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SNAP 8 EXPERIMENTAL REACTOR (S8ER)
FINAL SAFEGUARDS SUMMARY REPORT
(Title Unclassified)

AEC Research and Development Report

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SNAP 8 EXPERIMENTAL REACTOR (S8ER)
FINAL SAFEGUARDS SUMMARY REPORT
(Title Unclassified)

AEC RESEARCH AND DEVELOPMENT REPORT

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I. INTRODUCTION

A. SCOPE AND PURPOSE OF REPORT

This report is the final safeguards summary for the SNAP 8 Experimental Reactor (S8ER) Test. The report gives a brief description of the site, surrounding environment, facility, reactor, and the associated process and control systems. The normal and potential hazards associated with the operation of the facility are examined, and the inherent and built-in safety features provided to minimize the consequences of credible and hypothetical accidents, are analyzed and evaluated.

The purpose of this report is to show that the reactor and installation are so designed, constructed, and sited that the S8ER can be operated safely and will not effect any undue hazards to either the general population, operating personnel, or adjoining test facilities and personnel.

B. PROGRAM

The SNAP 8 (Systems for Nuclear Auxiliary Power) program is being developed jointly by the Atomic Energy Commission (AEC) and the National Aeronautics and Space Administration (NASA). The AEC is responsible for the reactor development, nuclear safety, and nuclear systems testing, and NASA is responsible for the power conversion system development and the overall systems engineering. The portion of the program described herein is known as the SNAP 8 Experimental Reactor (S8ER) test. This test is the first in a series of three reactor test systems which will be constructed and operated prior to launching the SNAP 8 system into orbit in 1965. The S8ER test is a test of the core only, without power-conversion-system equipment. The objectives of the test are (1) to determine the operating characteristics of the reactor; and (2) to demonstrate reactor performance at 300 kw_t, with a reactor coolant outlet temperature of 1300°F. Subsequent tests will be conducted at 600 kw_t for follow-on development of a 60 kwe system. This reactor test is being performed to verify the SNAP 8 reference design,¹ and to provide an experimental basis for any design improvement that may be indicated.

The test will be performed for the AEC by Atomics International (AI) in the former SNAP 2 Experimental Reactor Test Facility (SER), which is situated on AEC-optioned land at the Atomics International Nuclear Field Laboratory area,

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located in the Simi Hills in Ventura County, California. The successful operation of a 50-kwt reactor in a similar test was completed in the facility in 1960.² In 1961, the facility was improved and modified to its present status for the S8ER test. Operation with the 300 kwt reactor is scheduled to begin in May 1962. Prior to the reactor test, the fuel will be used in a critical experiment, which will be performed at another facility at the field laboratory. The S8ER tests and high power 600 kwt experiment should be completed in 1963.

D. S8ER SUMMARY DESCRIPTION

The reactor to be tested is a compact, hydrogen-moderated, beryllium-reflected reactor containing highly enriched uranium. The core contains 211 individual fuel rods consisting of homogenously combined zirconium, uranium, and hydrogen to form a hydrided Zr-U alloy. Each fuel rod is provided with a ceramic-lined Hastelloy cladding. The ceramic coating is applied to the inside surface of the cladding to prevent loss of the moderator as a result of hydrogen diffusion through the cladding. The core is contained in a stainless steel core vessel approximately 9-1/2 in. in diameter by 21 in. long. Heat is removed by a liquid NaK coolant which enters a plenum at the bottom of the core at 1100°F and exits from a plenum at the top of the core at 1300°F. The primary coolant inlet and outlet lines are 2-in. diameter, stainless steel pipe. The core vessel is surrounded by about 3 in. of beryllium, which forms the external radial reflector. Reactor control is accomplished by rotation of portions of this reflector (control drums) toward or away from the core.

The entire reactor and control system are contained in a shielded carbon-steel pressure vessel (containment vessel) which contains a helium atmosphere and remains sealed during reactor operation. The primary system coolant lines penetrate the containment vessel through welded seal caps and are then routed to a shielded, steel-lined vault where the heat is transferred from the primary NaK coolant loop to a secondary NaK coolant loop. A rupture diaphragm, located within the containment vessel is provided in the primary heat transfer loop to ensure that, in the event of overpressure, the primary piping will fail inside the containment vessel. Thus a possible pipe rupture outside the containment vessel (in the primary system vault) due to overpressure, will be prevented. As a result, the pressure vessel and primary coolant system comprise the primary containment system. A nitrogen atmosphere is maintained in the primary

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heat-transfer-system vault, which remains sealed during operation. Heat is removed from the nonradioactive, secondary, NaK loop by an air-blast heat exchanger located outside the building.

The S8ER building (See Figure I-1) consists of a single, rigid, steel-frame superstructure with corrugated metal siding. It contains a control room, mezzanine office space, and change room in one end of the building, and a reactor room in the remaining portion. The reactor and containment vessel, primary-system vault, and secondary-system-equipment pit are located below the reactor room floor level. Some secondary-loop equipment, the necessary power service, and a 7-1/2-ton bridge crane are located above the reactor room floor level. The building ventilation stack, secondary NaK surge tank and air-blast heat exchanger, and the equipment associated with shield-cooling systems and the helium- and nitrogen-supply systems are located on concrete slab areas exterior to the building.

Figure I-1. S8ER Facility (cutaway)

II. SUMMARY OF MAJOR SAFETY FEATURES AND CONCLUSIONS

A summary of the main safety features of the installation, results of the accident studies, and conclusions are presented below.

A. SITE

The Atomics International Nuclear Field Laboratories Area is located in the Simi Hills, which form a steep, rocky barrier about 900 ft high at the western edge of the San Fernando Valley. The test installation is located on an alluvial plain of about 1600 acres with an elevation varying from 1815 to 1830 ft above sea level. This area is unpopulated, save for other reactor and test installations related to North American Aviation programs. The nearest inhabited area is the town of Santa Susana, 3 miles north of the site. The nearest residence is approximately 2 miles from the facility (northerly direction).

B. FACILITY

The facility was originally constructed and successfully used for the 50-kwt SNAP 2 Experimental Reactor (SER) program which was completed in 1960.² The most significant safety feature of the facility is the primary containment system, which is designed to contain all credible reactor accidents. In addition, the entire primary coolant system (radioactive) is contained in a shielded vault. A helium atmosphere is maintained in the containment vessel, and a nitrogen atmosphere is maintained in the primary coolant system vault during all reactor operation (to exclude oxygen and prevent liquid metal fires).

The proposed 600-kwt S8ER program is similar in nature to the SER program, except for the higher power level. Significant facility improvements which have been made include the expansion of the control room and office space, improved handling equipment (hoists and bridge crane) in the reactor room, more effective shielding in the containment vessel and for the enlarged primary system vault, the addition of a facility stack and gas decay tank, air conditioning for the control room, and improvements in the auxiliary cooling and service systems.

C. SNAP 8 REACTOR

The SNAP 8 reactor has three inherent properties that minimize the power excursion possible during nuclear accidents.

- 1) Heat is generated directly in the homogeneous fuel moderator region, giving prompt negative temperature coefficient response to changes in reactor power level.
- 2) Because of the extremely compact core design, the heat capacity of the core is small, and a power surge results in an immediate temperature rise in the fuel moderator.
- 3) The hydrogen-retention properties of the fuel-moderator material are diminished at high temperature. At moderately high temperature (but well below the fuel melting point) the fuel cladding will rupture due to hydrogen pressure, thus causing rapid diffusion of hydrogen from the core. This results in rapid shutdown of the reactor.

D. HEAT TRANSFER AND CONTROL SYSTEM

The design of the heat transfer system is based on tried and proven design criteria. All the coolant lines are constructed of type 316 stainless steel. The system is designed to operate with internal pressures of 100 psig at 1400°F and is provided with a 50-psig rupture disc, which will relieve into the containment vessel in the event of overpressure. Both the primary and secondary NaK systems are completely closed loops using linear induction electromagnetic pumps as a driving force for fluid pumping.

The control drum drives are of a fail-safe design. In the event of a scram signal, the electromagnetic clutches are released and the control drums are driven full out by torsional scram springs. Any four of the six control drums will fully shut the reactor down under all temperatures and power conditions.

The safety instrumentation circuits are designed to fail safe. In the event of an electrical power failure, all circuits will fail to a safe condition and the reactor will be scrammed. Parallel neutron-flux-level monitoring channels are used to provide a cross check on the reactor power level.

E. OPERATING PERSONNEL AND PROCEDURES

A well trained, experienced staff will operate the reactor in accordance with approved procedures. The line of authority from division management to an individual operating crew is clearly specified (see Appendix B). The operating philosophy, training programs, qualifications of personnel, and the responsibilities of each individual in the operations unit will be described in detail in the addendum to this report.

Prior to any fuel handling, reactor operation, or major test changes, proposed designs and operating procedures will be reviewed by the appropriate committees of the Atomic International Reactor Safeguards Review Panel.

A comprehensive emergency procedure for safe evacuation of S8ER operating personnel will be integrated into the existing Santa Susana Radiological Emergency Plan. This procedure will comply with the requirements as established by "USAEC Manual, Chicago Operations Office, Part 0500, Health and Safety." Other emergency procedures will be provided as appropriate.

F. CONSEQUENCES OF CREDIBLE AND HYPOTHETICAL ACCIDENTS

To provide a means of evaluating the safeguards provided by the site-facility-reactor complex, several potential accidents were examined (see Sections VI and VII). The maximum credible accident (MCA) was determined to be a 76 Mw-sec nuclear excursion followed by cladding rupture and a release of primary NaK coolant and core-fission-product activity to the reactor containment vessel. The nuclear excursion was postulated to occur during reactor startup as a result of continuous insertion of reactivity at the maximum design rate of 2¢/sec. No melting of fuel or cladding occurs as a result of the incident. The incident assumes complete malfunction of corrective instrumentation and no remedial action by the operator. At the time of the incident the initial fission product inventory in the core is assumed to consist of 5.5×10^6 curies of gross gamma activity (1 sec decay) from 10,000 hr of full-power operation (600 kwt). For the purpose of dose calculations it was assumed that approximately 4.15×10^5 curies of activity (approximately 8% of the steady-state fission product inventory) are released to the containment vessel (including 3820 curies of I^{131}). Additional exposure from excursion-produced fission products was shown to be insignificant. Leakage from the containment vessel to the reactor room at a rate of 1.0%/day was assumed to occur as a result of the overpressure in the containment vessel. Exhaust of the reactor room atmosphere at a rate of 5000 cfm to the stack was assumed.

A summary of the radiation doses to individuals in the reactor room and control room following the MCA are shown in Table II-1.

TABLE II-1
TOTAL INTEGRATED DOSES TO INDIVIDUALS IN BUILDING DURING MCA

Time After Incident	Direct Radiation from Containment Vessel Dome		Radiation from Submersion in Activity Filled Reactor Room	
	3 ft from Dome r	Console r	Whole Body Dose r	Thyroid Dose* rem
1 min	0.2	0.01	0.003	6
10 min	1.2	0.05	0.02	73
1 hr	6.0	0.24	0.1	440

*Total integrated thyroid dose from all iodine isotopes at containment leakage rate of 1% per day. Personnel in the control room would not be exposed to this hazard.

Hazards to personnel outside of the building in a downwind direction will be due primarily to inhalation of iodine activity (direct radiation at 100 ft from the building is less than 0.01 r/hr at 1 min after the accident). These doses are shown in Table II-2. This table indicates what the doses would be if, (1) no

TABLE II-2
TOTAL INTEGRATED THYROID DOSE FROM ACTIVITY RELEASED FROM THE BUILDING UNDER UNFAVORABLE WEATHER CONDITIONS* (MCA)

Distance Downwind (meters)	Exposure Time and Dose (TID) [†]				
	1 hr (rem)	8 hr (rem)	1 day (rem)	1 week (rem)	2 weeks (rem)
<u>On-Site</u>					
100	0.92	7.2	22.0	130	220
500	0.21	1.7	5.0	30	50
1000	0.12	0.9	2.9	17	28
3000	0.05	0.4	1.2	7	12
<u>Off-Site</u>					
3220 [§]	0.005	0.04	0.12	0.7	1.2

*Fumigation conditions for on-site doses, lapse conditions for off-site doses. 1% per day containment vessel leakage rate.

[†]Total integrated thyroid dose from all iodine isotopes.

[§]Nearest residence.

plating out or absorption of iodine occurred in the containment vessel (initial release of 3820 curies of I^{131} from fuel during incident), (2) if the leak rate from the containment vessel did not decrease with time (constant leak rate of 1% per day at 68 psig assumed) and (3) if no steps were taken to decrease leakage from the vessel following the accident. [†]

A hypothetical accident was also evaluated which assumed that the fuel melts at the end of the core life and fission products are immediately dispersed into the reactor room and to the environment (4.15×10^5 curies released and all containment barriers voided). This accident is not credible and is presented for information only. Even with this incredible assumption, the total integrated lifetime doses at the nearest residence are only approximately 0.2 r whole-body gamma and 12 rem to the thyroid.

G. CONCLUSIONS

As a result of the enumerated safety features and accident study results, it is concluded that the reactor and installation are so designed, constructed, and sited that the S8ER can be operated safely without undue hazards to either operating personnel or the general population. Operating personnel located in the reactor room during the maximum credible accident would receive a maximum whole-body gamma dose of less than 1.3 r and a total integrated thyroid dose of 73 rem if they remained at a distance of 3 ft from the containment vessel for 10 min. A 10-min time to escape or perform necessary tasks should be more than adequate. Personnel in the control room would receive less than 0.3 r whole-body gamma dose in 1 hr. In 1 day the dose rate at the console drops to 0.08 r/hr.

Except for distances close to the building, the dose to personnel outside the building would be due primarily to iodine inhalation (maximum credible accident). Even if an individual remained in the center of the downwind plume at a distance of 100 meters from the facility for 8 hr, the total integrated lifetime thyroid dose would be only 7.2 rem. The corresponding dose to an individual at the nearest residence (3220 meters) would be 0.04 rem. In 2 weeks an individual

[†]Emergency action to decrease leakage will be described in the Addendum to this report.

at the nearest residence would receive a total lifetime dose of 1.2 rem if it is assumed that the unfavorable weather conditions prevailed throughout this period.

Even under the most adverse circumstances or faulty operation imaginable (hypothetical accident), the doses to the nearest off-site residence would be only 0.2 r whole-body gamma and 12 rem to the thyroid.

III. SITE AND FACILITY DESCRIPTION

The S8ER test facility was originally constructed in 1959 for the 50-kwt SNAP 2 Experimental Reactor Test.³ Following satisfactory completion of the SNAP 2 test in 1960,² the reactor and associated test equipment were removed from the building. In 1961, improvements and modifications were made to the facility and equipment to enable safe operation of the facility under a similar testing program with the proposed 600-kwt SNAP 8 reactor. Facility construction is based on normal practice and is designed in accordance with the applicable portions of the AEC Design Manual, Volume 6000. The "Uniform Building Code" (UBC), 1958 Edition, as published by the Pacific Coast Building Officials Conference, is the basic code governing the design.

A description of the site, environment, utilities, facility and the associated equipment and services are discussed in this section. Figure III-1 shows the location of the facility in the Atomics International Nuclear Development Field Laboratories area and Figures III-2 through III-8 illustrate the design of the facility.

A. SITE, ENVIRONMENT, POPULATIONS AND UTILITIES

1. Site and Environment

The S8ER site is located in the Atomics International Nuclear Development Field Laboratories area which is within the limits of North American Aviation's Field Test Area in the Simi Hills of Ventura County, California. Appendix A of this report discusses the general site environmental factors such as meteorology, hydrology, seismology, and off-site population aspects of the Field Laboratories and Test Area.

Figure III-1 shows the location of the S8ER facility within the Atomics International Nuclear Development Field Laboratories area and its relation to other installations in the local area. The S8ER facility is situated near the northwest boundary of the Field Laboratories area within an area known as the SNAP Complex. Table III-1 shows the distances from the facility to surrounding installations in the area.

Figure III-1. AI Nuclear Field Laboratories Area

TABLE III-1
FACILITIES IN THE VICINITY OF THE S8ER TEST FACILITY AND
THEIR ESTIMATED NORMAL DAYTIME POPULATION

Bldg No.	Building Name	Population	Distance from S8ER (ft)
003	Engineering Test Bldg	20	1620
005	Organic Lab	32	600
006	Sodium Lab	30	380
009	SGR-OMR Critical Experiments Lab	15	1790
010	SNAP Experimental Reactor Test Installation	5	0
011	Warehouse and Offices	38	950
012*	General Critical Bldg	-	100
013*	Component Assembly and Performance Test Bldg	-	210
019*	Flight Systems Nuclear Test Bldg	-	350
020	Components Development Hot Cell	26	1700
021	R/A Waste Decontamination and Office	6	700
022	R/A Waste Storage Vault	6	850
024	SNAP Environmental Test Facility	10	380
025	SNAP Experimental Lab	15	150
026	Large Component Test Loop	7	300
027	Non-nuclear Vibration and Shock Test Bldg	12	600
028	Shield Test Experiment Facility	5	300
030	AE-6 Counting Room and Workshop	3	1660
032	Thermal and Vacuum Environmental Test Bldg	10	720
035	AE-6 Office Annex	15	1710
036	SNAP Office Bldg	45	700
037	SNAP Office Bldg No. 2	48	700
038*	SNAP Office Bldg No. 3	-	620
039	AEC Office Bldg	5	420
041	SRE Component Storage Bldg	2	1250
049	OMR Prototype Experiment	3	650
056	Prototype System Nuclear Test Bldg	12	600

TABLE III-1 (Continued)

Bldg No.	Building Name	Population	Distance from S8ER (ft)
059*	Ground Prototype Test Bldg (S8DS)	--	490
073	Reactor Kinetics Test Bldg	6	950
083	Reactor Kinetics Control Bldg	4	1240
093	AE-6 Reactor Building	12	1600
100	Epithermal Critical Experiment Lab	15	1480
143	Sodium Reactor Experiment	30	1400
153	Sodium Service Bldg	2	1480
163	Site Service Bldg	15	1530
343	Guard Bldg	3	1570
353	Organic Reactor Development Bldg	10	1820
356	HNPF Fuel Handling Test Facility	3	250
357	HNPF Sodium Pump Loop	3	200
363	Mechanical Component Development Bldg	10	1890
373	Experimental Development Bldg	16	1520
375	OMR Control Shelter Bldg	5	1800
383	Instrumentation Bldg	18	1160
453	AE-6 Fuel Handling Bldg	2	1250
623	Guard Bldg	1	920
625	Non-nuclear Components Storage Bldg	--	550
663	Experimental Test Structure	2	490

*Proposed or under construction at present.

2. Test Area Population Distribution (Personnel Under North American Aviation and Atomics International Control)

In the Field Test Area, extending up to 2-1/2 miles east of the facility, approximately 3000 persons are employed, approximately 450 of whom are AI employees working within a radius of 2000 ft from the facility. Table III-1 shows the distances from the facility to surrounding installations and the distribution of personnel with building structures at the site. Off-site populations

and distributions are shown and discussed in Appendix A. It may be noted here that the nearest residence is approximately 2 miles from the S8ER.

3. Surface Drainage

Grading and paving of the site is arranged to direct surface runoff into the Field Test Area drainage system and outfall described in Appendix A.

4. Utilities and Access

Utilities serving the Nuclear Development Field Laboratories area are discussed and shown in Appendix A (see Figure A-3).

Figure III-2 shows the utilities and services serving the S8ER facility. An incoming underground 4160-volt feeder furnishes the necessary power for all operations, lighting, and heating. Water for the site is derived from local wells via an 8-in. main and the 50,000-gal storage tank shown in Figure A-3. Sanitary wastes are routed to a central sewage system which serves the Field Laboratories area. Telephone lines are provided to the facility. Access to the site is limited by a security fence which is integrated into the overall SNAP complex fencing system. Vehicle and pedestrian access is through a guard-controlled gate at the east boundary of the SNAP complex. Emergency exit gates are provided in the fence near the S8ER facility.

B. BUILDING

1. General Construction

The S8ER building is a rigid frame structure with corrugated metal decking and siding, and blanket insulation. The building is 60 ft long by 24 ft wide with a 17-ft eave height. The north 6 ft of the structure were added under the modification program to expand the control room space. A main pedestrian entrance and small vestibule is provided at the north wall which permits access to either the control room or mezzanine area above the control room.

The building foundation is of spread footing and grade beam construction, with a 6-in. mesh-reinforced floor slab. The floor slab is tied into the column footings to develop the rigid frame horizontal reactions. Figures III-3, III-4, and III-5 show a floor plan layout and elevation views of the facility.

Figure III-2. S8ER Plot Plan

Figure III-3. S8ER Facility Floor Plan Layout

Figure III-4. S8ER Facility Section View Looking East

Figure III-5. S8ER Facility End Section Views

2. Control Room

The control room occupies approximately 415 ft² of the north end of the building. This area houses control and readout instrumentation for the reactor, process instrumentation readout, electrical switchgear, and facility protective signal alarms. Observation of the reactor room is provided by a wire glass (double-strength) window in the SE wall. Exterior access to the control room is through the main entrance and a door in the wall separating the vestibule and control room.

Process and reactor instrumentation wiring is routed to the reactor room through covered floor trenches. Power and control wiring is routed from the control room to the reactor room in overhead cable trays.

3. Change Room

The change room is located adjacent to the control room and provides the only interior access to the reactor room within the building. This room is 14 ft long by 8-1/4 ft wide with an 8-ft ceiling. A lavatory, toilet, shower, and locker space are provided. A hand and foot counter is located in this room for personnel monitoring.

4. Mezzanine

The mezzanine area is located along the north and west walls of the building above the control room and change room. The mezzanine provides supplementary storage and office space for operations and health physics functions. Access to the mezzanine is by a spiral stairway from the vestibule. Observation of the reactor room is provided through a wire glass window in the SE wall.

5. Reactor Room

The reactor room occupies the south 34 ft of the building. This room provides a maintenance work area and space for miscellaneous switchgear and process equipment. A shielded containment vessel, primary system vault, and a secondary system equipment pit are located below floor level. Access to the shielded containment vessel and primary system vault is through removable shield plugs and sealed covers. These areas will be discussed separately below.

The reactor room is separated from the remainder of the building by a steel stud and dry wall partition. Material handling capability in this room is

provided by a hand-racked single-beam underhung bridge crane. A 7-1/2-ton motorized hoist and trolley and a 1-ton hand-operated chain hoist operate off the bridge. Either hoist may transfer from the bridge onto the monorail over the outside crane loading apron located south of the building.

Access to the reactor room from outside the building is through a 12 by 16 ft door in the south wall or through a 10 by 12 ft door in the west wall. A pilot door in the 12 x 16 ft door is provided for pedestrian and emergency use. Access from the control room into the area is normally routed through the change room, as previously described.

C. VAULT COMPLEX

The vault complex in the S8ER facility is composed of three separate enclosures: (1) shielded containment vessel, (2) primary system vault, and (3) secondary system equipment pit. The shielded containment vessel and primary system vault are contained in an integral concrete enclosure. The secondary system equipment pit is an open-topped concrete structure which is covered with bridge grating at the reactor room floor level and is separated from the primary system vault by 2-1/2 ft of compacted earth and the concrete walls.

The entire vault complex is provided with a foundation drainage system, consisting of circuits of perforated metal pipe surrounded by a gravel fill. The system drains into a pipewell sump located to the east of the building where the intercepted ground water may be monitored. The water may be either pumped into a tank for controlled disposal if contamination is detected or discharged to the site surface drainage system if no hazard exists. Exterior surfaces of the secondary system equipment pit are waterproofed with a membrane system.

1. Shielded Containment Vessel

The shielded containment vessel consists of a carbon steel pressure vessel embedded in concrete. The upper head of the vessel is removable and extends above the reactor room floor (see Figure III-6). The containment vessel houses the reactor, a shutdown shield, the reflector drives, and an operating shield. The reactor is located in a compartment below the shutdown shield and the controls are located in the upper compartment between the shutdown shield

Figure III-6. S8ER Reactor Installation and Containment Vessel

and operating shield, where low temperatures and radiation levels prevail. In order to minimize the possibility of NaK fires and hydrogen-oxygen reactions, the free space in the containment vessel is filled with helium and maintained at a positive pressure of 3 in. of water during reactor operation.

The exterior surface of the containment vessel is wrapped with two water cooling circuits to cool the concrete surrounding the containment vessel. Temperature buildup in the shutdown shield is limited by outside air which is pumped through cooling coils embedded in the concrete of the shutdown shield. Both systems cool the atmosphere above and below the shutdown shield. The cooling systems will be described in detail later.

A neoprene gasket is used in the containment vessel top flange assembly to retain helium in the vessel. All concrete in the containment vessel (shutdown and operating shield) is encased in steel to prevent any release of water vapor to the inert atmosphere. Both shield assemblies may be vented (valves normally closed) to prevent pressure buildup and damage to the assemblies. Separate gages are provided for each assembly to indicate pressure rises. The shutdown shield vent line is also equipped with a high-pressure alarm and is routed to the radioactive gas and vent system since it may contain small quantities of induced radioactivity.

The primary coolant inlet and outlet lines exit the containment vessel through two 6-in. sleeves which connect the upper compartment of the containment vessel with the primary system vault. In order to maintain containment vessel integrity in the event of pressure rises, the sleeves are seal-welded to the containment vessel wall and the coolant lines are seal-welded to the sleeves by means of seal caps. Smaller penetrations of the containment vessel include: inlet and outlet lines (2) for vessel atmosphere; shutdown shield air cooling lines (6); shield vent lines (2); and electrical and instrument conduits (4). All lines and conduits are seal welded to the containment vessel wall. Electrical and instrument conduits are sealed by high pressure electrical fittings (Pyle-Star-Line Environmental Series W electrical connectors). The lines external to the containment vessel are routed in covered floor trenches to provide physical protection and to minimize the possibility of damage. Associated equipment is located in isolated areas and is protected by appropriate guards.

The carbon steel pressure vessel was designed for a pressure of 305 psig and was hydrostatically tested at 456 psig in accordance with the ASME Boiler and Pressure Vessel Code for unfired pressure vessels containing lethal substances. The leakrate from the containment vessel was determined before and during SER operation with pressure up to 68 psig. The observed leakrate at 68 psig and with a free volume of 150 ft³ in the vessel (shield not in place during testing) was 0.1% per day. During S8ER operation the containment vessel will contain a free volume of approximately 50 ft³ and the corresponding leakrate would be 0.3% per day at 68 psig.

The fission products are normally contained in the fuel cladding. In case of a fuel element rupture, fission products may spread into the coolant stream where they would be contained within the core tank and primary cooling system. In the event of rupture of the core tank, the fission products would be contained by the containment vessel. In order to insure that no rupture of the system due to overpressure would occur outside the containment vessel, a rupture diaphragm which ruptures at 50 psig is connected to the primary coolant system in the upper compartment in the containment vessel. The 50-psig rupture pressure is well below the minimum rupture pressure (see page 70) for the primary system to insure that any rupture of the system from pressure buildup should release material only within the containment vessel (discharged NaK conveyed to lower compartment in containment vessel). Increased reliability of the pressure relief device against accidental rupture is obtained by use of a double rupture disc arrangement. As a result, the combination of the containment vessel and primary coolant system comprises the primary containment barrier against fission product release.

The containment vessel and primary coolant loop will be helium leak tested following completion of construction and modifications. In addition, the combined containment vessel and primary coolant loop will be leakrate tested prior to S8ER operation with all valves and penetrations in their normal operating condition (not blanked). A system leakrate of 1% per day at 68 psig will be considered acceptable under these testing conditions, although every effort will be made to obtain as low a leakrate as possible. The maximum credible accident will be based on this value (1% per day at 68 psig as described in Section VII).

2. Primary System Vault

The primary system vault is a shielded steel-lined room which contains the primary heat transfer system (including the EM pump, intermediate heat exchanger, surge tank, flowmeter, and all associated piping). The room is 9-1/2 ft long by 8 ft wide by 11 ft deep, with the ceiling of the room approximately at the same level as the reactor room floor.

The shielding walls below the reactor room floor level are made of ordinary, poured-in-place concrete. A minimum of 5 ft of concrete separates the primary system vault and containment vessel. Shielding above the reactor room floor level consists of blocks of high density (magnetite) concrete with a total effective thickness of 3 ft and a density of 6200 lb/yd³. The blocks immediately over the vault and on three working sides are removable. The fourth side, adjacent to the east building wall, is shielded by poured-in-place heavy concrete (magnetite as above).

Carbon steel plate 1/4 in. thick is anchored to the walls and floor of the vault. The ceiling of the vault consists of a removable metal cover (seal plate) located just below the concrete shield blocks. The seal plate is fabricated from 1/8-in. carbon steel plate, reinforced by a flat bar grid. Sealing of the plate to the vault liner is accomplished by seating a "knife edge" projection on the periphery of the plate against a silicone rubber gasket attached to the wall liner. Pressure for seating the plate is provided by 3/4-in. diameter bolts connecting the liner and plate. Power and instrument wiring, and valve operators are routed through sleeves to the secondary system equipment pit.

During normal operation, the vault will be filled with nitrogen at a positive pressure of 3-in. H₂O (~ 0.11 psig) to exclude oxygen. The vault is designed to withstand a nominal internal pressure of 3/4 psig, and recently was tested at 1.25 psig. Pressure relief for 3/4 psig or over is provided by a U-tube filled with oil, which vents to the stack. Although the vault is not designed to contain major coolant accidents, all penetrations are sealed to prevent loss of nitrogen from the vault, as shown in Table III-2.

The concrete surrounding the sides of the vault is water cooled by seven circuits of 1-in. diameter pipe attached to the outside surface of the wall liner. The vault atmosphere is cooled by a nitrogen-to-water heat exchanger (stream of vault nitrogen recirculated through external heat exchanger). Both of these systems will be discussed later in this section.

TABLE III-2
PRIMARY SYSTEM VAULT PENETRATIONS

Penetration	Method of Sealing
2-1/2-in. secondary coolant inlet to intermediate heat exchanger	Line passes through a pipe sleeve in the wall. A cap welded to the end of the sleeve has a hole cut in it to accommodate the line. A weld between the cap and the line completes the closure.
2-1/2-in. secondary coolant outlet from intermediate heat exchanger	
1-in. primary surge tank vent	
1-in. primary fill and drain line	
1-in. intermediate heat exchanger drain	
2-in. primary coolant reactor inlet	
2-in. primary coolant reactor effluent	
4-in. nitrogen supply (2 lines)	Line is welded to liner (externally self-contained system). An oil seal is provided which vents to the stack on overpressurization.
6-in. nitrogen return	
EM pump power leads	Electrical and instrument lines pass through pipe sleeves which are sealed with Chico A-5.
Pressure transmitter lines	
J-tube, thermocouples, flowmeters, and plugging meter wiring	Rod passes through a pipe sleeve in the wall. A cap welded to the end of the sleeve has a hole cut in it to accommodate the rod. O-rings fitted to the rod complete the closure between the rod and the cap.
Valve operators (5 rods)	
Primary vault cover	Vault cover plate edges rest on a 3/4-in. thick rubber seal. The cover plate is bolted down compressing the seal.

3. Secondary System Equipment Pit

The secondary system equipment pit is located at the south end of the reactor room and is separated from the primary system vault by 3 ft 9 in. of ordinary concrete and 2-1/2 ft of compacted earth. This pit is 16 ft long by 5-3/4 ft wide and has a clear depth of 7 ft 1 in.

The pit contains the secondary fill and drain tank, the primary drain tank, portions of the secondary piping, and valve operators for the remotely-controlled primary system. Local shielding is provided only for the primary drain tank which is located in the pit. Permanent shielding presently consists of 4 in. of lead on all exposed surfaces of the tank and drain lines. Additional shielding can be added if it is deemed necessary prior to or after transfer of NaK to the tank.

The pit is lined with 1/4-in. steel plate on the bottom and extending 12 in. up the walls. The pit is covered by an open bridge deck grating capable of supporting fork lift trucks. Valve operation is accomplished at the reactor room floor level above the grating.

D. EXTERIOR EQUIPMENT AREAS

Some equipment associated with the heat transfer systems and facility service systems is located on concrete pads outside the building. The major equipment and service systems at these locations are:

- Airblast heat exchanger (main heat transfer system)
- Vault nitrogen cooling system (primary system vault atmosphere)
- Shutdown shield cooling system (air compressors)
- Cooling water system (vault concrete cooling, etc.)
- Nitrogen and helium supply systems
- Electrical sub-station and emergency power unit
- Facility stack and R/A gas holdup tank.

The locations of these areas are shown in Figures III-2 and III-3. The service systems will be described below.

E. SERVICE SYSTEMS

In order to properly and safely operate the reactor and its heat transfer systems, certain supporting service systems are necessary. While portions of these systems are within the building, much of the equipment is located outdoors on concrete pads previously mentioned. Simplified P&I diagrams of the main service systems are shown in Figures III-7 and III-8. A brief description of these and other service systems associated with the facility is presented below.

1. Helium Supply Systems (see Figure III-8)

Helium is used in the facility for the following purposes:

- a) To purge the liquid metal systems prior to filling with NaK
- b) To transfer NaK to and from the heat transfer systems
- c) As an operating cover gas in the surge tanks of the primary and secondary NaK systems
- d) As an inert atmosphere for the reactor containment vessel.

Figure III-7. S8ER Service Systems P & I (Auxiliary Cooling Systems)

Figure III-8. S8ER Service Systems P&I (Gas Supply and Gas Vent Systems)

Three independent helium supply systems are provided to ensure adequate control and to provide isolation for systems servicing radioactive and nonradioactive areas and equipment.

One system consists of six 225 scf helium cylinders, a manifold, and 1/2-in. lines which distribute the helium to the primary fill tank, the primary surge tank, the primary drain tank, and the radioactive decay tank (contaminated system). A similar system, consisting of two 225 scf helium cylinders distributes helium to the secondary surge tank, and the secondary fill and drain tank (nonradioactive system). The third system consists of two 225 scf helium cylinders, a manifold and a 1/2-in. line which conveys helium to the reactor containment vessel (near top of vessel). An outlet line is provided at the bottom of the containment vessel to enable purging and venting of the vessel.

Each manifold is equipped with pressure control valves, pressure gages, low-pressure alarms, and pressure relief valves. The cylinders and manifolds are located outside the building on a concrete pad. Operation of the system is accomplished by manual control at the pad and/or from a control board in the reactor room.

Helium vented from the primary heat transfer system is routed to the radioactive gas decay tank where concentrations of radioactivity are determined prior to release to the stack. The secondary heat transfer system and the containment vessel atmosphere are vented directly to the stack.

2. Nitrogen Supply System (see Figure III-8)

Nitrogen is used in the facility to establish and maintain an inert atmosphere in the primary system vault. The system consists of six 225 scf nitrogen cylinders, a manifold and a 1/2-in. line which conveys the nitrogen to an area near the top of the vault. An outlet line is provided at the bottom of the vault to permit adequate purging prior to startup and after shutdown. The manifold is equipped with a pressure control valve, a pressure gage, low-pressure alarm, and pressure relief valve. The cylinders and manifolds are located outside the building on the same concrete pad as the helium supply system. Operation of the system is accomplished by manual control at the pad and/or from a control board in the reactor room.

3. Water Cooling System and Cooling Water Circuits (see Figure III-7)

Cooling water is used in the facility for the following purposes:

- a) To cool the concrete surrounding the containment vessel.
- b) To cool the concrete walls of the primary system vault.
- c) To provide cooling water to the aftercooler for the shutdown shield cooling system air compressor.
- d) To provide cooling for the vault nitrogen atmosphere cooling system.

The above functions are accomplished by a central water cooling system which cools and recirculates water to the remote cooling water circuits in the concrete or equipment cooling coils.

a. Water Cooling System

The water cooling system consists of an evaporative cooler, a cooling coil, a surge tank, two full capacity circulating pumps, a filter, and associated piping, valves, and instrumentation.

The evaporative cooler utilizes a raw water spray and forced draft air stream to remove heat from a cooling tube in the closed circuit cooling water circuits. Demineralized water treated with hydrazine to reduce scale buildup is used in the cooling water circuits. Cooling water recirculation through the closed cooling water circuits is provided by a centrifugal pump rated at 70 gpm at a head of 25 ft. A spare pump of like capacity and valving is provided.

Alarms are provided for (1) high temperature in the cooling water loop downstream of the evaporative cooler and (2) surge tank low level. The above cooling system components are located on a concrete pad outside the reactor building. The capacity of the cooler is 460,000 Btu/hr with a 20°F approach to a 70°F wet bulk temperature.

b. Cooling Circuits for Concrete Surrounding Containment Vessel and Vault

The cooling circuit which is utilized to cool the concrete surrounding the containment vessel consists of two parallel cooling coils attached to the

outer surface of the vessel (see Figures III-6 and III-7). The coils have a total combined length of 150 ft and are formed of 1/2-in. schedule 40 mild steel pipe. Cooling water flow is downward so that the coolest water is used to cool the area above the shutdown shield since the reactor controls and drive motors are located in this area. The design water flow in each circuit is 3 gpm (6 gpm total). Throttle valves are provided at the outlet of the circuit for manual control. A temperature indicator, a flow indicator, and a low-flow alarm are provided at the cooling circuit outlet. The bases for the design are inlet and outlet water temperatures of 90°F and 125°F, respectively. Temperature in the area above the shutdown shield will be kept below 200°F. The atmosphere temperature in the area below the shutdown shield will be kept below 400°F. The maximum concrete temperature (530°F) will occur on the core center plane approximately 8 to 10 in. from the containment vessel wall. The concrete shield surrounding the containment vessel is designed to permit operation under these conditions.⁴

The cooling circuit which is utilized to cool the concrete walls of the primary system vault consists of seven parallel coils attached to the outside of the steel vault liner (see Figures I-1 and III-7). The coils have a total combined length of about 1700 ft and are formed from 3/4-in. schedule 40 mild steel pipe. The design water flow in each coil is 0.44 gpm (3.1 gpm total). Each of the five lower coils is provided with valves and outlet temperature indication, and a flow indicator is installed to indicate the total flow through the five coils. The upper two coils are in parallel but are provided with a single throttle valve, temperature indicator, and a flow indicator at the circuit outlet. The bases for the design are inlet and outlet water temperatures of 90°F and 125°F, respectively. A maximum concrete temperature of approximately 140°F is anticipated.

c. Other Cooling Circuits

A 1-in. circuit supplies water to the cooling coil of the vault nitrogen atmosphere cooler. The bases for the design are inlet and outlet water temperatures of 90°F and 98°F, respectively.

A 1-1/2-in. circuit supplies cooling water to the air compressor aftercooler. The bases for the design are inlet and outlet temperatures of 90°F and 100°F, respectively.

Rated heat loads and flowrates for the entire cooling system are shown in Table III-3.

TABLE III-3
WATER COOLING SYSTEM HEAT LOADS AND FLOWRATES

Circuit	Heat Load (Btu/hr)	Flowrate (gpm)
Containment vessel	104,300	6.0
Primary vault	50,600	3.1
Vault N ₂ atmosphere cooler	45,100	11.2
Aftercooler	227,000	45.5
TOTAL	427,000	65.8

The entire closed circuit in the cooling water system contains approximately 100 gal of water.

4. Shutdown Shield Cooling System

The shutdown shield cooling system uses air to remove heat from the shutdown shield. The system is designed to maintain lead portions of the shutdown shield below 215°F and concrete portions below 460°F. (Two removable plugs in the inspection sleeves do not contain air-cooling coils. These plugs transfer internally generated heat to the main shield; consequently, their steady-state temperatures will be somewhat higher than the rest of the shield.) The system consists of two air compressors, two aftercoolers, two moisture separators, an air receiver, the cooling coils within the shield, connecting piping, valves, and appropriate instrumentation (see Figures III-6 and III-7).

The two air compressors are operated in parallel and provide 360 scfm of air at 125 psig. The compressors are two-stage units with air-cooled intercoolers. Each compressor is driven by a belt-connected, 50 hp electric motor. The compressed air flows through an aftercooler where it is cooled to about 100°F, then into the moisture separator, and finally into the air receiver tank. All of this equipment is located on a concrete pad about 30 ft SE of the building. A 2-in. carbon steel line transports the dried air to the building where connections are made to the three manifolds which supply the various cooling circuits within the shield.

A flow switch with a low-flow alarm is provided in the 2-in. supply line. The three manifolds on the air exit side of the shield are provided with manually operated flow control valves and flow indication instrumentation. The manifolds are then joined to a common 1-1/2-in. header and discharged to the building stack.

A description of the cooling coils and heat loads associated with the system are shown in Table III-4.

TABLE III-4
SHUTDOWN SHIELD COOLING SYSTEM CHARACTERISTICS

Coil Location	Copper Tubing OD and Wall Thickness (in.)	Linear Coil Length (ft)	Air Flow (scfm)	Heat Removed, Maximum (Btu/hr)
Center plug	3/8 x 0.035	25	19	1,635
40-in. plug	1 x 0.05	68	191	16,419
Coolant pipe sleeves (two in parallel)	5/8 x 0.035	46 (total)	114 (total)	4,905 (total)
Control drive shaft sleeves (six in parallel)	3/8 x 0.035	48 (total)	5 (total)	660 (total)
TOTAL			329	23,619

5. Vault Nitrogen Atmosphere Cooling System

The primary system vault atmosphere is cooled by a nitrogen-to-water cooling system located outside the vault (see Figure III-7). The system consists of a 6-in. nitrogen outlet duct leading from the vault, a tank containing a water cooling coil and blower, two 4-in. nitrogen return ducts, and control valves and instrumentation.

The tank housing the cooling coil and blower is mounted horizontally and is approximately 60 in. long by 24 in. in diameter. The equipment in the tank is arranged so that the nitrogen from the vault is drawn through the water cooling coil and is then returned to the vault by the blower. Two 6-in. glass inspection ports on the tank permit visual observation of the blower and cooling coil. The

tank is designed for an internal pressure of 15 psig and is located outside the building on a concrete pad. All metal joints are welded to either the ducts or vault liner to prevent leakage. One of the 4-in. return ducts is equipped with a flexible duct in the vault which distributes cool nitrogen to the EM pump coils.

A manual control valve is provided in each of the 4-in. ducts leading to the vault. Instrumentation consists of a temperature indicator in the return and supply ducts, a flow indicator in the return duct, a radiation detector and alarm in the supply duct, and a high-moisture alarm mounted downstream of the water cooling coil in the tank.

The blower is driven by a 3/4-hp motor and delivers 400 cfm at 4.5 in. water. The cooling coil has a capacity of 60,000 Btu/hr with a water flowrate of 15 gpm at a 90°F inlet and 98°F outlet temperature. The vault atmosphere is maintained at a temperature of 250°F or below.

The ducts leading from the vault to the concrete pad outside the building are appropriately routed (buried in concrete) to protect them from physical damage. Ducts and associated equipment on the pad are protected from physical damage by stanchions on exposed sides of the pad.

6. Building Ventilation and Air Conditioning System

The test facility is provided with ventilation, air conditioning, or heating in accordance with the specific requirements of the building areas. The systems are separated into two major areas served: the control room area and the reactor room area.

The control room is provided with year-round air conditioning consisting of a cooling unit, suspended from the ceiling within the control room, and electrical unit heaters. The condensing unit serving the cooling unit is located outside the building. Sufficient outside air is introduced to the room, through the cooling unit, to maintain a positive pressure in the area with respect to ambient, change-room and reactor-room pressures. A room temperature of 75°F is maintained year-round to ensure proper operation of the various instrumentation and electrical equipment and to provide working conditions consistent with alert reactor operations.

The change room is vented to the roof and is maintained at a slightly positive pressure with respect to the reactor room pressure. The reactor

room is ventilated by an air exhaust system. Air is drawn into the area through wall louvers and is exhausted with a fan which discharges into the 50-ft facility stack. The flowrate is 5000 cfm which provides 20 air changes per hour and maintains a negative pressure of 1/8-in. wg at all times. The exhaust fan and related ductwork are located outside the building on a concrete pad. Results of the analysis made to determine the release of radioactivity as a result of normal operation or accidents (see Sections VI and VII) indicate that filters are not required for the exhaust system.

7. Electrical Service

Power is supplied to the facility by an incoming underground 4160-volt feeder which leads to an outdoor concrete pad adjacent to the building. The incoming supply is stepped down to 480/277 volts by a 300-kva transformer and to 240 volts by a 150-kva transformer, both of which are located on the outdoor pad. Distribution throughout the building is accomplished by standard conduit and wiring practices (latest revision, National Electrical Code).

Lighting in the control room is accomplished by fluorescent fixtures. The reactor room, change room, and outdoor service areas are lighted by incandescent fixtures. Emergency lighting is provided during a power failure by Exide Model A battery lights located in all the rooms. The failure of normal power will automatically activate the system.

A 2-kw gasoline-powered generator provides emergency power for the stack monitor, area monitors, air monitors, and the extended range nuclear instrumentation channel. The generator is activated automatically upon an electrical power failure.

A local intercom system is provided which permits communications between the control room, reactor room, and the outside service area. One-way communication from the AI Industrial Security Control Center to the facility is provided by a speaker in the control room (tied into the site public address system). Emergencies, accidents, and evacuation directives are announced over this system.

Emergency power is not required to maintain shield cooling, since conduction of heat to the outside environment is sufficient to remove the small amount of heat generation after the reactor has scrammed.

8. Fire Protection Systems

No fire protection equipment employing water as a suppression medium will be stored or utilized in the facility. Fire detection and alarm is provided through local extensions to an existing central station site system. The control system is of the coded, electrically supervised type which receives alarm signals in positive succession and without interference in multiple signal circumstances. Well-trained fire personnel with suppression equipment are promptly dispatched from the central station in response to alarms. Local extensions are provided to the following equipment in the facility:

- 1 - smoke detector in reactor room
- 1 - smoke detector in the control room
- 1 - smoke detector in the electrical equipment area
- 1 - smoke detector in the change room
- 1 - smoke detector in the mezzanine
- 1 - key clock in the control room
- 1 - emergency telephone with direct tie-in to AI Industrial Security Control Center

The various smoke detectors will sound a local fire alarm horn in the control room.

The following manual fire suppression equipment is provided at the facility:

- 1 - 15 lb CO₂ extinguisher in the reactor room
- 1 - 30 lb Ansul extinguisher in the reactor room
- 1 - 15 lb CO₂ extinguisher in the control room
- 1 - 150 lb MET-L-X unit on the south pad
- 1 - 30 gal can of CaCO₃ for NaK fires

Auxiliary fire fighting equipment: This includes self-contained breathing apparatus, fireman's turnout clothing, and an emergency locker containing selected fire combat clothing and equipment for NaK fires.

A fire hydrant and safety shower are located just south-east of the facility.

9. Waste Collection and Disposal Systems

The production of radioactive wastes as a result of the test program is expected to be quite small; therefore, major waste collection or processing systems are not provided for the facility.

Vent lines to the stack are provided from the containment vessel and primary system vault spaces and from those service systems capable of producing airborne radioactivity. A 10-ft³ gas holdup tank and vacuum pump system is provided to permit holdup and monitoring of gases vented from the primary heat transfer system following shutdown. The tank is vented to the stack.

Drain lines are provided from the change room, sink, shower, and water closet to the central sewage system which has provisions for monitoring and handling low levels of radioactive liquid wastes. However, activity will not be deliberately disposed into the system. The change room floor and floor drain are designed to prevent leakage or overflow water to other building areas (dikes and slope of floor prevent overflow). To minimize the possibility of NaK-water reactions, no water connections are provided in other areas of the building. However, a floor drain is provided in the floor trench in the reactor room to remove water which might accidentally get into the area. The drain discharges to a pipewell sump, located just outside the building. The sump is part of the subsurface drainage system described previously. The sump is routinely monitored and may be pumped out if activity is detected.

10. Radiation Monitoring Systems

A 4-station remote area monitoring system is provided with detectors in the control room, change room, reactor room, and primary system vault. All 4 detectors indicate in the control room in view of the reactor operator. Visual and audible alarms are provided at the detector locations and in the control room to warn personnel if radiation levels exceed preset levels.

A stack monitor is provided to continuously monitor the stack effluent for gaseous activity. Monitoring is accomplished by drawing a representative sample of the stack effluent through a particulate filter and a gas chamber. Activity in the chamber is detected by a GM tube connected to a countrate meter and a strip chart recorder. Activity levels are continuously indicated and

recorded and visual and audible alarms are activated if predetermined levels of activity are exceeded. The filter is removable and may be analyzed periodically to determine accumulated activity. In addition, stack samples will be collected on a periodic basis by Health and Safety and analyzed in sensitive laboratory counting instruments to ensure that significant particulate activity is not being released to the environment.

A self-contained and semiportable airborne particulate monitor (constant air monitor) is provided for monitoring the atmosphere in the reactor room. The monitor may be moved to other areas where potential hazards may exist. The monitor is equipped with appropriate alarms to warn personnel of the existence of hazardous conditions.

A radiation monitor is provided to monitor the vault nitrogen atmosphere for the presence of radioactivity. The system consists of a GM tube, ratemeter, and speaker, and an annunciation and alarm circuit. The GM tube is located on the nitrogen cooling system circulating line in the reactor room. Any increase in radioactivity in the vault atmosphere will be detected by the monitor and warn operating personnel if preset levels are exceeded.

Two sirens, one mounted in the control room and a second one in the reactor room, are part of the AI radiological emergency plan. These sirens are to warn personnel of existing high-radiation areas. The sirens are actuated by other detectors in nearby buildings or manually from the central Industrial Security Control Center.

A normal complement of portable radiation survey meters and counting equipment is stored in the building. Additional equipment may be obtained from the central health physics office if necessary.

11. Reactor Operating Lights

Lights with yellow lenses and rotating mirrors are provided at each end of the building roof to inform personnel outside the building when the reactor is in operation. Two red lights, one in the control room and one in the reactor room, alert personnel of reactor operation. Control for the lights is initiated from the control room.

12. Fuel Unloading

Equipment and procedures to be used in fuel unloading operations will be described in a separate report.

13. Isolation of NaK and Water Systems

Particular attention was given in the design of the facility and service systems to prevent interaction between NaK and water systems. Control over water used in the change room was described in Section III-E-9. The only other source of water in the facility is contained in the cooling water system and the associated cooling water circuits (see Section III-E-3 and Figure III-7). The evaporative cooler and the surge tank and cooling water pumps which circulate demineralized water in the closed circuit cooling water loops are located on a concrete pad outside the building. Any leaks in this portion of the system would drain from the pad to the ground and surface drainage system. The closed circuit cooling water loops contain a total inventory of 100 gal of water. The water which is used to cool the concrete surrounding the containment vessel is in a piping loop located outside the steel pressure vessel; therefore, the possibility of interaction between NaK and water in this area is extremely remote. Similarly, the water which is used to cool the concrete surrounding the primary system vault is contained in pipes which are located outside the steel vault liner. The cooling water loop used to cool the air compressors (on the pad south of the building, Figure III-2) is located entirely outside the building. The cooling water loop used to cool the vault nitrogen is also located outside the building. Any leak in this system would be detected by the moisture detector in the vault nitrogen cooling system recirculation duct.

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IV. REACTOR, CONTROL, AND HEAT TRANSFER SYSTEM

The S8ER reactor is a compact nuclear reactor designed to operate for 10,000 hr at a power level of 600 kwt. The reactor consists of a homogenously combined uranium fuel and hydrogen moderator with movable beryllium reflectors for control and a liquid NaK as the heat transfer medium. The system is a direct extension of the technology developed and tested for the lower power SNAP2 reactors.^{2,3,5} The location of the reactor, with respect to other systems and the facility, was illustrated in Figure I-1. Design and operating characteristics are summarized in Table IV-1.

A. REACTOR

Figure IV-1 shows an isometric view of the reactor and external reflector assembly.

1. Fuel Elements

The reactor contains 211 fuel-moderator elements. Each element consists of an individual fuel rod, a hydrogen diffusion barrier containing burnable poison, exterior cladding, and grid plate indexing pins. (See Figure IV-2.)

The rods contain the enriched (93.15% U^{235}) fuel and hydrogen moderator in the form of a solid zirconium-uranium alloy (10% uranium) hydrided to a hydrogen density of 6×10^{22} atoms/cc. The rods are 0.532 in. in diameter and 14.0 in. long. The hydrogen diffusion barrier is a 0.003-in. -thick ceramic coating (AI-8763D) which is applied on the inside surface of the cladding to prevent the loss of the hydrogen moderator. A burnable poison (3.34 mg Sm_2O_3 per linear inch of fuel element) is added to the ceramic coating, to compensate for reactivity loss due to buildup of fission-product poison. The cladding consists of 0.540 in. ID Hastelloy N tubing with a 0.010-in. wall thickness. The cladding envelope is completed by Hastelloy end caps which are seal-welded to the tubing. The manufacturing process ensures that a continuous diffusion barrier is provided on the inside surface of the cladding. The finished fuel element is 0.560 in. in diameter and 14.47 in. long (excluding the 0.140-in. -diameter positioning pins located at the ends of each element). The 211 elements are arranged in a triangular pattern on 0.570-in. centers. The assembly results in a hexagonal right cylinder, with slightly chopped corners, approximately 8 in. across flats and 14.47 in. long.

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TABLE IV-1
S8ER DESIGN DATA SUMMARY

Design Parameters

Design power level	600 kw
Life	10,000 hr
Primary coolant	NaK 78 eutectic
Secondary coolant	NaK 78 eutectic
Number of fuel elements	211
Fuel loading	6.56 kg of U enriched 93.15%
Reflector	3 in. nominal thickness Be
Control	6 rotating control and safety drums

Fuel Elements

Composition	Hydrided Zr-U alloy
Hydrogen concentration	6×10^{22} atoms/cc
Alloy	90 wt % Zr - 10 wt % U
Cladding (Hastelloy N)	0.010-in. walls and 0.250-in. end caps
Fuel rod diameter	0.532 in.
Fuel rod length	14.0 in.
H ₂ diffusion barrier	0.003-in.-thick ceramic coating on inside of cladding surface
Burnable poison	48 mg of Sm ₂ O ₃ per element
Maximum cladding temperature at design power level	1440°F
Maximum core temperature at design power level	1542°F

Reactor Core

Core vessel size	9.34 in. OD by 21 in. nominal height
Core vessel material	316 SS
Core vessel wall thickness	1/16 in.
Upper grid plate material	316 SS
Upper grid plate diameter	9-3/16 in.
Upper grid plate thickness	11/32 in.
Upper grid plate coolant passages	420 5/32-in. -diameter holes

TABLE IV-1 (Continued)

Lower grid plate material	Hastelloy C
Lower grid plate diameter	9-3/16 in.
Lower grid plate thickness	5/16 in.
Lower grid plate coolant passages	420 1/8-in.-diameter holes
Internal reflector	18 BeO filler pieces
Internal reflector cladding	Hastelloy N
<u>Control and Safety Elements</u>	
Number of elements	6
Material	Beryllium
Length	14.5 in.
Drum radius of curvature	4.69 in.
Nominal thickness	3 in.
Weight of each control element (including support and shim)	26.5 lb
Radial bushing clearance (self aligning)	0.004 to 0.006 diametral
Friction torque	10 in. -lb
<u>Control Element Drives</u>	
Drive means	1 reversible a-c motor per control element (only one element can be inserted at one time)
Element rotation rate	0.055 rpm (2¢/sec)
Element rotation range	105°
Time to rotate full range	319 sec
Coast rotation	0.033° (0.2¢)
Rotation stops	Limit switches and mechanical stop
Overload protection	Slip clutch
Drive system backlash	None - taken out by scram spring
<u>Scram System</u>	
Scram power	Torsional spring
Scram torque	30 in. -lb total 20 in. -lb for acceleration
Scram activation	De-energizing of electromagnetic clutch

TABLE IV-1 (Continued)

Scram time (including 0.04-sec clutch release)	13° (2.50) = 0.169 sec, 85° = 0.342 sec, 105° = 0.405 sec
Number of elements scrambled	6
Number of elements required for shutdown	4
Scram velocity (maximum)	8.22 rad/sec
Scram acceleration (maximum)	27.3 rad/sec ²
Scram snubber	Pneumatic cylinder
Snubbing travel	20°
Deceleration during snub (maximum)	200 rad/sec ²
<u>Position Indication</u>	
Coarse	
Type	Potentiometer and vertical scale voltmeter
Number of elements sensed	6
Number of elements indicated	6 simultaneously
Accuracy of readout	±5° ~ ±30¢
Precision	
Type	Precision sensor and readout
Number of elements sensed	6
Number of elements indicated at one time	Active element plus any one of other five
Accuracy of readout	±0.070° = ±0.4¢
Type of readout	Digital lights
<u>Safety Circuit Trips</u>	
High neutron flux level (3 channels)	130% of design power level
Short reactor period (1 startup and 2 intermediate channels)	3 sec
Flux channel malfunction	2 high level channels
Low primary coolant flow	30% of design flow
Low primary coolant surge tank level	12 in.
High primary coolant outlet temperature	1375 °F

TABLE IV-1 (Continued)

Containment Vessel Pressure	15 psig
Earthquake	Grade VI on Modified Mercalli scale
High Primary System Pressure	40 psig
Manual pushbutton	
<u>Startup Source</u>	
Type	Po-Be
Size	0.5 in. diameter by 2 in. long
Activity	15 curies
Neutron emission	5×10^7 n/sec
Location	Outside of core vessel at top
<u>Primary Coolant</u>	
Flow rate	140 gpm
Reactor inlet temperature	1100°F
Reactor inlet pressure	38.7 psia
Reactor outlet temperature	1300°F
Reactor outlet pressure	36.8 psia
NaK inventory	315 lb
<u>Secondary Coolant</u>	
Flow rate	140 gpm
Heat exchanger inlet temperature	1050°F
EM pump inlet pressure	30.4 psia
Heat exchanger outlet temperature	1250°F
EM pump outlet pressure	54.5 psia
NaK inventory	622 lb
<u>Nuclear Characteristics</u>	
Lattice spacing	0.57 in. center-to-center
Mean prompt neutron lifetime	6.7 μ sec
Effective delayed neutron fraction	0.0077
Median fission energy	0.21 ev
Thermal flux	2×10^{12} n/cm ² /sec

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TABLE IV-1 (Continued)

S8ER REACTIVITY INVENTORY BASED ON
10,000 hr OPERATION AT 600 kwt

<u>Item</u>	<u>Reactivity Estimate (% $\Delta k/k$)</u>
1. Xe ¹³⁵ (equilibrium)	1.3
2. Sm ¹⁴⁹ ($\simeq$ equilibrium)	0.9
3. U ²³⁵ burnup	0.4
4. Other fission products	0.8
5. Cold to power	<u>1.6</u>
6. Total excess required	5.0
7. Initial unpoisoned excess (3-in. reflector; in vault)*	8.6
8. Sm ₂ O ₃ worth	<u>-2.6</u>
9. Initial poisoned excess	6.0
10. Reactivity available for H loss, contingencies†	3.6
11. Additional shim worth (beyond nominal 3 in.)	2.5
12. Control capability§ (six 3-in. drums; 0° to 105°)	14.9
13. Individual drum worth	2.7
14. Initial shutdown margin**	8.9

*The reflective properties of the reactor vault are expected to add 1.1% to the free space k_{eff} of 1.075.

†Deduced from Items 6 and 7.

§This value has been calculated to be less than six times the worth of an individual drum by 1.3% because of interaction effects.

**Deduced from Items 9 and 12.

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Figure IV-1. S8ER Reactor Assembly View

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Figure IV-2. S8ER Fuel Element

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Fuel element design characteristics are summarized in Table IV-1.

2. Grid Plates, Internal Reflectors and Coolant Baffle Plate

In addition to the fuel elements, the core contains an upper and lower grid plate, internal reflector filler pieces, and a coolant baffle plate (illustrated in Figure IV-3).

The grid plates support and position the fuel elements and internal reflector filler pieces in the core. The lower grid plate (inlet plenum grid plate) is a 9-3/16-in. diameter and 5/16-in. thick disc made of Hastelloy C plate. The upper grid plate (outlet plenum grid plate) is a 9-3/16-in. diameter, 11/32-in. thick disc made of 316 stainless steel plate. Both grid plates are pierced by 211 holes of 0.144-in. diameter, spaced in a triangular array on 0.570-in. centers. These holes are used to position and hold the fuel elements in the core. The lower grid plate is also pierced with 420 holes of 1/8-in. diameter, arranged in a hexagonal pattern superimposed on the triangular lattice of the fuel-element-positioning holes. The upper grid plate is provided with 420 similarly spaced holes, which are 5/32 in. in diameter. These holes permit the coolant to pass through the grid plates and around the fuel elements. In addition to the above holes, 30 holes are located on the periphery of each grid plate for positioning and holding 18 internal reflector filler pieces. Six holes are provided on the periphery of each plate for tie rods, which are used to assemble the core components and anchor the core to the reactor vessel.

The 18 internal reflector filler pieces occupy the space on the periphery of the core between the hexagonal fuel element pattern and the cylindrical reactor-vessel shell. These internal reflectors are made of BeO clad with 0.010 in. Hastelloy-N, and extend the full length of the core between the upper and lower grid plates.

A coolant baffle plate is located in the inlet plenum of the reactor core vessel, 5/8 in. below the lower grid plate and just above the top of the coolant inlet line. The plate is a 9-1/16-in. diameter disc of type 316 stainless steel, 1/16 in. thick. A pattern of openings in the plate has been established empirically by the SNAP 8 core hydraulic experiments to adjust the coolant flow profile through the core to a cosine shape with a peak-to-average-velocity ratio of 1.45, matching the calculated radial power distribution. This coolant flow distribution

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Figure IV-3. S8ER Core and Reflector Section and Plan View

is necessary to minimize the fuel-element cladding and hydrogen-barrier-coating temperature during full-power operation.

3. Core Vessel and Assembly

The core and core-structure components, (i. e., fuel elements, grid plates, internal reflectors and flow baffle plate) are integrated into a complete package by means of six vertical tie rods at the circumference of the core. The core assembly is fixed to the core vessel by the lower end of the tie rods which screw into brackets attached to the inner circumference of the lower plenum head of the core vessel. The core vessel is constructed of a cylinder of type 316 stainless steel, 9-3/16 in. in ID and approximately 15-1/2 in. in length, with a 1/16-in.-thick wall. The lower end of the cylinder is closed with a reversed dish, type 316 stainless steel head, 1/8 in. thick and approximately 4 in. deep. Coolant enters the cylindrical portion of the lower head via a side inlet nozzle and a 2-in. stainless steel pipe. The upper end of the vessel is closed by means of a 1/4-in.-thick, type 316 stainless steel cylinder (transition cylinder) with the lower end welded to the top of the reactor vessel wall and the upper end to the bottom of the shutdown-shield assembly above the core. This cylinder serves as the upper coolant plenum, with coolant discharging through a side nozzle and a 2-in., stainless steel pipe. To simulate the flight configuration, a dished core-vessel head is located in the plenum above the core. A 2-9/16-in.-diameter hole is located in the center of the dished head, which serves as the coolant exit port. Seven thermocouples are located in the plenum below the dished head to measure temperature distributions in the coolant leaving the core. Three thermocouples are located in the exit port to measure bulk-coolant exit temperature.

A summary of the reactor design characteristics is shown in Table IV-1.

B. REFLECTOR AND CONTROL SYSTEM

1. Reflector and Support

The core vessel is surrounded by 3 in. of beryllium (nominal thickness of equivalent annular cylinder) which forms the external radial reflector. The reflectors are approximately 14.5 in. long (approximately same as fuel element length) and are divided into six stationary and six movable sections. The portions

of the reflector which form the six movable sections (control drums) are segments of right circular cylinders with the axis outside the reflector assembly. Upper and lower bushings (shown on Figure IV-3) are located on the axis of each segment, permitting rotation of the control drums. Reactor control is accomplished by rotation of the control drums toward or away from the core, consequently changing the neutron economy and reactivity of the core. The control drums are also used as safety elements to scram the reactor. Removable shim plates are provided on the control drums to permit adjustment of the control drum worth prior to reactor startup.

The reflectors are supported by two stainless-steel support structures which are attached to a flange on the transition cylinder which connects the core vessel to the shutdown shield. The support structures are also pinned to brackets attached to the bottom of the core vessel to provide radial support. The movable control drums are connected to the support structures by means of the upper and lower bushings.

The bushings are of the self-aligning type, which permit operation with limited misalignment due to thermal expansion. The upper bushings carry both radial and thrust loads. The lower bushings carry only a radial load.

2. Drives

The control mechanisms (shown in Figures IV-4 and IV-5) which position and scram the control drums are located above the shutdown shield; each control drum has its individual drive system. The major components of the control system are: a drive shaft, an electromagnetic clutch, a scram spring, a reversible a-c motor, a pneumatic snubber, and a position indicator.

Each drive shaft is ball-bearing mounted in a removable sleeve housing mounted in the shutdown shield. The connection between this drive shaft and the control drum pivot shaft is made via two torsionally rigid, flexible couplings and an intermediate shaft. This arrangement allows relative motion between the drive shaft and the control drum due to differential thermal expansion of components, while maintaining angular register. A 1/32 in. minimum clearance (at operating temperatures) is provided between the drive shafts and the control rod sleeves in the shutdown shield to ensure clearances for operation. Clearances for thermal expansion and misalignment have been allowed, using conservative assumptions.

- 1 SNUBBER ASSEMBLY
- 2 OUTLIMIT SWITCH
- 3 DRIVER GEAR
- 4 SCRAM CLUTCH ASSEMBLY
- 5 SCRAM STOP ARM
- 6 "IN" ADJUSTING STOP
- 7 "IN" LIMIT SWITCH
- 8 LIMIT SWITCH ADJUSTING ACTUATOR
- 9 POSITION INDICATOR TRANSMITTER
- 10 SCRAM SPRING
- 11 DRIVE WORM
- 12 DRIVE SLIP CLUTCH
- 13 DRIVE MOTOR

Figure IV-4. Control Drive and Scram Mechanisms (Assembled View)

1 POSITION SENSOR
ELECTRICAL CONNECTOR	25 BRG RETAINING PLATE
2 TRANSMITTER MOUNTING
CLAMP & BOLT	26 BRG RETAINING RING
3 POSITION INDICATOR
TRANSMITTER	27 DUPLEX DRIVE SHAFT BRG
4 TRANSMITTER INPUT
SHAFT	28 SCRAMSTOP ARM LOCK KEY
5 MOUNTING BOLT	29 SCRAM CLUTCH LOCK KEY
6 TRANSMITTER SUPPORT	30 CONTROL DRUM DRIVE SHAFT
7 TRANSMITTER FLEXIBLE
COUPLING	31 DRIVE SHAFT BRG HSG
8 SCRAM SPRING	32 MOUNTING BOLT
9 LIMIT SWITCH ADJ.
ACTUATOR	33 "IN" LIMIT SWITCHES
10 ARM BRG	34 "IN" ADJ. STOP
11 SCRAM STOP ARM	35 MOUNTING BOLT
12 CLAMPING BOLT	36 STOP BRACKET
13 ELECTROMAGNETIC
SCRAM CLUTCH BODY	37 LEADS
14 LEADS	38 DRIVE MOTOR
15 SCRAM CLUTCH
ENGAGING TEETH	39 ATTACHING BOLT
16 SCRAM CLUTCH
ARMATURE	40 SLIP CLUTCH
17 ATTACHING BOLT	41 DRIVE WORM
18 LOCATING PIN	42 DRIVE SUPPORT
19 SCRAM CLUTCH
ADAPTER	43 SCRAM SPRING ANCHOR
20 SCRAM CLUTCH SPACER	44 SNUBBER RELIEF PADS
21 DRIVE GEAR BRG	45 SNUBBER BLEED ADJ.
22 DRIVE GEAR	46 SNUBBER PRESS TEST
PORT
23 ATTACHING BOLT
LOCKNUT	47 "OUT" LIMIT SWITCHES
24 DUST SHIELD	48 SNUBBER PLUNGER
& OUT STOP
	49 SNUBBER ASSY
	50 MOUNTING BOLT

Figure IV-5. Control Drive and Scram Mechanisms (Exploded View)

The drive shaft is keyed to an electromagnetic, rotary, scram clutch (Figure IV-4, Item 4) and to a scram drive arm (Figure IV-4, Item 5). It is also connected by a flexible coupling (with zero backlash) to the shaft of a position indicator transmitter (Figure IV-4, Item 9). Thus, the indicator senses the angular displacement of the control drum directly. The inner end of a flat, spiral torsion scram spring (Figure IV-5, Item 8) is connected to the scram drive arm (Figure IV-5, Item 11) so that the spring torque always tries to rotate the reflector toward the scram position. The solenoid portion (Figure IV-5, Item 13) of the scram clutch is keyed to the drive shaft; the armature section (Figure IV-5, Item 16) is attached to a worm gear (Figure IV-5, Item 22) meshed with the drive worm. The drive worm is connected through a torque limiting coupling to a reversible a-c drive motor.

The connection between the two sections of the scram clutch is made with crown tooth face gears, providing a creepless engagement. When the electromagnet is de-energized, the connection between the two halves of the clutch is broken, and the force of the scram spring drives the reflector to the full-out position. With the clutch energized, the reflector can be positioned by operating the drive motor. Because nonreversible worm gearing is used, the scram spring cannot drive the drive motor. However, the force of the scram spring on the drive worm eliminates any backlash in the worm gearing.

Dual, electrical, limit controls and physical stops limit the travel of the reflector in both the "in" and "out" directions. A pneumatic snubber is utilized to slow down the control drum at the end of the scram rotation. Mechanical keys are provided to lock each reflector shaft in the least reactive position during maintenance work. These keys fit milled slots on the shaft and are bolted to the top of the shutdown shield. Electrical connections at the control console allow only one of six drums to be positioned at any time. Actuation of a set-back switch, however, will drive all six drums out simultaneously. Additional circuitry is provided to drive any control drum to its full "out" position in case the spring fails to scram the drum. Only four of the six drums are needed to shut down the reactor.

Prior to installation of the control mechanism in the test facility, a simulated reflector and a control drive mechanism will be assembled, and operational tests will be performed to determine satisfactory operation. These tests

will be conducted at operating temperatures as well as room temperatures. Scram tests will also be made during reactor shakedown tests and as soon as normal environmental conditions are established during nuclear operation.

Data on the reflectors and their control system are summarized in Table IV-1.

C. REACTOR INSTALLATION

The reactor and control system are located below grade level in a shielded, carbon steel, pressure vessel (see Figure III-6). The pressure vessel contains the following:

- 1) A lower compartment containing the reactor core and external reflector assembly
- 2) A steel-encased, concrete, shutdown-shield assembly above the lower compartment, which supports the reactor assembly and reduces flux levels to equipment located above the shield
- 3) An upper compartment above the shutdown shield containing the control drive mechanisms and heat transfer system connections (inaccessible during reactor operation)
- 4) A steel-encased, concrete, operating shield above the upper compartment, which provides radiation protection for personnel in the reactor room during reactor operation.

The reactor core and external reflector assemblies are located in the lower compartment of the pressure vessel below the shutdown shield. The assemblies are suspended from the shield with the bottom of the core tank approximately 17 in. above the bottom of the pressure vessel. The free volume below the core tank is sufficient to hold the entire primary NaK inventory ($\sim 10 \text{ ft}^3$ NaK inventory). The total free volume of the compartment below the shutdown shield is approximately 20 ft^3 .

The shutdown shield is a stepped steel-lined plug which separates the upper and lower compartments. The plug is made of high-density borated concrete, lead, and insulation. The plug is approximately 40 in. in diameter and 4 ft thick overall (including 3-1/8 in. of insulation and 5 in. of lead on the bottom, 36 in.

of high-density concrete, and 2 in. of lead on top). A 10-3/4-in. diameter, removable, stepped plug is provided in the center of the shield to permit fueling of the reactor. The shield assembly is cooled by air which is circulated through copper cooling coils embedded in the concrete and lead. The inlet and outlet coolant lines from the core, and the drive rods which control the reflector control drums, are routed through the shutdown shield to the upper compartment in the pressure vessel.

The upper compartment is located between the shutdown shield and operating shield. The compartment is approximately 48 in. in diameter and 27 in. high and has an overall volume of approximately 30 ft³. Radiation levels in this compartment are relatively low as a result of attenuation of core gamma and neutron radiation by the shutdown shield. These low levels are necessary to minimize radiation damage and neutron activation of the rod drive mechanisms and controls located in this area. The low radiation levels also permit access to the equipment after shutdown of the reactor. Loading and removal of the core, and connections to the heat transfer system are effected from this area after removal of the operating shield and shutdown-shield center plug.

The operating shield is a stepped steel-lined plug located just below the pressure-vessel dome at floor level. The plug is made of ordinary concrete, approximately 4-1/2 ft thick, and reduces the radiation levels from the operating reactor to permissible levels at the reactor room floor level. The dome of the pressure vessel, which protrudes above the reactor room floor level, is also filled with concrete (approximately 2 ft thick) primarily to prevent accumulation of radioactive gases in the dome.

Four instrument thimbles, containing eight neutron detectors, are located in the concrete shield adjacent to the pressure vessel. The thimbles and instrumentation will be described in Section V.

D. HEAT TRANSFER SYSTEM

The function of the heat transfer system is to remove the heat generated in the reactor core (600 kw) and reject it to the atmosphere. The system consists of (1) a primary heat transfer loop which removes heat from the core by means of a circulating NaK coolant, (2) an intermediate heat exchanger which effects transfer of the heat from the radioactive primary coolant to a

secondary nonradioactive heat transfer loop, (3) the secondary heat transfer loop, also containing circulating NaK coolant which does not become radioactive, and (4) an airblast heat exchanger which removes heat from the secondary loop and rejects it to the atmosphere. Figure IV-6 shows a simplified P&I diagram of the heat transfer system.

1. Primary Heat Transfer Loop

The primary heat transfer loop consists of a NaK coolant, a surge tank, an intermediate heat exchanger, a linear induction electromagnetic pump, an electromagnetic flowmeter, a side loop containing a plugging meter, associated valves, piping, and instrumentation.

The coolant is eutectic NaK (78% potassium). A total inventory of approximately 315 lb of NaK is contained in the primary loop. All metal surfaces in contact with the NaK are type 316 stainless steel or Hastelloy. The primary system loop piping and equipment are located in a shielded, steel-lined vault containing an inert atmosphere.

During operation, the NaK coolant flows from the reactor core at 1300°F via a 2-in. pipe. The coolant stream enters the 5 ft³ primary surge tank which contains a helium cover gas maintained at 15 psig. From the surge tank, the coolant flows into the tubes of the intermediate heat exchanger. The NaK leaves the heat exchanger at 1100°F and flows through the electromagnetic pump. A small stream (about 0.5 gpm) is taken from the exit side of the pump to flow through the plugging loop, for NaK impurity determination, and back to the pump inlet. From the electromagnetic pump, the NaK flows through a bellows seal valve and back to the reactor core lower plenum. The average primary NaK coolant flow rate is about 140 gpm.

Normally, the flow rate through the primary system loop is varied by means of a manually operated auto-transformer which controls the output of the electromagnetic pump. The coolant flow may also be varied by the valve in the primary loop (remote manual operation); the main purpose of this valve, however, is to aid in draining and purging the primary loop.

Helium is used to pressure-transfer the NaK from the primary loop to a shielded primary drain tank located in the secondary system equipment pit (drain tank not part of primary heat transfer system) and to purge the system.

Figure IV-6. Heat Transfer System P & I Diagram

Vent and purge gases from the primary system are routed to a gas holdup tank for monitoring prior to release to the stack. Contact maintenance on the primary loop can be performed after the loop has been drained and/or an appropriate waiting period has elapsed to permit decay of the activity remaining in the primary loop (primarily 15 hr Na^{24}).

A rupture disc assembly is provided to protect the primary heat transfer system from overpressure. The assembly is installed on the NaK reactor inlet line inside of the containment vessel above the shutdown shield. It consists of two rupture discs in series with a leak detector located between the discs. The rupture disc assembly is appropriately installed in the system to prevent accumulation of gas under the disc (see Figure III-6) which might interfere with proper operation of the leak detector (conductivity type detector). The unit is designed to rupture at 50 psig and withstand a backpressure of 168 psig. The primary system is designed for 100 psig at 1400°F. During the maximum credible accident the primary system in the vault would be subjected to a maximum possible overpressure of 167 psig and a temperature of 1560°F (however, this pressure drops to 100 psig in approximately 1 minute). Stress calculations indicate that the EM pump is the controlling component. However, the primary system would withstand the maximum conditions for more than 200 hr, and would, in fact, withstand a pressure of 400 psig for 5 to 10 hr. Hence, disc rupture pressure has been conservatively selected to protect that portion of the primary system located outside the containment vessel.

2. Secondary Heat Transfer Loop

The secondary heat transfer loop consists of a NaK coolant, an intermediate heat exchanger, an airblast heat exchanger, a secondary surge tank, an electromagnetic pump, an electric heater, associated valves, piping, and instrumentation.

The coolant is NaK 78 (same as primary loop). All metal surfaces in contact with the NaK are made of 316 stainless steel. A total inventory of approximately 622 lb of NaK is contained in the loop. Except for the intermediate heat exchanger, the piping and equipment associated with the secondary loop are located in unshielded areas above grade level (secondary fill and drain tank located in the secondary system equipment pit). During normal operation, the pressure in the shell side (secondary coolant side) of the intermediate heat exchanger will be approximately 22 psi greater than the pressure in the tube side; if leaks should occur in the heat exchanger, therefore, the secondary system would not become contaminated with the radioactive primary NaK. With no flow, the shell-side pressure is 4.3 psi greater than the tube-side pressure owing to static head differences. The airblast heat exchanger is located outside the building on a concrete pad. Because the secondary loop contains no radioactivity, shielding is not necessary.

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During operation, the NaK coolant flows through the shell of the intermediate heat exchanger and exits at 1250°F. The flow is then to the airblast heat exchanger where the heat is removed and dissipated to the atmosphere. The NaK coolant exits the airblast heat exchanger at 1050°F and enters the secondary surge tank, which is maintained at 15 psig by a helium cover gas. From the surge tank, the coolant flows through the electromagnetic pump and back to the intermediate heat exchanger. The average normal flow in the secondary loop is about 140 gpm.

Flow through the loop may be varied by means of a manually operated auto-transformer which controls the output of the electromagnetic pump. A manually operated valve is provided in the loop to aid in draining and purging the loop. The secondary fill and drain tank is used to fill, drain, and purge the secondary loop. Helium is used to move NaK through the loop during draining, purging, and filling.

An electric heater is installed in parallel with the airblast heat exchanger. The reactor and its heat transfer system may be heated electrically with this heater by allowing NaK flow through the heater instead of the airblast heat exchanger. The capacity of the electric heater is 150 kw.

A rupture disc, designed to burst at 50 psig, is provided on the surge-tank vent line to protect the secondary heat transfer system from overpressure. The rupture disc discharges to the building stack.

E. OPERATING CHARACTERISTICS

1. General Nuclear Characteristics

The S8ER is the third in a series of reactors very similar in their nuclear design. The characteristics of the first two, the S2ER and S2DR, experimental and developmental reactors in the SNAP 2 program, have been compared in a previous safeguards report.⁵ Determinations of nuclear characteristics of both these reactors have given an experimental basis for the nuclear design of S8ER; the level of confidence in design methods has been enhanced.

In the S2ER, a 7 wt % U - ZrH_{1.68} alloy (N_H = 6.0) was used as a fuel moderator. The S2DR saw a change to a 10 wt % U - ZrH_{1.68} alloy (N_H = 6.5), and the incorporation of burnable poison Sm₂O₃ in the cladding coating. The

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amount of Sm_2O_3 added prepoisoned the S2DR to a reactivity level associated with equilibrium Sm^{149} , i. e., $-0.9\% \Delta k/k$.

For the S8ER, a 10 wt % U - $\text{ZrH}_{1.7}$ ($N_H = 6.0$) has been selected and the burnable poison concept retained. The amount of poison, however, has been increased to the extent that nearly three times the equilibrium Sm^{149} value is initially present, or $-2.6\% \Delta k/k$. Thus, the long-term, external, drum-control capability required for the S8ER has been reduced to approximately $2.6\% \Delta k/k$. Because the S8ER operates at a power level of 600 kw, 12 times that of the S2ER and S2DR, the excess reactivity requirements (see Table IV-1) are greater, necessitating a larger reactor. (The addition of burnable poison only reduces the external control needed. The size of the reactor for given excess requirements must be the same with or without prepoison.) The active core region of S8ER has therefore been lengthened to 14.0 in. In addition, upper and lower axial reflectors (1-1/2 in., BeO , end-cap inserts were used in S2DR fuel elements) have been eliminated. What little axial reflectivity exists is provided by the grid plates and the cladding end caps. Radially, the S8ER has a nominal 3-in. reflector outside the core vessel. The uranium inventory for this configuration is 6.56 kg.

The S8ER has a computed neutron mean prompt lifetime of $6.7 \mu\text{sec}$ and β_{eff} of 0.0077. These are very similar to the parameters computed for S2DR; $l = 6.4 \mu\text{sec}$ and $\beta_{\text{eff}} = 0.0080$. The experimentally determined lifetime for S2DR is $6.3 \mu\text{sec}$ for a $\beta_{\text{eff}} = 0.008$. The median fission energy of S8ER is computed as 0.21 ev. The S2DR median fission energy is 0.19 ev. These parameters appear with the geometric and material characteristics in Table IV-1.

2. Excess Requirements, Control Capability and Long Term Reactivity Variation

The excess requirements and the control capability of the S8ER are presented most easily in tabular form. They are shown in Table IV-1. The individual worth of a drum ($2.7\% \Delta k/k$) multiplied by six does not quite match the worth of six drums because of interaction or drum-coupling effects. The drum-worth curve is shown in Figure IV-7. The reactor itself has been designed for a maximum unpoisoned k_{eff} of 1.075. However, the concrete shielded containment vessel in which the S8ER will be operated contributes an additional reactivity of 1.1%. Thus, the effective, unpoisoned, excess reactivity

Figure IV-7. S8ER Control Drum Worth Curve

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of the reactor is 8.6%. The addition of the Sm_2O_3 then reduces the k_{eff} to 1.060 (at room temperature).

The effect of the prepoison on the lifetime excess reactivity, following full-power startup, is shown in Figure IV-8. The initial value of the excess at power is 4.4% $\Delta k/k$. The drop occurring in the first few days results from the xenon transient. As prepoison burns out, excess reactivity increases. After 100 days, the reactivity steadily decreases as a result of fission product accumulation, uranium burnup and hydrogen loss (5% in 10,000 hr).^{*} At the end of lifetime, if no fuel rods fail, the reactor should have an excess of 1.0%. The dashed line curve is included to show the reactivity change which would occur if no hydrogen were lost.

3. Temperature Coefficients

The calculated total isothermal temperature coefficient of the S8ER is displayed in Figure IV-9 as a function of temperature for both the prepoisoned and unpoisoned cores. It is seen that the addition of Sm_2O_3 reduces the magnitude of the coefficient. That part of the total coefficient which may be considered as acting promptly, results from the thermal spectrum shift, axial fuel-rod expansion, and upper-grid-plate expansion. The combined effect of these three mechanisms yields a prompt temperature coefficient of approximately $-0.07\%/^{\circ}\text{F}$ at the operating temperature of 1200°F core average coolant temperature. Temperature coefficients for the S8ER were obtained by a combination of theoretical calculations and conservative extrapolation of SER data as described in Appendix F.

4. Dissociation Pressure and Hydrogen Loss

Two important related characteristics of the fuel material are the dissociation pressure of zirconium hydride as a function of temperature and the hydrogen loss rate from ruptured elements or bare fuel.

At high temperatures, zirconium hydride is unstable and hydrogen dissociation occurs.⁶ Since the fuel is canned, the released hydrogen causes an increase in pressure in the fuel element and corresponding (higher) tensile stresses in the fuel cladding. At sufficiently high temperatures and pressures

^{*}The fuel rods are designed to have a hydrogen permeation rate which will result in a hydrogen loss of less than 5% in 10,000 hr.

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Figure IV-8. S8ER Time Variation of Excess Reactivity

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Figure IV-9. S8ER Total Isothermal Temperature Coefficient

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the fuel cladding will rupture. Figure IV-10 shows a plot of the cladding rupture pressure and fuel dissociation pressure vs temperature for the S8ER fuel element. The values for the dissociation pressure were obtained from Reference 6 and the ultimate tensile stress of the cladding material was obtained from Reference 7. Inspection of these curves indicates that cladding rupture would occur at approximately 1830°F under the assumed conditions (isothermal fuel and cladding temperatures).

Following cladding rupture, the free hydrogen immediately will be released or escape from the fuel and cause a corresponding loss in reactivity. The loss of hydrogen from the fuel, therefore, provides an ultimate shutdown mechanism which depends only on the rate of hydrogen release following cladding rupture. Figure IV-11 shows the rate of hydrogen loss from S8ER fuel as a function of temperature and time after cladding rupture. This figure is based on experimental data contained in a report by C. L. Huffine.⁸

Therefore, a nuclear excursion or other accident that causes overheating of the fuel will result in an increase in the dissociation pressure, cladding rupture at the limiting temperature, and ultimate reactor shutdown as a result of hydrogen loss. The significance of hydrogen loss on reactivity is described in Section VI-B. However, on the average a 2% hydrogen loss is approximately equivalent to a 1% reactivity loss.

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Figure IV-10. Dissociation Pressure of $\text{ZrH}_{1.71}$ and Cladding Rupture Pressure vs Fuel Rod Temperature

Figure IV-11. Hydrogen Loss from S8ER Fuel as a Function of Time and Isothermal Temperature

V. INSTRUMENTATION AND CONTROL

A. CONTROL ROOM AND CONSOLE

The control room is located at the north end of the reactor building immediately adjacent to the reactor room. A window located in the wall separating the control room and the reactor room gives the operator an unobstructed view of any operations taking place in the reactor room. Instrumentation and equipment required for both routine and emergency reactor operations are located within the control room.

A three-section control console commanding a good view of the instrumentation relay racks is the control center for reactor operations. Both the console and the relay racks are of modular construction. A cabinet containing the scram and interlock relays is located immediately to the rear of the instrumentation racks. Electrical connections between the control console, the instrument racks, the relay cabinet, and the reactor containment vessel are made via covered floor trenches. The trenches are divided into two sections by a vertical electrical shield plate. The signal wiring is routed on the opposite side of the shield from the control and power wiring. Figure V-1 shows the control room and the reactor instrumentation layouts.

B. NUCLEAR INSTRUMENTATION

Eight channels of nuclear instrumentation are used to monitor neutron flux and to provide signals representing reactor power and period for control and safety. A chart showing the expected flux range, and the equivalent reactor power, for each channel is shown in Figure V-2. A nuclear block diagram is given in Figure V-3. The characteristics of each of the nuclear channels is tabulated in Figure V-4. A control system diagram, showing the scram relay bus and interlocks, is given in Figure V-5.

Startup range Channels I and II indicate reactor power level from about 0.01 to 100 mw. In addition, Channel II also provides the minimum flux signal for the startup interlock system and a reactor period signal for the scram bus. Each channel consists of a General Electric type 546787065 fission chamber and a linear pre-amplifier feeding the associated linear (Channel I) or log (Channel II) CRM circuits. Specifically for Channel I, this consists of an ORNL 4-1 linear

THERMAL FLUX, n/cm^2 -sec

CHANNEL	DETECTOR	INSTRUMENT	
I	FISSION COUNTER, G.E. 5467870G5	DETECTOLAB DM 12(ORNL Q-1511)	LINEAR CRM
II	FISSION COUNTER, G.E. 5467870G5	GENERAL ELECTRIC 655C912C1	LOG CRM & PERIOD
III	COMP. ION CHMBR, G.E. 5467870G9	KEITHLEY MODEL 420A	LOG N & PERIOD
IV	COMP. ION CHMBR, G.E. 5467870G9	KEITHLEY MODEL 420A	LOG N & PERIOD
V	COMP. ION CHMBR, G.E. 5467870G9	KEITHLEY MODEL 410	MICROMICROAMMETER
VI	UNCOMP. ION CHMBR, A.I.-HUNTSINGER	DETECTOLAB DR3(ORNL Q-1565)	SAFETY AMPLIFIER - A.I. MODIFIED FOR RELAY SCRAM
VII	UNCOMP. ION CHMBR, A.I.-HUNTSINGER	DETECTOLAB DR3(ORNL Q-1565)	SAFETY AMPLIFIER - A.I. MODIFIED FOR RELAY SCRAM
VIII	UNCOMP. ION CHMBR, A.I.-HUNTSINGER	DETECTOLAB DR3(ORNL Q-1565)	SAFETY AMPLIFIER - A.I. MODIFIED FOR RELAY SCRAM

*RANGES OF POWER CHANNELS MAY BE EXTENDED DOWNWARDS BY REPOSITIONING CHAMBERS.

Figure V-2. Nuclear Channels and Ranges

Figure V-3. Nuclear Block Diagram

CHANNEL NUMBER	MANUFAC- TURER	INSTRUMENT PURPOSE	INSTRUMENT RANGE LIMITS AT DETECTOR POSITION	SCRAM TRIP CIRCUITS	INSTRUMENT INDICATOR RESPONSE TIME	TIME CONSTANT OF TRIP	TIME DELAY SIGNAL TO CLUTCH RELEASE	TOTAL EST. TIME INCLUDING 129 MSEC ROTATION TIME
I	DETECTOLAB (ORNL Q-1511)	LINEAR CRM AUXILIARY START-UP CHANNEL (SCALER AND POPPY)	5×10^4 CPS	NONE	100-1000 CPM < 1 SEC 2% STD DEVIATION 3.7 SEC 0.2% STD DEVIATION	—	—	—
II	GENERAL ELECTRIC	START UP LOG CRM AND PERIOD MINIMUM FLUX	$1-10^5$ CPS 5 DECADE	PERIOD 3 SEC 5 SEC WARNING			40 MSEC	
III	KEITHLY 420 A	INTERMEDIATE RANGE	10^{-15} TO 10^{-7} AMP	PERIOD 3 SEC 5 SEC WARNING	CURRENT 10^{-12} TO 10^{-7} < 2 SEC 90% PERIOD 3 SEC MAX TO 90% FINAL	10 MSEC	40 MSEC	179 MSEC
IV			IDENTICAL TO CHANNEL III					

Figure V-4. Nuclear Instrument Characteristics
(Sheet 1 of 2)

V	KEITHLY MOD 410	EXTENDED RANGE	10^{-3} AMPS @ 10^2 N/CM ² 10 DECADES	NONE	10^{-12} TO 10^{-8} 67% FINAL VALUE 1 SEC 10^{-9} TO 10^{-3} 90% FINAL VALUE IN 0.5 SEC	—	—	—
VI	DETECTOLAB (ORNL Q-1565) AI MODIF. 8ERI 23004	POWER RANGE	8×10^{-6} AMP @ 1.6×10^9 N/CM ²	130% DESIGN POWER 120% WARNING	RECORDER 4 SEC F.S.	12 MSEC	40 MSEC	181 MSEC
VII	DETECTOLAB (ORNL Q-1565) AI MODIF. 8ERI 23004	POWER RANGE	8×10^{-6} AMP @ 1.6×10^9 N/CM ²	130% DESIGN POWER 120% WARNING	RECORDER 4 SEC F.S.	12 MSEC	40 MSEC	181 MSEC
VIII	DETECTOLAB (ORNL Q-1565) AI MODIF. 8ERI 23004	POWER RANGE AND AUTO FLUX CONTROL	8×10^{-6} AMP @ 1.6×10^9 N/CM ²	130% DESIGN POWER 120% WARNING	RECORDER < 4 SEC F.S.	12 MSEC	40 MSEC	181 MSEC

Figure V-4. Nuclear Instrument Characteristics
(Sheet 2 of 2)

Figure V-5. Control System Diagram

amplifier driving an ORNL linear countrate meter. A second pulse output circuit in the linear amplifier is used to drive a Baird Atomics 4-decade scaler. The scaler has been modified so that the outputs of the first three decades can be selected and fed to an audio amplifier. This "poppy" amplifier is equipped with control room and reactor room speakers to provide an audible indication of count rates. The output of the Channel II pre-amplifier is fed into a General Electric logarithmic countrate meter equipped with a period circuit. The period circuit extracts the derivative of the log countrate, displays the information, and operates a scram relay. The adjustment range of the period trip circuit is from ∞ to 3 sec. An adjustable electronic relay is used in conjunction with the log CRM to provide the minimum flux interlock for reactor startup.

Channels III and IV provide neutron flux level signals in the intermediate range from about 1×10^{-3} to 1×10^4 watts. Each channel also provides a period signal for the scram systems which causes reactor scram on short period. The components and circuitry of both channels are identical. The neutron detectors are General Electric type 546787069 gamma-compensated ionization chambers. The chamber outputs are connected to Kiethly model 420A Log N and Period amplifiers. The power levels are recorded on a Bristol dual-pen recorder. The period signal has associated electronic trips for short-period warning and scrams.

All period scrams from Channels II, III, and IV have a common bypass that is actuated whenever the reactor power level exceeds 5% rated power for 1 min. The bypass is automatically de-actuated without delay whenever the power level is below 5% of rated.

Channel V is an extended range monitoring system consisting of a General Electric gamma-compensated ionization chamber connected to a Kiethly model 410 micro-microammeter located at the control console. The range of this instrument extends from about 1×10^{-6} to 7×10^4 watts.

Channels VI, VII, and VIII are power range instruments providing neutron flux level signals from about 1 watt to 200% full power. Each channel consists of an Atomics International uncompensated ionization chamber, an ORNL pre-amplifier, and modified ORNL safety amplifiers. The outputs of Channels VI and VII are recorded on a Bristol dual-pen recorder. The safety amplifiers of all three channels provide power level signals for the scram system. The Minneapolis-Honeywell recorder associated with Channel VIII has a retransmitting slide wire to be used in conjunction with an automatic flux controller.

The nuclear detectors for all 8 channels are located in 4 instrumentation thimbles adjacent to the reactor containment vessel, two on a side. The thimbles (4-3/4 in. diameter by 12 ft long) are fabricated from steel piping and have stainless steel liners. Each thimble is provided with a canned 3-ft paraffin, lead and boral shield plug containing a spiral cable passage. Additional local shielding can be added above the thimbles and shield plugs if streaming is excessive at the reactor room floor level. The associated detector pre-amplifiers are enclosed near the top of the instrument thimble.

C. REACTOR CONTROL

Reactivity control in the reactor is provided by the six rotating control elements. A selector switch at the operator's console allows rotation of only one drum at a time, but they may be rotated in any sequence. A separate control allows simultaneous outward rotation of all 6 drums for setback control.

Under a scram condition the electromagnetic clutch on each drum drive mechanism is disengaged, allowing a heavy torsional spring to rotate the control element rapidly to its least reactive position. Any one of the scram signals will open two series-wired relays de-energizing the magnetic clutches. The total time for any scram signal to remove \$2.50 of reactivity is 194 msec (design requirement = \$2.50 in 200 msec not including relay dropout time). This is ample negative reactivity to turn over the neutron power during any excursion and start the reactor well on its way to shutdown. This control system response time must be added to the nuclear instrumentation response time to obtain the overall system "dead" time to an abnormal condition. Table V-1 shows the response time of the various control system components.

TABLE V-1
RESPONSE TIME OF CONTROL
SYSTEM COMPONENTS

Item	Time* (msec)
Dropout time of scram relays	25
Clutch release time	40
Rotation time	<u>129</u>
Total	194

*Dropout time of scram relay determined experimentally. Other times are calculated values. The calculated values agree with times observed in the laboratory with simulated equipment.

If a drum does not scram to the full "out" condition, the clutches may be re-energized and the drum driven out. To re-energize the clutches following a scram, the safety circuit must be energized and a switch turned which interrupts the control drum "in" drive power.

Both coarse and fine position indication systems are provided to determine the position of each reflector element. A potentiometer incorporated into each sensor unit coupled to a vertical scale voltmeter indicates the position of each drum within $\pm 5^\circ$ for coarse indication. A resolver-control-transformer servo system incorporating a servo-motor-driven digitizer indicates the reflector position with an overall accuracy of $\pm 0.07^\circ$ or $\pm 0.4\%$. This precise system has two readouts. One readout is always connected to the element being used for reactor control. The other readout can be used to check the position of any of the five other drums.

D. PROCESS INSTRUMENTATION

Reactor coolant pressure, flow, level, and temperature are the process variables monitored in the primary and secondary NaK heat transfer systems. Flow is measured using permanent-magnet flowmeters placed downstream of the circulating pump in each loop. These flows are recorded on a two-pen strip-chart recorder located in the control room. The recorder contains micro-switches to actuate the low-flow scram and the approaching low-flow alarm. The same type of flowmeter is used on the primary system plugging meter. The plugging meter in the primary system is used to determine the oxide and/or hydride concentration in the primary NaK. Plugging meter flow and temperature signals are received by a modified two-pen recorder located in the control room.

NaK pressure and tank levels are recorded in the control room on multipoint strip chart recorders. NaK pressures are measured on the suction and discharge of the circulating pump for each loop using electric pressure transmitters provided with NaK-type diaphragm seals. All of the NaK fill, drain, and surge tanks are provided with "J"-tube-type (nichrome-resistance) level detectors. Alarms indicating "off" normal conditions are provided.

Seven chromel-alumel thermocouples have been provided in the coolant outlet plenum and 3 have been placed in the discharge port. The plenum temperature signals are connected to a multipoint strip chart recorder. The three outlet thermocouples are connected as follows: one to a temperature switch, one to a

temperature recorder, and one to a temperature controller indicator. Any one of these will initiate a scram upon receipt of a high-temperature signal. In the event of the failure of an outlet thermocouple, one of the plenum thermocouples may be substituted (after appropriate adjustment of its safety circuit trip setting). At least three thermocouples (either discharge port or core outlet thermocouples) must remain operable during the approach to design power level. After operating experience has been gained at design power level and coolant flow and temperature characteristics have been demonstrated, the reactor may be operated with a single operable discharge port or core outlet thermocouple. The temperature controller also provides the capability of utilizing automatic "on - off" temperature control for flight reactor control simulation.

The 150-kw NaK heater and the heaters which provide the required temperature differential for cold trapping on the primary drain tank, primary fill tank, and secondary fill and drain tank are provided with indicating temperature controllers (millivolt type) located in the control room. Remote control of one of the air blast heat exchanger louvers is possible from the control room.

Service systems (cooling water, cooling air, nitrogen, and helium) instrumentation employ standard industrial components, (i. e., pressure gages, switches and regulators; thermometers; variable-area-type flowmeters; and level sight glasses) locally mounted. Various alarms for low manifold pressures, low cooling water flow, etc., are installed in the control room.

E. REACTOR SCRAMS, ALARMS, AND INTERLOCKS

A 10-point annunciator with red windows is mounted on the instrument rack which gives visual and audio indication of reactor scrams.

Reactor scrams which are indicated on the annunciator panel are:

- a) High neutron flux, set at 130% full power on Channels VI, VII, and VIII
- b) Short period, set at 3 sec on Channels II, III, and IV
- c) High reactor temperature, set at 1375°F, highest of 3 thermocouples in the discharge port
- d) Low NaK flow, primary loop, set at 30% of rated flow or 42 gpm. The primary pump is also shut off on this scram

- e) Low level primary surge tank, set at 12 in. minimum depth. The primary pump is shut off on this scram.
- f) Malfunction nuclear instruments, 2 of 3 coincidental failure of power Channels VI, VII, and VIII
- g) Manual shutdown
- h) Earthquake, set at VIII Mercalli. Both primary and secondary pumps are turned off on this scram.
- i) Containment vessel pressure, set at 15 psig
- j) High primary system pressure, 40 psig

A 20-point annunciator is mounted on the instrument rack which gives visual and audio indication of the following pre-scrams and system malfunction. The warnings a) through e) are pre-scram warnings and are distinguished from the balance of the annunciator system by yellow painted window frames on the annunciator assembly.

- a) Approaching high flux, set at 120% full power Channels VI, VII, and VIII
- b) Approaching short period, set at 5 sec, Channels III and IV
- c) Approaching high temperature, set at 1325°F (any of 7 thermocouples in reactor outlet plenum)
- d) Approaching low primary NaK flow, set at 75 gpm
- e) Approaching low primary surge tank level, set at 18 in.
- f) Containment vessel pressure, set at 10 psig
- g) Reflector clutches de-energized
- h) Leak primary rupture disc
- i) Low flow, secondary NaK, set at 30% of rated flow or 42 gpm
- j) High radioactivity vault atmosphere, set to a value as close as possible to the normal operating background determined from operating experience
- k) High radiation area monitor signals, individual detectors set to a value as close as possible to the normal operating background determined from operating experience

- l) Off automatic control
- m) Low nitrogen pressure, primary vault, set at 3 in. W.G.
- n) Low reactor outlet temperature, set with operating experience to a value as close as possible to operating temperature
- o) High radiation stack monitor signal, set to a value as close as possible to the normal operating background determined from operating experience
- p) Shutdown shield vent line high pressure, set at 10 psig
- q) Spare
- r) Spare
- s) Spare

Another 20-point annunciator mounted on the instrument rack gives the following visual and audio alarms:

- a) High shell temperature, secondary NaK heater, set at 1375°F
- b) High discharge pressure, NaK pumps, set at 35 psig
- c) High primary system temperature, any of several thermocouples at miscellaneous primary system locations, set with operating experience to values as close as possible to operating temperatures
- d) Off-normal helium pressure, containment vessel, set at 3 and 10 in. W.G.
- e) High nitrogen pressure, primary vault, set at 10 in. W.G.
- f) High temperature, plugging line, set at 1300°F
- g) High temperature, intermediate heat exchanger, set at 1300°F for outlet of secondary system intermediate heat exchanger, and 1100°F for inlet of secondary system intermediate heat exchanger
- h) Off-normal helium pressure, primary surge tank, set at > 20, or < 10 psig
- i) Off-normal helium pressure, secondary surge tank, set at > 20, or < 10 psig
- j) Low pressure, helium supply, set at 50 psig

- k) Low pressure nitrogen supply, set at 10 psig
- l) High temperature, evaporative cooler outlet water, set at 120°F
- m) High humidity, vault cooling outlet, set at 10%
- n) Low level, secondary surge tank, set at 18 in.
- o) Low cooling water level, surge tank, set at 8 in. above bottom of tank
- p) Low water flow, cavity coolant, set at 3 gpm
- q) Low gas flow, shutdown shield coolant, set at 300 cfm
- r) Spare
- s) Spare

The audio alarm for the scram annunciator is a motor-interrupted chime. The pre-scram and process audio alarm is a horn. Upon receipt of a trip signal, the light flashes and the audio alarm is activated. The operator can silence the alarm by pressing the silence button and the light ceases flashing, but remains lit until the situation is corrected and the reset button is depressed.

1. Prestartup Interlocks

The reactor startup interlocks are those which must be satisfied before the reactor can be operated. The operational interlocks are: (a) the console master key must be turned to the "on" position energizing the reactor control circuits, clutch power supply, and "reactor operating" beacons; (b) the count rate on Channel II must be greater than the minimum set point of 30 counts per min; (c) primary coolant NaK flow must be above 10 gpm; (d) secondary NaK coolant flow must be above 10 gpm; (e) all control elements must be in the least reactive "out" position; (f) period scram circuit must not be bypassed; (g) scram bus must be clear.

2. Automatic Control Features

Interlocks that are on the automatic flux or temperature control provide maximum assurance that the automatic power controlling system may be overridden by the reactor operator at all times. Operational interlocks are: (a) an interlock which prevents automatic control if the control drum used is at either limit switch; (b) an interlock which prevents automatic flux control unless the reactor power level is above 5% rated value; (c) an interlock which prevents

automatic temperature control unless the reactor coolant outlet temperature is above 500°F; (d) an interlock which prevents manual insertion of control drum during automatic control; and (e) an interlock which transfers control automatically to "manual" and provides a visual and audible signal that control has been transferred to manual whenever the controlled parameter varies from the set point by more than 5%, or any drum other than the control drum is moved to a less reactive position. The flux and temperature control set points are rate limited to 5% rated power per min, and 10°F per min, respectively.

3. Process System Interlocks

Interlocks prevent starting the primary pump unless: (a) the earthquake switch is closed; (b) the primary surge tank is filled to at least 12 in. depth; or (c) the pump powerstat is set at zero. When these conditions are met, the pump contactor may be actuated and the pump driving voltage raised to the desired operating level.

Interlocks prevent starting the secondary pump unless: (a) the secondary surge tank is filled to at least 12 in. depth; (b) the pump powerstat is set at zero; or (c) the airblast heat exchanger louvers are closed and the fan is off.

The heater is automatically de-energized in the event: (a) the secondary NaK flow drops below 10 gpm; (b) the secondary surge tank drops below 12 in. depth.

4. Scram and Interlock Bypass

Circuits listed in this section may have to be bypassed for operating reasons or to permit instrument maintenance during power operation. A bypass panel of key switches is located on one of the instrument panels in view of the reactor operator seated at the control console. By administrative control only those scram circuits for which key-switch bypasses are provided will be bypassed during reactor operation. This type of panel has two advantages: (a) bypass keys in use are on display within vision of the reactor operator and his alternate; and (b) administrative controls to prevent bypassing of vital circuits behind the panels are feasible.

The bypass panel consists of two rows of six key-operated switches. In both rows, the locks have interchangeable keys within a row, but lower row keys are different than those for the upper locks. The lower row contains first order bypasses; the upper, second order.

Lower row switches (2 keys available) are:

<u>Switch</u>	<u>Bypassing</u>
1	Power Channel VI, high flux safety circuit trip
2	Power Channel VII, high flux safety circuit trip
3	Power Channel VIII, high flux safety circuit trip
4	Reactor outlet temperature switch, TIS-11
5	Reactor outlet temperature recorder, TR-6
6	Reactor outlet temperature recorder, TR-1
7	Spare

Upper row switches (a key for each lock) are:

<u>Switch</u>	<u>Bypassing.</u>
1	Low primary flow scram (30% flow) Low primary flow reactor startup interlock (75% flow)
2	Low secondary flow reactor startup interlock (75% flow)
3	Low level primary surge tank
4	Containment vessel pressure
5	High primary system pressure
6	"Drums out" contact on startup interlock (also disconnects "in-drive" circuit)
7	Spare

Upper switch No. 1 must be keyed to permit startup of the primary pump. After primary flow exceeds 30% full flow (42 gpm), the key must be removed so that the low flow scram circuit will be operable. Bypasses are available for low primary and secondary flow reactor startup interlocks for use during low-flow experiments.

Only two keys will be available to the shift supervisor for use in the lower row of switches. This limitation will prevent the simultaneous bypassing of more than two of three reactor outlet high-temperature scrams. However, the availability of two keys will not permit simultaneous bypassing of two power channel scrams, due to the special circuitry provided. Lower switches 1, 2, and 3 each simultaneously bypasses the safety circuit trip and opens the malfunction relay for the appropriate power channel. During the time one power channel is bypassed, malfunction of any other power channel will trip the safety circuit through the nuclear instrument malfunction scram. Likewise, bypassing of a second power channel will also cause a nuclear instrument malfunction scram.

VI. POTENTIAL HAZARDS AND HAZARDS CONTROL

A. HAZARDS ASSOCIATED WITH NORMAL OPERATION

1. Radiation in Accessible Areas with Reactor Operating at 600 kw

Operation of a nuclear reactor produces neutrons, gamma rays and fission products. During normal operation of the S8ER, the fuel and fission products are contained by the fuel cladding, and the hazards associated with this source are due primarily to the direct gamma and neutron radiation escaping the core. To protect personnel from these direct radiation hazards, the reactor is completely surrounded by biological shielding (the concrete and earth surrounding the containment vessel, the shutdown shield, and the operating shield as described in Sections III and IV).

Radioactivity is also produced as a result of neutron activation of materials exposed to neutron fluxes. In the S8ER, the major production of induced radioactivity will result from neutron activation of the NaK coolant during its passage through the core. During normal operation, the primary NaK coolant is contained in a closed loop, and the only hazard associated with this source is the direct gamma radiation escaping through the walls of the primary heat-transfer-system piping and equipment. To protect personnel from this hazard, all portions of the heat transfer system are located in the shielded primary-system vault or in the compartment below the operating shield. These areas are inaccessible during operation.

Table VI-1 shows the maximum radiation levels expected in accessible areas with the reactor operating at full power (600 kw for 10,000 hr). Inspection of the table indicates that radiation levels are insignificant to operating personnel.

2. Radiation Levels in Primary System Vault

In the event of malfunction of equipment, it may be desirable or necessary to enter the primary system vault for inspection or maintenance. This can be accomplished only by removal of the top shield blocks and cover plate. The necessary sequence of events will be:

- a) Shutdown reactor
- b) Allow an adequate shutdown period for reactor cooling and to permit radioactive decay of NaK coolant.

TABLE VI-1

RADIATION LEVELS IN ACCESSIBLE AREAS DURING FULL
POWER OPERATION (600 KWT)

Source	Maximum Dose Rates in Accessible Areas (Reactor Room)		
	Above Reactor at Floor Level (mrem/hr)	Above Primary System Vault (mrem/hr)	Equipment Pit (mrem/hr)
Neutron	0.012	-	-
Core Gammas	Negligible	-	-
NaK*	0.090	0.002	1.4 [†]
Total	0.1	0.002	1.4

*Activation of NaK results in a total inventory of 257 curies of Na²⁴ and 116 curies of K⁴² in the primary coolant (see Appendix G). Approximately 1.26 ft³ of NaK are in the containment vessel (0.18 ft³ above shutdown shield and 8.82 ft³ in the primary system vault).

[†]Maximum dose rate in pit from streaming through wall penetrations. Dose rates at the floor level are insignificant.

- c) Drain NaK coolant to drain tank if necessary (~10% of NaK will remain in the vault piping after draining).
- d) Purge vault atmosphere to the stack.
- e) Remove top shield blocks and seal plate.

Table VI-2 shows the radiation levels in the vault during operation and after shutdown. Since the reactor and vault are separated by a minimum of 5 ft of concrete, neutron activation of materials in the vault is insignificant, and the residual radiation level in the vault, after a normal shutdown, will be due primarily to the activated coolant remaining in the vault (as table indicates). Inspection of the table indicates that limited access to the vault for inspection or minor maintenance should be possible approximately 4 or 5 days after shutdown. After one week's decay, the general radiation levels should be low enough to permit limited access for general maintenance.

Dose rates at the floor level above the equipment pit as a result of NaK in the drain tank (90% of total inventory) would be approximately 300 mr/hr at shutdown, 100 mr/hr one day after shutdown, and 11 mr/hr 3 days after shutdown.

TABLE VI-2

RADIATION LEVELS IN PRIMARY SYSTEM VAULT DURING AND FOLLOWING
FULL-POWER OPERATION FOR 10,000 HR

Source	Dose Rate in Vault				
	During Operation (mrem/hr)	1 day decay (mrem/hr)	3 day decay (mrem/hr)	1 wk decay (mrem/hr)	10 days decay (mrem/hr)
Neutrons [*]	14,000	0	0	0	0
Core Gammas	5,600	4.3	3.5	2.8	2.5
NaK [†]	590,000	200,000	22,000	250	9.0
Total	609,600	200,000	22,000	253	11.5

*Less than 10^5 n/cm²/sec activation flux level

†Dose rates based on 3-ft distance from unshielded point source. Approximately 45% of the NaK inventory (315 lb total inventory) will be contained in the primary surge tank, with the remaining portion in other piping and equipment. Following shutdown, the doses in the vault can be reduced by approximately a factor of 10, by draining the NaK to the shielded drain tank.

3. Radiation Levels Above Shutdown Shield

In the event of the malfunction of control system equipment, it may be necessary to enter the containment vessel for inspection or maintenance. This can be accomplished only by removal of the containment vessel head and operating shield. The necessary sequence of events will be as follows:

- a) Shut down the reactor.
- b) Allow adequate shutdown period for reactor cooling and to permit radioactive decay of the NaK coolant.
- c) Drain NaK coolant to drain tank if necessary (including NaK coolant lines above shutdown shield).
- d) Purge containment vessel atmosphere to stack.
- e) Remove containment vessel head and operating shield.

Table VI-3 shows the radiation levels above the shutdown shield during full-power operation and after shutdown.

TABLE VI-3

RADIATION LEVELS ABOVE SHUTDOWN SHIELD DURING AND FOLLOWING
FULL-POWER OPERATION FOR 10,000 HR

Source	Dose Rate Above Shutdown Shield			
	During Operation (mrem/hr)	1 day decay (mrem/hr)	3 days decay (mrem/hr)	1 wk decay (mrem/hr)
General Radiation Level				
Neutrons*	7,300	-	-	-
Core Gammas	Negligible	-	-	-
NaK [†]	49,200	16,200	1,740	20
Total	56,500	16,200	1,740	20
Streaming (Coolant Lines)				
Neutrons*	1,760,000	-	-	-
Core Gammas	14,000	460	360	280
NaK	49,200	16,200	1,740	20
Total	1,810,000	16,700	2,100	300

*Less than 6×10^4 n/cm²/sec activation flux level in general area and 1.53×10^7 n/cm²/sec from streaming.

[†]Approximately 6-1/2 ft of 2.0-in. pipe containing NaK located between shutdown shield and operating shield. NaK assumed to be distributed about periphery of equipment chamber above shutdown shield.

Inspection of the above table indicates that the activation flux levels are too low to cause significant activation of materials above the shutdown shield. The major radiation levels following shutdown will be due to activity from the core fission products and the NaK in the piping above the shutdown shield. Limited access to the area for inspection or maintenance should be possible approximately 4 or 5 days after shutdown.

4. Activity in Containment Vessel Atmosphere

A high purity helium atmosphere will be maintained in the free space in the containment vessel and the vessel will remain sealed during all reactor operation. Neutron activation of helium does not result in the production of significant radioactive products. There will, therefore, be no buildup or release of hazardous radioactivity as a result of neutron activation of the containment vessel atmosphere. Following shutdown, the containment vessel atmosphere can be vented to the stack and purged at a controlled rate.

5. Activity Release from Shutdown Shield Cooling System

The shutdown shield is cooled by a small stream (360 cfm) of outside air which is dried and routed through cooling coils embedded in the concrete and lead in the shutdown shield (see Figure III-6). The exit stream is routed directly to the stack where dilution with the 5000 cfm stack exhaust occurs. Because air contains approximately 1% argon, and the cooling coils pass through a neutron flux field, significant A^{41} activity is produced.

A comparison of the calculated A^{41} activity at the stack exit with MPC values is shown below:

$$A^{41} \text{ concentration at stack exit} = 1 \times 10^{-7} \mu\text{c/cc}$$

$$\text{MPC (submersion) for controlled persons} = 2 \times 10^{-6} \mu\text{c/cc}$$

$$\text{MPC (submersion) for uncontrolled persons} = 4 \times 10^{-8} \mu\text{c/cc}$$

The A^{41} concentration at the stack exit is therefore 1/20 of the MPC allowed for exposure of persons in controlled areas (40-hr week) and 2.5 times the MPC allowed for persons in uncontrolled areas. By the time the activity reaches an exposure point (on the ground or in adjacent facilities) the average concentration will be well below the MPC allowed for uncontrolled areas. Actual doses near the facility will be less than indicated, because the MPC values are based on direct radiation exposure from submersion in an infinite cloud. Activation of nitrogen was investigated and does not result in any significant hazard.

6. Radioactivity in Cooling Water System

Portions of the shield cooling water system are subjected to neutron activation, particularly in the concrete surrounding the pressure vessel below the shutdown shield. Because the cooling water system (see Section III-E-3) contains demineralized water and operates as a closed loop, radioactivity in the system will be due primarily to activated corrosion products. Corrosion of the system is minimized by the addition of hydrazine (ph of 9 maintained), and buildup of corrosion products in the circulating coolant is limited by a particulate filter in the loop. Table VI-4 shows the calculated equilibrium curie quantities of radioactivity in the system, resulting radioactivity levels expected in the cooling water, and radiation levels from external system components.

TABLE VI-4

ACTIVITY AND RADIATION LEVELS FROM COOLING WATER SYSTEM*
(Maximum Levels in Water and External Components)

Source	Water (Equilibrium Levels)		Filter (Equilibrium Level)	
	mc	$\mu\text{c/cc}^\dagger$	mc	mr/hr (1 ft)
Fe 59	0.002	5.3×10^{-9}	2.2	15.3
Mn 56	0.082	2.17×10^{-7}	0.2	1.3
Total	0.084	2.2×10^{-7}	2.4	16.6

*Based on a 1 mg/dm^2 per day corrosion rate

$^\dagger\text{MPC}_w$ for Fe 59 = $5 \times 10^{-5} \mu\text{c/cc}$ (uncontrolled areas)

MPC_w for Mn 56 = $1 \times 10^{-4} \mu\text{c/cc}$ (uncontrolled areas)

Inspection of the above table indicates that the radioactivity concentrations in the cooling water are insignificant. The loop filter is located outside the building and will not cause exposure of personnel to significant radiation fields.

7. Activity in Soil Surrounding Reactor Pit

The concrete surrounding the lower portion of the pressure vessel is not of sufficient thickness to stop all neutrons which escape from the reactor. The neutrons which escape the shield will be captured by the soil in the local area adjacent to the reactor and shield and will produce induced radioactivity.

Radioactivity retained in the immediate area results in no hazard except for the potential radiation hazard during future excavation. Migration of the activity to water supplies as a result of ground water movement would be the major radiological concern. However, because of the limited rainfall in the area and the hydrological characteristics of the area (see Appendix A), interconnection with underground water-bearing strata or streambeds does not occur. Rainfall penetrating the substrata would do so very slowly and would remain in the local area unless pumped from one of the local wells within the NAA field test area. The water in the wells is monitored regularly for radioactivity and is not used for drinking.

To prevent any migration of the activity induced in the local soil, the facility is provided with a drainage system which surrounds the vault complex

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below the concrete structure. Rainfall or water passing near the affected neutron flux area is collected by the system and routed to a sump, where the water can be monitored and pumped out for processing and disposal if activity is detected.

8. Hazards from Minor Cladding Defects

Owing to the care employed in the manufacture of the SNAP 8 fuel elements, it is not expected that any fuel elements will have cladding defects or develop failures during the S8ER test. Nevertheless, because the reactor could continue to operate (knowingly or inadvertently) with cladding defects or incipient failures, it is of interest to determine the potential hazards associated with such operation.

Operation of the reactor under the above conditions would be expected to release fission products to the NaK coolant. Because the primary system operates as a closed loop, the released fission products would be contained in the primary loop, and the immediate operating hazard associated with this activity would be due to direct gamma radiation escaping from the primary system and vault. Following shutdown or during venting, the activity may present waste disposal and handling problems, and radiation levels may limit access to the vault. The magnitude of the potential hazard from continued operation of the reactor with cladding defects will depend on (1) the number of fuel elements with cladding defects, and (2) the fraction of the fission product inventory released from the affected fuel element.

The number of fuel elements which can have cladding defects, and yet allow operation of the reactor, would be limited by the loss of reactivity as a result of the escape of hydrogen moderator through the defect. For example, the equilibrium N_H of a defective fuel rod would be expected to drop from 6×10^{22} to 2.7×10^{22} atoms/cc. If 10% of the rods develop failures, this would result in a reactivity loss of approximately $2.8\% \Delta k/k$. This would cause shutdown (considering other reactivity losses such as xenon, samarium, and temperature defect) and would prevent further power operation of the reactor. It should be emphasized again that fuel elements are not expected to fail as a result of S8ER reactor operation, and the 10% figure is merely a convenient value for the purpose of demonstrating the safety of even this unlikely occurrence.

Fission products are released from exposed fuel as a result of the recoil and diffusion processes. Because the fuel is in the form of a solid rod (0.532 in. diameter) and moderate temperatures are involved (normal operating temperatures)

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the recoil process would be expected to be the major contributor to fission product release. If it is assumed that all the activity associated with recoil from the surface of a rod is released, a calculated release value of approximately 0.17% of the inventory in a failed element is obtained. Therefore, the fraction of the fission product inventory which would be released to the S8ER coolant as a result of operation with cladding defects or failures is:

$$\begin{aligned}\text{fraction of fission products released to coolant} &= 0.1 \times 0.0017 \\ &= 0.017\%.\end{aligned}$$

Since the operating reactor contains approximately 6×10^6 curies of fission product activity, the total activity in the circulating coolant or vault with failures in 10% of the elements is:

$$\begin{aligned}\text{activity in coolant} &= 6 \times 10^6 \times 0.017 \times 10^{-2} \\ &\approx 1,000 \text{ curies.}\end{aligned}$$

The calculated dose rate in the reactor room above the concrete shield from this source is:

$$\text{dose rate in reactor room} = 0.00274 \text{ mr/hr.}$$

Direct radiation levels to personnel, therefore, as a result of operation of the reactor with fission products in the coolant, would be insignificant. Following shutdown, the 1000-curie fission product source would decay according to the power function $t^{-0.2}$, where t is in seconds. In one day the activity would decay to 100 curies and to 50 curies in one week. Because significant portions of the remaining activity could be transferred to the shielded drain tank or vented to the gas holdup tank, the activity would not result in insurmountable access or waste handling problems to the facility.

9. Leaks, Spills and Backups

It is possible that gas leaks and coolant spills or backups may occur as a result of operating errors or equipment failures. These conditions can cause abnormal radiation levels or air activity levels in the facility and can lead to unnecessary personnel exposure or activity releases if not detected and corrected.

Abnormal radiation levels from these conditions, however, will be detected by the remote area monitoring system probes, and alarms associated with the system will indicate the existence of these conditions. Excessive or unnecessary exposure can, therefore, be avoided. The abnormal condition can then be investigated with the aid of a Health Physicist and portable instrumentation, and appropriate corrective action can be undertaken safely.

Abnormal air activity levels will be detected by the nitrogen system monitor, the stack monitor, and the facility air-particle monitor (constant air monitor). Alarms associated with these systems will warn operating personnel of the existence of these conditions. Appropriate action can be taken to avoid exposure and to minimize release of activity to the environment (scram reactor, evacuate unnecessary personnel, utilize protective equipment, isolate systems, etc.).

10. Radioactive Waste

Only small quantities of radioactive waste are expected to be generated as a result of S8ER operation. Control of solid radioactive wastes of relatively low radiation intensities is accomplished by providing suitably marked containers at the facility for the immediate collection of these wastes. Collected wastes are packaged, sealed, monitored and tagged according to specifications established by a special waste disposal unit which services the entire Atomic International Nuclear Field Laboratories Area. The waste disposal unit collects the accumulated material on scheduled days and transfers it to the Radioactive Material Disposal Facility for final preparation and disposal.

Control and disposal of high radiation intensity radioactive waste may require special shielded containers, depending on the physical size and radiation intensity of the waste material or item. These containers will be supplied by the waste disposal unit. The waste disposal unit will provide necessary transportation and immediate pickup following loading of the high intensity waste into the disposal container.

If the liquid waste sump is determined to contain radioactivity in excess of release quantities, the waste disposal unit will be notified and the contents of the sump will be transferred to the Radioactive Materials Disposal Facility for final processing and disposal. Contaminated water flushed from the cooling water system will be handled in a similar manner if significant activity is present in the water.

11. Fires

The materials used in the construction of the building are fire resistant. The design features employed to prevent interaction between NaK and facility water systems were described in Section III. During operation of the S8ER, the atmosphere in the containment vessel and primary vault will be filled with helium and nitrogen, respectively, to prevent a fire in case of a NaK spill in these areas. Hazardous amounts of combustible materials will not be allowed in or near the building. Fire could accidentally start as a result of NaK spills or faulty wiring. If such a fire should occur within or near the building and is considered a jeopardy to safe operation of the reactor, the reactor will be scrammed immediately. Smoke detectors in the building will actuate local alarms and alarms at the Industrial Security Control Center. The operations personnel will be well trained in the handling of NaK to minimize spills. They will also be trained in combating NaK fires and electrical fires.

The facility and surrounding site have sufficient fire protection systems to extinguish any plausible fire in the facility. Section III details the fire prevention system for the facility.

12. Fuel Handling

Prior to handling fuel material at the facility, a detailed fuel handling procedure must be submitted to the AI Reactor Safeguards Review Panel (see Appendix B). Following study and approval of the operation by this office and the appropriate approval authorities, the fuel may be transferred from a central fuel storage vault to the facility according to the approved procedure. Nuclear instrumentation will warn personnel against the occurrence of accidents during this phase of the program. No fuel will be stored in the facility (except for the minimum amount required in the reactor room during the loading operation). All fuel not used will be returned to the central vault following loading to the proper reactivity value. The detailed fuel loading procedure will be covered in an addendum to this report.

Removal of the fuel from the reactor core at the completion of the reactor testing program also requires the preparation of a detailed removal procedure, review of the procedure by the Safeguards Review Panel, and approval by the appropriate approval authorities. Upon approval, the removal operation will proceed exactly as specified in the removal procedure. If some unforeseen

development dictates any deviation from the established procedure, the operation will cease until all possible consequences of the deviation have been investigated, and, if necessary, the procedure has been resubmitted for approval. As the fuel is removed, it will be placed in shielded containers which will then be transferred to an appropriate facility for examination or storage. Final disposition of the fuel will be as directed by the AEC. Nuclear instrumentation will warn personnel of potentially hazardous situations during this phase of the program. Equipment used in fuel unloading, storage and shipment will be described in detail in a separate report.

B. ACCIDENT STUDIES

1. Nuclear Incidents

The occurrence of a nuclear excursion in a reactor represents one of the major potential hazards associated with the operation of a nuclear facility, because it may lead to the release of fission products accumulated in the core from normal reactor operation, as well as those formed during the excursion. The occurrence of such an accident is a direct result of an uncontrolled insertion of reactivity. This insertion of reactivity can be studied by assuming either that the reactivity change occurs at a given rate over a period of time, or instantaneously. The severity of the occurrence will depend on such conditions as when the accident is initiated and terminated; the magnitude of nuclear parameters, such as the prompt neutron lifetime; the effective delayed neutron fraction; and the partial temperature coefficients of reactivity.

For the purpose of this study, various conditions which could lead to or cause an excursion in the SNAP 8 Experimental Reactor were assumed and the results of such incidents were determined by means of analog computer techniques. These studies determined the magnitude of major incidents as well as the adequacy of the scram system. The results of these studies follow.

a. Ultimate Shutdown Mechanism

During a power excursion, the fuel element cladding will rupture, due to hydrogen pressure at moderately high temperatures. Calculations indicate that the fuel cladding rupture occurs at approximately 1830°F. Loss of approximately 12% of the hydrogen is sufficient to shut down the reactor irreversibly.

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Hence, the reactor cannot achieve criticality, even with all the drums fully inserted. Loss of approximately 4% of the hydrogen (equivalent to approximately 2% $\Delta k/k$) is sufficient to effect reactor shutdown without changing the position of any reflector elements. At 1830° F, this shutdown would occur in about 2 sec and at 2000° F in about 1/2 sec. The amount of hydrogen released for various fuel element temperatures as a function of time following rupture were shown in Figure IV-11.

b. Ramp Insertion of Reactivity at Startup

This incident assumes that the reactor is initially subcritical ($-\$0.50$) and isothermal at room temperature ($\sim 70^\circ \text{F}$). A control drum is then assumed to rotate in continuously at its maximum design reactivity input rate of 2¢/sec. The initial power level is 6 mw with a source of 5×10^6 neutrons/sec. The initial flow rate is 10% of rated (600 kw system).

S8ER contains many safeguards in the design of its instrumentation and control system which will prevent any reactor runaway during startup. These safety features are in the startup interlock, alarm, and scram circuit. The operation of these features is illustrated in Table VI-5. The first scram (3-sec period) occurs when the reactor power is approximately 6 watts. A signal from any one of two nuclear instrumentation channels will effect this scram, which occurs before any appreciable heat is generated. In the unlikely event that the short-period scram does not take place, a second alarm (high power) will sound. Immediately thereafter, a high neutron-flux signal from any one of three power-range channels will shutdown the reactor at a power level of 850 kw. This second scram provides adequate protection, because only enough energy has been generated to raise the average fuel temperature approximately 50° F. These circuitry features, coupled with the fact that the reactor will be subcritical with a cold, clean core (the most reactive core), even if three drums remain in the most reactive position and only three drums spring out, make a reactor runaway very improbable.

In the extremely unlikely event neither scram action occurs, the fuel element cladding will rupture about 5 sec after the start of the sensible heat generation. (This is about 74 sec after the start of the incident.) The release of hydrogen from the fuel elements, plus the rise of the coolant temperature,

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TABLE VI-5
STARTUP ACCIDENT SEQUENCE

Event	Time after start (sec)*
Start of incident	0.0
Reactor critical	25.0
Period alarm	50.0
Period scram	56.0
Power alarm	69.5
Power scram	69.7

will cause the coolant rupture disc located in the reactor coolant inlet line to rupture (releasing NaK to the compartment below the shutdown shield). This results in a loss of coolant flow a few seconds later as the EM pump cavitates and also prevents the establishment of any natural convective flow. Six sec after start of sensible heat generation, the hydrogen loss is sufficient to shut down the reactor, and by the time 10 sec have elapsed, the hydrogen loss is sufficient to prevent any further criticality even with all control drums fully inserted. The loss of hydrogen occurs preferentially across the core according to the temperature distribution. The loss of hydrogen from the center fuel elements tends to hold down the peak fuel temperatures; appreciable energy is required to remove the hydrogen from the fuel element (72,000 Btu/lb-mole are required to release hydrogen from the fuel element and there is approximately 1-1/4 lb-mole of hydrogen in the core). The cooling effect causes a decrease in the peak-to-average temperature ratio and holds the hot spot fuel temperature to less than 2100°F. Once the reactor power has decayed to decay heat power (~ 10 seconds after sensible heat generation) the heat losses from the reactor are at least 10 kw greater than the 30-kw decay heat power, and the overall reactor temperatures start to decay. There is, therefore, no melting of fuel elements as a result of the assumed incident.

Figure VI-1 shows a time-history plot of reactor power, average fuel element temperature, and coolant outlet temperature, as computed on the electronic analog computer for the first 5 sec of the incident after sensible heat generation. The dotted portions of the hand-calculated curve are based upon the Way-Wigner decay heat generation curve and the hydrogen loss curve (Figure IV-11).

Figure VI-1. S8ER Startup Incident
(2¢/sec reactivity ramp, 10% rated coolant flow)

The incident results in a 76-Mw-sec excursion. It should be noted that more than 200 Mw-sec of energy (including 95 Mw-sec to release all the hydrogen) is required to raise the core to the melting point (3300°F assumed). Since only 76 Mw-sec of energy is available from the excursion, core temperatures will not reach the melting point.

c. Ramp Insertion of Reactivity at Power

This incident assumes that the reactor is initially at rated conditions (600 kw, 1300°F reactor outlet temperature, and 140 gpm NaK flow) when a control drum is rotated in at its maximum design reactivity input rate of 2¢/sec.

Again, design safety features will normally shutdown the reactor. The sequence of events is shown in Table VI-6. Each scram can be triggered by a signal from one of three parallel instrument circuits. Again there are two scram modes to ensure protection. At the time of the high-power scram, the average fuel temperature would be 1380°F and the hot spot fuel temperature only 1430°F. In the unlikely event that neither operator action or the overflux scram circuit fail to correct the situation, the coolant-outlet-temperature scram circuit will scram the reactor when the outlet coolant temperature reaches 1375°F. This scram action will limit the average fuel temperature to approximately 1490°F and the hot spot fuel temperature only 1580°F, well below the point where any core damage can occur.

TABLE VI-6
POWER ACCIDENT SEQUENCE

Event	Time after start (sec)
Start of Incident	0.0
Power alarm	6.9
Temperature alarm	8.0
Power scram	8.4
Temperature scram	13.5

In the extremely unlikely event that this incident continues unchecked, the coolant rupture disc ruptures 27 sec after the start of the incident. Approximately 1 sec later the fuel element cladding ruptures due to internal hydrogen pressure. This rupture causes reactor shutdown. The rapid loss of approximately 10% of the hydrogen causes the fuel element temperature to decay to,

and level out at, about 1800° F. At this time the reactor is irreversibly shut down. The hydrogen loss cooling effect tends to prevent the center fuel temperatures from rising to excessive values, while the heat losses from the reactor are 10 kw greater than the decay heat power generation. Both of these effects act to prohibit any fuel-element melting.

Figure VI-2 shows a time-dependent plot of reactor power, average fuel element temperature, and core-coolant-outlet temperature as determined from the electronic analog computer. The dotted lines show the effect of hydrogen loss and are based upon hand calculations using the hydrogen loss curve, Figure IV-11. The incident results in a 55-Mw-sec excursion and is therefore, less severe than the startup accident.

2. Loss of Coolant Flow

There are, essentially, three ways in which coolant flow may be lost. These are (1) loss of coolant pump power, (2) inadvertant closing of the primary loop throttle valve, and (3) rupture of the primary system. The system rupture is discussed in Section VI-B-3. The other two incidents were studied and are discussed separately below.

a. Loss of Coolant Pump Power

In the event of power failure to the EM pump, primary coolant flow decays rapidly, causing the low primary-coolant-flow scram (set at 30% of rated) to operate, scrambling the reactor. In the unlikely event that the low-flow scram fails to operate, the high reactor-outlet-coolant-temperature scram provides sufficient backup protection to prevent any core damage. The system has sufficient natural convective flow, 5.1% of rated with a 200° F driving head, to prevent any excessive temperature rises. The outlet coolant temperature peaks at approximately 1400° F, the average fuel temperature peaks at about 1420° F, and the hot spot fuel temperature peaks at about 1450° F. Hence, there is no core damage.

In the extremely unlikely event that no scram action, automatic or manual, occurs, the increased fuel element temperature and upper grid plate temperature continue to cause the reactor power to decrease, due to the negative temperature coefficient of reactivity. At 25 sec after the start of the incident, the system pressure, owing to the increased coolant temperatures, reaches the rupture pressure of the rupture disc located on the inlet coolant line to the reactor. This breaks the natural convective flow path, resulting in complete

Figure VI-2. Ramp Insertion of Reactivity at Power
(2¢/sec reactivity ramp, rated 600 kw conditions)

flow stoppage. At 40 sec after the start of the incident, the fuel-element cladding ruptures, owing to the release of sufficient hydrogen by internal hydrogen pressure, to essentially shut the reactor down in the next few seconds. The loss of hydrogen also cools the fuel elements. Once shutdown has occurred, the fuel temperature rises are suppressed by added hydrogen loss. Also, after shutdown due to hydrogen loss, the core radiates approximately 10 kw more than is generated by the decay heat, causing a slow decrease in reactor temperatures. Hence, there will be no fuel element melting as a result of this assumed incident.

b. Inadvertant Closing of the Primary System Throttle Valve

The primary coolant system of the S8ER contains only one valve, the hand operated throttle valve. This incident assumes that the reactor is operating at rated power, rated temperature, and rated flow, when the throttle valve is closed linearly in 10 sec. As the throttle valve is hand operated through three or four remote linkages and requires 8 to 10 revolutions to close, it is improbable that closure in less than 10 sec could occur. As before, the first scram protection is provided by the low-flow scram circuit, and backup protection is provided by the high coolant-outlet-temperature scram, which operates about 8 sec after the start of the incident and before the coolant flow has ceased completely. This scram action prevents fuel element cladding failure as the heat losses from the system are slightly in excess of the decay heat generated. For either scram action, the coolant temperature peaks at about 1450° F, and the peak fuel temperature peaks at about 1500° F. Hence, there is no core damage.

In the extremely unlikely event that scram action, automatic or manual, does not occur, the rupture disc fails 25 sec after the start of the incident and the fuel-element cladding ruptures 10 sec later due to internal hydrogen pressure. This means that natural convective flow cannot be established even if the throttle valve were re-opened. Approximately 5 sec after fuel element cladding rupture occurs, the reactor is irreversibly shutdown. The preferential loss of hydrogen from the hotter fuel elements tends to decrease the peak-to-average temperature distribution across the core. This, combined with the fact that the heat radiated from the core exceeds the decay heat generation by approximately 10 kw, keeps the fuel element temperatures well below the melting point.

3. Loss of Coolant Incident

A loss of coolant is caused by a rupture of the primary coolant system. It is unlikely that the integrity of the primary system will ever be breached, as it operates at between 25 to 50 psia at between 1100 and 1300° F, while it is designed for 100 psig at 1400° F. The effect of a coolant rupture can be broken down into two problem areas, one involving the reactor core, and the other involving the pressure buildups in the various containment areas. These effects are discussed separately below.

a. Effect on the Reactor Core

This incident was studied by assuming the reactor to be initially at rated power, rated temperature and rated flow (600 kw system) when all the coolant is instantaneously lost. The core is protected by scram action, owing to loss of level in the surge tank, and by loss of coolant flow shortly after the EM pump cavitates. In the unlikely event that no scram action occurs, the fuel element temperature immediately begins to rise, causing the reactor power to decrease, owing to the negative temperature coefficient of reactivity. The fuel temperature continues to rise until it reaches 1830° F, the cladding rupture temperature, 25 sec after the start of the incident. At this time, hydrogen begins to leave the core rather rapidly, effecting a reactor shutdown. Once the reactor is shut down, the preferential loss of hydrogen from the hotter fuel elements tends to average the core temperature. The radiant heat losses from the reactor to the containment vessel are approximately 40 kw, while the decay heat generation is only 30 kw 10 sec after shutdown. Hence, the core temperatures will not exceed approximately 1850° F and no fuel element melting results from this assumed incident.

b. Effect in the Primary Coolant Vault

If a coolant system rupture occurs in the primary coolant vault, approximately 270 lb of NaK at 1300° F are released to the vault. For the purpose of this assumed incident, the vault liner and atmosphere cooling systems are assumed inoperative, and the EM pump is not shut off during the accident. The resulting pressure rise opens the pressure relief system (oil trap) at an overpressure of 0.75 psig. A fraction of the NaK and nitrogen atmosphere is then released to the outside atmosphere via the stack. The amount of NaK and

nitrogen released as a function of time is shown in Figures VI-3 and VI-4, respectively. If the relief system fails to operate, the pressure seal on the primary vault lid fails in excess of 1.25 psig, and approximately the same amount of NaK and nitrogen is expelled to the reactor room and stack.

Also shown in Figures VI-3 and VI-4 is the amount of NaK and nitrogen released from the primary coolant vault when the secondary system ruptures. In this incident, it is assumed that 630 lb of nonradioactive NaK at 1200° F is dumped into the vault. Table VI-7 shows the peak temperature and total amount of NaK and nitrogen released. (Calculation methods are shown in Appendix D).

TABLE VI-7

RELEASE OF NaK AND NITROGEN DUE TO PIPE RUPTURE IN THE
PRIMARY VAULT

Loop Broken	Atmosphere Overpressure (psi)	Atmosphere Temperature (° F)	Total NaK Expelled (lb)	Total N ₂ * Expelled (lb)
Primary	0.75	1090	6.69	25.3
Secondary	0.75	980	4.73	25.3

*44 lb nitrogen in vault at start of accident

c. Effect in Reactor Containment Vessel

The effect of a rupture in the reactor containment vessel is discussed as a part of the maximum credible accident described below (VI-B-5).

4. Potential Systems Interaction in Containment Vessel

Because portions of the primary NaK system, service systems, and control systems are in close proximity to each other in the containment vessel (water cooling system outside of vessel), the interactions between these systems which could lead to hazardous conditions were investigated. The possibility of interaction between NaK, water and/or air systems is of particular concern, since fires or explosions, as a result of NaK-water or air reactions, could violate the integrity of the system and lead to release of radioactivity. The events which could lead to unsafe conditions and the safeguards employed to prevent their occurrence are discussed in the following paragraphs.

Figure VI-3. Mass of NaK Lost From Primary Vault During Loss of Coolant Accident in Vault

Figure VI-4. Mass of Nitrogen Loss from Primary Vault During Loss of Coolant Accident in the Vault

An air-system line rupture or leak could be postulated to occur in the compartment above the shutdown shield or in the shutdown shield. A rupture above the shutdown shield would be detected by the high-pressure alarms and scrams on the containment vessel atmosphere (scram set at 15 psig).^{*} If it is assumed that the operator fails to take action or the scram fails, a maximum pressure of 120 psig could result in the containment vessel. Since the vessel and piping will withstand much greater pressures than 120 psig, and the rupture diaphragm will withstand a backpressure of 169 psig, there would be no interaction with other systems or hazardous consequences as a result of this incident.

A major rupture of an air cooling line within the shutdown shield would not be expected, because the lines are embedded in the lead and concrete in the steel-encased shield. An air-line leak within the shield, however, would cause a pressure rise which would be detected by the pressure detector in the shutdown-shield vent line. This detector would sound an alarm at 10 psig and the compressors could then be shut down by the operator in the control room. In the event that operator corrective action is not taken, it can be postulated that local high-pressure areas could develop in the shutdown shield, owing to these leaks. Investigation of this possibility indicates that local failure could occur at weld points and result in leakage of air to the containment vessel atmosphere. The consequences of this leakage are no different than those resulting from an air line rupture above the shutdown shield, as described above. The local stresses or deformations produced in the shutdown shield were also investigated and determined to be of insufficient magnitude to interfere with the operation of the control or safety system (particularly since there are six independent control systems). Air system rupture, therefore, would not cause or lead to hazardous conditions.

The possibility of primary NaK-system rupture leading to air system rupture and resulting in NaK-air reactions was also considered. Primary system rupture or rupture disc failure in the containment vessel is minimized by the primary system high pressure scram which will shut down the reactor (scram set at 40 psig, rupture disc set at 50 psig; the primary system is designed for 100 psig service). If the pressure scram fails to protect the system, the rupture disc would burst and NaK would be released to the area below the shutdown

^{*}It is presently planned to also add an oxygen analyzer on the containment vessel atmosphere. This will be described in the addendum to this report.

shield (See Figure III-6). Similarly, it may be postulated that the primary system core tank could rupture below the shutdown shield during a nuclear excursion and NaK would again be released below the shutdown shield. Because there are no air system cooling lines below the shutdown shield, neither of these two accidents could result in direct interaction between the NaK and air systems. Failure of the air system lines as a result of transient overpressure below the shutdown shield and movement of the shield are prevented by the holddown lugs on the upper surface and periphery of the shutdown shield assembly and plugs. The hold-down fixtures are capable of preventing movement of the shield and shield plugs during all credible accidents. It is concluded, therefore, that there would be no air-system rupture or NaK-air reactions as a result of NaK release.

Physical effects to the containment vessel and control system as a result of the release of hot NaK were also investigated. Because accidents can release NaK only to the area below the shutdown shield, and the control system drives and mechanisms are located above the shutdown shield, no adverse physical effects would be expected to occur to the control system. The puddle of hot NaK in the lower portion of the containment vessel would cause local stress in this section of the vessel (3/4-in. carbon-steel wall). Because the vessel cannot expand, due to the concrete surrounding it, some deformation (buckling) might result, however, the stresses would not be sufficient to cause failure of the containment vessel. There would be no possibility, therefore, of interaction between the NaK and the water in the cooling coils surrounding the containment vessel.

As a result of the above discussions, it is concluded that there are no credible interactions between the NaK, air, or water systems which would violate the integrity of the containment system or cause hazardous conditions.

5. Maximum Credible Accident

The maximum credible accident (MCA) in the S8ER is defined as a nuclear excursion followed by a release of NaK into the reactor containment vessel as the nuclear excursion continues unchecked. The incident assumes complete malfunction of corrective instrumentation and no remedial action by the operator. The following sequence of events occurring during the incident has been postulated.

A nuclear excursion occurs during startup of the reactor when reactivity is increased continuously at the maximum design rate of 2¢/sec (see Section VI-B-1, b). The fuel elements rapidly heat up to 1830°F approximately

25 sec after sensible heat generation commences, causing the cladding to rupture, which in turn causes rupture of the coolant system pressure relief disc located on the inlet coolant line. The power burst, 76 mw-seconds, heats the core up to the cladding bursting temperature and drives off approximately 6.7% of the hydrogen, effecting a reactor shutdown. The core then loses another 10% of its hydrogen, cooling down from 2050° F to 1850° F in the process, as decay heat generation is not sufficient to provide all the energy required (9000 Btu) to release 10% of the hydrogen. At this time, the hydrogen in the fuel is at equilibrium with the hydrogen in the vessel, and further hydrogen loss is prevented.

The pressure in the containment vessel rises rapidly to 167 psig, 20 sec after the fuel elements burst, and then decays as shown in Figure VI-5. During the pressure transient, there will be a maximum differential pressure of 25 psig across the shutdown shield. This differential pressure arises from the resistance to the flow of gas through the openings in the shutdown shield. Holddown lugs on the shutdown shield have sufficient strength to prevent any motion of the shield during the pressure transient. The temperature of the atmosphere in the containment vessel reaches 1790° F during the incident, if it is assumed that there is no containment vessel cooling. The vessel and surrounding concrete, however, provide a heat sink sufficient to absorb all the heat transferred from the reactor core and surrounding atmosphere during the incident. The maximum containment vessel wall temperature is 460° F, only 60° F above its normal operating temperature. Both the containment vessel and primary piping system are capable of withstanding this pressure for the short period involved.

The loss of hydrogen occurs preferentially across the core according to the temperature distribution. The loss of hydrogen from the center (hotter) fuel elements tends to hold down the peak fuel temperatures as it requires appreciable energy to remove the hydrogen from the fuel element. This cooling effect causes a decrease in the peak-to-average temperature ratio and holds the hot spot fuel temperature to less than 2100° F. Once the reactor power has decayed to decay heat power (at time $\sim$ 10 sec after sensible heat generation), the heat losses from the reactor are at least 10 kw greater than the 30 kw-decay heat power and the overall reactor temperatures start to decay. Therefore, there is no fuel element melting as a result of this assumed incident.

Figure VI-5. Containment Vessel Pressure Following Maximum Credible Accident

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During the course of this postulated incident, hydrogen loss from the fuel elements is not rapid enough to cause a violent disassembly of the core. Hence there is no possibility of creating "missiles".

VII. RADIOLOGICAL SIGNIFICANCE OF MAJOR ACCIDENTS

A. INTRODUCTION

The most serious potential hazard associated with a nuclear power reactor is that due to the fission product activity the reactor contains as a result of continuous long-term operation and/or the fission product activity which may be generated as a result of an accidental power excursion. The release of this activity from the fuel or normal containment barriers could present a significant hazard to operating personnel, personnel working in the immediate vicinity of the facility, and populations residing in the local environment. The ability to contain locally and/or effectively disperse the activity which might be released in the event of a reactor accident is, therefore, a most important safety feature of any reactor installation.

The basic containment barriers (described in previous sections) are the fuel cladding, the core vessel and closed loop primary system, the reactor containment vessel (pressure vessel), and the sealed primary system vault. In addition, local containment and dispersion is provided by the reactor building, which is maintained at a negative pressure by continuous ventilation to a stack.

If the normal barriers are penetrated following an accident, several accidents of significant radiological consequences are conceivable. In this section, an analysis of these consequences is made to evaluate the overall safety aspects of the S8ER reactor-facility-site complex. The assumptions used and a summary of the results of these analyses are presented herein.

B. SUMMARY

Four accidents and their radiation levels and resulting doses were postulated as follows:

- 1) Contained accident (activity released from fuel but contained in primary system loop)
- 2) Maximum credible accident (activity released to containment vessel atmosphere)
- 3) Loss-of-coolant accident (rupture of primary system in primary system vault)
- 4) Hypothetical accident (fuel melts, activity released to environment)

1. Contained Accident (activity released from fuel but contained in primary system loop)

In this accident it was assumed that the fuel cladding ruptures, and 4.15×10^5 curies of activity ($\sim 8\%$ of steady-state fission product inventory) are released from the core and are contained and circulated in the primary coolant loop. The only hazard from this accident is the direct radiation escaping from the vault to occupied areas. Approximately 10 sec after the accident, the maximum dose rate directly above the vault top shield would be approximately 0.7 mr/hr. Therefore, personnel exposures as a result of this accident are insignificant.

2. Maximum Credible Accident (activity released to containment vessel atmosphere)

In this accident it was assumed that 4.15×10^5 curies of fission product activity ($\sim 8\%$ of steady-state fission product inventory) is released to the containment vessel atmosphere as a result of failure of both the cladding and the primary system (i. e. core vessel failure or rupture disc relief due to excursion). Doses resulting from the excursion-produced fission products (76 Mw-sec) are negligible compared to those resulting from steady-state fission product inventories (full-power operation for 10,000 hr). Exposure of individuals as a result of this accident is due to (a) direct radiation escaping from activity contained in the dome of the containment vessel; (b) whole-body gamma exposure from submersion in the activity-filled reactor room or from the stack exhaust plume, both as a result of leakage from the containment vessel; and (c) inhalation doses resulting from exposure while in the reactor room or in the stack exhaust plume.

A summary of the direct radiation doses to individuals in the reactor room and control room as a result of activity contained in the dome of the containment vessel are shown in Table VII-1. Doses as a result of submersion in the activity-filled reactor room are also shown in the table.

Inspection of this table indicates that personnel could be in the reactor room for 10 min following the MCA without incurring undue risk under these emergency conditions (the individual in the reactor room would receive a 1.2 r whole-body gamma dose and a lifetime thyroid dose of ~ 73 rem). An individual at the control room would receive a whole-body gamma dose of less than 0.1 r in 10 min and 0.3 r in one hr.

TABLE VII-1
RADIATION DOSE RATES AND INTEGRATED DOSES TO
INDIVIDUALS IN THE BUILDING DURING MCA

Time After Incident	Direct Radiation From Containment Vessel Dome				Radiation From Submersion in Activity-Filled Reactor Room			
	3 ft From Dome		Console		Whole Body Dose		Inhalation Dose *	
	r/hr	r	r/hr	r	r/hr	r	rem/hr	rem
10 sec	15.	0	0.60	0	0.18	0	440	0
1 min	10.	0.2	0.40	0.01	0.12	0.003	440	6.
10 min	6.	1.2	0.24	0.05	0.07	0.02	440	73.
1 hr	4.	6.0	0.16	0.24	0.05	0.1	440	440.
24 hr†	2.	80.	0.08	3.2	0.03	1.0	400	10,000

*Total integrated lifetime thyroid dose from all iodine isotopes. Personnel in the control room would not be exposed to this hazard.

†Included in table to show trend in dose only since exposure for more than 10 min is unnecessary.

Hazards to personnel outside of the building in a downwind direction will be due primarily to inhalation of iodine activity (direct radiation at 100 ft from the building is less than 0.01 r/hr at one minute after the accident). Doses to on-site and off-site personnel as a result of stack release of activity released from the containment vessel are shown in Table VII-2.

TABLE VII-2
TOTAL INTEGRATED THYROID DOSE FROM ACTIVITY RELEASED
FROM BUILDING (On-site Doses for Fumigation Conditions
and Off-site Doses for Lapse Conditions)

Distance Downwind	Exposure Time and Dose (TID)				
	1 Hr	8 Hr	1 Day	1 Week	2 Weeks
Meters	rem	rem	rem	rem	rem
On-site					
100	0.92	7.2	22.0	130.	220.
500	0.21	1.7	5.0	30.	50.
1000	0.12	0.9	2.9	17.	28.
3000	0.05	0.4	1.2	7.	12.
Off-site					
3220*	0.005	0.04	0.12	0.7	1.2

*Nearest residence

Inspection of the table indicates that even if an individual remained in the center of the downwind plume at a distance of 100 meters from the facility for 8 hr, the total integrated lifetime thyroid dose he would receive would be only 7.2 rem. The corresponding dose to an individual at the nearest residence (3,220 meters) is 0.04 rem. In two weeks an individual at the nearest residence would receive a total lifetime dose of 1.2 rem, if it is assumed that the unfavorable weather conditions remained constant during this period and no corrective measures were taken to prevent leakage after the incident.

3. Loss-of-Coolant Accident (rupture of primary system in primary system vault)

In this accident it was assumed that the primary system ruptures in the vault and NaK coolant vapors are released to the stack as a result of the over-pressure in the vault. Doses to on-site or off-site personnel as a result of this accident are small; (5 mr whole-body gamma dose at 100 meters from facility as a result of the released activity). If fission products were also present in the coolant as a result of operation with defective fuel elements in the core (defects in 10% of elements as described in Section VI. A. 8) the additional full-body gamma dose would be about equal to the dose resulting from the Na^{24} and K^{42} . The inhalation dose from the released fission products is also insignificant. Therefore, the doses resulting from the loss-of-coolant accident are small and insignificant compared to doses from the maximum credible accident.

4. Hypothetical Accident

In this accident, it is assumed that the fuel melts at the end of the core life and all the released fission products are immediately dispersed into the reactor room and to the environment (4.15×10^5 curies released and all containment barriers voided). This accident is not credible and is presented for information only. Even under this incredible condition the total integrated lifetime doses at the nearest residence (3,220 meters) are only approximately 0.2 r whole-body gamma and 12 rem to the thyroid. Doses at closer distances and under unfavorable on-site weather conditions are shown in Figures VII-7 and VII-8.

C. FISSION PRODUCT RELEASE RATES FROM FUEL

Fission products released from fuel involved in reactor accidents depend primarily on the type of fuel and the conditions of time, temperature, and environment to which the fuel is subjected during an accident. The fuel used in the S8ER is a hydrided zirconium-uranium alloy with Hastelloy cladding. During

postulated accidents, the fuel may be subjected to either a NaK coolant or an inert atmosphere, under a variety of temperatures. Experimental evidence to indicate the fission product release rates under the accident conditions which are assumed in this hazards analysis (particularly temperatures) are not presently available for SNAP 8 fuel. Upper release rates, however, for the most severe accident conditions may be inferred from the results of the planned meltdown experiments performed at ORNL.⁹ The most representative data which may be obtained from these experiments appear to be those resulting from the melting of Zr-alloy-type fuel elements in air. These results are summarized in Table VII-3.

TABLE VII-3
FISSION PRODUCT RELEASE RATES
FROM MELTED FUEL
(ORNL Meltdown Experiment, 3300°F,
STR Element, 15% Burnup)

Isotope	Release Percent
Xe and Kr	100.0
Iodine	26.0
Cesium	10.0
Strontium	1.0
Barium	0.5

Because of the inherent shutdown mechanism associated with SNAP 8 fuel (cladding rupture and hydrogen release) it is unlikely that accident conditions can occur in the S8ER which will cause fuel temperatures as high as 3300°F. At lower temperatures (~2100°F reached in Maximum Credible Accident) it would be expected that fission product release would be less than those shown in Table VII-3. The best available calculated values for release fractions indicate that probably less than 1% of the total fission product inventories would be released as a result of overheated fuel caused by postulated excursions (no fuel melting). These release fractions are calculated on the basis of experimental measurements of hydrogen release from overheated, unirradiated SNAP fuel. These calculations, however, may not be applicable to S8ER, because accidents are postulated to occur at the end of long periods of full-power operations (10,000 hr at 600 kw). Release fractions (hydrogen or fission products)

under these conditions are not known and may be affected (higher than those predicted above) by radiation damage, thermal damage, or other effects associated with long periods of full-power operation. The more conservative values shown in the table, therefore, will be used for all major accidents studied in the analysis. The results of the experimental program, now in progress,¹⁰ to determine the release rates from SNAP fuel will show the degree of conservatism in this assumption. It may be noted that the figures in the above table are not significantly different than those resulting from the application of the AEC standard release.¹¹

D. FISSION PRODUCT INVENTORIES

The SNAP 8 core is designed to provide continuous full-power operation (600 kw) for 10,000 hr in space. The experimental program for the S8ER operation will be limited to a series of short low-power tests followed by two power-endurance runs of approximately 1500 and 2000 hr. Nevertheless, for the purpose of hazards analysis in this section, the total fission product inventory will be assumed to be that accumulated as a result of 10,000 hr of continuous operation at 600 kw. Additional fission-product activity as a result of a nuclear excursion will be based on a 76 Mw-sec excursion (MCA).

The gross fission-product gamma activity in the core, following an accident, may be approximated from the relationships shown in Reference 12. Specific fission product inventories may be obtained from the compilations of Blomeke and Todd.¹³

Figure VII-1 shows a plot of the total fission product inventories in the core as a function of time after shutdown. Fission product activity from steady-state operation is assumed to decay according to the function $t^{-0.21}$, and activity from the excursion according to the function $t^{-1.21}$, where t is in seconds. Inspection of the curve indicates that fission product activity resulting from the excursion is less than that resulting from the power run in approximately 30 sec, and rapidly becomes insignificant. Because doses over relatively long periods of time are under discussion in this analysis (5 min or greater) the doses resulting from release of excursion-produced fission products are not as important as those resulting from release of the steady-state fission products (integrated dose from excursion is negligible compared to doses from steady-state activity). Doses resulting from the excursion will therefore be ignored in the following analysis.

Figure VII-1. Total Fission Product Activity Due to 600 kw for 10,000-hr Steady-State Operation and a 76 Mw-sec Excursion

Table VII-4 shows a summary of specific fission product inventories in the core immediately after a 10,000 hr run. The table shows selected fission product groups and specific isotopes which may be of particular radiological significance for this study.

TABLE VII-4
FISSION PRODUCT INVENTORIES IN S8ER REACTOR
FOLLOWING 600-kwt OPERATION FOR 10,000 HR

Fission Product Group or Specific Isotope	Activity in Core Immediately After Shutdown	Mode of Decay
	curies	half-life as noted
Gross Gamma (max)	5,500,000	($t^{-0.21}$, t in sec)
Xe and Kr (total Beta)	324,000	
Xe 133	32,200	5.3 day
Kr 85	109	10.3 yr
I and Br (total Beta)	272,000	
I 131	14,700	8.05 day
I 132	22,200	2.4 hr
I 133	30,800	20.8 hr
I 134	38,400	52.5 min
I 135	29,900	6.68 hr
Cs (total Beta)	177,900	
Cs 137	835	26.6 yr
Sr (total Beta)	177,100	
Sr 89	24,200	54.0 day
Sr 90	840	28.0 yr
Ba (total Beta)	180,000	
Ba 140	31,900	12.8 day

Based on the release rates shown in Table VII-3, a maximum of 415,000 beta curies of fission product activity may be released from the core during an accident. For purposes of cloud gamma dose, it will be assumed that each disintegration of this mixture emits a 0.7-mev gamma ray, and that the activity decays according to the power function $t^{-0.21}$, where t is in sec. Because the released activity consists primarily of volatile fission products, these are conservative assumptions.

E. METEOROLOGICAL ASPECTS AND PARAMETERS

A description of the site and the general meteorological environment is given in Appendix A. For the purpose of evaluating the environmental hazards resulting from an accident, it is convenient to divide the affected areas into two separate regions called the "on-site area" and the "off-site area." This is done because the meteorological and topographical conditions of the site (mountainous) and populated areas (valley) are quite diverse, and the most unfavorable weather and activity release conditions do not occur simultaneously for the two regions. Because of this condition, the usual meteorological analysis of airborne activity transportation is not entirely suitable. The conditions discussed below are considered the most hazardous to each of the two regions.

1. On-Site

The on-site area is defined as the North American Aviation Field Test Area, which includes facilities of the Atomics International and Rocketdyne Divisions of North American Aviation. The boundaries of the area extend up to a maximum distance of 2-1/2 miles from the S8ER facility. Access is controlled and restricted to employees of North American Aviation, Inc. or authorized visitors.

For elevated releases, local experience¹⁴ has indicated that the most unfavorable on-site dispersion conditions exist when the weather type known as fumigation occurs. The base of the temperature inversion, which is present quite often at the site, generally rises to an elevation of 200 to 1000 ft during the day and then falls to ground level during the evening and early morning hours. If the released cloud or plume of radioactivity was trapped just below the inversion base, diffusion upwards would be very limited. In addition, because a neutral or mild lapse condition usually exists between the ground and the inversion base, diffusion of the cloud downward toward the ground would be fairly rapid. Therefore, to maximize the on-site doses from elevated releases, the inversion base will be assumed to be slightly higher than the elevation at which the release takes place (15 ft above stack exit assumed).

For ground release, the most unfavorable on-site dispersion conditions exist when inversion conditions prevail.

2. Off-Site

The minimum distance from the facility to the North American Aviation, Inc. Field Test Area boundary is approximately 300 ft. The terrain at this location, however, is steep (drops off rapidly to the surrounding valley) and uninhabited. The nearest residence is about 2 miles from the reactor, and the generally populated areas start at the valley floors at a minimum distance of 3 to 4 miles from the facility. Therefore, for practical purposes, the off-site areas can be assumed to start well beyond the site boundaries (several thousand feet). Because of these conditions, it is believed that the most unfavorable conditions for off-site exposure may be approximated by assuming that a mild lapse condition exists (Condition B in Appendix A) which results in an effective release height of about 200 ft relative to the valley floor. In other words, the off-site doses may be obtained by assuming a 200 ft release height with mild lapse conditions prevailing.

A summary of the conditions described in 1 and 2 above, and applicable meteorological parameters which will be used in the diffusion and dose calculations in this section, are shown in Table VII-5. Selection of the parameters is based on observations of local conditions and the recommendations of Atomics International and Rocketdyne meteorologists.*

TABLE VII-5
METEOROLOGICAL CONDITIONS AND PARAMETER SUMMARY[†]

Region	Release Type	Weather	C	C _y	C _z	n	u	H	h
on-site	elevated	fumigation	0.21	0.21	0.21	0.29	3	20	15
on-site	ground level	inversion	-	0.05	0.03	0.5	0.5	-	0
off-site	all types	mild lapse	0.21	-	-	0.29	3	-	66

[†] C = generalized diffusion coefficient for isotope turbulence ($m^{n/2}$)

C_y = diffusion coefficient in the cross wind plane ($m^{n/2}$)

C_z = diffusion coefficient in the vertical plane ($m^{n/2}$)

n = parameter associated with stability (dimensionless)

u = average wind speed (m/sec)

H = height of inversion base (m)

h = release height (m)

*See also Reference 14 and Appendix A.

F. ACCIDENT CONDITIONS AND CONSEQUENCES

1. Contained Accident (activity released from fuel but contained in primary coolant system)

In this accident, it is assumed that the cladding has ruptured and released fission product activity (maximum of 4.15×10^5 curies) which is circulated in the primary coolant system. Since the activity would be retained in the primary system, the only hazard from this accident would be that due to the direct radiation emanating from the vault to occupied areas (reactor room above vault or equipment pit). Figure VII-2 shows a plot of the radiation level in the reactor room above the vault top shield as a function of time after the accident. It can be seen from this graph that personnel exposures in this area are insignificant (0.68 mr/hr 10 sec after the accident). Maximum dose rates in the equipment pit could approach 1000 times the dose rates shown in the curve (<0.68 r/hr at 10 sec) as a result of streaming through wall penetrations. This area, however, is not an occupied area. The dose rates above the pit grating as a result of scattering are insignificant.

2. Maximum Credible Accident (activity released to containment vessel atmosphere)

In this accident (MCA described in Section VI.-B-5) it is assumed that the fission product activity is released to the containment vessel atmosphere as a result of failure of both the cladding and the primary system (i. e. core vessel failure or rupture disc relief). The hazard from this accident would be due to direct radiation to the reactor room as a result of the redistribution of the fission product activity in the containment vessel. In addition, exposure would result from fission product leakage from the containment vessel to the reactor room and environment.

a. Direct Radiation From Contained Activity

Since the activity released from the core (415,000 curies as described in Section VII C and D is composed primarily of volatile fission products and the compartments in the containment vessel are not isolated (not hermetically sealed), it is expected that a major portion of the released fission products might eventually be uniformly distributed throughout the entire containment vessel atmosphere. Portions of the released activity would, therefore, be contained (1) below the shutdown shield, (2) between the shutdown shield and the operating shield, and

Figure VII-2. Radiation in Reactor Room Due to Fission Product Activity Contained in Primary System

(3) in the void space between the operating shield and the concrete in the dome of the containment vessel (See Figure III-6). The free volume in these three spaces is 20 ft^3 , 30 ft^3 , and 1.5 ft^3 , respectively. Therefore, if uniform distribution is assumed, approximately 60% of the released activity would be contained in the compartment above the shutdown shield and 3% in the void space above the operating shield. Because of the massive shielding provided by the operating shield (4-1/2 ft of concrete), the major dose rates in the reactor room would be due to the activity in the void space above the operating shield.

Figures VII-3 and -4 show the calculated dose rates in the reactor room and control room as a result of the activity contained in the dome of the containment vessel. Integrated doses are also shown on the curves, assuming that it takes 10 sec for the released activity to reach the void space above the operating shield. For the purpose of these calculations, the activity was assumed (1) to be concentrated at a point source at the center of a cylindrical volume, (2) to have an effective energy of 0.7 mev, and (3) to be shielded by 4-3/4 in. of steel (See Figure III-6).

Inspection of the curves indicates that an individual would receive 1.2 r in 10 min if he were to remain at a distance of 3 ft from the containment vessel dome during the maximum credible accident. An individual at the console would receive only 0.04 r in 10 min and 0.23 r in 1 hr. In one day the direct radiation level 3 ft from the containment vessel dome would be approximately 2 r/hr and the corresponding dose rate at the console would be 0.08 r/hr.

b. Hazards From Fission Product Leakage

(1) Fission Product Release From Core to Containment Vessel

Fission product activities released from the core to the containment vessel may be obtained from Tables VII-3 and -4. For this analysis, both the hazards associated with gross fission products and I^{131} will be computed. The activities released from the core are, therefore:

$$\begin{aligned}\text{Gross fission products} &= 4.15 \times 10^5 \text{ curies} \\ \text{Iodine 131} &= 3.82 \times 10^3 \text{ curies}\end{aligned}$$

(2) Leakage From Containment Vessel

The containment vessel leakage will be based on the leak rate test specification of 1% loss of the free volume per day at 68 psig (See Section III. C.1).

Figure VII-3. Radiation in Reactor Room Due to Fission Products Contained in Dome of Containment Vessel (Individual 3 ft From Dome During MCA)

Figure VII-4. Radiation at Control Room Console Due to Fission Products Contained in Dome of Containment Vessel (MCA)

The pressure-time curve for the maximum credible accident is shown in Figure VI - 5. Inspection of this curve indicates that the pressure in the containment vessel builds up to a peak pressure of 167 psig in 20 sec following the accident and decays rapidly as a result of heat loss to the vessel and structures. The pressure falls to 68 psig in 3 min and to less than 50 psig in 10 min. After 10 min, the pressure decay is very slow, because it depends primarily on leakage from the containment vessel. Since the leak rate is not known as a function of pressure, an average overpressure of 68 psig will be assumed for this accident analysis. The corresponding leak rate will, therefore, be 1% per day for the duration of the incident. This results in a constant leak rate and constant release of fission products (no decay). The constant leak rate assumption will result in conservative downwind dose estimates, since relatively long exposure times (hours or days) are of primary interest at this point. Estimates of doses to personnel in the building should also be conservative, because a significant time period is required for the activity to escape from the fuel to the reactor room and to build up to the maximum fission-product concentration in the area.

(3) Activity in Reactor Room and at Stack Exit

The equilibrium fission product activity in the reactor room will depend on the room volume and ventilation rate. The ventilation rate is 5000 cfm ($2.36 \text{ m}^3/\text{sec}$) and the room volume is 425 m^3 . Thus, the air in the room will change every 3 min. In view of this rapid air flow rate it may be reasonably assumed that:

- a) Fission products will be removed from the reactor room at the same rate as they leak in from the containment vessel.
- b) There will be a uniform mixing of fission products in the reactor room.
- c) The concentration of fission products in the reactor room will be the same as in the exhaust air (stack exit).

The assumed, initial, fission-product concentrations in the reactor room (and at stack exit), as a result of a leakage rate of 1% per day and the above assumptions, would be:

$$\begin{aligned}\text{Gross fission products} &= \frac{4.15 \times 10^5 \text{ c} \times 10^6 \mu\text{c/cc} \times 1 \times 10^{-2} / \text{day}}{8.64 \times 10^4 \text{ sec/day} \times 2.36 \times 10^6 \text{ cc/sec}} \\ &= 2.04 \times 10^{-2} \mu\text{c/cc}\end{aligned}$$

$$\begin{aligned}\text{I}^{131} &= \frac{3.82 \times 10^3 \times 10^6 \times 1 \times 10^{-2}}{8.64 \times 10^4 \times 2.36 \times 10^6} \\ &= 1.87 \times 10^{-4} \mu\text{c/cc}\end{aligned}$$

These values and the appropriate decay factors were used to plot the curves shown in Figure VII-5 which depict the fission product and I^{131} -activity concentrations in the reactor room and at the stack exit as a function of time after the incident.

(4) Doses to Reactor Room Occupants

Table VII-6 summarizes the activity concentration (See Figure VII-5) and corresponding dose rates to occupants in the reactor room. Direct gamma radiation levels correspond to submersion dose rates that an occupant would receive if he stood in the center of the activity-filled room. For this calculation, the reactor room was assumed to be a hemispherical volume equal to the room volume with the receptor located at the center of a sphere twice the volume of the room. An effective gamma energy of 0.7 mev per disintegration was assumed. Beta dose rates to the surface of the bare skin were also investigated for the above geometrical conditions and for an assumed effective beta energy of 0.4 mev. The beta dose rates are approximately 8 times the gamma dose rates; these exposure rates, however, are not included in the table, because the whole-body gamma dose is considered to be more restrictive than local beta exposure (clothing significantly reduces the total potential beta exposure). The iodine dose rate is based on a breathing rate of $300 \times 10^{-6} \text{ m}^3/\text{sec}$, an I^{131} dose conversion factor of 1.45 rem/ μc inhaled and a factor of 1.5 to account for all other iodine isotopes associated with the activity release.^{15,16}

Inspection of Table VII-6 indicates that an occupant in the center of the activity-filled room would be exposed to a submersion (external gamma radiation) dose rate of 0.12 r/hr at one minute and 0.07 r/hr at 10 min. In 10 min he would receive a total whole-body dose of less than 0.02 r. These doses are

Figure VII-5. Fission Product Activities in Reactor Room and at Stack Exit Following the Maximum Credible Accident

small compared to the direct radiation dose that he would receive from the activity contained in the pressure vessel dome (See Section VII. F. 2. a).

In 10 min an occupant in the reactor room would receive a total integrated lifetime thyroid dose of approximately 73 rem as a result of inhalation of all the iodine isotopes. Following an accident there would be no need to remain in the reactor room for 10 min. In 5 min the integrated thyroid dose would be approximately 36 rem.

TABLE VII-6
ACTIVITY IN REACTOR ROOM AND DOSE RATES TO OCCUPANTS

Time After Accident (as noted)	Gross Fission Products			Iodine	
	$\mu\text{c/cc}^*$	curies	r/hr	$\mu\text{c/cc}^*$	rem/hr
1 min	8.8×10^{-3}	3.8	0.12	1.9×10^{-4}	440
10 min	5.4×10^{-3}	2.3	0.07	1.9×10^{-4}	440
1 hr	3.7×10^{-3}	1.6	0.050	1.9×10^{-4}	440
8 hr	2.4×10^{-3}	1.0	0.032	1.8×10^{-4}	420
1 day	1.9×10^{-3}	0.81	0.026	1.7×10^{-4}	400

*Also corresponds to stack exit concentrations. The listed concentration for iodine is that due to I^{131} only; the corresponding dose rates includes the dose from all iodine isotopes.

(5) Downwind Doses

Cloud gamma and inhalation dose rates to personnel outside the building or in a downwind direction will be less than those shown in Table VII-6 because of geometry and meteorological dilution factors. In addition, cloud gamma dose rates will also be less as a result of the rapid decay of the gross fission product activity in the cloud. Therefore, because the direct submersion dose hazards to an occupant in the building are small compared to the hazard from the inhalation dose, the major downwind hazard will be due to the dose resulting from iodine inhalation.

To obtain the total downwind inhalation dose, a dose rate based on the initial I^{131} concentration and a one-hr exposure time was calculated from the meteorological formulas¹² described below. (Results obtained in rem/hr of exposure for maximum and constant I^{131} concentration.) The total integrated dose was then obtained by multiplying the dose rate by 1.5 to account for exposure

from other iodine isotopes and graphically integrating the resulting dose rate as a function of exposure time and the I^{131} decay rate.

(a) For On-Site Doses

$$TID = \frac{Q}{\pi^{1/2} C_y \bar{u} H x^{(2-n)/2}} (Btk) ,$$

where

TID = total integrated lifetime dose from I^{131} activity inhaled in the first hour of exposure, rem

Q = curies released/ sec, 4.48×10^{-4} c/sec (from Table VII-6 and a 5,000 cfm exhaust rate)

$C_y, \bar{u}, H, n$ = appropriate conditions as defined in Table VII-5 for on-site doses under fumigation conditions

x = distance from reactor to downwind receptor, meters

B = breathing rate, 300×10^{-6} m³/sec

t = exposure time, 3600 sec

k = dose conversion factor, 1.45×10^6 rem/c

(b) For Off-Site Doses

$$TID = \frac{2Q}{\pi C^2 \bar{u} x^{(2-n)}} \exp \left[\frac{-h^2}{C^2 x^{(2-n)}} \right] Btk$$

where

C, $\bar{u}$, h, n = appropriate conditions as defined in Table VII-5 for off-site doses under lapse conditions,

and other parameters are as defined above.

Calculated doses as a function of downwind distance and selected exposure times are shown in Table VII-7.

3. Loss-of-Coolant Accident (rupture of primary system in primary system vault)

In this accident it is assumed that the primary system ruptures in the vault and releases NaK coolant to the vault (See Section VI. B. 3 for description

TABLE VII-7
TOTAL INTEGRATED THYROID DOSE FROM ACTIVITY
RELEASED FROM BUILDING DURING MCA
(On-site Doses for Fumigation Conditions
and Off-site Doses for Lapse Conditions)

Distance Downwind Meters	Exposure Time and Dose (TID)				
	1 Hr rem	8 Hr rem	1 Day rem	1 Week rem	2 Weeks rem
On-site					
100	0.92	7.2	22.	130.	220.
200	0.51	4.0	12.	72.	120.
500	0.21	1.7	5.	30.	50.
1000	0.12	0.94	2.9	17.	28.
2000	0.07	0.55	1.7	10.	17.
3000	0.05	0.40	1.2	7.	12.
Off-site					
3220*	0.005	0.04	0.12	0.7	1.2

*Nearest residence

of accident). The vault overpressure resulting from the accident is relieved to the stack through an oil-filled U-tube. Because the reactor would scram and remain intact in the event of the accident, the potential hazard is due to Na²⁴ and K⁴² activity released from the vault to the environment. The total activity inventory in the primary coolant is 257 curies of Na²⁴ and 116 curies of K⁴². The amount of NaK released from the vault to the environment was determined to be approximately 7 lb (See Section VI. B. 3) or 2.22% of the total NaK inventory (total NaK inventory = 315 lb). Therefore, approximately 5.7 curies of Na²⁴ and 2.58 curies of K⁴² are released to the environment as a result of the accident.

The total integrated whole-body cloud gamma dose to a downwind receptor, under unfavorable weather conditions, was approximated as follows.

a. For Off-Site Doses

$$TID = \frac{2Qf}{\pi C_y C_z \bar{u} x^{(2-n)}} \left\{ \exp \left[- \frac{h^2}{C_y C_z x^{(2-n)}} \right] \right\} (0.246 E_\gamma)$$

where

TID = total integrated whole-body cloud gamma dose (rem)

Q = total activity inventory, 257 curies of Na²⁴ and 116 curies of K⁴²

f = fraction of inventory released to environment, 2.22%

$h, C_y, C_z, n, \bar{u}$ = appropriate parameters and weather conditions from Table VII-5

x = distance from stack to downwind receptor (meters)

E_γ = average photon energy per disintegration, 2.8 mev¹⁷

b. For On-Site Doses

$$TID = \frac{Qf}{\pi^{1/2} C_y \bar{u} H x^{(2-n)/2}} (0.246 E_\gamma)$$

where

H = 20 meters (from Table VII-5), and other parameters are as defined above.

Doses resulting from the above accident are shown in Figure VII-6. It can be seen from this curve that personnel exposures as a result of this accident are small.

It is also conceivable that fission products would be released to the environment as a result of this accident, if the reactor has been operating with defective fuel elements. For the purpose of this accident, it will be assumed that the fraction of I, Br, and nonvolatile activity released from the vault will be proportional to the fraction of the NaK coolant expelled from the vault, and the fraction of the Xe and Kr activity released from the vault will be proportional to the fraction of the nitrogen atmosphere expelled from the vault. The fraction of NaK expelled from the vault was determined to be 2.22% of the total inventory (see calculations for NaK release above). The fraction of the nitrogen atmosphere expelled from the vault during the accident is 57.5% (Section VI.B.3). In addition, from the discussion and calculations in Section VI.A.8, it was postulated that the primary NaK system could contain a maximum of 0.017% of the gross fission inventory at shutdown. Therefore, the activity released to the environment as a result of this incident and the shutdown inventories shown in Table VII-4 are:

Figure VII-6. Downwind Whole Body Cloud Gamma Dose Due to Stack Release of Primary NaK Coolant

Xe and Kr = 31.6 c

I and Br = 1.0 c (including 0.06 curies of I^{131})

Other = 16.6 c (based on 80% of gross inventory)

Total Released = 49.2 c

Downwind doses resulting from the above release were investigated by using the formula described above (for NaK release), except that Q_f was replaced by $49.2 t^{-0.21}$, where t is the time for the activity to reach downwind distance x in seconds, and E_γ was replaced by 0.7 mev, the assumed effective gamma energy of the fission product mixture. On-site cloud gamma doses for the release are shown on Figure VII-6. Inspection of the curve indicates that the doses from the fission product release are about equal to the doses resulting from the Na^{24} and K^{42} activity in the cloud. Because a major portion of the released activity is Xe and Kr and only a small fraction of the iodine is released, the inhalation dose is negligible. Therefore, the doses resulting from the loss-of-coolant accident are small compared to doses from the maximum credible accident.

4. Hypothetical Accident

To fully evaluate the safety aspects of the site and ultimate risk (particularly with regard to off-site areas) it is of interest to consider the consequences of a hypothetical accident which assumes that all containment barriers are voided. This can be conservatively accomplished by assuming that the fuel melts at the end of the core life and all the released fission products are immediately dispersed into the reactor room and environment (release of 4.15×10^5 curies from fuel as described in Section VII. C and D).

For this accident the total integrated dose to a downwind receptor as a result of inhaled activity and under inversion and lapse weather conditions was calculated as follows:

$$TID = \frac{2 Q_o f_o e^{-\lambda t}}{\pi C_y C_z \bar{\mu} (x + x_o)^{2-n}} \left\{ \exp \left[-\frac{h^2}{C_y C_z (x + x_o)^{2-n}} \right] \right\} (Bk)$$

where

TID = total integrated dose (rem)

Q_o = inventory in reactor at shutdown, [curies (from Table VII-4)]

f_o = fraction of inventory released from fuel (from Table VII-3)

λ = decay constant for isotope of interest, [sec (from Table VII-4)]

t = time for cloud to reach downwind receptor (sec)

$h, C_y, C_z, n, \bar{u}$ = appropriate weather conditions as defined in Table VII-5

$x = \bar{u}t$ = distance from reactor to downwind receptors (meters)

$x_o = (\bar{u}t)_o$ = distance required from reactor to virtual upwind source point to produce original concentration in cloud (meters)

$$= \left(\frac{2V}{\pi^{3/2} C^3} \right)^{2/[3(2-n)]}$$

V = volume of reactor room, 425 meters³

B = breathing rate, 300×10^{-6} m³/sec

K = dose conversion factor (rem/curie inhaled as tabulated below)^{15,16}

Isotope	rem/curie Inhaled
I 131	1.45×10^6
I 132	0.053×10^6
I 133	0.399×10^6
I 134	0.025×10^6
I 135	0.125×10^6

For fumigation conditions,

$$TID = \frac{Q_o f_o e^{-\lambda t}}{\pi^{1/2} C_y \bar{u} H (x + x_o)^{(2-n)/2}} (Bk)$$

where

H = height of inversion base, 20 meters (from Table VII-5), and other parameters are as defined above.

The whole-body cloud gamma dose to a downwind receptor was approximated from the above formula by making the following substitutions.

- Replace $Q_o f_o$ with Q , where Q = total gamma curies released or 4.15×10^5 curies
- Replace $e^{-\lambda t}$ with $(t + t_o)^{-0.21}$, where $t_o = 10$ sec (time for activity to escape from fuel to reactor room), and t is as defined above
- Replace Bk with $0.246 E_\gamma$, where E_γ = photon energy in Mev or 0.7 Mev

Doses derived from the above formula are shown in Figures VII-7 and -8. Inspection of these curves indicates that the total integrated doses at the nearest residence (3220 meters) are approximately 0.2 r whole-body gamma and 12 rem to the thyroid. However, the total integrated on-site doses (particularly thyroid dose) would be excessive unless evacuation could be effected. For example, the total integrated doses, if a person remained at 100 meters while the center of the plume (stack release) passed over his position, would result in a whole-body gamma dose of 20 r and a thyroid dose of 1000 rem. The inversion dose curve is included in the figure for reference only, since it is impossible to conceive of an accident which would destroy the building and stack. It should be noted that these accidents are not credible and are presented here for information only.

Figure VII-7. Downwind Whole Body Cloud Gamma Dose Due to Hypothetical Accident

Figure VII-8. Downwind Thyroid Dose Due to Maximum Hypothetical Accident

APPENDIX A

DESCRIPTION OF SITE (NAA FIELD TEST AREA)

A. LOCATION

The Atomics International Field Test Laboratories area is located within the limits of North American Aviation's Field Test Area in the Simi Hills of Ventura County, California. The Test Area is approximately 30 mi west-northwest of downtown Los Angeles. Figures A-1 and -2 show the relation of the Test Area to Greater Los Angeles and to neighboring small towns.

The Field Test Lab area is in the western portion of the NAA Test Area and almost entirely in a flat region of the Simi Hills known as Burro Flats.

B. LOCAL TERRAIN

The Simi Hills are a rugged outcropping of sandstone strata, defining the west border of the San Fernando Valley and the south and east borders of the Simi Valley. The maximum elevation of the Simi Hills is about 2400 ft above mean sea level. The elevation of the San Fernando and Simi Valley Floors is approximately 900 ft.

C. POPULATION DISTRIBUTION

Figure A-1 shows the surrounding population distribution, including the Greater Los Angeles area. Each dot represents 1000 persons; the solid dots denote the population in October 1958, and the open dots give the projected distribution for 1980.

The area immediately surrounding the AI Field Test Lab area is extremely hilly and contains no residences. The Simi Valley to the north is the nearest populated region and contains the towns of Simi and Santa Susana (Figure A-2). The closest residences are approximately 2 mi north of the Field Test Area. The entire Simi Valley, ranging from 2 to 6 mi from the Test Area, contains approximately 12,000 persons at the present time. The nearest towns to the east are each about 7 mi from the site and have the following estimated populations as of July 1, 1960: Chatsworth, 21,700; Canoga Park, 56,000; and Woodland Hills, 23,700. Population distribution is relatively sparse in other directions.

Figure A-1. Distribution of Population, Los Angeles and Ventura Areas

Figure A-2. Location of the North American Aviation Field Test Area Relative to Surrounding Communities

D. UTILITIES

Figure A-3 shows the electrical and water utilities serving the AI Field Lab area. With the exception of the 66-kv transmission line traversing the area, all utilities shown are solely for the service of test and operational facilities within the NAA Field Test area.

E. SEISMOLOGY

A map showing the active earthquake fault locations in the region is shown in Figure A-4. The nearest active fault is the San Gabriel Fault, approximately 15 mi northeast of the Field Test area.

F. GEOLOGY

The Simi Hills are composed primarily of marine sediments consisting of tightly cemented sandstones and shales (Chico formation) of Upper Cretaceous Age. These are relatively impervious layers tilted so that they have a strike of approximately 325° (35° west of north) and a dip of approximately 22° . A geologic map of Zone 3, Ventura County, bordering the site is presented in Figure A-5.

The exposed strata have a maximum thickness of about 30 ft. In many places they are badly fractured, and large monoliths have broken loose and fallen from their original strata and lie scattered on the hillsides. The shales decompose rapidly when exposed to the weather. The sandstones also decompose but at much slower rate.

Burro Flats consists of alluvial deposits of a recent geological age.

G. GROUND WATER HYDROLOGY

Compared to the quite impervious sedimentary strata, the alluvium derived from it is relatively pervious to water. The alluvial deposits under the Field Test Area are restricted to the small flat valley in the Burro Flats region. Prehistoric faults in the nearby rocks hold water well, indicating that they are not interconnected with any underground water-bearing strata. These faults are believed to be short and discontinuous.

The first two wells drilled on the land used for the NAA Field Test Area produced 2 and 11 gpm. Wells were then drilled in inactive fault zones in the

Figure A-3. Utilities in the Atomics International Nuclear Development Field Lab Area

Figure A-4. Active Earthquake Faults, Los Angeles and Vicinity

Figure A-5. Geologic Map, Zone 3, Ventura County, California

area and found to be good, one producing 185 gpm (the capacity of the pump) for 8 hr with only an 8-ft drawdown. The fissures and fractures of the faults contain a water-bearing aquifer, but are apparently tightly enclosed by relatively impervious rock. The aquifers are apparently not replenished by rain water percolating directly through the soil to the aquifer or by stream beds crossing the pervious fault zone.

Even though it is believed that the permeability of the stratified sedimentary rock is very low, this rock is many times more permeable parallel to the bedding plane than perpendicular to the bedding plane. It is possible that some of the strata could be semipervious and allow a fluid to move parallel to the bedding plane. However, planes intersecting the Field Test area slope away from the San Fernando Valley, and dip far below the Simi Valley (assuming a constant dip of 22° , corresponding to the dip near the surface).

It is concluded that any radioactive material being accidentally released into the surrounding substrata would very slowly permeate far below the land surface, and remain there unless pumped out of one of the local wells within the NAA Field Test area. The waters in the wells are regularly monitored for radioactivity and are not used for drinking.

H. SURFACE WATER HYDROLOGY

The principal watershed areas and the general surface drainage in the Los Angeles area are shown in Figure A-6. The Los Angeles River drains the San Fernando Valley, and its drainage area includes the Field Test area. Most of the water comes from the higher San Gabriel, and Santa Monica Mountains.

The manner in which surface drainage from the Field Test area reaches populated areas is shown in the insert of Figure A-7. The solid line shows the direct line of flow from the Burro Flats region. Bell Creek, which carries the drainage from the facility, flows into Canoga Park about 2 mi south of Chatsworth Reservoir. The possibility of contaminating the Chatsworth Reservoir is extremely remote, since neither ground water nor surface water drainage from the site passes near the reservoir or its source.

In Canoga Park, Bell Creek flows into the Los Angeles River, which passes through Canoga Park and the southern edge of Reseda. Flow from there is into the Sepulveda Flood Control Basin, where water may be held up before flowing through the City of Los Angeles.

Figure A-6. Principal Watershed Areas, Los Angeles County

Figure A-7. Surface Water Drainage from the SNAP Building Complex

I. METEOROLOGY AND CLIMATOLOGY

The Los Angeles basin is a semi-arid region controlled for the most part by the semi-permanent Pacific high-pressure cell. The seasonal changes in the position of this cell influence to a great extent the weather conditions over the area. The Pacific high-pressure cell extends from Hawaii to the Southern California coast, with the associated subsidence inversion tilting downward from Hawaii to the California coastline. During the summer season the high is displaced to the north, resulting in mostly clear skies with little precipitation. In the winter the Pacific high moves south far enough to allow some of the Aleutian lows and the associated fronts to move through the area, producing moderate precipitation. Due to stronger pressure gradients and portal passages, the winds are predominantly from a northerly direction.

The U.S. Weather Bureau takes complete surface observations at 3 locations in the Los Angeles area: The City Office in downtown Los Angeles, 29 mi southeast of the reactor site; the International Airport, 28 mi southeast of the site; and the Lockheed Air Terminal in Burbank, 22 mi east of the site. The Burbank station is the nearest to the reactor site but has an elevation of 700 ft, which is about 1100 ft below the site.

It is estimated that the average winter temperature at the Field Test area is approximately 4° cooler than at Burbank, and the summer temperature about 4° warmer. The daily range of temperature is approximately the same as at Burbank. Approximately 30 days of the year the temperature at the site may be expected to go above 90°F, to a high of about 100°F each year, while the temperature may drop below 32°F on 5 days each year to a low of about 28°F. In 50 years the maximum may be expected to be 106°F and the minimum 18°F. The temperature, in general, is mild as the morning stratus and sea breeze prevent it from rising unduly in the summer and the low latitude and ocean influence keep it from falling in winter.

The precipitation is extremely variable, with many years having less than half the mean amount. Similarly, many years have no snowfall at all, the occasional year with a moderately heavy fall accounting for the measurable average. The average and heaviest precipitation statistics are shown in Table A-I. Observations in the vicinity of the site, combined with related data from other areas, indicate approximate surface wind conditions as shown in Table A-II.

TABLE A-I

MEANS AND EXTREMES IN PRECIPITATION FOR PERIOD OF RECORD

	Mean	Heaviest (50-year probable) (in.)
Annual	16.0	40
Summer half yr (May-Oct)	1.0	7
Winter half yr (Nov-Apr)	15.0	38
Rainiest month (Feb)	3.5	15
Rainiest 24 hr in 50 years	-	8.5
Annual snow	1	7
Max. 24-hr snow	-	6
Number of days with 0.01 in. or more	40	-

TABLE A-II

SURFACE WIND CONDITIONS

	Summer	Winter
Prevailing afternoon direction	WNW	NW
Prevailing early morning direction	ESE	E
Average daytime speed	8 mph	6 mph
Average nighttime speed	3 mph	2 mph
Max. wind (1-min average)	30 mph at least once per yr 55 mph once in 50 yr	

During the early morning the surface wind passes over Burro Flats into the Simi Valley. In the afternoon the wind is more inclined to be directed toward the San Fernando Valley.

At a considerable elevation above the mountains the influence of the ground on the wind becomes unimportant, so that upper winds (perhaps above 6500 ft) are controlled almost entirely by the large-scale pressure systems and may not vary much over a few miles of distance. On this basis the upper winds above Burbank may be presumed to be quite similar to the upper winds above the site. Typical upper wind data from the Burbank station are shown graphically in Figure A-8. The estimated upper winds above the site are given in Table A-III.

BURBANK STATION (APPROXIMATELY 22 MILES FROM FIELD TEST AREA)

Figure A-8. Typical Upper Wind Data, Los Angeles Area

TABLE A-III
UPPER WINDS

	Summer			Winter		
Elevation (ft)	3,250	9,750	16,500	3,250	9,750	16,500
Prevailing direction	SSE	SW	SW	NNW	NW	NNW
Average speed in prevailing direction (mph)	5	12	15	5	15	20

A subsidence inversion is present almost every day during the summer months and frequently in other months. However, the elevation of the facility is several hundred feet greater than the height of the average inversion base that exists over the populated areas in the valleys below. A ground inversion is produced by radiation cooling during clear nights in fall, winter, and spring. The nocturnal inversions produced on clear nights are associated with a tendency for the cooled air to drain downward from the slopes of the hills near which the site is located. It is unlikely, however, that the air moving down the slope is ever initially cold enough to drain all the way to the valley floor without compressionally heating to the point where it would move out over the colder air layer covering the valley floor. During the day, the diurnal heating tends to establish a stirred layer with adiabatic lapse rate immediately above the hills.

The site experiences clear days and nights through most of the summer. In other seasons, particularly in spring, stratus is present over the site, when inversions are high. Spring and fall are periods of greatest frequency of fog. Cold air drainage reduces the likelihood of fog in winter. During winter, higher clouds are frequent, associated with fronts and storms.

Table A-IV presents an estimate of the average number of days per month which are clear, partly cloudy, and cloudy during the daytime at the Field Test area.

1. Factors Governing Effluent Cloud Paths

Any effluent released into the atmosphere at the site will follow a course dictated by the wind flow over the site. The idealized flow over an obstruction such as the Simi Hills Range, considering only fluid dynamics, is shown in Figure A-9. This indicates that the downward penetration of a cloud into the

valley is dependent upon which streamline the cloud is forced to follow, i.e., the height above the site at which vertical equilibrium is established. From this consideration, it appears that a hot cloud, which achieves a rise due to thermal buoyancy, presents a less serious problem than a cold release.

TABLE A-IV
AVERAGE OVERCAST CONDITIONS

Month	Number of Days per Month		
	Clear	Partly Cloudy	Cloudy
January	15	8	8
March	14	9	8
May	13	11	7
July	25	5	1
September	22	7	1
November	19	7	4

Figure A-9. Idealized Air Flow From Burro Flats

It is fortunate that flow over the Simi Hills is not governed strictly by fluid dynamics. Local meteorological conditions, for the most part, tend to inhibit the downward component of the wind toward the valley floors, and consequently reduce the number of potentially unfavorable situations.

This alteration of the idealized flow path is accomplished in a number of ways.

a. Wind Direction and Turbulence

The WNW wind at Burro Flats is primarily a flow from the Simi Valley extending into the western San Fernando Valley both by momentum and pressure gradients set up by the San Fernando Valley's heating more rapidly than the Simi Valley. This flow has depth, and its movement across the Simi Hills can be depicted as a column of air traversing an obstruction (Figure A-10). This column can be thought of as being made up of many air particles, each of which possesses its own vorticity. The vorticity, in this case, is a measure of the curvature of the flow in the vertical plane; a particle on streamline B has less (anticyclonic) vorticity than one on streamline A. In the zone of packed streamlines over the site, turbulence in the flow serves as a mechanism for interchange of particles from one streamline to another. This results in a trajectory which is higher than one derived from purely laminar flow, with the height of the trajectory directly related to the degree of turbulence.

Figure A-10. Air Flow Over the Simi Hills

b. Elevation

The fortunate location of Burro Flats approximately 935 ft above adjacent valley floors, provides an environment which, most of the time, naturally protects the San Fernando Valley population from site-generated pollution. As shown in Figure A-11 situations (a), (c), and (d) all provide a measure of protection, by either restricting diffusion by stability over the valley, as in (a) and (c), or by achieving a high cloud trajectory as well as good diffusion by instability over Burro Flats, as in (d).

The flow toward the Simi Valley normally does not occur under such favorable conditions. Actually, the ESE winds, primarily an early morning phenomenon in spring and summer, are weaker than the WSW winds, and are coincident with more stable conditions over the site. The situation most closely associated with flow toward the Simi Valley would be Figure A-11 (b) (with the direction arrow pointing WNW).

c. Location

The flow over Burro Flats is so predominately ESE or WNW that locations other than the eastern Simi and western San Fernando Valleys have little opportunity to become involved with pollution from the site. During the autumn and winter there is a high incidence of north or northeast wind which would carry the effluent toward the Agoura-Thousand Oaks area. However, hazard to these areas is negligible because of:

- 1) Distance (6 to 8 mi from Burro Flats).
- 2) Flow being parallel to Simi Hills spine, fostering good diffusion and little downward component.
- 3) Morning conditions, which would closely resemble the situation on Figure A-11 (a) (with direction arrow pointing SW).

Winds from directions other than those previously mentioned are generally of short duration and quite rare. Therefore, it appears that no additional off-site locations should be concerned with site-generated effluents.

d. Release Time

A complete analysis of the consequences of a radioactive discharge must include the possibility of a release over an extended period of time. The

Figure A-11. Short Burst Cloud Dispersion

effluent cloud would take on the appearance of an elongated plume (Figure A-12) the size, shape, and path of which result from the same variations in atmospheric stability as on those depicted in Figure A-11.

As in the case of the instantaneous or short-duration release, the worst case is the one with stable air over Burro Flats and unstable air over the valleys, Figure A-12 (b). This condition exists less than 5% of the normal operating time. The resulting pollution at any particular location is determined by the following:

- 1) The rate of release which governs the concentrations along the plume axis. Slow rates will hold amounts to a harmless level.
- 2) Residence time at a particular location.
- 3) Diffusion in the direction of wind flow. This is negligible in an elongated plume.

Every effort will be made to minimize the hazard by holding the release of pollutants to the atmosphere to the slowest possible rate and under favorable meteorological conditions.

Figure A-12. Continuous Cloud Dispersion

APPENDIX B

OPERATING ORGANIZATION, RESPONSIBILITIES, PROCEDURES, AND QUALIFICATIONS OF PERSONNEL

The S8ER will be operated by the SNAP 8 reactor experimental unit of the Advanced Reactor Development Department. The organization structure of the operating component and its relation to safety organizations is shown in Figure B-1.

The shift supervisor on duty will be responsible to the reactor experimental unit supervisor for the safe and efficient operation of the S8ER and for the safety of personnel in the building. He is given sufficient authority to ensure that he can easily control personnel access within the building and can terminate the reactor operation at any time for reasons of safety. A list of personnel qualified to fill shift operating positions will be maintained at all times in the control room.

Detailed responsibilities of operating personnel will be clearly delineated in the operating procedures manual. An up-to-date copy of the manual will be maintained in the control room. This manual will contain all operating limits and any mandatory actions required to operate the reactor safely. Emergency procedures and a list of responsibilities assigned to operating personnel will be included in the manual. The contents of this manual will be specifically limited to emphasize its importance, and to provide directives which operating personnel can follow without reference. Revisions to the operating procedures manual must be approved by the appropriate supervision and approval authorities defined by Atomics International Standard Operating Procedures.

A current operating reference book will be maintained in the control room. This book will be a guide to methods of achieving desired results and an aid in training operating personnel. Stepwise procedures for performing repetitive operations or for performing operations which are important enough to merit written instruction will be detailed in this manual. In addition to procedural information, descriptions of the SNAP 8 reactor and experimental facility will be included in the operating reference book.

Operating personnel for the S8ER have been selected to provide experienced, technically competent operating crews. Each shift supervisor has had previous

*APPROVAL AUTHORITY

Figure B-1. Organization Chart Showing Relationship Between Safeguards Review Panel and Operating Division

reactor operating experience. At least two members of each initial startup crew have undergone training on shift at the SNAP 2 Development Reactor to ensure their familiarity with compact reactor operation. In addition, operating personnel will study appropriate reference material, and will attend lectures to familiarize them with the technical details of their jobs. Qualifications of operating personnel will be based on written examination, observed performance, and oral examination. After successful completion of the training program, prospective operators, engineers, and shift supervisors will be recommended by the supervisor of the reactor experimental unit to the approval authority for designation as qualified to operate S8ER.

More detailed operating procedures, responsibilities, and qualifications and training of operating personnel, and the operating organization will be described in an addendum to this report.

APPENDIX C

REACTOR ACCIDENT CALCULATIONS

The analog model used in the study of the various accidents consisted of the reactor kinetic equations, heat-transfer equations, and coolant-flow equations. These equations are summarized below.

1. Reactor Kinetics Equation

The dynamic equations used to simulate the reactor kinetics were

$$\dot{n} = \frac{\rho - \beta}{\ell^*} n + \sum_{i=1}^3 \lambda_i C_i \quad \dots (C-1)$$

$$\dot{C}_i = \frac{\beta_i}{\ell^*} n - \lambda_i C_i, \quad \dots (C-2)$$

where

n = neutron power

C = delayed neutron precursor concentration

ℓ^* = prompt neutron lifetime

λ = decay constant

β = fraction of delayed neutron precursor

ρ = reactivity

The equation for reactivity was

$$\rho = \rho_o + \alpha_f \Delta T_f + \alpha_{gu} \Delta T_{gu} + \alpha_{gl} \Delta T_{gl}, \quad \dots (C-3)$$

where

ρ_o = initial reactivity

α = temperature coefficient

ΔT = temperature change

subscripts

f = fuel

gu = upper grid plate

gl = lower grid plate

Since α_f is not a constant with temperature, the reactivity change $\alpha_f \Delta T_f$ as a function of fuel temperature was incorporated into the analog model by using a function generator. This curve is shown in Figure C-1.

2. Heat Transfer Equations

The heat transfer equations used in the model simulated an average fuel element which was broken down into five equally spaced axial nodes. The equations were based upon a heat balance of each node. The model used solved for the outlet coolant temperature of each node. This results in an average fuel temperature which is too high, as it is based upon the temperature rise above the nodal outlet coolant temperature, instead of the nodal average coolant temperature. However, the use of one additional fuel temperature nodal equation based upon the reactor inlet coolant temperature corrects this error and brings the average fuel temperature down to within 5°F of the "exact" value as determined by a standard central difference model. The equations are:

$$\dot{T}_{fe} = \frac{k_e}{M_{fe} C_f} \frac{n}{n_o} - \frac{UA}{M_f C_f} (T_{fe} - T_{cin}) ; \quad \dots (C-4)$$

$$\dot{T}_{fi} = \frac{k_i}{M_{fi} C_f} \frac{n}{n_o} - \frac{UA}{M_f C_f} (T_{fi} - T_{ci}) ; \quad \dots (C-5)$$

$$\dot{T}_{ci} = \frac{UA}{M_c C_c} (T_{fi} - T_{ci}) - \frac{W}{M_{ci}} (T_{ci} - T_{ci-1}) ; \quad \dots (C-6)$$

where

T = temperature

k = power constant

n/n_o = normalized reactor power

U = heat transfer coefficient

A = heat transfer area

M = mass

C = specific heat

W = coolant flow rate

Figure C-1. Reactivity Change As a Function of Fuel Temperature

subscripts

e = entrance fuel node

f = fuel

c = coolant

i = number of node

in = inlet

3. Flow Equations

There are three potential incidents in which reactor coolant flow changes during the incident. These are loss of pump power, inadvertent closure of the throttle valve, and loss of coolant. Each of these incidents is discussed separately below.

a. Loss of Pump Power

The equation used to simulate the loss of pump power is based upon a force balance where the motive force equals the head force minus the friction force.

$$\dot{v} = K \frac{Ah\Delta T}{M} - \frac{A}{M} \Delta P v^2, \quad \dots (C-7)$$

where

v = coolant velocity

K = density constant

A = flow area

M = coolant mass

h = density head

T = temperature difference

P = rated flow pressure drop

b. Inadvertent Closure of Throttle Valve

The equation describing the inadvertent closure of the throttle valve is a simple linear reduction of flow as a function of time. It takes 10 sec to fully close the linear throttle valve. The equation used was

$$v = v_0 - 0.1t, \quad \dots (C-8)$$

where

v = coolant velocity (normalized)

v_o = rated coolant velocity (normalized)

t = time

c. Loss of Coolant

The time-dependent-flow equation for a loss-of-coolant accident depends largely upon where the system rupture occurs. This dictates whether or not flow through the core reverses. If reverse flow were to be considered, considerable changes in the problem analog mechanization would have been necessary. Hence, it was decided to make two pessimistic assumptions regarding the study of the assumed incident. The first was that coolant flow instantaneously goes from rated flow to zero flow; the second was that at the same time as flow goes to zero, coolant is removed from the core. The latter assumption does not affect the results appreciably, as less than 6% of the core heat capacity is tied up in the coolant.

APPENDIX D

PRIMARY VAULT LOSS OF COOLANT CALCULATIONS

The two assumed primary vault incidents were studied using the same general equations. Mass and heat balances were written assuming a system composed of NaK (liquid and vapor), nitrogen, steel floor liner, concrete floor, steel wall and ceiling liner, concrete walls and ceiling, vault equipment, floor cooling capacity, wall and ceiling cooling capacity, and vault atmosphere cooling capacity. The equations used were:

$$\dot{m} = \dot{m}_{kf} + \dot{m}_{vk} \quad \dots (D-1)$$

$$\dot{Q}_{c1} = (mc)_{c1} \dot{T}_{c1} + \dot{Q}_{a1} \quad \dots (D-2)$$

$$\dot{Q}_{c2} = (mc)_{c2} \dot{T}_{c2} + \dot{Q}_{a2} \quad \dots (D-3)$$

$$\dot{Q}_{s1} = (mc)_{s1} \dot{T}_{s1} + \dot{Q}_{c1} \quad \dots (D-4)$$

$$\dot{Q}_{s2} = (mc)_{s2} \dot{T}_{s2} + \dot{Q}_{c2} \quad \dots (D-5)$$

$$\dot{h}_{kf} = \dot{Q}_{vk} + \dot{Q}_{ka} + \dot{Q}_{s1} \quad \dots (D-6)$$

$$\dot{Q}_{ck} + \dot{Q}_{ka} + \dot{Q}_{vc} + \dot{h}_{vk} = \dot{Q}_{s2} + \dot{Q}_{a3} + \dot{h}_{n2} \quad \dots (D-7)$$

where

- $\dot{m}$ = mass flowrate of coolant spilled
- $\dot{m}_{vk}$ = mass rate of NaK vaporization
- $\dot{m}_{kf}$ = mass rate of change of NaK on vault floor
- $\dot{Q}_{c1}$ = rate of heat flow, steel floor liner to concrete floor
- $\dot{Q}_{a1}$ = rate of heat flow, floor-to-floor cooling system
- $\dot{Q}_{c2}$ = rate of heat flow, wall steel liner to wall concrete
- $\dot{Q}_{a2}$ = rate of heat flow, wall and ceiling to cooling system
- $\dot{Q}_{s1}$ = rate of heat flow, NaK on floor-to-floor steel liner
- $\dot{Q}_{s2}$ = rate of heat flow, vault atmosphere to wall and ceiling steel liner
- $\dot{Q}_{a3}$ = rate of heat flow, vault atmosphere to vault cooling system
- $\dot{Q}_{ck}$ = rate of heat flow, NaK heat of condensation

$\dot{Q}_{vk}$ = rate of heat flow, NaK heat of vaporization
 $\dot{Q}_{ka}$ = rate of heat flow, NaK on floor to vault atmosphere
 $\dot{Q}_{vc}$ = rate of heat flow, equipment to vault atmosphere
 $\dot{h}_{kf}$ = rate of change of enthalpy of spilled NaK
 $\dot{h}_{n2}$ = rate of change of enthalpy of N_2 in vault atmosphere
 mc = heat capacity
 $\dot{T}$ = rate of change of temperature

and subscripts

$c1$ = concrete floor
 $c2$ = wall and ceiling concrete
 $s1$ = steel floor liner
 $s2$ = steel wall and ceiling liner

The equation for the vault overpressure was limited to the pressure relief system setting of 0.75 psi overpressure. The equation used was:

$$\Delta P_{vt} = \frac{M_{n2} P_k}{M_{n2_o}} + \frac{M_{n2} R (T_{at} - T_{at_o} + 460)}{V} \quad \dots (D-8)$$

where

ΔP_{vt} = vault overpressure
 M_{n2} = mass of N_2 in vault
 M_{n2_o} = initial mass of N_2 in vault
 P_k = NaK vapor pressure at temperature T_{at}
 R = gas constant
 V = vault volume
 T_{at} = N_2 temperature
 T_{at_o} = initial N_2 temperature

Equations D-1 through D-8 were programmed on an electronic analog computer and solved.

APPENDIX E

CORE CONTAINMENT VESSEL PRESSURE CALCULATIONS

A. REACTOR HEAT LOSSES

The heat loss from the reactor during the transient accident was calculated by:

$$q = A \sigma \epsilon \left(T_R^4 - T_W^4 \right) \quad \dots (E-1)$$

where

q = heat loss (Btu/hr)

A = the reactor vessel area (2.95 ft^2)

σ = Stefan-Boltzman constant $\left[1.73 \times 10^{-9} \text{ (Btu)/(hr-ft}^2 \text{ - } ^\circ\text{R)}^4 \right]$

ϵ = the vessel emissivity (0.8)

T_R and T_W = the reactor surface and wall temperature respectively ($^\circ\text{R}$)

It was determined that the reactor at 2050°F will stay at this equilibrium temperature during its initial decay heat period. Further calculations show that the reactor will slowly cool from 2050°F as its decay heat drops.

B. HYDROGEN LOSS

The core hydrogen release was calculated in the following manner:

1. Burst During Transient

$$\% \text{H}_2 \text{ lost} = \frac{100 (P_B)}{QH_2} = 6.67\% \quad \dots (E-2)$$

where

P_B = the post fuel element burst decay energy ($6 \times 10^3 \text{ Btu}$)

QH_2 = the heat required to vaporize all the hydrogen in the core ($90,000 \text{ Btu}$)

2. Hydrogen Loss after Shutdown

The hydrogen loss after shutdown was found by a trial and error technique to determine the amount of hydrogen that may be lost from the core in such an amount that the hydrogen pressure of the core is equal to that in the atmosphere.

The following equations were employed:

a. Reactor temperature

$$T_R = 2050 - \frac{q_{H_2}}{C_P M} M_H \quad \dots (E-3)$$

where

T_R = the final reactor temperature (found to be 1800°F)

q_{H_2} = heat absorbed by the hydrogen release (36,000 Btu/lb)

M_H = lb of H_2 released (found to be 0.252 lb)

$C_P M$ = heat capacity of the core (Btu/°F)

b. Hydrogen pressure in fuel

Determined from the vapor pressure data given by Reference 18.

c. Hydrogen pressure in containment vessel

$$P = \frac{M_H}{2} R \frac{T_V}{V} \quad \dots (E-4)$$

R = the universal gas constant $[10.7(\text{psi-ft}^3)/(\text{lb mole} \cdot ^\circ\text{R})]$

V = the vessel volume (50 ft³)

T_V = the final vault temperature $[^\circ\text{R}, (\text{found to be } 1790^\circ\text{F})]$

d. Vault temperature

$$T_V \cong \frac{\left(T_o M C_V\right)_{\text{NaK}} + \left(T_o M C_V\right)_{\text{He}} + \left(\bar{T}_o M_H C_V\right)_H}{(M C_V)_{\text{NaK}} + M C_{\text{He}} + (M_H C_V)_H} \quad \dots (E-5)$$

where

$(T_o M C_V)_{\text{NaK}}$ = the initial temperature, mass, and heat capacity of the NaK vaporized, assumed to be at equilibrium with the entering stream at 1600°F

$(T_o M C_V)_{\text{He}}$ = the initial temperature (1600°F), heat capacity, and mass of the vessel's helium atmosphere

$(\bar{T}_o M_H C_V)_H$ = the average release temperature of the hydrogen (1925°F), heat capacity, and mass of the hydrogen

- e. The pressure decay is found by determining the temperature decay of the vault, as

$$T_V(t) = T_{V_0} - \frac{h}{(M C_V)_V} A_V \int_0^t (T_V - T_W)^{1.25} dt \quad \dots (E-6)$$

where

$$T_{V_0} = 1790^\circ\text{F}$$

h = the convective heat transfer coefficient of the gas¹⁹

A_V = the wall area (75 ft²)

$(M C_V)_V$ = the heat capacity of the vault atmosphere (2.4 Btu/°F)

T_W = the maximum wall temperature (440°F)

- f. The overpressure as a function of time was found by

$$P = \frac{R [T_V(t) + 460]}{V} - 14.7 \quad \dots (E-7)$$

where

n = the number of gas moles present in the atmosphere.

APPENDIX F

TEMPERATURE COEFFICIENTS

The breakdown of the total isothermal coefficient into the partial coefficients "fuel," "grid plate," "reflector," etc. was dictated by their importance as input parameters for reactor dynamics analyses. The calculation of such temperature-dependent coefficients is inherently a difficult and lengthy process and, insofar as the spectral shifts are concerned, not yet well understood. For this reason the purely theoretical results had to be evaluated in the light of such relevant data as had been taken on the SNAP 2 reactors, SER (primarily) and S2DR (to a lesser extent).

The results of these investigations are summarized in Figure F-1. The "upper grid" coefficient is due to the radial expansion of the upper grid plate (caused by a rise in coolant outlet temperature). This expansion spreads the rods slightly more apart, increasing the effective core radius and neutron leakage. Although not implied in its label, the "upper grid" coefficient is also taken to include the heating effects on the upper part of the core vessel and the transitional internal reflector region. Since no heating is assumed in the external Be reflector, the core vessel, although under constraint at its upper and lower ends, expands into the small 90-mil gap separating the vessel and external reflector. By the same token, the "upper and lower grid" coefficient accounts for the analogous effects resulting from lower grid plate expansion, in addition to those just mentioned.

The "reflector" coefficient is the effect produced by heating the radial reflector only. The strong negative coefficient is somewhat surprising but not unreasonable. Previous analyses of the SER have indicated that a large coefficient with a long time constant is necessary to reconcile a rather small prompt no-flow coefficient with the much larger isothermal coefficient.

The remaining partial coefficient, the "fuel" coefficient, effectively results from two causes: the shifts of the thermal spectrum, and the axial expansion of the fuel rods.* The axial expansions reactivity worth is felt to be quite predictable from theory; however, the worth associated with the spectrum shifts is

*The radial expansion of the rods is not considered to contribute appreciably, since the net result is only to displace a small amount of NaK from core. The reactivity worth of NaK has been shown to be very small.

Figure F-1. S8ER Partial Temperature Coefficients

another matter and, if one uses the SER as a yardstick, existing theory predicts a too positive coefficient. For this reason the spectrum coefficient of S8ER has been extrapolated from the SER data in the following manner: the presence of Sm_2O_3 in the S8ER fuel has a positive effect on the coefficient, and its magnitude, theoretically calculated, has been added to the experimentally determined SER spectrum coefficient, to yield the S8ER spectrum coefficient. This spectrum effect, when added to the axial expansion effect, gives the "fuel" curve shown in Figure F-1.

Several experiments were performed to measure the SER fuel coefficient. The effect of axial expansion was calculated yielding data for the SER spectrum coefficient. The smallest (in magnitude) values of the SER data were conservatively assumed to apply to the S8ER without prepoisoning.

APPENDIX G

EQUILIBRIUM ACTIVITY OF THE NaK COOLANT

The calculation of the Na^{24} and K^{42} activity induced in the NaK was based upon the following equations:

$$\text{Na}^{24} : A_j^{24} = K_{j24} \cdot \sum_{i=1}^{16} \Sigma_{ij} \cdot \int_{j^{\text{th}} \text{ region}} \phi_i dV , \quad \dots (G-1)$$

$$\text{K}^{42} : A_j^{42} = K_{j42} \cdot (0.18) \sum_{i=1}^{16} \Sigma_{ij} \int_j \phi_i dV , \quad \dots (G-2)$$

where A_j is the gamma activity (dps) produced in the j^{th} axial region of the reactor. K_j is the ratio of captures in Na or K, to the total number of captures, for the j^{th} region, Σ_{ij} is the total macroscopic absorption cross-section in the j^{th} region, for neutron energy group i (AIM-6 sixteen group cross-sections) and $\int_j \phi_i dV$ is the flux in the i^{th} group, integrated over the volume of the j^{th} region.

As noted, a factor of 0.18 is included, since 18% of K^{42} disintegrations emit the 1.53-Mev gamma.

The total activity is the sum over all regions.

Assumptions: 1) The activities are saturated.

2) Decay of Na^{24} and K^{42} during the time spent outside of the flux is neglected.

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