a correction factor was employed to determine the release for the 920 MWd/T burnup in LOFT. The correction factor was obtained by considering data from two experiments performed at ORNL under conditions in which only the irradiation period was varied. These data are presented in Table IV-3⁽²⁵⁾.

TABLE IV-2
FISSION PRODUCT RELEASE FROM FUEL

Zone	UO ₂ Melted (%)	Atmosphere	Irradiation	Temperature °C	Material Released (%)							
					Xe-Kr	I	Te	Cs	Ru	Sr-Ba	Zr-Ce	UO ₂
I and II	100	87% Steam 12% Helium 1% Hydrogen	Trace		100	95	90	89	21	19	5.3	3.2
III	0	Pure Helium	1000 MWd/T	1980	48.5	63	90	69.5	4.8	~ 10 Sr 51 Ba		

TABLE IV-3

FISSION PRODUCT RELEASE FROM TRACE LEVEL AND 2800 MWd/T FOR
IRRADIATED SS-CLAD UO₂ MELTED IN IMPURE HELIUM

Exp.	Irradiation	Clad	Atmosphere	Material Released (%)									
				Xe-Kr	I	Te	Cs	Ru	Sr	Ba	Rare Earths	UO ₂ Vaporized	
R _t	Trace	SS	Impure Helium	99.5	89.7	92.0	91.3	61.0	2.1	4.6	2.2	-	-
R ₂₈₀₀	2800 MWd/T			99.9	92.2	98.2	98.5	90.4	2.1	6.6	5.1	21.2	

The data from stainless steel-clad UO_2 melted in a helium atmosphere is employed because similar data for zirconium-clad melting in a steam atmosphere are not available. The results are considered to be applicable in correcting for burnup in Zones I and II since all the fuel in these zones is assumed to melt and because the release in a steam atmosphere is not expected to differ significantly from that in a helium atmosphere.

The correction factor (F) for the fission products released from Zones I and II is obtained by:

$$F = \frac{(R_{2800} - R_t) \frac{920}{2800} + R_t}{R_t}$$

where:

F = Correction factor

R_t = Release for trace-level irradiation

R_{2800} = Release for 2800 MWd/T irradiation

The fission product release data used for Zone III was obtained with the fuel temperature held at 1980°C . This temperature closely approximates the melting temperature of zirconium cladding ($\sim 1950^\circ\text{C}$) and is the temperature at which fission product release becomes significant. As can be seen from Figure IV-10, temperatures in the end regions of Zone III (Regions 1C and 5C, Figure IV-9) do not quite reach 1980°C , whereas temperatures in inner regions of Zone III (Regions 2C, 3C, and 4C) slightly exceed 1980°C .

The quantity of fission products predicted to be released from each zone is summarized in Table IV-4. The release percentages were obtained for Zones I and II by multiplying the percent of the total core activity contained in the given zone by the appropriate fission product percentages from Table IV-2 after these percentages have been corrected for burnup as described above. The release

TABLE IV-4

FISSION PRODUCT RELEASE FROM CORESTEAM-LIMITED METAL-WATER REACTION

Core Zone	UO ₂ Melted (%)	Percent of Activity in Zone	Time of Release (sec)	Material Released from Fuel (%)								
				Xe-Kr*	I-Br**	Te	Cs	Ru	Sr	Ba	Zr-Ce	Rest
I	100	62.5	544	62.6	60.1	57.1	57.4	15.2	5.9	6.8	3.3	0.626
II	100	14.5	1760	14.5	13.9	13.2	13.3	3.5	1.4	1.6	0.8	0.145
III	0	22.9	2137	11.1	14.4	20.6	15.9	1.1	2.3	11.7	- -	0.229
Total Release from Core				88.2	88.4	90.9	86.6	19.8	9.6	20.1	4.1	1.000

*100% release of both Xe and Kr was assumed for Zones I and II.

**Bromine was assumed to be released similarly to iodine.

percentage for Sr and Ba in Zones I and II was assumed to be equally divided between the two elements. Since the data presented in Table IV-3 do not include results for Zr and Ce, it was assumed that 5.3% (see Table IV-2) of each of these elements is released.

The percentages in Table IV-3 for Zone III were obtained simply by multiplying the percent of the core activity in Zone III by the appropriate percentages given in Table IV-2.

Although data are not available for elements other than listed, a 1% release of the rest of the fission products in the core was assumed in accordance with previous radiological analyses and in accordance with TID-14844⁽²⁹⁾. A check calculation was performed to compare the results of this assumption with those assuming 0% release of the remaining elements; the resulting doses were not significantly different from those for a 1% release.

The 3-zone fission product release model presented in Table IV-4 differs from that previously presented in IDO-16981 in that data recently made available by ORNL has been used to consider specific release percentages for isotopes other than iodine. The data used for the major portions of the assumed fission product release (Zones I and II) applies to a steam-helium-hydrogen atmosphere which is believed to be more nearly representative of the LOFT atmosphere than the previously assumed helium atmosphere. The applied data is believed to result in a conservative release model since modifications were not employed to account for differences in fuel surface-to-volume ratios and percentages of UO₂ vaporization. These modifications were not made due to insufficient data for the case of LOFT.

The percent of the core activity in each of the three release zones and the time of release vary from those previously presented for the stainless-steel clad core due to the changes in the temperature model brought about through the effects of the zirconium cladding and resultant metal-water reactions.

2. Fission Product Transport

Steam and hydrogen currents from water trapped in the reactor vessel bottom head and the steam-metal reactions provide an effective mechanism for transport of fission products from the reactor vessel to the containment building. Studies on the flow rates through the core and the subsequent fission product holdup and plateout in the reactor system are not sufficiently refined to permit the assignment of specific values to these phenomena. Therefore, all of the fission products released from the fuel are assumed to be instantaneously transported to the containment building. This assumption is believed to be conservative since experimental studies such as those conducted in the ORNL Hot-Cell Confinement Mockup Facility (CMF)⁽²⁷⁾ indicate that a significant portion of the fission products may be retained in the reactor vessel and coolant piping.

3. Plateout of Fission Products in the Containment Building

The fission product inventory is assumed to be instantaneously released from each of the three core zones to the containment vessel in accordance with the data presented in Table IV-4. Except for the iodine and bromine products, the conservative assumption is made that all fission products released from the core remain airborne and immediately become available for release to the atmosphere. The iodines and bromines are assumed to deposit or plateout on the inner containment surfaces as expressed by the following equation:

$$C = C_0 e^{-at}$$

where:

C = Fraction of iodine present at any time t

C_0 = Initial fraction of iodine present

a = Exponential constant of $0.693/140 \text{ min}^{-1}$

t = time, min.

This equation is derived from the results of several experiments conducted in the United Kingdom⁽²⁸⁾ to investigate the plateout of elemental iodine on various surfaces as a function of time. The specific deposition velocity used, represented by the constant " a ", was obtained for plateout on a chlorinated rubber-base paint in a steam atmosphere.

The negative exponential function in the above equation describes the time dependence of iodine concentration in the containment atmosphere over an initial period of approximately 8 to 10 hours, after which time the concentration decreases steadily by deposition to approximately 1% in a period of 1 day⁽²⁸⁾.

A plot of the exponential function is presented in Figure IV-12. This curve represents the assumed concentrations of iodines available for release from the containment as a function of time, out to approximately 7.7 hours at which time the concentration reduces to 10% of the initial concentration. Thereafter the percentage of airborne iodines is assumed to remain constant for the remainder of the containment release period. This seemingly conservative assumption is made in order to give credit for the possible formation of iodine compounds by the reaction of free iodine with impurities in the atmosphere or on the containment surfaces.

The assumed minimum concentration of 10% is based upon results of experiments performed to determine the quantities of iodine compounds formed in a CO/CO₂ atmosphere⁽³⁰⁾. While the value of

approximately 10% was determined for this atmosphere, a significant reduction was observed in a similar experiment for a CO₂/air atmosphere⁽³⁰⁾. Due to the unavailability of similar data for a steam atmosphere, the higher results for the CO/CO₂ atmosphere are assumed to apply for LOFT.

Although the experimental data discussed above pertain only to the plateout and radioactive compound formation of iodines, bromine is considered to behave similarly. However, calculations indicate that the effect of this assumption on the dose results is negligible.

Since the computer code used in the radiological analysis is not capable of handling the exponential plateout function, the area under the curve in Figure IV-12 is divided into sections corresponding to the time at which meteorological conditions change and to the time when the I and Br concentration is reduced to 10%. Therefore, the average airborne I and Br activity assumed for computations is 46.5% for the first 6 hours, 13.25% for times between 6 hours and 7.7 hours, and 10% for times greater than 7.7 hours.

4. Fission Product Behavior Model

The data presented in the previous discussions for the fission product release from the core and the plateout on inner containment surfaces have been combined to obtain the total fission product inventory available for release from the containment vessel as presented in Table IV-5. Except for I and Br the percentages in the table for the various time intervals are simply the results in Table IV-4 accumulated as time progresses; the I and Br percentages in Table IV-4 have been multiplied by the average fractions of activity remaining airborne corresponding to each time interval as presented above.

As noted previously the fission product release model used in the zirconium-cladding analysis differs from that presented in IDO-16981 for the stainless-steel-cladding analysis in that specific release percentages have been assumed for fission products other than iodine. However, the plateout model is similar to that in IDO-16981.

TABLE IV-5

FISSION PRODUCTS AVAILABLE FOR RELEASE FROM CONTAINMENTSTEAM-LIMITED METAL-WATER REACTION

Release Time Interval After Blowdown	Percent of Core Inventory Available for Release to the Atmosphere								
	Xe-Kr	I-Br	Te	Cs	Ru	Sr	Ba	Zr-Ce	Rest
544 sec $\leq$ t < 1760 sec	62.6	28.0	57.1	57.4	15.2	5.9	6.8	3.3	0.626
1760 sec $\leq$ t < 2137 sec	77.1	34.5	70.3	70.7	18.7	7.3	8.4	4.1	0.771
2137 sec $\leq$ t < 6 hr	88.2	41.2	90.9	86.6	19.8	9.6	20.1	4.1	1.000
6 hr < t $\leq$ 7.7 hr	88.2	11.7	90.9	86.6	19.8	9.6	20.1	4.1	1.000
7.7 hr < t $\leq$ end of test	88.2	8.8	90.9	86.6	19.8	9.6	20.1	4.1	1.000

D. Core Disassembly and Energy Release to the Containment

There are three basic sources of energy associated with the LOFT loss-of-coolant test; they are: (1) internal energy in the primary coolant, (2) decay energy from the fission products, and (3) metal-water reaction energy from reaction of the Zircaloy cladding and subsequent combustion of the gaseous hydrogen produced. The effects decay and metal-water reaction energy on heating up the core have been determined; however, the effects of all the energy sources on the release of fission products from the containment vessel must be examined. From the time history of pressure and temperature within the containment one can estimate the leakage of fission products from the containment vessel. In order to obtain this pressure-temperature history, however, the time, the rate and the mechanism by which energy is supplied to the containment atmosphere must be found.

A further extension of the heatup analysis to predict core disassembly and subsequent release of energy to the containment is then necessary despite the inherent uncertainties and the simplifying assumptions which must be made. For the purpose of analysis the core was modeled as having several distinct radial rings. The results of the thermal analysis in Section IV-B were used to describe the axial temperature profile for each ring as a function of time. When the temperature in the lowest axial region of each ring reaches the melting point of a fuel element restraining grid ($\sim 2550^{\circ}\text{F}$), the grid is assumed to fail. Since the grid in each element supports the fuel pins for that element, failure of the grid allows the material in the element to fall through the holes in the lower support structure. Thus, when the melting point is reached in any ring, the grid is assumed to fail and all the material in that ring then falls into the water reservoir remaining in the bottom of the reactor vessel.

The release of energy to the containment will then proceed as follows: (1) when each ring falls, the accumulated decay and metal-

water reaction energy stored in the core materials as sensible heat and latent heat of fusion, will rapidly be transferred to the water and steam generated; (2) decay energy generated by the fuel in each region after they have fallen will appear continuously as steam generation; and (3) the energy from hydrogen combustion will be deposited directly as sensible heat in the containment atmosphere because the reaction occurs in the containment.

The rate of addition of hydrogen combustion energy is 44.6% of the total metal-water reaction energy generation rate as given by Figure IV-7. The time-history of the core disassembly and the associated energy releases is given in Table IV-6. Also shown is the total accumulated fraction of the total decay energy generation to be added continuously after the specific region falls into the water. The two different enrichment zones for the LOFT core result in a radial decay heating profile with a peak in outer enrichment zone as shown in Figure IV-1. The rings are numbered outward radially from the core center and, because of the outer peak in the heating, ring 4 is calculated to fall before ring 3, thus accounting for the earlier time at which this region falls. Since the principal support structure, the lower core support grid, has not failed when the ring falls, this behavior can actually occur within the core. After approximately 5 hours, all the water will be gone from the reactor vessel, and at this time 25% of the total decay heating rate is arbitrarily assumed to appear as direct heat in the containment.

E. Effect of Core Disassembly on the Reactor Vessel

During the core migration phase, extremely high temperature differentials can exist between the migrating fuel-cladding mixture and the residual water in the vessel. Fuel dropping into the residual water will cause rapid local generation of steam and an attendant pressure pulse. The local pressure can be related by the first law

TABLE IV - 6

DECAY AND METAL-WATER REACTION
ENERGY RELEASES

Ring No.	Time At Which Ring Falls Hrs.	Total M-W Energy To Boiling H ₂ O Btu.	Total Decay Energy To Boiling H ₂ O Btu.	Total Fraction of Decay Energy Rate For All Regions In The Water Reservoir
1	0.207	9.38×10^4	3.44×10^4	0.0518
2	0.224	2.71×10^5	1.09×10^5	0.2042
3	0.288	3.85×10^5	2.18×10^5	0.6676
4	0.279	5.52×10^5	2.19×10^5	0.4596
5	0.385	5.58×10^5	6.55×10^5	1.00

of thermodynamics to the time duration of the pressure disturbance (pressure pulse width) and the amount of energy absorbed by the water⁽³¹⁾, assuming that this energy is manifested only in compressing the liquid. Figure IV-13 shows this relationship graphically for a triangular pressure-time shape. The amount of energy that can be deposited within a given pulse time is, of course, dependent on the rate of cooling of the core components. To determine an upper bound for this rate, it was assumed deposition that all of the core is instantaneously submerged in the residual water, that all of the core components are molten and that it retains all of its original heat transfer area, i.e., no agglomeration has occurred. Latent heat effects and heat transfer surface resistances were disregarded. The resulting rate of energy deposition is therefore dependent only on the heat transfer characteristics of the molten core. The calculated peak pressure and thermal energy are shown in Figure IV-14 as a function of pulse width.

The time required for the core to become completely submerged would not be less than approximately 200 milliseconds, based solely on gravitational effects. Buoyant forces and steam pressure buildup at the leading edge of the core would considerably extend this time and could, in fact, reverse the core trajectory or expel sections of the molten material. Accounting for these effects would result in the calculated energy release being spread over a longer time span within several smaller pressure pulses. For the purposes of this analysis, it was assumed that the time for energy deposition in the water, or pulse time, would be 100 milliseconds.

A technique employed by Newmark and Cole⁽³²⁾ was used to predict the effects of the dynamic pressure loading on the reactor vessel. This involved the use of the following empirical formula developed by Newmark:

$$\frac{P_{\max}}{q} = \frac{(2\mu - 1)\tau}{\pi\Theta_1} + \frac{1 - \frac{0.5}{\mu}}{1 + \frac{0.7\tau}{\Theta_1}}$$

where:

$P_{\max}$ = maximum dynamic pressure

q = equivalent static load

$$\mu = \text{material ductility} = \frac{\text{failure deflection}}{\text{yield deflection}}$$

Θ_1 = time of load application (pressure pulse width)

τ = natural vibration period of vessel

Solutions to this equation for various material ductility values, μ , are shown in Figure IV-15. For the LOFT vessel, a value of 50 was used, with a natural vibration period calculated to be 0.9 milliseconds. The resulting ratio of dynamic to static load is 1.0 and from Figure IV-14 the maximum dynamic pressure, accounting for the reflective shock, is 1680 psi. Equating this to be effective static load gives a safety factor of 1.6 for the ratio of design to applied pressure. It must be realized that the assumption of no pressure relief during the period of pulse formation is highly unrealistic since the acoustic propagation time to the free surface of the water, and hence, pressure relief times is no more than one or two milliseconds. To investigate the pressure loading effects on the vessel in the time span before relief requires a prediction of the rate of core entry into the water and the entry characteristics as they affect energy deposition rate from the core to the water. As stated earlier, the minimum time required for complete immersion of the core is 200 milliseconds, disregarding buoyancy and back-pressure effects. Since the natural frequency of the reactor vessel is approximately one millisecond the dynamic loading effects would be most pronounced for pressure pulse durations in this range. For this reason, the dynamic loading effects were also investigated for pressure pulse width of one millisecond. During that period, 1/200 of the core would have entered the water and would have transferred energy at the maximum rate. The resulting maximum dynamic pressure, including reflected shock, is 1600 psi, or approximately the same as found for the 100 millisecond pulse duration. For all other values of energy deposition time, the peak pressure will not exceed that value.

Based upon this highly conservative analysis, it is concluded that the migration of the core components into the residual water pool below it will not result in over pressurization of the reactor vessel.

F. Containment Vessel Response

The pressure and temperature transients that will be experienced within the LOFT containment building due to the loss-of-coolant from a zirconium-clad core have been investigated for the planned loss-of-coolant test. The pressure history for this case was computed using a digital computer code⁽³³⁾ developed by Phillips Petroleum Company. This pressure history was used as input data to another digital computer code which calculates the instantaneous leak rate from the containment building. With this information, it was possible to calculate the fission-product leakage to the atmosphere.

The pressure history for the test case (Figure IV-16) includes the containment response to all the forms of energy release that will be manifested during the test sequence. These include, in chronological order, the blowdown energy from the primary system only, the chemical reaction energy associated with the combustion of hydrogen, the energy released to the containment by the generation of steam from any portion of the core which has migrated to the reactor vessel bottom head, and decay heat that is generated after all the water in the bottom head has been boiled away. In determining the containment response, it is assumed that no heat is transferred from the containment atmosphere during the blowdown period.

Regarding the energy release due to hydrogen combustion, it is assumed that the hydrogen gas given off during the metal-water reaction will recombine with free oxygen present within the containment vessel. Depending upon the concentrations existing within

the containment air-steam-hydrogen mixture, combustion may or may not occur. If combustion does take place, it will occur as a fire. The energy release from any hydrogen-air reaction will appear in the containment atmosphere, and thus, will have little effect on the core heatup; however, it can have a significant effect on the temperature and pressure occurring within the containment. Studies have been made to determine the concentration limits for flammability and detonation of hydrogen-air-steam mixtures under the conditions likely to exist in a containment vessel after blowdown⁽³⁴⁾. For LOFT, the maximum hydrogen concentration for a uniform containment atmosphere is so low that the composition of the mixture is outside the flammable and detonation regions for the entire range of possible steam concentrations. The only way for a hydrogen-air reaction to occur under these circumstances is for the hydrogen to react before it can be diluted beyond the flammable limit.

The hydrogen gas will be given off at a rate proportional to the metal-water reaction rate, and because of the high temperatures associated with the zirconium-water reaction the hydrogen is released from the cladding at a high temperature. The hot hydrogen gas mixed possibly with some steam will flow from the core and out the break in the coolant line. During this migration, the hydrogen will not be cooled below its spontaneous ignition temperature. Thus, when the hot hydrogen contacts the air, ignition will occur with the hydrogen burning before it has a chance to diffuse within the containment atmosphere. This energy release is treated as a direct energy conversion within the containment vessel.

That portion of the decay energy released to the containment vessel after the water in the reactor vessel has been boiled away is also assumed to be a direct energy conversion. This is a conservative assumption, but it adequately covers the energy release to the containment due to removal of the decay heat by any mode of heat transfer.

The pressure response data obtained for the planned loss-of-coolant test show that under no circumstances will the LOFT containment pressure exceed the design pressure. In fact, the containment peak pressure occurs as a result of blowdown only and the addition of the metal-water reaction energy serves only to delay or modify the general pressure decay.

The pressure history as determined for the case representing the test conditions with a zirconium-clad core is quite different than that presented for a stainless-steel clad core (Figure X-18, page X-60) of the Preliminary Safety Analysis Report. The energy release for the zirconium clad core is greater due to the additional energy supplied by the metal-water reaction. For this reason, it (would naturally) follow that the pressure decay in the containment would be less rapid. The difference in the results of the two investigations can be explained, not in terms of the total or rate of energy release, but by considering the method used in calculating the pressure-time histories. The pressure history for the case with a stainless-steel clad core was determined by performing a series of "hand-calculations." In order to reduce the number of iterations and, thereby, the time required to perform the calculations, it was necessary to make a number of simplifying assumptions. The assumptions used were conservative, so that the consequences of the test would not be minimized. The result of the previously employed analytical technique was a compounding of conservative assumptions which yielded a gross underestimate of the rate of pressure decay.

The digital computer code used to obtain the pressure-time history for the zirconium clad core permits a more detailed analysis and inclusion of other analytical variables. Hence, simplifying assumptions are not required in order to facilitate problem manageability. Thus, it is possible to more accurately describe the

problem and a pressure-time history can be obtained which is believed to more closely approximate the true situation. The relative effect of the cladding conversion on the containment behavior cannot, therefore, be determined directly from a comparison of the pressure-time history calculated previously for the stainless-steel clad core and that calculated from the Zircaloy clad core. In order to provide this comparison, the pressure-time history for the loss-of-coolant test with a stainless-steel core has been recalculated with the new analytical techniques, but the original assumption regarding core heatup behavior have been retained. The two pressure-time histories are compared in Figure IV-16. As would be expected, the difference in the two curves is principally due to the fact that metal-water reactions were considered only for the Zircaloy clad core. Since the energy source from these reactions is terminated within the first few hours, the two curves essentially coincide after approximately 7 hours.

Heating of the containment vessel due to radiation heat transfer from the hydrogen flame was also investigated. The only area which may undergo significant localized heating is that closest to the simulated break in the primary system - the containment door. Adequate protection measures are contingent upon the final design of the door and the design of the dolly-mounted system which will set the distance from the simulated break to the inner surface of the door. In addition other criteria have been imposed upon the design to insure that local heating of the reactor dolly or primary system will not prevent transport of the dolly to the hot shop following the test or prevent closure of the primary system prior to transport.

G. Fission Product Leakage From the Containment Building

The fission product leakage rate from the containment vessel is largely dependent on the geometry of the leakage path. The geometries found in containment vessel can be any combination of orifice-to-channel type openings, resulting in separate and agglomerate flow regimes of nozzle, turbulent, laminar, and molecular flows. Comparing the flow regimes, it can be shown that maximum leakage is obtained by employing the simple nozzle leakage equations.

The development of the nozzle equations with respect to fission product leakage is based on a conservative, but realistic, approach. This conservative approach consists of defining several areas of physical and mathematical phenomena with realistic assumptions concerning flow rate characteristics and fission product behavior. The basic assumptions used are summarized below:

1. Fission Product Behavior

- a. The fission products released to the containment building are uniformly distributed throughout the gaseous environment.
- b. No fission products are considered to be absorbed in the water that condenses out in the containment.
- c. The release of fission products is proportional to the product of the mass leakage rate (lbs of mixture/sec) and the curie concentration (curies/lb of mixture) in the containment building.

2. Flow Rate Characteristics

- a. The gaseous mixture has the same characteristics as a perfect gas mixture.
- b. The expansion of the environment gases is defined by an isentropic or reversible adiabatic process.
- c. The flow of the mixture is an instantaneous flow depending on the variables of temperature, pressure, density and gas

constant of the gaseous environment, the outside atmospheric pressure and the area of the leakage hole.

d. The flow characteristics are considered to be similar to those of a sharp edged orifice.

e. No water condenses out the gaseous mixture during the isentropic expansion.

The development of the leakage equation consists of employing the fundamental energy equations, the perfect gas law and the isentropic relationships. With these equations, the following leakage equation can be obtained for subsonic flow:

$$\frac{W_2}{A_2} = \sqrt{\frac{P_o}{T_o}} \sqrt{\frac{K_g}{R}} \left[\frac{P_2}{P_o} \right]^{\frac{K+1}{2K}} \left[\frac{\left(\frac{P_2}{P_o} \right)^{\frac{1-K}{K}} - 1}{\frac{K-1}{2}} \right]^{1/2}$$

This equation developed for sonic flow is as follows:

$$\frac{W_2}{A_2} = \sqrt{\frac{P_o}{T_o}} \sqrt{\frac{K_g}{R}} \left[\frac{2}{K+1} \right]^{\frac{K+1}{2K-2}}$$

The boundary conditions used in determining the transitional sonic-subsonic region can be obtained by the following equation:

$$\left(\frac{P_2}{P_o} \right)_{\text{crit}} = \left(\frac{2}{K+1} \right)^{\frac{K}{K-1}}$$

The resulting leakage equations have been plotted as a function of time as shown in Figure IV-17. These equations incorporate the instantaneous pressure, density and temperature in the containment building which are obtained by an energy balance technique described earlier in this report.

The leakage data reported in the Preliminary Safety Analysis Report expressed the leakage as the square root of pressure. This, however, does not take into consideration such items as the variation of leakage with respect to critical flow, changes in temperature and changes in density. These items are, however, considered in the equations developed above.

Terms:

W = Mass flow rate (lb/sec)
A = Area of the opening in containment building (ft²)
T = Temperature (°R)
R = Gas constant (lb_f/lb_m) (°R/ft)
P = Pressure (lb_f/ft²)
K = Isentropic constant (C_p/C_v)

Subscripts:

o = Containment conditions
2 = Throat conditions
3 = Atmosphere conditions
()_{crit} = Transitional subsonic-sonic conditions at throat

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Fig. IV-1 LOFT Radial Decay Heating Profile.

Fig. IV-2 LOFT Decay Heat Source P/P_0 vs. Time.

Fig. IV-3 LOFT Axial Power Profile

Fig. IV-4 Steam Generation Rate vs. Time.

Fig. IV-5 Steam Generation Rate vs. Time.

Fig. IV-6 Total Steam Generation Rate vs. Time.

Fig. IV-7 Metal-Water Reaction Energy Generation Rate vs. Time

Fig. IV-8 Steam Limited Metal-Water Reaction Energy vs. Time.

	A	B	C
1	*C=830 *F=440	C= 1500 F= 2020	C=3600 *
2	C=375 F=352	C= 533 F= 651	C=1845 *
3	C=325 F=352	C= 458 F= 528	C=1552 *
4	C=375 F=440	C= 533 F= 651	C=1845 *
5	C=830 F=916	C= 1500 F= 2020	C=3600 *

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C = TIME AT WHICH CLADDING STARTS TO MELT
 F = TIME AT WHICH FUEL STARTS TO MELT
 * FUEL DOES NOT MELT IN ANNULUS C.

Fig. IV-9 Cladding and Fuel Initial Melting Times.

Fig. IV-10 Core Temperature vs. Time.

Fig. IV-11 LOFT 3 Zone Core Model.

Fig. IV-12 Percent of Activity Released to Containment Which Remains Airborne and Available for Release to Atmosphere.

Fig. IV-13 Peak Pressure vs. Pressure Pulse Duration and Energy Absorbed.

Fig. IV-14 Peak Pressure and Thermal Energy vs. Pressure Pulse Duration - LOFT Core.

Fig. IV-15 Equivalent Static Load Curves

Fig. IV-16 Containment Building Pressure Decay.

Fig. IV-17 Leakage Rate - Steam Limited.

CHAPTER V

RADIOLOGICAL ANALYSIS OF THE LOSS-OF-COOLANT TEST

V. RADIOLOGICAL ANALYSIS OF THE LOSS-OF-COOLANT TEST

A. Analysis

The radiological exposures are tabulated as a function of distance and sectors to indicate the related effects of wind persistency with respect to the calculated doses. Table V-1 is included for the purpose of relating the sectors shown in Figure V-1 to various towns and facilities for which the radiological exposures could occur.

The radiological doses presented in Tables V-2 through V-15 are based on the assumption that favorable meteorological conditions exist immediately following initiation of the loss-of-coolant test and are in accordance with the conditions shown in Table V-16 for the remainder of the release period. The conditions shown in Table V-16 are identical to those employed in IDO-16981 for the corresponding release period. The meteorological conditions upon which Table V-16 is based were obtained from a five year recorded history by the United States Weather Bureau at the NRTS and are considered to be the most realistic conditions which could exist for the test release period during the month of February.

In addition to the meteorological conditions and parameters listed in Table V-16 the following assumptions and analytical techniques are used to determine the radiological doses.

1. Assumptions

- (a) The reactor has operated at 50 MW for 800 hours.
- (b) The fission products released to the containment building and available for release to the atmosphere are presented in Section IV-C. These releases are based upon a limited metal-water reaction.

- (c) Fission product leakage external to the containment shell is proportional to the leak rate curve as shown in Figure IV-17.
- (d) Depletion of the radioactive cloud from fallout is neglected; however, it can be shown analytically that if this reduction factor is taken into account, the doses presented in Tables V-2 through V-15 may be significantly reduced, the degree of reduction being dependent upon such parameters as meteorological conditions, release height, deposition velocity, and downwind distances^(1,2).
- (e) The radiological dosages calculated by the use of the following equations have been modified to account for cloud meandering and wind persistency. The cloud meandering factor is to account for centerline deviation during any given release period and is obtained by averaging the cloud concentration in the crosswind direction, as given by Sutton's diffusion equation (Equation 1), over a $22\frac{1}{2}^\circ$ sector. The wind persistency factor accounts for wind shift during any given release period and is in accordance with the basic meteorological assumptions as shown in Table V-16 as well as the wind rose data for winter lapse and inversion conditions as shown in Figures V-2 and V-3. The numerical values for wind persistency, cloud meandering, and the product of the two are shown in Table V-17, V-18 and V-19.

The assumptions listed above are identical to those employed in the analysis presented in IDO-16981 except for (b) and (c). The different fission product release and containment leakage quantities assumed for this analysis are discussed in Section IV.

2. Analytical Techniques

The analytical techniques upon which the cloud gamma, deposition gamma, and inhalation doses are based employ a modified Sutton's diffusion equation for a continuous point source.

a. Sutton's Diffusion Equation for a Continuous Release of Radioactive Effluents⁽³⁾

$$\frac{X}{Q} = \frac{2}{U \pi C_y C_z D^{2-N}} e^{-\left[\frac{1}{D^{2-N}} \left(\frac{Y^2}{C_y^2} + \frac{H^2}{C_z^2} \right) \right]} \text{ (sec/m}^3\text{)} \quad (1)$$

where:

X = airborne concentration at any detector position downwind
(units/m³)

U = wind speed (m/sec)

C_y = horizontal diffusion coefficient (m)^{N/2}

C_z = vertical diffusion coefficient (m)^{N/2}

D = distance to downwind detector position (m)

N = Sutton's stability parameter (dimensionless)

Y = distance to crosswind detector position (considered as zero
in all cases) (m)

H = height of release (considered as zero in all cases) (m)

Q = source strength (units/sec).

It is necessary to integrate Sutton's equation with respect to time to determine the integrated doses for the various cases considered. Since Q (activity released at any time t) is assumed to vary with time, it is necessary to include this varying parameter in the analytical model and integrate the equation between the appropriate time intervals.

The modified Sutton's equation is identical to that upon which the radiological analysis in IDO-16981 was based. However, the techniques used in this analysis to obtain the external doses have been improved. The changes in techniques are discussed below.

b. Cloud Gamma Dose

The cloud and deposition gamma doses presented in Table V-2 through V-15 are based upon mathematical models which are considerably different from those presented in the Preliminary Safety Analysis Report for the LOFT Facility (IDO-16981).

In IDO-16981 the external doses were based upon gross conversion factors which were applicable for a semi-infinite radioactive cloud uniform in concentration throughout and emitting gamma rays at one specific energy per disintegration.

The calculational techniques presented below and those used to compute the external doses for this addendum have been refined to include 10 gamma energy groups as well as integration techniques which compensate for the use of gross conversion factors.

$$D_i = \sum_{j=1}^{j=n} \sum_{k=1}^{k=m} f_k \lambda_k \left[\int_{x_1}^{x_2} \int_{-\infty}^{\infty} \int_{-\infty}^{\infty} G(r) \left[\int_{t_1}^{t+t_1} C_k(x,y,z,t) dt \right] \cdot dzdydx \right] \quad (2)$$

where:

D_i = Cloud gamma dose in sector i and is equal to the summation of all the separate meteorological cases which contribute to the dose in sector i

n = Number of meteorological cases j
 m = Number of isotopes k
 f_k = Radiological decay constant of isotope k (sec^{-1})
 $G(r)$ = Dose attenuation kernel which is defined as follows:

$$G(r) = \frac{v e^{-\mu r}}{4\pi r^2} \left[1 + \mu r + \frac{(\mu r)^2}{7E^{2.4}} \right] \quad (3)$$

where:

v = Flux to dose conversion factor ($\text{rem}\cdot\text{m}^2/\gamma$)
 μ = Linear absorption coefficient for air (m^{-1})
 E = Average gamma energy per photon (MeV/γ)
 r = Distance (m) from source to receptor
 $r = \sqrt{(x-D)^2 + (y-E)^2 + (z+h)^2}$

where:

D = Downwind distance to detector position (m)
 E = Crosswind distance to detector position (m)
 h = Height of release (m)

$C_k(x,y,z,t)$ = concentration function in atoms per cubic meter and is defined as follows:

$$C_k(x,y,z,t) = N_k(t) \cdot L(t-t_1) \cdot \frac{X}{Q} \quad (4)$$

where:

$N_k(t)$ = Number of atoms of the k^{th} isotope at any time t
 $L(t-t_1)$ = Leakage function (sec^{-1})
 t_1 = Time to reach downwind detector position (sec)

When D (the distance from the point of release to the downwind detector position) is $\geq 10^3$ meters

$$x_1 = D-10^3$$

$$x_2 = D+10^3$$

When D is $< 10^3$ meters

$$x_1 = 0$$

$$x_2 = D+10^3$$

The mathematical models for the cloud gamma dose as well as the models for the deposition and inhalation dose calculations have been programmed for the IBM 7040 computer.

c. Inhalation Dose

The inhalation dose to the various organs shown in Tables V-2 through V-15 is the total dose contribution from the individual elements and isotopes listed in Table V-20. The differential equation which has been programmed for the IBM-7040 computer and that which is used to determine the integrated dose to any individual organ from any selected isotope is shown below.

$$\frac{dA_K(t)}{dt} = \lambda_b A_K(t) - \lambda_r A_K(t) + B_R \cdot C_f \cdot \frac{X}{Q} \cdot L(t-t_1) \cdot N_K(t) \quad (5)$$

where:

$A_K(t)$ = Number of atoms in the organ at any time t

λ_b = Biological decay factor (sec^{-1})

λ_r = Radioactive decay factor (sec^{-1})

B_R = Breathing rate ($2.32 \times 10^{-4} \text{ m}^3/\text{sec}$)

C_f = Fraction of inhaled material which reaches the critical organ

The techniques used to compute the inhalation doses to the various critical organs are the same as presented in IDO-16981.

d. Deposition Gamma Dose

The deposition gamma doses presented in Tables V-2 through V-15 are based upon a 5 day exposure period as was assumed in IDO-16981. These doses have been calculated by the use of the following mathematical models.

$$D_{g-i} = \sum_{j=1}^{j=n} \sum_{k=1}^{k=m} f_k \int_0^{\infty} \int_{-\infty}^{\infty} \left[\bar{A}_k(x,y) + \bar{A}_k^*(x,y) \right] G(R) \, dx dy \quad (6)$$

where:

D_{g-i} = Deposition dose in sector i which is the sum of all the separate meteorological conditions contributing to the dose in this sector

f_k = Number of photons per disintegration (γ 's/dis.)

R = Distance (m) from source to receptor

$R = \sqrt{(x-D)^2 + (y-E)^2}$, D and E as previously defined

$G(R)$ = Dose attenuation kernel equivalent to that presented for cloud gamma dose with r replaced by R

$\bar{A}_k$ = Dose contribution during deposition period

$$\bar{A}_k = \int_{\tau_1}^{\tau_{dep} + \tau_1} \lambda_k A_k(\tau) \, d\tau$$

where:

λ_k = Radioactive decay constant of isotope k (sec^{-1})

τ_1 = Time for released effluent to reach downwind detector position (sec)

τ_{dep} = Time over which materials are deposited (sec)

$$A_k(\tau) = \sum_{j=1}^{j=n} K_j \cdot V_g \cdot \frac{X}{Q} \cdot E_k \frac{(e^{-\lambda_i \tau_1})}{\lambda_i + P_j - \lambda_k} \left[e^{-\lambda_k(\tau - \tau_1)} - e^{-(\lambda_i + P_j)(\tau - \tau_1)} \right] \quad (8)$$

where:

k_j = Linear approximation to leakage rate curve (sec^{-1})

V_g = Deposition velocity (m/sec)

λ_i = Radiological decay constant of the i^{th} isotope

P_j = Leakage rate decay constant (sec^{-1})

E_k = Linear operator

$E_k = N_k(\tau)/e^{-\lambda_i \tau}$

$N_k(\tau)$ = Number of atoms of the k^{th} isotope at decay time τ

$A_k^*(\tau)$ = Dose contribution from the deposition period plus τ_1 to the final exposure period

$$A_k^*(\tau) = \int_{\tau_{\text{dep}} + \tau_1}^{\tau_2 + \tau_{\text{dep}} + \tau_1} \lambda_k \cdot A_k(\tau) d\tau \quad (9)$$

where:

$$A_k^*(\tau) = A_k(\tau_{\text{dep}} + \tau_1) e^{-\lambda_k(\tau - \tau_{\text{dep}} - \tau_1)} \quad (10)$$

Other terms are previously defined.

The radiological doses calculated by the use of the previous mathematical model have been corrected to account for cloud meandering and wind variation which will occur during the release period. The cloud meandering factor is obtained as follows:

$$C_m^{11} = \frac{2Q}{\pi u C_y C_z D^{2-N}} \int_{-\infty}^{+\infty} \exp\left[\frac{-y^2}{C_y^2 D^{2-N}}\right] dy \quad (11)$$

Averaging the results from the equation above over a $22-1/2^\circ$ sector, the following equation is obtained:

$$C_m^1 = \frac{16Q}{\pi^{3/2} U C_z} D^{-\left(\frac{4-N}{2}\right)} \quad (12)$$

Since the radiological doses have been calculated by the use of Sutton's equation (Equation 1) for a crosswind distance of zero meters, the cloud meandering factor is then a ratio of Equation 12 to Equation 1.

$$C_m = \frac{C_m^1}{\text{Eq. 1}} = \frac{\frac{16QD^{-\left(\frac{4-N}{2}\right)}}{\pi^{3/2} U C_z}}{\frac{2Q}{U \pi C_y C_z D^{2-N}}} \quad (13)$$

$$C_m = \frac{8C_y}{\pi^{1/2} D^{N/2}} \quad (14)$$

The numerical values obtained by the above mathematical manipulations are presented in Table V-17 and are identical to those presented in IDO-16981.

The wind variation factor is obtained by correcting the wind rose data, as presented in Figures V-2 and V-3, by the meteorological conditions presented in Table V-16. The numerical values for wind variation are presented in Table V-18 while the product of cloud meandering and wind variation is presented in Table V-19.

e. Ingestion Dose

The ingestion doses presented in Tables V-2 through V-15 are based on the release of isotopes I-131 and I-133. The contribution to the ingestion dose from each of these isotopes is considered to be directly proportional to the fraction of the total inhalation thyroid dose due to I-131 and I-133, respectively. The ratios of ingestion to inhalation, as obtained from results of the AEC-ID CERT Test⁽⁴⁾, are 1200 for I-131 and 100 for I-133. These values apply for the ingestion dose to a child as compared to the inhalation dose from these isotopes to an adult.

The ratios have been reduced by a factor of 20 to allow for the appropriate reduction in the amount of food obtained from grazing during the winter months⁽⁵⁾. Therefore, the ingestion doses presented represent the sum of 60 times the part of the inhalation dose due to I-131 and 5 times that due to I-133.

Each of the ingestion doses presented is that which would be received by a child if he were to consume one quart of milk per day for an infinite period of time. Since the mass of a child's thyroid is approximately a factor of ten smaller than that of an adult's, the ingestion dose to an adult would be approximately a factor of ten less than the child's dose.

f. Direct Radiation from the Containment Building

The potential radiation exposures from gamma rays emanating from fission products inside the containment building following the core meltdown are presented in Figure V-4. Several curves are presented showing total gamma dose as a function of distance from the building for a number of exposure times. As may be seen in Figure V-4, the total exposure to personnel at the Technical Support

Facility, the nearest unshielded facility to the containment vessel (approximately 1900 m), is less than one mrem for a five day exposure. Therefore, it is concluded that the direct radiation from the containment building will not disrupt the normal routine at the other facilities in the TAN complex. The area within 1000 ft of the containment vessel will be considered as an exclusion area and access to this area will be regulated by the TAN Health and Safety Group.

The direct radiation doses are based on the assumption that 15 percent of the gross fission products are distributed uniformly within the containment volume above the ground floor. For analytical purposes, the containment building is separated into three sections depending on the vessel wall thickness or shielding available to reduce the radiation from the building. The first evaluation considered only the hemispherical dome which is shielded with six inches of concrete. The second calculation considered the cylindrical volume from 35 ft above the grade to the spring line of the containment building. This region is shielded with twelve inches of concrete. The region below the 35 ft level was also considered; however, because of the concrete biological shield (2 ft of concrete) surrounding this region, the doses were found to be negligible.

B. Conclusions

The radiological analysis for the loss-of-coolant test is, where possible, based on realistic assumptions and analytical models. Areas in which sufficient experimental data are not available to justify the use of realistic assumptions are of necessity based on conservative values. Two of these areas in particular are cloud depletion due to fallout and the relationship of leakage rate to pressure. Cloud depletion is neglected due to the unavailability of data which would indicate what the true particle size and chemical state of the

released particles and halogens will be after they have migrated through various leakage paths, and hence filtering media, to the external atmosphere. The relationship of leakage rate to pressure is in all probability proportional to a combination of a linear and square root function of the pressure differential. However, until the actual leak rate tests are conducted on the LOFT facility the true proportionality constants cannot be determined. It is believed that the combination of these two factors alone (leak rate and cloud depletion) will result in the off-site doses being reduced by at least a factor of five.

A comparison of the inhalation doses presented in Tables V-2 through V-15 to the corresponding values presented in IDO-16981 shows that the muscle doses in this analysis are in general approximately a factor of three greater than those in the stainless-steel cladding analysis, whereas doses to the other critical organs are less than those in IDO-16981. This seeming discrepancy is primarily due to the differences in assumed release percentages for the cesium and barium isotopes. In this analysis, the assumed percentages are 86.6 percent for cesium and 20.1 percent for barium; only 1 percent release of each was assumed in IDO-16981. Since comparatively few isotopes are considered as contributing to the muscle dose (see Table V-20), the effect of the increased release percentages is significant for this organ. The fact that the doses to the other critical organs are less than those in IDO-16981 is primarily due to the reduction in predicted containment leakage which is discussed in Section IV-G.

It is extremely difficult to compare the external gamma doses in this analysis to those presented in IDO-16981. This is due to the differences in (1) the assumed fission product release percentages, discussed in Section IV-C, (2) the technique used to calculate the containment pressure decay history, discussed in Section IV-G, and (3) the analytical techniques used to determine the external gamma doses, discussed in Section V-A.

In order to attain a better understanding of the effects of the cladding material on the radiological doses, the containment pressure decay history for stainless-steel cladding was computed using the same improved techniques as were used for the zirconium analysis. A comparison of the two pressure histories indicates that all radiological doses for the stainless-steel analysis would be approximately the same as those presented in this analysis for zirconium cladding if identical analytical techniques and fission product release percentages were assumed. However, if the containment fission product concentration used in the IDO-16981 analysis were assumed and the improved analytical techniques and pressure history calculation applied, the inhalation, ingestion, and deposition doses for stainless-steel would be at least a factor of 4 lower than those presented for zirconium. The cloud doses would be approximately the same in both cases.

Examination of the radiological doses indicates that the maximum off-site exposure occurs in the Howe vicinity (Sector K, Table V-11). The nearest farming area in the Howe direction is approximately seven miles southwest of the LOFT facility. The calculated doses at this location are compared in the following table to the corresponding doses as presented in IDO-16981.

MAXIMUM OFF-SITE EXPOSURES FOR LOSS OF COOLANT TEST		
<u>Types of Exposure</u>	<u>Calculated Dose (rem)</u>	
	<u>Zr</u>	<u>SS</u>
External Gamma Dose	4.9×10^{-4}	8.0×10^{-4}
Inhalation Thyroid Dose to a Child*	1.5×10^{-2}	4.3×10^{-1}
Ingestion Thyroid Dose to a Child**	3.1×10^{-1}	2.0×10^0

* Child's inhalation dose is 1.5 times adult's inhalation dose.

** Child's ingestion dose is 10 times adult's ingestion dose.

The radiological doses at the site boundary for zirconium cladding are well below the exposure criteria in AEC Manual Chapter 0524. The conclusion can therefore be drawn that the loss-of-coolant test can be safely conducted at the NRTS. It should be emphasized, however, that extensive radiological measurements will be made during the test both in the containment building and downwind of the facility. If the fission product release rate to the atmosphere at any time exceeds that which would result in an integrated dose which is above the pre-determined limits, the test can be quickly and safely terminated. It should also be noted that since monitored penetrations will be installed whereby the fission products leaking through such penetrations can either be stopped or collected the potential doses should be significantly reduced.

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- (1) Walter M. Culkowski, "Calculations of the Deposition of Aerosols from Elevated Sources," ORO-171.
 - (2) A. C. Chamberlain, "Aspects of Travel and Deposition of Aerosol and Vapor Clouds," AERE HP/R 1261.
 - (3) G. A. DeMarrais, N. F. Islitzer, "Diffusion Climatology of the NRTS," IDO-12015, U. S. Weather Bureau, April 1960.
 - (4) C. A. Hawley, "Controlled Environmental Radioiodine Test," IDO-12035.
 - (5) Private Communications, J. R. Horan, Director Health and Safety, AEC-ID to D. E. Williams, Director Reactor Division, AEC-ID, February 25, 1964.
 - (6) Chapter 0524, Standards for Radiation Protection, USAEC.

TABLE V-1

DISTANCE AND RELATIVE LOCATIONS OF VARIOUS TOWNS
AND FACILITIES WITH RESPECT TO LOFT

Town or Facility	Distance From LOFT		Sector Where the Town or Facility is Located													
	Miles	Meters	A	B	C	E	F	G	H	I	J	K	M	N	O	P
TSF	1.2	1.93×10^3				x										
Montevieu	12	1.93×10^4			x											
Howe	15	2.41×10^4										x				
NRF	17	2.74×10^4								x						
EBR-II	18	2.90×10^4						x								
CFA	25	4.02×10^4								x						
Hamer	29	4.67×10^4				x										
Atomic City	30	4.83×10^4							x							
Roberts	31	4.99×10^4				x										
Idaho Falls	42	6.76×10^4					x									

TABLE V-2

Radiation Dose in North-Northeast Sector

DISTANCE FROM LOFT (METERS)	SECTOR A INHALATION DOSE						
	TOTAL BODY (rem)	THYROID (rem)	MUSCLE (rem)	LUNG (rem)	LIVER (rem)	KIDNEY (rem)	BONE (rem)
10^2	4.7×10^{-1}	1.3×10^1	1.4×10^{-2}	7.5×10^0	5.5×10^{-1}	1.9×10^0	4.9×10^0
5×10^2	2.8×10^{-2}	7.8×10^{-1}	8.4×10^{-4}	4.5×10^{-1}	3.3×10^{-2}	1.1×10^{-1}	3.0×10^{-1}
10^3	8.2×10^{-3}	2.3×10^{-1}	2.4×10^{-4}	1.3×10^{-1}	9.7×10^{-3}	3.4×10^{-2}	8.6×10^{-2}
5×10^3	4.9×10^{-4}	1.4×10^{-2}	1.5×10^{-5}	8.0×10^{-3}	5.8×10^{-4}	2.0×10^{-3}	5.2×10^{-3}
10^4	1.5×10^{-4}	4.2×10^{-3}	4.4×10^{-6}	2.4×10^{-3}	1.8×10^{-4}	6.0×10^{-4}	1.6×10^{-3}
5×10^4	8.9×10^{-6}	2.4×10^{-4}	2.5×10^{-7}	1.4×10^{-4}	1.1×10^{-5}	3.5×10^{-5}	9.4×10^{-5}
10^5	2.4×10^{-6}	6.3×10^{-5}	7.6×10^{-8}	3.8×10^{-5}	2.9×10^{-6}	9.1×10^{-6}	2.6×10^{-5}

DISTANCE FROM LOFT (METERS)	EXTERNAL GAMMA DOSE		INGESTION DOSE
	CLOUD GAMMA DOSE (rem)	DEPOSITION GAMMA DOSE (rem)	
10^2	4.5×10^{-3}	1.3×10^0	*
5×10^2	1.6×10^{-3}	1.1×10^{-1}	*
10^3	1.2×10^{-3}	4.0×10^{-2}	*
5×10^3	1.3×10^{-4}	1.1×10^{-3}	*
10^4	4.5×10^{-5}	2.7×10^{-4}	1.4×10^{-1}
5×10^4	4.8×10^{-6}	1.4×10^{-6}	8.4×10^{-3}
10^5	1.3×10^{-6}	3.8×10^{-7}	2.3×10^{-3}

* Controlled Grazing Land

TABLE V-3

Radiation Dose in Northeast Sector

DISTANCE FROM LOFT (METERS.)	SECTOR B INHALATION DOSE						
	TOTAL BODY (rem)	THYROID (rem)	MUSCLE (rem)	LUNG (rem)	LIVER (rem)	KIDNEY (rem)	BONE (rem)
10^2	3.2×10^{-1}	8.7×10^0	9.4×10^{-3}	5.1×10^0	3.7×10^{-1}	1.3×10^0	3.3×10^0
5×10^2	1.9×10^{-2}	5.3×10^{-1}	5.7×10^{-4}	3.1×10^{-1}	2.2×10^{-2}	7.8×10^{-2}	2.0×10^{-1}
10^3	5.9×10^{-3}	1.6×10^{-1}	1.8×10^{-4}	9.5×10^{-2}	7.0×10^{-3}	2.4×10^{-2}	6.2×10^{-2}
5×10^3	3.4×10^{-4}	9.5×10^{-3}	1.0×10^{-5}	5.4×10^{-3}	4.0×10^{-4}	1.4×10^{-3}	3.5×10^{-3}
10^4	1.0×10^{-4}	2.9×10^{-3}	3.0×10^{-6}	1.6×10^{-3}	1.2×10^{-4}	4.1×10^{-4}	1.1×10^{-3}
5×10^4	5.9×10^{-6}	1.6×10^{-4}	1.6×10^{-7}	9.5×10^{-5}	7.1×10^{-6}	2.3×10^{-5}	6.3×10^{-5}
10^5	1.6×10^{-6}	4.3×10^{-5}	5.2×10^{-8}	2.6×10^{-5}	2.0×10^{-6}	6.2×10^{-6}	1.8×10^{-5}

DISTANCE FROM LOFT (METERS.)	EXTERNAL GAMMA DOSE		INGESTION DOSE
	CLOUD GAMMA DOSE (rem)	DEPOSITION GAMMA DOSE (rem)	THYROID (rem)
10^2	3.0×10^{-3}	8.5×10^{-1}	*
5×10^2	1.1×10^{-3}	7.6×10^{-2}	*
10^3	8.4×10^{-4}	2.9×10^{-2}	*
5×10^3	8.8×10^{-5}	7.6×10^{-4}	*
10^4	3.1×10^{-5}	1.9×10^{-4}	9.5×10^{-2}
5×10^4	3.2×10^{-6}	9.7×10^{-7}	5.6×10^{-3}
10^5	8.6×10^{-7}	2.6×10^{-7}	1.6×10^{-3}

* Controlled Grazing Land

TABLE V-4

Radiation Dose in East-Northeast Sector

DISTANCE FROM LOFT (METERS)	SECTOR C INHALATION DOSE						
	TOTAL BODY (rem)	THYROID (rem)	MUSCLE (rem)	LUNG (rem)	LIVER (rem)	KIDNEY (rem)	BONE (rem)
10^2	2.4×10^{-1}	6.6×10^0	7.1×10^{-3}	3.8×10^0	2.8×10^{-1}	9.7×10^{-1}	2.5×10^0
5×10^2	1.4×10^{-2}	3.9×10^{-1}	4.2×10^{-4}	2.3×10^{-1}	1.7×10^{-2}	5.7×10^{-2}	1.5×10^{-1}
10^3	4.2×10^{-3}	1.2×10^{-1}	1.2×10^{-4}	6.7×10^{-2}	4.9×10^{-3}	1.7×10^{-2}	4.4×10^{-2}
5×10^3	2.5×10^{-4}	7.1×10^{-3}	7.5×10^{-6}	4.1×10^{-3}	3.0×10^{-4}	1.0×10^{-3}	2.6×10^{-3}
10^4	7.7×10^{-5}	2.2×10^{-3}	2.3×10^{-6}	1.2×10^{-3}	9.1×10^{-5}	3.1×10^{-4}	8.0×10^{-4}
5×10^4	4.4×10^{-6}	1.2×10^{-4}	1.2×10^{-7}	7.1×10^{-5}	5.3×10^{-6}	1.7×10^{-5}	4.7×10^{-5}
10^5	1.2×10^{-6}	3.2×10^{-5}	3.9×10^{-8}	2.0×10^{-5}	1.5×10^{-6}	4.6×10^{-6}	1.1×10^{-5}

DISTANCE FROM LOFT (METERS)	EXTERNAL GAMMA DOSE		INGESTION DOSE
	CLOUD GAMMA DOSE (rem)	DEPOSITION GAMMA DOSE (rem)	THYROID (rem)
10^2	2.3×10^{-3}	6.4×10^{-1}	*
5×10^2	8.2×10^{-4}	5.6×10^{-2}	*
10^3	6.0×10^{-4}	2.0×10^{-2}	*
5×10^3	6.6×10^{-5}	5.7×10^{-4}	*
10^4	2.3×10^{-5}	1.4×10^{-4}	7.1×10^{-2}
5×10^4	2.4×10^{-6}	7.1×10^{-7}	4.2×10^{-3}
10^5	6.4×10^{-7}	2.0×10^{-7}	1.2×10^{-3}

* Controlled Grazing Land

TABLE V-5

Radiation Dose in East-Southeast Sector

DISTANCE FROM LOFT (METERS)	SECTOR E INHALATION DOSE						
	TOTAL BODY (rem)	THYROID (rem)	MUSCLE (rem)	LUNG (rem)	LIVER (rem)	KIDNEY (rem)	BONE (rem)
10^2	8.1×10^{-1}	2.2×10^1	2.4×10^{-2}	1.3×10^1	9.6×10^{-1}	3.3×10^0	8.5×10^0
5×10^2	4.6×10^{-2}	1.3×10^0	1.4×10^{-3}	7.4×10^{-1}	5.4×10^{-2}	1.9×10^{-1}	4.8×10^{-1}
10^3	1.3×10^{-2}	3.7×10^{-1}	4.0×10^{-4}	2.2×10^{-1}	1.6×10^{-2}	5.5×10^{-2}	1.4×10^{-1}
5×10^3	8.3×10^{-4}	2.3×10^{-2}	2.5×10^{-5}	1.3×10^{-2}	9.8×10^{-4}	3.4×10^{-3}	8.7×10^{-3}
10^4	2.5×10^{-4}	7.1×10^{-3}	7.5×10^{-6}	4.0×10^{-3}	3.0×10^{-4}	1.0×10^{-3}	2.6×10^{-3}
5×10^4	1.5×10^{-5}	4.1×10^{-4}	4.1×10^{-7}	2.4×10^{-4}	1.8×10^{-5}	5.8×10^{-5}	1.6×10^{-4}
10^5	4.0×10^{-6}	1.1×10^{-4}	1.3×10^{-7}	6.5×10^{-5}	4.9×10^{-6}	1.5×10^{-5}	4.4×10^{-5}

DISTANCE FROM LOFT (METERS)	EXTERNAL GAMMA DOSE		INGESTION DOSE
	CLOUD GAMMA DOSE (rem)	DEPOSITION GAMMA DOSE (rem)	
10^2	7.8×10^{-3}	2.2×10^0	*
5×10^2	2.7×10^{-3}	1.8×10^{-1}	*
10^3	1.9×10^{-3}	6.5×10^{-2}	*
5×10^3	2.2×10^{-4}	1.9×10^{-3}	*
10^4	7.6×10^{-5}	4.6×10^{-4}	2.3×10^{-1}
5×10^4	8.0×10^{-6}	2.4×10^{-6}	1.4×10^{-2}
10^5	2.1×10^{-6}	6.4×10^{-7}	3.9×10^{-3}

* Controlled Grazing Land

TABLE V-6

Radiation Dose in Southeast Sector

DISTANCE FROM LOFT (METERS)	SECTOR F INHALATION DOSE						
	TOTAL BODY (rem)	THYROID (rem)	MUSCLE (rem)	LUNG (rem)	LIVER (rem)	KIDNEY (rem)	BONE (rem)
10^2	1.8×10^0	4.8×10^1	5.2×10^{-2}	2.8×10^1	2.1×10^0	7.1×10^0	1.8×10^1
5×10^2	1.0×10^{-1}	2.9×10^0	3.1×10^{-3}	1.7×10^0	1.2×10^{-1}	4.2×10^{-1}	1.1×10^0
10^3	2.9×10^{-2}	8.2×10^{-1}	8.8×10^{-4}	4.8×10^{-1}	3.5×10^{-2}	1.2×10^{-1}	3.1×10^{-1}
5×10^3	1.8×10^{-3}	5.1×10^{-2}	5.4×10^{-5}	2.9×10^{-2}	2.1×10^{-3}	7.3×10^{-3}	1.9×10^{-2}
10^4	5.5×10^{-4}	1.6×10^{-2}	1.7×10^{-5}	8.9×10^{-3}	6.5×10^{-4}	2.2×10^{-3}	5.8×10^{-3}
5×10^4	3.2×10^{-5}	8.9×10^{-4}	9.0×10^{-7}	5.2×10^{-4}	3.9×10^{-5}	1.3×10^{-4}	3.4×10^{-4}
10^5	8.8×10^{-6}	2.3×10^{-4}	2.8×10^{-7}	1.4×10^{-4}	1.1×10^{-5}	3.3×10^{-5}	9.6×10^{-5}

DISTANCE FROM LOFT (METERS)	EXTERNAL GAMMA DOSE		INGESTION DOSE THYROID (rem)
	CLOUD GAMMA DOSE (rem)	DEPOSITION GAMMA DOSE (rem)	
10^2	1.7×10^{-2}	4.7×10^0	*
5×10^2	6.0×10^{-3}	4.1×10^{-1}	*
10^3	4.2×10^{-3}	1.4×10^{-1}	*
5×10^3	4.7×10^{-4}	4.1×10^{-3}	*
10^4	1.7×10^{-4}	1.0×10^{-3}	5.1×10^{-1}
5×10^4	1.7×10^{-5}	5.2×10^{-6}	3.0×10^{-2}
10^5	4.6×10^{-6}	1.4×10^{-6}	8.5×10^{-3}

* Controlled Grazing Land

TABLE V-7

Radiation Dose in South-Southeast Sector

DISTANCE FROM LOFT (METERS)	SECTOR G INHALATION DOSE						
	TOTAL BODY (rem)	THYROID (rem)	MUSCLE (rem)	LUNG (rem)	LIVER (rem)	KIDNEY (rem)	BONE (rem)
10^2	3.5×10^0	9.6×10^1	1.0×10^{-1}	5.6×10^1	4.1×10^0	1.4×10^1	3.6×10^1
5×10^2	2.1×10^{-1}	5.7×10^0	6.1×10^{-3}	3.3×10^0	2.4×10^{-1}	8.4×10^{-1}	2.2×10^0
10^3	6.0×10^{-2}	1.7×10^0	1.8×10^{-3}	9.7×10^{-1}	7.2×10^{-2}	2.5×10^{-1}	6.3×10^{-1}
5×10^3	3.6×10^{-3}	1.0×10^{-1}	1.1×10^{-4}	5.8×10^{-2}	4.3×10^{-3}	1.5×10^{-2}	3.8×10^{-2}
10^4	1.1×10^{-3}	3.1×10^{-2}	3.3×10^{-5}	1.8×10^{-2}	1.3×10^{-3}	4.5×10^{-3}	1.2×10^{-2}
5×10^4	6.3×10^{-5}	1.7×10^{-3}	1.8×10^{-6}	1.0×10^{-3}	7.5×10^{-5}	2.4×10^{-4}	6.7×10^{-4}
10^5	1.8×10^{-5}	4.6×10^{-4}	5.6×10^{-7}	2.8×10^{-4}	2.1×10^{-5}	6.7×10^{-5}	1.9×10^{-4}

DISTANCE FROM LOFT (METERS)	EXTERNAL GAMMA DOSE		INGESTION DOSE
	CLOUD GAMMA DOSE (rem)	DEPOSITION GAMMA DOSE (rem)	THYROID (rem)
10^2	3.3×10^{-2}	9.3×10^0	*
5×10^2	1.2×10^{-2}	8.2×10^{-1}	*
10^3	8.6×10^{-3}	2.9×10^{-1}	*
5×10^3	9.5×10^{-4}	8.1×10^{-3}	*
10^4	3.4×10^{-4}	2.0×10^{-3}	1.0×10^0
5×10^4	3.4×10^{-5}	1.0×10^{-5}	5.9×10^{-2}
10^5	9.2×10^{-6}	2.8×10^{-6}	1.7×10^{-2}

* Controlled Grazing Land

Radiation Dose in South Sector

DISTANCE FROM LOFT (METERS)	SECTOR H INHALATION DOSE						
	TOTAL BODY (rem)	THYROID (rem)	MUSCLE (rem)	LUNG (rem)	LIVER (rem)	KIDNEY (rem)	BONE (rem)
10^2	3.6×10^0	1.1×10^2	1.0×10^{-1}	5.7×10^1	4.2×10^0	1.4×10^1	3.6×10^1
5×10^2	2.1×10^{-1}	6.1×10^0	6.2×10^{-3}	3.3×10^0	2.5×10^{-1}	8.6×10^{-1}	2.2×10^0
10^3	6.1×10^{-2}	1.8×10^0	1.8×10^{-3}	9.9×10^{-1}	7.3×10^{-2}	2.5×10^{-1}	6.4×10^{-1}
5×10^3	3.6×10^{-3}	1.1×10^{-1}	1.1×10^{-4}	5.9×10^{-2}	4.4×10^{-3}	1.5×10^{-2}	3.9×10^{-2}
10^4	1.1×10^{-3}	3.2×10^{-2}	3.3×10^{-5}	1.8×10^{-2}	1.3×10^{-3}	4.6×10^{-3}	1.2×10^{-2}
5×10^4	6.4×10^{-5}	1.8×10^{-3}	1.8×10^{-6}	1.0×10^{-3}	7.6×10^{-5}	2.4×10^{-4}	6.8×10^{-4}
10^5	1.8×10^{-5}	4.7×10^{-4}	5.6×10^{-7}	2.8×10^{-4}	2.1×10^{-5}	6.9×10^{-5}	1.9×10^{-4}

DISTANCE FROM LOFT (METERS)	EXTERNAL GAMMA DOSE		INGESTION DOSE
	CLOUD GAMMA DOSE (rem)	DEPOSITION GAMMA DOSE (rem)	
10^2	5.0×10^{-2}	9.3×10^0	*
5×10^2	1.7×10^{-2}	8.2×10^{-1}	*
10^3	1.1×10^{-2}	2.9×10^{-1}	*
5×10^3	1.1×10^{-3}	8.1×10^{-3}	*
10^4	3.9×10^{-4}	2.0×10^{-3}	1.0×10^0
5×10^4	3.5×10^{-5}	1.0×10^{-5}	6.1×10^{-2}
10^5	9.2×10^{-6}	2.8×10^{-6}	1.8×10^{-2}

* Controlled Grazing Land

TABLE V-9

Radiation Dose in South-Southwest Sector

DISTANCE FROM LOFT (METERS)	SECTOR I INHALATION DOSE						
	TOTAL BODY (rem)	THYROID (rem)	MUSCLE (rem)	LUNG (rem)	LIVER (rem)	KIDNEY (rem)	BONE (rem)
10^2	2.6×10^0	7.4×10^1	8.2×10^{-2}	4.5×10^1	3.3×10^0	1.2×10^1	2.9×10^1
5×10^2	1.7×10^{-1}	5.0×10^0	4.8×10^{-3}	2.6×10^0	1.9×10^{-1}	6.6×10^{-1}	1.7×10^0
10^3	4.8×10^{-2}	1.5×10^0	1.4×10^{-3}	7.6×10^{-1}	5.6×10^{-2}	2.0×10^{-1}	5.0×10^{-1}
5×10^3	2.9×10^{-3}	8.6×10^{-2}	8.4×10^{-5}	4.6×10^{-2}	3.4×10^{-3}	1.1×10^{-2}	3.0×10^{-2}
10^4	8.7×10^{-4}	2.6×10^{-2}	2.6×10^{-5}	1.4×10^{-2}	1.0×10^{-3}	3.5×10^{-3}	9.1×10^{-3}
5×10^4	4.9×10^{-5}	1.4×10^{-3}	1.3×10^{-6}	7.8×10^{-4}	5.9×10^{-5}	1.9×10^{-4}	5.2×10^{-4}
10^5	1.4×10^{-5}	3.8×10^{-4}	4.5×10^{-7}	2.2×10^{-4}	1.7×10^{-5}	5.3×10^{-5}	1.5×10^{-4}

DISTANCE FROM LOFT (METERS)	EXTERNAL GAMMA DOSE		INGESTION DOSE
	CLOUD GAMMA DOSE (rem)	DEPOSITION GAMMA DOSE (rem)	THYROID (rem)
10^2	5.4×10^{-2}	7.1×10^0	*
5×10^2	1.6×10^{-2}	6.2×10^{-1}	*
10^3	9.6×10^{-3}	2.2×10^{-1}	*
5×10^3	8.9×10^{-4}	6.2×10^{-3}	*
10^4	3.3×10^{-4}	1.6×10^{-3}	8.6×10^{-1}
5×10^4	2.8×10^{-5}	7.7×10^{-6}	4.8×10^{-2}
10^5	7.3×10^{-6}	2.2×10^{-6}	1.4×10^{-2}

* Controlled Grazing Land

TABLE V-10

Radiation Dose in Southwest Sector

DISTANCE FROM LOFT (METERS)	SECTOR J INHALATION DOSE						
	TOTAL BODY (rem)	THYROID (rem)	MUSCLE (rem)	LUNG (rem)	LIVER (rem)	KIDNEY (rem)	BONE (rem)
10^2	2.3×10^0	7.7×10^1	6.9×10^{-2}	3.7×10^1	2.7×10^0	9.4×10^0	2.3×10^1
5×10^2	1.4×10^{-1}	4.3×10^0	3.9×10^{-3}	2.1×10^0	1.6×10^{-1}	5.4×10^{-1}	1.4×10^0
10^3	3.8×10^{-2}	1.2×10^0	1.1×10^{-3}	6.2×10^{-1}	4.6×10^{-3}	1.6×10^{-1}	4.0×10^{-1}
5×10^3	2.3×10^{-3}	6.2×10^{-2}	6.6×10^{-5}	3.6×10^{-2}	2.7×10^{-4}	9.2×10^{-3}	2.4×10^{-2}
10^4	7.0×10^{-4}	2.2×10^{-2}	2.1×10^{-5}	1.1×10^{-2}	8.3×10^{-5}	2.9×10^{-3}	7.3×10^{-3}
5×10^4	4.2×10^{-5}	1.2×10^{-3}	1.1×10^{-6}	6.7×10^{-4}	5.1×10^{-5}	1.6×10^{-4}	4.5×10^{-4}
10^5	1.1×10^{-5}	3.1×10^{-4}	3.5×10^{-7}	1.8×10^{-4}	1.4×10^{-5}	4.2×10^{-5}	1.2×10^{-4}

DISTANCE FROM LOFT (METERS)	EXTERNAL GAMMA DOSE		INGESTION DOSE
	CLOUD GAMMA DOSE (rem)	DEPOSITION GAMMA DOSE (rem)	THYROID (rem)
10^2	5.7×10^{-2}	5.8×10^0	*
5×10^2	1.6×10^{-2}	5.0×10^{-1}	*
10^3	9.1×10^{-3}	1.8×10^{-1}	*
5×10^3	7.8×10^{-4}	4.9×10^{-3}	*
10^4	3.0×10^{-4}	1.2×10^{-3}	7.2×10^{-1}
5×10^4	2.5×10^{-5}	6.5×10^{-6}	4.2×10^{-2}
10^5	5.6×10^{-6}	1.7×10^{-6}	1.1×10^{-2}

* Controlled Grazing Land

TABLE V-11

Radiation Dose in West-Southwest Sector

DISTANCE FROM LOFT (METERS)	SECTOR K INHALATION DOSE						
	TOTAL BODY (rem)	THYROID (rem)	MUSCLE (rem)	LUNG (rem)	LIVER (rem)	KIDNEY (rem)	BONE (rem)
10^2	1.3×10^0	3.4×10^1	3.8×10^{-2}	2.1×10^1	1.5×10^0	5.3×10^0	1.3×10^1
5×10^2	7.8×10^{-2}	2.5×10^0	2.2×10^{-3}	1.3×10^0	9.1×10^{-2}	3.1×10^{-1}	8.1×10^{-1}
10^3	2.1×10^{-2}	5.7×10^{-1}	1.8×10^{-3}	3.4×10^{-1}	2.5×10^{-2}	8.8×10^{-2}	2.2×10^{-1}
5×10^3	1.3×10^{-3}	4.1×10^{-2}	3.8×10^{-5}	2.0×10^{-2}	1.5×10^{-3}	5.2×10^{-3}	1.4×10^{-2}
10^4	4.0×10^{-4}	1.3×10^{-2}	1.1×10^{-5}	6.4×10^{-3}	4.7×10^{-4}	1.6×10^{-3}	4.1×10^{-3}
5×10^4	2.3×10^{-5}	6.8×10^{-4}	6.4×10^{-7}	3.7×10^{-4}	2.8×10^{-5}	8.9×10^{-5}	2.5×10^{-4}
10^5	6.4×10^{-6}	1.8×10^{-4}	2.1×10^{-7}	1.0×10^{-4}	7.7×10^{-6}	2.4×10^{-5}	6.9×10^{-5}

DISTANCE FROM LOFT (METERS)	EXTERNAL GAMMA DOSE		INGESTION DOSE
	CLOUD GAMMA DOSE (rem)	DEPOSITION GAMMA DOSE (rem)	THYROID (rem)
10^2	3.4×10^{-2}	3.2×10^{-0}	*
5×10^2	9.6×10^{-3}	2.9×10^{-1}	*
10^3	5.3×10^{-3}	9.8×10^{-2}	*
5×10^3	4.6×10^{-4}	2.7×10^{-3}	*
10^4	1.7×10^{-4}	6.9×10^{-4}	4.1×10^{-1}
5×10^4	1.4×10^{-5}	3.6×10^{-6}	2.4×10^{-2}
10^5	3.2×10^{-6}	1.0×10^{-6}	6.6×10^{-3}

* Controlled Grazing Land

TABLE V-12

Radiation Dose in West-Northwest Sector

DISTANCE FROM LOFT (METERS)	SECTOR M INHALATION DOSE						
	TOTAL BODY (rem)	THYROID (rem)	MUSCLE (rem)	LUNG (rem)	LIVER (rem)	KIDNEY (rem)	BONE (rem)
10^2	1.6×10^{-1}	4.4×10^0	4.7×10^{-3}	2.5×10^0	1.9×10^{-1}	6.4×10^{-1}	1.7×10^0
5×10^2	9.6×10^{-3}	2.7×10^{-1}	2.8×10^{-4}	1.5×10^{-1}	1.1×10^{-2}	3.9×10^{-2}	1.0×10^{-1}
10^3	2.8×10^{-3}	7.9×10^{-2}	8.4×10^{-5}	4.5×10^{-2}	3.3×10^{-3}	1.2×10^{-2}	3.0×10^{-2}
5×10^3	1.7×10^{-4}	4.7×10^{-3}	5.0×10^{-6}	2.7×10^{-3}	2.0×10^{-4}	6.8×10^{-4}	1.8×10^{-3}
10^4	5.1×10^{-5}	1.4×10^{-3}	1.5×10^{-6}	8.2×10^{-4}	6.0×10^{-5}	2.1×10^{-4}	5.3×10^{-4}
5×10^4	3.0×10^{-6}	8.2×10^{-5}	8.2×10^{-8}	4.7×10^{-5}	3.5×10^{-6}	1.2×10^{-5}	3.1×10^{-5}
10^5	8.1×10^{-7}	2.1×10^{-5}	2.6×10^{-8}	1.3×10^{-5}	9.8×10^{-7}	3.1×10^{-6}	8.8×10^{-6}

DISTANCE FROM LOFT (METERS)	EXTERNAL GAMMA DOSE		INGESTION DOSE
	CLOUD GAMMA DOSE (rem)	DEPOSITION GAMMA DOSE (rem)	
10^2	1.5×10^{-3}	4.3×10^{-1}	*
5×10^2	5.5×10^{-4}	3.8×10^{-2}	*
10^3	4.0×10^{-4}	1.4×10^{-2}	*
5×10^3	4.4×10^{-5}	3.8×10^{-4}	*
10^4	1.5×10^{-5}	9.3×10^{-5}	4.8×10^{-2}
5×10^4	1.6×10^{-6}	4.7×10^{-7}	2.8×10^{-3}
10^5	4.2×10^{-7}	1.3×10^{-7}	7.7×10^{-4}

* Controlled Grazing Land

TABLE V-13

Radiation Dose in Northwest Sector

DISTANCE FROM LOFT (METERS)	SECTOR N INHALATION DOSE						
	TOTAL BODY (rem)	THYROID (rem)	MUSCLE (rem)	LUNG (rem)	LIVER (rem)	KIDNEY (rem)	BONE (rem)
10^2	8.1×10^{-2}	2.2×10^0	2.4×10^{-3}	1.3×10^0	9.6×10^{-2}	3.3×10^{-1}	8.5×10^{-1}
5×10^2	5.0×10^{-3}	1.4×10^{-1}	1.5×10^{-4}	8.0×10^{-2}	5.8×10^{-3}	2.0×10^{-2}	5.2×10^{-2}
10^3	1.3×10^{-3}	3.7×10^{-2}	4.0×10^{-5}	2.2×10^{-2}	1.6×10^{-3}	5.5×10^{-3}	1.4×10^{-2}
5×10^3	8.4×10^{-5}	2.4×10^{-3}	2.5×10^{-6}	1.4×10^{-3}	9.9×10^{-5}	3.4×10^{-4}	8.8×10^{-4}
10^4	2.6×10^{-5}	7.2×10^{-4}	7.6×10^{-7}	4.1×10^{-4}	3.0×10^{-5}	1.0×10^{-4}	2.7×10^{-4}
5×10^4	1.5×10^{-6}	4.1×10^{-5}	4.1×10^{-8}	2.4×10^{-5}	1.8×10^{-6}	5.8×10^{-6}	1.6×10^{-5}
10^5	4.0×10^{-7}	1.1×10^{-5}	1.3×10^{-8}	6.5×10^{-6}	4.9×10^{-7}	1.5×10^{-6}	4.4×10^{-6}

DISTANCE FROM LOFT (METERS)	EXTERNAL GAMMA DOSE		INGESTION DOSE
	CLOUD GAMMA DOSE (rem)	DEPOSITION GAMMA DOSE (rem)	
10^2	7.8×10^{-4}	2.2×10^{-1}	*
5×10^2	2.9×10^{-4}	2.0×10^{-2}	*
10^3	1.9×10^{-4}	6.5×10^{-3}	*
5×10^3	2.2×10^{-5}	1.9×10^{-4}	*
10^4	7.7×10^{-6}	4.7×10^{-5}	2.4×10^{-2}
5×10^4	8.0×10^{-7}	2.4×10^{-7}	1.4×10^{-3}
10^5	2.1×10^{-7}	6.4×10^{-8}	3.9×10^{-4}

* Controlled Grazing Land

TABLE V-14

Radiation Dose in North-Northwest Sector

DISTANCE FROM LOFT (METERS)	SECTOR 0 INHALATION DOSE						
	TOTAL BODY (rem)	THYROID (rem)	MUSCLE (rem)	LUNG (rem)	LIVER (rem)	KIDNEY (rem)	BONE (rem)
10^2	1.2×10^{-1}	3.3×10^0	3.6×10^{-3}	1.9×10^0	1.4×10^{-1}	4.9×10^{-1}	1.2×10^0
5×10^2	7.3×10^{-3}	2.0×10^{-1}	2.1×10^{-4}	1.2×10^{-1}	8.5×10^{-3}	2.9×10^{-2}	7.6×10^{-2}
10^3	2.1×10^{-3}	6.0×10^{-2}	6.4×10^{-5}	3.5×10^{-2}	2.5×10^{-3}	8.8×10^{-3}	2.3×10^{-2}
5×10^3	1.2×10^{-4}	3.4×10^{-3}	3.6×10^{-6}	1.9×10^{-3}	1.4×10^{-4}	4.9×10^{-4}	1.3×10^{-3}
10^4	3.8×10^{-5}	1.1×10^{-3}	1.1×10^{-6}	6.1×10^{-4}	4.5×10^{-5}	1.5×10^{-4}	4.0×10^{-4}
5×10^4	2.2×10^{-6}	6.1×10^{-5}	6.2×10^{-8}	3.6×10^{-5}	2.7×10^{-6}	8.6×10^{-6}	2.4×10^{-5}
10^5	6.0×10^{-7}	1.6×10^{-5}	1.9×10^{-8}	9.7×10^{-6}	7.3×10^{-7}	2.3×10^{-6}	6.6×10^{-6}

DISTANCE FROM LOFT (METERS)	EXTERNAL GAMMA DOSE		INGESTION DOSE
	CLOUD GAMMA DOSE (rem)	DEPOSITION GAMMA DOSE (rem)	
10^2	1.2×10^{-3}	3.2×10^{-1}	*
5×10^2	4.2×10^{-4}	2.9×10^{-2}	*
10^3	3.1×10^{-4}	1.0×10^{-2}	*
5×10^3	3.2×10^{-5}	2.7×10^{-4}	*
10^4	1.1×10^{-5}	6.7×10^{-5}	3.5×10^{-2}
5×10^4	1.2×10^{-6}	3.6×10^{-7}	2.1×10^{-3}
10^5	3.2×10^{-7}	1.0×10^{-7}	5.8×10^{-4}

* Controlled Grazing Land

TABLE V-15

Radiation Dose in North Sector

DISTANCE FROM LOFT (METERS)	SECTOR P INHALATION DOSE						
	TOTAL BODY (rem)	THYROID (rem)	MUSCLE (rem)	LUNG (rem)	LIVER (rem)	KIDNEY (rem)	BONE (rem)
10^2	3.2×10^{-1}	8.7×10^0	9.4×10^{-3}	5.1×10^0	3.7×10^{-1}	1.3×10^0	3.3×10^0
5×10^2	1.9×10^{-2}	5.3×10^{-1}	5.7×10^{-4}	3.1×10^{-1}	2.2×10^{-2}	7.8×10^{-2}	2.0×10^{-1}
10^3	5.9×10^{-3}	1.6×10^{-1}	1.8×10^{-4}	9.5×10^{-2}	7.0×10^{-3}	2.4×10^{-2}	6.2×10^{-2}
5×10^3	3.4×10^{-4}	9.5×10^{-3}	1.0×10^{-5}	5.4×10^{-3}	4.0×10^{-4}	1.4×10^{-3}	3.5×10^{-3}
10^4	1.0×10^{-4}	2.9×10^{-3}	3.0×10^{-6}	1.6×10^{-3}	1.2×10^{-4}	4.1×10^{-4}	1.1×10^{-3}
5×10^4	5.9×10^{-6}	1.6×10^{-4}	1.6×10^{-7}	9.5×10^{-5}	7.1×10^{-6}	2.3×10^{-5}	6.3×10^{-5}
10^5	1.6×10^{-6}	4.3×10^{-5}	5.2×10^{-8}	2.6×10^{-5}	2.0×10^{-6}	6.2×10^{-6}	1.8×10^{-5}

DISTANCE FROM LOFT (METERS)	EXTERNAL GAMMA DOSE		INGESTION DOSE
	CLOUD GAMMA DOSE (rem)	DEPOSITION GAMMA DOSE (rem)	THYROID (rem)
10^2	3.0×10^{-3}	8.5×10^{-1}	*
5×10^2	1.1×10^{-3}	7.6×10^{-2}	*
10^3	8.4×10^{-4}	2.9×10^{-2}	*
5×10^3	8.8×10^{-5}	7.6×10^{-4}	*
10^4	3.1×10^{-5}	1.9×10^{-4}	9.5×10^{-2}
5×10^4	3.2×10^{-6}	9.5×10^{-7}	5.6×10^{-3}
10^5	8.6×10^{-7}	2.6×10^{-7}	1.6×10^{-3}

* Controlled Grazing Land

TABLE V-16

METEOROLOGICAL CONDITIONS AND DURATION OF EACH
FOR THE RELEASE PERIOD

Time Interval After Blowdown (hr)	Percent of Time Interval	Parameters Assumed for Dose Calculations					
		Condition	Wind Direction	Wind Velocity (mph)	Stability Parameter N	Diffusion Coefficient $C_y(m)^{N/2}$	Diffusion Coefficient $C_z(m)^{N/2}$
$0 \leq t < 6$	100	Strong Lapse	350° to 70°	4	0.20	0.334	0.334
$6 \leq t < 24$	10	Strong Inversion	110° to 240°	3	0.50	0.280	0.035
	90	Strong Inversion	300° to 70°	3	0.50	0.280	0.035
$24 \leq t \leq 27^*$	100	Strong Lapse	350° to 70°	4	0.20	0.334	0.334

* Containment pressure is reduced to zero in approximately 27 hours.

TABLE V-17

CORRECTION FACTOR TO ACCOUNT FOR CLOUD MEANDERING OVER A $22-1/2^\circ$ SECTOR

Distance (meters)	Cloud Meandering Factor (C_m)	
	Lapse Condition $N = 0.20$	Inversion Condition $N = 0.50$
10^2	0.95	0.40
5×10^2	0.81	0.27
10^3	0.75	0.22
5×10^3	0.65	0.15
10^4	0.60	0.13
5×10^4	0.51	0.09
10^5	0.47	0.07

TABLE V-18

CORRECTION FACTOR TO ACCOUNT FOR WIND VARIATION DURING THERELEASE PERIOD

Wind Occurs From	Sector Dose In	Wind Frequency Based on Winter Time Conditions (Figs.VI-6 & -7) (percent)	Wind Frequency Modified to account for Conditions Specified in Table XI-P (percent)	Atmosphere Conditions Considered
N	H	6.5	17.11	Lapse
NNE	I	10.5	27.63	
NE	J	13.0	34.21	
ENE	K	8.0	21.05	
ESE	M	2.0	0.93	Inversion 10% of Time From 110°-240°
SE	N	1.0	0.47	
SSE	O	1.5	0.70	
S	P	4.0	1.86	
SSW	A	6.0	2.78	
SW	B	4.0	1.86	
WSW	C	3.0	1.40	
WNW	E	3.0	4.64	Inversion 90% of Time From 300°-70°
NW	F	6.5	10.1	
NNW	G	13.0	20.2	
N	H	13.0	20.2	
NNE	I	10.0	15.5	
NE	J	8.0	12.4	
ENE	K	4.5	6.96	

TABLE V-19

CORRECTION FACTOR - PRODUCT OF WIND PERSISTENCY AND CLOUD MEANDERING FACTORS.

Wind From	Sector Dose Occurs In	Correction Factor for Inversion Conditions (N = 0.50) and Following Distances(m)						
		10^2	5×10^2	10^3	5×10^3	10^4	5×10^4	10^5
SSW	A	1.1×10^{-2}	7.4×10^{-3}	6.1×10^{-3}	4.1×10^{-3}	3.5×10^{-3}	2.4×10^{-3}	1.9×10^{-3}
SW	B	7.4×10^{-3}	5.0×10^{-3}	4.4×10^{-3}	2.8×10^{-3}	2.4×10^{-3}	1.6×10^{-3}	1.3×10^{-3}
WSW	C	5.6×10^{-3}	3.7×10^{-3}	3.1×10^{-3}	2.1×10^{-3}	1.8×10^{-3}	1.2×10^{-3}	9.7×10^{-4}
WTW	E	1.9×10^{-2}	1.2×10^{-2}	1.0×10^{-2}	6.9×10^{-3}	5.9×10^{-3}	4.0×10^{-3}	3.2×10^{-3}
NW	F	4.1×10^{-2}	2.7×10^{-2}	2.2×10^{-2}	1.5×10^{-2}	1.3×10^{-2}	8.7×10^{-3}	7.0×10^{-3}
NTW	G	8.1×10^{-2}	5.4×10^{-2}	4.5×10^{-2}	3.0×10^{-2}	2.6×10^{-2}	1.7×10^{-2}	1.4×10^{-2}
N	H	8.1×10^{-2}	5.4×10^{-2}	4.5×10^{-2}	3.0×10^{-2}	2.6×10^{-2}	1.7×10^{-2}	1.4×10^{-2}
NNE	I	6.2×10^{-2}	4.1×10^{-2}	3.4×10^{-2}	2.3×10^{-2}	2.0×10^{-2}	1.3×10^{-2}	1.1×10^{-2}
NE	J	5.0×10^{-2}	3.3×10^{-2}	2.7×10^{-2}	1.8×10^{-2}	1.6×10^{-2}	1.1×10^{-2}	8.6×10^{-3}
ENE	K	2.8×10^{-2}	1.9×10^{-2}	1.5×10^{-2}	1.0×10^{-2}	8.9×10^{-3}	6.0×10^{-3}	4.9×10^{-3}
ESE	M	3.7×10^{-3}	2.5×10^{-3}	2.1×10^{-3}	1.4×10^{-3}	1.2×10^{-3}	8.0×10^{-4}	6.4×10^{-4}
SE	N	1.9×10^{-3}	1.3×10^{-3}	1.0×10^{-3}	7.0×10^{-4}	6.0×10^{-4}	4.0×10^{-4}	3.2×10^{-4}
SSE	O	2.8×10^{-3}	1.9×10^{-3}	1.6×10^{-3}	1.0×10^{-3}	8.9×10^{-4}	6.0×10^{-4}	4.8×10^{-4}
S	P	7.4×10^{-3}	5.0×10^{-3}	4.4×10^{-3}	2.8×10^{-3}	2.4×10^{-3}	1.6×10^{-3}	1.3×10^{-3}
Correction Factor for Lapse Conditions, N = 0.20								
N	H	1.6×10^{-1}	1.4×10^{-1}	1.3×10^{-1}	1.1×10^{-1}	1.0×10^{-1}	8.8×10^{-2}	8.1×10^{-2}
NNE	I	2.6×10^{-1}	2.2×10^{-1}	2.1×10^{-1}	1.8×10^{-1}	1.6×10^{-1}	1.4×10^{-1}	1.3×10^{-1}
NE	J	3.3×10^{-1}	2.8×10^{-1}	2.6×10^{-1}	2.2×10^{-1}	2.0×10^{-1}	1.8×10^{-1}	1.6×10^{-1}
ENE	K	2.0×10^{-1}	1.7×10^{-1}	1.6×10^{-1}	1.4×10^{-1}	1.3×10^{-1}	1.1×10^{-1}	9.9×10^{-2}

TABLE V-20

FISSION PRODUCTS AND ORGANS CONSIDERED FOR
INHALATION DOSE CALCULATIONS

Isotope	Atomic No.	Mass No.	Organ Affected						
			Bone	Kidney	Liver	Lung	Muscle	Thyroid	Total Body
Ga	31	72	x	x	x	x			x
As	33	77		x	x	x			x
Br	35	82				x			x
Rb	37	86			x	x	x		x
		87			x	x	x		x
Sr	38	89	x			x			x
		90	x			x			x
		91	x			x			x
		92	x			x			x
Y	39	90	x			x			x
		91 ^m	x			x			x
		91	x			x			x
		92	x			x			x
		93	x			x			x
Zr	40	93	x	x	x	x			x
		95	x	x	x	x			x
		97	x	x	x	x			x
Nb	41	95	x	x	x	x			x
		97	x	x	x	x			x
Mo	42	99		x	x	x			x
Tc	43	99 ^m	x	x	x	x			x
		99	x	x	x	x			x
Ru	44	103	x	x		x			x
		105	x	x		x			x
		106	x	x		x			x
Rh	45	105	x	x	x	x			x
Pd	46	107		x	x	x			x
Ag	47	111	x	x	x	x			x
Cd	48	115		x	x	x			x
In	49	115	x	x	x	x		x	x
Sn	50	125	x		x			x	x
Sb	51	125	x		x	x		x	x

TABLE V-20 (Continued)

FISSION PRODUCTS AND ORGANS CONSIDERED FOR
INHALATION DOSE CALCULATIONS

Isotope	Atomic No.	Mass No.	Organ Affected						
			Bone	Kidney	Liver	Lung	Muscle	Thyroid	Total Body
Te	52	127m	x	x	x	x		x	x
		127	x	x	x	x		x	x
		129m	x	x	x	x		x	x
		129	x	x	x	x		x	x
		131m	x	x	x	x			x
		132	x	x	x	x		x	x
I	53	129	x	x	x	x		x	x
		131	x	x	x	x		x	x
		132	x	x	x	x		x	x
		133				x		x	x
		134				x		x	x
Cs	55	135	x	x	x	x	x		x
		137	x	x	x	x	x		x
Ba	56	140	x	x	x	x	x		x
La	57	140	x		x	x			x
Ce	58	141	x	x	x	x			x
		143	x	x	x	x			x
		144	x	x	x	x			x
Pr	59	143	x	x	x	x			x
Nd	60	147	x	x	x	x			x
		149	x	x	x	x			x
Pm	61	147	x	x	x	x			x
Sm	62	153	x	x	x	x			x
Eu	63	155	x	x	x	x			x
Gd	64	159	x	x	x	x			x

Fig. V-1 Area Surrounding LOFT Showing Sectors Used for Radiological Analysis.

WINTER

SPRING

SUMMER

FALL

TOTAL

Fig. V-2 TAN 20 Foot Level Lapse Wind Roses.

Fig. V-3 TAN 20-Foot Level Inversion Wind Roses.

Fig. V-4 Integrated Gamma Dose From the Containment Shell.

CHAPTER VI

SAFETY ANALYSIS OF THE MAXIMUM CREDIBLE ACCIDENT

VI. SAFETY ANALYSIS OF THE MAXIMUM CREDIBLE ACCIDENT

A. Identification of the Maximum Credible Accident

The maximum credible accident for the LOFT facility is defined as a rupture of the primary coolant circuit during adverse meteorological conditions after the reactor has been operating at full power (50 MW) for 800 consecutive hours. It is assumed that as a consequence of this rupture, both the primary and secondary system coolants are expelled to the containment volume. Because of the margin of safety incorporated in the primary coolant system design to accommodate multiple blowdowns and the rigid inspection and acceptance tests required prior to nuclear operation, the probability of a system rupture is very low. In addition, this probability is reduced by the relatively short operating history planned for the reactor.

Since the primary coolant system is not encased in a biological shield, the effect of unbalanced forces which could be imposed upon the system as a result of rapid coolant expulsion must be considered. Such forces could severely stress the system and conceivably cause missiles to be generated. Although recent investigations and reviews of reported data⁽¹⁾⁽²⁾⁽³⁾ indicate that the generation of a missile with sufficient energy to completely penetrate the containment is highly improbable, the current design and supporting analysis is not sufficiently developed to allow a conclusion that such a penetration is incredible. For this reason, the rupture of the primary system is postulated to result in the generation of a missile which penetrates the containment barrier leaving an assumed hole of six square inches in cross sectional area. As the pressure increases in the containment volume, the containment pressure reduction spray system will be automatically energized to provide a condensing mechanism for reduction of internal pressure in the building. The condensate sump will be

filled to $2/3$ its capacity (~5000 gal) prior to the test to insure that an adequate supply of water is available for immediate operation of the spray system. An independent spray system designed to remotely decontaminate the internal surfaces of the containment building is utilized as a redundant pressure reduction spray system and incorporates many of the fail-safe features existing in the primary spray system. (Section VII - page 34 of the Preliminary Safety Analysis Report - IDO 16981).

The model used to describe containment response for the maximum credible accident is similar to the model used for the test case. The differences, as discussed above, are in the assumptions used for blow-down, containment building integrity, and the use of the pressure reduction spray system. In the analysis of the maximum credible accident, it is also assumed that the core water-injection system is inoperative in order to evaluate the upper range of the consequences of this accident. Operation of this system would retard core heat-up and hence greatly reduce or completely eliminate the metal-water reaction and the fission product release.

A computer code ⁽⁴⁾ was used to determine the effect of the recirculating spray systems on the pressure decay within the containment vessel. The operation of the spray system produces a rapid pressure decay, and, when coupled with the effect of the hole in the containment vessel, causes the pressure to reach a pre-incident level in a short time. With a penetration area of six square inches, the internal pressure as shown in Figure VI-1 will be reduced to atmospheric in approximately 0.7 hour. The computer code and the assumptions presented in Section IV-G, except for the hole size, were used to determine the leakage rates from the containment vessel as a function of pressure (Figure VI-2). These leakage rates were used to determine the amount of fission products released from the containment vessel to the atmosphere.

The analysis of the maximum credible accident includes the assumption that the contents of the secondary coolant system are also discharged to the containment building. Secondary system rupture is postulated to occur when primary system hydrodynamic forces across the steam generator during blowdown violate the integrity of the heat exchanger tubes and tubesheet. However, the maximum pressure attained during simultaneous blowdown of both the primary and secondary coolant systems will not exceed the design pressure of the containment vessel.

It is further postulated that the maximum credible accident occurs during strong inversion conditions with the wind blowing to the north-east. This set of conditions results in the maximum off-site radiological dose.

After the pressure in the containment building has been reduced to approximately atmospheric pressure, the containment atmosphere will be recirculated through the halogen and particulate filter system to reduce the fission product concentration in the building. When lapse meteorological conditions are obtained, with the wind blowing to the on-site direction, the remaining fission products in the containment atmosphere will be exhausted through the filter system to the stack.

Based on this postulation of the maximum credible accident, the potential radiological exposures to both on-site personnel and the general public are presented in the following sections, along with the assumptions employed in arriving at these doses.

B. Radiological Analysis

The radiological analysis associated with the maximum credible accident is based upon the analytical models presented in Section V and the following basic assumptions:

- (1) Rupture and expulsion of all coolant from the primary and secondary coolant systems.

- (2) Generation of a missile which penetrates the containment building and results in a six-square-inch opening to the external atmosphere.
- (3) The pressure reduction spray system is automatically energized at the first indication of a pressure increase in the containment shell.
- (4) Fission product release from the fuel to the containment building is identical to that presented in Section IV-C for the loss-of-coolant test. This release is based upon a steam-limited metal-water reaction.
- (5) 50% of the particulates and halogens released to the containment shell are removed by the washout effects of the pressure reduction sprays.
- (6) Fission product leakage through the six-square-inch hole is proportional to the leakage curve as shown in Figure VI-2.
- (7) Release occurs under strong inversion conditions with the wind blowing at 5 mph, $c_y = 0.280$, $c_z = 0.035$.
- (8) Deposition velocity is 1 cm/sec for the halogens and 0.2 cm/sec for the particulates.
- (9) A reduction factor of five for the particulates and halogens has been used to account for cloud depletion due to fallout for distances greater than 10^4 meters (~6 miles) from LOFT⁽⁵⁾.
This is based on the belief that the particle size and chemical state of the fission products are relatively unchanged in going through the postulated 6 in. penetration, as compared to the test case in which they migrate through various small leakage paths which act as filtering media.
- (10) An analysis of the fission product diffusion through the postulated aperture in the containment building after the internal pressure reached zero indicates the additional contribution to the total dose is insignificant; therefore, this factor is neglected.
- (11) Fission product plateout inside the containment vessel is neglected due to the relatively short hold-up time and the

turbulence which exists in the containment building.

The assumptions listed above are identical to those employed in the analysis of the maximum credible accident presented in IDO-16981 except for (4) and (6). The differences in assumed percentages of fission products released from the fuel and methods for calculating containment leakage are discussed in Section IV.

The radiological doses calculated by the use of the above assumptions and previously described analytical models are presented in Table VI-1.

Personnel located in the immediate vicinity of the LOFT facility may be exposed to direct radiation from the fission products contained in the containment building. However, the levels to which personnel will be exposed will be no greater than those shown in Figure V-4, and in all probability, will be less, due to the large leakage rate and removal effects of the source term by the pressure reduction spray system.

C. Conclusions

The radiological doses calculated for the postulated maximum credible accident are presented in Table VI-1. The deposition doses presented are based on a 2-hour exposure time, as were the corresponding doses presented in IDO-16981. Calculations indicate that these doses would be increased by approximately a factor of 20 if a 1-year deposition exposure time is considered.

The values presented in Table VI-1 have been extrapolated to determine the doses at the nearest off-site area as well as those at the nearest population center (Idaho Falls). These doses are compared to the corresponding values, as presented in IDO-16981, in Table VI-2.

TABLE VI-1
RADIOLOGICAL DOSES FOR THE MAXIMUM
CREDIBLE ACCIDENT

DISTANCE FROM LOFT (METERS)	INHALATION DOSE (rem)						
	TOTAL BODY	THYROID	MUSCLE	LUNG	LIVER	KIDNEY	BONE
10^2	7.2×10^3	1.5×10^6	1.5×10^2	9.5×10^4	9.4×10^3	2.7×10^4	4.9×10^4
5×10^2	6.5×10^2	1.4×10^5	1.3×10^1	8.5×10^3	8.4×10^2	2.4×10^3	4.4×10^3
10^3	2.3×10^2	4.8×10^4	4.7×10^0	3.0×10^3	3.0×10^2	8.6×10^2	1.6×10^3
5×10^3	2.0×10^1	4.2×10^3	4.2×10^{-1}	2.7×10^2	2.6×10^1	7.5×10^1	1.4×10^2
10^4	1.4×10^0	2.9×10^2	2.9×10^{-2}	1.9×10^1	1.9×10^0	5.4×10^0	9.8×10^0
5×10^4	1.2×10^{-1}	2.2×10^1	2.6×10^{-3}	1.6×10^0	1.6×10^{-1}	4.5×10^{-1}	8.6×10^{-1}
10^5	3.8×10^{-2}	6.6×10^0	9.3×10^{-4}	5.2×10^{-1}	5.5×10^{-2}	1.5×10^{-1}	3.0×10^{-1}

DISTANCE FROM LOFT (METERS)	EXTERNAL GAMMA DOSE (rem)	
	CLOUD GAMMA DOSE	DEPOSITION GAMMA DOSE
10^2	6.5×10^2	9.5×10^3
5×10^2	3.5×10^2	1.3×10^{-3}
10^3	3.1×10^2	5.0×10^{-2}
5×10^3	4.0×10^1	1.3×10^1
10^4	1.1×10^1	5.4×10^{-1}
5×10^4	7.3×10^{-1}	5.0×10^{-3}
10^5	2.4×10^{-1}	1.3×10^{-3}

TABLE VI-2

RADIOLOGICAL DOSE SUMMARY

Location Considered	External Gamma Dose (rem)				Inhalation Dose (rem)	
	Cloud Gamma		Deposition Gamma			
	Zr	SS	Zr	SS	Zr	SS
Nearest inhalabited off-site area (7 miles east of LOFT)	9.3	22	0.42	0.31	250	9.0
Idaho Falls (42 miles southeast of LOFT)	0.37	1.1	0.003	0.01	14	0.55

It is difficult to draw any conclusions from Table VI-2 regarding the effect of the cladding material on the doses. This is due to the different assumptions regarding fission product release and the improved techniques for calculating containment leakage and downwind doses which are employed in this analysis (these differences were previously discussed in Sections IV and V). However, if the same percentages of fission products released from the fuel used in the IDO-16981 analysis were assumed and the improved analytical techniques and pressure history calculation applied, it is estimated that the inhalation and deposition doses for stainless-steel would be approximately a factor of 40 lower than those presented for zirconium. Since the primary contributors to the cloud dose are the noble gases, this dose would be approximately the same in both cases.

The radiological doses presented in Tables VI-1 and VI-2 for distances greater than 10^4 meters (~ 6 miles) are within the exposure guide lines set forth in the Code of Federal Register (10CFR100). Since evacuation of the TSF area (nearest unshielded area to LOFT) can be accomplished within the time required for the radioactive cloud to reach the TSF area, the dose to on-site personnel is of no immediate concern.

It can be concluded, therefore, that if the postulated maximum credible accident should occur, the resulting radiological exposures will be below the AEC's exposure guide lines.

-
- (1) "Final Hazards Summary Report, Humboldt Bay Power Plant, Unit Number 3," Pacific Gas and Electric Company, September 1, 1961.
 - (2) N. R. Fabel, "Containment of Fragments from a Runaway Reactor," SRIA-5, October 1, 1958.
 - (3) R. W. White, N. B. Botsford, "Containment of Fragments from a Runaway Reactor," SRIA-6, September 15, 1963.
 - (4) Phillips Petroleum Company Program Number 40.0414.
 - (5) Private communications, U. S. Weather Bureau, NRTS.

Fig. VI-1 Containment Building Pressure Decay.

Fig. VI-2 Leakage Rate from Containment Building.

CHAPTER VII

CONCLUSIONS

VII. CONCLUSIONS

Since the issuance and review of the LOFT preliminary safety analysis report, the reactor design has been modified to replace stainless steel with zirconium as the fuel cladding material. A safety analysis of the consequences attendant to the conduct of a loss-of-coolant test and the postulated maximum credible accident on a LOFT reactor system utilizing zirconium-clad UO_2 fuel indicates that the associated hazards constitute an acceptable risk and will not jeopardize the health and safety of the general public.

From the analysis of the loss-of-coolant situation with zirconium, two major differences are apparent from the results reported for the stainless steel clad core. Because of the added energy from the metal-water reaction, the fuel temperatures are much higher within a shorter time period. This effect results in a greater fission product release from the fuel and, coupled with better transport mechanisms, results in a greater fission product release to the containment building. The higher concentration effects on leakage and downwind radiological doses are offset, however, by a much lower building leakage than reported for the stainless steel clad core. The lower leakage is primarily due to a calculated faster rate of pressure reduction, which has resulted from the development of better analytical tools to describe the containment response. The effects of liberated hydrogen burning within the containment volume have been treated in the analysis and do not have a significant influence on the pressure response of the containment.

From the analysis, the potential radiological doses during the test are shown in Table VII-1.

TABLE VII-1

RADIOLOGICAL DOSES
FOR THE PLANNED LOSS-OF-COOLANT TEST

Location	External Gamma Dose (rem) (Cloud + Deposition)***	Thyroid Dose (rem)	
		Inhalation	Ingestion
TAN Complex	1.7×10^{-2}	1.2×10^{-1}	
NRF	1.3×10^{-4}	4.3×10^{-3}	
**Site Boundary	4.9×10^{-4}	$1.5 \times 10^{-2*}$	$3.1 \times 10^{-1*}$

*Dose to a child.

**In the Howe direction.

***Deposition dose is for 5-day exposure.

If credit is allowed for the capture of fission product leakage through the monitored penetrations, the calculated doses will be considerably below the values presented. In addition, radiological measurements will be made both inside and external to the containment building throughout the test period. If it appears that the permissible off-site exposure will be exceeded, the test will be terminated by use of the pressure reduction spray system. In addition, certain additional procedures are available to insure that permissible exposures are not exceeded. These include the purchase of milk, limitations on cattle grazing, and evacuation of people in the downwind areas.

Direct radiation from fission products enclosed within the containment building is calculated to be negligible at the TAN complex (1.2 miles) which is the closest facility to the LOFT site.

The potential radiological doses to the general public at the nearest site boundary in the event of the postulated maximum credible accident are summarized in Table VII-2.

TABLE VII-2
RADIOLOGICAL DOSES
FOR THE MAXIMUM CREDIBLE ACCIDENT

Location	External Gamma Dose (rem)*	Thyroid Inhalation Dose (Iodine) (rem)
Site Boundary (Nearest)	9.7	250
NRF	2.1	58

*External gamma dose is the sum of the cloud gamma dose and 2-hour deposition dose.

On the basis of this addendum to the preliminary safety analysis, it is concluded that the loss-of-coolant test using a zirconium-clad core can be safely performed in the LOFT facility at its present location. It is further concluded that in the unlikely event of the postulated maximum credible accident the consequences will not present undue risk to on-site personnel or the general public.