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SNAP 10A Launch and Reentry Hazards Due to Steady-State Operation

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SNAP 10A

LAUNCH AND REENTRY HAZARDS

DUE TO STEADY-STATE OPERATION

By

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CONTENTS

TABLES

FIGURES

ABSTRACT

The safety analysis performed in support of the SNAP 10A flight tests has included consideration of potential hazards due to quasi steady-state operation of the SNAP 10A reactor under accident conditions. Steady-state operation is considered possible, although highly improbable, following launch pad accidents that could occur during or after fueling of the launch and orbital stage vehicles. In the extremely improbable event that reentry aerodynamic and impact forces fail to render the core permanently subcritical, steady-state operation is again considered possible. Potential radiation hazards due to accidents of this nature are examined in this report.

I. INTRODUCTION

The launch of a nuclear power unit (NPU) into a polar earth orbit, as planned for the SNAP 10A Flight Tests, will be similar in every respect to a routine launch of a nonnuclear payload. Ground handling tasks during final countdown will not be seriously affected by the presence of the NPU in the missile. The potential for accidental nuclear radiation in biologically significant quantities will be essentially nonexistent until routine pad evacuation is enforced prior to fuel loading. The fuel loading operations will introduce hydrogenous fluids into the vicinity of the reactor that could, in the event of an abort, cause accidental criticality. The rate of reactivity addition under these conditions would most likely be rapid, causing a supercritical excursion and concomitant destruction of the reactor. However, it is conceivable that reactivity could be added slowly enough to permit a stable rise to power, i.e., steady-state operation. Such an accident, involving missile fuel, would be expected also to involve deluge water that is used during pad aborts to control the ensuing fire. The deluge water would drain into a retention basin of sufficient size to permit a comparatively slow water level rise. Although obviously problematical, it is possible that abort forces could place the reactor in the basin prior to the arrival of the water. From an investigation of potential fluid retention cavities at the proposed launch pad (Pad 4, PALC 2, NMFPA), the retention basin is apparently the only cavity large enough to permit a slow fluid level rise under missile abort conditions.

Personnel would not be exposed to radiation from the operating reactor except by deliberate choice to enact emergency procedures intended to effect shutdown. From the radiological viewpoint, the major informational requirements to support emergency procedure planning are:

a) Direct dose rates near the basin edge, with sufficient detail to facilitate shielding calculations

b) Indirect dose rates near the basin edge, but within the shadow formed by the basin structure

c) Dose rate reduction with time due to reflector loss (water vaporization from reactor heat)

d) Duration of reactor operation if action to effect shutdown is delayed

e) Airborne contamination potential due to failure of fuel element cladding or to core destruction (excursion) during operation.

The required information has been obtained analytically, and the results are reported in the following pages. Although analyses of improbable accident consequences are useful, a successful launch is, of course, anticipated; and this anticipation is justified by the impressive reliability record of the particular launch and orbital stage vehicles to be employed.

As soon as the NPU is transported above the earth's atmosphere, disassembly of the reactor core due to aerodynamic heating will become an effective safeguard against postimpact criticality. However, certain design safeguards are necessary to the utilization of reentry ablative forces for this purpose. In particular, the beryllium reflector assembly that surrounds the vessel must be removed to expose the vessel walls, and the vessel must then ablate sufficiently to permit release of the fuel elements. To achieve these conditions, automatic reflector ejection early in the reentry phase will be provided by the ablation of a thin retention band. This method is not dependent on the reliability of the ground command or on-board automatic reflector release systems. A thin wall reactor vessel has been designed which also includes a forward lip weld. Due to these design features, sufficient disassembly of the reactor to eliminate postimpact criticality is virtually certain. The inherent capability of hydrogen moderator loss in a high temperature environment also has a significant role during reentry. The loss of only 20% of the hydrogen would eliminate the possibility of postimpact criticality, assuming that impact forces do not actually improve the core geometry. Loss of four central or seven peripheral fuel elements at impact would be sufficient to prevent criticality.

In the unlikely event that all design reflector ejection mechanisms fail, with the reactor continuing to operate in orbit for 100 years or longer, self-welding of the reflector assembly to the reactor structure is at present considered to be a possibility. The protection from aerodynamic heating and ablation afforded by the assembly could then conceivably leave the core intact and capable of criticality at the moment of impact on the earth. If impact is into water or moist soil, either the impact forces will separate the fuel elements, preventing criticality, or an excursion will occur. In either case, steady-state operation will be prevented. If impact is into dry soil, criticality is possible only if the core geometry survives, and reactivity is subsequently added (unless added slowly, an excursion will occur). Under conditions of slow reactivity addition, such as a slowly rising water reflector, the fuel may go critical, and remain so as long as reflector losses (due to boiling, percolation, etc.) are replenished, or until reactor damage causes irreversible shutdown. Damage of this type could include an excursion, or overheating of the upper, dry core section, with resulting hydrogen moderator release. Two additional restrictive conditions are required for such quasi steady-state operation. First, the fuel element cladding must be capable of retaining the hydrogen moderator during the rise to power, despite damage by reentry heat and impact forces. Second, the rise to power must occur in a stable manner which prevents core damage by initial power pulses. Should the steady-state operation actually occur, it is unlikely that personnel would be injured by the associated radiation. The probability of steady-state operation and personnel injury has been calculated as between 1×10^{-5} and 2×10^{-4} per reentry event,[1] depending on orbital lifetime.

Evaluation of the consequences of the quasi steady-state accident has required information concerning:

a) Dose rates near the reactor while operating

b) Duration of reactor operation

c) Conditions of personnel exposure

d) Airborne contamination potential.

With the exception of the third question, analytical estimates have been made to provide the required information. The conditions of personnel exposure are conjectural and not subject to technical analysis. Therefore, the report text does not deal specifically with the question. However, a few pertinent comments have been included in Appendix A.

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II. ANALYSIS RESULTS

A. LAUNCH PAD ACCIDENT

The reactor could assume the following terminal positions after a missile failure on the launch pad (Figure 1):

1) In the gantry crane track well at locations providing rectangular basins of 5-ft width and 8-in. depth between the rails

2) In the gantry crane track well pockets, near the drainage points, which provide rectangular basins of 1-ft, 10-in. width and 7-in. depth

3) On approximately level surfaces of the flume in a current produced by debris (Figure 2)

4) In the flume channel, possibly elevated by structural components

5) In the retention basin, elevated by structural components.

In these positions, fluids could be maintained at a level sufficient for criticality, but insufficient for a terminating excursion, i.e., partial submersion of the core. Since the position and attitude of the reactor with respect to the fluid level cannot be defined for any specific accident, the power level must be taken for analytical purposes as the maximum allowed by nuclear, thermodynamic, thermal, and hydraulic effects. Similarly, the operating time must be considered as limited by fluid availability only. Under these assumptions, reactor operation could be terminated by hydrogen moderator loss following failure of the upper, dry fuel element cladding. The maximum operating power level would be limited by either power oscillations of increasing amplitude, or by heat transfer. An upper limit analysis has been performed, assuming reactor stability to the point of cladding failure due to heat transfer. If the reactor remains stable with water in the core, the cladding is expected to fail at 130 kw.[2] Stability analyses have indicated that the reactor may become unstable before the 130-kw level is achieved.[3,4] However, instability has not yet been demonstrated, and a power level of 130 kw has been assumed for purposes of safety analysis.

In Reference 2, longitudinal conduction of heat from the dry to the immersed section of the core, and the evaporation of water carried by steam in the immersed section, are shown to be the only significant cooling mechanisms. The water level outside the vessel is critical, with reactor operation possible only

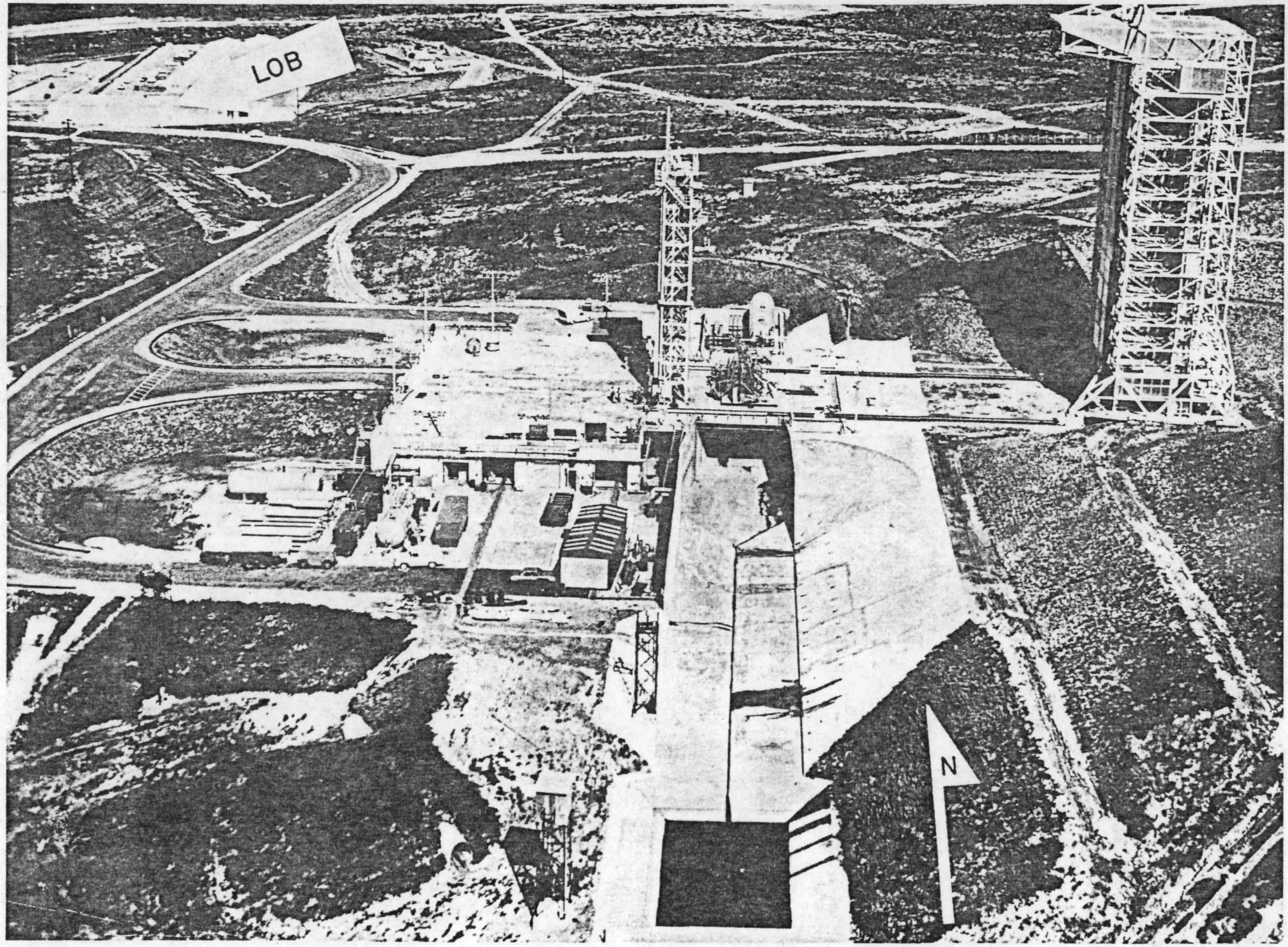

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Figure 1. Launch Complex (PALC-2)

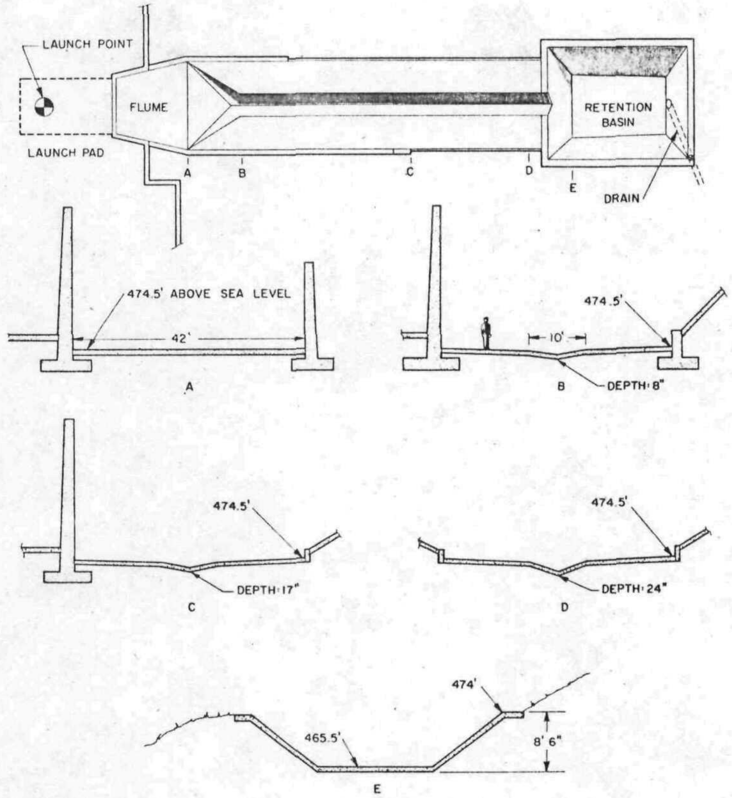

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Figure 2. Flume and Retention Basin

between about 6.5 to 8.8 in. water depth with the reactor in a vertical attitude. At the higher level, water is converted into steam at the rate of about 460 lb/hr, or approximately 1 gpm. If reactivity is added slowly, in the form of a gradually rising hydrogenous fluid, the reactor becomes cold critical at 6.5 in., hot critical at 6.9 in., and reaches 130 kw at 8.8 in. A rate of 20 in./hr would permit a power rise to 130 kw without an excursion, assuming stability. A stability analysis to determine the maximum rate of water addition, above which the cladding will fail due to the initial power pulse, has been made,[5] although conclusive data on this point are not yet available.

If an accident on the launch pad involving missile fuel should endanger the umbilical mast, an emergency water deluge and washdown system will be used. This system flows at 10,000 gpm. A minute of operation would put 2 ft of water in the low end of the retention basin. With the system operating, the water flow in the flume channel leading to the basin has been calculated to be 14 in. deep at midlength. The nature of a 10,000-gpm deluge, maintained for 1 min, together with the water rise limit indicated above, virtually eliminates steady-state operation in the flume or tracks. If the reactor were to fall into the retention basin before complete drainage of the deluge water to the basin, and if the reactor were to be elevated by structural components so that its closest point to the basin floor were near 9 in., the terminal deluge drainage, or rain drainage, could bring the reactor to power without an excursion. The final drainage volume required to raise the water level from 6.5 to 8.8 in. would be about 2000 gal, and could not exceed a rate of 20 in./hr without causing a terminating pulse. The elapsed time for final drainage, 2.3 in. at 20 in./hr, would be about 7 min. That 7 min could be required for the final drainage appears extremely unlikely aside from rainfall, and countdown during threatening weather is not anticipated. Although steady-state operation of the reactor in the retention basin is improbable, the possibility cannot be eliminated, and a radiation hazard analysis has been performed, assuming a power level of 130 kw and further that the power level is decreased only by loss of moderator due to steam formation.

Since the reactor would be partially submerged, the degree of shielding afforded by the water would depend on the power density in the core. To avoid eliminating fissions occurring above the water level, and to avoid unwarranted relaxation lengths for lower sections of the core, a sinuisoidal power density was assumed over the entire vessel length. The dose rate was calculated, at a fixed point, as a function of time. Following the achievement of 130 kw at an 8.8-in. water level, the dose rate would be decreased by the falling power level, and increased by the fission product buildup and shielding loss as retention basin water was converted to steam. A fixed point was chosen on the basis of access during the emergency, because persons would be exposed only if required to enact emergency procedures. A point at the edge of the direct beam, 3 ft from ground level, and 10 ft from the basin edge, was selected. This point is 39 ft from the reactor surface, as shown in Figure 3. Initially, the dose rate would be near 10^4 rem/hr, and would be reduced by power loss as shown in Figure 4. The total energy release would be about 2×10^4 Mw-sec, with 96% released during the first 100 hr. Indirect radiation would yield an initial dose rate of about 1600 rem/hr at a point slightly removed from the direct beam. These dose rates indicate that emergency action could be taken to effect shutdown, provided that the direct beam was avoided, and that sufficient shielding was

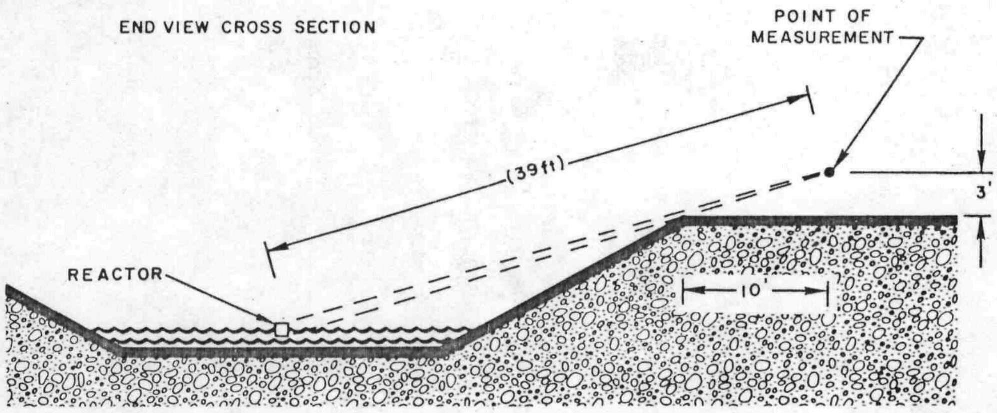

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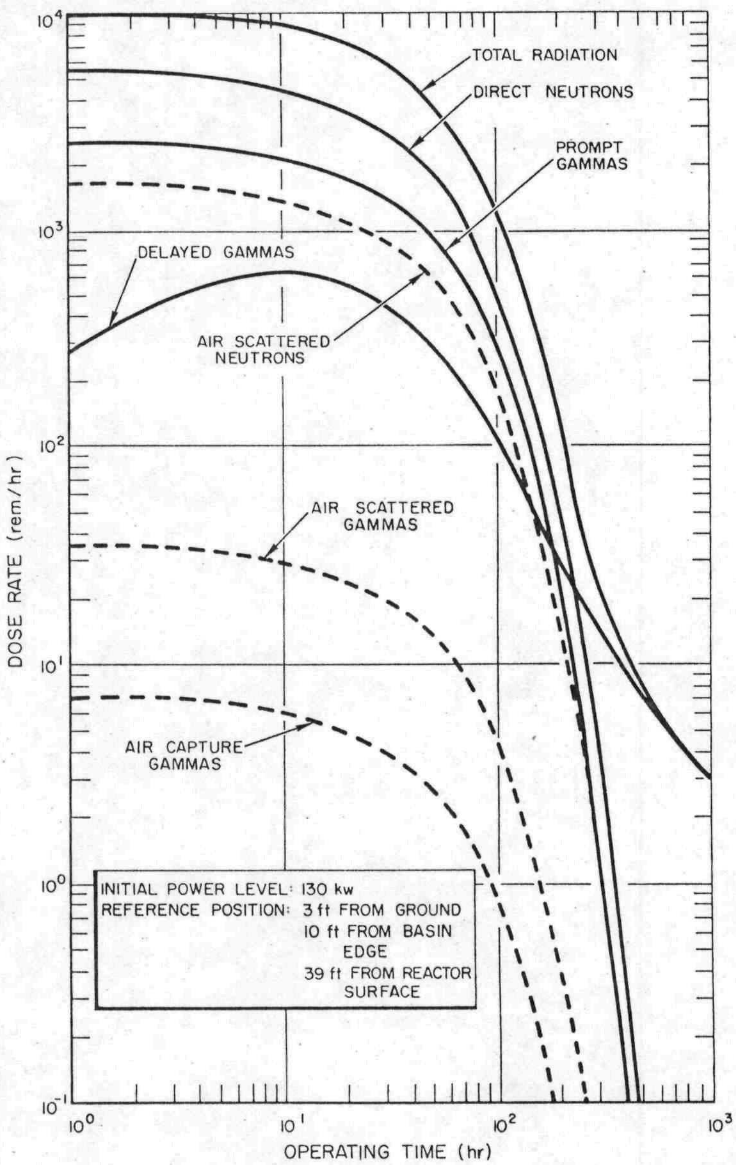
Figure 3. Quasi Steady-State Operation Configuration Assumed for the Retention Basin

Figure 4. Dose Rate Near Edge of Retention Basin



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provided for the indirect neutrons. Figure 4 indicates that dose rate reduction by power loss is slow, so that little would be gained by delaying emergency action.

Two important factors would reduce these dose rates in any actual case. A vertical attitude was assumed for the reactor, whereas a horizontal attitude would have a lower maximum power level.[2] Instability may set the upper power level limit considerably below 130 kw. Actually, the rate of reactivity insertion is likely to be too large to allow reactor operation for more than a few milliseconds.

Airborne contamination could result should reactor operation be terminated by fuel element cladding failure or by an excursion. The thyroid dose to unprotected persons downwind would be the most serious consequence. In Figure 5, the thyroid dose is shown as a function of distance downwind assuming 1 hr of operation at 130 kw, and 100% iodine release. Following one or more hours of operation, an excursion (70 Mw-sec) would generate a negligible increase in the dose, and the curve in Figure 5 is applicable to release by cladding failure as well as by an excursion. The dose would be increased by operating periods longer than 1 hr. Figure 6 presents the thyroid dose as a function of reactor operating time at distances of: 1400 ft, the distance to the Launch Operations Building; 3000 ft, the distance to Range Safety Personnel; 6000 ft, the fallback area radius; 10,000 ft, the impact limit line distance; and 18,000 ft, the distance to the nearest public community, Surf. The effect of a terminating 70-Mw-sec excursion is included. For operation periods not exceeding 1.7 hr, the dose would be less than 300 rem to exposed personnel. The calculations indicate larger doses at the Launch Operations Building distance, but all personnel would be protected by the building and ventilation equipment. The dose in areas of public access would be < 200 rem for operation periods of < 10 hr, and < 24 rem for periods of < 1 hr.

Assuming a reactor operating period of 1 hr, terminated by an excursion, the external cloud gamma dose at distances of particular interest would be: 11 r at the Launch Operations Building (unshielded dose); 4.2 r at the position of the Range Safety Personnel; 1.5 r at the fallback area radius; 0.7 r at the nearest impact limit line distance; and 0.2 r at Surf.

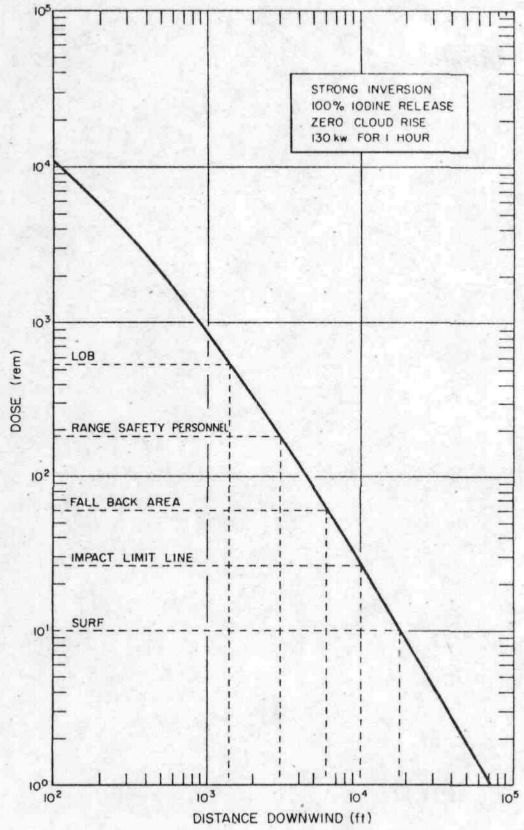

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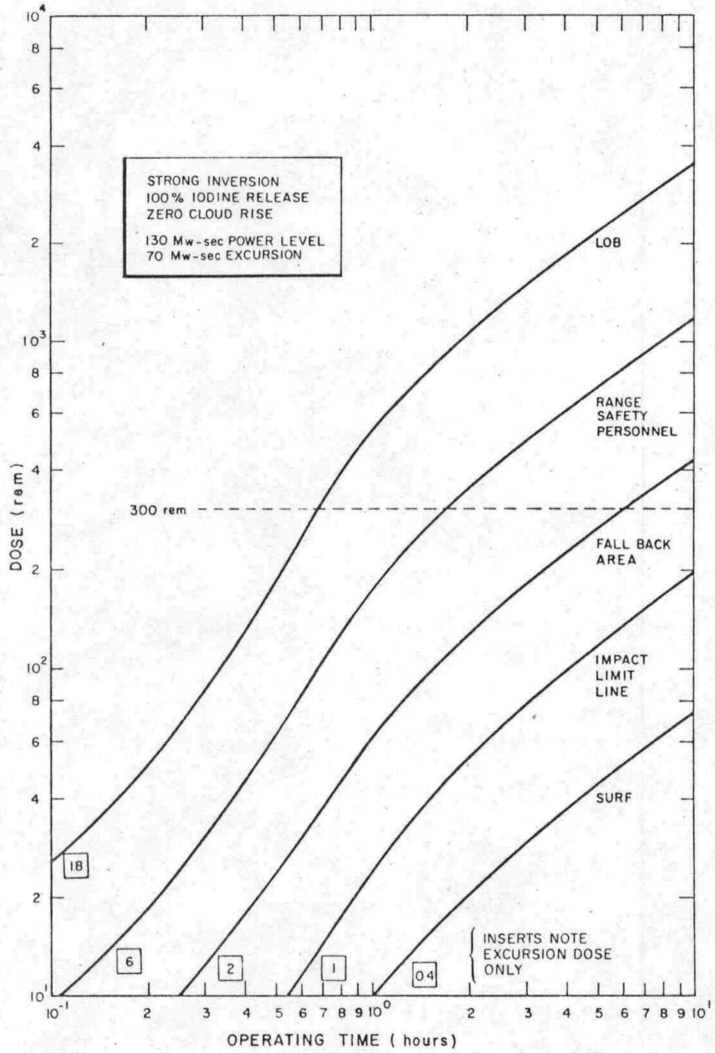
Figure 5. Thyroid Dose, Steady-State Accident on Launch Pad

Figure 6. Thyroid Dose vs Operating Time Steady-State Accident on Launch Pad



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Shutdown through enactment of emergency procedures would be expected to prevent air contamination, i.e., removal or poisoning of retention basin water could effect shutdown without fuel element cladding damage. After shutdown gamma dose rates from the core are shown in Figure 7 as a function of decay time for operating periods of 1, 3, and 10 hr at 130 kw. These data indicate the extent of shielding requirements for recovery procedures.

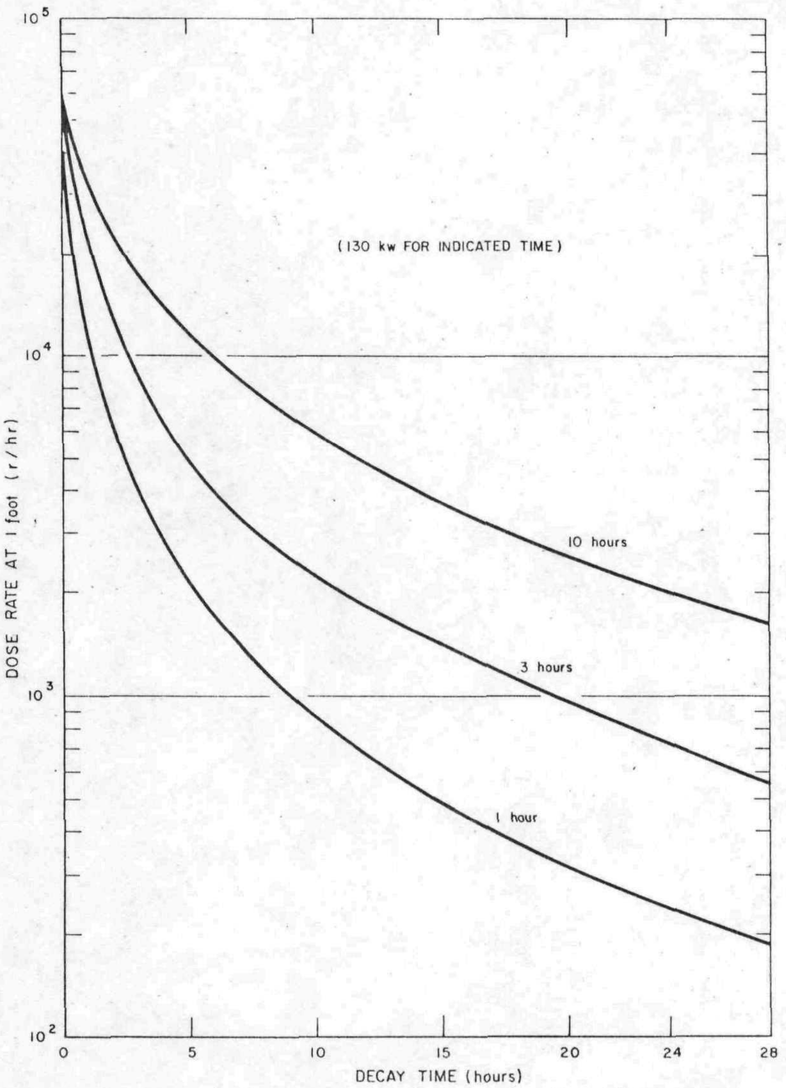

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Figure 7. After Shutdown Gamma Dose Rate

B. REENTRY ACCIDENT

Reentry with the reflector assembly self-welded to the reactor theoretically could result in survival of the core geometry following soil impact. An excursion would be expected if the reactor became completely buried in soil of sufficient neutron reflector content. Otherwise, impact in soil could leave the core subcritical, possibly in a crater. At a later time, additional reflector such as rainwater could become available. Under certain, specialized conditions, the reactivity input provided by this reflector could conceivably bring the reactor to power without core damage. Reactor operation would continue until exhaustion of the reflector, or until a change in reflector replenishment abruptly terminated the reactor operation by overheating the fuel and thereby releasing the hydrogen moderator, or by causing a supercritical excursion.

An estimate of the exposure magnitude may be obtained from Figure 8, which presents isodose curves around a particular crater for a reactor operating at a power level of 1 kw. The assumed crater configuration is shown in Figure 9. An analysis has been performed to determine the power level at which the upper core would achieve a temperature of 1600°F (at which hydrogen released from the fuel elements is expected to cause shutdown). The results indicate that it would be possible for 4.2 kw to be conducted from the upper core by the fuel, while radiation and steam cooling losses would be 0.9 and 0.8 kw, respectively. Thus, the reactor could maintain a maximum power output of ~6 kw. A person conducting a 1 min close inspection of the reactor would receive a dose of ~500 rem. Since this dose is between the lethal dose of 600 rem and the mid-lethal dose of 400 rem, circumstances only slightly more favorable than these would permit survival. Specifically, a power level $<$ 6 kw, an exposure duration of $<$ 1 min, a more distant point of closest approach, or combination of these would reduce the dose considerably.

The conditions for this accident are listed below:

1) Reentry survival of core geometry following impact on low reactivity soil

2) Subsequent reactivity addition, such as moisture content increase in soil around the core at a rate not exceeding the equivalent of a 20-in./hr water level rise

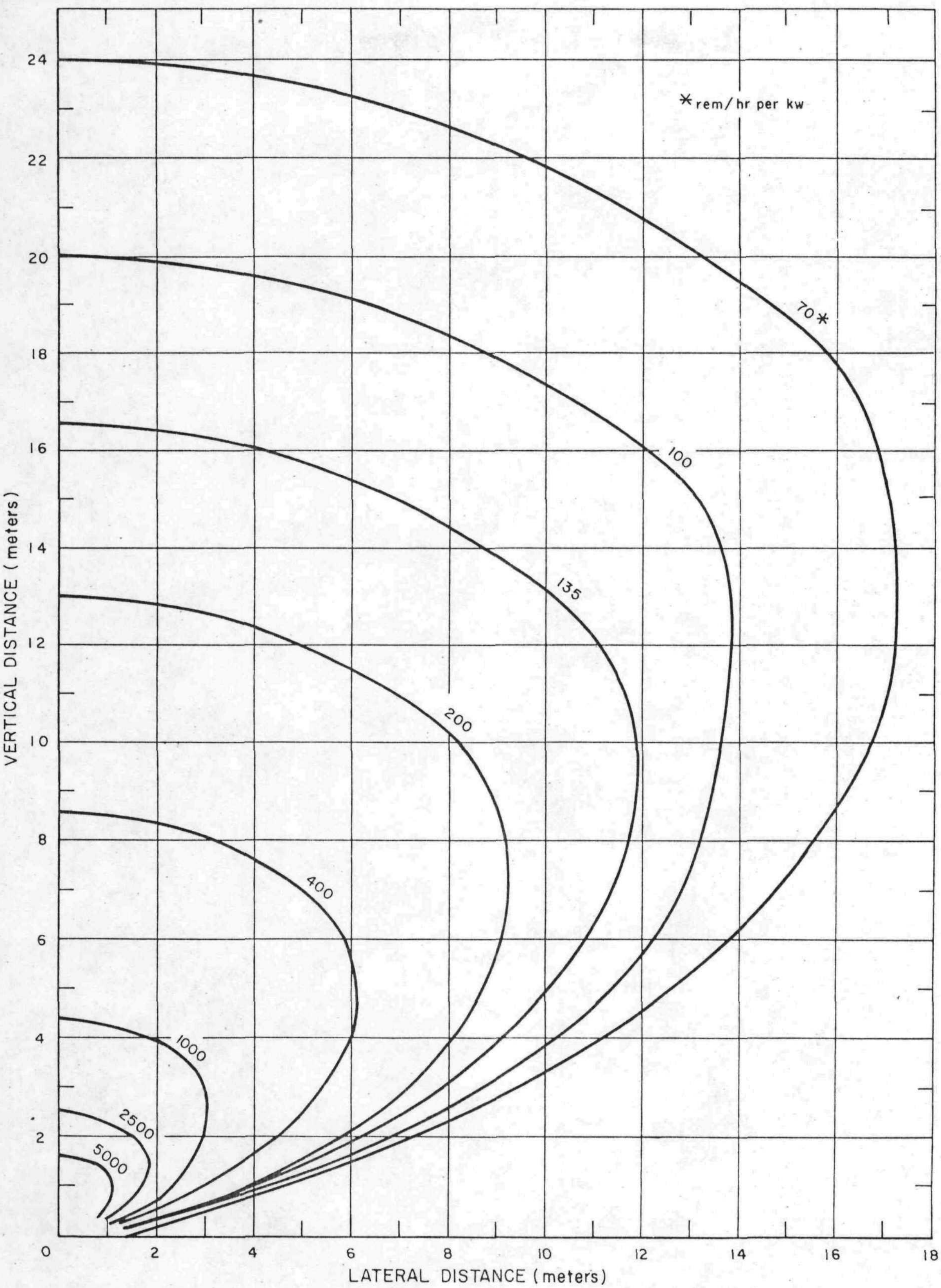

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Figure 8. Isodose Rate Curves for SNAP 10A Operation in a Crater

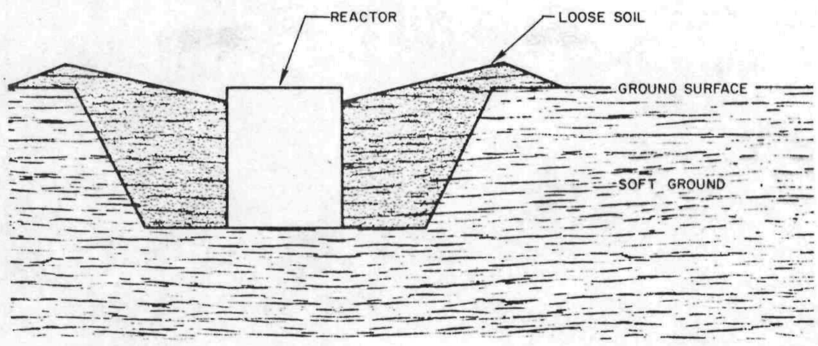

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Figure 9. Assumed Crater Configuration

3) Replenishment of water converted to steam by reactor heat

4) Total moderator/reflector height on vessel exterior, vertical attitude, not >10.8 in. with all water removed from the core interior.

5) Presence of personnel prior to irrevocable shutdown.

An independent study has been made of earth impact of a SNAP 10A core following reentry, considering the shape and depth of the crater, probable damage to the core for various earth types, with conclusions as to the credibility of a rain-filled crater incident.[6,7] The results of this study indicate that impact in soft dirt would permit survival of the core geometry, but the resulting crater would contain loose dirt sufficient to bury or partially bury the core, and bottom core inlets would be significantly plugged. This backfill and plugging would prevent unrestricted water and steam flow through the core. Under these conditions, the reactor could not operate at 130 kw,[2] but would be limited to 6 kw as mentioned. The crater would most probably be conical, about 3 ft in diameter, and about 30 in. in depth. Possible sources of moisture include rain, ground water, irrigation, river or lake overflow, or accidental, large-scale spill. The last three sources mentioned would cause rapid reactivity additions, destroying the core. Rain would provide the conditions for short-term, maximal energy release.

Those conditions are:

1) Cladding failure prevented during period of operation

2) Moisture replenished at $\sim$0.05 gpm for 6 kw power level during period of operation

3) Excursion caused at end of operation by increase in rate of moisture replenishment (increased precipitation rate, drainage breakthrough).

Ground water would provide the conditions for long-term maximal energy release:

1) Excursion prevented

2) Cladding failure prevented

3) Moisture replenished at about 0.05 gpm for 6 kw power level for long periods.

The 0.05 gpm replenishment rate limit was derived from Reference 2, in which it is shown that operation at 130 kw will cause a steam conversion rate of about 1 gpm.

The duration of the operating period, as caused by rain, would depend on the precipitation (moisture replenishment) rate, on the crater dimensions, soil percolation rates for water, and the volume capacity of the soil for water. The precipitation rate would also determine the power level, somewhat as indicated below, neglecting the hydraulic properties of the soil:

<0.05 gpm	Power rises until reflector loss equals replenishment; power is 6 kw.
0.05 gpm	Power rises to 6 kw and remains steady.
>0.05 gpm	Power rises as rising moisture level approaches 10.8 in.; continued rise increases power; upper core temperature exceeds 1600°F; shutdown by hydrogen loss.
>>0.05 gpm	Water reaches core interior from above or below, causing a supercritical excursion.

It can be readily shown for a 3-ft diameter crater that 0.05 gpm is roughly equivalent to a 0.6-in. /hr rainfall, which would be the maximum moderator replenishment rate that the reactor could withstand after a power level of 6 kw had been attained. In Section III, a procedure for estimating the most likely duration of a 0.6-in. /hr rainfall is described. The procedure yields a reactor operation duration of 5 hr, most of which would be at a power level considerably below 6 kw.

From the material presented to this point, it is possible to construct the probable sequences for this accident which would result in maximal energy releases. Six conditions for maximum energy release are common to all of the sequences.

1) The reactor core survives reentry impact on relatively dry, soft soil, forming a crater of conical shape, and leaving the reactor partially buried in soil loosened by the impact.

2) The loose soil forms a berm at the edge of the crater of sufficient height to prevent drainage of water into the crater.

3) The reactor is in a vertical attitude, with its top approximately at ground level.

4) The soil level on the vessel is 10.8 in.

5) The soil tapers upward toward the crater edge.

6) Inlets to the core from the bottom are soil filled to the extent that water must percolate through the soil to reach the core interior.

Sequence 1.

At some future time, rainfall into the crater saturates loose soil around the vessel; allowing water to rise slowly inside the core. The water level inside the core is approximately at the same height as the supersaturated soil on the outside. When this moisture level reaches about 6.5 in., the reactor goes critical. At about 6.9 in., power generation begins. The heat is sufficient to boil the water out of the core, while the moisture level on the outside continues to rise. The upper core temperature reaches 1600°F at an outside moisture level of 10.8 in. (the soil level; above 10.8 in., the cladding fails, releasing hydrogen and causing shutdown). Direct rainfall replenishes the moisture content of the loose soil at a rate of 0.05 gpm, allowing the reactor to operate at 6 kw until precipitation begins to diminish. The reactor is shut down when the moisture content of the soil falls sufficiently, probably about 5 hr after startup. This sequence may be repeated.

Sequence 2.

At some future time, the ground water table in the crater area rises to a level which allows drainage into the loose soil. The water level inside the core rises, while the loose soil absorbs moisture from below. When power generation begins, all of the water in the core is converted to steam. The reactor continues to operate at 6 kw until the ground water table recedes. This sequence may be repeated.

Sequence 3.

Ground water and rainfall combine to provide the required reflector, without damaging the core.

Sequence 4.

Before shutdown occurs in Sequence 1, 2, or 3, the fuel element cladding fails due to a precipitation rate increase, ground water flow increase, or mild drainage breakthrough. The escape of hydrogen shuts down the reactor. Volatile fission products are released and carried downwind.

Sequence 5.

Before shutdown occurs in Sequence 1, 2, or 3, a supercritical excursion is caused by a large increase in the precipitation or ground water flow rate, or by a drainage breakthrough, allowing water to enter the core. Hydrogen expansion destroys the core geometry. Fission products are released and carried downwind.

Data presented in Figure 10 represent a 70 Mw-sec excursion preceded by 5 hr of reactor operation at 6 kw. The prompt dose rate is shown as a function of lateral distance, assuming the configuration indicated by Figure 11. The dose rate in a parallel plane 3 ft above the ground surface would be <25 rem/hr at distances >32 ft. The prompt dose from the excursion (not shown) would be about 6×10^3 rem at 10 ft lateral distance, but <25 rem beyond 30 ft. The external cloud gamma exposure would be <25 rem at distances >110 ft. Should the reactor operation be terminated by cladding failure and hydrogen release (Sequence 4) rather than by an excursion, the cloud gamma exposure would be approximately one order of magnitude lower. The thyroid dose would be determined primarily by the tellurium-iodine buildup during the 5-hr operation. The thyroid dose curve in Figure 10 is therefore applicable to both Sequence 4 and Sequence 5. This dose would be <300 rem beyond 430 ft.

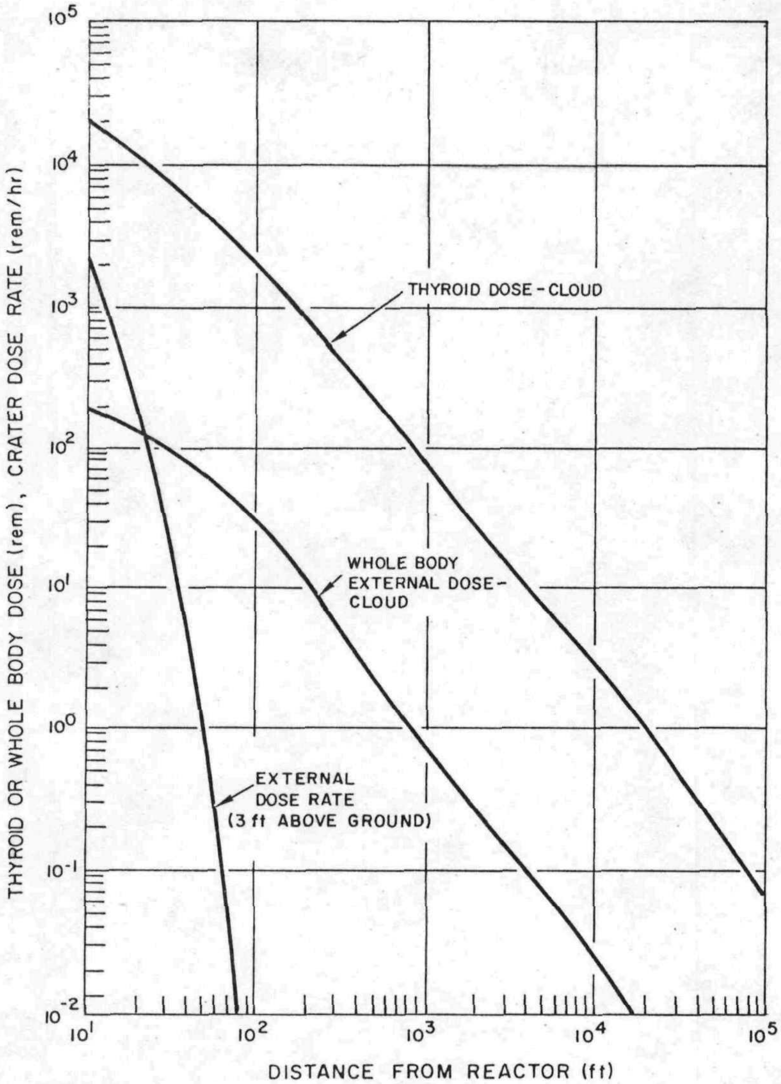

Figure 10. Radiation Hazards Due to Steady-State and Excursion—No Orbital Operation

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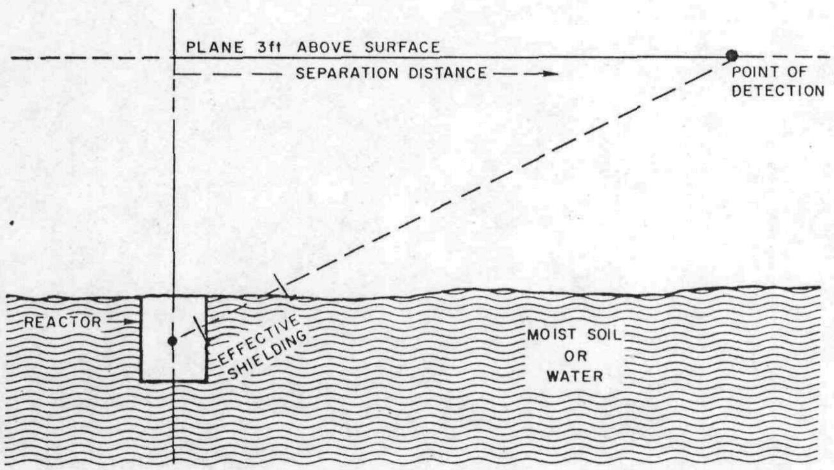

Figure 11. Shielding Geometry for Excursion Prompt Dose Calculation

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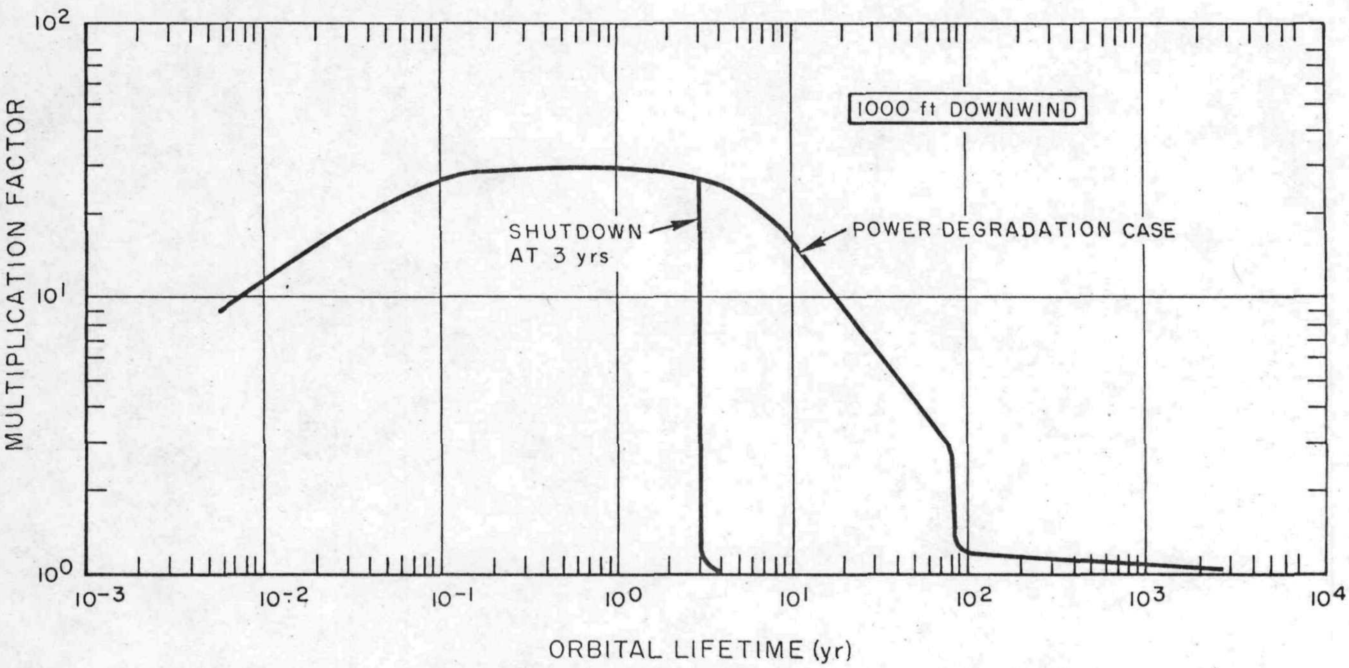
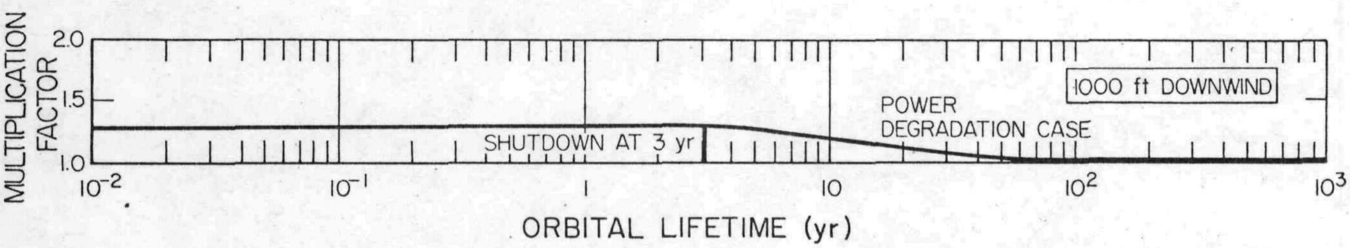
The fission product inventory generated by reactor operation in orbit would increase the thyroid and external cloud doses. The inventory effect is shown in Figure 12 in the form of multiplication factors vs orbital lifetime. Multiplication of data in Figure 10 by these factors provides the increased dose due to the inventory. Two power histories are indicated: (1) shutdown at 3 years, and (2) power degradation with coolant stagnation at 76 years.



6-17-64 Thyroid Dose 7561-01640A



6-17-64 Cloud Gamma Dose 7561-01634A

Figure 12. Multiplication Factors for Orbital Lifetimes Less Than 3800 Years

III. ANALYTICAL METHODS

A. BASIC DOSE DATA, EXCURSION

Figure 13 summarizes the variation of the gamma and neutron dose with distance as a result of an energy release of 70 Mw-sec. It is assumed that the reflector assembly is not present, and that the reactor is submerged in water or moist soil so that the upper surface is flush with the ground or water surface. The upper curve provides the dose at vertical distances above the reactor. The lower curves provide the dose in a parallel plane 3 ft above the ground or water surface. The amount of effective shielding associated with various distances from the reactor in this configuration is indicated. It is assumed that the reactor remains essentially intact during the excursion. The shielding geometry for these calculations is shown in Figure 11.

1. Gamma Radiation

The prompt fission gamma and neutron capture gamma dose rates at the reactor surface were obtained using the 14-0 code.[8] The fission product decay gamma dose rate at 1 ft, as a function of time, was obtained using the CURIE-DOSE code.[9] The dose variation with distance in the radial direction from the cylindrical source was obtained using geometrical techniques provided by Rockwell,[10] and air attenuation and buildup factors provided by Glasstone.[11] Broad-beam transmission factors obtained for prompt and fission product gammas were used to calculate the dose reduction due to water and moist soil shielding.[12] The moist soil was assumed 30% water by volume. These dose rates were integrated for a 70 Mw-sec excursion. The total dose calculations shown in Figure 13 include fission products gammas as calculated for equilibrium conditions of reactor operation, and are therefore maximal. These gammas contribute a small fraction of the total dose, e.g., <15% of the unshielded total at small distances.

2. Neutron Radiation

Neutron currents from the reactor vessel side surface were obtained using the AIM-6 diffusion theory code.[13]

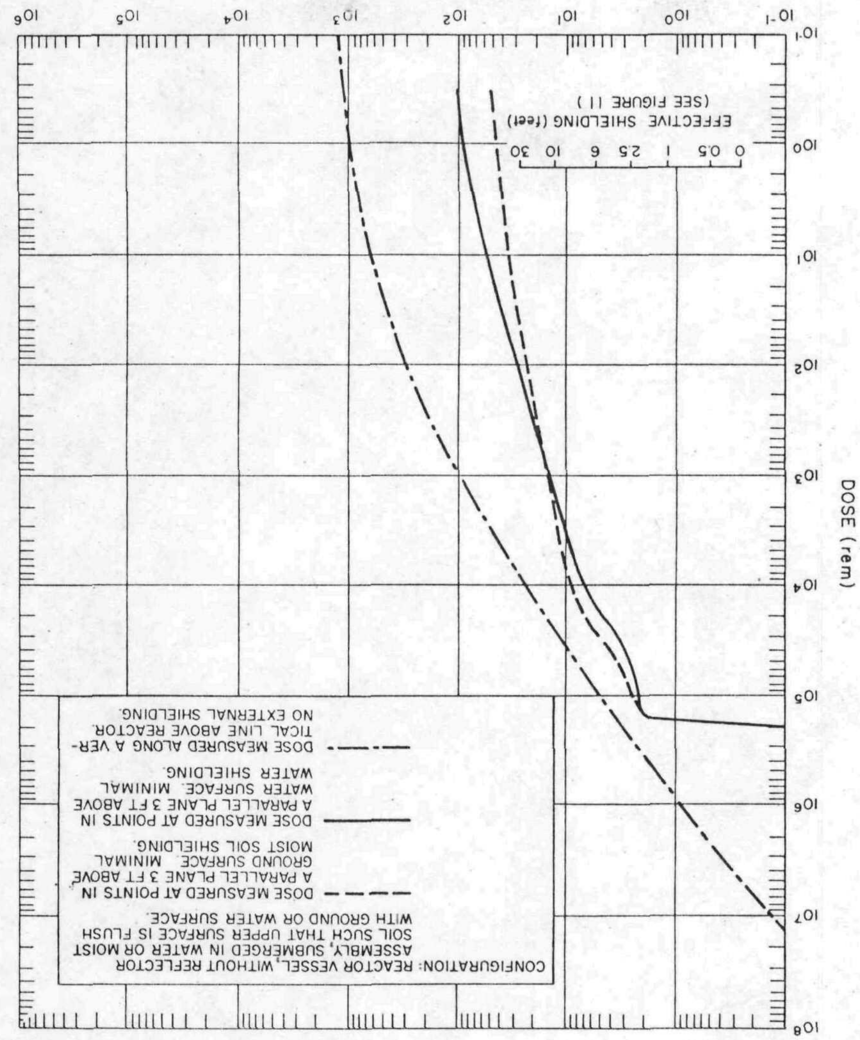


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Figure 13. Gamma and Neutron Dose as a Function of Distance for a 70 Mw-sec Excursion

a. Source Strength

Since the AIM-6 code assumes isotropic neutron flux and scattering in all regions, currents in outer regions are not rigorously accurate. Therefore, only currents given in the vicinity of the vessel surface were used for source strength data. Table 1 presents the method used for converting the current in various energy intervals (E + dE), to surface dose or dose rate. The dose due to each energy interval is the ratio of the total integrated flux per watt-sec φ, nvt (ΔE)/w-sec, to the dose conversion factor δ, nvt/rem) associated with the

TABLE 1

NEUTRON DOSE AT VESSEL SIDE SURFACE

Energy Interval E + dE (Mev)	$\bar{E}$ (Mev)	δ (nvt/rem)	φ [nvt(ΔE)/w-s]	Neutron Dose (rem/w-s)
10 - 3	6.5	2.5×10^{7}	2.09×10^{6}	8.36×10^{-2}
3 - 1.4	2.2	2.9×10^{7}	3.35×10^{6}	1.16×10^{-1}
1.4 - 0.9	1.15	2.7×10^{7}	1.33×10^{6}	4.93×10^{-2}
0.9 - 0.4	0.65	3.6×10^{7}	1.06×10^{6}	2.94×10^{-2}
0.4 - 0.1	0.25	6.5×10^{7}	9.35×10^{5}	1.44×10^{-2}
0.1 - 17 kev	59 kev	1.7×10^{8}	4.4×10^{5}	2.58×10^{-2}
17 - 3	10	6.7×10^{8}	3.3×10^{5}	4.93×10^{-4}
3 - 0.55	1.78	8×10^{8}	3.3×10^{5}	4.12×10^{-4}
0.55 - 100 ev	325 ev	7.8×10^{8}	3.2×10^{5}	4.1×10^{-4}
100 - 30	65	7.6×10^{8}	2.0×10^{5}	2.64×10^{-4}
30 - 10	20	8.2×10^{8}	1.5×10^{5}	1.83×10^{-4}
10 - 3	6.5	8.8×10^{8}	1.5×10^{5}	1.71×10^{-4}
3 - 1	2	9×10^{8}	1.4×10^{5}	1.56×10^{-4}
1 - 0.4	0.7	9.4×10^{8}	9.8×10^{4}	1.04×10^{-4}
0.4 - 0.1	0.25	9.7×10^{8}	8.0×10^{4}	8.25×10^{-5}
			Total	2.98×10^{-1} rem/w-s

average energy $\bar{E}$ of the interval. The surface dose is given by the sum. The dose at the reflector surface is approximately 1/3 the vessel side surface sum, while the surface dose at the ends is about 1/5 that of the vessel side surface.

The dose conversion factor, δ, was obtained directly from the Code of Federal Regulations, Title 10, Part 20, Standards for Protection Against Radiation, "Neutron Flux Dose Equivalents."[14] Factors for neutron energies not given in that document were obtained by interpolation.

b. Distance and Air Attenuation

The dose variation with distance in the radial direction from the cylindrical source was obtained using geometrical techniques provided by Rockwell.[10] Monte Carlo computations[15] performed at General Dynamics/ Fort Worth for determining neutron transport in air near the air-earth interface were used to obtain the effects of air scattering and attenuation. The General Dynamics data were selected because of excellent agreement with measurements from weapons tests. The data, as presented in Reference 15, are multiplied by the square of the distance from a point source, permitting direct application to any geometry.

c. Transmission

Broad-beam transmission factors obtained for 7.5 Mev neutrons incident on water and on sandy loam from Reference 12 were used to calculate the dose reduction due to water and moist soil shielding. These transmission factors include contributions from secondary gamma radiation. The moist soil was assumed to consist of 30% water by volume.

B. BASIC DOSE RATE DATA, REACTOR OPERATING

Dose rates from an operating SNAP 10A reactor are given as a function of distance in Figure 14. Prompt neutrons and gammas are shown on separate curves. A third curve shows prompt gammas plus equilibrium fission product gammas. The calculations were performed as described above in Section III-A.

Figure 14 provides working curves which do not apply to any actual configuration. The configuration used for the calculations may be described as a reactor operating in an infinite air medium with the reflector assembly removed. Thus the curves can be used with any shielding configuration.

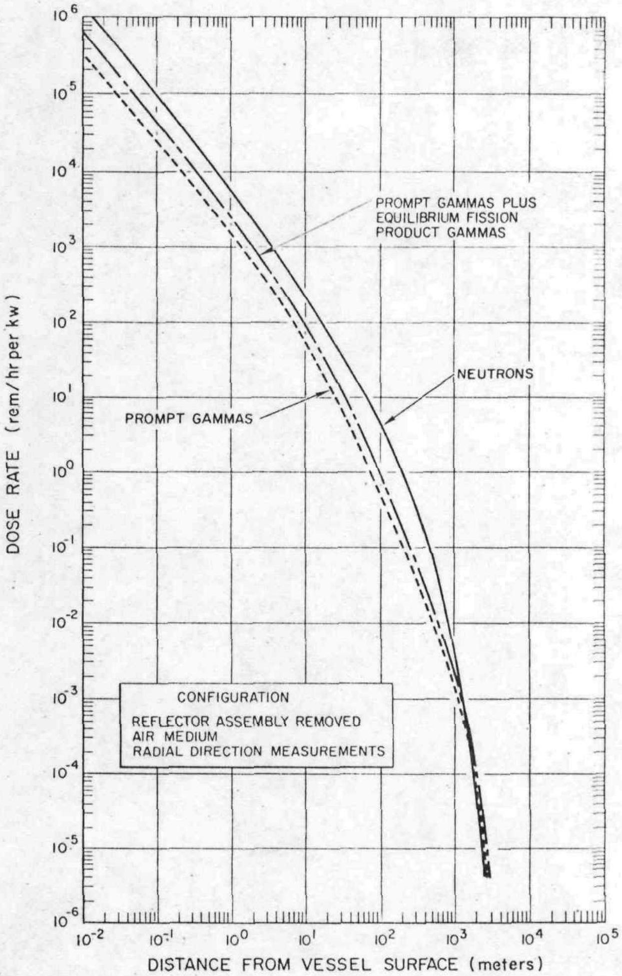

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Figure 14. Neutron and Gamma Dose Rate From Operating SNAP 10A Reactor

The unshielded gamma dose rate from an operating SNAP 10A reactor in the same configuration is shown in Figure 15 with fixed distance and variable operating time. The fission product gamma buildup is shown. Prompt gamma data were obtained using the 14-0 code, and CURIE-DOSE was used to obtain fission product gamma data.

C. BASIC AIRBORNE CONCENTRATION DATA

Airborne concentrations were determined using standard Sutton equations, and using a fission product release model of 100% halogens and noble gases, and 1% particulates. Ground release under inversion weather conditions was assumed. For the launch pad area, the following meteorological parameters were employed: windspeed $\overline{u} = 2$ m/sec; $n = 0.5$; $C_x = C_y = 0.12$; $C_z = 0.05$. These parameters reflect prevalent meteorological conditions at Point Arguello. For atmospheric releases following reentry of the system, the parameters recommended in the calculational procedures (TID-14844) for 10CFR100[16] were employed as worldwide averages: $\overline{u} = 1$ m/sec; $n = 0.5$; $C_x = C_y = 0.4$; $C_z = 0.07$. Thyroid doses were computed using the CURIE-DOSE code,[9] while cloud gamma doses were obtained from the code, THUNDERHEAD.[17]

D. DOSE RATE CALCULATIONS

The quasi steady-state operation analysis was based on the nuclear, thermodynamic, thermal, and hydraulic effects presented in Reference 2. Throughout the analysis it was assumed that the reactor had been brought to power by

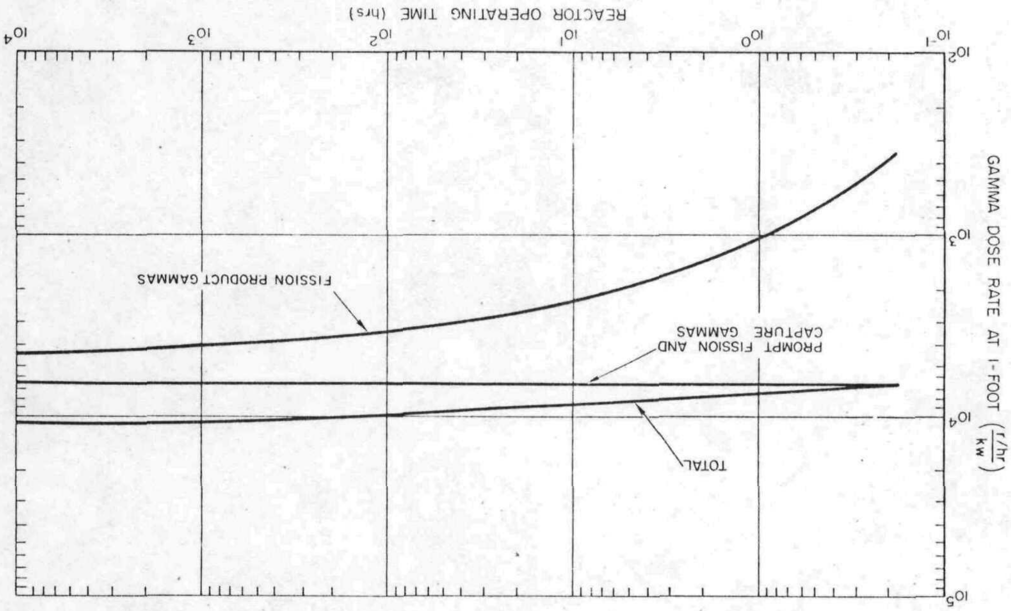

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Figure 15. Variation of Unshielded Gamma Dose Rate From Operating SNAP 10A With Time

reactivity additions at a rate sufficiently slow to prevent core damage due to an initial power pulse. It was further assumed that reactor stability would be maintained during the power rise to the point at which heating of the upper core would result in hydrogen release and shutdown.

1. Retention Basin Case

In this case, the reactor is positioned in a vertical attitude in the center of the retention basin, and elevated by structural components such that terminal fluid drainage brings the fluid level inside and outside the core to a height of 8.8 in. The power level is 130 kw.

a. Operating Time

It is assumed that no emergency procedures are enacted to effect shutdown. Degradation of power occurs as the fluid level is lowered by vaporization due to reactor heat. The fluid is assumed to be water rather than vehicle fuel. The assumption is conservative, allowing a longer period of operation.

Three equations were used to obtain the power level as a function of time. The rate at which water is converted to steam is proportional to the power level.

$$\frac{dV}{dt} = KP \qquad \ldots(1)$$

where

V = volume of water evaporated (gal)

t = reactor operating time (min)

P = power level (Mw)

K = steam generation constant (gpm/Mw).

The relation between the water level on the vessel and the power level is assumed to be of the form

$$w = ma^{P} \qquad \ldots(2)$$

where

w = water level on vessel exterior (in.)

m, a = constants for Equation 2.

Thus, the time rate of change for w is

$$\frac{dw}{dt} = ma^{P} \ln a \frac{dP}{dt} \quad . \qquad \ldots(3)$$

The volume of water that has been converted to steam is a function of an area factor and Δw.

$$V = A\Delta w \quad , \qquad \ldots(4)$$

where

A = area factor (gal/in.)

$\Delta w = w_{max} - w$

Here the time rate of change for w is

$$\frac{dw}{dt} = -\frac{1}{A}\frac{dV}{dt} \quad ; \qquad \ldots(5)$$

or, from Equation 1

$$\frac{dw}{dt} = -\frac{1}{A}KP \quad . \qquad \ldots(6)$$

The desired differential equation in P and t is obtained by equating Equations 3 and 6,

$$ma^P \ln a \frac{dP}{dt} = -\frac{1}{A}KP \qquad \ldots(7)$$

which has the solution

$$\ln\left|P\ln a\right| + P\ln a + \frac{1}{4}(P\ln a)^2 + \frac{1}{18}(P\ln a)^3 + \ldots = -K(Am\ln a)^{-1} t + C \qquad \ldots(8)$$

The series on the left converges rapidly for (Plna) less than one, so that use of only the first few terms is required. The constants for Equation 8 were determined as follows for the retention basin case.

From Reference 2, K may be derived as 7 gpm/Mw.

The constants m and a for Equation 2 are obtainable by graphing w(P) from points provided in Reference 2, and finding an approximate equation in the form of Equation 2 for the curve, using a common numerical analysis technique, and yielding

$$w = 6.1\ (20)^P$$

The area factor A is 810 gal/in. for the height considered in the retention basin. The water height for a 10,000-gal deluge establishes an effective area in the trapezoidal shaped basin of approximately 1300 ft^2 which is converted to gal/in. by the factor 7.5 per 12 gal/ft^2-in.

The integration constant C is -0.51, as calculated from initial conditions of 0.13 Mw power at zero time.

The equation for the operating time is

$$t = -2.35 \times 10^3 \left(\ln 3P + 3P + \frac{9}{4}P^2 + \frac{3}{2}P^3 + 0.51\right) \qquad \ldots(9)$$

and is shown graphically in Figure 16.

b. Energy Release

The total energy release, assuming shutdown by reflector loss due to steam generation, was determined by numerical integration as about 2×10^4 Mw-sec. Approximately 96% would be released during the first 100 hr.

c. Neutron Dose Rate

The reactor configuration is shown in Figure 3. The dose rates from the submerged and exposed sections were calculated separately, as a function of time, and summed. Only one distance was considered — 39 ft, as shown in Figure 3. The following general equation was used for the submerged section.

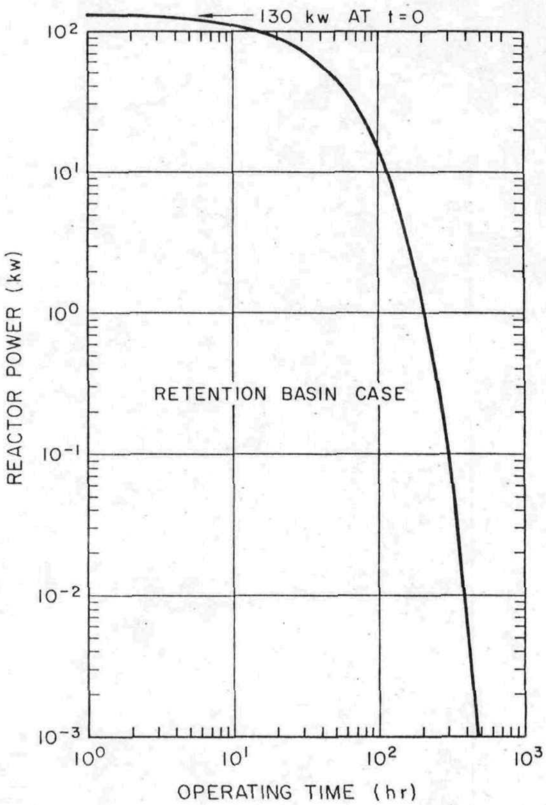

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Figure 16. Reactor Power Reduction From Loss of Water Reflector Due to Steam Generation

$$N_s\left[d, t(w)\right] = C(d)\ P(t)\ N(d_1)\ F(t) \int_0^w F'(h)\ T(h)\ dh \qquad \ldots(10)$$

where

$N_s\left[d, t(w)\right]$ = neutron dose rate from submerged section at d meters as a function of operating time, t hr, determined by the water level, w in. (rem/hr)

$C(d)$ = geometry factor for distances other than d_1, the reference distance

$P(t)$ = reactor power as a function of operating time (kw)

$N(d_1)$ = unshielded neutron dose rate in radial direction at d_1 (rem/hr-kw)

$F(t)$ = fraction of reactor power density submerged, as a function of operating time

$F'(h)$ = element of reactor power density shielded by water relaxation length associated with h

$T(h)$ = transmission through water for any h

h = distance from bottom of vessel to point of interest (in.)

For the exposed section,

$$N_e\left[d, t(w)\right] = C(d)\ P(t)\left[N(d_1)\ f(t) + N'(d_1, P, \overline{K}, \varphi)\right] \quad \ldots(11)$$

where

$N_e\left[d, t(w)\right]$ = neutron dose rate from exposed section at d meters as a function of operating time (rem/hr)

$f(t)$ = fraction of reactor power density exposed as a function of operating time

$N'(d_1, P, \overline{K}, \varphi)$ = scattered neutron dose rate from <u>axial</u> direction as a function of distance, power, polar angle $\overline{K}$, and azimuthal angle φ (rem/hr-kw)

The functions of Equations 10 and 11 were obtained as follows:

1) The geometry factor C(d) for this case is simply $C(d_1) = 1$, where d_1 is 39 ft. For other distances, cylindrical source data available from Rockwell (Reference 10) may be used.

2) The power factor P(t) is obtainable from Equation 9.

3) The unshielded dose rate $N(d_1)$ is obtainable from Figure 14.

4) The submerged fraction of the power density in the core, F(t), may be estimated by assuming a sinusoidal distribution over the entire 12-in vertical dimension of the vessel, i.e., Power Density Distribution = sin $\left[(\pi/12)h\right]$.

The cumulative distribution to a height h is

$$\int_o^h \sin\left(\frac{\pi}{12}h\right) dh \quad ,$$

and the fractional cumulative distribution, F(h), is

$$\frac{\int_o^h \sin\left(\frac{\pi}{12}h\right) dh}{\int_o^{12} \sin\left(\frac{\pi}{12}h\right) dh} = \frac{1}{2}\left(1 - \cos\frac{\pi}{12}h\right) \quad . \qquad \ldots(12)$$

The exposed fraction is

$$f(h) = 1 - F(h) \quad .$$

To express F(h) and f(h) as functions of time, it must be noted that the point of interest h is the water level w in this derivation, i.e., F(h) = F(w). Substitution of the function w = w(t) provides the required function F(t). From the established relation between power and time, Equation 9, and between power and water level, Equation 2, a very good approximation of w = w(t) is

$$w = 8.8t^{-0.053} \quad , \qquad \ldots(13)$$

and F(t) is readily obtained from Equations 12 and 13.

To obtain an expression for an element of power density, $F'(h)$, which would be shielded by a water relaxation length, h may be divided into divisions, h_i, separated by a distance Δh. Then

$$F'(h) = \frac{\int_{h_i}^{h_i+\Delta h} \sin\left(\frac{\pi}{12} h\right) dh}{\int_{o}^{12} \sin\left(\frac{\pi}{12} h\right) dh} = \frac{1}{2}\left[\cos \frac{\pi}{12} h_i - \cos \frac{\pi}{12}(h_i + \Delta h)\right] .$$

The accuracy of the computation is not seriously compromised by taking Δh as unity, yielding

$$F'(h) = \frac{1}{2}\left[\cos \frac{\pi}{12} h - \cos \frac{\pi}{12} (h + 1)\right] .$$

The transmission factor through water, T(h), involves a geometrical determination of x(h), the length of path through water between an element of power density, and the water surface, in the direction of the point of measurement. The product of x(h) and an absorption coefficient, μ, provides the relaxation length, such that

$$T(h) = e^{-\mu x(h)}$$

The distance from the vessel bottom to the point of measurement is 39.3 ft; from the 8.8 in. level to the measurement point is 39.1 ft. Assuming that these lines are parallel,

$$x = 2.2 \ (8.8 - h)$$

In Reference 18, the dose transmission of fission neutrons in water is shown to be essentially a straight line function, in a semi-logarithm plot, with a slope of about 0.254 in.$^{-1}$. Thus, the transmission at any height is

$$T = e^{-0.56\,(8.8-h)} \quad .$$

The scattered neutron dose rate at the point of measurement, $N'(d, P, \overline{K}, \varphi)$, due to neutrons emitted from the top surface of the vessel was calculated from Monte Carlo data published in References 19 and 20. The method of calculation is described below in the section on indirect dose rates.

With appropriate trigonometric substitutions, the integral of Equation 10 may be expressed as

$$1.37 \times 10^{-3} \left[e^{-0.56\,w} (\sin 0.26\,w - 0.465 \cos 0.26\,w) + 0.465 \right] \quad .$$

The expressions presented above may be substituted into Equations 10 and 11, solved for variable t, and summed to provide the total neutron dose rate at the point of measurement. The results are shown in Figure 4. Comparison with Figure 16 reveals that the effect of the variation in water shielding is almost completely masked by the power variation. Thus, it is demonstrated that for additional work (excepting fission product gammas), it is necessary to calculate only the initial dose rate, which is then reduced in time according to the power level. Initial conditions are

$$P(o) = 130 \text{ kw} \quad ,$$

$$N(39 \text{ ft}) = 160 \text{ rem/hr-kw} \quad ,$$

$$F(o) = 0.84 \quad ,$$

$$f(o) = 0.16 \quad ,$$

$$w = 8.8 \text{ in.} \quad ,$$

$$N'(39 \text{ ft},\ 130,\ 2.88,\ \pi) = 0.46 \text{ rem/hr-kw} \quad ,$$

yielding

$$N_s(39\text{ ft}, 0) + N_e(39\text{ ft}, 0) = 3.5 \times 10^3 + 3.4 \times 10^3 = 6.9 \times 10^3 \text{ rem/hr} \quad .$$

d. Gamma Dose Rate

The procedure for calculating the gamma dose rate is similar to the neutron dose rate procedure. The following general equations were used for prompt (prompt fission and capture) gammas.

$$G_{s,p}\left[d, t(w)\right] = C(d)\ P(t)\ G_p(d_1)\ F(t) \int_o^w F'(h)\ T_p(h)\ dh \quad , \qquad \ldots(14)$$

$$G_{e,p}\left[d, t(w)\right] = C(d)\ P(t)\ G_p(d_1)\ f(t) \quad , \qquad \ldots(15)$$

where

$G_{s,p}\left[d, t(w)\right]$ = prompt gamma dose rate from submerged section at d meters as a function of operating time, t hr, determined by water level, w in. (r/hr)

$G_{e,p}\left[d, t(w)\right]$ = gamma dose rate (as above) from exposed section (r/hr)

$G_p(d_1)$ = unshielded prompt gamma dose rate in radial direction at d_1 (r/hr-kw)

$T_p(h)$ = prompt gamma transmission through water for any h.

The new functions of Equations 14 and 15 were obtained as follows:

Values for $G_p(d_1)$ are obtainable from Figure 14.

The transmission factor through water, $T_p(h)$, may be estimated from the half-value-layer (HVL) of 6.8-in. of water for 2.5 Mev gammas.

$$T(x) = 2^{-n} \quad ,$$

$$x = x(h) \quad ,$$

where x(h) is the length of water path as previously defined, and n is the number of HVL's.

$$n = x/HVL \quad ,$$

$$x = 2.2(8.8 - h) \; .$$

$$T_p(h) = 2^{-0.324(8.8-h)} \quad .$$

The integral of Equation 14 is reducible to

$$0.036\left[2^{0.324\,w}(\sin 0.26\,w - 1.16\cos 0.26\,w) + 1.16\right] \; .$$

With these expressions for functions in Equations 14 and 15, the prompt gamma dose rate may be computed for variable time at the point of measurement, Figure 4. As with the neutron dose rate, variable shielding is masked by variable power level.

Initial conditions are

$$P(o) = 130 \text{ kw} \quad ,$$

$$G_p(39 \text{ ft}) = 36.6 \text{ r/hr-kw} \quad ,$$

$$F(o) = 0.84 \quad ,$$

$$f(o) = 0.16 \quad ,$$

$$w = 8.8 \text{ in.}$$

yielding a prompt gamma dose rate of

$$G_{s,p}(39 \text{ ft}, 0) + G_{e,p}(39 \text{ ft}, 0) = 1710 + 760 + 2.5 \times 10^3 \text{ r/hr} \quad .$$

The delayed gamma dose rate, to which fission products are the only significant contributer, was calculated from Equations 14 and 15, substituting the subscript f for p, considering the unshielded dose rate per kilowatt as a function of operating time, $G_f(d_1, t)$, and modifying $T_f(h)$ for the softer fission product gammas.

To simplify the determination of the unshielded fission product gamma dose rate per kw at 1 ft, $P(t)G_f(d_1, t)$ may be graphed as a function of operating time, as shown in Figure 15. From this figure the ratio G_f/G_p may be obtained for constant P. The product of this ratio and the total prompt gamma dose rate with variable P(t), as shown in Figure 4, provides an estimate of $P(t)G_f(d_1, t)$ for $0 < t < 100$ hr. (During the first 100 hr, 96% of the total energy is generated.) For $t > 100$ hr, fission product decay may be considered as controlling. The relation

$$I_2 = I_1 \left(\frac{\Delta t_2}{\Delta t_1}\right)^{-1.2}$$

where

I_1 = fission product gamma intensity at time of first measurement

I_2 = fission product gamma intensity at time of second measurement

Δt_1 = elapsed time between formation and first measurement

Δt_2 = elapsed time between formation and second measurement

may be used to estimate $P(t)G_f(d_1, t)$ for $t > 100$ hr by assuming that the fission products are formed at $t = 50$ hr.

The transmission factor through water, $T_f(h)$, may be obtained as for $T_p(h)$, using the HVL of 3.5 in. for 0.7 Mev gammas.

$$T_f(h) = 2^{-0.63\,(8.8-h)} \quad .$$

With these changes, Equations 14 and 15 may be solved for variable t and summed to provide the delayed gamma dose rate as shown in Figure 4.

e. Indirect Dose Rate

The air scattered neutron dose rate, the gamma dose rate due to air capture of thermal neutrons, and the gamma dose rate due to air scattered gammas were calculated to determine the feasibility of emergency reactor shutdown procedures.

From Reference 19, the total scattered neutron dose rate, $D_n(d)$, at a distance d from a point source in an infinite, homogeneous medium of air is given by

$$D_n(d) = \int_{E=o}^{E_{max}} \int_{o}^{\pi} \int_{o}^{2\pi} S(\overline{K}, \overline{\varphi}, E_o)\, \hat{D}_n\, (\overline{K}, \overline{\varphi}, d, E_o) \sin \overline{K} d\overline{K} d\overline{\varphi} dE \qquad \ldots(16)$$

where

$D_n(d)$ = total scattered neutron dose rate, rem/hr per source neutrons/sec

E = neutron energy (Mev)

E_o = initial neutron energy (Mev)

$S(\overline{K}, \overline{\varphi}, E_o)$ = neutron current of energy E_o moving in the direction $(\overline{K}, \overline{\varphi})$ at a point on the surface of a unit sphere

$\overline{K}$ = polar angle, measured with respect to source-detector axis

$\overline{\varphi}$ = azimuthal angle, measured between y, a normal coordinate to the source-detector axis, and the projection of the neutron direction on the plane formed by y and the third coordinate, z

$\hat{D}_n(\overline{K}, \overline{\varphi}, d, E_o)$ = Monte Carlo estimate of the neutron dose rate (at a detector on the source-detector axis) from a monodirectional source emitting one neutron per second of energy E_o in the direction $(\overline{K}, \overline{\varphi})$.

For a monoenergetic, isotropic source, Equation 16 may be written as

$$D_n(E_o, d) = 2\pi \sum_{1}^{j} S(\overline{K}_j, E_o)\, \hat{D}_n(\overline{K}_j, E_o, d) \sin \overline{K}\, (\Delta\overline{K}_j) \;, \qquad \ldots(17)$$

to permit a numerical integration technique, Reference 20. Values for $D_n(E_o, d)$ appear in Reference 20 for distances of 10, 35, 64, and 100 ft. The product of $D_n(E_o, d)$ and the source strength, $\sigma(E_o)$ for neutrons of energy E_o from the SNAP 10A vessel, provides the desired dose rate for a given E_o. A summation of the dose rate due to eight energy ranges then provides the total dose rate at a specified distance.

$$D_n(d) = \sum_{i=1}^{8} D_n(E_o, d)_i \, \sigma(E_o)_i \quad . \qquad \ldots(18)$$

Since interpolation of the data presented in Reference 20 appears inaccurate between energies other than the eight used, the SNAP 10A leakage spectrum was adjusted to match those eight energies, as shown in Table 2. For the spectrum adjustment for energies greater than 3 Mev, the U^{235} fission spectrum given in Reference 21 was used.

Table 2 also presents values of Equation 18. The source strengths, $\sigma(E_o)$, were obtained as described below. The total integrated flux per watt-sec was summed for the vessel side and end surfaces for each energy range. The sums were then multiplied by the vessel surface area, 3100 cm^2. Multiplying by the area is equivalent to converting a sphere of 3100 cm^2 area to a unit sphere, and thus provides a good approximation. The percentage of the total neutron dose contributed by scattering is shown in the final row of Table 2.

To estimate the air scattered neutron dose rate at 39 ft in the retention basin configuration, water shielding must be considered along with the integration limits for $\overline{K}$ and $\overline{\varphi}$. With respect to water shielding, it was assumed that the dose rate reduction would be equivalent to that for the direct beam. At 39 ft initially, this factor is 0.35. Concerning the integration limit for $\overline{K}$, the source-detector axis behind the reactor forms an angle of about 15° with the water level. The contribution to the dose rate at the point of measurement from neutrons emitted within this angle would be negligible. From Reference 20, this contribution in an infinite air medium is approximately 8%. For the integration

TABLE 2

SCATTERED NEUTRON DOSE RATE IN INFINITE AIR MEDIUM

E_o (Mev)	$\sigma(E_o)$ (n/sec-w)	$D_n(E_o, d)\,\sigma(E_o)$			
		d = 10 ft (rem/hr-w)	d = 35 ft (rem/hr-w)	d = 64 ft (rem/hr-w)	d = 100 ft (rem/hr-w)
0.33	5.0×10^{9}	1.95×10^{-2}	5.0×10^{-3}	2.5×10^{-3}	1.4×10^{-3}
1.1	1.2×10^{10}	9.25×10^{-2}	2.4×10^{-2}	1.3×10^{-2}	7.3×10^{-3}
2.7	1.6×10^{10}	8.00×10^{-2}	2.3×10^{-2}	1.3×10^{-2}	7.4×10^{-3}
4.0	2.3×10^{9}	2.07×10^{-2}	6.0×10^{-3}	3.2×10^{-3}	2.0×10^{-3}
6.0	2.3×10^{9}	1.60×10^{-2}	4.4×10^{-3}	2.4×10^{-3}	1.5×10^{-3}
8.0	1.8×10^{8}	1.12×10^{-3}	3.2×10^{-4}	1.8×10^{-4}	1.1×10^{-4}
10.9	1.6×10^{7}	1.07×10^{-4}	3.0×10^{-5}	1.6×10^{-5}	9.8×10^{-6}
14.0	3.6×10^{6}	3.14×10^{-5}	9.0×10^{-6}	4.5×10^{-6}	2.8×10^{-6}
$D_n(d)$		0.23	0.063	0.034	0.020
Percent of total		13%	26%	34%	50%

limit for $\overline{\varphi}$, all neutrons emitted below the source-detector axis were neglected, a reduction of 0.5. Thus the initial air-scattered neutron dose rate would be approximately

$$0.06 \text{ rem/hr-watts} \times 1.3 \times 10^{5} \text{ watts} \times 0.35 \times 0.92 \times 0.5 = 1600 \text{ rem/hr}.$$

Degradation of this dose rate with time would follow the power level degradation, as shown in Figure 4.

References 22 and 23 were consulted for the calculation of the gamma dose rate at the point of measurement due to air capture of thermal neutrons. Reference 22 provides Monte Carlo estimates of $\hat{D}_g(\overline{K}, \overline{\varphi}, d)$, in units of r/hr per source neutron/sec. The air capture gamma dose rate, $\hat{D}_g$, due to one thermal neutron emitted in the $(\overline{K}, \overline{\varphi})$ direction from a unit sphere, monodirectional source, is graphed there as a function of $\overline{K}$. To obtain an effective source strength to dose conversion factor at the three distances provided, the curves of $\hat{D}_g$ vs $\overline{K}$ were integrated numerically for $0° < \overline{K} < 180°$, and then divided by 180.

$$C_{eff} = \frac{1}{180°} \int_0^{\pi} \hat{D}_g(\overline{K}, \overline{\varphi}, d) \sin \overline{K} dK \quad .$$

To convert the reactor source to a unit sphere, monodirectional source, the thermal flux per watt-sec, φ, was multiplied by the area, providing a unit sphere source, and by 4π, providing a monodirectional source. This calculation yielded a source strength of 1.4×10^{13} thermal n/sec-kw. The air capture gamma dose rate, $D_g(d)$, was obtained as 1.4×10^{13} C_{eff}, r/hr-kw, as shown in Table 3. The value of $D_g(39$ ft) was obtained by smooth curve interpolation as 3×10^{-2} r/hr-kw.

TABLE 3

AIR CAPTURE GAMMA DOSE RATE IN AIR MEDIUM

d (ft)	C_{eff} (r/hr-N_o/s)	Source Strength (N_o/s-kw)	$D_g(d)$ (r/hr-kw)
33	2.2×10^{-15}	1.4×10^{13}	3.2×10^{-2}
64	1.1×10^{-15}	1.4×10^{13}	1.6×10^{-2}
100	4.4×10^{-16}	1.4×10^{13}	6.5×10^{-3}

To estimate the additional contribution of neutrons thermalized by the air, the leakage spectrum from the reactor for energies less than 10 kev was normalized to the thermal flux. Reference 23 provides $D_g(E_o, d)$ from a point, isotropic, monoenergetic source as a function of energy, for d = 50 ft. These data were normalized to the thermal neutron value of D_g. The sum of the products of these normalized values provided an estimate of the factor of increase over D_g (50 ft) for thermal neutrons only, as shown in Table 4. The factor of

TABLE 4

INCREASE FACTOR FOR AIR THERMALIZED NEUTRONS

E_o (ev)	Normalized Spectrum (n_o/thermal n_o)	Normalized D_g (50 ft) $[D_g(E_o)/D_g(0.025)]$	Product
0.025	1.00	1.00	1.00
0.25	0.38	0.54	0.21
0.7	0.29	0.38	0.11
2.0	0.38	0.29	0.11
6.5	0.47	0.20	0.09
20	0.47	0.14	0.07
65	0.65	0.10	0.07
325	0.88	0.06	0.05
1,780	1.00	0.04	0.04
10,000	1.68	0.02	0.03
		Sum	1.8

1.8 for D_g (50 ft) was applied to D_g (39 ft) without distance correction, yielding 5.4×10^{-2} r/hr-kw. Thus, it appears that D_g (39 ft) for 130 kw would not be more than 7 r/hr for the retention basin configuration, and would degrade with power as shown in Figure 4.

Concerning details of the source strength calculation, it was assumed that the flux and spectra from the submerged section would be equivalent to leakage with the reflector assembly in position, and AIM-6 code values for that configuration were used. For the exposed section, flux and spectra data for the bare core configuration were used. The area of the submerged section was obtained using the reactor diameter with the reflector assembly in position, while the vessel diameter was used for the exposed section. The reflector leakage spectrum was used to obtain the normalized spectrum in Table 4.

Reference 22 data include thermal neutrons reflected from concrete, assuming a source height of 12.5 ft. This contribution is very significant, and would be greater in the retention basin configuration. However, this increase was counterbalanced, at least in part, by neglecting the attenuation effects of the water in the basin.

Reference 24 provided a ready estimate of air scattered gammas. Calculations performed by the 14-0 program[8] and reported in Reference 24 provide ratios of air scattered gammas to air scattered neutrons, and of air scattered gammas to neutron air capture gammas. As shown in Figure 4, air scattered gammas initially would yield a dose rate of 32 r/hr at the point of interest.

2. Crater Case

In this case the reactor is positioned in a vertical attitude in a soil filled crater as shown in Figure 9. Rainfall into the water, or ground water seepage, provides the reactivity required for operation at 6 kw. With the wide variation of possible reactor attitudes and crater shielding configurations, detailed dose rate calculations are not justified.

a. Operating Time

World record rainfall data, plotted as amount vs duration, indicate that a 0.6 in./hr rainfall would never exist longer than about 20 days.[25] However, the procedure for averaging rainfall data eliminates peak periods which would terminate reactor operation. Also, such rainfall has been recorded only in Jamaica and India, so that the 20-day upper limit does not represent a large fraction of the earth's land surface. A more realistic approach to a duration limit would be to establish a representative percentage of time during which the precipitation rate is less than 0.6 in./hr but greater than some small rate associated with negligible power generation. For example, in Washington, D.C., 0.6 in./hr rainfall has been exceeded, on the average, 9 hr/yr adjusted instantaneous rate. A rate of 0.06 in./hr has been exceeded, on the average, 167 hr/yr. These numbers define a time period of 158 hr/yr for significant reactor operation. The ratio of this period to the frequency of rainfalls with an average precipitation rate between 0.06 and 0.6 in./hr would provide a reasonable estimate of a single reactor operation duration. Reference 25 reports that the total annual precipitation in Washington occurs over an integrated period of about 650 hr. Apparently, the precipitation rate is between 0.06 and 0.6 in./hr ~24% of the time. The precipitation is measurable an average of 124 days/yr. The frequency of rainfalls with an average precipitation rate between 0.06 and 0.6 in./yr may then be estimated as about 30 times per year. This estimate yields a reactor operation duration of about 5 hr, most of which would be at a power level considerably below 6 kw. U.S. Air Force designers consider the rainfall in Washington to be fairly representative of world-wide rainfall conditions.[25]

b. Dose Rate

The isodose curves appearing in Figure 8 were obtained by application of appropriate transmission factors to the unshielded dose rates shown in Figure 14. Sources for the transmission factors are given in Section III-A. The moist soil was assumed 30% water by volume. Thus, the total shielding distance d', as determined by an angle, θ, provided the water shielding as $0.3\ d'$, and the soil shielding as $0.7\ d'$. The dose rate along a line formed at a given angle, θ, is

$$Z(d, \theta) = Z_n(d)T_n^s(\theta)T_n^w(\theta) + Z_{p\gamma}(d)T_{p\gamma}^s(\theta)T_{p\gamma}^w(\theta) + Z_{d\gamma}(d)T_{d\gamma}^s(\theta)T_{d\gamma}^w(\theta)$$

where

$Z(d,\theta)$ = total shielded dose rate at distance d along angle θ (rem/hr-kw)

$Z_n(d)$ = total unshielded neutron dose rate at d (rem/hr-kw)

$Z_{p\gamma}(d)$ = unshielded prompt gamma dose rate at d (r/hr-kw)

$Z_{d\gamma}(d)$ = unshielded delayed gamma dose rate at d (r/hr-kw)

$T^s(\theta)$ = broad beam transmission through soil for radiation indicated by subscript, as a function of θ

$T^w(\theta)$ = broad beam transmission through water for radiation indicated by subscript, as a function of θ.

The unshielded dose rates in Figure 14 include air scattering. Application of the transmission factors to the scattered radiation components has a compensating effect. Radiation emitted in upper hemispherical directions would yield scattered components not necessarily shielded by soil or water. Radiation emitted in lower hemispherical directions would have some scattered components completely eliminated.

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APPENDIX A
EXPOSURE CONDITIONS

The consequences of the quasi steady-state accident may be considered from the viewpoint of accumulated dose at specified distances from the crater, or from the viewpoint of the dose rate as a function of distance. The accumulated dose is of particular interest in a populated zone, whereas the dose rate is more significant in rural areas where deliberate, short-term investigation would be the expected mode of exposure. Repeated personnel exposures in a populated zone would imply the absence of any action on the part of persons in the zone following impact of the comparatively large object. Such inaction is considered incredible, and only dose rates from an operating reactor in a rurally located crater have been considered in this report.

The number of persons exposed, and the exposure duration, are conjectural. However, it is reasonable to develop a logical theory based on present day experience. Accordingly, it may be presumed that steam rising above the reactor could attract the notice of a small group. The phenomenon would be investigated; natural precautions would probably limit the number of persons making a close approach. The sight of a metallic object in a crater, generating steam, would arouse a natural fear of danger, such as an explosion, probably resulting in rapid withdrawal from the immediate area. The exposure would probably range between one minute at one meter, and one hour within 10 meters. The object would in all likelihood be reported to authorities before additional close exposures occurred. Since radiation detection equipment is already present in virtually every nation, authorities would be expected to establish nuclear energy as the source of heat prior to further examination of the object.