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SPECIAL

SUPPLEMENT TO SRE SAFETY ANALYSIS REPORT

ATOMICS INTERNATIONAL
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Sodium–Cooled Reactor Experiment at Santa Susana

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I. CREDIBLE ACCIDENTS

A. INTRODUCTION

This report is a preliminary study; the final version will be a supplement to SRE Safety Analysis Report.[1] Modifications to the existing SRE reactor core involve the replacement of the central seven moderator elements with a fast-fuel test section. This installation provides a fast-flux trap in the center of the reactor capable of material testing in a representative fast-reactor environment. The report is an evaluation of the possible effects on the safety and containment features of the SRE facility, associated with the operation of the reactor with the fast fuel test section installed. It describes the building and containment features of the SRE and gives the results of the kinetic analysis before and after proposed core modifications. Effects of "noncredible" reactor excursions on the SRE containment structure are given in Section II. Means of providing additional containment for these excursions and environmental control are described in Section III.

The Sodium Reactor Experiment (SRE) has been operating since 1957 and in the last seven years has accumulated considerable operational and maintenance experience in addition to the generation of 31 million kilowatt hours of electrical power in 27,000 hours of successful operation. As a test facility, fuel elements of U - 10 Mo, UC, Th-U, and U-oxide have been irradiated in a thermal neutron flux. The SRE has served to confirm the design technology of the integrated sodium-cooled process systems. Experience accumulated through years of SRE power operations was used in the SRE-PEP design and greatly improved the heat removal and instrumentation system reliability. Experience with the SRE system provides the sound basis for the safe, reliable operation of the proposed Two-Region Sodium-Cooled Reactor Experiment (TR-SRE).[2]

The TR-SRE is specifically designed to function as a "Fast-Flux Trap" fuel and material irradiation facility. In this capacity, the TR-SRE will be testing materials in a fast-reactor environment.

The required facility modifications for the TR-SRE involve the reactor, secondary heat transfer system, fuel handling, and service facilities. The reactor modifications include removal of the central seven PEP moderator elements and the insertion of a central core structure with a hexagonal array of nineteen fast-fuel channels (please refer to Figure 1). There are six decoupler elements between the fast-flux and thermal-flux sections of the core. A new 40-in. diameter central plug will be installed in the reactor loading-face to accommodate the smaller close-packed instrumented fuel elements and material tests in the central test region.

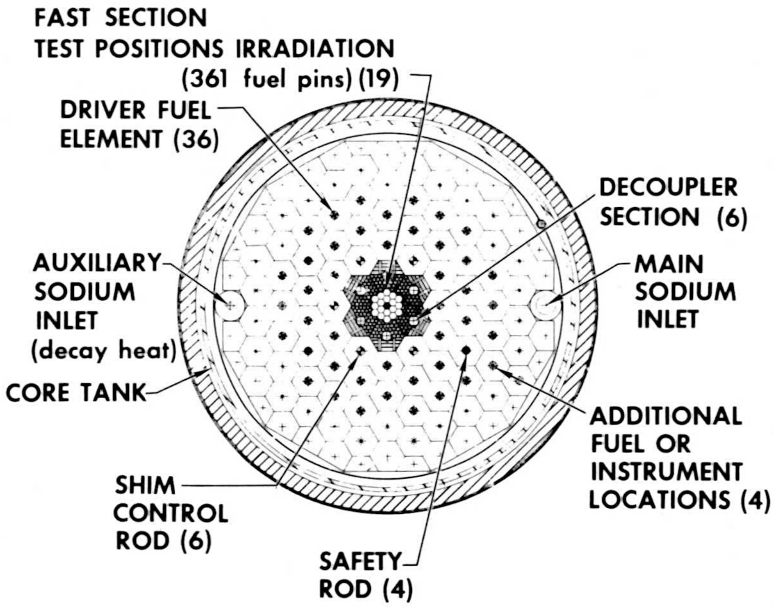


Figure 1. Cross-Section of Two-Region SRE Core

The fuel handling equipment will be modified to improve environmental control around the fast fuel elements during fuel handling.

The principal modification to the sodium heat transfer system is the addition of a 20-Mwt heat sink to allow 50-Mwt reactor operation. The existing SRE main airblast heat exchanger will be modified and included in the main secondary-sodium heat transfer system. The main airblast will be operated in parallel with the existing turbine plant, providing the overall 50-Mw thermal capacity.

All other existing SRE-PEP systems will be used in the TR-SRE operations; SRE-PEP system descriptions are applicable to the TR-SRE.

B. SUMMARY AND CONCLUSIONS

Core dynamic studies show that the kinetic response of the SRE with the fast-fuel test section remains that of a thermal reactor (please see Section I-E).

The proposed control and plant protective systems are those existing for SRE with the addition of two shim rods. Present safety studies show that the installation of a fast-fuel test section (FFTS) will not compromise the operational reliability or safety characteristics of the reactor power plant.

The present containment structure surrounding the SRE core provides a conservative degree of confinement for radioactivity release under credible accident conditions. As now conceived,[2] no modifications to building containment is required for the installation of the fast-fuel test section in the SRE reactor core. SRE containment and radioactive waste disposal systems are applicable to the operation of the SRE with the FFTS installed.

C. SRE BUILDING AND REACTOR CONTAINMENT DESCRIPTION

A brief description of salient containment features is presented below.

1. Building

The existing SRE building is constructed of reinforced steel and tilt-up concrete slabs (see Figure 2). The reactor building has an area of more than 10,000 square feet. Exhaust fans and high efficiency filters on the reactor building roof are sized to maintain the reactor room at a negative pressure compared to all the surrounding environments. Two identical vents of exhaust fans and filter banks are provided. Only one unit is required to maintain reactor room atmospheric control. Although it provides some measure of environmental control, the building itself is not considered as a containment structure. With all ventilating equipment shut off, the building leak is estimated to be 4,000%/day; consequently, the main factors affecting fission product management are associated with the reactor structure, auxiliary systems, and handling equipment.

All sodium components in the building (pipes, pumps, and vessels) containing radioactive materials are maintained as a completely enclosed system in below grade vaults and galleries sealed from outside atmosphere (see Figure 3).

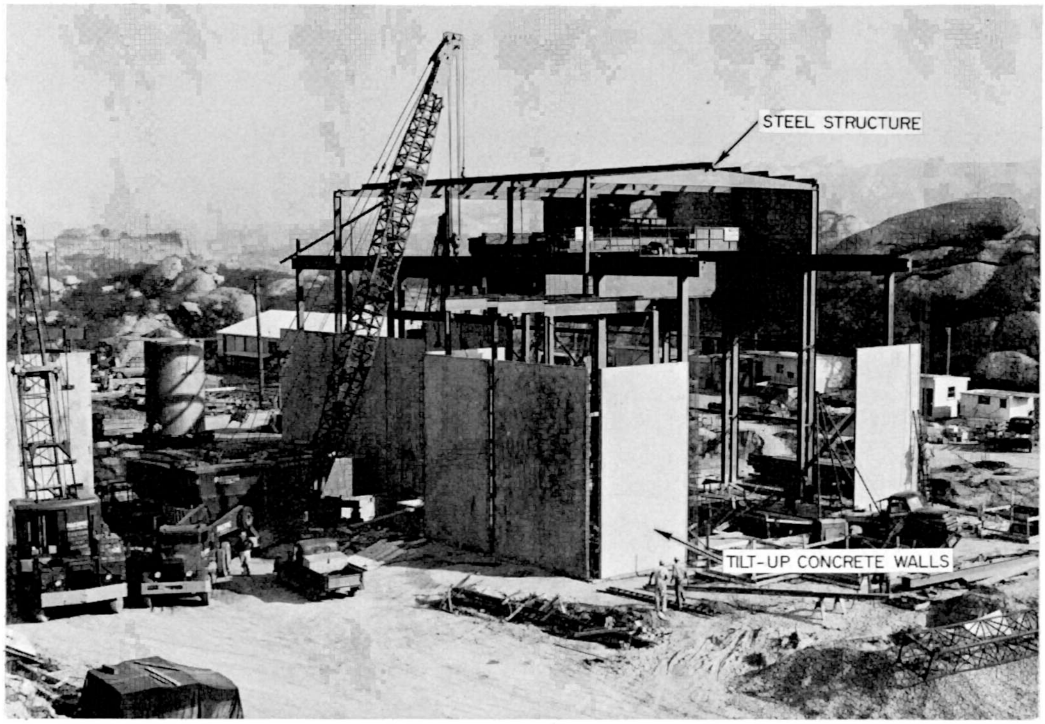


Figure 2. SRE Reactor Building During Construction

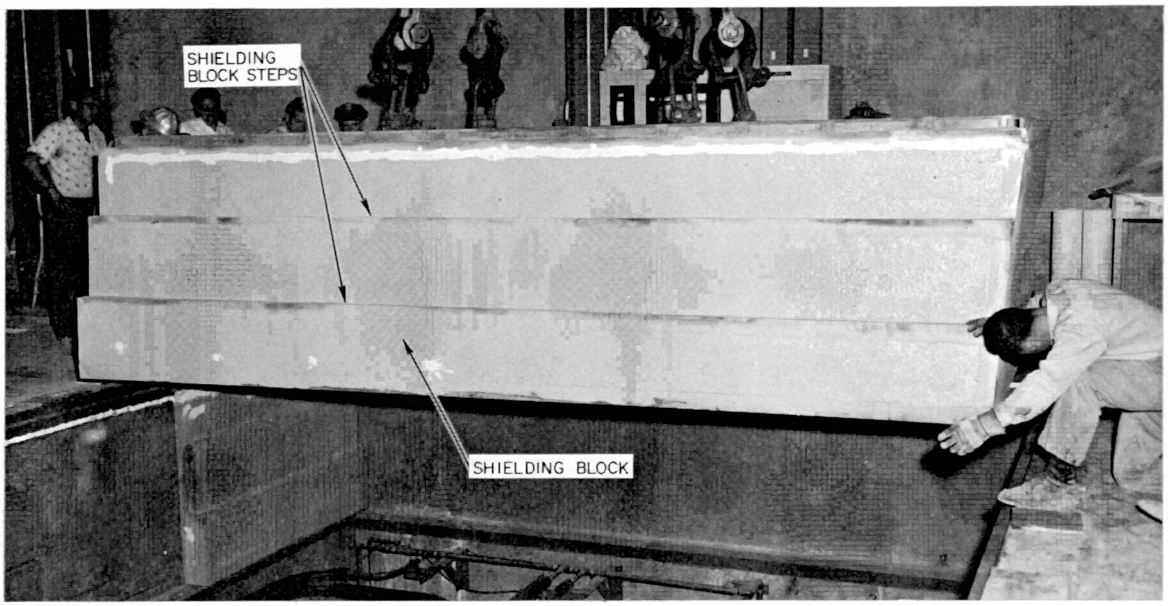


Figure 3. Primary Sodium Vault

2. Reactor Complex

The combined active core of the TR-SRE is contained in a volume six feet in diameter and six feet high. The proposed fuel test fast-section is to be located in the center of the core (see Figure 4). The reference-design, fast test section is composed of 19 hexagonal-shaped fuel elements, each containing 19 fuel rods. The reference core fuel material is a mixed plutonium-uranium oxide. Surrounding the fast section is a ring of six decoupler units containing a combination of depleted UO_2 and boron steel.

A honeycomb structure consisting of thin-wall 304-stainless-steel hexagonal-shaped tubes positions the fast-fuel elements within the core. The fast-fuel elements are axially supported by hanger rods fixed to individual shield plugs extending through the loading face shield. Surrounding the fast-decoupler sections is the SRE lattice. There are 36 uranium carbide fuel elements, 6 shim, and 4 safety rods located in the SRE section. All of these elements are suspended from the loading face shield above the reactor. All elements in the core are supported on a grid plate at the base of a 1-1/2-in. type 304 stainless steel core tank, 11 ft in diameter and 19 ft high.

Proceeding radially outward from the core tank is a 5-1/2-in. thermal shield of cast steel and a 1/4-in. steel outer tank which can contain sodium in the event of a leak in the core tank. Metal bellows (Figure 5) are installed at the top of both the reactor tank and the outer tank to provide gas seals and to allow for thermal expansion. Twelve inches of thermal block insulation, Figure 6, occupies the space between the outer tank and 1/4-in. -thick, carbon-steel, concrete liner (see Figure 7). The 3-ft thick concrete foundation provides the biological shield for the surrounding soil.

The top shield is made of high-density concrete, 6 ft thick. It consists of a fixed, stainless-steel, stepped-ring shield (Figure 8), 15 ft in diameter, with a central loading-face stepped shield 11 ft, 8 in. in diameter (Figures 9, 10), and weighing approximately 85 tons. A low-temperature melting alloy (cerrobend) cast into steel troughs at floor level is used as a gas seal between the ring shield and the surrounding foundation, and between the ring shield and loading face shield. The sodium coolant enters the reactor above the core,

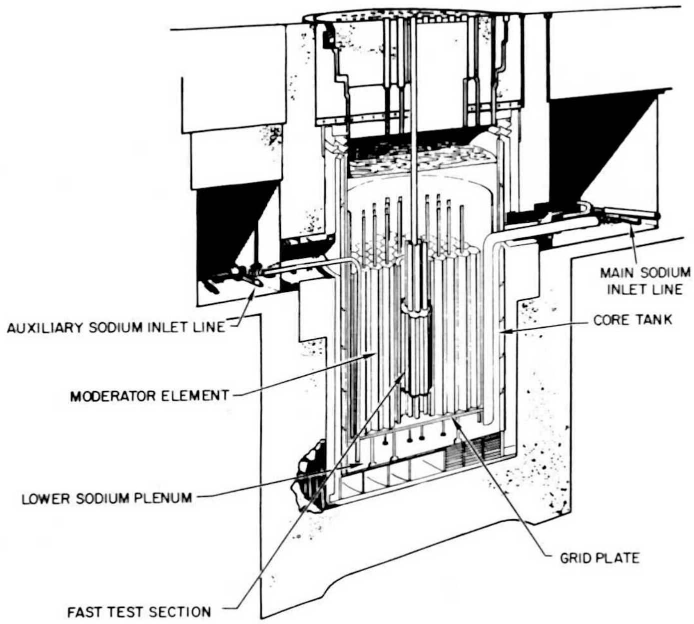


Figure 4. Cutaway View of Two-Region SRE

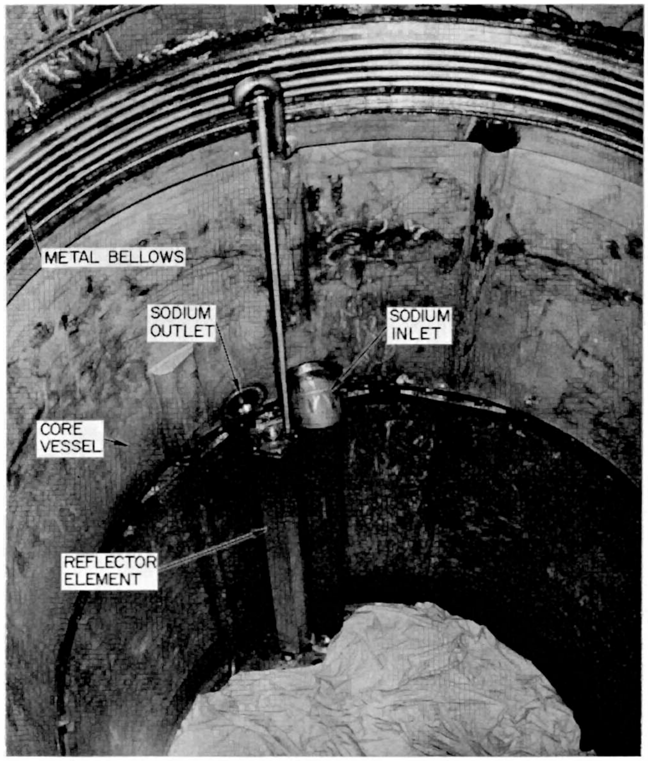


Figure 5. Inside of the SRE Core Vessel

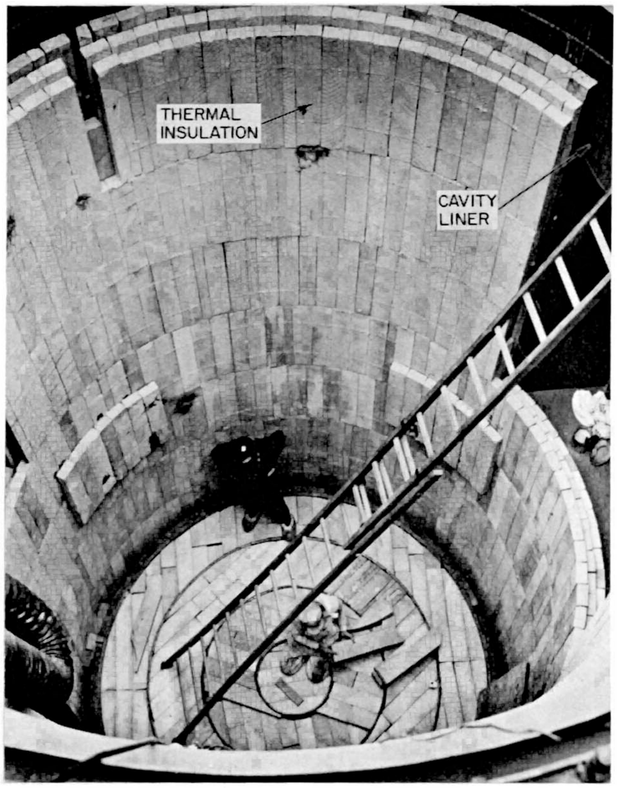


Figure 6. Thermal Insulation Inside Cavity Liner

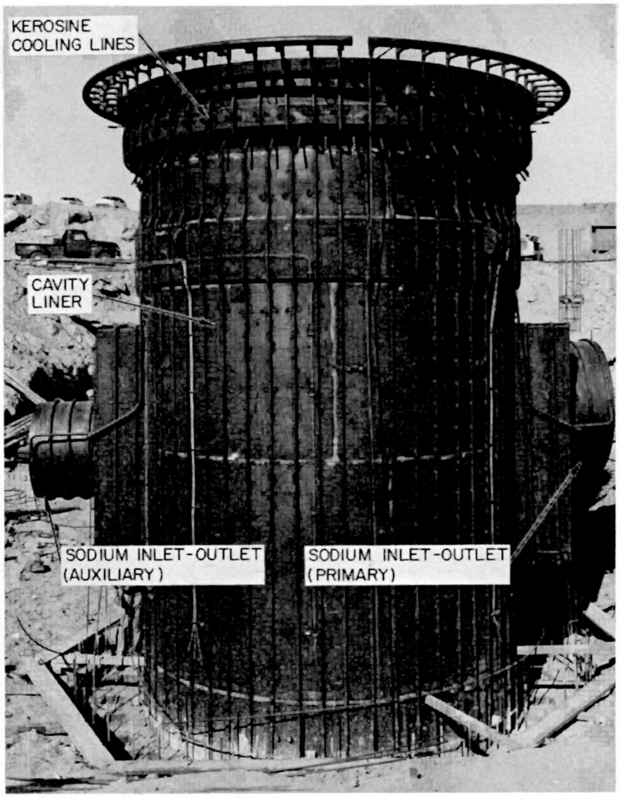


Figure 7. Cavity Liner

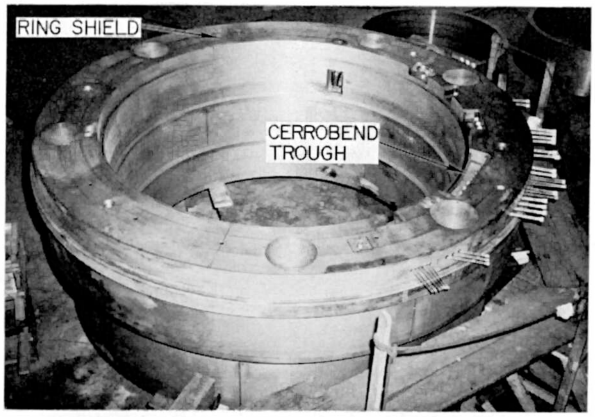


Figure 8. The Ring-Shield Section of the Top-Shield

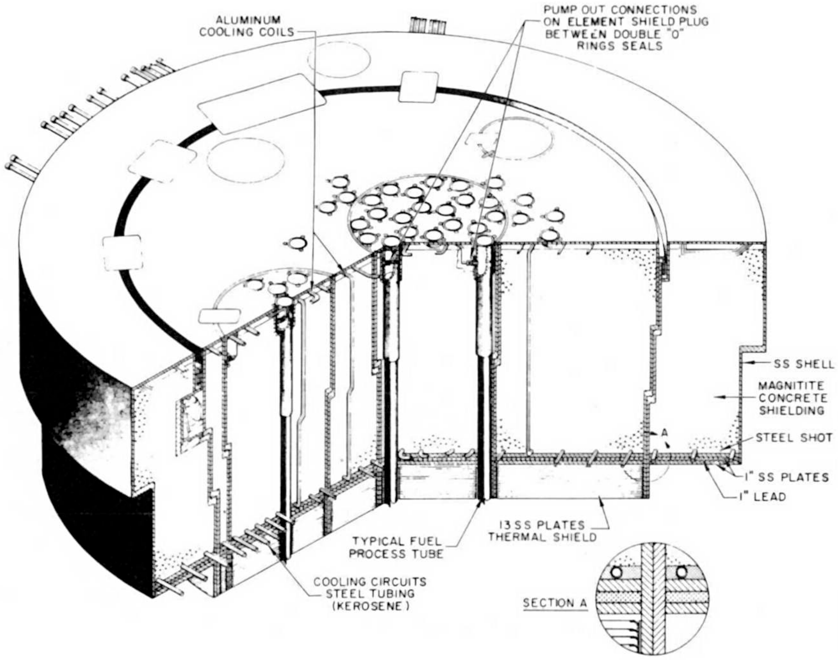


Figure 9. Design of the SRE Loading - Face Shield

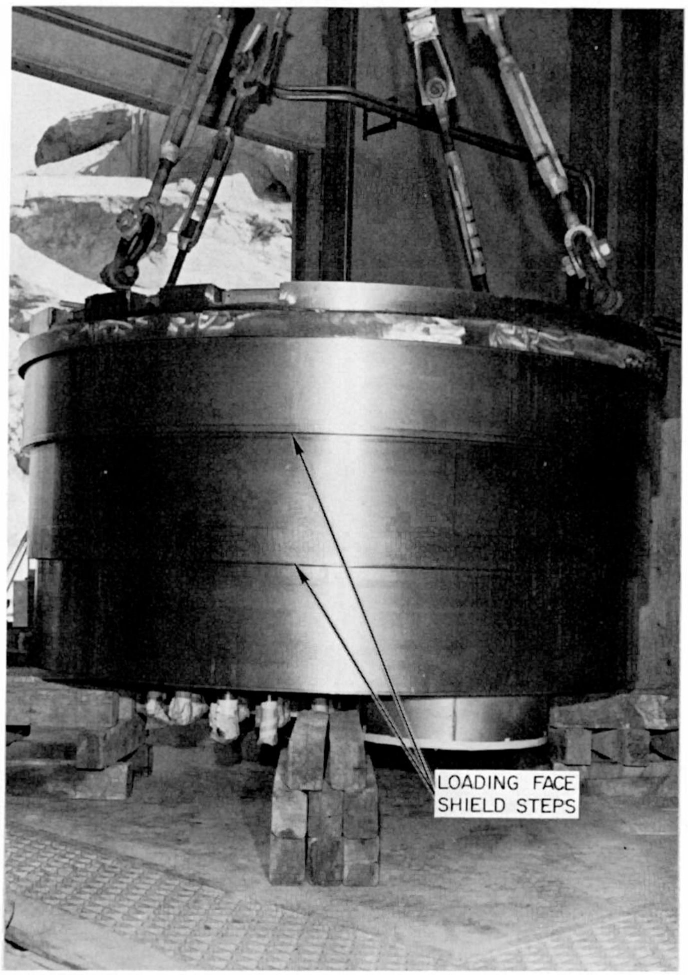


Figure 10. Loading-Face Shield

flows through a downcomer inside the tank to the lower plenums, up through the fuel channels into the 6-ft deep sodium pool, and out through the core tank outlet nozzle. A more detailed description of the system is given in References 1 and 2.

D. OPERATIONAL CHARACTERISTICS OF THE SRE CORE WITH A FFTS

SRE plant performance characteristics with the fast test section are presented in Table I. Although other core arrangements may be contemplated, this core is considered the most reactive from a safeguard analysis standpoint (highest fuel enrichment); it is, therefore, referenced in this safety study.

Operational parameters of the reference SRE core are given in Table II. These values include the uncertainty in the estimates taken in the direction to worsen the accident. The criticality values used for th TR-SRE were estimated from the following equations taken from Reference 3.

$$(1 - k_{ff})\,(1 - k_{ss}) = k_{fs}\,k_{sf} \qquad (1)$$

where

k_{ff} = The fast section multiplication,

k_{ss} = The thermal section multiplication,

k_{fs} = The coupling from fast (f) to slow(s),

and

k_{sf} = The coupling from slow(s) to fast(f).

The effective prompt neutron lifetime for steady state coupled systems is given by

$$l_{eff} = \alpha_f l_f + \alpha_s l_s \qquad (2)$$

where α's are reactivity partition factors in the fast and thermal TR-SRE sections (please see Table III). The effective delayed neutron fraction is given by

$$\beta_{eff} \approx \alpha_f \beta_f + \alpha_s \beta_s \qquad (3)$$

Given in Table III, are the estimated kinetic characteristics of the reference TR-SRE core.

TABLE I

PERFORMANCE CHARACTERISTICS TR-SRE REFERENCE PuO_2-UO_2 CORE

Reactor Power (MWT)	
Fast Fuel Section	22.7
Decoupler	4.1
Thermal Section	23.2
Total	50.0
Sodium Temperature (°F)	
Reactor Inlet	650
Reactor Outlet	1200
Fast Fuel Section	
Active Diameter (in.)	10.9
Active Length (in.)	43.0
Number of Fuel Elements	19
Fuel Slug Diameter (in.)	0.25
Number of Fuel Rods	361
Fuel Loading (kg Pu) 90% Pu	106
Max. Specific Power (kw/kg fissile)	270
Avg. Specific Power (kw/kg fissile)	214
Averaged Neutron Flux (n/cm^2-sec)	1.3×10^{15}
Max. Rod Power (kw/ft)	23.0
Average Rod Power (kw/ft)	20.9
Decoupler Section	
Diameter (in.)	23.7
Length (in.)	72.0
Number of Rods	636
Fuel Slug Diameter (in.)	0.605
Fuel Loading (kg UO_2)	1220
Borated Steel Thickness (in.)	5
Thermal Section	
No. of Driver Elements	36.0
Fuel Material	UC
Initial Enrichment	6.5
No. of Rods/Element	8
Weight of Fuel/Element (kg UC)	34.0
Slug Diameter (in.)	0.60
Active Length (ft)	6.0

TABLE II

OPERATIONAL CHARACTERISTICS OF TR SRE CORE

Enrichment	
Fast Section (% Pu^{239})	90
Thermal Section (% U^{235})	6.5
Total Mass	
Fast Section (kg Pu^{239})	106
Thermal Section (kg of U^{235})	80
Initial Core K_{eff}	
Dry	1.044
Wet	1.053
Hot	1.005
Equilibrium Xenon and Samarium Worth	3.05
Xenon Override ($)	0.55
Available for Burnup ($)	9.50
Burnup Characteristic ($) (1000 Mwd/MT) (Fast Fuel)	0.095
Number of Shim Rods	6
Number of Safety Rods	4
Worth of Rods ($)	22.4
Shim	15.0
Safety	7.4
Fast Core Neutron Flux (n/cm^2 - sec)	1.3×10^{15}
Average Gross Radial Peak-to-Average Power (Fast Section)	1.10
Average Gross Axial Peak-to-Average Power	1.20
Temperature Coefficients of Reactivity (Estimated) ($\Delta k/°F \times 10^{-5}$)	
a) Fast Section	
Fuel	<0.08
Coolant	-0.40
b) Decoupler	
Fertile	-0.30
Coolant	-0.50

TABLE II

OPERATIONAL CHARACTERISTICS OF TR SRE CORE

Temperature Coefficients of Reactivity (Estimated)	($\Delta k/°F \times 10^{-5}$)
c) Thermal Section	
Fuel	-0.70
Moderator	0.80
Fuel Coolant	0.30
Moderator Coolant	0.80
Effective Delayed Neutron Fraction	0.0036
Prompt Neutron Lifetime (μsec)	48

TABLE III

TR-SRE: KINETICS CHARACTERISTICS

k_{ff} = 0.95	l_f = 0.5 μs
k_{ss} = 0.66	l_s = 365 μs
k_{fs} = 0.05	l_{eff} = 48 μs
k_{sf} = 0.34	β_f = 0.0031
α_f = 0.87	β_s = 0.0070
α_s = 0.13	β_{eff} = 0.0036

As noted in Table III, the fast fuel test section in SRE is approximately \$14.0 subcritical (i.e. $1-k_{ff}/\beta_{eff}$ = \$14.0). In parameter studies of various fast-fuel test sections in the SRE core, subcriticality of the fast section varied from \$10 to \$15.0. Thus, the fast-fuel test section must remain coupled neutronically to the SRE core in all accident conditions for the system to remain critical; the test section cannot under any circumstances go critical alone.

1. Thermal SRE Section

The effective prompt neutron lifetime of the SRE core is approximately 500 μs. The effective prompt neutron lifetime of the thermal section of SRE with fast fuel test section installed is 365 μs. The decrease is due to increased neutron leakage.

The Dopper coefficient in the thermal SRE section possess a characteristic $T^{-1/2}$ temperature dependence and has been measured in SRE Core I, II, and III to be large ($\sim -1.0 \times 10^{-5}$ $\Delta k/°F$).

The sodium-temperature and void coefficients are positive (i. e. void $\sim$ \$10.00) and result from the removal of a poison (sodium) from the thermal section. In the SRE with fast-section, sodium-voiding from the thermal section would result in an increase of \$1.30 due to the split in reactivity (87% fast; 13% thermal).

2. Fast-Fuel Test Section

The effective prompt neutron lifetime of the TR-SRE fast section is approximately 0.5 μs. The effect of neutronically coupling this section to the SRE thermal section is to produce an effective prompt neutron lifetime of 48 μsec (more than twice as large as that for typical thermal PWR systems).

The Dopper coefficient of the fast fuel has a $T^{-1/2}$ temperature, is positive, and small (i. e. $\sim 0.08 \times 10^{-5}$ $\Delta k/°F$).

The computed effect for voiding sodium from the fast fuel test section in SRE is -\$4.95, as is typical of very small fast systems. This provides a highly desirable feedback which will provide a delay or minimizing effect in any reactor transient.

E. REACTOR TRANSIENTS

1. Kinetic Response

The dynamic behavior of the system was investigated by using analog computer simulation of the generalized two-region reactor kinetics equations developed by Avery.[3] The basic dynamic model included:

a) Fast and thermal region neutron kinetics (with coupling in each direction between regions),

b) Six delayed neutron precursor groups,

c) Thermal kinetics for fuel elements in each region, and

d) Feedback reactivity to the neutron kinetics equation from each region.

The behavior of the SRE with a fast-test section, from a reactor control standpoint, is nearly identical to that of the SRE alone, and it has many of the properties of all thermal systems, requiring no additional safety features. This is typical of all coupled systems whose multiplication of fast neutrons is small compared to the total steady-state reactivity. Reactor control is, therefore, by conventional methods and utilizes standard hardware.

The controlling kinetic effect is similar to the action of delayed neutrons as found in conventional dynamic studies. This effect permits control of a reactor core on the relatively slow precursor decay. The delaying effect in a coupled reactor results from the long lifetime of thermal neutrons due to the precursor decay constant.

In the analysis of the two-region system, the dynamic behavior relative to that of the SRE was studied. Credible accidents studied in this analysis are listed in Table IV. These classic accidents are studied with the assumption that the plant protective system is operating with one level of protection failed.

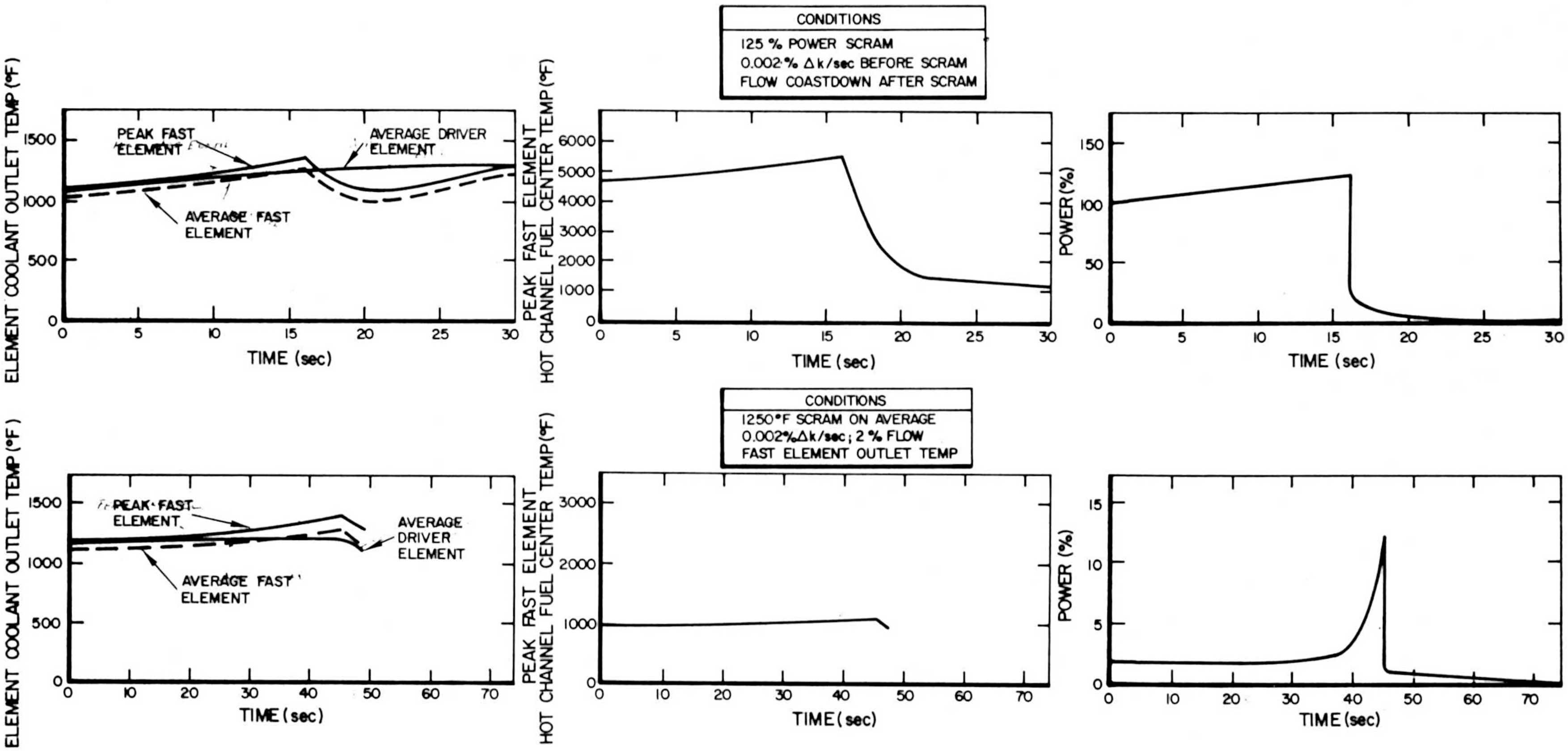
TABLE IV

NUCLEAR INCIDENTS WITH SCRAM PROTECTION

Shim rod withdrawal at low power
Shim rod withdrawal at high power
Loss of coolant flow at full power
Cold inlet coolant transient

The summary results presented in Table V indicate a considerable similarity in the transient temperature response of the two systems and demonstrate that the overall reactor control is by the existing thermal section. The fast fuel temperature response is indicative of the higher power density of this particular fast fuel test.

These results permit the use of the existing SRE control and protective systems. The following is a detailed discussion of these results.



Figures 11 & 12. Shimrod Withdrawal at Full- and Low-Power

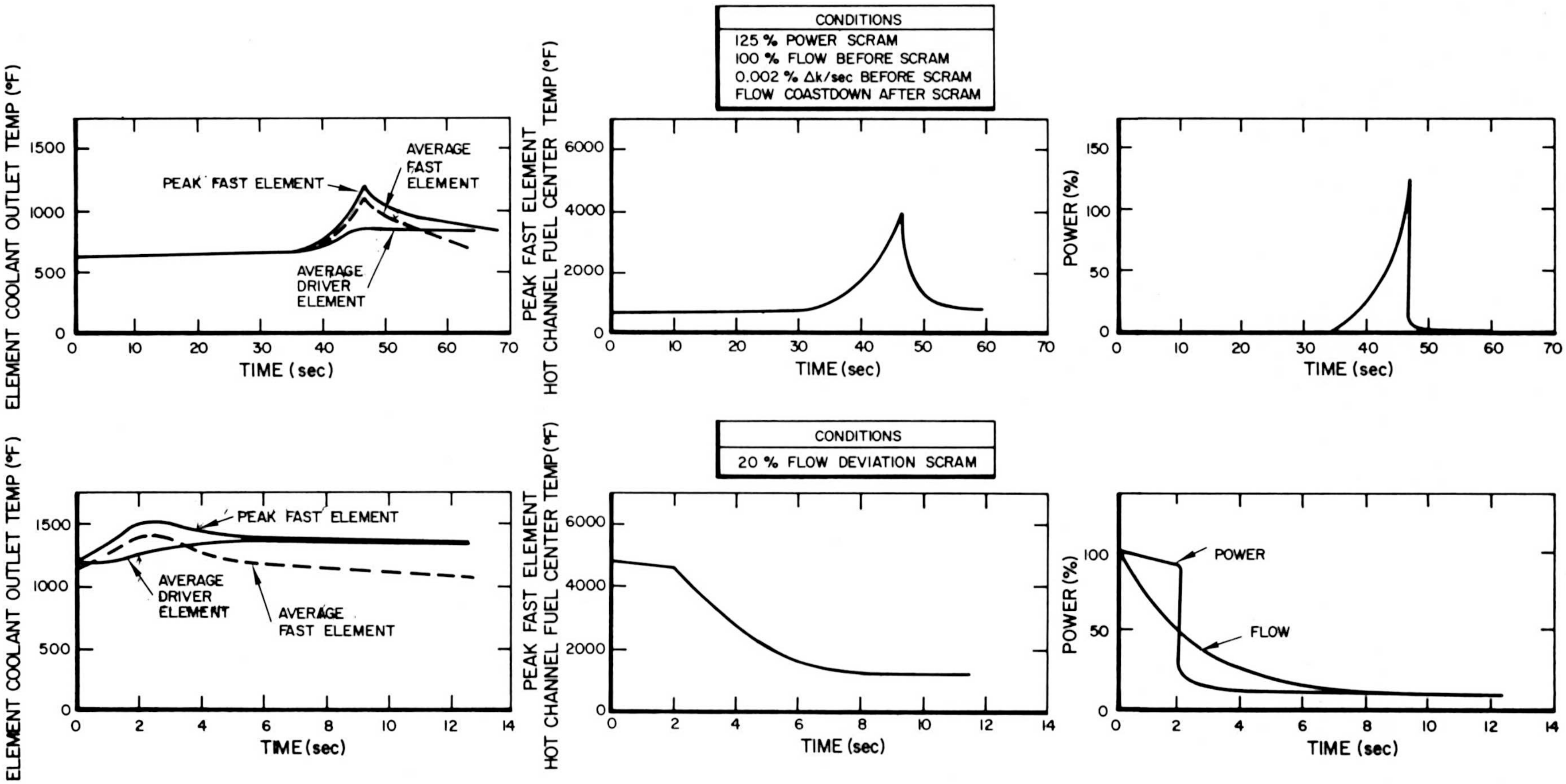
TABLE V

CREDIBLE ACCIDENT SUMMARY TEMPERATURES (°F)

Accident	SRE		TR-SRE			
	Thermal UC		Fast Oxide		Thermal UC	
	Max Na Exit	Max Clad	Max Na Exit	Max Clad	Max Na Exit	Max Clad
One Shim Rod Withdrawal 100% Power	1245	1333	1350	1550	1280	1360
One Shim Rod Withdrawal 2% Power; 2% Flow	1271	1208	1500	1360	1250	1175
One Shim Rod Withdrawal 2% Power; 100% Flow	914	931	1100	1120	900	920
20% Flow Deviation 100% Power	1287	1340	1480	1680	1300	1360

2. Shim Rod Withdrawal at Full Power

Power level control is normally maintained with a single shim rod in a high neutron importance region. A rod drive or control system malfunction can be postulated as resulting in an uncontrolled shim rod drive activation. This malfunction would produce a corresponding range of reactivity insertion rates. In this study, the peak, shim-rod, reactivity insertion rate of $0.55/sec was used. The results of this for a reactor initially at full power and flow is shown in Figure 11. The transient was terminated by a reactor scram at 125% of full power. The results are very similar to those for the single shim rod withdrawal transient in the SRE-PEP. As shown in Figure 11, the fuel element coolant outlet temperatures in the fast and thermal regions remain below 1350°F (below the boiling point of sodium). The maximum cladding temperature is approximately 200°F above the coolant exit temperature and remains well below its melting temperature. Peak fuel centerline temperatures are above the fuel melting point throughout the excursion. The additional melting that occurs



Figures 13 & 14. Shimrod Withdrawal at Low Power and Loss of Coolant Flow at Full Power during the power excursion does not progress to the fuel clad interface. A high coolant outlet temperature scram in the driver element will initiate scram action approximately four seconds before the 125% power level scram is reached. A less severe excursion would occur with normal protective system action. In this case, high power scram serves as backup for coolant outlet temperature protection.

3. Shim Rod Withdrawal at Low Power

Two transients of this type were considered assuming the reactor to be operating initially at 2% power (1% due to fissioning of fuel and 1% due to decay of fission products). In the first case, the coolant flow is assumed to be 2% of full flow, and in the second case, at full flow. Figure 12 presents the results for the first case protected by a 1250°F scram on the basis of coolant outlet temperature from the average power element in the fast section. Fuel and coolant exit temperatures in the three elements shown remain below limiting temperatures and no fuel damage occurs. Uncontrolled shim rod withdrawal at 2% power and full flow is shown in Figure 13 with 125% power level scram protection. Initial fuel and coolant temperatures for this incident result in a power scram protection preceding outlet temperature protection (just the reverse situation from the low flow case). In this case, power level protection would prevent fuel element damage.

4. Loss of Coolant Flow at Full Power

With the reactor initially at full power and full flow, a flow coastdown due to loss of pump power is assumed. The transient shown in Figure 14 occurs when the reactor is protected by a scram at 20% deviation from flow setpoint. This transient for the average driver element is very similar to the loss of flow transient shown in the SRE-PEP report. No damage to the reactor results from this transient.

5. Cold-Inlet Coolant Transient

This transient was included in the study to show the effect of a postulated but highly improbable coolant inlet temperature incident. The excursion shown on Figure 15 is based on a step decrease of 220°F in the coolant temperature at

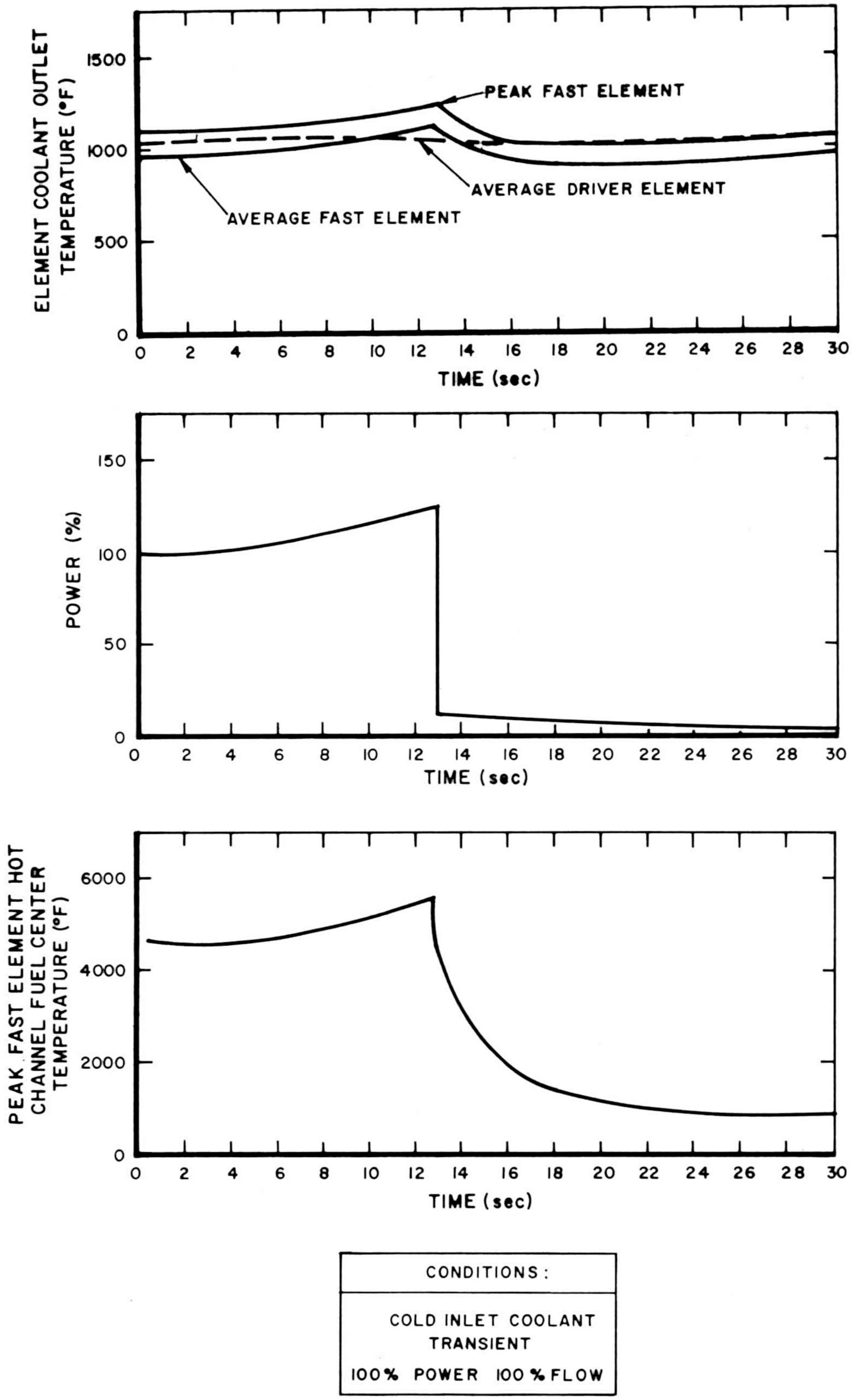


Figure 15. Cold Inlet-Coolant Transient

the reactor inlet when operating at full flow and power. Little temperature reduction occurs in the outlet coolant in the initial stages of the excursion. Resulting reactivity feedback effects increase reactor power with an eventual high power or high outlet temperature scram. Protection by either parameter is shown to occur at approximately identical times with no excessive fuel or coolant temperatures.

The above safety analysis shows that the kinetic response of the TR-SRE is similar to that for the SRE. Installation of the fast fuel test section in the SRE will not compromise the demonstrated safety of the power plant.

Further discussions on the SRE protective system are presented in Appendix A.

F. RADIOLOGICAL ANALYSIS

A study of sodium accidents was made with the following assumed conditions.

1) Time of entry to a vault or gallery varying from 0 to 10 days after shutdown from full power

2) A subsequent sodium leak release of 12,400 lb of sodium which covers the 475 ft^2 vault floor area

3) A fire results and burns for 20 hr consuming all of the sodium

Studies covered the effect of a spill-and-fire in the primary fill tank vault which has a floor area of 150 ft^2. For this area, about 3900 lb of sodium could burn in 20 hours. Of the fires studied, this has the greater probability and worst consequence since the vault is external to the building. In both fires, ground release of radioactive material is assumed. Important parameters and assumptions for these studies are listed in Table VI.

The total (direct-cloud and inhalation) whole-body irradiation dose for 2 hr and for 30 days after the accident is plotted in Figure 16 as a function of cooling time for both major areas. The Pu which was postulated to be in the sodium does not significantly affect the radiological dose at the site boundary. Siting requirements can be noted from the graph of minimum decay time required before shield plugs may be lifted. On the basis of the 30-day dose, removal of the block from the primary fill tank vault must be delayed at least four days

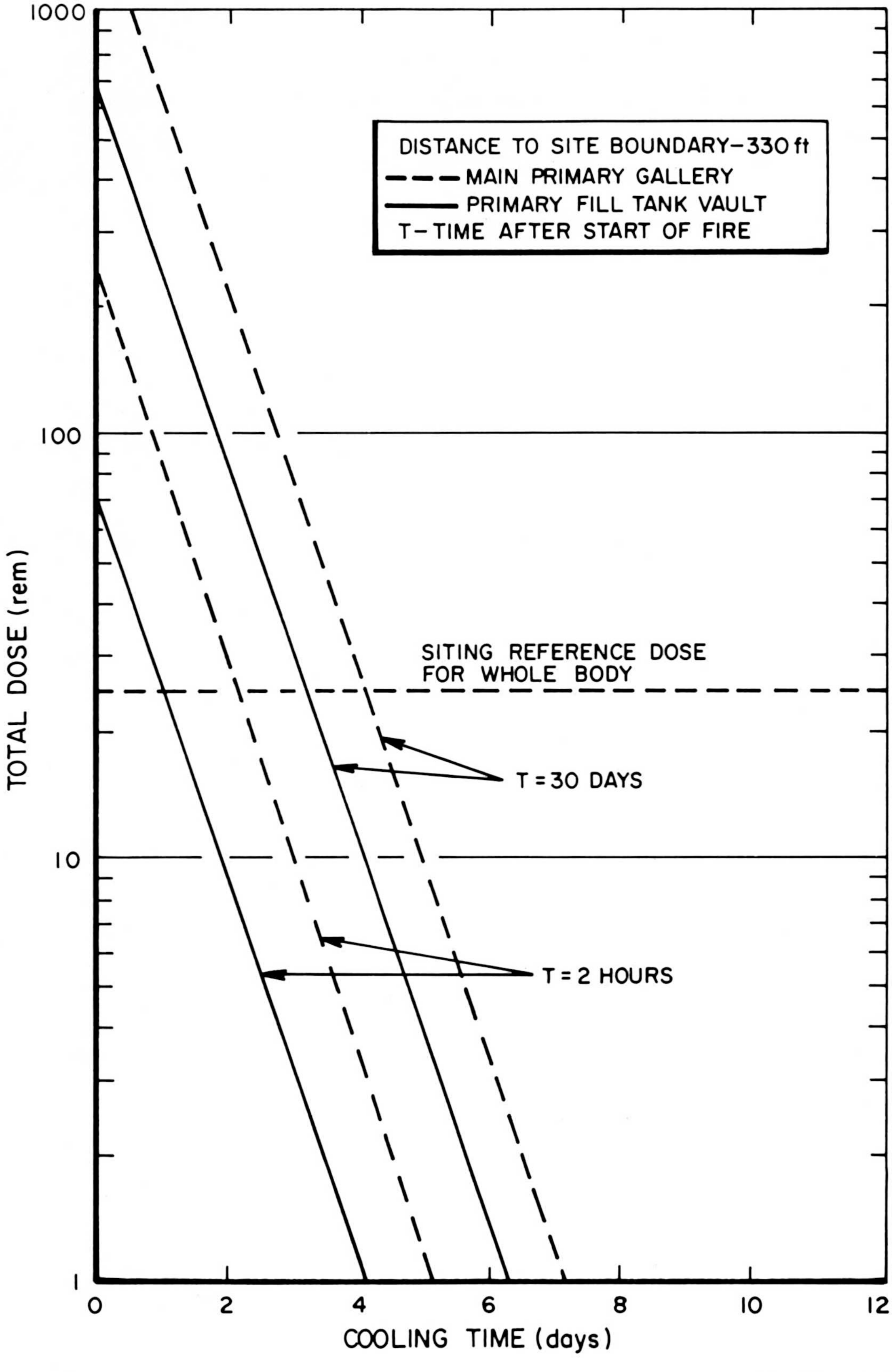


Figure 16. Total Irradiation Dose from Sodium vs Cooling Time

TABLE VI

SODIUM FIRE ACCIDENT PARAMETERS

Reactor Operating Time at Full Power (days)	300
Core Power (Mwt)	50
Cooling Time, Variable (days)	0 to 10
Fraction of Fission Products in Coolant	10^{-4}
Pu in Coolant*	0.1 ppm
Coolant Burning Time (hr)	20
Coolant Burning Rate† ($lb/hr\text{-}ft^2$)	1.3
Specific Activity of Na^{24} (c/gm)	0.45
Total Sodium Inventory, approx. (lb)	53,000
Release Fractions:	

Radioactive Material	Released on Burning (%)
Halogens	50
Noble Gases	100
Na^{24}	50
PuO_2	5

*Arbitrarily taken as 1/2 the measured value of the U fraction measured in the sodium coolant following the SRE fuel element damage episode, NAA-SR-6999, page IA10.

†Presently considered the base value; obtained from large sodium-burning experiments at the AI Field Laboratory.

after shutdown from equilibrium full power activity, and at least five days must be allowed for the main primary gallery. SOP access procedure, however, requires a much longer time before anyone could enter the vault safely due to the direct dose restrictions.

Siting criteria is thus seen to be less stringent than the personnel access criteria and hence is not the regulating factor for these accidents.

1. Plutonium Fuel

The total plutonium inventory of this referenced reactor is 106 kg in the form of mixed PuO_2-UO_2. For the unirradiated reactor, all plutonium fuel will be in the fast test-section. Power generation will not decrease the total inventory. About as much plutonium is bred in the decoupler and thermal region as is burned in the original fast fuel.

All TR-SRE plutonium containing fuel will be enclosed and sealed in stainless steel rods. The handling of fuel is limited to core loading, storage, washing, surface examination, packaging, and shipping. Because of the additional toxicity factor involved with Pu and recognition of the possible need to handle an occasional faulted fuel element, fuel handling system modifications are proposed.

The additional safety features proposed for TR-SRE system fuel handlings are:

a) Provision of double containment for the FHM,

b) Addition of a Xe^{133} detector to the fuel handling machine, and

c) Provision of environmental control for the hot-cell.

2. Fuel Handling

Fast-fuel removal will be performed after fuel radiation has decayed. The required decay period for the fast fuel is four weeks. The referenced fast fuel will remain in the core at least two years. Approximately twice a year, the reactor will be shut down to change the thermal driver fuel.

The fuel handling machine will be modified to provide a double seal to improve environmental control. The gas space between the 40-in. gate valve and the indexing ring will be the secondary barrier. There will be provisions for purging this space before and after each fuel change. Fuel handling procedures for the thermal driver elements will be the same as those for PEP core loading. Supplementary environmental control will be covered by operating procedures.

3. Fuel Storage, Washing, and Hot-Cell Operation

Fuel storage cell thimbles for all fuel, are 4-in. Schedule-40 carbon steel pipe with a wall thickness of about 1/4 in. set in concrete on 12-in. centers. Although designed for low enrichment SRE fuel, the original factor of safety is sufficiently large to allow safe storage of all TR-SRE fast fuel even under the most adverse conditions.

Storage cell area was analyzed for the reactivity of fresh fuel stored in the cells. It was assumed that at least 2 core loadings of fast fuel are stored. The concrete surrounding the cells was assumed to be saturated with water from ground water seepage. The analysis indicates that there is no possibility of a criticality accident.

The cells were visually inspected for moisture and purged with helium before insertion of fuel elements. Therefore, no possibility for a Na-H_2O reaction in the storage cells exists.

Fuel cleaning will be accomplished in the same way as with present SRE fuel. Only sound elements with no Xe^{133} indications will be cleaned.

The SRE hot-cell improvements will include environmental control by use of an inert atmosphere. Secondary environmental control will be provided in the hot-cell for the encapsulation of irradiated plutonium bearing fuels. Hot-cell operation on fast fuel will be limited to removal and preliminary inspection of the active section.

II. NONCREDIBLE ACCIDENTS

A. INTRODUCTION

The following noncredible accidents cannot occur but have been postulated and studied for the purposes of:

1) Establishing the margin of safety inherent in the SRE plant,

2) Providing a design basis for adding to the building containment, and

3) Establishing accident limits for which containment is feasible.

For these purposes, only the core meltdown accidents were studied because they lead to the more difficult containment problems. These studies assumed that:

1) Unknown reactivity insertion takes place,

2) The total protective system fails,

3) The core collapses and reassembles resulting in a more reactive geometry which can detonate due to rate of reassembly,

4) The core sodium is expelled from the fast section,

5) Fast fuel and structure material (all or partially) is isothermally at the boiling point of the fuel, and

6) Further energy additions will explosively vaporize fuel and steel.

These conditions ultimately lead to a vaporization of the fuel, disassembly of the core, and a reduction of the reactivity which terminates the accident. The effects of the associated shock and blast wave on the system were computed. Calculations were then made of the breach of containment, the fission fragment and Pu escape, and the associated 2-hr dose at the site boundaries. No credit was taken for:

1) Sodium voiding from fast section which decreases the reactor core reactivity by ~ \$5.00,

2) The dispersal of fuel mixture by internal pressures reducing reactivity (i. e., fuel and sodium pressures), and

3) Oscillatory behavior of fuel and sodium which disassembles the core with little energy release.

Any one of these effects would, in any practical case, limit the accident before detonation.

The modified Bethe-Tait (M-B-T) approach, as proposed by Jankus(4), was used in these studies. The studies show that a range of energy release equivalent to 3.5 to 22 lbs of TNT is possible. A summary of the results is presented in the appendix.

In view of the uncertainties in the techniques used in evaluating the nuclear explosive energy releases, a parametric study was made using varying amounts of TNT energy release (i. e., 0 to 100 lb of TNT equivalent energy release). Explosive pressures and associated damage were used in the estimates of the dispersal of gaseous core products. Five alternate containment features and associated leakage rates were used in computing the 2-hr dose at the site boundaries. The five designs are discussed in the following section.

B. VARIOUS CONTAINMENT DESIGNS

The following specific design arrangements were evaluated for their ability to contain various releases of radioactivity. Five designs of containment were conceived and associated leakage rates were calculated.

Design 1

The existing SRE building with a 4000%/day leakage rate is considered. A membrane, plastic or sheet metal, is assumed to be over the reactor loading face and leakage from this enclosure over the reactor is taken to be ~5%/day.

Design 2

The first additional containment considered is sealing the existing building by caulking. The estimated leakage rate would be 100%/day.

Design 3

In this design, we assume that the reactor core is covered with a pressure dome having a leakage rate of 1%/day. The building leakage rate is assumed to remain at 4000%/day.

Design 4

In this containment arrangement, the pressure dome having the 1%/day leakage rate in Case 3 is combined with a sealed building having the 100%/day leakage rate of Case 2 and atmospheric control is added to filter the released air from the building.

Design 5

Finally, a containment arrangement is considered that includes a sealed steel liner inside the present building (assumed leakage rate of 1%/day) and atmospheric control within the pressure dome (assumed leakage rate of 1%/day) to minimize the site boundary dose.

These designs are summarized in Table VII.

TABLE VII

CONTAINMENT DESIGNS & LEAKAGE

Reactor Safety Designs	Leakage Rate (%/day)	
	Building	Reactor
Existing SRE building	4000	5
Caulking of existing building	100	5
Pressure dome in existing SRE building	4000	1
Pressure dome and caulked building atmospheric control	100	1
Pressure dome and steel liner inside building atmospheric control	1	1

C. SUMMARY OF EXPLOSIVE ANALYSIS

The resulting shock wave from an explosive energy release would deform the present reactor vessel (please see Appendix B). The maximum radial deformation computed is 3.45 in. at 50 lb of TNT energy. At 100 lb of TNT energy, the vessel would be deformed approximately 6.89 in. It is concluded that up to energy releases equivalent to that of 100 lb of TNT, there would be no rupturing of the existing vessel. The bellows section at the top of the reactor vessel would be greatly deformed and local cracks might occur at the weld connections. The reactor vessel is not the weak point in the present SRE reactor complex.

In the explosion model, the initial shock wave is followed by a rapid pressure rise or blast due to the expanding gaseous core products. This blast effect is that of a rapidly increasing pressure front that would be applied to the inside of the vessel and the top loading face shield (LFS). An energy release equivalent to that of 50 lb of TNT would not lift the LFS. However, energy releases in excess of this would shear the cerrobend seal, lift the LFS, and allow the gases to vent around the plug into the surrounding outside atmosphere. At an energy equivalent to 100 lb of TNT, the plug jump would be greater, allowing more gases to vent. The above accident cases were used as the basis for evaluation of the effect of various designs of containment previously listed.

<u>Case 1, Containment Design 1, F $<$50 lbs of TNT</u>

The explosive gas pressure inside the vessel would increase from 20 psi for a 10 lb of TNT energy equivalent to 80 psi at a 50 lb of TNT energy equivalent. At 50 lb of TNT energy, the blast wave pressure would not lift the LFS or damage the vessel. Therefore, dispersal of vaporized core products would be by leakage around existing seals in the LFS. Assuming that the proposed plastic or sheet metal membrane limits the leakage rate from the LFS to $\sim$ 5%, the 2-hr whole body dose at the site boundary would be $<$30 rem. A higher leakage rate would result in an increased whole body dose. At leakage rates approaching 50%/day, the 2-hr whole body dose at the site boundary would be $<$ 300 rem. It is concluded that explosive energy releases would require additional containment, if the LFS leakage rate is greater than 5%/day.

Case 2, Containment Design 3, F > 50 lbs but < 100 lbs of TNT

By placing a pressure dome over the reactor with a design leakage rate of 1%/day, we can greatly improve the control of released core gaseous products to the site boundary. As previously indicated, the blast wave from energies between 50 and 100 lbs of TNT would lift the plug and vent the gases into the dome. Since leakage out of the dome is limited to 1%/day, the 2-hr whole body dose at the site boundary would not exceed 49 rem, even with a building leakage rate of 4000%/day.

Case 3, Containment Design 4, F > 50 lbs but < 100 lbs of TNT

Similarly, if Case 2 is reconsidered with a caulked-and-sealed building having a leakage rate of 100%/day, the 2-hr whole body dose would be reduced to ~ 0.2 rem.

Case 4, Containment Design 5, F > 100 lbs, but < 200 lbs of TNT

Explosive equivalent energy releases between 100 and 200 lb of TNT might increase leakage from the reactor dome to the building. Assuming, first, that the dome leakage remains at 1%/day and that the building leakage is reduced to 1%/day, the resulting 2-hr whole body site dose is less than 0.1 rem. Assuming that the dome leakage rate is increased to 100%/day with that of the building at 1%/day, the resulting 2-hr site whole body dose would not exceed 10 rem.

In the 100 lb of TNT energy, or greater case, the explosive gas pressures would approach 160 psi and give the reactor vessel a permanent radial set of 6.84 in. Additional cracking would take place at weld connections, and the reactor bellows would deform extensively; however, the primary vessel would not be violated at 100 psi. At greater explosive energy releases, primary vessel tearing would be possible unless additional steps were taken to protect it.

III. ADDITIONAL CONTAINMENT

A. GENERAL

By use of the maximum potential damage data for the SRE core, conceptual engineering design studies scoped the additional reactor and building containment.

The present SRE building was inspected to evaluate its leakage containment capabilities. Because of the many utilities penetrating the building shell, because of the types of doors in the building, and because of the many cracks in the concrete tilt-up wall panels, the current leakage rate of the building is quite large and is estimated to be greater than 4,000% per day at 1/4 inch water pressure.

The methods of decreasing the building leakage rate and of obtaining pressure containment are defined later. In general, according to the contemplated design, pressure containment is obtained by providing an all-welded steel dome over the reactor per se. It is designed to withstand an internal pressure of 20 psig while maintaining a leak rate of less than 1% per day. Within the dome are steel beams and a 1-1/4-inch thick steel plate provided to stop missiles and secure the loading face shield plug in case of a pressure buildup inside the reactor core.

Two methods of decreasing the building leakage rate are defined. The first decreases the leakage rate to less than 1% per day at 1/4 inch water pressure. This is accomplished by lining the walls and ceiling of the existing building with all-welded steel plates and reworking all of the building penetrations. The second method decreases the leakage rate to less than 100% per day at 1/4 inch water pressure. This is accomplished by caulking all seams and cracks in the existing walls, sealing the building penetrations, and painting the walls and ceiling with a special coating.

B. REACTOR TOP CONTAINMENT DOME

The reactor top containment dome covers the top of the reactor and attaches to the concrete structure around the reactor at the floor elevation, please see figures 17 and 18 for dome details.

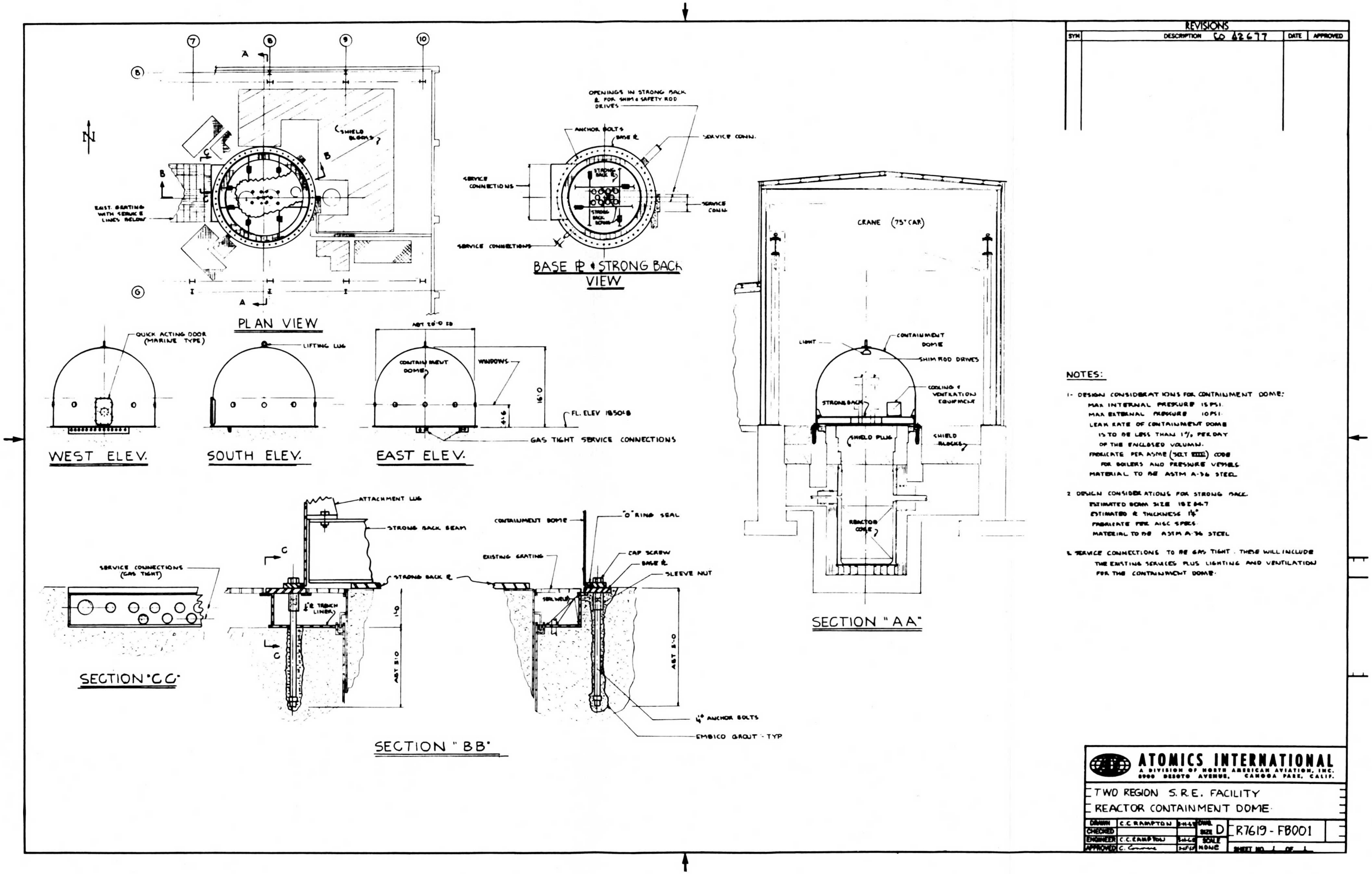
It is estimated that the dome will be fabricated as an all-welded leak-tight structure from 1/2-inch thick ASTM A36 steel plates. The concrete around the reactor shield plug will be modified to allow the placing of a base plate and anchor bolts which will be flush with the floor surface. A gas-tight, steel, pit liner will be installed between the shield plug cerrobend seal and the new base plate. This will require the removal and replacement of many utilities around the reactor top. An O-ring will seal the joint between the dome and the new base plate. The leak rate of this system will be less than 1% of the dome volume per day at an internal pressure of 20 psi. Should the internal pressure exceed 20 psi, this system would not be practical, and another approach of anchoring the dome would have to be found.

A strong-back structure will be placed over the reactor shield plug to make the shield plug and parts thereof act as one large mass. Holes will be provided in the strong-back structure to allow the placing of the shim and safety rod drives.

C. STEEL PLATE LINER

The first design of building containment was that of installing a steel liner plate on the high bay roof and walls and of adding a steel plate wall between the high and low bays. To complete the isolation of the high bay, quick-acting marine type doors for personnel entry and a sealed sliding steel equipment access door must be installed. The high bay of the SRE can be made to hold 1/4 inch water pressure with less than 1% leakage per day, please see figures 19 and 20.

Design engineering calculations for the pressure containment are given in the appendix.


REVISIONS
SYM
DESCRIPTION
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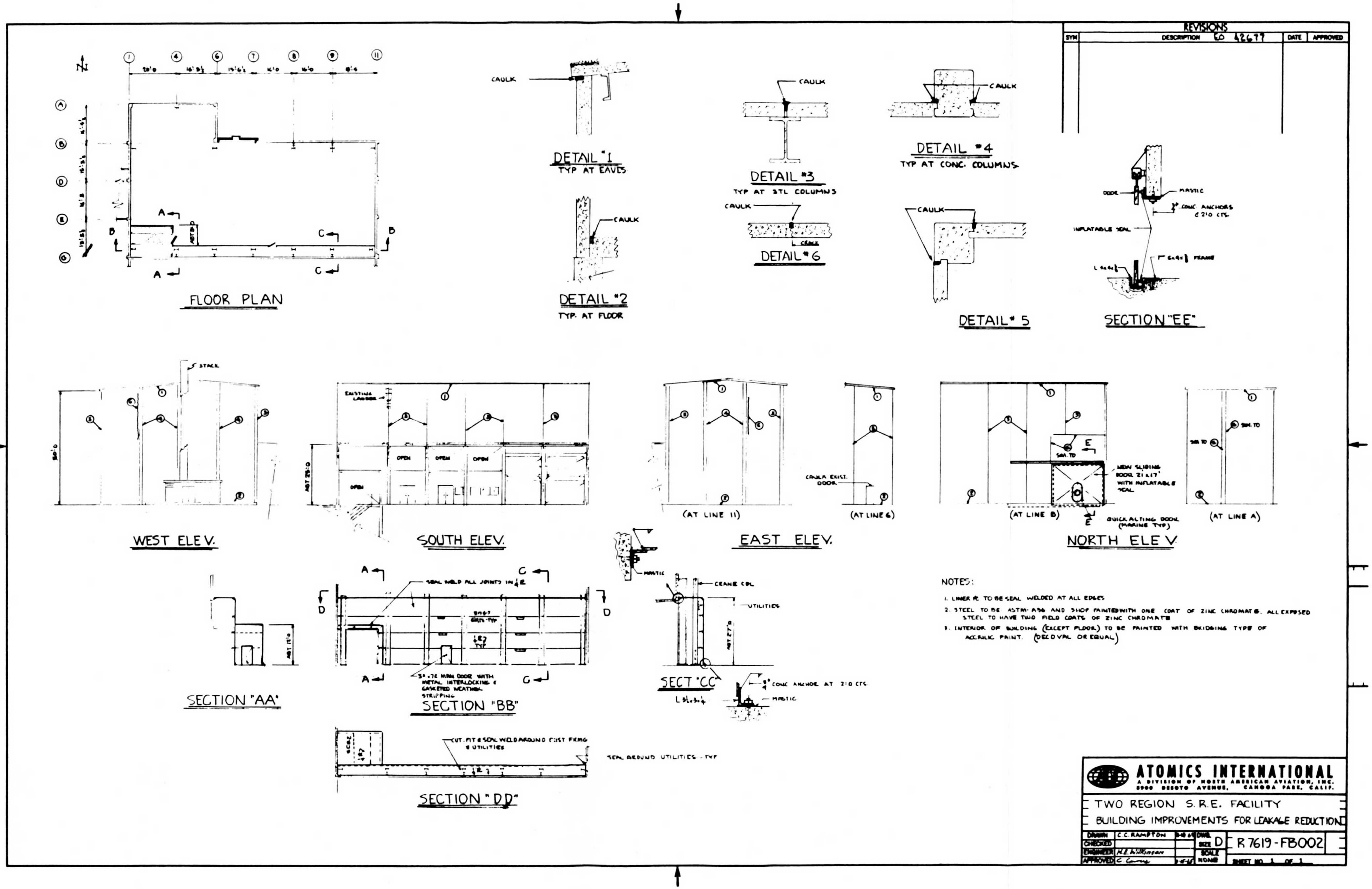
APPROVED
PLAN VIEW
N
EXIST. GRATING WITH SERVICE LINES BELOW
SHIELD BLOCKS
OPENINGS IN STRONG BACK & FOR SHIM & SAFETY ROD DRIVES
ANCHOR BOLTS
BASE ℞
SERVICE CONN.
SERVICE CONNECTIONS
STRONG BACK ℞
STRONG BACK BEAMS
BASE ℞ & STRONG BACK VIEW
QUICK ACTING DOOR (MARINE TYPE)
LIFTING LUG
CONTAINMENT DOME
WINDOWS
FL. ELEV 185018
GAS TIGHT SERVICE CONNECTIONS
WEST ELEV.
SOUTH ELEV.
EAST ELEV.
ATTACHMENT LUG
STRONG BACK BEAM
STRONG BACK ℞
TRENCH LINER
SERVICE CONNECTIONS (GAS TIGHT)
SECTION "CC"
CONTAINMENT DOME
"O" RING SEAL
EXISTING GRATING
CAP SCREW
BASE ℞
SLEEVE NUT
ANCHOR BOLTS
EMBICO GROUT - TYP
SECTION "BB"
CRANE (75" CAP)
LIGHT
CONTAINMENT DOME
SHIM ROD DRIVES
COOLING & VENTILATION EQUIPMENT
STRONG BACK
SHIELD PLUG
SHIELD BLOCKS
REACTOR CORE
SECTION "AA"
NOTES:
1- DESIGN CONSIDERATIONS FOR CONTAINMENT DOME:
MAX INTERNAL PRESSURE 15 PSI.
MAX EXTERNAL PRESSURE 10 PSI.
LEAK RATE OF CONTAINMENT DOME IS TO BE LESS THAN 1% PER DAY OF THE ENCLOSED VOLUMN.
FABRICATE PER ASME (SECT VIII) CODE FOR BOILERS AND PRESSURE VESSELS
MATERIAL TO BE ASTM A-36 STEEL
2 DESIGN CONSIDERATIONS FOR STRONG BACK.
ESTIMATED BEAM SIZE 18 I 54.7
ESTIMATED ℞ THICKNESS 1¼"
FABRICATE PER AISC SPECS.
MATERIAL TO BE ASTM A-36 STEEL
3 SERVICE CONNECTIONS TO BE GAS TIGHT. THESE WILL INCLUDE THE EXISTING SERVICES PLUS LIGHTING AND VENTILATION FOR THE CONTAINMENT DOME.
ATOMICS INTERNATIONAL
A DIVISION OF NORTH AMERICAN AVIATION, INC.
8900 DESOTO AVENUE, CANOGA PARK, CALIF.
TWO REGION S.R.E. FACILITY
REACTOR CONTAINMENT DOME
DRAWN C.C. RAMPTON
CHECKED
ENGINEER C.C. RAMPTON
APPROVED
DWG SIZE D
R7619 - FB001
SCALE NONE
SHEET NO. 1 OF 1


REVISIONS
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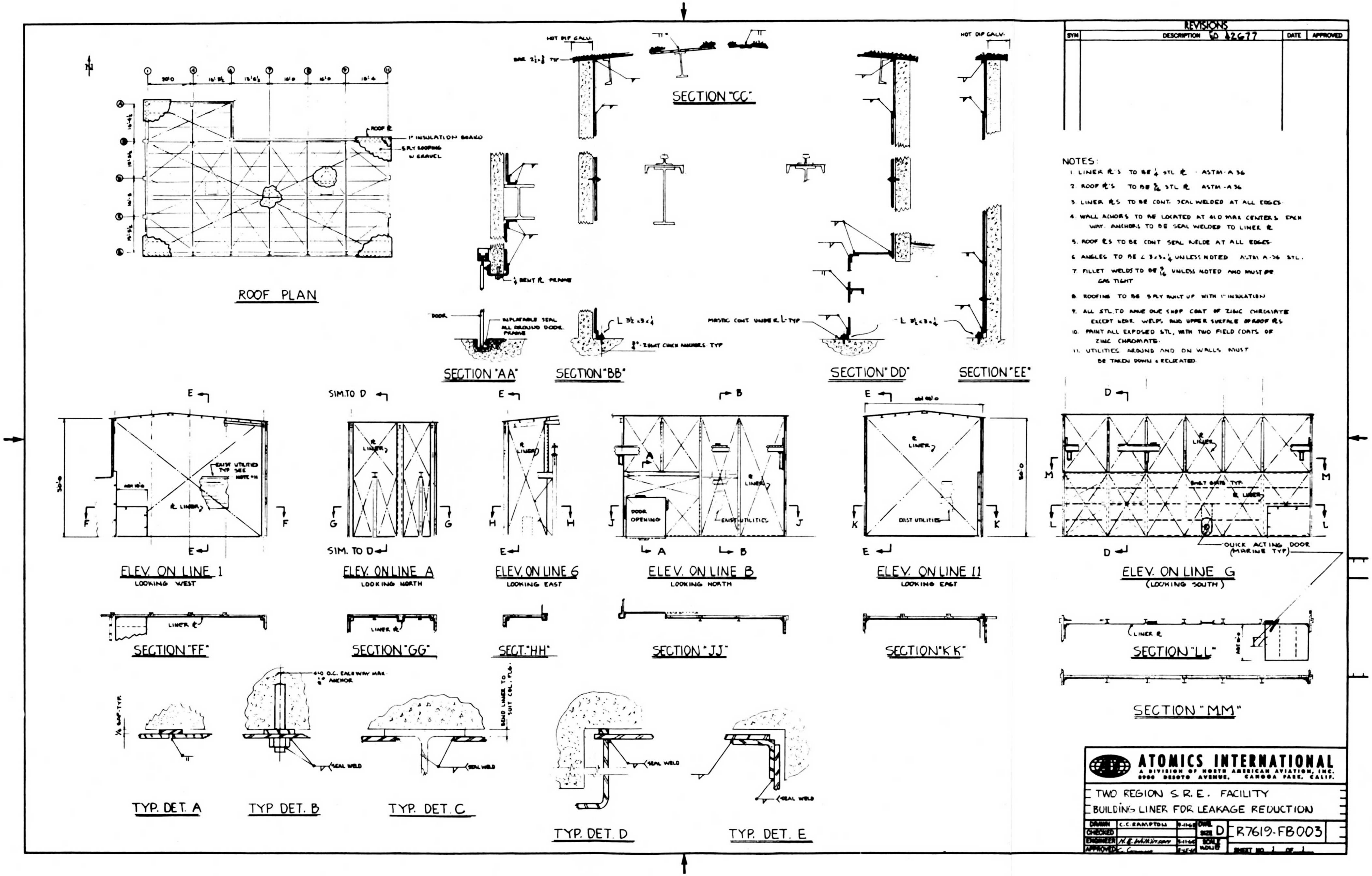
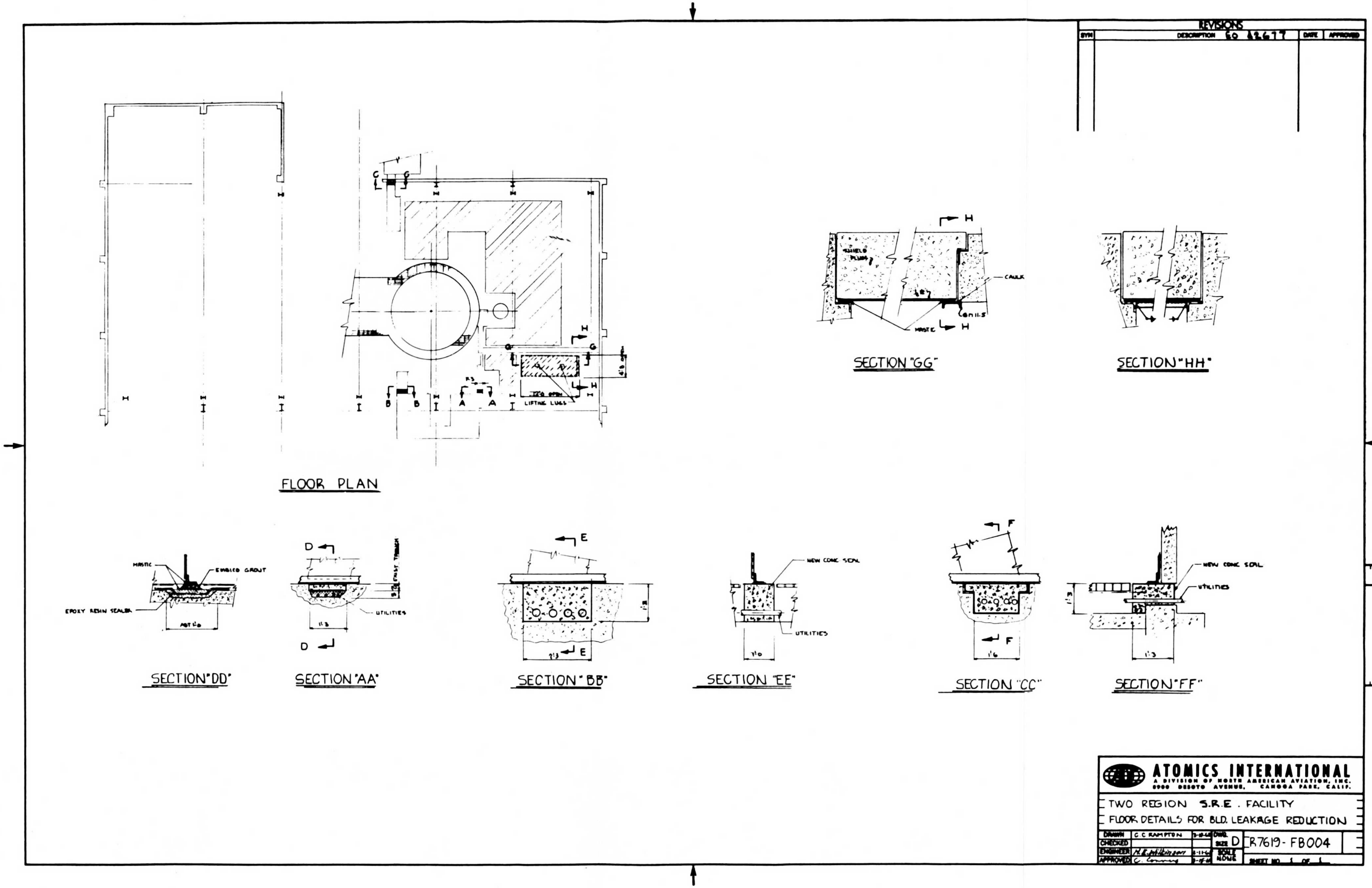
CAUK
DETAIL #1
TYP AT EAVES
DETAIL #2
TYP. AT FLOOR
DETAIL #3
TYP AT STL COLUMNS
DETAIL #4
TYP AT CONC. COLUMNS
DETAIL #5
DETAIL #6
SECTION "EE"
FLOOR PLAN
WEST ELEV.
SOUTH ELEV.
EAST ELEV.
NORTH ELEV.
(AT LINE 11)
(AT LINE 6)
(AT LINE B)
(AT LINE A)
SECTION "AA"
SECTION "BB"
SECT "CC"
SECTION "DD"
NOTES:
1. LINER PL TO BE SEAL WELDED AT ALL EDGES
2. STEEL TO BE ASTM A36 AND SHOP PAINTED WITH ONE COAT OF ZINC CHROMATE. ALL EXPOSED STEEL TO HAVE TWO FIELD COATS OF ZINC CHROMATE
3. INTERIOR OF BUILDING (EXCEPT FLOOR) TO BE PAINTED WITH OXIDIZING TYPE OF ACRYLIC PAINT. (DECOVAL OR EQUAL)
ATOMICS INTERNATIONAL
A DIVISION OF NORTH AMERICAN AVIATION, INC.
TWO REGION S.R.E. FACILITY
BUILDING IMPROVEMENTS FOR LEAKAGE REDUCTION
R7619-FB002



ROOF PLAN
SECTION "CC"
SECTION "AA"
SECTION "BB"
SECTION "DD"
SECTION "EE"
NOTES:
1. LINER PL'S TO BE 1/4 STL PL · ASTM-A 36
2. ROOF PL'S TO BE 3/16 STL PL ASTM-A 36
3. LINER PL'S TO BE CONT. SEAL WELDED AT ALL EDGES.
4. WALL ANCHORS TO BE LOCATED AT 4'-0 MAX CENTERS EACH WAY. ANCHORS TO BE SEAL WELDED TO LINER PL
5. ROOF PL'S TO BE CONT SEAL WELDE AT ALL EDGES.
6. ANGLES TO BE ∠ 3×3×1/4 UNLESS NOTED ASTM A-36 STL.
7. FILLET WELDS TO BE 3/16 UNLESS NOTED AND MUST BE GAS TIGHT
8. ROOFING TO BE 5 PLY BUILT UP WITH 1" INSULATION
9. ALL STL TO HAVE ONE SHOP COAT OF ZINC CHROMATE EXCEPT NEAR WELDS AND UPPER SURFACE OF ROOF PL'S
10. PAINT ALL EXPOSED STL, WITH TWO FIELD COATS OF ZINC CHROMATE.
11. UTILITIES AROUND AND ON WALLS MUST BE TAKEN DOWN & RELOCATED.
ELEV. ON LINE 1
LOOKING WEST
ELEV. ON LINE A
LOOKING NORTH
ELEV. ON LINE 6
LOOKING EAST
ELEV. ON LINE B
LOOKING NORTH
ELEV. ON LINE 11
LOOKING EAST
ELEV. ON LINE G
(LOOKING SOUTH)
QUICK ACTING DOOR (MARINE TYP)
SECTION "FF"
SECTION "GG"
SECT. "HH"
SECTION "JJ"
SECTION "KK"
SECTION "LL"
SECTION "MM"
TYP. DET. A
TYP. DET. B
TYP. DET. C
TYP. DET. D
TYP. DET. E
ATOMICS INTERNATIONAL
A DIVISION OF NORTH AMERICAN AVIATION, INC.
8900 DESOTO AVENUE, CANOGA PARK, CALIF.
TWO REGION S.R.E. FACILITY
BUILDING LINER FOR LEAKAGE REDUCTION
DRAWN C.C. RAMPTON
D
R7619-FB003


REVISIONS
FLOOR PLAN
LIFTING LUGS
SECTION "GG"
SECTION "HH"
CAULK
MASTIC
SECTION "DD"
SECTION "AA"
SECTION "BB"
SECTION "EE"
SECTION "CC"
SECTION "FF"
MASTIC
EMBED GROUT
EPOXY RESIN SEALER
UTILITIES
NEW CONC SEAL
ATOMICS INTERNATIONAL
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TWO REGION S.R.E. FACILITY
FLOOR DETAILS FOR BLD. LEAKAGE REDUCTION
C.C. RAMPTON
R7619-FB004

D. SEALING BY CAULKING

The alternate design to building containment was that of sealing all the joints and cracks of the existing concrete panels of the high bay walls and installing a wall between the high and low bays. By completing the following work, the high bay of the SRE building can be made to hold 1/4 inch water pressure at 100% per day leakage, please see figures 17-20.

E. VENTILATION, COOLING, HEATING, AND ENVIRONMENTAL SYSTEM

1. High Bay

The exhaust system should consist of prefilters, high efficiency air particulate filters, halogen filters, exhaust fans, all ductwork, valves, and controls necessary for a complete system. Two identical full capacity systems could be provided. Either system would be standby for the other and could be connected to the emergency power.

2. Containment Dome

A controlled inert gas atmosphere would be provided within the dome. An environmental system could maintain internal pressures below 1/4 inch water relative to the high bay. The atmosphere would be controlled to limit the oxygen content of the gas to less than 1%.

The environmental system would consist of a unit located within the dome, a unit located on the high bay floor, nitrogen gas supply and vent piping, and all controls and valves necessary for a complete system.

F. ANCHORAGE OF CONTAINMENT DOME AND STRONG-BACK

The allowable uplift force on the forty-eight 1/4-inch anchor bolts (please see figures 21 and 22) computed is 837,000 lb. This force is based on the AISC specifications for tensile stress area of a bolt and the ACI specifications for allowable stress for billet steel reinforcing bars in tension. The bond stress between the concrete and a deformed anchor bolt and the shear stress in the concrete were also checked but found not to be limiting values for the proposed anchorage.

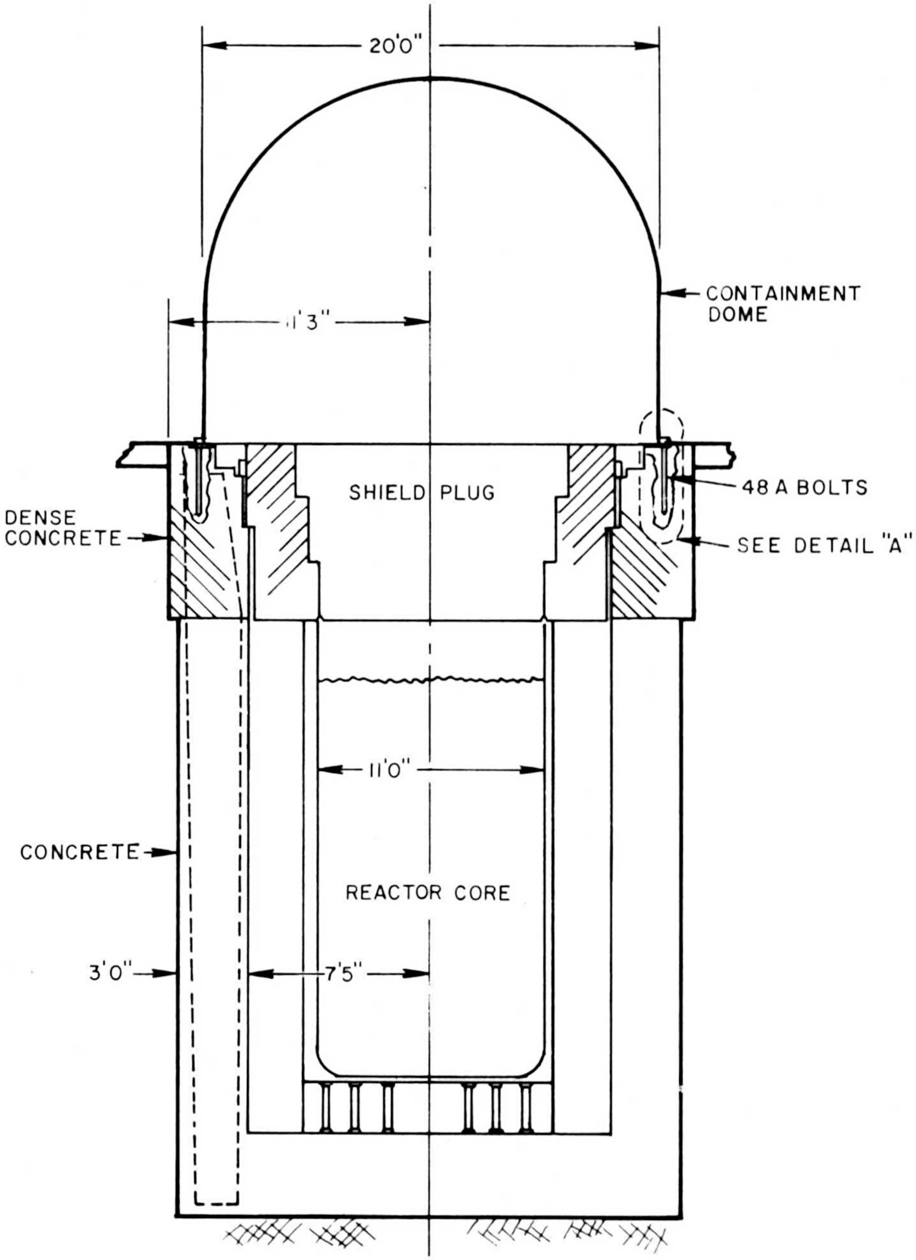

Figure 21. Bolt Placement in Reactor Structure

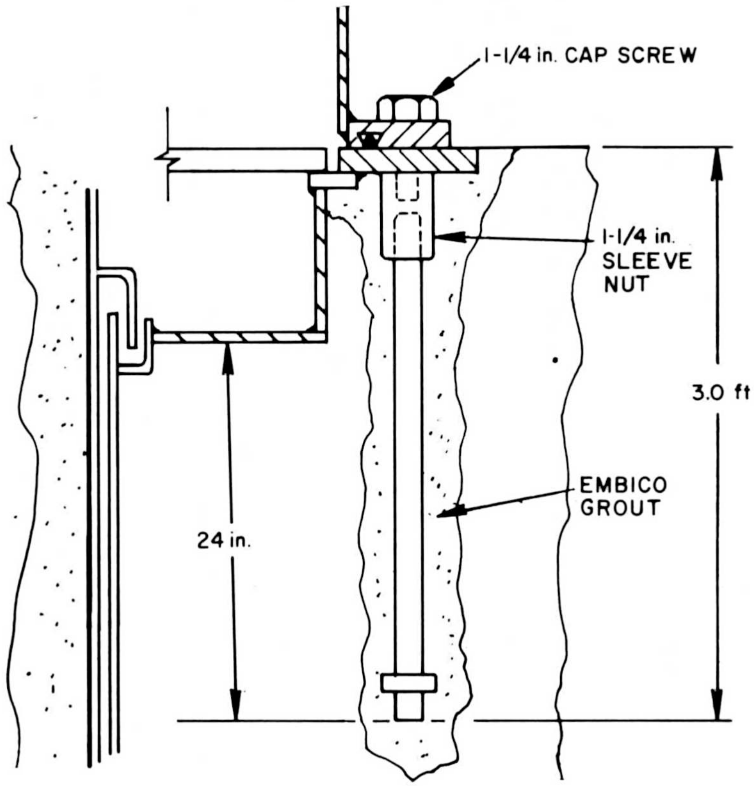

Figure 22. Detail of Pressure Dome Support Bolt

The uplift force given above and calculated in the attachment is based on building code allowables, and therefore inherent in the analysis is a factor of safety against yield, of at least 1.5.

G. SRE CERROBEND SEAL STRENGTH IN TOP SHIELD

The pressure required inside the reactor vessel to lift the top shield and break the cerrobend seal (please see figures 23 - 25) has been calculated equal to $P_{cr} = 56.5$ psi.

The critical pressure (P_{cr}) in the reactor vessel is that pressure which causes one of two possible modes of cerrobend seal failure. First, failure could occur when the cerrobend interfacial bond strength is exceeded. Second, failure could occur when the structural members of the seal form the "plastic mechanism". The analysis indicates that the failure is controlled by "interfacial bond strength" of the cerrobend alloy.

The average interfacial bond strength for an untinned contact surface is equal to about $q = 300$ psi. The tests indicate that there is no clear cut effect of the rate of load application upon the interfacial bond strength. Nevertheless, it was observed that in extremely rapid-rate-of-loading tests, the mode of fracture changed from ductile to brittle, which would indicate that the bond strength is somewhat reduced.

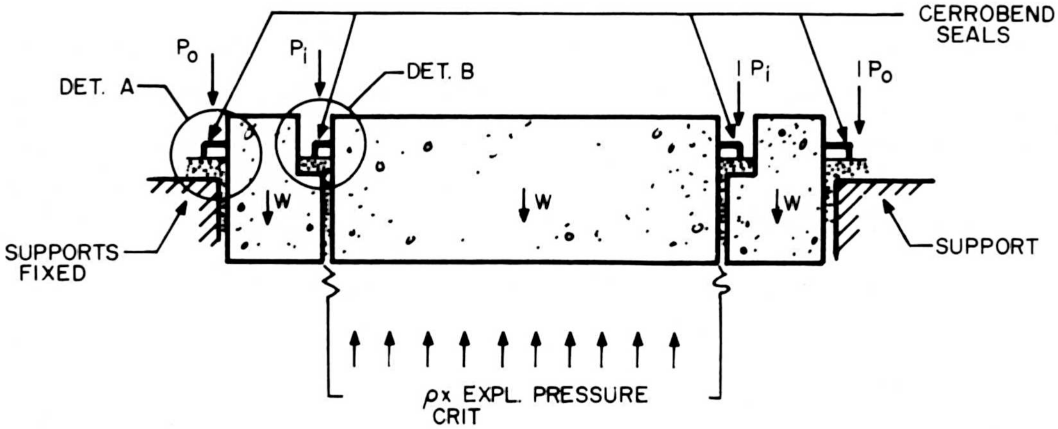

Figure 23. Loading-Face Shield

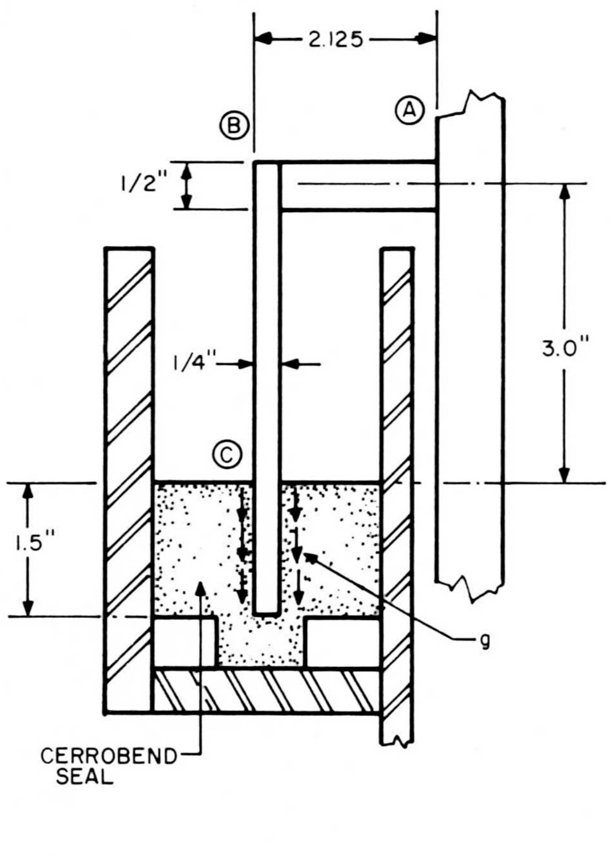

Figure 24. Outside Cerrobend-Seal

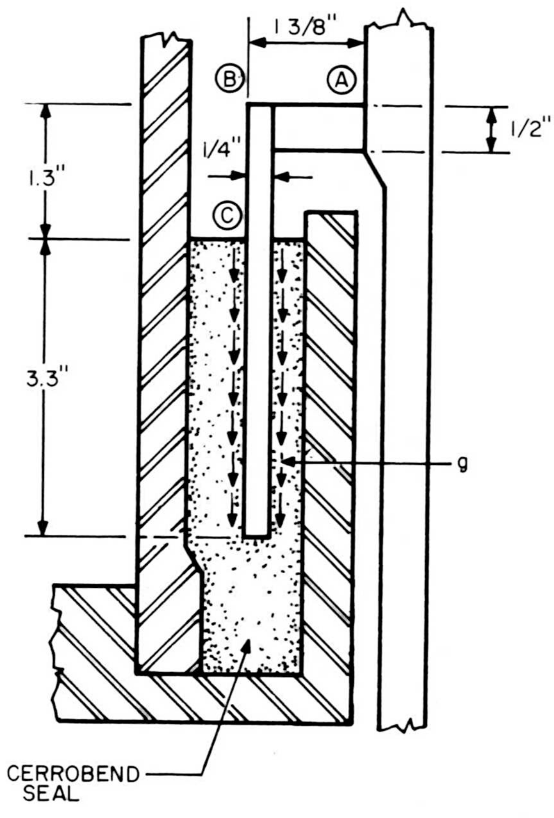

Figure 25. Inside Cerrobend-Seal

IV. CONCLUSIONS

1. On a technical basis, no additional building containment is required because of the addition of the fast test section in the SRE.

2. For the unmodified reactor, an energy release up to the equivalent of 50 lbs of TNT will not lift the loading face shield or damage the core tank. With a loading face shield leakage rate of 5%/day the boundary dose is just tolerable.

3. For larger energy releases, additional constraint of the loading face shield and additional building containment are indicated. Containment of accidents with energy releases up to 200 lbs of TNT equivalent appear to be feasible.

4. The presence of plutonium does not materially affect the site boundary dose rate with the noncredible detonation accident up to 50 lbs of TNT energy equivalent.

5. The addition of PuO_2 fuel to the system does not significantly affect the radiological considerations at the site boundary.

APPENDIX A
RELIABILITY OF SRE INSTRUMENTATION

SUMMARY

The protective instrumentation for the SRE has been designed to provide high reliability and thorough protection for the reactor. The following sections discuss the features incorporated to protect against various types of malfunctions in the instruments. This material supplements that contained in the SRE-PEP Reactor Safety Analysis Report.

1. Nuclear Instruments

The incoming 115 volt AC is normally the power supplied to the building; however, in the event of loss of building power, a battery bank temporarily supplies 115 volts AC through a diverter-pole MG set. Loss of building power also automatically starts a diesel generator which assumes the load as soon as synchronized with the emergency power from the battery.

The trip signal from the nuclear instruments to the logic system is 12 volts DC under normal conditions, and drops to 0 volts DC in the event of a trip. Complete loss of power to the chassis, therefore, automatically produces a trip signal from that chassis.

The high voltage power for the detection chambers is generated internally in each chassis. A monitor in the chassis generates an alarm if power drops below a pre-set value. The monitor also generates an alarm whenever any module is removed from the chassis, or if the function switch on the front panel is not in the operate position.

A flux comparator unit external to the nuclear instruments provides protection against loss of input signal to the power range monitors. The flux comparator unit compares the signals from the three power range monitors and actuates a visual and audible alarm if any one of the signals deviates from the other two by more than a pre-set value.

Upon receipt of this alarm, the operator determines which channel is deviating from the other two and trips this channel until the cause for the deviation is eliminated. Since the logic circuits require a positive signal to maintain them in untripped condition, loss of signal or disconnecting the leads between the nuclear instruments and the logic circuit will automatically cause a trip.

The nuclear instruments are interlocked to ensure that one type of instrument is always on scale during reactor operation. The startup interlock is actuated by the log count rate channel when this channel indicates >1.5 counts per second. This interlock must be satisfied before rods can be withdrawn. If, after rods have been withdrawn, the log count rate channel drops below 1.5 cps before other instruments are on scale, shim and safety rods will automatically drive into the reactor. The interlock is automatically bypassed when the two intermediate range monitors (Log N) are on scale or if two of the three power range monitors indicate greater than 1% power.

2. Temperature Protection System

In addition to the nuclear instruments, primary protection for the reactor is provided by two types of temperature circuits. Three channels monitor fuel channel coolant exit temperature and three channels monitor the difference between the fuel channel coolant exit temperature and the reactor outlet temperature. The latter are referred to as temperature deviation monitors. They actuate a scram if the fuel channel coolant exit temperature exceeds the reactor outlet temperature by a pre-set amount, and actuate an alarm if the coolant exit temperature drops below the reactor outlet temperature by a pre-set amount. This provides an early indication of temperature deviation from normal operating conditions.

The six temperature monitors are mounted in six individual chassis. The amplifier in each chassis is supplied from the emergency power bus, and regulated internally in the amplifier. Power for other circuits in each chassis is supplied from two redundant external power supplies. The trip units for the temperature monitors, as well as for other protective signals, are contained in a separate chassis which obtains power from the nuclear instrumentation power supply.

Both types of temperature monitors are protected against loss of input signal. The high temperature circuits are built with up-scale burnout. The temperature deviation circuits will automatically trip of the reactor outlet thermocouple is lost, and will actuate an alarm if the fuel channel exit thermocouple is lost. Loss of power to the trip unit chassis automatically causes a trip by the same mechanism described for the nuclear instrumentation.

The temperature monitors automatically trip in the event that the function switch on the front panel is not in the operate position.

3. Logic Circuit

The three redundant logic circuits (see figure 26) receive power from two redundent power supplies which generate 60 volts DC. The magnet switches shown in Figure 26 are actually located in the associated logic circuit chassis. Loss of power to a chassis automatically opens the associated magnet switches.

Triple redundancy in the logic circuit coupled with the matrix of magnet switches shown in Figure 26 provides protection against loss of a logic chassis or failure of magnet switch in the unsafe direction. Any two of the logic circuits will scram the reactor. A manual malfunction monitor is also provided on each logic chassis. This is periodically used by the operator to test the operability of each solid state switch and each of the magnet switches in the chassis. The design of the system permits switches on one chassis to be operated freely without dropping the rods. Thus, any failure of a solid state switch in the unsafe direction will be detected by the periodic monitoring.

Power for the magnets is provided by two redundant magnet power supplies. Either supply can hold the magnets if the other fails. Failure of both supplies will drop the rods. The system is wired so that a short-to-ground anywhere in the system cannot prevent the rods from dropping on receipt of a scram signal.

The manual scram button on the console actuates a trip through the logic system, and also interrupts the power to the magnets as shown in Figure 26. This provides final backup in the event of an incredible series of failures which prevent the protective system from instigating a scram.

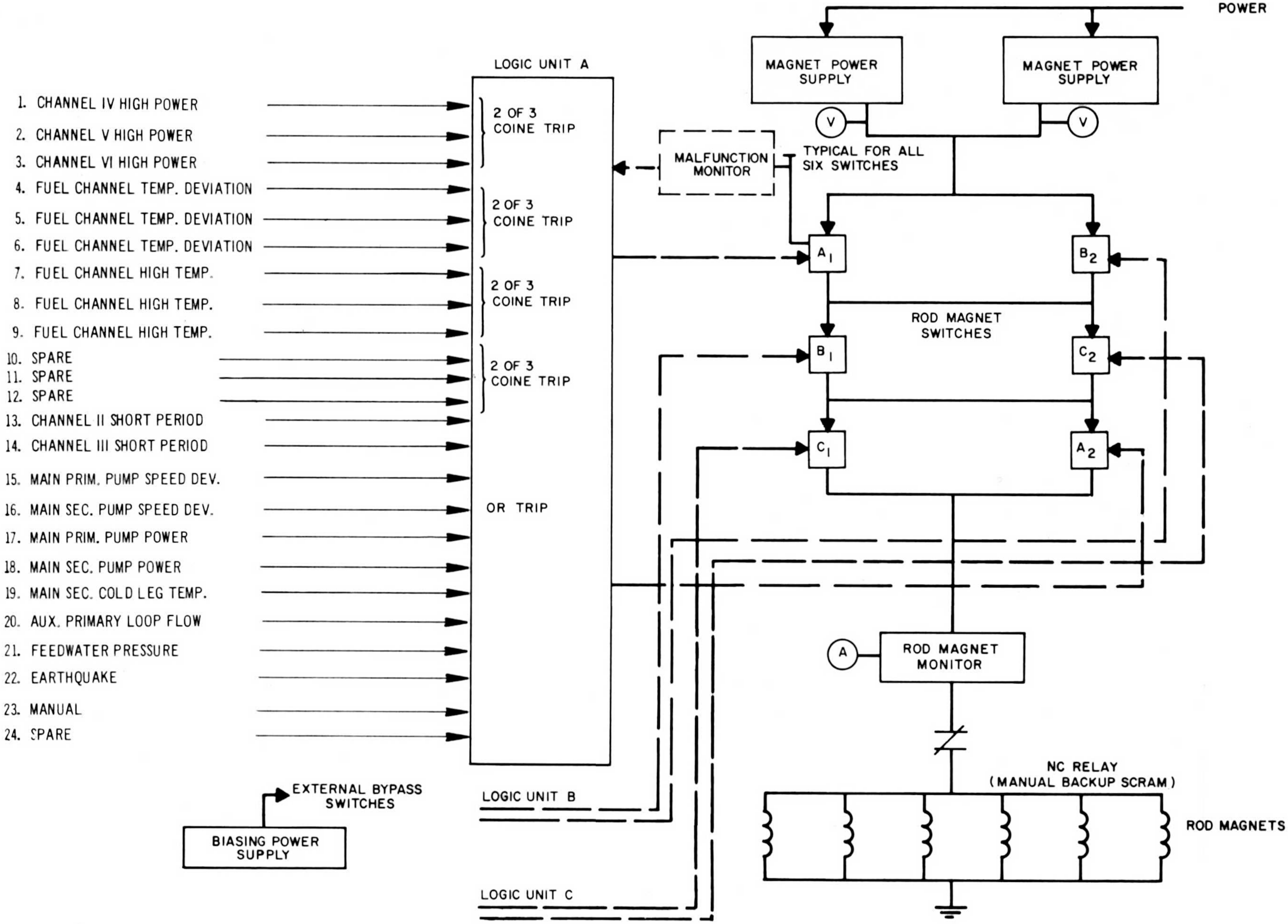

Figure 26. Logic Switching and Magnet Power Supply

4. Conclusion

The primary protective units for the reactor are the power range monitors, the high temperature monitors, and the temperature deviation monitors. Each of these systems contains three independent units connected in a two-of-three coincidence scram. Failure of one unit does not prevent scram from the other two units. Protection against loss of power, loss of signals, or operator error has been incorporated in all systems. Even if more than one unit of any type fails, a second, different, system provides adequate protection against any credible reactor excursion.

This instrument system thus provides complete protection for the reactor under all conditions.

5. Nomenclature and Table of Constants for TNT Explosion Study

A	=	Top Shield Area = 95 ft^2
A_c	=	Top Shield Annulus Flow Area = 1.5 ft^2 $\quad 0 < x < L/3$
		= 3.3 ft^2 $\quad L/3 < x < 2L/3$
		= 4.5 ft^2 $\quad 2L/3 < x < L$
C	=	Conversion Factor from Pounds of TNT to ft-lbs for Blast Wave = 1.125×10^6 ft-lb/lb
C_{na}	=	Velocity of Sound in Na
C_{cb}	=	Circumferential Distance of Cerrobend Seal = 44 ft.
E	=	Shock Wave Energy (ft-lb/ft^2)
E_p	=	Shock Wave Energy Transmitted to Top Shield (ft-lb)
G	=	Acceleration of Gravity = 32.2 ft/sec^2
H_{cb}	=	Height of Cerrobend Seal = 2 in.
L	=	Height of Top Shield - 6 ft.
$M(t)$	=	Explosion Gases Remaining in Reactor (lbs)
M_o	=	Total Weight of Fuel Evaporated
M	=	Weight of TNT Charge (lbs)
M_s	=	Weight of Top Shield = 184,000 lbs.
P_a	=	Ambient Pressure Above Top Shield (psi)
P_b	=	Initial Blast Wave Pressure (psi)
P_s	=	Shock Wave Pressure (psi)
$P(t)$	=	Time Dependent Blast Pressure (psi)
R	=	Radial Distance from Center of Explosion (ft)
V	=	Time dependent Gas Volume (ft^3)
V_o	=	Initial Gas Volume (ft^3)
$W(t)$	=	Time Dependent Gas Flow Rate (lbs/sec)
X	=	Time Dependent Top Shield Movement (ft)
σ_{ob}	=	Cerrobend Seal Shear Stress = 300 psi
ρ	=	Na Density = 52 lbs/ft^3
γ	=	C_p/C_v = 1.25
τ	=	Shock Wave Time Constant (sec)

APPENDIX B
EXPLOSIVE ENERGY RELEASE STUDIES

The following analysis represents a parametric study of the effects of explosive energy release from nuclear hypothetical accidents in the SRE. The magnitudes of released energy are measured in units equivalent to the energy of "pounds-of-TNT." This is the conventional method of evaluating large nuclear excursions in fast or coupled reactors, and allows a ready comparison with other reactor excursion studies or experimental explosion results (Reference 5).

The range of energy release* investigated were 10 to 100 lbs of TNT, equivalent to accidents of 20 to 200-Mw-sec magnitudes. TNT explosions are characterized by two unique forms of energy release, namely shock wave and blast wave. The two phenomena can be treated as different destructive mechanisms, and can by analyzed separately and independently.

Shock wave properties of chemical explosions in a liquid medium used in this analysis have been adapted from Reference 6. Although determined experimentally from underwater explosions, the general mathematical relationships are not believed to be radically different for sodium and in terms of charge weight and radial distance from center of explosion are as follows:

$$P_s \approx 2.16 \times 10^4 \left(\frac{M^{1/3}}{R}\right)^{1.13} \tag{B-1}$$

$$E \approx 2.89 \times 10^4 \, M^{1/3} \left(\frac{M^{1/3}}{R}\right)^{2.05} \tag{B-2}$$

and

$$\tau = 0.04976 \, M^{1/3} \left(\frac{M^{1/3}}{R}\right)^{-0.314} \tag{B-3}$$

*It should be noted that to bring all of the fast-section fuel and structural stainless steel to the point of melting from operating conditions requires an existent energy equivalence of approximately 130 lbs of TNT.

The pressure energy functions of Equations B-1 and B-2 have been calculated as a function of charge weight at the vessel wall in the center of the reactor. The results are summarized in graphical form in Figure 27.

The energy absorbed by the shield plug was calculated from relationships generated from Reference 7, and is equal to

$$E_p \approx 4\, E_u A(a)^{\frac{1+a}{1-a}} \tag{B-4}$$

where

$$a = \frac{Ms}{A\rho\, C_{na}\, \tau}$$

and E_u the incident shock wave energy from Equation B-2.

By making allowance for the minimum energy required to shear the cerrobend seal, the shield plug displacement from the shock wave alone has been determined.

The blast wave properties have been calculated from the following relationships. The total energy input is cM, which has to be equal to the change in internal energy and work done, and has been approximated by

$$cM = \frac{1}{\gamma - 1} P_b V_o + W_{na} \tag{B-5}$$

where W_{na} is the work done against the sodium. The volume V_o is the total gas space and was estimated as the sum of the original cover gas volume, the volume of the fast section, the increase of core vessel volume due to deformation of the shock wave, and the increase due to the deformation of the bellows and of moderator cans. The pressure p was calculated from Equation B-5 and compared with the force required to rupture the cerrobend seal and lift the top shield. When p is larger, the top shield acceleration is

$$\frac{d^2x}{dt^2} = g\left[\frac{A}{M_s} p(t) - \frac{\sigma_{cb}\, C_{cb}}{M_s} (H_{cb} - x) - 1\right] \tag{B-6}$$

when

$$X \geq H_{cb} \; ; \; \sigma_{cb} = 0$$

where

$$p(t)V(t)^{\gamma} = C'' \left[m(t) - \int w(t)dt \right] \qquad (B\text{-}7)$$

$$m(t) = m_o - w(t)dt$$

and

$$\frac{dw(t)}{dt} = \frac{A_{sg}}{L - X} \left[p(t) - p_A - c^1 w^2 \right] \qquad (B\text{-}8)$$

The constants A_s and c^1 vary as a function of X because of the three steps in the top shield and were adjusted accordingly.

Equations B-1 through B-8 were programmed for a digital computer and solved for TNT explosion energies from 10 to 100 pounds.

RESULTS

Figures 27 through 30 show the shock wave properties for various weights of TNT. Figure 30 is the shock wave displacement of the top shield if the cerrobend seal had no strength at all. Normally, even considering only its minimum shear strength of 300 psi, the top shield would not be lifted by the shock wave alone.

Figures 31 through 33 are graphical representations of the blast wave effects; in this case for an explosion equivalent to 100 lbs. of TNT. The pressure (Figure 32) does not change appreciably before the cerrobend seal has sheared off; after that, however, particularly since the explosion gases are being blown out past the moving shield (Figure 31) the pressure decreases rapidly.

Figures 34 and 35 are the initial blast pressures and mass of fuel evaporated for the weights of TNT considered, while Figure 36 is the short time gas leakage to the high bay area if the shield is being held down.

Figure 37 represents the height to which the top shield would be blown by the blast pressures generated. Three different conditions were investigated;

Figures 27-37 inclusive.
Explosive Energy
and Effects Data

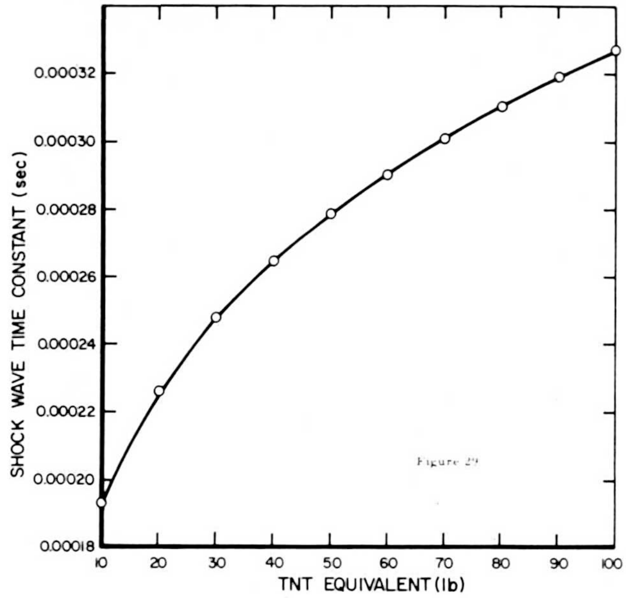

Figure 29

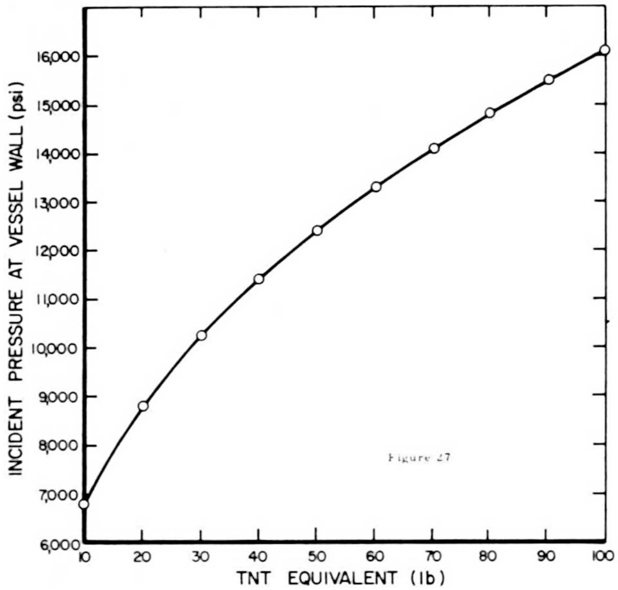

Figure 27

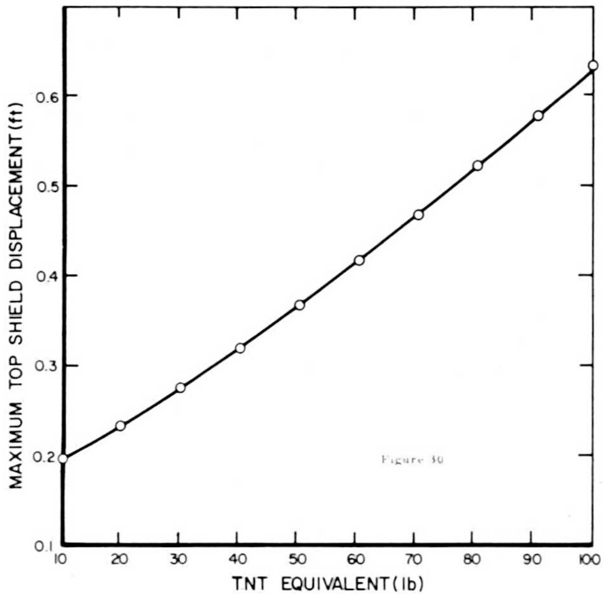

Figure 30

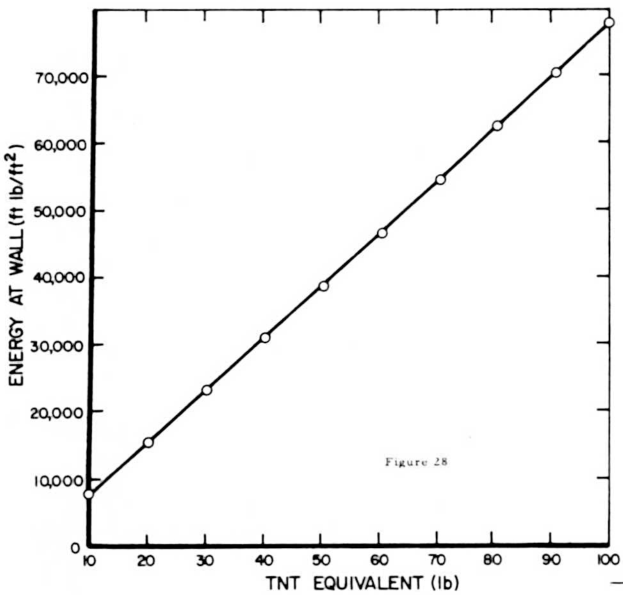

Figure 28

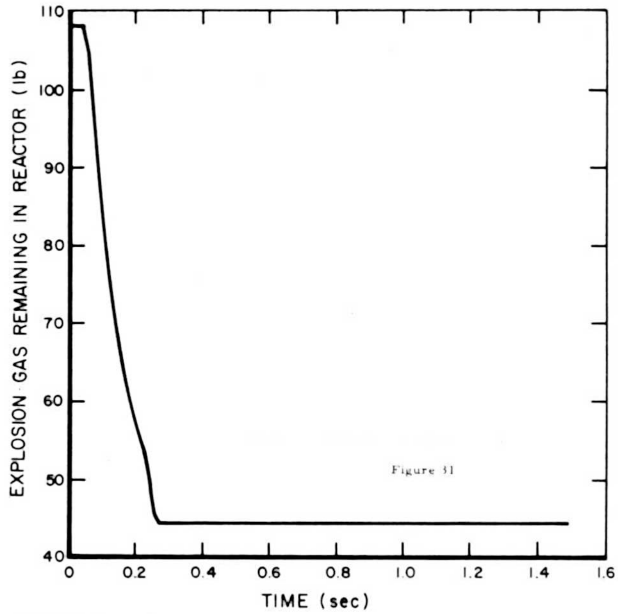

Figure 31

APPENDIX

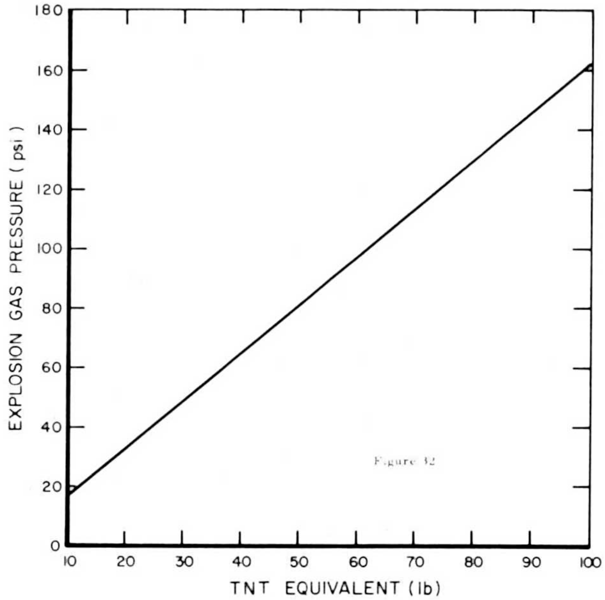

Figure 32

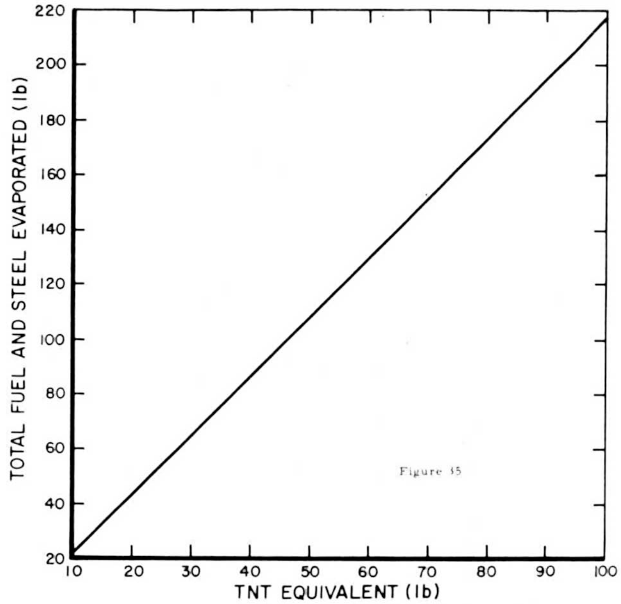

Figure 35

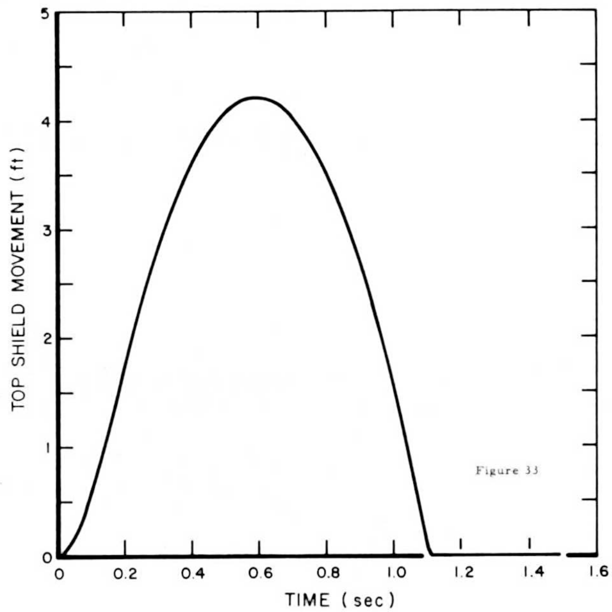

Figure 33

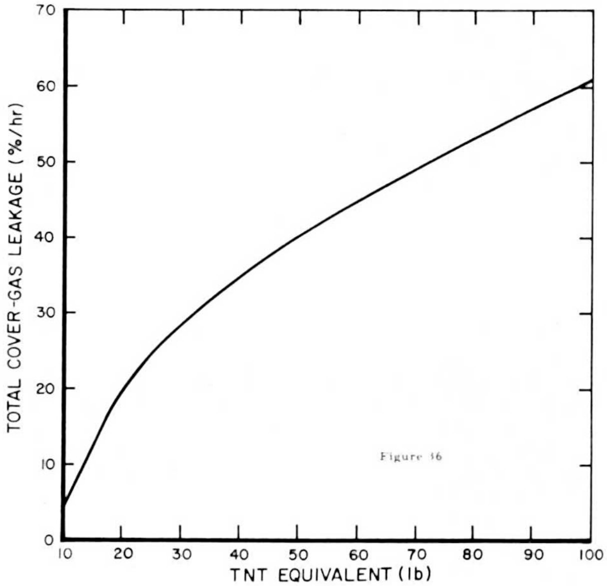

Figure 36

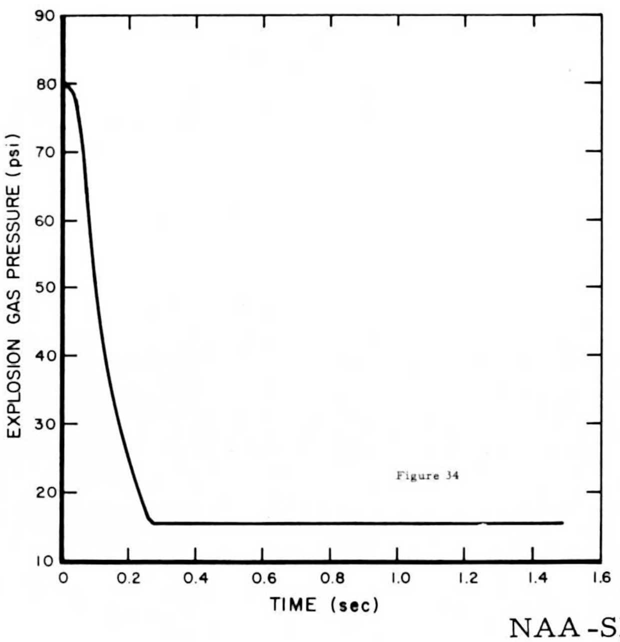

Figure 34

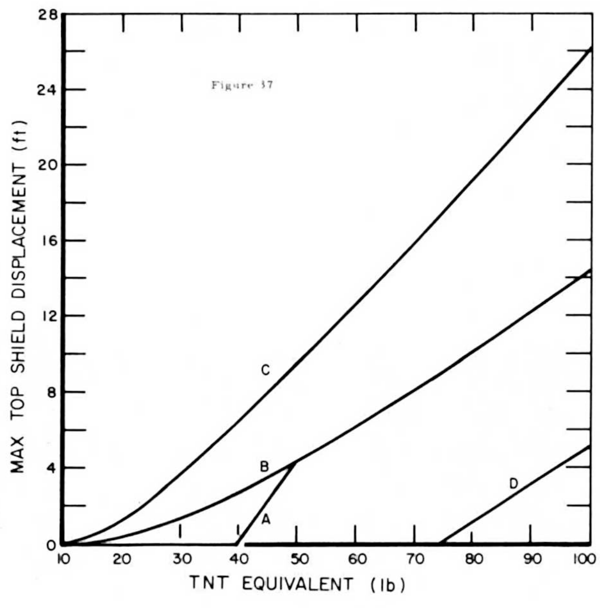

Figure 37

the curve A represents the normal conditions. In condition B it was assumed that the cerrobend seal has no strength; i.e., no energy is expended in rupturing it. In C, the cerrobend seal has not been included and it was assumed that no gas leaks past the top shield until it has cleared the loading face. This rather unrealistic case was investigated only for the sake of comparison with previous analyses which had made this simplified assumption.

Comparison of A and B shows that the cerrobend seal would prevent any lifting of the top shield for a charge below 80 lbs of TNT. Above that, however, its restraining capacity becomes progressively less effective and, in fact, negligible as the charge weight approaches the maximum of 100 lbs of TNT.

Table VIII is a summary of the shock and blast wave effects for 10 to 100 lbs of TNT considering only the cerrobend seal as the restraining force on the top shield. Column 2 is the radial deformation of the core tank from the shock wave in the center of the reactor. These values were calculated from the shock wave energy releases of Figure 30.

TABLE VIII

SHOCK AND BLAST WAVE EFFECTS

Pounds of TNT Energy Equivalent	Shock Wave Effects		Blast Wave Effects	
	Top Shield Movement (ft)	Core Tank Deformation (in.)	Maximum Pressure (psi)	Top Shield Movement (ft)
10	0.	0.7	16.1	0
20	0.015	1.33	32.2	0
30	0.025	2.05	48.3	0
40	0.035	2.71	64.4	0
50	0.045	3.45	80.5	4.22
60	0.060	4.07	96.6	6.01
70	0.10	4.80	112.7	7.94
80	0.155	5.46	129.0	9.98
90	0.210	6.15	145.0	12.12
100	0.27	6.84	161.0	14.36

APPENDIX C
SHOCK WAVE EFFECT ON REACTOR VESSEL

The SRE reactor vessel was analyzed for loads resulting from large energy releases at the center of the core, (please see Figure 38). Weights of high explosives from 20 lb to 100 lb were used in the analysis.

1. Assumptions

The analysis made was based on the following assumptions:

a) The average temperature of the reactor vessel is 700°F.

b) The blast and shock loads are those given by energy release studies modified to give a triangular pulse.

c) Fifty percent of the shock energy is absorbed in deformation of the SRE moderator elements.

d) The response of the vessel may be described as a one degree of freedom system.

e) Material properties at 700°F are

$E = 24 \times 10^6$ psi,

YP = 21,000 psi, and

Elongation 50%.

f) The strain rate resulting from the high explosive energy release will not significantly affect the material properties of Assumption No. e above.

A brief literature search was made and the results incorporated in the analysis. This is a "preliminary study" and intended only to give an estimate of the situation. The models used were the simplest and many of the results are based on our best judgment.

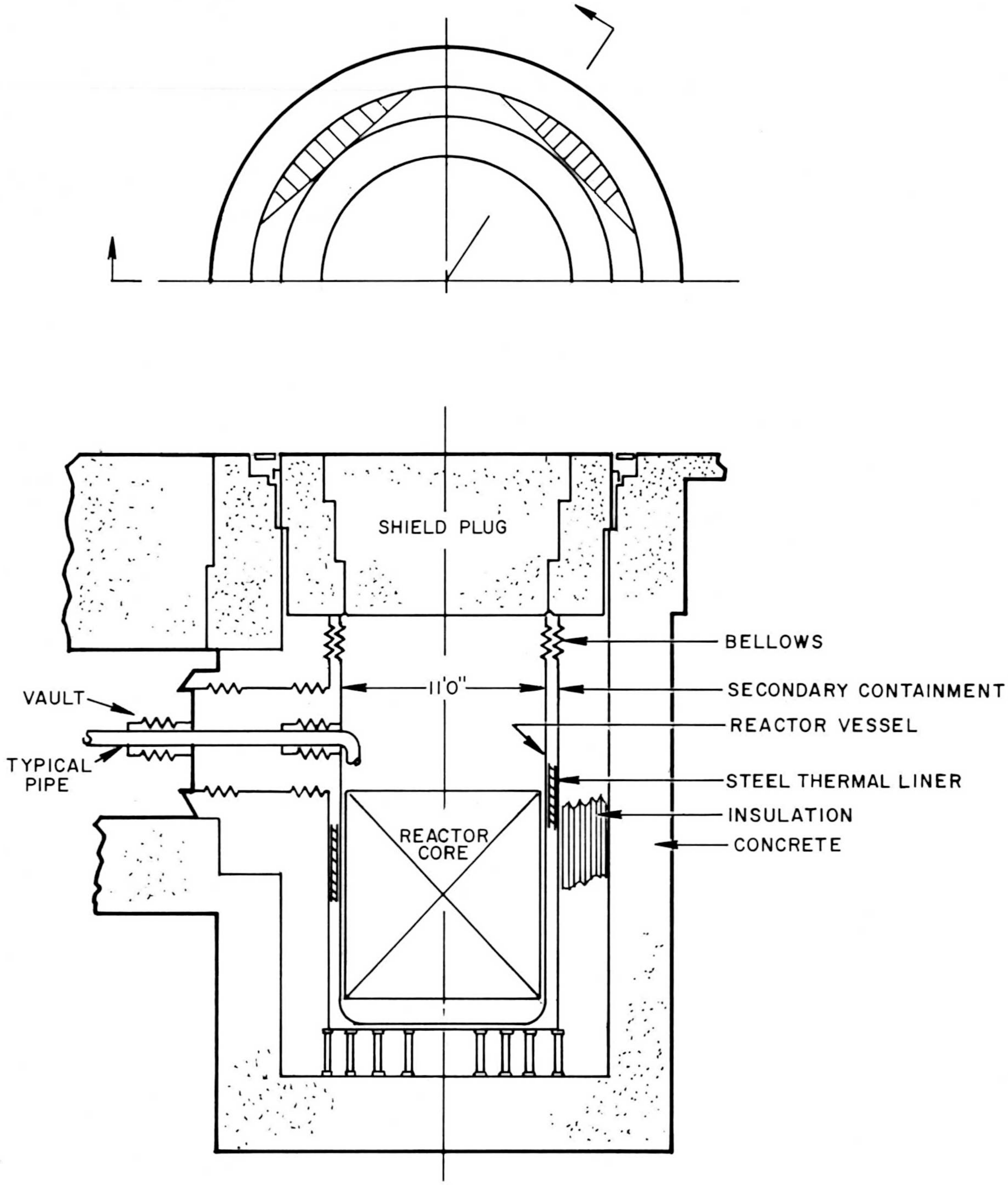

Figure 38. Typical Section Through Reactor Structure Used in Explosion Study

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Since the loading of the reactor vessel (see Figure 38) from the shock wave acts for a very short duration, it is treated as a dynamic problem with a time dependent pressure or load. The response of a simple one degree of freedom spring-mass system to a time dependent load was determined and from this an "equivalent static load" was determined. This load caused the vessel to yield and take permanent set. The magnitude of this set was determined from the amount of shock energy that was available to displace the wall. This displacement caused the vessel wall to have an elongation as follows:

TNT (lb)	Elongation (%)
20	2
40	4
60	6
80	8
100	10

It is noted that the elongations or strains do not exceed the ultimate for this material (50%).

To obtain information about the action of the SRE reactor vessel when exposed to a serious accident, such accident effects were calculated on the basis of an equivalent TNT explosion. Loading data for various amounts of TNT are given in previous section of this report.

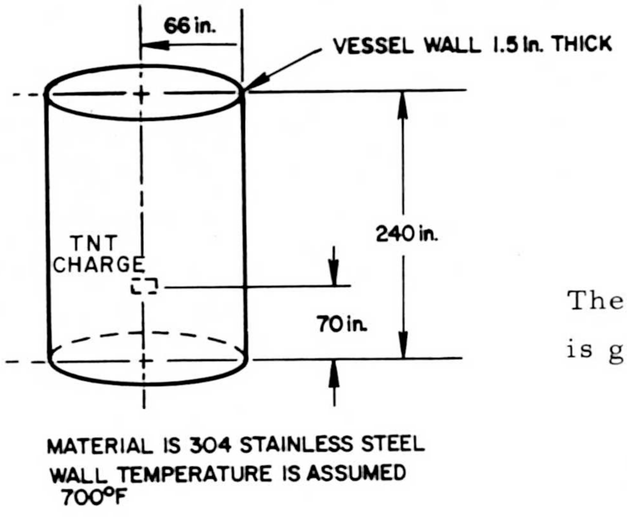

Figure 39. Vessel Geometry

The idealized reactor vessel geometry is given in Figure 39.

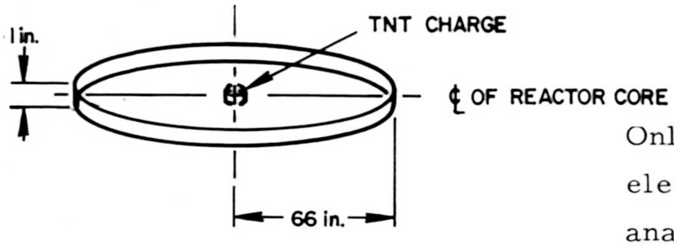

Figure 40. Analysis Model

Only the portion of the vessel at the elevation of the core centerline was analyzed (see Figure 40).

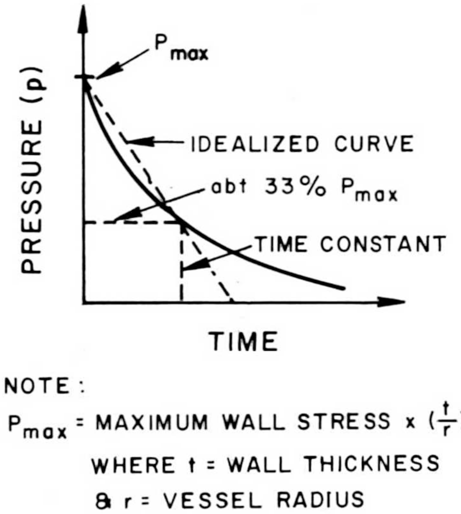

Figure 41. Shock Loading

The loading data given previously are idealized as illustrated in Figure 41.

Since the vessel loading is a time dependent function, a single degree of freedom spring mass system as shown in Figure 42 was used as the mathematical model.

The natural period of vibration (T) of this system is 23.4×10^{-4} seconds and was found from;

$$T = \frac{2\pi R}{\sqrt{\frac{E}{Z}}}$$

where

E = modulus of elasticity in psi,

Z = unit weight/384,

and

R = radius of vessel.

The response of the single degree system for the "idealized" loading is shown in Figure 43.

In order to estimate the amount of yielding, the following procedure is used. It is assumed that the vessel material has a load-deformation curve as shown in Figure 44.

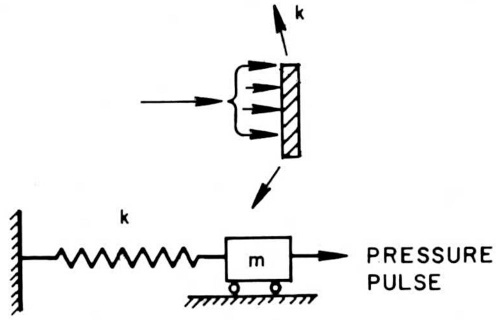

Figure 42. Loading Model

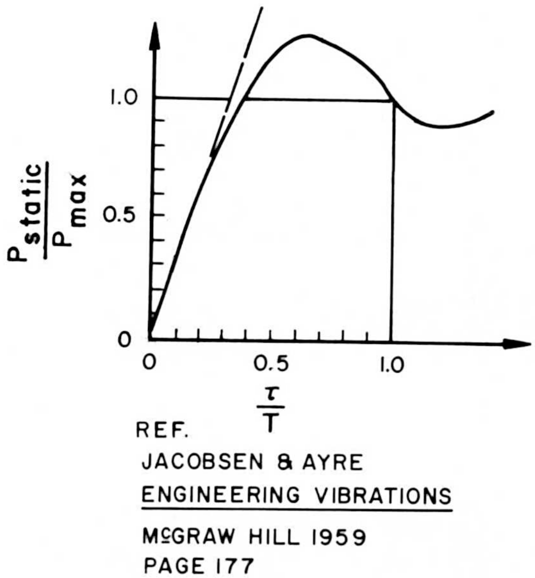

Figure 43. Shock Response

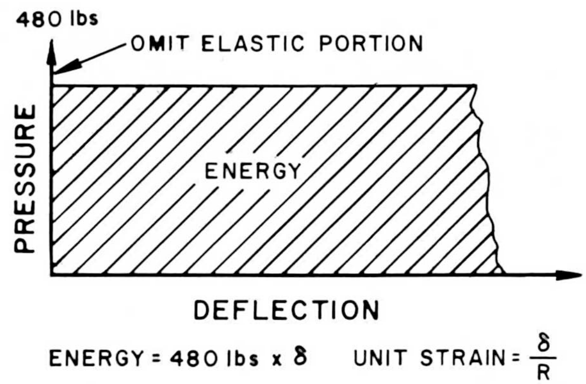

Figure 44. Load Deformation

TABLE IX

SHOCK WAVE PROPERTIES

TNT (lbs)	Maximum Pressure (psi)	Time Constant (sec)	Total Energy (Ft lb/sq ft^2)	Idealized Pulse Duration (sec)
20	9,088	2.26×10^{-4}	1.53×10^{4}	3.39×10^{-4}
40	10,633	2.65×10^{-4}	3.13×10^{4}	3.97×10^{-4}
60	13,814	2.90×10^{-4}	4.70×10^{4}	4.35×10^{-4}
80	15,381	3.10×10^{-4}	6.30×10^{4}	4.65×10^{-4}
100	16,722	3.26×10^{-4}	7.90×10^{4}	4.89×10^{-4}

TABLE X

BLAST WAVE PROPERTIES

TNT (lbs)	Maximum Pressure (psi)	Time Constant* (sec)
20	273	1
40	391	1
60	510	1
80	615	1
100	718	1

*This value is estimated

Table III summarizes the data.

$$P_{yp} = \frac{s\,\sigma_{yp}}{R} = \frac{1.5 \times 21,000}{66} = 480 \text{ psi},$$

where

s = thickness of vessel wall,

σ_{yp} = yield point of vessel wall,

and

R = radius of vessel.

TABLE XI

SUMMARY OF SHOCK DATA

TNT (lbs)	P_{max} (psi)	t/T	$\frac{t/T}{0.3}$	P_{static}* (psi)
20	9,088	0.145	0.483	4,400
40	10,633	0.170	0.566	6,000
60	13,814	0.186	0.620	8,560
80	15,381	0.199	0.663	10,200
100	16,722	0.208	0.693	11,600

*All of these pressures will cause yielding

The energy $\mathcal{N} = P_{yp} \times d$,

where

d = radial deflection of vessel wall

and

the unit strain $\epsilon = d/R$.

It is also assumed that 50% of the available energy is absorbed by the moderator elements, thermal shield, and other core element. This appears to be a conservative estimate in light of reference 9. The calculations used in Figure 30 are summarized in Table XII.

TABLE XII

SUMMARY OF ENERGY DATA

TNT (lbs)	Energy (lb/in.^{2})		Deflection (in.)	Unit Strain (%)
	Total	50%		
20	1275	638	1.33	2
40	2608	1304	2.71	4
60	3916	1958	4.07	6
80	5250	2625	5.46	8
100	6582	3291	6.84	10

Unit strains of the amount found in Table XII indicate that no general disintegration or cracking will occur. However, at discontinuities (pipe and nozzle attachments) there is a good probability that some local cracks will occur.

It is possible that gas leakage through cracks that may occur around vessel discontinuities could be contained by the containment diaphragms and prevented from leaking into the vault; however, to give this area a detail numerical analysis would require time and effort beyond the scope of this preliminary analysis.

2. Results

a) The loading resulting from the shock wave will cause the most damage.

b) The reactor vessel will bulge outward and take a permanent set (from 1-1/2 to 6 inches in radius; however, no general disintegration or cracking is to be expected.

At vessel discontinuities (pipe and other vessel attachments) local cracks would occur. A great deal of the available energy would be expended in deforming the vessel and forming these cracks. However, it is probable that the secondary containment will not be ruptured as a result of the deformations and reduced loadings that will be imposed on it. This is an area for further investigation.

The bellows at the top of the SRE reactor vessel would deform a greatly and local cracks would occur at the weld connections; however, this would depend on the ductility of the weld joints and the radius of bend at the joints. This could result in a rupture of the secondary containment; however, no gases would be released to the vaults as a result of this mode of failure.

APPENDIX D
BETHE–TAIT ACCIDENT RESULTS

A parametric study was made using the modified Bethe-Tait method developed by Jankus[4]. TR-SRE kinetics and reactivity characteristics were used in computing the associated reactivity ramp rate. A maximum reactivity ramp rate of $27/sec was computed for core compaction. However, ramp rates from 0 to 100 $/sec were used in this survey. The results are presented in Figures 45 through 48.

1. Summary of Equations

a) Maximum Explosive Energy (Mw-sec)

$$W_{max} \simeq \frac{4}{3}\pi a^3 \rho Q^* \left(\frac{Q-Q^*}{Q^*}\right)^{5/2} \left(\frac{Q^*}{qQ}\right)^{3/2} \frac{2}{5} \; ; \; Q^* < Q < Q^*(1-q)^{-1} .$$

$$W_{max} \simeq \frac{4}{3}\pi \rho Q^* \left[\frac{Q}{Q^*}(1 - 0.6q) - 1\right] \; ; \; Q > Q^*(1-q)^{-1} .$$

b) Ramp Rate

$$(k_1 - 1 - \beta_{eff}) \simeq \alpha t .$$

c) Total Energy Released

$$E_t \simeq \frac{4}{3}\pi a^3 \rho Q (1 - 3/5q) .$$

2. Nomenclature

a	=	The effective radius of fast section
ρ	=	The fuel density in fast section
Q^*	=	The threshold energy for pressure buildup
Q	=	The energy density in fast section
q	=	The parameter used to numerically represent power distribution
k_1	=	The multiplication in the fast section
β_{eff}	=	The effective delayed neutron fraction
E_t	=	The total energy release
α	=	Constant for ramp rate insertion

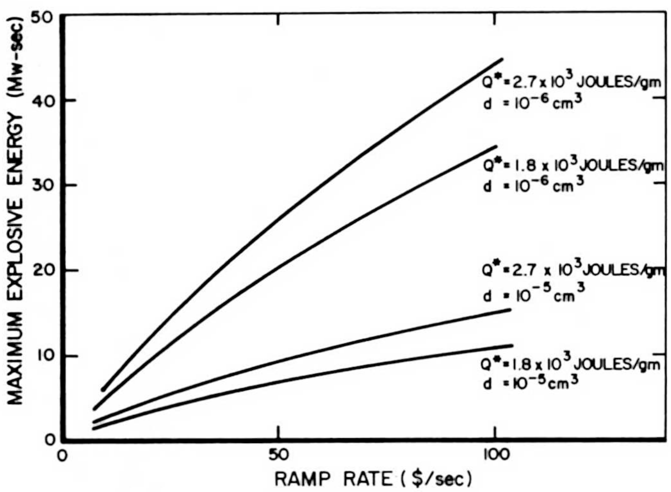

Figure 45. Maximum Explosive Energy Vs Ramp Rate, $\ell p = 25\ \mu s$

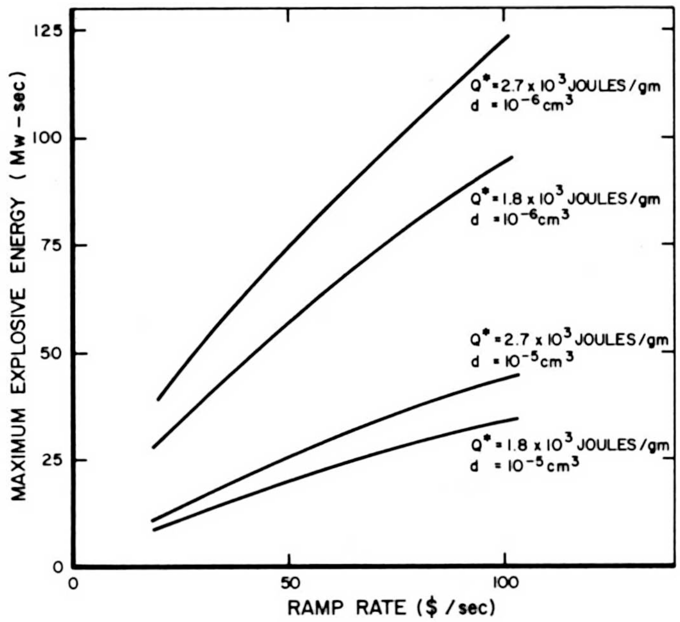

Figure 46. Maximum Explosive Energy Vs Ramp Rate, $\ell p = 0.5\ \mu s$

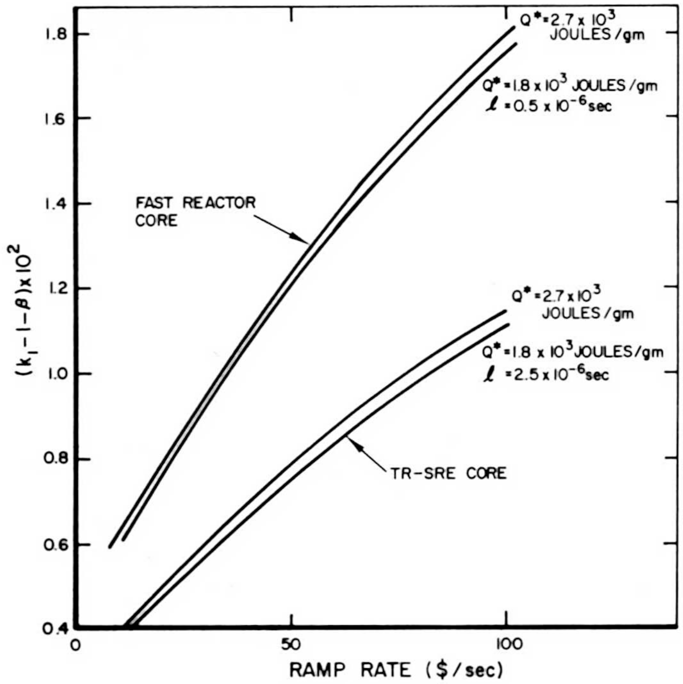

Figure 47. $k-1-\beta$ Vs Ramp Rate

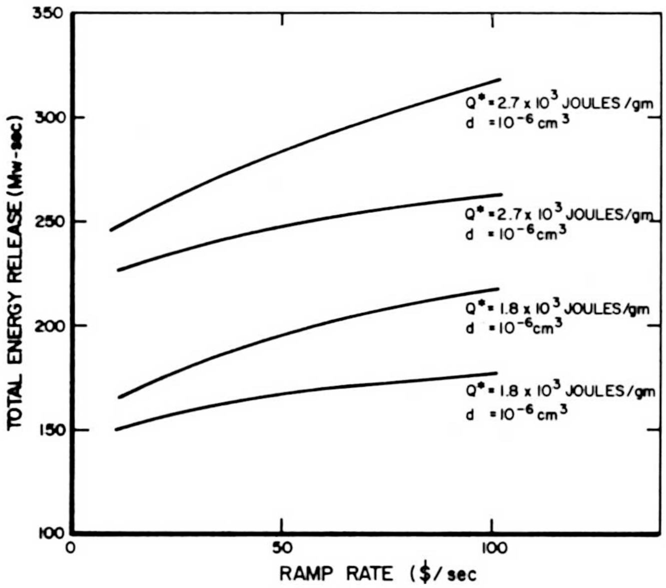

Figure 48. Total Explosive Energy Release Vs Ramp Rate

APPENDIX

APPENDIX E
RADIOLOGICAL ANALYSIS FOR MELTDOWN ACCIDENT

1. Hypothetical Meltdown Accidents (With Additional Containment)

Two hypothetical meltdown accidents have been analyzed for associated radiological hazards. It was assumed that all fuel in the fast section is at its boiling point and that a reactor excursion occurs releasing 200 Mw-sec (100 lb of TNT) of energy for one case, and 100 Mw-sec (50 lb of TNT) for the other. All of this released energy is assumed to vaporize fuel. Assuming that the loading face shield does not have a hold-down device, the 200-Mw-sec release lifts the top shield, whereas the 100 Mw-sec release does not. All noble gases and iodine in the fast section are assumed to be released to the core cover gas for both cases. However, it is expected that only a small fraction of iodine would be released from the sodium pool. The fission product inventory is based on a burnup of 300 days at full power.

The AISITE Code[10] and Pasquill's[11] type F meteorology were used to calculate irradiation doses. The reactor building inleakage is 115%/day. Building effluent passes through a prefilter, a high efficiency absolute filter, and a halogen filter before it is discharged out the stack. Two filter and fan assemblies are provided. One assembly is a standby and both fans are on emergency power. A redundant radiation monitoring system in the exhaust system will detect an abnormal air activity level and automatically shut off the building air supply. The calculated radioactive release from the building is based on a 99.99% absolute filter efficiency and a 98% halogen (charcoal) filter efficiency. The manufacturer's rated charcoal filter efficiency is greater than 99%. A more conservative number was used to account for a possible, but unproven, decrease in efficiency with use.

Since the reactor is located in rugged terrain, a ground release was assumed for the calculations. The containment shell located over the reactor is designed for a 1%/day leakage rate.

2. <u>200 Mw-sec Energy Release</u>

In an energy release of 200 Mw-sec it was assumed that 34 lb of this fuel and its associated fission products were instantaneously released to the containment shell. The assumptions for this analysis are as follows:

1) All iodine and noble gases in the core cover gas are instantaneously released to the containment shell,
2) All of the noble gases in the containment shell leak out of the shell,
3) 50%[16] of the iodine in the containment shell leaks out of the shell,
4) Thirty-four lb of sodium is released instantaneously to the containment shell, and
5) Vaporized materials with high melting points that were released to the shell readily solidify and settle or plate out. Therefore, 1%[16] of the Pu, solid fission products, and sodium in the shell are available for release from the shell.

The 2-hr irradiation dose at the site boundary was calculated for various shell and building leakage rates (see Figure 49). For a reasonable range of leakage rates, the cloud plus inhalation dose at the site boundary is small. This dose is 3 rem to the thyroid and 0.28 rem whole body for the design leakage rates of 1%/day and 115%/day for the shell and building, respectively.

Due to the postulated high release of fission products to the shell for this hypothetical accident, the 2-hr direct dose at the site boundary is ~ 200 rem. At a distance of 1050 ft from the reactor, the 2-hr direct dose is ~ 25 rem. The terrain in the vicinity of the site boundary is rugged and desolate, and only a small area can be viewed from the reactor. Beyond ~ 600 ft from the reactor, the elevation decreases rapidly (initially the elevation decreases 125 ft in ~225 ft from the mountain would be between the receptor and the reactor. Of course, additional shielding in the vicinity of the containment shell would reduce the direct dose. The direct dose at the site boundary would be less than 20 rem for 1 inch of lead shielding.

At a distance of ~ 1640 ft from the reactor, the total (cloud, inhalation, and direct) dose for 30 days is 25 rem to the whole body and 250 rem to the

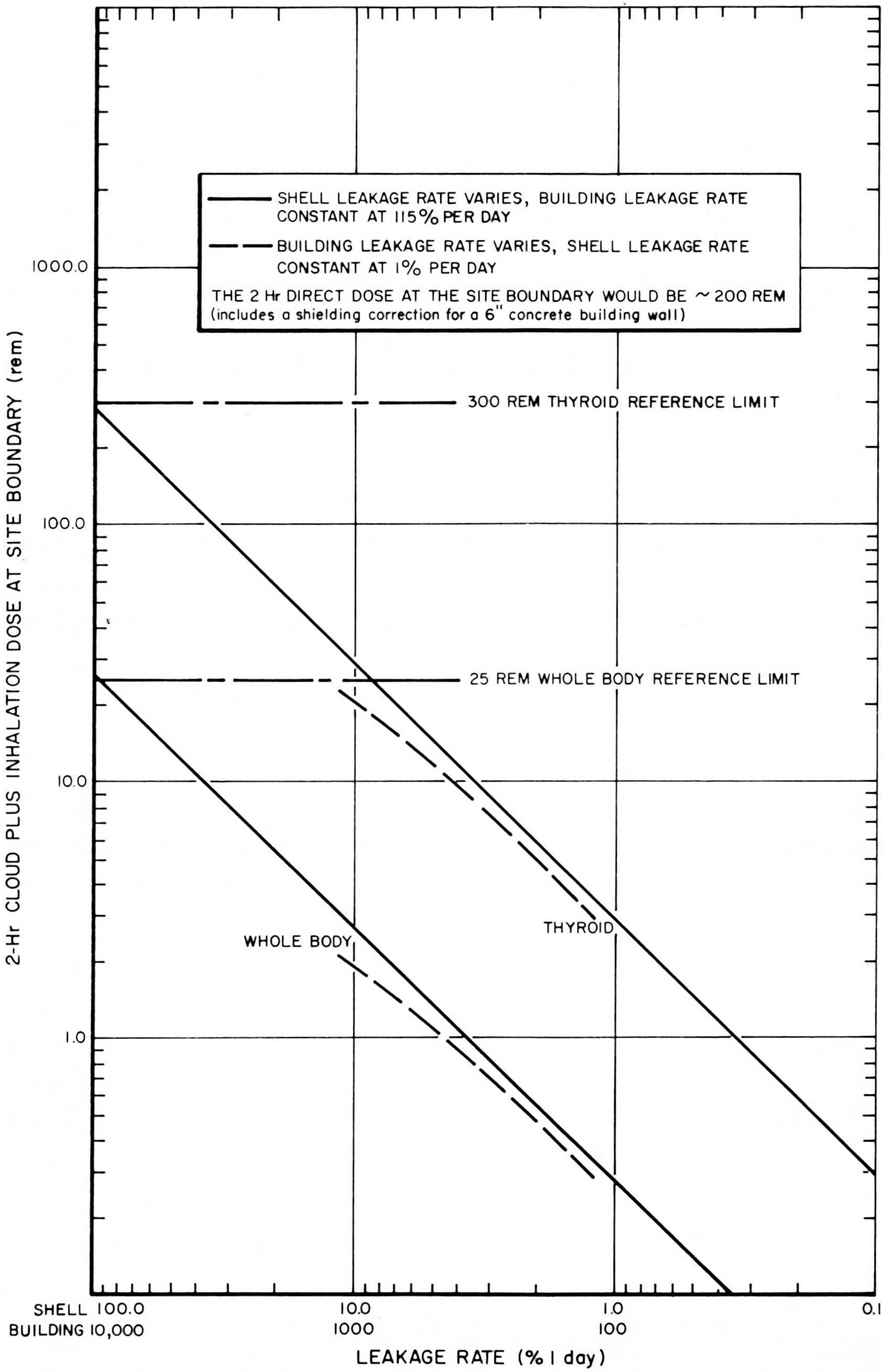

Figure 49. Irradiation vs Leakage Rate for a 200-Mw-sec Transient

APPENDIX

thyroid. This includes a 19 rem direct dose based on no additional shielding. At a distance of ~ 1340 ft from the reactor, the cloud plus inhalation dose for 30 days is 7.5 rem to the whole body and 300 rem to the thyroid.

In this hypothetical accident case, fission products are the major contributor to the 2-hr inhallation dose at the site boundary. For the design shell and building leakage rate and building ventilation system, this dose is 0.007 rem to the bone from plutonium compared to ~ 2.7 rem to the thyroid from iodine. Thus, at the site boundary the dose* from plutonium is not controlling compared to that from iodine.

Extrapolation of data in Figure 49 shows that both the thyroid and whole-body 2-hr dose at the site boundary are less than 0.1 rem, if it is assumed that the building and shell leakage rates are 1% per day. Thus, if sufficient shielding is provided to reduce the direct radiation from the building, the amount of fission products released to the dome could be greater than that produced by the 100 lb of TNT case without exceeding the suggested limits* for emergency radiation doses at the site boundary.

3. 100 Mw-sec Energy Release

In an energy release of 100 Mw-sec, it was assumed that all the vaporized fuel and its associated fission products are available to leak out of the core to the containment shell for 15 minutes following the energy release. Thereafter, 1% of the fission products (excluding noble gases and iodine) and plutonium, from the fast fuel remaining in the core, will leak to the shell. The initial leakage rate is estimated to be 26.7%/hr from the core to the shell. It is assumed that this leakage rate reduces to the present maximum assumed operating leakage rate of 45%/day within 15 minutes. (The normal operating leakage rate is less than 45%/day.) Also, it is assumed that 100% of the noble gases and 25% of the iodine in the fast core will leak out at these rates. Experimental data at Atomics International shows that only 25% of the iodine released to the cover gas is transported to a cooler temperature region (such as the containment shell). Due to model limitations in the computer code, it was necessary to assume that all fission products released to the shell in a 2-hr

*Suggested limits for emergency radiation doses: 25 rem to the whole body and 300 rem to the thyroid.

period were released immediately. This assumption results in an overestimation of the 2-hr dose at the site boundary. It is assumed that 1%[16] of the vaporized fuel and associated fission products (except iodine and noble gases) leaking to the shell are released from the shell. All of the other fission products and plutonium reaching the shell are released from the shell.

Figure 50 shows the 2-hr dose at the site boundary for various shell and building leakage rates. The cloud plus inhalation dose at the site boundary is 0.105 rem to the thyroid, and 0.019 rem whole-body for the design leakage rates of 1%/day and 115%/day for the shell and building, respectively. The 2-hr direct dose at the site boundary is 6.4 rem.

The cloud plus inhalation dose at the site boundary is ~ 2.5 rem to the thyroid, and ~ 0.35 rem to the whole body for a 1% per day shell leakage rate and 4000% per day building leakage rate. The present building leakage rate is 4000% per day.

Assuming a 115% per day building leakage rate and no containment dome, the 2-hr site boundary doses* (cloud plus inhalation) are 255 rem to the thyroid and 49 rem to the whole body. The 2-hr direct dose at the site boundary is 6.4 rem.

*It is assumed that 1% of the vaporized fuel and associated fission products (except iodine and noble gases) leaking to the building are available for release from the building.

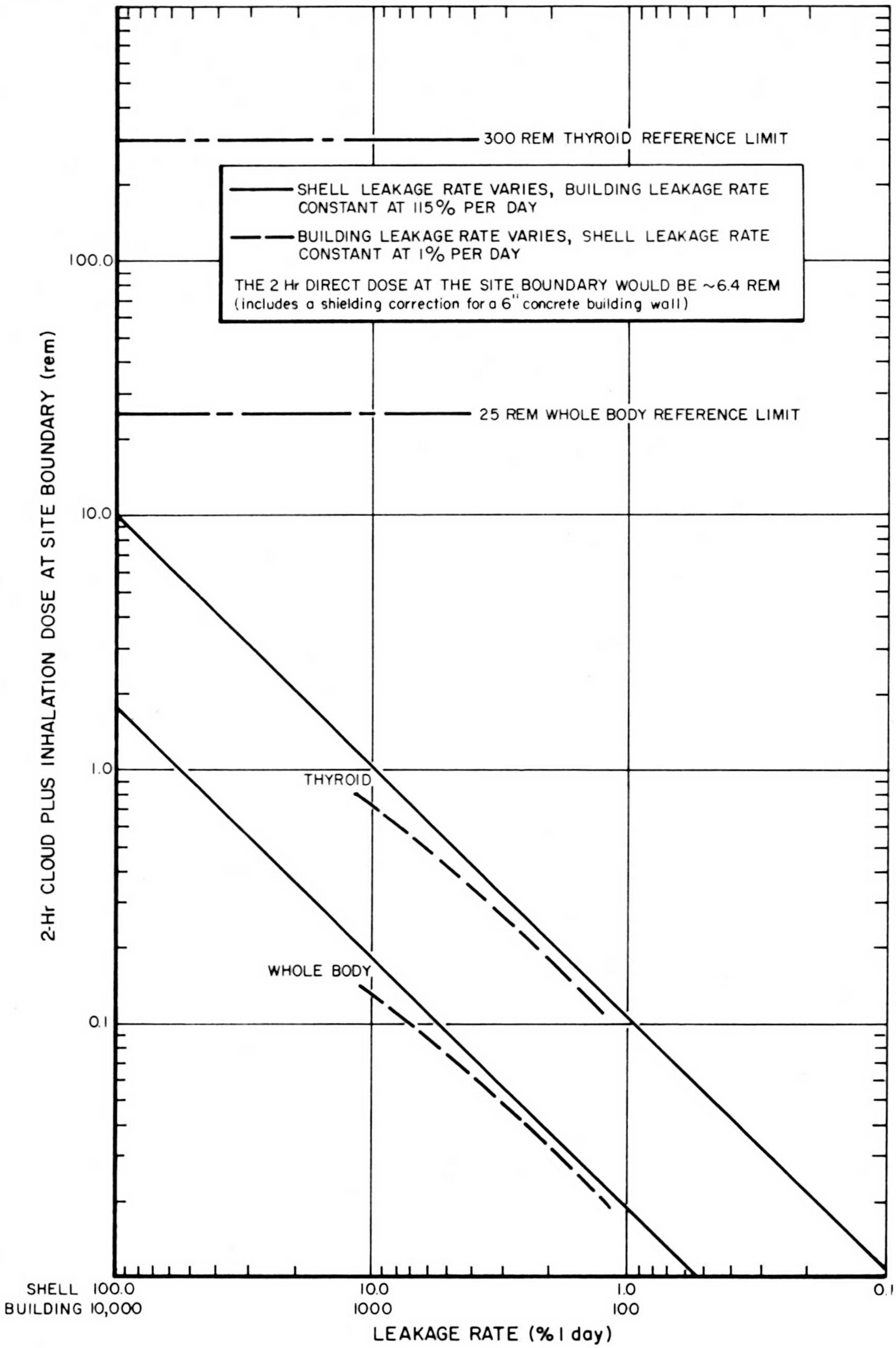

Figure 50. Irradiation vs Leakage Rate for a 100-Mw-sec Transient

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