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SNAP 8 DEVELOPMENTAL REACTOR (S8DR)

SAFETY ANALYSIS REPORT

ADDENDUM 1

ATOMICS INTERNATIONAL

A DIVISION OF NORTH AMERICAN ROCKWELL CORPORATION

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**SNAP 8 DEVELOPMENTAL REACTOR (S8DR)
SAFETY ANALYSIS REPORT
ADDENDUM 1**

By

JOHN H. ROECKER

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A DIVISION OF NORTH AMERICAN ROCKWELL CORPORATION

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FOREWORD

This addendum to the SNAP 8 Developmental Reactor (S8DR) Safety Analysis Report (SAR), NAA-SR-Memo-11373, Sept. 6, 1967, has been prepared to answer specific questions or requests for additional information resulting from AEC review of the report. Each question or request is answered as an individual item.

The main section headings in this addendum are identical to those in the original SAR, and the questions are answered in the sections to which they apply. The subheadings in the addendum, however, give the subject of the questions answered; they do not correspond in specific order or wording to those in the SAR. Sections VII through X of the addendum contain no further information, as none was requested.

The conclusions of the original SAR are in no way altered as a result of this further review. The S8DR test can be conducted in Building 059 without providing an undue hazard to neighboring facilities, employees, or the general public.

I. INTRODUCTION AND SUMMARY

A. CONTAINMENT

Question: "The last paragraph on page 15 dealing with "levels of containment" should be clarified as to the relationship between the primary and secondary containment. "

Answer: Fission products formed in the operation of the reactor are contained, in order, by the fuel alloy, the fuel-element cladding, and the reactor vessel and primary coolant system. Further levels of containment are provided by the primary containment (vacuum system) and secondary containment (test vault). These final two levels of containment are not consecutive under all conditions.

If a breach of the reactor vessel or reactor inlet/outlet piping were to occur, fission products would be contained within the primary containment.

If a breach of the primary NaK system were to occur outside the primary containment, the fission products would all be contained within the secondary containment.

II. SITE

A. EMERGENCY PROCEDURES FOR ON-SITE PERSONNEL

Question: "Please provide additional information describing how the on-site personnel will be affected in the remote event of a loss-of-coolant accident. The answer should include an outline of the emergency procedures to be followed, a statement describing the degree of pre-operational review that the detailed emergency procedures will receive, and the number of on-site personnel which could be affected within a 0.75-mile radial distance from Building 059. Will the control room be evacuated? If not, what information will the operators have as to dose levels in the control room following this postulated accident?"

Answer: In an emergency arising within the S8DR facility the shift supervisor on duty has the responsibility for assessing the extent of the emergency and starting corrective action. If he judges the situation to be controllable, action is taken by the operating crew to counter the emergency, and the Santa Susana control center is so informed. At any time the shift supervisor may reassess this emergency situation and order system shutdown and facility evacuation. Upon evacuation, assessment of an incident as a possible major emergency will be made by the leader of the master emergency team which responds. The numbers of site personnel who could be affected in a major emergency are listed in Table 1 as a function of a direction and distance from Building 059.

Evacuation of the facility can be started by action other than that of the shift supervisor. Evacuation automatically activates the master emergency team. Radiation probes in the control room set off alarms at a reading of 20-mr/hr. When such an alarm sounds, procedures for facility evacuation start. Evacuation can also be ordered orally from the Santa Susana control center over the PA system or mechanically by manually actuating the alarm system.

TABLE 1
EMPLOYEE POPULATION DISTRIBUTION AT BUILDING 059 SITE*

Distance from Building 059 (mi.)	Compass Direction																Total Employees
	N	NNE	NE	ENE	E	ESE	SE	SSE	S	SSW	SW	WSW	W	WNW	NW	NNW	
0 to 1/4	0	0	9	28	41	1	86	86	13	9	0	0	0	0	0	0	273
1/4 to 1/2	0	0	8	14	3	5	8	28	2	13	13	0	0	0	0	0	94
1/2 to 3/4	0	0	0	25	2	0	4	0	0	0	0	0	0	0	0	0	31
Total within 3/4 mi.	0	0	17	67	46	6	98	114	15	22	13	0	0	0	0	0	398

* Data obtained from Atomics International and Rocketdyne Industrial Engineering Departments, November 1967.

In the specific case of a loss-of-coolant accident the shift supervisor should exercise his judgment in the evaluation of the situation; however, as soon as a 20-mr/hr or greater field is registered in the control room, evacuation of the facility is mandatory.

With the ratio of the gamma to the thyroid dose as presented in the SAR, a 20-mr/hr gamma doserate would imply a thyroid doserate of 2.1-rem/hr. This assumes that the control room detectors are reading only the gamma from the released cloud and not direct gamma from the high bay, and does not include the advantage of the isolation from the cloud offered by the control room itself.

After evacuation from the facility, remote monitoring of reactor and facility systems parameters is carried out from the data logger readouts in Building 024 (0.2 mi away). A gamma probe in the control room also is remotely indicated in the Santa Susana control center (0.6 mi away).

Atomics International's Standard Operating Procedures establish a Master Emergency Plan. Coordination of this overall plan is the responsibility of plant protective services. Each facility having areas of possible criticality and/or high radiation must establish a facility emergency plan which is integrated into the Master Emergency Plan. The facility emergency procedures for the S8DR test facility comply with these requirements.

Emergencies are classified under this system according to their extent and the personnel and facilities that are directly affected. Three classes have been defined as follows:

Class I

An emergency of a localized nature but large enough to require stopping of normal operations of the facility; control of the situation can be handled by personnel assigned to the facility with possible backup from the master emergency teams.

Class II

An emergency great enough to require evacuation of the area involved and which is an imminent threat to other facilities in the immediate area; control of the situation requires active assistance from master emergency teams. Evacuation is to a preselected area where a local command post for the emergency teams is established.

Class III

An emergency of such size that other facilities and areas are definitely affected and major emergency procedures are required.

For facilities under the cognizance of the Reactor Safeguards Review Panel, as is the S8DR facility, review of the facility emergency plan is required by the following:

- (1) Director of the department responsible for operating the reactor facility or activity.
- (2) Plant protective services to ensure their adequacy and compatibility with the Master Emergency Plan.
- (3) Reactor Safeguards Review Panel committee responsible for the reactor facility or activity.

After such review the plan is submitted to the approval authority for final review and approval.

Following is an outline of the facility emergency procedure.

- I. Purpose
- II. Definition
- III. Classification
- IV. Reporting
- V. Notification
- VI. Actions to be Followed on Designation of an Emergency
- VII. Facility Emergency Team
- VIII. Facility Hazards and Associated Containment, Suppression, or Termination Measures

B. POPULATION DISTRIBUTION

Question: "Source and date of the prediction of population distribution should be supplied. "

Answer: The present and predicted population in Tables 2.1 and 2.2, page 23 of the SAR, were prepared by the Radiation Safety Group at AI and originally reported in "Radiation and Nuclear Safety at AI."^{(1)*} The basic data were obtained from the site evaluation report⁽²⁾ and were updated with the latest population data by city district from References 3 through 7. These were then worked into the desired form in the tables.

* Superscript numbers in parentheses are for References at back of report.

III. REACTOR

A. CONTROL DRUMS

Question: "What is the design basis used to give assurance that one or more of the control drums will not become stuck in a position such that force from the scram springs is insufficient to rotate them? Please discuss potential causes of failure of a drum to rotate and state the design considerations taken to preclude them. For instance, the drum actuator must remain static to provide a fixed reference to assure that the spring force is directed to moving the drum segment out and not to simply rotating the actuator rotor.

Discuss the assurance you have that the scram springs are functional, i. e., that they are not broken or disconnected at the end of the wind up operation or that they do not break during the operations, thus rendering such a drum unscramable. Discuss ramifications."

Answer: Failure of an S8DR control-drum scram system to function once a scram signal has been given can occur only if its scram spring is unable to exert sufficient torque to overcome the frictional and inertial resistances of the control drum. The possible failures modes are: (1) failure of the scram clutch to release; (2) breakage of the scram spring; (3) excessive spring relaxation; (4) excessive increase of the frictional resistance; and (5) loss of the spring anchor at the actuator end. The probability that any of these failures will occur has been minimized by conservative design and verified by extensive developmental testing. Each of the possible failure modes is discussed.

1. Failure of Scram Clutch to Release

This failure could occur as a result of clutch face self-welding or clutch lockup due to residual magnetic holding or lack of sufficient clutch separation forces. These possibilities have been minimized by careful detail design and material selection as follows:

(1) Toothed engaging surfaces of the scram clutch face and the armature ensure positive retention of the stored energy of the scram spring before scram. Gravity provides the primary separating forces when the scram signal cuts the clutch magnet power. Positive clutch separation is aided by (1) the geometry of the teeth which resolve part of the torsional reaction of the scram spring into a clutch separating force, (2) a separating pull force from the bellows-coupling whose free position corresponds to the clutch released position, and (3) the porous chrome plating on the tooth surfaces which form a positive magnetic path gap between the teeth to minimize the residual holding force when the clutch is deenergized.

(2) Material self-welding tests and the scram system development tests described below have demonstrated that the porous chrome-plating forms adequate nonself-welding surfaces for the S8DR test environment of 850°F (clutch design temperature) and 10^{-5} torr.

(3) Development tests⁽⁸⁾ have shown that at room temperature a minimum of 2.08 volts dc are required to keep the clutch engaged against the above operational separating forces. This corresponds to a required holding voltage of approximately 5 volts dc at the clutch design temperature. A scram signal cuts the clutch voltage to zero. Tests reported in Reference 9 showed the residual holding force between the clutch face and the armature to be only 7.0 oz even after maximum excitation of the coil.

(4) The maximum change in the bellows-coupling free length due to relaxation after 900 cyclic deflections at 700°F was found to be only 0.002 in. The design value for clutch face and armature separation with the clutch deenergized is 0.020 in. Thus there is assurance that the bellows contribution to the total clutch separating force will be available throughout the test.

2. Breakage of the Scram Spring

The scram spring design value for initial scram torque at the spring design temperature of 775°F is 19.0 in. -lb.⁽¹⁰⁾ This value includes a 25% margin over the required torque to allow for possible spring relaxation. The torque is produced at a bending stress of 20,200 psi, which compares favorably with a tensile yield stress for Rene 41 of 90,000 psi and an ultimate stress of 132,000 at 800°F. Hence the probability of scram spring breakage is remote.

3. Excessive Spring Relaxation

Relaxation tests on representative scram springs⁽⁸⁾ have indicated that the maximum expected relaxation is less than 2.5% at 850°F in a nonirradiated environment. Inpile tests on Rene 41 compression springs, stressed to much greater percent of the torsional yield strength, has shown that force relaxation is not affected by irradiation (2×10^{19} nvt_f, 5×10^{10} r) at temperatures below 1050°F.⁽¹¹⁾ S8DR expected doses are 3×10^{19} nvt_f and 5×10^{10} r gamma. Thus the design torque margin should be available throughout the S8DR operation. The low stress levels plus the above component tests and the scram system development tests discussed below give further assurance that the springs will not relax unduly or break in service.

4. Excessive Increase of Frictional Resistance

The total frictional resistance of all the scram system rotating members was assumed to be 6.0 in. -lb when design requirements were established.⁽¹⁰⁾ This value included the bearing frictions and the friction forces generated between turns of the scram spring. The maximum frictional resistance of the scram system assembly has been measured to be less than 3.5 in. -lb at design operating conditions.⁽¹²⁾

The following factors will further minimize frictional resistance buildup:

- (1) The spring retainer is designed to minimize contact between the turns, and the spring is coated with dry film lubricant to reduce friction.

(2) Previous tests on carbon-graphite Al_2O_3 bearings have demonstrated that friction decreases significantly as the bearing temperature increases above room temperature⁽⁸⁾ and that friction remains nearly constant up to 10,000 hr of endurance.⁽⁹⁾ Thus the operational friction resistance should always remain substantially below the design limit, increasing the actual operating margin of the scram spring torque.

5. Loss of Spring Anchor

The anchor for the end of the scram spring which absorbs the torque reaction is provided by the actuator brake which mechanically locks the rotor. The inertia of the rotor, amplified by the 13.8-to-1 gear ratio, is sufficient to anchor the spring if the scram occurs while the actuator is stepping (brake unlocked). This was verified by conducting scrams on the test assembly with the actuator brake energized (brake off). There were no measurable differences between the scram times made with the brake locked and unlocked.

Extensive developmental tests were conducted on three test assemblies which included the complete control-drum drive and scram system⁽¹³⁾ (see Table 2). The tests were made with the scram clutch and spring at or above the design temperatures. No failure of the control drum to scram occurred during the test period.^(9, 13 through 16)

The original design analysis of the scram system included thorough evaluation of failure modes. Large design margins were incorporated where adequate performance data were not then available. Extensive developmental tests on components and complete systems have subsequently verified that the margins are more than adequate. The demonstration of over 1200 successful scram operations on each of the three test systems provides high confidence of successfully meeting the S8DR design requirement of 55 scrams. In the unlikely event that one drum scram system should become inoperative there would still be five others which would scram. Only simultaneous failure of two or more individual drum scram systems, an extremely improbable event, would prevent reactor shutdown. Shutdown would still occur with one drum stuck full-in.

TABLE 2
SCRAM TESTING ON THREE TEST ASSEMBLIES

Test Conditions	Unit No.		
	SN 001	SN 002	SN 003
Total test duration, hr	13,102	15,508	9,820
Scram cycles	1,389	1,200	1,376
Thermal cycles	200	200	198
Static dwell at temperature	-	2,000 (twice)	-
Clutch temperature, °F	850	850	950
Scram spring temperature, °F	870	870	970
Scram time to 31°, sec	0.30	0.27	0.32
Maximum scram time to 109°, sec	0.62	0.57	0.62

B. COMPONENT TESTING

Question: "Significant results of component testing which has been completed since the issuance of the SAR should be summarized. Control drum drive system components are of particular interest."

Answer: Significant amounts of control-drum drive component test data have been accumulated as a result of both the S8DR reflector design verification test and the continuing component-development program. These test data are summarized in this section.

1. Actuator

The component test program has shown that actuator performance could be improved with a modified coil design. The modification made consisted of (1) changing the chrome-plated copper to stainless-steel-clad copper wire, (2) changing from the SAP-2 encapsulate to AI-1, (3) increasing the coil wire

size, and (4) wiring phase coils in parallel rather than series. The expected performance improvement with the revised coil design was demonstrated, providing added confidence in the capability of an actuator to perform successfully and greater reliability in the S8DR control system.

For this demonstration three actuator test beds were subjected to 200 thermal cycles at 1025, 1125 and 1225°F, respectively, at the rate of 3 cycles per day. The test beds differed from the S8DR actuators in that they did not have rotors and therefore were not operated with a load. This in no way affected the thermal-cycle testing. No coil failures occurred and only the coils tested to 1225°F showed any measurable coil resistance change. The insulation resistances of all coils were higher than those of the previous coil design and remained high throughout the test. The 1125°F test bed actuator was subsequently subjected to 63 additional thermal cycles and 2193 hr of exposure to 1100°F without failure. The design requirements for the actuator in the S8DR environment is 950°F and 55 thermal cycles; thus the actuator test bed performance clearly demonstrates that the actuator will perform satisfactorily in S8DR.

To obtain information on the amount of actuator performance margin a detailed examination of actuator coils operating at higher than S8DR conditions was made. The coils of the 1225°F test bed actuator showed an increase in the coil resistance of 7 to 14%. In addition the oxalloy conductor had elongated and distorted in cross-section, but the coil assemblies maintained their mechanical and electrical integrity; they could still operate the actuator. Coil samples of the same design were later thermal-cycled 405 times to 1300°F to determine failure modes. The coil resistance of these coils increased approximately 56% and the coil conductors were elongated almost 18% but, again, the mechanical and electrical integrities were maintained. These results clearly demonstrated that possibility of coil conductor fracture was eliminated by the design modification, thus eliminating actuator failure due to such fracture.

In addition to the testing of test bed actuators and separate actuator coils, 10 actuators of the modified design were assembled and acceptance-tested. Three of these are now undergoing the design-verification test consisting of rapid thermal cycles to 1125°F and operational endurance at that temperature. All performances have been well within design requirements during testing listed in Table 3.

TABLE 3
STATUS OF DESIGN-VERIFICATION TESTING

Unit	Total Thermal Cycles Completed to 1125°F	Operations and Dwells Completed at 1125°F (hr)
SN 001	60	540
SN 002	57	421
SN 003	42	158

This testing has shown that the S8DR actuators will operate successfully in the S8DR test.

2. Scram Mechanism

Extensive thermal and operational cycling have demonstrated that the scram mechanism will operate successfully under S8DR conditions (see Table 2).

During the design verification testing of the first S8DR reflector-half assembly the bushing on the scram mechanism snubber shaft became displaced twice. In both instances, however, the snubber continued to operate satisfactorily. It was concluded that the displacement of the bushings was due to the decrease of the shrink-fit between the bushing and the housing and the limited clearance between the snubber shaft and the bushing bore.

The design of the snubber for S8DR has been modified to guarantee retaining the bushing in the housing and to increase the clearance between the snubber

shaft and the bushing bore. Six snubber assemblies reworked to the modified design were acceptance-tested with the snubber heated to 950°F, approximately 100°F higher than the expected operational temperature. Each unit successfully completed the 10 snubbing operations without any malfunction.

Special tests to evaluate the effects of a possible snubber malfunction demonstrated that neither a stuck snubber nor snubber shaft latch failure would be detrimental to the control-drum integrity or operational safety. A total of 2400 scrams from the full-in position against a stuck snubber were made on test assemblies which included control-drum bearing brackets, flexible coupling, and simulated drum mass. The flexible coupling, which is the weakest link, was heated to 1070°F; no failures nor permanent deformation of any part resulted. Tests were also conducted to simulate scrams against a snubber with a latch failure. In all tests the maximum initial inward rebound was to within 54° of the full-in position; the initial bounce was followed by a short highly damped oscillation as the scram spring continued to force the drum to the full-out position. These tests showed that the drum always came to rest in the full-out position within a couple of seconds.

Because of the design modification, malfunctions of the snubber mechanism are not expected. Even if a malfunction were to occur it has been demonstrated that the control drums will still operate and effect reactor shutdown.

3. Position Sensor

Two sensors of the modified design (coil design similar to that of the actuator) have completed design-verification tests consisting of 200 thermal cycles to 800°F followed by a 5000-hr operational endurance at that temperature. Two other units have completed 6981 hr of operational endurance at 800°F; they are thermal-cycled weekly and have completed 33 cycles. The electrical performance of these modified-design units meets the S8DR operational requirements.

4. Control-Drum Bearings

Three types of tests were conducted to establish the basis for selecting an alternate bearing material combination to replace the Al_2O_3 P5N carbon-graphite combination. This combination had experienced degradation when tested at 1250°F, 100 above the S8DR design temperature. Three selection tests were designed to evaluate alternate materials for its compatibility, friction characteristics, and performance in bearings.

The material compatibility was evaluated by testing potential material combinations in static contact for 500, 1000, 2000, and 5000 hr at test temperature of 1250, 1350, and 1450°F. The combination included various carbide coatings in contact with LiF-impregnated carbon-graphite (Pure Carbon Co. P5N) and plain carbon-graphite (Pure Carbon Co. P5). Metallurgical examination of the samples after 500, 1000, and 2000 hr of testing (5000-hr testing is still in progress) has indicated:

(1) All variations of Al_2O_3 coating in contact with the P5N carbon-graphite experienced gross failure similar to that experienced on bearing tests.

(2) The reference Al_2O_3 coating (detonation-gun-applied) remained dense, adherent, and essentially unaffected when tested in contact with the P5 carbon-graphite.

(3) The Cr_3C_2 and WC coatings were apparently stable and remained adherent. There was evidence that some vaporization of the chrome binder had occurred; some chrome material had transferred by this mechanism to the surface of the carbon-graphite.

(4) The TiC appeared to weaken after the initial high-temperature exposure. Numerous voids were evident on the metallographic samples. Because no binder had been used in the application of this coating, no material transfer to the carbon-graphite occurred.

The sliding friction characteristics at 1250°F of the carbide coatings were higher and more erratic ($\mu = 0.62$ for TiC, 0.39 for WC, 0.85 for Cr_3C_2) compared to the Al_2O_3 vs P5N ($\mu = 0.25$ to 0.31). One sample each of Cr_3C_2 and WC either seized or exhibited friction coefficient greater than 1.0 during these 1000-hr tests. Some of the chrome binder material was transferred to the surface of carbon-graphite on both of these samples, possibly accounting for the high friction. On the other hand the friction characteristics of Al_2O_3 against P5 carbon-graphite matched closely with those of Al_2O_3 against P5N carbon-graphite; after 1203 hr at 1250°F the maximum friction coefficient is 0.31 with no evidence of erratic behavior.

The nominal torques at 1250°F of WC-, TiC-, and Cr_3C_2 -coated shafts, when tested with the reference-design S8DR bearing ball, were similar to those of the reference-design Al_2O_3 -coated shaft. However, the torques of both WC- and Cr_3C_2 -coated shafts increased significantly after several hundred hours of static dwell at 1250°F (2.6 to 7.0 in.-lb for WC and 2.0 to 3.6 in.-lb for Cr_3C_2). The TiC-coated shaft behaved similarly to the Al_2O_3 -coated (2.0 to 2.7 in.-lb compared to 2.0 to 2.5 for Al_2O_3).

The tests also confirmed the previous indication that the failure of the Al_2O_3 coating was attributable to the LiF impregnant in the P5N carbon-graphite. No evidence of impending failure could be detected on the Al_2O_3 when tested in contact with nonimpregnated P5 carbon-graphite. The friction characteristics of the Al_2O_3 against P5 proved to be similar to those against P5N.

The design of the control-drum bearing has been changed to incorporate the Al_2O_3 P5 carbon-graphite material combination. Testing on this material combination has shown that it will operate successfully in the S8DR environment. Also, design-verification tests of S8DR bearings have started. Three have completed 30 thermal cycles, two sets to 1150°F and one to 1250, and are now entering a thermal-soak phase at the same temperatures.

IV. REACTOR COOLANT SYSTEM

A. PRIMARY SYSTEM STRESS ANALYSIS

Question: "Please provide additional information on thermal and total stress levels and the limiting number of stress cycles for critical areas of the primary system. Include a consideration of heat-up and cool-down cycles as well as operational temperature transients, particularly those following a scram from maximum power."

Answer: As a basis for this discussion, it must be pointed out that the main components of the primary system are the reactor, primary surge tank, intermediate heat exchanger (IHX), primary pump, and interconnecting piping (see Figure 4.2 of the SAR).

1. Critical Areas

For this discussion, critical areas are defined as locations where the operating stress level approaches most closely the design-allowable.

The most critical portion of the reactor vessel when subjected to steady-state stresses is the transition section of the upper plenum where the calculated stress intensity is 3,050 psi and the design-allowable is 3,300. Under transient conditions the vessel wall at the reflector support lugs is most critical having a design-allowable fatigue life of 140 cycles compared to the maximum of 55 causing the stress.

The most critical area of the surge tank is the inlet nozzle connection to the tank wall when subjected to hot- or cold-shock transients. The most severe cyclic stress range (including a local stress concentration of 1.5) of 105,000 psi results from the combination of cold shocks due to high-power scrams and hot shocks due to rapid startups and loss-of-flow scrams. This gives a fatigue life of 120 cycles compared to an expected maximum of five cycles causing stress of this magnitude. It is significant that some improvements have been made in the design of this nozzle over the one in the similar SNAP 8 Experimental Reactor surge tank that performed satisfactorily.

The primary inlet and outlet nozzles are the most critical areas of the IHX. It is estimated that the fatigue lifetime for the primary outlet nozzle is 200 cycles as a result of stresses generated by the automatic startup. These transients are conservatively estimated to occur no more than 55 times. Again, the favorable S8ER experience with an identical IHX should be noted.

In the pump the most critical area is the header end-cap joint. Two analyses were performed to bracket the stress range in this joint. One indicated low elastic stresses and no fatigue problem, the other showed high bending stresses that are expected to shake down to elastic action and have adequate design margins.

The most critical pipeline in the system is the line between the IHX and the primary-coolant pump. This line has an allowable number of peak stress cycles of 230 compared to the estimated maximum number of cycles of 55.

In summary a thorough review of stresses, calculated over the last four to five years, has been performed by personnel other than those doing the original work reported in the SAR. This check showed that the primary loop does have adequate design margins. More refined analyses are in progress on the surge-tank inlet, IHX inlet, and the pump header to further substantiate earlier judgments.

2. Operating Conditions

The steady-state operating conditions for the primary loop are summarized in the SAR, Figure 4.2 for 600-kw operation and 4.3 for 1000-kw operation. At 600 kwt the maximum design temperature and pressure for the reactor vessel are 1350°F and 35 psia; at 1000 kwt, 1150°F and 40 psia.

The critical transients for the fuel element are shown in Figure 3.5 of the SAR. Additional data on transient operation are in NAA-SR-MEMO-12272, "SNAP 8 Briefing to AEC Washington, D. C., December 1, 1966" and NR-7568-02, "Model Specification for SNAP 8 Nuclear System," August 5, 1964. The resulting design operating transients are summarized in Table 4, herein.

TABLE 4
DESIGN OPERATING TRANSIENTS

<u>Transient Description</u>	Test Frequency (number)		
	Shakedown and Startup	Endurance	Total
Automatic startup (150°F/min. for 4 min. , maximum idealized transient)	40	15	55
Loss of heat-rejection in secondary loop	5	5	10
Scram on loss of primary pump	8	10	18
Scram for high power (or temperature)	2	3	5
Cold slug due to pump restart following scram	15	10	25

These transients have different effects on the various components in the primary loop. For example, at the thermocouple feedthrough bulkhead of the reactor vessel the automatic startup transient causes the highest stress. At the surge tank inlet the maximum stresses are the result of scrams for high power (or temperature), and in the interconnecting piping the maximum stresses result from the total change in temperature in bringing the loop up to operating temperature. Since the transients result in different rates of change in coolant temperature, depending on which component is being considered, the critical transient will be described in conjunction with the particular component.

3. Irradiation Effects

The primary-loop components which receive significant irradiation are the reactor vessel and the coolant piping next to the vessel. The vessel and primary-loop piping material is Type 316 SS. The effects of irradiation on the structural evaluation of the reactor vessel have been described in the SAR. The highest flux areas for the coolant piping are where (1) the outlet line joins the upper plenum, (2) the outlet line runs downward opposite the reflector,

and (3) the inlet line attaches to the lower plenum. The outlet line adjacent to the reflector receives the greatest dose. For 600-kwt operation, the maximum total dose in the outlet line is less than 3×10^{20} nvt in 12,000 hr. Even if all operation were at 1000 kwt the total dose would be less than 5×10^{20} nvt. At irradiation and testing temperatures above about one-half the melting temperature ($\sim 600^\circ\text{C}$ or 1100°F) the main effect of irradiation is loss of ductility; the yield and ultimate strength are not greatly changed. According to the AEC Authorizing Legislation, Fiscal Year 1967, Appendix 12, "Preliminary data indicate that ductility at 650°C is reduced from 60% elongation in the unirradiated condition to 16 to 30% elongation in the irradiated ($1 - 10 \times 10^{20}$ total nvt) condition." From the standpoint of the structural integrity of the piping, an elongation of 16% is an adequate amount of ductility. In the ASME Power Piping Code, the established allowable stress does not allow maximum fatigue strain range to exceed about 0.2%. This conclusion is supported by data from Shober and Muir which is discussed by Bush and Tobin in the ASTM 1963 Symposium on "Radiation Effects on Structural Materials." Here the conclusion is drawn that "If the strain levels were below 0.6% the irradiation steel (347) appeared to have an improved resistance to fatigue, and higher stresses were always required to cause failure in the irradiated stainless steel at a given number of cycles" A. B. Holt of the Reactor Standards Staff, Division of Licensing and Regulations, states in TID-17682, "Survey of 21 Reactor Vessels in Light of the Anticipated Effects of Neutron Irradiation on the Brittle-Rupture Properties of Materials of Construction," 1962, that ". . . . we have used a criterion of 10^{19} nvt as the amount of accumulated fast-neutron dosage which brings a reactor vessel made of carbon steel into a state of brittleness which seems to warrant a closer scrutiny from the point of view of safe operation. . . . Several of the reactor vessels are made of stainless steel which is much less affected by neutron irradiation. For this case we have used the 10^{21} nvt criterion." Based on the above statements, the loss of ductility in the irradiated S8DR coolant piping is not considered deleterious to the safe operation of the system.

4. Reactor Vessel

Upper Plenum and Thermocouple Bulkhead. As discussed in the SAR the upper plenum and thermocouple bulkhead were designed and evaluated in accordance with the criteria of Section III of the ASME Boiler and Pressure Vessel Code. The configuration of the upper plenum and bulkhead and their areas of critical stress are shown in Figure 1. Table 5 lists: the locations and sources of stresses due to internal pressure; the maximum total or peak stress which includes the effects of pressure, transient temperature distribution and stress concentrations; and the design-allowable stress or number of cycles. The highest stresses in the bulkhead assembly occur during the startup transient; hence these were the only ones calculated for this assembly. The thermal stresses are the highest in the center of the upper bulkhead, Location 3 in Figure 1 (which is not the primary NaK containment), and the junction between the spacer cylinder and the lower bulkhead, Location 2. The lower bulkhead is nearly isothermal. The calculated stress intensities and cyclic stresses are

Figure 1. S8DR Upper Plenum and Bulkhead Assembly Stress Areas

TABLE 5
UPPER PLENUM AND BULKHEAD ASSEMBLY STRESSES

Location	Source of Stress	Calculated Stress (psi)	Design Allowable*
Center of lower bulkhead	Internal pressure 40 psia, 1150°F (1000 kwt)	2,800	11,350 psi at 1150°F
	35 psia, 1350°F (600 kwt)	2,400	3,450 psi at 1350°F
Junction of lower bulkhead and cylinder	Peak stress intensity (+150°F/min. for 4 min.)	29,200	1100 cycles [†]
Center of upper bulkhead	Peak stress intensity (+150°F/min. for 4 min.)	29,600	1100 cycles [†]
Transition section of upper plenum	Internal pressure 40 psia, 1150°F (1000 kwt)	3,500	11,200 psi at 1150°F
	35 psia, 1350°F (600 kwt)	3,050	3,300 psi at 1350°F
Nozzle penetration	Piping reactions + internal pressure	6,045	30,000 psi at 1350°F

* Per Section III, ASME Boiler and Pressure Vessel Code

† Assumes startup cycles of 150°F/min. for 4 min. followed by shutdown cycles of similar rate and duration.

within the criteria of Section III of the Code, which demonstrates the adequacy of this design for S8DR operation. This conclusion was further substantiated by a series of tests in which approximately twice the expected gradients were imposed on the bulkhead a total of 55 times, the maximum expected number of startups, without any evidence of cracking.

A comparison between the thermal gradients resulting from a scram from high power and those from a rapid startup is shown in Table 6 and it can be seen that the startup transient is more severe.

TABLE 6
TRANSIENT TEMPERATURE DIFFERENCES IN UPPER
BULKHEAD ASSEMBLY

Location	Transient Temperature Difference (°F)	
	Automatic Startup (600 to 1200°F at 150°F/min.)	High-Power Scram (1350 to 1035°F in 40 sec)
Center of upper bulkhead to outer edge	101	9
Junction of lower bulkhead and cylinder	142	103

Lower Plenum. The design temperature and pressure for the lower plenum (Figure 2) are discussed in the SAR. Summarized in Table 7 are the stresses in the lower plenum due to internal pressure, and those in the vessel reflector support-lug and wall-thickness-transition area due to inlet temperature

Figure 2. S8DR Vessel Lower Plenum Stress Areas

TABLE 7
LOWER PLENUM ASSEMBLY STRESSES

Location	Source of Stress	Calculated Stress (psi)	Design Allowable*
Center of lower plenum	Internal pressure 35 psia at 1350°F (600 kwt)	1390	3,500 psi at 1350°F
	40 psia at 1150°F (1000 kwt)	1590	11,500 psi at 1150°F
Edge of lower plenum	Internal pressure 35 psia at 1350°F (600 kwt)	4005	30,000 psi at 1350°F
	40 psia at 1150°F (1000 kwt)	4650	38,000 psi at 1150°F
Inlet nozzle	Piping reactions + internal pressure	8000	30,000 psi at 1350°F 32,000 psi at 1150°F
Vessel wall at reflector supports	Thermal transient: Automatic startup (including stress concentrations)	56,000 peak	140 cycles at 1100°F †
	Loss of heat-rejection	34,000 peak	600 cycles §
Vessel	Thermal transients due to loss of heat-rejection	2000	> 10 ⁶ cycles §

* Per Section III, ASME Boiler and Pressure Vessel Code.

† Assumes heating cycle followed by cooldown cycle of similar rate and duration (1100°F fatigue life from Code Case 1331).

§ See Section A-8 (following) for discussion of fatigue life.

transients. Two transients were considered in calculating the thermal stresses. These were the automatic startup condition, 150°F/min. for 4 min., and the loss of heat-rejection in the secondary loop which causes the uncooled NaK in the primary loop to reenter the core as a hot slug. Also shown are the effects of piping loads on the inlet nozzle. The resulting stresses and cyclic life are within the ASME Code Section III design criteria applied to S8DR.

5. Surge Tank

The surge tank was originally designed and constructed in accordance with Section VIII of the ASME Code and was the same as the unit used in the S8ER test. The S8ER tank was at temperature over 10,000 hr and was subjected to thermal transients of at least the same order of magnitude as those expected for S8DR. The original design pressure and temperature were 100 psig at 1450°F. The primary membrane stress in both the cylindrical shell and the ellipsoidal head is 1700 psi at a net internal pressure of 100 psig. In S8DR the maximum system pressure is 35 psia for 600-kw operation and 40 for 1000. Because the surge tank is located outside the vacuum chamber, the maximum net operating pressure is about 26 psig, which results in a primary membrane stress of 440 psi.

The surge tank for S8DR was modified slightly from the S8ER design by removal of the doubler plate on the inlet nozzle to minimize the effect of thermal transients. A thermal shield was also added in the inlet line to smooth out the temperature gradients. With the doubler plate removed the maximum allowable design pressure in the surge tank at 1450°F is approximately 75 psig and at 1350, reactor design temperature, is 135 psig.

The critical thermal transients for the surge tank inlet are combinations of the scram on high power (or outlet temperature), the automatic startup, and the scram on low flow. The scram on high-power transient (cold shock) includes the effects of a pump setback interlock which reduces the coolant flow and attenuates the transient to the surge tank and IHX. The stress intensity at the inlet nozzle to the tank joint is 42,000 psi, calculated on an elastic basis. The thermal gradients due to the hot shocks are about 60% as severe as the cold shock due to the scram on high power and the resulting stress range from the combined hot and cold shocks is about 70,000 psi. Applying a stress concentration factor of 1.5 raises this stress range to 105,000 psi. The alternating stress level, which is half of this, would indicate a fatigue life of 120 cycles

(see Figure 4, Section A-7, following) compared to the estimate of 5 of these maximum stress-range cycles. It should be noted that for the same combination of transients at the S8ER surge tank, which was operated without the modifications discussed above, the stress calculated on an elastic basis would be over 500,000 psi.

The 70,000-psi stress range for S8DR indicates that a plastic hinge will form at the joint, and the cyclic strain will be greater than that indicated by the elastic analysis. Considering the favorable experience on S8ER and the limited number of maximum-stress operational transients (5 high-power scrams), it is judged that the design-allowable cyclic life exceeds the number of applied transients. To substantiate this, a plastic analysis is now being performed as permitted under Paragraph N-41706 of the Winter 1967 Addendum to Section III of the ASME Code.

6. Intermediate Heat Exchanger (IHX)

The IHX was designed and constructed to comply with the provisions of Section VIII of the ASME Code, and is identical to the IHX used for the S8ER test. The design pressure and temperature are 100 psig at 1450°F. At 100 psig the primary membrane stress in the casing shell is 1800 psi. In operation the maximum pressure in the IHX is from the secondary-loop coolant. In the secondary loop the maximum pressure occurs during 1000-kw operation and is equal to 55 psia. The net pressure in the IHX is about 41 psig and the resulting primary membrane stress in the shell is 740 psi.

The design specification for the IHX required 100 transients at a rate of 300°F/min. for 30 sec. The effects of such transients were analyzed in the region of the primary coolant outlet nozzle. The maximum calculated alternating stress intensity due to this transient thermal gradient at the nozzle penetration is 91,000 psi. This stress is at the root of the weld joining the nozzle to the head and includes a stress concentration factor of 5 to account for the specific configuration. At the end of the 30-sec transient the temperature

differences in the structure have stabilized. Therefore the effects of the S8DR design transient, the startup transient, can be approximated by dividing the above-calculated stress by two. The resultant alternating stress intensity, 45,500 psi, results in a design fatigue life of 200 cycles at 1200°F, according to the design fatigue strength curve for austenitic stainless steels from Code Case 1331 of the Interpretations of the ASME Boiler and Pressure Vessel Code.

As mentioned in the surge tank discussion, an interlock is incorporated in the primary loop which cuts back the primary-coolant flow when there is a scram on high power, reducing this transient considerably. Also, because of the mixing volume in the surge tank, this thermal transient is greatly attenuated at the IHX inlet.

Another potentially critical transient for the IHX is due to pump restart following a shutdown. NaK flowing from the surge tank could have a significantly different temperature from that in the IHX. To minimize this transient, operating restrictions have been applied which limit the pump restart rate and the maximum temperature difference between the primary surge tank and secondary-loop NaK to 150°F at the time of restart. These restrictions were invoked during S8ER operation and proved to limit transients to low values compared with startups and scrams.

The analysis of this component and, again, the favorable experience with S8ER show that it will satisfactorily meet the S8DR test requirements. As an added precaution, however, some additional analyses of the IHX inlet are being performed to further verify this conclusion.

7. Electromagnetic (EM) Pump

The original design pressure for the EM pump was 100 psig. However, at the maximum primary system pressure of 40 psia for the S8DR test the net pressure in the pump throat is 26 psig.

A cross-section of the throat is shown in Figure 3. At the center of the throat cell span the primary-membrane plus bending-stress intensity is 650 psi at 26 psig. The design-allowable for this type of loading at 1150°F is 11,500 psi and 3590 psi at 1350. The maximum-discontinuity stresses in the pump throat occur at the edge of the span where the stresses are a factor of two greater than at the center. The design-allowable for these stresses is in excess of 30,000 psi and the fatigue life is greater than 10^6 cycles (see Figure 4).

The most critical area of the pump appears to be the header end-cap joint. Two analyses were performed to bracket the stress range in this joint. One indicated low elastic stresses and no fatigue problem; the other, high bending stresses and possible formation of a plastic hinge. These bending stresses are expected to shake down to elastic action and have adequate design margins. Again, as a precautionary measure, a more refined analysis has been started to provide added backup for this conclusion.

Figure 3. Electromagnetic Pump Throat, Header, and Cell

Figure 4. Fatigue Strength for Austenitic Stainless Steel to 1500°F

8. Interconnecting Piping

The main coolant lines of the interconnecting piping were designed and evaluated in accordance with the ASME Power Piping Code. The allowable-stress range for piping expansion stresses is given by the formula:

$$S_A = f (1.25 S_c + 0.25 S_h)$$

where S_c is the basic material allowable-stress at the minimum temperature, S_h is the allowable-stress at the hot temperature, and f is the stress range reduction factor for cyclic operation. For the number of S8DR operational cycles, $f = 1.0$. Using allowable-stress values from Section VIII of the ASME Boiler and Pressure Vessel Code for temperatures above 1200°F, and with a maximum design temperature of 1450°F, the value of S_A is 23,600 psi. The hot and cold stress-allowables, S_h and S_c , were determined by the least value of 62.5% of the yield strength, 25% of ultimate strength, 1% creep in 10^5 hr, or

80% of the minimum stress for creep rupture in 10^5 hr. For long-term high-temperature stresses, evaluation of S_A by the use of Section VIII allowables is valid in order to preclude large strain concentrations (since the creep considerations govern). However, for short-duration transients, creep deformation is not significant and the yield strength is of primary concern. This rationale leads to a second value of S_A for transient conditions:

Evaluation of allowable stress for short-term transient ($T_{\max} = 1450^\circ\text{F}$).

$$S_c = 0.625 \times S_{yc} = 0.625 \times 34,000 = 21,300 \text{ psi}$$

$$S_h = 0.625 \times S_{yn} = 0.625 \times 17,000 = 10,700 \text{ psi}$$

$$S_A = 1.25 S_c + 0.25 S_h = 1.25 \times 21,300 + 0.25 \times 10,700 = 29,500 \text{ psi}$$

(Values are from ASTM DS5-S1.)

The piping stresses have been determined for long-term and transient operation (see Table 8). For the transient conditions, stress values were not noted if they were not significantly different from the values for long-term operation. Table 8 shows that for such operation all stresses fall below the S_A of 23,600 psi. For the transient conditions all stresses are below the S_A of 29,500 psi.

In the above computations, piping secondary stresses, e. g., axial bending stresses due to differential expansion, are taken into consideration. For fatigue evaluation the peak stresses in the circumferential direction must be analyzed. As noted by the M. W. Kellogg Company ("Design of Piping Systems," Sylie, 1956), large circumferential stress intensification factors are encountered in the region of piping elbows. The stress ranges noted in Table 9 are obtained with these values. The table also shows the fatigue evaluation for the system, with large margins shown for all lines.

The curves shown in Figure 4 were used for the fatigue evaluation. The solid curves were constructed by the universal slope method recommended by Manson and Halford (NASA-TMX-52270) with the number of cycles reduced to

TABLE 8
MAXIMUM SECONDARY STRESS RANGES FOR S8DR PRIMARY PIPING

Line No.	Line Description	Stress Ranges (psi)			
		Long-Term Operation ($S_A = 23,600$ psi)	Transient Conditions ($S_A = 29,500$ psi)		
		Ambient to Maximum Temperature	High-Power Scram	Low-Flow Scram	Loss of Heat-Rejection
101	Reactor outlet vacuum vessel wall	11,650	12,800	13,300	-
102	Vacuum vessel wall, surge tank	14,300	13,100	13,800	-
103	Surge tank, intermediate heat exchanger	7,900	-	-	-
104	Intermediate heat exchanger, pump	23,100	-	-	29,000
105	Pump, vacuum vessel wall	6,100	-	-	7,600
106	Vacuum vessel wall, reactor inlet	13,000	-	-	16,300

TABLE 9
FATIGUE EVALUATION FOR S8DR PRIMARY PIPING

Line No.	Peak Stress (Circumferential) (psi)	Alternating Stress* (psi)	Cycles		
			Allowable Number (N) †	Maximum Number Expected §	Design Factor
101	26,600	20,000	4×10^3	55	73
102	28,600	21,500	3×10^3	55	55
103	15,800	11,900	16×10^3	55	290
104	58,000	43,500	230	55	4
105	15,200	11,400	18×10^3	55	330
106	32,600	24,400	2×10^3	55	36

* Per Paragraph N 415.2, Section III, ASME Boiler and Pressure Vessel Code

† From Figure 4

§ Number of cycles over full temperature range

5% of predicted value. Data used are those on Type 316 SS tube and pipe from ASTM-DS5-S1, "Elevated Properties of Stainless Steel." Although the data indicate increasing ductility (reduction in area) at temperatures above 1200°F, no advantage was taken of this in the analysis. NASA-TMX-52270 also reports fatigue data of several investigators for Type 304 SS at temperatures from 1000 to 1500°F. The 1200 and 1500°F results are plotted in Figure 4 as dashed lines. Also shown in the figure is the 1200°F fatigue strength line from ASME Code Case 1331-1. The evaluation of S8DR piping stresses was made by comparing the alternating stress intensities with the lowest curve of the figure; for stresses below 25,000 psi this was the 1500°F curve which includes the ASME design safety factor. For those above 25,000 psi the 1200°F ASME curve was used.

Another potentially critical area for the piping is the point at which the coolant lines pass through the walls of the vacuum chamber. The vacuum vessel feedthrough consists of a length of pipe with an expanded section welded on at one end. This assembly fits around the coolant line. The large end of the feedthrough is welded to the vessel wall, the other to the coolant line. At operating temperature the peak stress at the joint between the feedthrough and the line is 20,000 psi (alternating stress intensity, 10,000 psi) which includes a stress concentration factor of 5 for the weld-joint configuration. From Figure 4, the resulting fatigue life is 3×10^4 cycles.

B. SEPARATION OF WATER AND NaK

Question: "Please state the design criteria used to insure isolation of water and NaK. "

Answer: In the design used to assure the separation of water and NaK in the S8DR primary and secondary containment, three barriers exist between these materials. These are the pipe or container holding the water, those holding the NaK, and a third separate barrier.

The only possible exception to this assurance is the water-baffle system located in the vertical riser section of the diffusion pumps. If one of these two baffles were to rupture, the water would fall into the diffusion pump where it would boil and be withdrawn through the roughing pump. Because this baffle system uses only 10 gal of water and a pump body has a 50-gal capacity, this incident could not spill water over into the vacuum ducts and on into the reactor vacuum vessel. Above the water baffle, and between it and the reactor, is an optically opaque liquid-nitrogen baffle which would further trap any water from this source. Still further protection would be given by the automatic closure of the 48-in. vacuum valves when a loss of vacuum is detected, as would be the case if either NaK or water vapor is released to the vacuum system. Therefore a rupture in either the NaK or water-baffle system could not result in more than the vapor of one liquid reaching the containing barrier of the other because of:

- (1) the limited inventories of the water and NaK systems,
- (2) their physical locations,
- (3) the distances to be traveled through the vacuum ducts, and
- (4) the optically opaque liquid-nitrogen baffle between them.

The only other source of water in the test vault area is the vacuum vessel and shield cooling system. In this case the design criteria for assuring

separation of water and NaK are met in the following manner. First, all cooling coils are located outside the vacuum vessel, thus utilizing the vessel wall as a third barrier for the part of the NaK system located inside the vacuum system. Second, a steel plate will be installed over the vacuum vessel pit separating all cooling coils from that portion of the NaK system located in the test vault. Third, a steel sleeve and plate separate the cooling coils and NaK lines in the pipe chase below the vault floor where NaK lines enter the vacuum vessel.

V. CONTAINMENT SYSTEM

A. LOSS-OF-COOLANT ACCIDENT LEAKAGE

Question: "If the loss-of-coolant accident is due to a primary-coolant pipe rupture in the secondary containment vault, what are potential leakage paths and possible leakage rates to the atmosphere? Discuss the assumptions and analytical models used to determine the peak pressure in the containment."

Answer: The potential leakage paths out of the secondary containment are at the shield plugs, vault doors, valves, wires and conduit leading out of the vault, and through liner porosity and seams. It is not possible, without extensive and selective leaktesting, to determine the individual leakrates for the above-mentioned paths. Because of the underground location of the vault and the location of the penetrations, that leakage which might exist would either diffuse through several feet of concrete to an underground release point or leak to areas serviced by the facility stack. Most of the leakage would probably be through the shield plugs, wires, and conduit to the high bay, and then out the stack.

The secondary-containment vault will be tested, after all equipment is installed, before nuclear operation. The objectives of the vault test are to:

- (1) Verify that the vault leakrate is less than 1.5 vol %/hr at 4-psig vault overpressure.
- (2) Determine the relationship between vault leakrate and pressure at about 4, 3, 2, and 1 psig.
- (3) Test the vault negative- and positive-pressure protection system. The negative-pressure protection system will be tested at -0.5 psig. The overpressure protection system will be set and tested to assure that the 6-in. relief line opens at 4 psig and the 30-in. at 4-1/4.

Because of this testing the integrity of the vault is assured and a leakrate of less than 1.5 vol %/hr will be confirmed. A highly conservative analysis of the loss-of-coolant accident shows that the peak pressure will be less than that tested and that a pressure greater than 1 psig will exist for less than 12 min.

The second part of the question is answered in Section XI-A-2 of this addendum, where loss-of-coolant accident is described in detail.

B. SECONDARY CONTAINMENT

Question: "If the design criteria for the secondary containment is 4 psi, why is the pressure relief valve set at 5 psi? Discuss the possible effects on the secondary containment if either of these pressure levels occurs in the secondary containment vault."

Answer: The secondary-containment vault will be leakrate tested at the design criterion to assure a leakage rate of less than 1.5 vol %/hr. This overpressure capability assures complete containment of the worst accident that could be accompanied by a loss of fuel-clad integrity (see Figure 11.2 and Par. 11.3.5.3 of the SAR.

During non-nuclear operation or when insufficient decay heat is present to rupture the clad in the event of a total coolant loss, the vault atmosphere may have an oxygen concentration of greater than 0.1 %. A rupture of the NaK system at this time, if followed by a NaK fire, could result in vault pressures greater than 4 psi if completely contained. Therefore, to protect the structure, an overpressure relief system activates and vents the vault through the stack filter plenum to be discharged out the stack.

The operating mode and setpoints for this pressure-relief system have been recently revised to the following:

- (1) At a vault pressure of $4 \pm 1/4$ psi a 6-in. line will open to the stack filter plenum.

(2) At a vault pressure of $4\text{-}1/4 \pm 1/4$ psi a 30-in. line will open to the stack filter plenum.

(3) The valves will close these lines when the pressure is relieved to the setpoint values.

(4) The system will be active at all times to assure that under any circumstances leading to a vault pressure greater than 4 psi, the relief will be through the facility stack. This protects against possible structural failure and a potential ground release of activation, and/or fission products that may exist in the coolant.

As stated in Section V-A this system will be performance-tested before nuclear operation to verify the operating setpoints. The pressurization of the vault to $4\text{-}1/2$ psi is allowable because the actual design overpressure is $4\text{-}1/2$ psi. Therefore no adverse effects on the vault are expected by these pressure levels. To assure the vault integrity after the overpressure relief test, the vault leakrate tests will be performed after the performance of this system is demonstrated.

C. POSSIBLE LEAKAGE FROM PRIMARY AND SECONDARY CONTAINMENT

Question: "Provide a schematic drawing of all penetrations through the vacuum vessel or secondary vault which could be potential leakage paths in an accident. Discuss provisions taken either in design, hardware, or containment to prevent leakage to the atmosphere from these paths either directly or indirectly. Discuss valve closure procedures in case of an accident for the critical systems."

Answer: Schematic drawings of the penetrations through the primary vault (vacuum vessel) and secondary vault are shown by Figures 5 and 6. These penetrations are listed in Tables 5.1 and 5.2 of the SAR. All are shown without regard to their potential as leakage paths since every effort has been taken to minimize leakage throughout the critical systems.

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Figure 5. Primary-Containment (Vacuum System) Penetrations

Figure 6. Overall S8DR Vault Layout Showing Penetrations

For the primary-containment system (the vacuum vessel) all penetrations with the exception of the two roughing pump discharge lines and the eight pressure sensors are welded (Table 5.1). The nonwelded penetrations are O-ring sealed. The primary-containment system will be given an ASME Code test at 42 psig for 24 hr. It will be leakrate tested at 10 psig to assure leakage of less than 0.35 vol %/day. More important to the demonstration of system integrity throughout the duration of the test, the vacuum system will be operated at a pressure of about 10^{-5} torr, which implies a maximum leakrate of <0.01 vol %/day.

The secondary-containment system penetrations (Table 5.2) are all seal-welded to the vault liner. Lines containing instrumentation leads are potted with epoxy to prevent leakage. The two access doors are double O-ring sealed and bolted shut; the access plugs are single O-ring sealed and bolted. The purge valves are gate-type with inflatable seals to prevent leakage.

Valve-closing procedures in an accident in the primary-containment system consist of the automatic closing of the 48-in. vacuum valves (Figures 5.1, SAR, and 5, this addendum) when the pressure in the system reaches 200 microns. These valves are designed to hold to an exceedingly low leakage rate ($<1 \times 10^{-8}$ Std cc/sec). If these valves fail to close during an accident, the effluents will go to one of the vacuum system holdup tanks. The primary-containment system is a closed system at all times; no valve action is required to provide system integrity. When one of the vacuum system holdup tanks is vented to the low-volume exhaust system it is isolated from the rest of the vacuum system, and the alternate holdup tank is in service (see Par. 5.1.2.3 of the SAR). Because of this arrangement for the primary-containment system, complete isolation is achieved at all times regardless of manual or automatic valve action.

In the secondary containment the only valves which open to systems which are not the closed type are the 6-in. and 30-in. purge valves (Figure 5.5, SAR)

and the valving on the vault exhauster line. The 6- and 30-in. purge valves are closed during operation and sealed by inflatable seals (see answers in V-A and V-B). These valves open if the vault is pressurized to 4 psi and vent directly to the stack filter plenum. The isolation valve on the vault exhauster line, which also vents to the stack filter plenum, closes automatically if the vault pressure exceeds 0.5 psig or if the radiation level in the exhaust line exceeds a setpoint level. This level will be set when operating background is determined, but will probably be of the order of 2 times background. The monitor on this line is so located that it receives negligible direct radiation from the reactor or test vault. If a failure in this isolation system occurs in which the isolating valve does not close, the pressure-control valve may be closed manually from the control room. This valve is in series with the isolation valve and would independently isolate the vault.

VI. INSTRUMENTATION, CONTROL, AND ELECTRICAL SYSTEMS

A. IEEE PROPOSED STANDARDS FOR REACTOR PROTECTION SYSTEMS

Question: "If any of the criteria contained in the IEEE Proposed Standards for Reactor Protection Systems are not met, show why they are not necessary for this reactor facility. In particular, each of the items in Section 4 of the IEEE Proposed Standards should be addressed. Based on descriptive information contained in the safety analysis report and from on-site discussions, the reactor protection system does not appear to meet the single-failure criterion in that it contains two final solid-state logic amplifiers which both must agree that the reactor should be scrammed. It appears that a failure of either of these two amplifiers could prevent automatic reactor scram."

Answer: It can be best demonstrated that the S8DR instrumentation system meets the standards proposed by the IEEE for nuclear power plant protection systems by showing that all of the specific requirements of Section 4 of the referenced document are met. To facilitate discussion here, a short description of each of the requirements has been extracted from the document.

"4.01 General Functional Requirement. The nuclear power plant protection system shall, with precision and reliability, automatically initiate appropriate protective action whenever a plant condition monitored by the system reaches a trip point level. This requirement applies for the full range of conditions and performance requirements that could be met during plant life."

The instrumentation furnished by the manufacturer was built to AI specification NES8DS-24-001, Nuclear Instrumentation. Part of the specification required that tests be performed and inspections made per AI-approved procedures to demonstrate that the specification performance requirements were met. The AI inspection department documented the results of these efforts and indicated that the instrumentation did meet the specification. The specification included:

(1) performance characteristics such as response time, accuracy, and ranges; (2) environmental conditions under which the systems would be required to operate; (3) bypass and trip features required; (4) number of channels and type detectors; and (5) required redundancy.

It is the judgment of AI that demonstration of meeting the purchase specification together with previous experience of the manufacturer in supplying satisfactory instrumentation of a similar type to the nuclear industry reasonably assures that this functional requirement will be met.

"4.02 Single Failure Criterion. Any single failure within the protection system shall not prevent proper protection system action when required.

NOTE: 'Single failure' includes such events as the shorting or open circuiting of inter-connecting signal or power causes. It also includes single credible malfunctions or events which cause a number of consequential component, module, or channel failures. For example, the overheating of an amplifier module is a 'single failure' even though several transistor failures result. Mechanical damage to a mode switch would be a 'single failure' although several channels might become involved. "

The single-failure criterion has been met through redundancy and logic circuit design, the basis of which is to assure that a backup circuit is available to respond to a scram signal or that the system is "fail-safe."

The two logic amplifiers and two clutch-power supplies were provided in parallel to assure reliability during long-term runs of the order of 10,000 hr without interruption. With the duplicate configuration of amplifier and clutch-power supplies, a failure in any one of these units would not cause a shutdown. This combination provides for removal and repair in case of failure and additionally for the manual testing of each half of the shutdown system without interference with the operations. The failure problem between manual test

periods was recognized during design preparation. Consequently, a pulse testing feature was built into the system to continuously monitor the operability of each circuit during operation. A fault in either circuit is audibly alarmed and visually indicated. AI believes that the protection given by this feature of the logic circuit amplifiers meets the intent of the single-failure criterion established by the IEEE Proposed Standards.

Failure of a single switch would not be a violation of the criterion in that a scram signal would still scram the other five control drums, which control sufficient reactivity to shut down the reactor.

Further discussion of this criterion is in VI-C, -D, and -E.

"4.03 Quality of Components and Modules. Components and modules shall be of a quality which is consistent with minimum maintenance requirements and low failure rates. Quality levels shall be achieved through the specification of requirements known to promote high quality, such as requirements for design, for the derating of components, for manufacturing, quality control, inspection calibration and test. "

Past performance of the manufacturer in providing quality products and the inspection department verification of adherence to the specifications are adequate to demonstrate quality. All circuits were conservatively designed with components operating at derated characteristics to provide longer life. For instance, the transistors were designed to operate to have one-half minimum gain specified, a maximum leakage current twice that specified at 65°C, and a maximum power dissipation less than one-tenth that allowed by the manufacturer.

"4.04 Equipment Qualification. Type test data or reasonable engineering extrapolation based on test data shall be available to verify that equipment, which must operate to provide protection system action, will meet, on a continuing basis, the performance requirements determined to be necessary for achieving the system requirements. "

The nuclear protection equipment was subjected to the performance tests specified in the Nuclear Instrumentation System Specification (NES8DS-24-001). The verification of the performance of the instrumentation to these tests was observed and verified by the AI Product Inspection Department. Similar equipment provided by the same manufacturer has proven to operate reliably in similar operating environments in both the SNAP 2 Developmental System and SNAP 10 Programs for similar operating periods. In prototype tests a trip circuit was turned on and off 5 times per second while being maintained at 65°C for over a half million operations; no evidence of deterioration appeared except to an associated relay which had no effect upon the trip circuit. Operation remained normal at 150°C.

"4.05 Channel Integrity. All protection system channels shall be designed to maintain necessary functional capability under extremes of conditions (as applicable) relating to environment, energy supply, malfunctions and accidents."

The specification required functional capability under extreme of environmental conditions, namely that the design will allow the system to operate satisfactorily in a temperature range from 50 to 110°F and in humidities from 0 to 70%. Line voltage and frequency transients are avoided by taking power from the uninterruptible power supply (UPS) through the flywheel motor-generator set.

"4.06 Channel Independence. Channels which provide signals for the same plant protective function shall be independent and physically separated to accomplish decoupling of the effects of unsafe environmental factors, electrical transients, and physical accident consequences documented in the design basis and to reduce the likelihood of interactions between channels during maintenance operations or in the event of channel malfunction."

Each of the nuclear detectors is placed in its individual thimble around the core and supplied with its own polarizing power supply. Cable separation and

vessel penetration are discussed in IV-C. The thermocouple and signal conditioning amplifiers are separate and unique to each of the three signals required for the reactor outlet temperature protective circuits. These thermocouples are individually sheathed in stainless steel penetrating the vacuum vessel in a common header to separate conduits to the amplifiers. Separate backup thermocouples reading the reactor outlet temperature are continuously displayed on a recorder and will generate alarm signals at off-normal conditions.

"4.07 Control and Protection System Interaction. Where a plant condition which requires protective action can be brought on by a failure or malfunction of the control system and the same failure or malfunction prevents proper action of a protection system channel or channels designed to protect against the resultant unsafe condition, the remaining portions of the protection system shall independently meet the requirements of Sections 4.01 and 4.02. "

The automatic control system used in operating the plant derives its signals from the Resistance Temperature Detectors used exclusively for this purpose. There is no common utilization of components or systems for either protective and information systems with the control system.

The manual control system is independent in action from the automatic controller. The manual controller and its operation are described fully later in Section VI-I.

"4.08 Derivation of System Inputs. To the extent feasible and practical, protection system inputs shall be derived from signals which are direct measures of the desired variables. "

All protective system inputs are derived from signals which are direct measures of the desired variables.

- "4.09 Capability for Sensor Checks. Means shall be provided for checking with a high degree of confidence, the operational availability of each system input sensor during reactor operation. This may be accomplished in various ways, for example:
- by perturbing the monitored variable; or
 - within the constraints of Section 4.11,
 - by introducing and varying, as appropriate, a substitute input to the sensor of the same nature as the measured variable;
 - or by cross checking between channels which bear a known relationship to each other and which have readouts available."

During extended power operation substitute inputs of the same nature can be introduced as necessary for system checkout, such as millivolt signals for temperature or test signals for each nuclear channel and also by cross-checking between redundant readouts. A regularly scheduled checkout and preventative maintenance program is established for the facility.

- "4.10 Capability for Test and Calibration. Capability shall be provided for testing and calibrating channels, their associated trips and the devices used to derive the final system output signal from the various channel signals. For those parts of the system where the required interval between testing will be less than the normal time interval between plant shutdowns there shall be capability for testing during power operation."

Capability for test and calibration has been built into the nuclear channels. Test signals can be inserted into all temperature channels.

- "4.11 Channel Bypass or Removal from Operation. The system shall be designed to permit any one channel to be maintained, and when required, tested or calibrated during power operation without system trip. During such operation the active parts of the system shall of themselves continue to meet the single failure criterion."

A bypass system is available with administrative controls to allow check-out, maintenance, and calibration during operation.

"4.12 Operating Bypasses. Where operating requirements necessitate automatic or manual bypass of a protective function, the design shall be such that the bypass will be removed automatically whenever permissive conditions are not met. Devices used to achieve automatic removal of the bypass of a protective function are part of the protection system and must be designed in accordance with this standard.

"4.13 Indication of Bypasses. If the trip function of some part of the system has been bypassed or deliberately rendered inoperative for any purpose, this fact shall be continuously indicated in the control room."

All automatic bypasses are indicated, annunciated, and automatically reset when permissive conditions are not met. Manual bypasses are administratively controlled and indicated.

"4.14 Access to Means for Bypassing. The design shall permit the administrative control of the means for manually bypassing channels or protective functions."

Access to bypasses are administratively controlled through a key-lock control panel.

"4.15 Multiple Trip Settings. Where it is necessary to change to a more restrictive trip setting to provide adequate protection for a particular mode of operation or set of operating conditions, the design shall provide means of assuring that the more restrictive trip setting is used. The devices used to prevent improper use of less restrictive trip settings shall be considered a part of the protection system and shall be designed in accordance with the other provisions of this standard regarding performance and reliability."

With a single exception (see below) all restrictive trip settings appropriate for a mode of operation require modification of the nuclear instrumentation logic cards. Such change cannot be made without approval and knowledge of the operations supervisor. The single exception is the choice of range for the power scram setpoint. In this case the range is continuously indicated.

"4.16 Completion of Protection Action Once it is Initiated.
The protection system shall be so designed that, once initiated, a protection system action shall go to completion. Return to operation shall require subsequent deliberate operator action. "

Once started, all protective actions are locked on and require a manual action to clear the protection circuit.

"4.17 Manual Actuation. Means shall be provided for manual initiation of protection system action. Failures in an automatic protection circuit shall not prevent the manual actuation of protective functions. Manual actuation shall require the operation of a minimum of equipment. "

The manual scram button has two actions through the logic circuit to interrupt magnet power at the magnet switch and remove ac input from the scram clutch magnet supply. As discussed in VI-E later, three independent means exist for effecting a manual scram of the S8DR system.

"4.18 Access to Trip Settings, Calibration and Test Points.
The design shall permit the administration control of access to all trip settings, module calibration adjustments and test points. "

The S8DR design permits administrative control access to all trip settings, calibration adjustments, and test points.

"4.19 Identification of Trips. Trips shall be indicated and identified down to the channel level. "

Trips are indicated and identified down to the channel level by an annunciator panel and indicator lights.

"4.20 Information Readout. The protection system shall be designed to provide the operator with accurate, complete, and timely information pertinent to its own status and to plant safety. The design shall minimize the development of conditions which would cause meters, annunciators, recorders, alarms, etc. to give anomalous indications confusing to the operator. "

The operator is given unambiguous continuous recording of reactor power, reactor outlet temperature, flow, system pressures, and drum positions.

"4.21 System Repair. The system shall be designed to facilitate the recognition, location, replacement, repair or adjustment of malfunctioning components or modules. "

The protection system was designed to monitor and annunciate for malfunctions of system integrity from detector to readout, and to utilize a continuous monitoring system for functions vital for continued operation. The design provides for a degree of interchangeability to facilitate repair. Non-nuclear indicators (temperature, pressure, and flow) are provided with up-scale or down-scale drive if a sensor is open-circuited.

B. AUTOMATIC STARTUP

Question: "Provide additional information on how the reactor is automatically brought through criticality and heat-up to power. Include a discussion of the basic sensors relied upon and indicate whether any sensors are shared with the reactor protection system. The discussion should clearly show that single faults in the control system cannot cause loss of redundancy in ability to scram the reactor safely. "

Answer: The automatic controller and the reactor protection system do not share any common components. Failure of the automatic controller or its sensors will not in any way compromise the operation of the reactor protection system.

The automatic controller controls the motion of all six drums. Three drums, Nos. 1, 3, and 5, are designated as control drums and the other three, 2, 4, and 6, are startup drums. During a reactor startup the latter are driven in to a preselected angle and stopped before any motion of the control drums is started.

The controller sequence is set up to drive only one drum at a time and to make only one 0.26° step at a time. For example, the automatic controller will drive No. 2 one step (0.26°), pause 1.1 sec, drive No. 4 one step, pause for 1.1 sec, then drive No. 6 one step, pause 1.1 sec and repeat the sequence. The driving time (~ 1.4 sec) and the length of the pause (~ 1.1 sec) are set to yield one step every 2.5 sec. This continues until the startup drums have been driven to the full-in position. To stop the startup drums at some position other than full-in a set of preset angle switches is provided; these switches can stop the startup drums at any preselected angle between full-out and full-in. The controller then starts to drive the control drums in sequence, one step at a time.

Stepping of the three control drums can be performed at a variety of rates. Although the startup drums move one step every 2.5 sec, the stepping rate for the control drums changes as the startup progresses. The control drums start their stepping from the full-out position at the rate of one step every 2.5 sec (the same as the startup drums), but in rising to temperature three changes in rate are made. As the reactor approaches the critical drum configuration, the rate changes to Intermediate Rate 1 which is adjustable to allow one step every 10 to 180 sec. After passing through the critical drum configuration the rate changes to Intermediate Rate 2 which also allows one step every 10 to 180 sec

but usually is set for a faster stepping rate than Rate 1. Drum insertion will continue at Rate 2 until the reactor outlet temperature reaches the desired operating point as indicated by temperature switches controlled by RTD's in the reactor outlet line.

Once the operating temperature is reached the controller begins functioning as a deadband controller, and the temperature switches stop further drum insertion until the temperature falls below or rises above a preset deadband. If the temperature falls below the low end of the deadband the controller re-institutes the stepping of drums at Intermediate Rate 2 until the temperature reenters the deadband. If the temperature rises above the deadband the controller changes to a Long-Term Rate which is adjustable to allow one step every 110 to 720 sec. The controller then begins stepping drums out at this rate until the temperature drops to within the deadband. Once the controller has switched to the Long-Term Rate all subsequent drum motion, either in or out, will be at this rate.

The controller receives its startup drum stop signal and Rate 1 and Rate 2 rate-change signals from the precision drum-position indicator and a preset angle switch assembly. The preset angle switch assembly has registers which allow adjusting the switches to stop the startup drums or change the drive-rate at any angle between full-out and full-in. Variability in these settings allows compensation for changes in the critical position as the excess reactivity changes with time.

The following operations must be performed before starting up with the automatic controller:

- (1) Determine the true critical position.
- (2) Preset the auto controller to drive the startup drums to full-in or whatever other position is desired. The current excess reactivity at this time may dictate only partial startup drum insertion. Partial insertion requires setting the preset angle switch assembly to stop at the desired angle.

(3) Set the preset angle switches to change to Intermediate Rate 1 before reaching the predetermined critical drum position, and to Intermediate Rate 2 after passing through the critical drum position.

(4) Preset the rates for Rate 1, Rate 2, and the Long-Term Rate. These adjustments will be covered by individual test procedures. Having multiple rates provides flexibility in studying the response of the controller and reactor system.

(5) Set up the temperature switches and their deadband to control at the desired temperature. This action will also be covered by specific test procedures. Flexibility in setpoint and controlling deadband will also be useful in studying system responses.

The only sensor used by the automatic controller in its operation is the coolant outlet temperature-sensing system. Signals from this system are not used in any manner by the reactor protection system. The temperature-sensing system consists of a platinum resistance temperature detector and a temperature switch module. One module is provided for each of six detectors installed in the reactor coolant outlet line. Two detectors and associated modules are utilized as a redundant pair to provide control. The other four are available on a standby basis.

Because of the long-term drift characteristics of the platinum RTD's, two detectors and switch modules are utilized as a pair in providing "high" and "low" deadband signals to the controller. The switch logic insures that the reactor will be controlled by the "undrifted" or "least-drifted" detector.

Based on development test data, the detectors are expected to increase in resistance, i. e., indicate a measured temperature which is higher than actual. Therefore, the low deadband switches are wired in parallel so that the switch indicating the lowest temperature, the least drifted, causes the controller to function. The high deadband switches are wired in series so that both switches must close in this case. This insures that controller again functions only when the lowest temperature or least-drift switch actuates.

The auto controller also uses the precision drum-position indicator (PDPI), but this is not a true sensor. It counts drum-controller step pulses and adds or subtracts 0.26° with each step. A check circuit in the PDPI verifies the actually detected drum motion of the reactor coarse drum-position indicators (CDPI) following each controller pulse; to determine whether to add or subtract 0.26° the PDPI also checks in what direction the CDPI's sensors detected motion. The PDPI indications for the six drums provide the input signals which trip the preset angle switches. Neither the PDPI nor the CDPI's have anything in common with the reactor protective circuit. Therefore, failure in either of these circuits will not compromise safety circuit operation.

C. SINGLE-FAILURE CRITERION

Question: "Please discuss the single failure criterion regarding the subject of penetrations and cable run isolation to show that no single failure will prevent a reactor scram."

Answer: This criterion was not specifically invoked on a lead-by-lead basis at the time of basic facility and protective system design selection in 1964; the criterion as it exists today was not formulated at that time. The design did, however, make use of the total experience accumulated at Atomics International in designing and operating more than a dozen similar facilities and test systems. Careful review of the S8DR facility and protective system shows that design features have been used which make the systems capable of meeting this requirement.

First, the reactor protective system has been designed to safely execute reactor shutdown required by one or two failures.

Second, signal-type wiring has been run separately from power-type to eliminate spurious signals on low-level scram instrumentation. This has been done by designing specific cable trays or conduits for specific usages. Further protection has been given by using shielding braid on all signal leads that are a part of the protection system.

Third, the cabling for nuclear instrumentation is continuously monitored by the pulse test circuitry. Any failure of this cabling would be immediately annunciated as failure of a nuclear instrumentation channel. Redundant channels are provided so that the failure of a single channel does not render this part of the overall protective system inoperative.

Fourth, special protection is given to the vacuum vessel conduit headers. Because of the need for the feedthrough of several signal and control leads from the vacuum vessel, it was necessary that conduit headers be installed at the vault floor level. A steel cover-plate eliminates the possibility of damaging these headers.

A steel cover plate also protects all cable runs in the high bay. Cable runs in the control room are located under a computer-room type floor in covered cable trays.

Although cable separation to avoid single failures was not specifically used, it is believed that with the existing design features a single failure which would disable the protective system cannot occur.

D. REACTOR PROTECTION SYSTEM INSTRUMENTATION

Question: "The description and block diagram provided for the Reactor Protection System instrumentation are not in sufficient detail to review for safety adequacy. Please provide diagrams or schematics and show how all individual trips (relays, contacts or pressure sensors, etc.) are obtained, how they are incorporated into each level of logic, where and how bypasses are introduced and how the logic acts to interrupt power to the scram clutches."

Answer: The following schematic drawings are shown here to more clearly describe the operation of the reactor protection system instrumentation:

S8DS-94206: Nuclear System (Figure 7)
D636721: Trip Circuit Card (Figure 8)
S8DS-94204: Auxiliary System (Figure 9)

S8DS-94207: Reactor Startup and Protective System (Figure 10)
S8DS-94203: Process Primary System (Figure 11)
D639175: Logic Amplifier and Magnet Switch, Main Chassis (Figure 12)
639091: Logic Amplifier with Pulse Testing (Figure 13)
B638595: Magnet Switch (Figure 14)

1. Nuclear Instrumentation System Trip Circuits

A standard physically and functionally interchangeable trip circuit module in each nuclear channel provides an indication of a scram condition to the logic amplifiers, Figure 7. A schematic of the trip circuit module is given by Bendix drawing Figure 8.

The trip circuit compares an input voltage signal with an externally supplied reference voltage. The card may be connected to act as either a high- or low-level trip, on signals of either positive or negative polarity. In the first case, if the input voltage does not exceed the trip level the output of the module is +15 volts; when the trip level is exceeded the output suddenly drops to about 0 volts. In the second case, when the input exceeds the trip level the module is untripped and the output is +15 volts, dropping to about 0 volts when the input goes below the trip level.

In addition to providing the electronic outputs, the trip circuit also supplies power for a backup relay for annunciation. The relay is energized whenever the trip output is +15 volts, but is deenergized when the trip output drops to 0 volts.

The trip circuit is one of the most critical units in reactor instrumentation because it generates signals that control the scram function of the reactor. The Bendix trip circuit is the result of intensive study and development. The fast (electronic) output is safe against the shorting of any transistor or the opening of any diode; that is, a single occurrence of either type will invariably cause the circuit output to drop to 0 volts. Further, the design of the circuit causes transistors to fail short if they fail at all. Failure in the relay circuit (which is not within the failsafe provisions) will not prevent the circuit from

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NOTES:
 1. ITEM 26 (1N914) MAY BE ALTERNATE FOR ITEM 22 (1N457) BUT ALL (S) DIODES ON ANY ONE CARD MUST BE SAME TYPE.
 2. USE PLUG GUIDE NO. 126638-2.

REVISIONS			
LTR	DESCRIPTION	DATE	APPROVAL
A	CHANGE RESISTOR NO. R18 TO R10 ADD NOTES 1 & 2	4-22-65	S.H.H.

		55		
		54		
AR	CW70-22-90	53	WIRE-HOOK UP (TEFLON)	WHT/BLK
AR	CW70-22-7	52		V10
AR	CW70-22-6	51		BLU
AR	CW70-22-2	50		RED
AR	CW70-22-1	49		BRN
AR	CW70-22-0	48	WIRE-HOOK UP (TEFLON)	BLK
		47		
AR	CT90-22-4	46	TUBING-TEFLON	YEL
AR	CT90-22-5	45	TUBING-TEFLON	GRN
AR	CT90-22-6	44	TUBING-TEFLON	BLUE
AR	CT90-16-9	43	TUBING-TEFLON	TRANS
AR	A000668-1	42	WIRE-BUS	#22 AWG
AR	A001654-2	41	SOLDER	
2	HC275502-56	40	NUT-HEX	#2-56
1	HC275504-40	39	NUT-HEX	#4-40
2	HC299602-M	38	WASHER-LOCK	#2
1	HC299604-M	37	WASHER-LOCK	#4
2	HC850516-256	36	SCREW-BINDER HD.	#2-56 x 1/2"
1	HC450508-440	35	SCREW-BINDER HD.	#4-40 x 1/2"
1	A001045-2	34	TERMINAL-SOLDER	
58	B002354-3	33	TERMINAL-FEEDTHRU	
5	B001964-1	32	CLIP-INSULATED	
		31		
		30		
1	A002016-11	29	TRANSISTOR 2N1132	Q3
4	A002016-9	28	TRANSISTOR 2N1711	Q1, 2, 4, 5
		27		
5	B001237-6	26	DIODE 1N914 (ALT. TO ITEM 22)	SEE NOTE 1
1	B001237-19	25	DIODE 1N9658	CR6, 8
1	B001237-18	24	DIODE 1N9678	CR5
1	B001237-12	23	DIODE 1N9638	CR3
5	B001237-8	22	DIODE 1N457	CR1, 2, 4, 9
		21		
1	C000701-102	20	RESISTOR 1K 1/2W 5%	R10
1	B001407-8	19	RESISTOR, VARIABLE 10K	R17

1	C000701-103	18	RESISTOR 10K 1/2W 5%	R16
1	C000701-153	17	15K 1/2W 5%	R15
1	C000701-682	16	6.8K 1/2W 5%	R13
1	C000701-152	15	RESISTOR 1.5K 1/2W 5%	R12
		14		
		13		
1	C000701-472	12	RESISTOR 4.7K 1/2W 5%	R9
1	C000701-222	11	2.2K 1/2W 5%	R8
1	C000701-392	10	3.9K 1/2W 5%	R7
1	C000701-333	9	33K 1/2W 5%	R6
2	C000701-332	8	3.3K 1/2W 5%	R3, R4
2	C000701-473	7	47K 1/2W 5%	R2, R5
3	C000701-104	6	RESISTOR 100K 1/2W 5%	R1, R14, R11
		5		
		4		
		3		
1	CL38534-1	2	CONNECTOR & GUIDE PLUG ASS'Y	
1	EP1224279-1	1	CARD-COMPONENT	

LIST OF MATERIAL

Figure 8. Trip Circuit Card
(Dwg. D636721)

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operating, or affect the electronic output. The relay circuit is used for annunciation only.

When the circuit is untripped (see Figure 8), Q2 and -5 are conducting, and Q1, -3, and -4 nonconducting. When the voltage at the base of Q1 becomes positive to the voltage at the base of Q2, Q1 conducts. The current through Q1 flows to the base of -3 and saturates -3 and -4. When Q4 is turned on, the voltage at its collector (and at pin 15) changes from +15 volts to about 0 volts, indicating that a trip has occurred.

If either Q1, -3, or -4 shorts, the action is the same as though a high input were furnished the input terminal, i. e., a trip occurs. If Q5 shorts, the relay output remains energized but the electronic output is not affected. Operation of the electronic output is not dependent upon Q5. If Q2 shorts, breakdown diode CR3 conducts, driving Q4 into saturation. Thus, if any of the transistors affecting the electronic output should short, a tripped output is presented to the external logic.

If CR5 opens, the voltage at the emitter of Q3 becomes more positive. The circuit would continue to operate normally except at high temperatures. At high temperatures a transistor with very high leakage currents will turn on, causing a failure in the safe direction.

The failure of a single active component in the expected mode (i. e., shorting) thus cannot cause the electrical output to fail in an unsafe direction. Since the relay circuit is used for annunciation output only, failsafety in this stage is not as required.

If a failure in the unexpected mode (i. e., open circuit) should occur, a malfunction would be detected and annunciated by the pulse test monitoring system discussed in the SAR. The channel would then be immediately bypassed, which would in turn put it in the scrambled condition.

2. High Reactor Outlet Temperature Scram

Scram from high reactor outlet temperature originates from three separate and independent electronic trip units connected in two-of-three logic. The three thermocouples for these units serve only this function. The units are shown schematically on Figure 9, and the location of the scram contacts in the scram system in Figure 10. Removal of any one trip unit for maintenance or calibration is permissible. If this becomes necessary, that unit goes to a tripped condition and the scram logic requirement is reduced to one-of-two.

3. Loss of Primary Coolant Flow

The loss-of-primary-flow scram system is shown schematically on Figure 11. The input for the scram logic system is provided by the opening of a set of contacts in flowswitch FS-102. The flow signal for FS-102 is provided by flowmeter FR 11-102.

4. Earthquake Scram (Figure 10)

The input for the earthquake scram is obtained directly from the earthquake switchblock. This switch is normally closed.

5. Manual Scram

The manual scram switch has separate switchblocks which perform the following functions:

1. Break the +15-volt logic into the "OR"-logic input.
2. Break the ac input supply to that for the scram magnet clutch.
3. Deenergize the relay whose contacts break the ac supply to the scram magnet clutch.

6. Loss of Primary Pump Power (Figure 11)

Whenever power is lost to the primary coolant system pump a reactor scram is started by the open-phase relay (OPR-1). This scram is faster-responding than that from loss of flow whenever pump power is lost.

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REVISIONS				
SYM	ZONE	DESCRIPTION	DATE	APPROVED
A		1) GENERAL REVISIONS	10/17/66	WILSON
B		1) GENERAL REVISIONS	10/17/66	WILSON
C		1) CHANGED FUT. PLS CONTACTS TO CPM-A AT PB-48	10/17/66	WILSON
D		1) ADDED RELAY CONTACTS K510 & K520 TO CLUTCH PWR SUPPLY	10/17/66	WILSON
E		SEE REVISION RECORD	10/17/66	WILSON

Figure 10. Schematic Diagram, Reactor Startup and Protective System (Dwg. S8DS-94207)

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7. Logic Amplifier and Magnet Switch (Figure 12)

Logic inputs from the various monitoring channels are +15 volts for a safe indication, and about 0 volts for a trip output. The 0-volt signal may be represented by either a ground or an open circuit; thus, whether the logic line becomes broken or shorted to ground a trip input always results.

Figure 12 shows that the inputs are connected directly to the logic amplifier through 2.7k resistors. Switch S2 operates on these input lines to permit manual test of the entire channel. Input signals are combined in the logic amplifier module (Figure 13), the outputs of which connect directly as control signals to the magnet switches. One side of all the magnets is common and is connected to high-current power supplies, providing from +24 to 32 volts. The other side of each magnet is connected directly to the magnet switches in the two magnet switch channels. The magnet switches contain OR diodes in the power lines to the magnets, and each switch is capable of handling the full current requirement of a magnet without approaching rated dissipations of the circuit elements.

The logic amplifier module receives signals from the channel trip circuits and combines them in a predetermined manner to control the magnet-switch modules. The schematic shows that the input lines connect to various gates to effect the desired logic. These gates are arranged so that either opening or grounding the input will cause the gate output to drop to near 0 volts.

So long as all inputs are at the safe level of about +15 volts, the junctions at such diode pairs as CR8 and -19 will be at +15 because current will flow toward the -25-volt supply through the 47k resistors R26 to -31 and the diodes CR8 through -18 from the input terminals. Removing the proper combination of inputs causes the junction to drop toward 0 volts. For example when the input to CR8 is removed, the junction between CR8 and -19 tends to drop. The inputs to both CR9 and -10 must be removed for the junction of CR9, -10, and -20 to drop; CR9 and -10 therefore form an AND gate.

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- NOTES:
1. ALL RESISTORS $\frac{1}{2}$ W $\pm 5\%$ TOLERANCE EXCEPT AS INDICATED.
 2. ALL DIODES 1N457 EXCEPT AS INDICATED.
 3. ALL RESISTANCE IN OHMS.

Figure 13. Schematic, Logic Amplifier With Pulse Testing (Dwg. 639091)

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In the safe condition, transistors Q9, -10 and -12 through -14 are all turned off. The emitter of Q9 is held at about +9 volts by R36 and CR31, while the collector is held near 0 volts by R37, CR30, and CR32. Diode CR32 also assures that Q10 is turned off and diode CR33 assures that Q12 through -14 are turned off. When the proper combination of logic inputs drops to near 0 volts, one or more of the diodes CR19 through -24 conducts and turns Q9 on. This causes Q10 to conduct and turns on -12 through -14. The collectors of these last transistors will drop to about 0 volts.

Diodes CR27 prevent transistor Q9 from saturating in the ON condition. Diodes CR28 and -29 serve the same function for Q10.

The logic amplifier module has been provided with a pulse-testing feature that continually checks the operability of transistors Q9 through -14. This is an important feature of the logic amplifier because of the importance of these transistors in scrambling the reactor.

The pulse tester consists of a pulse generator which effectively gives out the equivalent of about 1000 5- μ sec scram signals every second. These signals are coupled into the logic network through CR25 in exactly the same manner as any other logic input. Transistors Q9, -10, -12, -13, and -14 are turned ON as though a scram signal had been fed into the logic amplifier, but the subsequent transistors in the magnet switch are too slow to react to these fast pulses. The collectors of Q12 through -14 are connected through an AND gate into the base of Q11; the pulses must appear at all three collectors before they are fed to Q11. From the collector of the latter, a buffer amplifier which inverts and amplifies the pulses, comes a continuous series of positive pulses which go into the pulse monitor circuit. If the pulses fail to reach the monitor circuit an alarm is given.

The magnet switch module (Figure 14) is a solid-state high-speed switch controlling the current through the control-drum magnets or clutches. One end of each magnet winding is commoned and connected to the magnet power supply,

NOTES:

1. USE FLUG GUIDE NO. 126638-75

Figure 14. Schematic, Magnet Switch
(Dwg. B638595)

which is also connected to terminal 19 of the magnet switch. The other end of each of the windings is connected to an individual magnet switch card, by means of paralleled terminals 22 to 25. The windings are shunted by CR6 and -7. These two diodes keep inductive surges within limits that will not damage the transistors, at the same time permitting maximum speed of clutch release.

The magnet is in the collector circuit of transistor Q2, whose emitter is returned to the negative terminal of the magnet power supply. When the magnet is energized the collector is near 0 volts, and Q3, whose emitter is grounded, is cut off. Releasing the magnet causes the collector of Q2 to rise to the voltage of the magnet supply, and Q3 turns on. The latter is used to control an associated indicator lamp on the main chassis.

The state of Q2 is controlled by emitter-follower Q1. The emitter of Q1, when it is conducting, is at about the same potential as the base. In the safe condition the base receives current from the output of the logic amplifier, and both Q1 and -2 are turned on. When a scram output exists, the base of Q1 can be pulled about 1.5 volts negative to ground, which positively turns it and Q2 off.

Diodes CR1 and -4 both prevent destructive reverse biasing of their associated transistors. Diode CR5 is an OR diode which permits two magnet switch cards to be connected to a single magnet and accordingly allows either card to be checked for proper operation without actually releasing the magnet. When one of the two cards is being checked, its terminals 22 to 25 will remain near 0 volts and its CR5 will not conduct. The second trip card will continue to function, thus assuring scram capability. The collector of Q1 being at about +12 volts, diode CR2 will furnish about +7 volts to the collector of Q2 and sufficient current to the base of Q3 to light the indicator lamp. Diode CR3 disconnects -2 when the switch is operating normally, i. e., when the magnet has been turned off by a scram signal entering both paralleled magnet switches.

E. NON-NUCLEAR SINGLE-FAILURE SCRAM

Question: "Explain and provide assurance that a single failure in any non-nuclear scram input circuit can be tolerated and the system still retain ability to prevent the reactor from operation beyond protective set point. "

Answer: The non-nuclear scrams provided for the S8DR facility are:

1. High reactor-outlet temperature (two-of-three requirement)
2. Low primary flow
3. Earthquake
4. Manual scram

The single-failure criterion is discussed for each of these in this section.

1. High Reactor-Outlet Temperature

Three independent thermocouples feeding separate amplifiers are used. These thermocouples are solely for the protective circuitry and not for reactor control or operating data. Their output signals are fed to a relay matrix whose logic hookup provides a scram signal on a two-of-three overtemperature situation. The circuit is failsafe in that loss of power or failure in the common line will open the relays and cause a scram. Upscale protection is provided in the case of an open-circuited thermocouple. A temperature signal less than the setpoint value is required to engage the relay. The single-failure criterion is satisfied by the signal redundancy.

2. Low Primary Flow

Low primary flow is detected by a single sensor and its associated scram circuit. For this parameter the reactor outlet-temperature circuits discussed above provide backup in a flow detector circuit failure. Any reduction in primary flow would immediately cause an increase in reactor ΔT . As soon as the reactor outlet temperature reaches its setpoint limit these high outlet-temperature safety circuits would start a scram. The single-failure criterion has been met in this manner.

3. Earthquake

The earthquake scram is a nonredundant circuit so that it does not by itself meet the single-failure requirement. However, this scram is incorporated in the protective system to anticipate damages which might occur in a severe earthquake. Such damages are in themselves covered by scram circuits or protective system design as in the case of loss of system power. Thus in this situation, detection of an off-normal condition (e. g. , low flow, high temperature) that the earthquake scram would normally anticipate would provide the backup signal.

4. Manual Scram

For manual scrams the single-failure criterion is met in the S8DR system by the provision of three independent means:

a. The Manual Scram Button

This is a push-pull switch which simultaneously cuts power to the scram clutches and the logic power supplies. Once the switch is pushed, scram action cannot be interrupted. The entire protective system must be reactivated before the reactor can be restarted.

b. The Scram Evacuate Switch

This switch causes manual scram through a relay which cuts power to the scram clutches as well as starting vault isolation, heater shutdown, control center alarm, and evacuation signals. Again, this action cannot be stopped once started.

c. Reactor Console Key Lock

This key controls power to the reactor startup and protective systems. In the off position all power is cut to the protective system and the scram clutches.

F. PERIOD SCRAM CRITERIA

Question: "Is the period scram required to prevent cladding failure? If so, state the criteria that will be used to determine how long the period by-pass may be tolerated. The reply should take into account the probability of an unsafe failure in the single operative channel during bypass. Also, it is noted that the bypass switch is common to both intermediate channels. Provide assurance that failure of this single component cannot occur such that both channels would be bypassed simultaneously."

Answer: The period scram is not required to prevent cladding failure. This is discussed in Section 11, page 177 of the SAR, with a simulated startup accident without period scram shown in Figure 11.1. The SAR concludes, "If this startup accident were to occur with the reactor at its highest temperature condition (1300°F), a maximum fuel temperature of 1500°F would occur. Therefore, these incidents would not result in damage to the cladding." The highest fuel temperature which would be reached in this type of accident is well below the 1635°F minimum average fuel temperature required for cladding damage.

The startup analysis has clearly shown that any failure of a single operative channel will not result in an unsafe reactor condition. This factor was recognized in the design of the bypass switch into this system, and therefore the single-failure criterion was not invoked.

G. SCRAM CIRCUIT TESTING

Question: "Describe what tests have been or will be performed prior to reactor criticality to assure that no failure of a test circuit will prevent proper scram trip action."

Answer: Before reactor criticality the nuclear safety circuits will be thoroughly tested and checked to assure that all instruments and sensors are wired properly, calibrated, and will meet the safety system specifications. These tests will include checks of the operation of the built-in test circuits. These self-checking

test circuits pass signals through the nuclear power channels and cause the scram trip cards to respond as described on page 106 of the SAR.

For these prenuclear checkout tests the test circuits will be adjusted or disabled as necessary to check the following effects:

- 1) Lack of test pulse completion and/or continuity.
- 2) Variable frequencies, pulse widths, and amplitudes.
- 3) Random noise in the circuits.

At this time no credible failure in the test circuit has been postulated which will cause this circuit to prevent proper scram action. During nuclear operation a check of the pulse monitors will be made on a weekly basis to ensure that the test circuits and the malfunction alarms are functioning properly.

A failure of the test signal pulse generator which either reduces the amplitude of the test pulse or eliminates it will result in an alarm signal from the pulse monitor, but will have no effect on the safety circuit response. A pulse generator failure which applies voltage continuously at the maximum level available to it (25 volts) will not cause a significant enough change in the 750-volt ion chamber voltage to affect the chamber's operation. This type of failure will also result in an alarm from the pulse monitor circuit as soon as the pulses stop. A failure which increases the pulse frequency to the point of continuous oscillation will cause the trip card to remain in the tripped condition; the channel concerned will issue a scram signal, which will result in an operational but not a safety problem.

H. NUCLEAR CHANNEL COMPONENT REMOVAL

Question: "Provide assurance that an evaluation has been or will be performed to determine that removal of any channel component and/or detector cable can be accomplished safely with only an alarm being annunciated."

Answer: The requirement for being able to remove any channel component and/or detector cable without starting scram is only necessary while the plant is at power; consequently only the nuclear power channels are involved. The ability to perform maintenance, checks, and repairs on individual components on any one of these three channels without interference with the long-term safe operation at power is a design feature of the nuclear protective system. This can be done safely since the two-of-three power level criteria is reduced to a one-of-two scram condition, with either of the two remaining channels capable of giving a scram signal when the third channel is out of service. Two channels out of service automatically initiates a scram. Demonstration of the system's ability to perform as designed in this regard is a scheduled part of the facility preoperational checkout.

I. DRUM ACTUATION SYSTEM

Question: "Discuss the drum actuation system in sufficient detail to show how multiple drum movement is precluded in both automatic and manual modes. The analysis performed should justify a conclusion that no single failure could cause a reactivity insertion rate in excess of the safety limit (1.5¢/sec)."

Answer: Several features have been incorporated into the design of the manual and automatic control systems to preclude multiple drum movement. These are:

- (1) Each of the six control drum actuators is powered by its own separate controller-power supply.
- (2) The output of each controller-power supply is disconnected from its actuator by relay contacts until it is selected by the drum selector switch when operation is in manual mode, or by the auto controller sequencer when in the automatic control mode.
- (3) The input circuitry to each controller-power supply is disconnected until selected by either the drum selector switch or the auto controller sequencer depending on which mode of operation is being used. This

feature and the output selection feature discussed above function on contacts from separate relays. Therefore a failure in one does not disable both systems.

(4) In the manual mode the drive-in and -out commands for the controller-power supply are selected by a single "joy stick" type switch. In addition to switch disengagement, the drive-in function is disabled by the drive-out. In the automatic mode the drive-in and -out commands are received from independent relay contacts. Again, the drive-in function is disabled whenever the drive-out function is enabled.

The above design features were installed specifically to prohibit the actuation of more than one drum during automatic and manual operation modes.

A still further feature which has been designed into the actuator controller-power supply is that which limits the reactivity insertion rate to less than 1.5¢/sec. Two separate circuits control the time required for the actuator to step. The first of these is the pulsing circuit. This controls the pulse duration time which is variable from 400 to 500 msec. An adjustable pulse overlap of 25 to 75 msec is also part of this circuit. If the pulse width and overlap adjustments were set up in the worst combination (i. e., 400-msec pulse duration and 75-msec overlap) the minimum time to complete a quarter of a degree drum rotation (3 pulses) would be 1050 msec (1.05 sec). A quarter of a degree of rotation corresponds to 1.5¢ reactivity at the point of maximum differential worth. Thus for the case of continuous pulsing of the most pessimistic settings the maximum reactivity insertion rate would be less than 1.5¢/sec.

A failure of the pulsing circuit could generate pulses of a shorter duration than the 400 msec minimum. Tests to find out the shortest-duration pulses which would still operate the actuator showed that with pulses of less than 50-msec duration the actuator would not function. Therefore, the minimum time required for a quarter degree of rotation would be 150 msec. To meet this possibility of too-short pulses a second time-delay circuit is included. This

circuit delays the pulsing train after each third pulse for 1.25 sec with the result that the minimum time required for a quarter degree of rotation is 1.4 sec. This is well beyond the minimum of 1 sec which is the 1.5¢/sec safety limit.

J. NEUTRON FLUX ERRORS

Question: "Provide assurance that the maximum errors expected in the neutron flux signals furnished to scram trip units from the IRM (intermediate range) and PRM (power range) channels would not result in cladding failure."

Answer: The intermediate range channels furnish scram trip signals for reactor periods only. Power-level scram trip signals are not provided.

The power-channel scram trip points will be set at $120 \pm 5\%$ of the expected operating power level. This permits 1200 ± 60 kw for 1-Mw operation and 720 ± 36 for 600-kw operation before a power scram occurs. An additional 2% nuclear channel calibration variance from the calorimetric power determinations is assumed. The scram trip points for the power instrumentation will be checked monthly.

The maximum fuel temperatures would occur if the power were to rise to just below the scram trip point and then come to equilibrium. Fuel temperatures can be estimated from the reactor power and from the average fuel-to-coolant heat-transfer coefficient UA, which is shown in Figure 15. These UA values are taken from a recently completed evaluation of the S8DR core⁽¹⁷⁾ which takes into account the most recent information on the as-built fuel elements, hydraulic tests, hydrogen-loss tests, and fuel-growth tests. The most conservative average fuel temperature at which an initial cladding rupture would occur in the peak temperature location is approximately 1635°F. This estimate is conservative, since the irradiated (end-of-life) cladding properties were assumed, as was the highest as-built fuel-rod hydrogen concentration, which determines the internal pressure.

Figure 15. Reactor Heat-Transfer Coefficient UA at End-of-Life

The 600-kw and 1-Mw cases will be considered separately. Normal inlet and outlet temperatures for 600-kw operation are 1100 and 1300 $^\circ\text{F}$ respectively. If the power were to rise to the scram trip point with allowance made for the above-mentioned errors, the following conditions would apply:

$$\begin{aligned}
 P &= 771 \text{ kw} \\
 T_{\text{inlet}} &= 1100^\circ\text{F} \\
 T_{\text{outlet}} &= 1357^\circ\text{F}
 \end{aligned}$$

Allowing an additional uncertainty of 5% in the calculated calorimetric power gives a power of 810 kw to use with the minimum UA from Figure 15 for the fuel-temperature determination. The results show a UA of 3.95 kw/°F and a fuel ΔT of 205°F. This is added to the average coolant temperature $[(1100 + 1357)/2]$ to obtain 1434°F as the average fuel temperature. This value is well below the 1635°F minimum damage point. In addition, the system is further protected in that a rise of 43°F in system temperature would result in a reactor outlet-temperature scram at 1400°F.

Normal inlet and outlet temperatures for 1-Mw operation are 830 and 1100°F respectively. Following the same procedure as for 600-kw operation we obtain an outlet temperature of 1177°F, a maximum credible reactor power of 1349 kw, a UA of 3.7 kw/°F, and a fuel temperature of 1368°F. Again, the system is nowhere near the 1635°F damage point. In this case the system temperature would rise only 23°F before the reactor outlet-temperature scram setpoint at 1200°F is reached.

K. STACK MONITOR DELAY TIME

Question: "Specify delay time between actual release of activity to the stack and time of receiving alarm or annunciation. What effect would this have on safety considerations?"

Answer: Vault exhaust gases are monitored before they reach the stack by a gamma probe on the test vault exhaust line. This gamma probe has a time constant of 20 sec for a slow change in gamma level. It has an appreciably faster response to step changes. Any release of fission products would be detected by this monitor, which will then automatically seal off the test vault nitrogen inlet and reroute the exhaust gas from the stack to the underground holdup tanks. Since this time delay is small it does not have an appreciable effect on the assumptions of the radiological analysis; therefore, the conclusions of that analysis are not altered.

The stack monitor is thus not relied upon for primary safeguards protection. The delay time during which particulate material travels up the stack to the sample point and then through the sample line to the stack monitor is about two seconds. The time constant for the stack monitor itself is variable between 0.5 and 50 sec. Thus the longest time delay would be about one minute for alarm or annunciation of any release to the stack itself. For large activity increases such as 2 decades above the initial reading, the delay would only be about 3 sec, even with the long-time constant. Thus for any accident situation there are two monitoring points for potential stack release activity. Both monitors have step-change response times of a few seconds.

VII. SECONDARY COOLANT AND AUXILIARY SYSTEMS

No questions.

VIII. RADIATION PROTECTION SYSTEMS

No questions.

IX. CONDUCT OF OPERATIONS

No questions.

X. INITIAL TESTS AND OPERATIONS

No questions.

XI. SAFETY ANALYSIS

A. MAXIMUM REACTIVITY TRANSIENT AND LOSS-OF-COOLANT ACCIDENTS

Question: "Please provide additional detail information concerning the sequence of events for both the maximum reactivity transient and the loss-of-coolant accidents. For the loss-of-coolant accident, the answer should include a discussion of how fuel and core are expected to fail and the volumetric flow rates of hydrogen out of the reactor, the components of pressure rise in the containment chambers (NaK and H₂) vs time for postulated primary system failures, both within the vacuum chamber and the secondary containment vault."

Answer: The following information covers the two situations mentioned in the question.

1. Maximum Reactivity Transient

a. Initial Conditions

The conditions assumed to exist at the time of the maximum reactivity insertion are as follows:

- (1) Coolant at 1100°F.
- (2) Reactor at 1000 kw.
- (3) Maximum core operating history.
- (4) Reactivity insertion rate of 1.5¢/second, starting with a control drum which is initially full-out.
- (5) Vacuum vessel (primary containment) at vacuum.

Assumptions (1) through (4) result in the largest possible energy and fission-product contribution to this accident. Assumption (5) is an essential condition for reactor operation, so it must be considered an initial condition for the primary containment.

b. Sequence of Events

The reactivity insertion is assumed to continue throughout the course of the transient accident until the control drum is fully inserted, at which time the drum is stopped mechanically. The course of this accident transgresses several reactor scram points, namely, high reactor power, high reactor-outlet temperature, and loss of flow, without causing a scram. Reactor fuel temperatures are increased until the average reaches 1750°F, which is the highest that can be reached without rupturing elements in the peak-temperature locations by internal hydrogen pressure. Table 10 shows the sequence of this particular accident, unprotected by scram, with starting conditions of 1000-kw 1100°F outlet temperature.

TABLE 10
MAXIMUM REACTIVITY TRANSIENT

Elapsed Time (sec)	Power (kw)	Fuel Temperature (°F)		Average Coolant Outlet Temperature (°F)
		Average	Peak	
0	1000	1230	1440	1100
10	1280	1275	1490	1125
20	1700	1380	1650	1200
30	2160	1540	1870	1290
40	2590	1715	2100	1410
42	2700	1750	2160	1430

As H_2 is released from the fuel elements, reactivity is lost rapidly and the reactor shuts down. This mechanism is fully described later in XI-B of this section, where it is shown that at fuel temperature of 1750°F reactivity can be lost by H_2 diffusion at a rate of 2% $\Delta k/k$ per second.

Pressure buildup from H_2 lost from the fuel can be relieved in any of three ways:

- (1) Pressure relief of the primary-NaK system to the system holdup tank.
- (2) Rupture of the core vessel, the weakest part of the primary-NaK system, into the primary containment.
- (3) Rupture of the primary-NaK piping system into the secondary containment.

The first of these is the most likely.

This would not violate the integrity of the primary system, and no fission products would be released to the primary or secondary containment. However, if the second way listed above occurred, spilling NaK and releasing H_2 , a pressure increase in the vacuum chamber would occur.

c. Effect on Containment

From consideration of (1) tabulation of outlet temperature vs time for this accident, (2) a loop time of about 36 sec at 1000 kw, and (3) a ΔT of 270°F, the average temperature of NaK in the loop is 1100°F (without consideration of colder NaK in the cold trap). Putting this inventory of 1100°F NaK into the vacuum vessel (at 10^{-5} torr) will raise its pressure to ~2 psia. Although the energy required to release all the H_2 is not available, the dumping of all H_2 from the core into the vacuum vessel would increase the pressure to ~21 psia or ~7 psig. The high heat capacity of the vacuum vessel and ducting would rapidly cool this overpressure. Thus, assuming no vessel cooling, the containment system would be above atmospheric pressure for only about 2 min. The leakage from the primary containment during this time would be about $1.4 \times 10^{-5}\%$ of its volume.

d. Fission Product Release

Assuming that the fission products are released with the H_2 , even though their diffusion times are orders of magnitude greater than for hydrogen, only a small amount, $\sim 10^{-7}$, are available to create a radiological hazard. Because this is significantly less hazardous than the loss-of-coolant accident in the secondary-containment vault, the radiological effects were not calculated.

e. Conservatism of Analysis

The conservative factors in the above analysis are summarized below:

- (1) Continuous insertion of reactivity is due only to gross operator negligence.
- (2) Fuel damage is caused by absence of scram action at high power and high outlet temperature.
- (3) Hydrogen lost from the fuel can be diffused out the NaK piping or tank without seriously overpressuring the primary system.
- (4) Insufficient energy is available to drive off enough hydrogen to raise the containment pressure to atmospheric.
- (5) The fission products, except those existing between the fuel and ruptured cladding, cannot diffuse out in the same time as the hydrogen.

2. Loss of Coolant

a. Initial Conditions

The conditions assumed to exist at the time of the loss-of-coolant accident are as follows:

- (1) Coolant at 1200°F (half at 1300°F and half at 1100°F).
- (2) Reactor at 1000 kw (although not consistent with 1300°F coolant outlet).
- (3) Maximum core operating history.
- (4) Double-ended failure of primary-coolant pipe.
- (5) Containment vault atmosphere at 0.1% oxygen or less.

Assumptions (1) through (4) result in the largest possible energy transfer to the vault atmosphere and the largest fission-product inventory available for release. Assumption (5) is an essential condition for reactor operation so it must be considered an initial condition for the secondary containment.

b. Sequence of Events

The mechanism for starting a loss-of-coolant accident is not defined. Because of the design, construction, and testing of the primary system, and because of the short pipe lengths, low pressure, and pipe ductility, it is difficult to imagine a failure causing a large opening and rapid loss of coolant. Further, almost all of the primary system within the test vault is covered by four or more inches of thermal insulation, adding an additional barrier for containing NaK from spray discharging into the vault. However, to examine a limiting case, it was assumed that the primary system ruptured at the pump inlet and that all available NaK was sprayed into the vault to cause perfect heat transfer between the NaK and the vault atmosphere. Spraying the NaK, as opposed to dumping it into a pool on the vault floor, would cause rapid transfer of heat to the vault atmosphere and a peak pressure near 4 psig (see Figure 11.2 of the SAR) before thermal equilibrium is established. At equilibrium the pool and spray cases would be identical.

The reactivity status of the system during the course of this accident is favorable even without a reactor scram. As coolant flow is lost, fuel temperature will increase tending to shut down the reactor. As boiling and voiding occur, further reactivity is lost; measurements on S8ER showed a 25¢ reactivity loss from voiding the core and upper and lower plenums. If fuel temperatures in excess of ~1750°F are reached, appreciable H₂ loss will occur through cladding ruptures, irreversibly shutting down the reactor (see XI-B, later in this section). Power or decay heat generation of 1 Mw-sec is required to release 0.93% of core hydrogen.

If a reactor scram is obtained from loss of flow, high outlet temperature, or loss of pump power, it is possible that the core would be kept cooled by natural convection and NaK boiling without suffering cladding failure. However, for the sake of this analysis reactor scram is not assumed.

Following a loss of coolant and core voiding, the evolution of H_2 and attendant cooling of the fuel continues for ~ 4 hr. During this time sufficient energy is generated to release all core H_2 . During this time most of the core H_2 is lost and the reactor reaches a shutdown power of ~ 10 kw. It has been calculated that this amount of decay power, namely 10 kw, can be radiated, in the absence of other cooling, with a core vessel temperature of 1500°F . Such temperature, with only radiative heat processes, would result in peak fuel temperatures of less than 3000°F at 10 kw. Since the fuel does not reach melting temperatures ($\sim 3375^\circ\text{F}$) under the above circumstances, fuel melting will not result from any accidents to the S8DR system.

c. Effect on Containment

To calculate the pressure increase in the secondary containment due to the loss of coolant the following assumptions were made:

- (1) Rupture provides the maximum inventory of NaK for a spill.
- (2) Spray heat transfer to vault atmosphere occurs.
- (3) Heat transfers to the vault atmosphere and liner only. No allowances are made for the vault cooling system or conduction to the vault walls.
- (4) Following the rupture the core becomes vapor-locked and fuel temperatures increase until clad failures occur and hydrogen is released.
- (5) Hydrogen release continues at a rate dependent on decay heat generation.

Based on the above assumptions of spray release of NaK to the secondary containment and H_2 evolution at the maximum rate, the transient pressure in the secondary containment is as shown by Figure 11.2 of the SAR. Initially

the overpressure results from heating of the vault atmosphere. Heat transfer to the liner cools the atmosphere, thus reducing the pressure. For several hours, release of H_2 contributes a small portion of overpressure to the system. A quasi-equilibrium overpressure of ~ 0.5 psi exists following thermal equilibrium. It is assumed that further reduction in pressure is by leakage at a rate of 0.38%/hr, which assumes a constant overpressure of 1 psi throughout the incident. Based on this assumption the containment reaches atmospheric pressure in 19 hr, having released 7.2% of its volume.

d. Fission-Product Release

It is assumed that the fission products are all released at the time of rupture to create the radiological hazard depicted in the SAR. If the fission products were to be released as quickly as the highly mobile hydrogen, it would require ~ 4 hr for their release, and the integral of their release would be substantially diminished.

e. Conservatism of Analysis

The conservative factors in the above analysis are summarized below:

(1) The possibility of a large rupture of the primary system, which consists of a compact array of small, heavily insulated, ductile components operating at low pressure, is exceedingly small. Substantial experience on these systems bears this out.

(2) In most instances of leak location, size, and prompt scram action, excessive core heating and fuel damage will not occur.

(3) The operation of the vault cooling system would decrease the duration of the overpressure to 18 min.

(4) Fission-product release would take a long time in comparison with the duration of the vault overpressure.

B. LOSS OF HYDROGEN FROM FUEL

Question: "Provide additional detailed information describing how the loss of hydrogen from the fuel (or transfer of hydrogen within the fuel if the clad does not rupture) relates to reactivity and causes an irreversible shutdown of the reactor. The discussion should include substantiating data and numerical results of analyses. "

Answer: In the SAR, pages 173 and 174, two inherent safety features of the hydride fuel are discussed. One is the endothermic (> 40 kcal/mole) process of the hydride dissociation, which absorbs energy as the H_2 is being released, and thus limits the maximum fuel temperature. The second feature is the irreversible shutdown associated with loss of the H_2 moderator.

If an accident were to occur which drove a control drum from its least to most reactive position, a total of \$3.90 would be added to the reactor. For the calculated worth of 62¢/% H_2 a total of 6.3% of the core H_2 needs to be removed to negate the \$3.90 reactivity addition. The calculated H_2 worth has been verified by the testing of low- H_2 -content fuel elements in the S8ER critical tests.⁽¹⁸⁾

Diffusion rates of H_2 from the fuel material were developed for the S8ER fuel. Addendum II of the S8ER Safeguards Report⁽¹⁹⁾ states:

"Ordinary diffusion theory^{(20)*} provides the following equation to describe the time-dependent loss of a gas from a cylinder:

$$L = 1 - \sum_{n=1}^{\infty} \frac{4}{\xi_n^2} \exp - \left[\frac{\xi_n^2 Dt}{r_o^2} \right] \quad \dots (1)$$

* Reference number in this addendum.

where

L = fraction of gas lost from cylinder,

n = summation variable,

ξ_n = roots of the equation $J_0(x) = 0$, where $J_0(x)$ is the Bessel function of zero order,

D = diffusion coefficient of the gas in the cylinder,

t = time, and

r_o = radius of the cylinder.

Equation 1 was modified as follows to approximate an infinite series:

$$L = 1 - \sum_{n=1}^{10} \frac{4}{\xi_n^2} \exp \left[\frac{\xi_n^2 Dt}{r_o^2} \right] - L_o \quad \dots (2)$$

where, n , the summation variable, was varied from one through ten to yield a 10th order approximation of the infinite series. L_o is the loss at time zero, as predicted by a 10th order solution to Equation 1:

$$L_o = 1 - \sum_{n=1}^{10} \frac{4}{\xi_n^2} \quad \dots (3)$$

Equation 2 provides a conservative, ie., low, estimate of loss as a function of time since L_o forces the loss to zero at time zero, and the 10th order approximation yields lower rate of change of loss than the more rigorous infinite series. The solution of Equation 2 is shown in Figure 16* where loss is plotted vs the dimensionless time parameter Dt/r_o^2 .

* Figure number in this addendum.

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Figure 16. Gas Loss from Cylinder

Figure 16, derived from fundamental diffusion theory, can now be used to predict the H_2 loss from S8DR at the high-temperature condition where cladding failure occurs. The required H_2 diffusion coefficients were extrapolated from the formulation of Reference 21, which gives

$$D = 599 e^{-34,800/RT}$$

where

D = diffusion coefficient (cm^2/sec)

R = universal gas constant ($\text{cal/mole}^\circ\text{K}$), and

T = temperature ($^\circ\text{K}$).

Table 11 lists the times required to release 6.3% of the core H_2 at temperatures from 1700 to 2600°F.

TABLE 11
TIME REQUIRED TO LOSE 6.3% HYDROGEN
FROM S8DR CORE

Fuel Temperature ($^\circ\text{F}$)	Required Time
1700	3.3 sec
1800	1.7 sec
2000	550 msec
2200	210 msec
2400	92 msec
2600	45 msec

The above analytical predictions of H_2 loss-rate based on experimentally measured diffusion coefficients were substantiated by experimentally measured H_2 loss from zirconium-hydride cylinders.

These pulse-heating experiments were performed at Atomics International⁽²²⁾ on 1/8-in. diam zirconium-uranium hydride cylinders. The cylinders were isothermally heated in an evacuated chamber from room temperature to about 2200°F in less than 1/2 sec. For pulses corresponding to 130-Mw-sec transients the hydride specimens rapidly attained a temperature of about 2200°F, and maintained it by H_2 evolution. The experiment also demonstrated that 100% of the H_2 evolved within 400 msec, approximately 75% of it after the

temperature reached 2200°F. A square-law extrapolation of the experimental results yields 7 sec as the time required to evolve 100% of the H₂ from the larger-diameter S8DR fuel rods. Assuming that the loss rate is exponential with time after 2200°F is reached, the time required for 6.3% of the H₂ to be released from a fuel rod is 85 msec. Table 11 shows 210 msec for the same situation, a very conservative figure. The conservatism may be explained by a phase change from δ to β zirconium-hydride at high temperatures. The diffusion coefficient in β -phase is known to be larger than in δ -phase.⁽²³⁾

Hydrogen transfer within the fuel material of intact elements acts as a power coefficient of about 0.1%/kw with a time constant of several hours at normal operating conditions. Since the time constant has an inverse dependence upon the diffusion coefficient it will decrease very rapidly as fuel temperature increases. However, the H₂ distribution effect is reversible and thus does not have the significance that H₂ loss does in reactor shutdown following a transient.

Therefore, the conclusion is reiterated that the hydride fuel provides inherent safety features to both limit the temperature of the fuel and provide irreversible shutdown reactivity in accidents which result in temperatures high enough to cause cladding damage.

C. WHOLE BODY AND INHALATION DOSE CALCULATIONS

Question: "For the maximum operating power level and operating time, please provide the results of calculations for potential external whole body doses and internal thyroid (via inhalation) doses as a function of downwind distance from Building 059. In making these calculations assume the fractional release of fission products to be those in TID-14844 and similar meteorological conditions. These calculations should include the following release parameters:

1. release location
 - a. at ground level
 - b. from stack height

2. release rate and time

- a. entire release over a 20-30 minute period
- b. entire release at a uniform rate over the evacuation time period of up to two hours.
- c. entire release at the anticipated design leak rate of secondary containment over the course of the accident.

Estimate the maximum leak rate from the building that could occur without exceeding 10CFR100 dose guide values. "

Answer: The internal thyroid dose has been calculated for the requested conditions using the method and fission-product values from TID-14844.⁽²⁴⁾ The atmospheric dispersion functions were obtained from ORO-545,⁽²⁵⁾ which gives similar values to those in Reference 24 except for short distances. At 100 meters the dispersion function for type-F conditions is twice as high as that obtained from Reference 24.

Type-F meteorological conditions were assumed for all except the 100-meter distance for stack release, for which type-A conditions were used because of their more significant effect. A breathing rate of $3.47 \times 10^{-4} \text{ m}^3/\text{sec}$ was assumed for the 20-min and 2-hr release, and a breathing rate of $2.32 \times 10^{-4} \text{ m}^3/\text{sec}$ was assumed for the release at the design leakrate conditions. The design leakrate of 0.38%/hr was used in the analysis. This is the maximum leakrate which occurs during the maximum credible accident as analyzed in Section 11 of the SAR. The thyroid doses are listed here in Table 12.

TABLE 12
 THYROID DOSE FOR GROUND AND STACK RELEASES
 OF FISSION PRODUCTS

Ground			
Distance (meters)	Release (roentgens)		
	20-Minute	Two-Hour	Design
100 (site boundary)	2.90×10^5	2.81×10^5	6.31×10^4
1000	4.23×10^3	4.10×10^3	907
3000 (nearest residence)	725	702	155
Stack			
100 (site boundary)	1335	1294	286
1000	635	615	136
1400 (peak type-F)	730	707	156
3000 (nearest residence)	477	462	102

The above analysis assumed that all fission products released to the test vault were released to the environment, with the exception of the 50%-iodine plate-out in the test vault. Actually only 1/14 of the test vault contents are released to relieve the 1-psi overpressure, as discussed below.

The direct dose was calculated for fission products which escape into the high bay from the shielded test vault. The source terms from TID-14844 were used in the calculation, except 50% of the iodines were assumed to have plated-out in the test vault and thus did not contribute to the direct dose. In addition it was assumed that 0.38% of the fission products leaked into the high bay each hour until the end of 263 hr, at which time all fission products had escaped. The average fission-product concentration in the high bay during this time was used to compute the long-term dose. For stack release the fission products are continuously removed from the high bay according to the model in Section 11

of the SAR. For this case the direct dose is always very low (0.6 r for the worst condition investigated). When ground release occurs the ventilation system is apparently not operating; the direct doses, accordingly, are given in Table 13.

TABLE 13
DIRECT DOSE FROM HIGH BAY WITHOUT VENTILATION
SYSTEM, RELEASE OF ALL FISSION PRODUCTS

Distance (meters)	Release (roentgens)		
	20-Minute	Two-Hour	Design
100	2.73	11.7	47.2
1000	0.0003	0.0013	0.0054
3000	0.00003	0.00015	0.0006

The direct dose from a semi-infinite cloud was calculated according to the method of Burnett,⁽²⁶⁾ again using the source terms from TID-14844. The results are given in Table 14.

These doses (Table 14) also reflect the conservatism of releasing all fission products from the test vault, even though only 1/14 of this leakage will reduce the vault to atmospheric pressure, eliminating a driving force for leakage.

The controlling dose, as determined by 10CFR100 guide lines, is the iodine dose to the thyroid. With the same assumptions as in the above calculations, the design leakrate could increase by 10% to 0.42%/hr before the thyroid dose from a stack release would exceed the 300-r limit.

As pointed out in Sections 11.4 of the SAR and V-B of this addendum, the rapid release of H₂ from the fuel at temperatures near 2000°F, limits the fuel temperature for any credible excursion and thus prevents fuel melting. For this reason it does not appear reasonable to apply the standard fission-product release for fuel melting. The Cegelski data, obtained specifically for SNAP

TABLE 14
CLOUD DOSE FOR GROUND AND STACK RELEASES
OF ALL FISSION PRODUCTS

Ground			
Distance (meters)	Release (roentgens)		
	20-Minute	Two-Hour	Design
100	2340	1757	104
1000	34.1	25.6	1.5
3000	5.8	4.4	0.26
Stack			
100 (type -A)	10.8	8.1	0.48
1000	5.1	3.9	0.23
1400 (peak type -F)	5.9	4.4	0.26
3000 (nearest residence)	3.9	2.9	0.17

reactor fuel, shows releases which are factors of 50 to 250 lower than the standard release.

Likewise, it cannot be assumed that the entire release escapes from the subterranean test vault, since the vault is under atmospheric control and the contents can be pumped to underground holdup tanks. It is more realistic to assume that only the overpressure, 1/14 of the total in the vault, can possibly be released to the high bay and to the outside environment.

D. LITHIUM HYDRIDE SHADOW SHIELD

Question: "Will the lithium hydride shadow shield be present below the reactor during the SNAP 8-DR operation? If so, what effect will its involvement in a loss-of-coolant accident have on the consequences for the design basis accidents?"

Answer: The LiH shadow shield will be present during all phases of test operations. Its involvement in a loss-of-coolant accident will have little effect on the consequences of the accident. During a loss-of-coolant accident in the vacuum vessel the NaK would spill onto the shield confinement vessel, depending on the leak location, and give up some of its heat energy to this structure. This would have some effect in reducing the pressure and temperature of the vacuum vessel during the accident. No allowance for it, however, has been considered necessary in the accident analysis.

The presence of the shadow shield will not contribute adversely to the consequences of the accident because of its enclosure in two separately sealed stainless-steel containers. These containers would not be expected to fail. Even if they did, however, and the NaK impinged on the LiH, no reactions would occur. The temperature of the shield would increase somewhat but would not change the consequences of the accident.

If at 1300°F, all of the available NaK were to come into thermal equilibrium with the LiH shield (which has reached its highest temperature because of reactor operation), the shield temperature would reach 610°F. Since the H₂ dissociation pressure is 13 mm of Hg at 1300°F, this factor would add only negligibly to the consequences of the accident described in Section 11 of the SAR.

APPENDIX ANALYTICAL MODEL

A. PRIMARY HEAT LOSSES, COLD LEG

Question: "Correct Table A.12. "

Answer: Table A.12, page 221 of the SAR lists the expected heat losses for the cold leg of the primary loop. The tabulated values should total 28,969 Btu/hr, not 33,969 as shown. However, missing from the tabulation is the heat loss from the intermediate heat exchanger, calculated to be 5400 Btu/hr. With this addition the total should then be 34,369 Btu/hr. The difference in the corrected total and that used in the original calculations is so small that it has no effect whatsoever on the analysis.

REFERENCES IN ADDENDUM

1. R. E. Alexander, "Radiation and Nuclear Safety at AI," AI-67-3, March 1, 1967
2. R. L. Ashley, "Evaluation of the Atomics International Nuclear Development Field Laboratory as a Location for Reactor Facilities," NAA-SR-7300, May 25, 1962
3. "Population Estimate County of Ventura," Ventura County Bulletin No. 17, April 1965
4. "San Fernando Valley 1960," Security First National Bank Research Department, 1960
5. "Planning Report of the Department of City Planning," City of Los Angeles (January 1, 1961 to June 30, 1962)
6. "Population Estimate," City of Los Angeles Bulletin 1965-1, January 1, 1965
7. Telecon with members of the Los Angeles County Regional Planning Commission
8. D. G. Mason, "SNAP 8 Progress Report, Feb-April 1967," NAA-SR-12414
9. D. G. Mason, "SNAP 8 Progress Report, Feb-April 1966," NAA-SR-11898
10. C. E. Johnson, "SNAP 8 Quarterly Progress Report, May-July 1964," NAA-SR-9992
11. D. S. Cowgill, N. F. Davies, and W. J. Kurzeka, "Irradiation Test of SNAP 8 Candidate Spring Materials," NAA-SR-11519
12. D. G. Mason, "SNAP 8 Progress Report, Oct. 1965-Jan. 1966," NAA-SR-11758
13. D. G. Mason, "SNAP 8 Progress Report, Aug-Oct 1966," NAA-SR-12188
14. D. G. Mason, "SNAP 8 Progress Report, May-July 1967," NAA-SR-12521
15. D. G. Mason, "SNAP Reactor Programs Progress Report, Aug-Oct 1967," NAA-SR-12583
16. D. G. Mason, "SNAP 8 Progress Report, May-July 1966," NAA-SR-12068

17. P.M. Magee, "S8DR Core Thermal and Hydraulic Performance," NAA-SR-12564
18. D.E. Crouter, "SNAP 8 Experimental Reactor (S8ER) Critical Experiments," NAA-SR-9642, September 15, 1964
19. "SNAP 8 Experimental Reactor (S8ER) Final Safeguards Summary Report Addendum II," NAA-SR-6958, Add. II, January 14, 1963
20. W. Jost, "Diffusion," Academic Press, 1952
21. W.M. Albrecht and W.D. Goode, Jr., "The Diffusion of Hydrogen in Zirconium Hydride," BMI-1426, March 3, 1960
22. R.E. Taylor, H.G. Weidberg, and R.J. Albee, "Pulse Heating of Zirconium-Uranium Hydrides," NAA-SR-7398, December 30, 1962
23. W.M. Albrecht and W.D. Goode, Jr., "The Diffusion of Hydrogen in Zirconium," BMI-1373, August 25, 1959
24. J.J. DiNunno, et al, "Calculation of Distance Factors for Power and Test Reactor Sites," TID-14844, March 23, 1962
25. W.F. Hilsmeier and F.A. Gifford, "Graphs for Estimating Atmospheric Dispersion," ORO-545, July, 1962
26. T.J. Burnett, Nucl, Sci Eng, 2, p. 382, 1957.