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ADDITIONAL SAFEGUARDS INFORMATION FOR HALLAM NUCLEAR POWER FACILITY

SUPPLEMENT 4

NAA-SR-5700

ATOMICS INTERNATIONAL

A DIVISION OF NORTH AMERICAN AVIATION, INC.

P.O. Box 309 Canoga Park, California

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CONTRACT: AT(11-1)-GEN-8

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DISCLAIMER

Enclosure
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CERTIFICATION

I certify that the statements made herein are true, complete, and correct to the best of my knowledge and belief, and are made in good faith.

J. J. Flaherty
Executive Vice President
Atomics International Division

Subscribed and sworn to before me this 1st day of December 1961.

George C Pappas
Notary
George C. Pappas
My Commission Expires February 15, 1964

INTRODUCTION

A detailed description of the HNPF and the hazards summary are given in the parent report (NAA-SR-5700). This supplement answers questions of Reference (1) concerning core kinetics and reactivity accidents (Questions II a, b, c, d), primary system confinement cavities (V a, b, c, d, e, f, g), effluent control (VI c, e), fire and/or explosion (VII a, b, c, d), casualty conditions (IX a, b, c, e), and plant testing and operation (XI a, c, d, e). This supplement, together with Supplements 1, 2, and 3, completes the answers to questions of Reference (1).

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(1) Letter to Atomics International (4275AT) from the Division of Licensing and Regulation of the Atomic Energy Commission, August 11, 1961.

Question II(a): Explain why the time constants for the graphite temperature coefficient of reactivity are considerably shorter than previously reported (Report NAA-SR-MEMO-3379). What effect do these new constants have on previous accident analysis with respect to time avilable for corrective measures by operator action?

Answer: A preliminary discussion of the change in moderator time constant was presented in Appendix A of Supplement I, NAA-SR-3379. The graphite time constants are shorter than originally reported in NAA-SR-3379. The moderator can scallops are designed to expand and come into close contact with the graphite at 600°F, eliminating the helium gap which exists at room temperature. The helium gap was assumed to exist in the earlier calculations and its elimination gives a lower thermal resistance. Since the time constant is proportional to thermal resistance, the time constants were reduced. The mathematical model of analysis has been changed from a cylindrical graphite log to a more appropriate equivalent annular configuration because heat transfer analysis indicates that most of the heat flux is through the scallops of the moderator cans instead of across the flats, but this change did not result in a significant change in temperature difference in the graphite or in time constants. The closing of the helium gap in the scallop also decreases the average graphite temperature, thereby decreasing the magnitude of the positive graphite contribution the power coefficient.

The shorter time constant has only slight effect on previous accident analysis. Studies presented in the parent report, based on the shorter graphite time constant, show that protective system action occurs before the graphite contribution becomes significant. For the completely uncontrolled and unprotected power excursion leading to core meltdown, the graphite time constant has a slight influence on the time required to reach melting. The magnitude of this influence depends on the cause of the accident. For example, with continuous control rod withdrawal from full power, both the earlier studies (NAA-SR-3379) and those in the parent report show that sodium boiling commences about one minute after start of rod movement. This time is less than any of the moderator time constants used, so the effect of the reduced graphite time constant would be to shorten the time available for operator action by a few seconds, at most. Therefore operator corrective action will not be seriously impaired by the change in time constant.

The reactor stability analysis presented in the parent report is based upon the shorter graphite time constant. This analysis shows that there is no problem with regard to reactor stability, since the total power coefficient of reactivity acts very slowly. Even when the total power coefficient is positive the reactor power drifts slowly enough to allow for corrective operator action.

Question II(b): Provide a breakdown of all the factors that were considered in estimating temperature coefficients of reactivity and their time constants (e.g., Doppler, expansion effects, spectral shift, streaming, etc.

Answer: There are three individual temperature effects which affect the total reactivity. These are: (a) temperature changes in the fuel, resulting in reactivity changes due to the Doppler effect; (b) changes in sodium temperature, hence density, resulting in changes in parasitic neutron absorption; and (c) changes in graphite moderator temperature, resulting in changes in the unit cell flux distribution and neutron cross sections. The results of these individual effects have been calculated separately over the temperature range $300°F \leq T \leq 1350°F$ for the clean core. In each case, the temperature of the component in question was varied while the other two were assumed to maintain their operating temperatures. Complete reactor calculations were performed for each case in such a manner that spectral density and dimensional variations were included. Thus, not only are the details of variation in absorption included, but also changes in neutron leakage and diffusion properties of the reactor core. Details of these calculations appear in the succeeding paragraphs. Studies of reactor kinetic properties are continuing as data from the HNPF exponential and critical experiments are received. These studies will also incorporate data received from the site when available.

In Table II(b)-1 the clean core partial coefficients at operating temperatures are presented where these operating temperatures are taken as: (1) $T_{fuel} = 890°F$, (2) $T_{graphite} = 850°F$, and (3) $T_{sodium} = 778°F$.

TABLE II(b)-1

HNPF TEMPERATURE COEFFICIENTS OF REACTIVITY

	Table 4.10 of the parent report	Current Values
Fuel	$-1.556 \times 10^{-5}/°F$	$-1.85 \times 10^{-5}/°F$
Graphite	1.33×10^{-5}	1.55×10^{-5}
Sodium	0.556×10^{-5}	0.55×10^{-5}

The coefficients presented above differ from those appearing in the parent report and represent a more recent set of calculations. The coefficients appearing in the parent report, calculated in 1959, are the parameters upon which the kinetic studies are based. As can be seen from inspection of the two sets, there are no major changes. Therefore, the kinetic studies were not re-run since no significant variations in the predictions of reactor kinetic behavior are anticipated. Temperature coefficients change somewhat with fuel burnup, as shown in the parent report. This is due to the change in isotopic ratios within the core, and coefficients in the irradiated core are calculated by the same methods presented here.

A. DETERMINATION OF FUEL TEMPERATURE COEFFICIENT

Changes in reactivity due to fuel temperature variations are caused primarily by variation of epithermal resonance neutron absorption in U^{238}. This variation of resonance absorption with temperature (The Doppler effect) has been measured extensively, the Doppler coefficient used in this study being $1.5 \times 10^{-4}/°C$ ($0.83 \times 10^{-4}/°F$). In evaluating the fuel coefficient for HNPF, the fuel temperature was varied between 300°F and 1350°F. For each case, epithermal neutron constants were calculated with the FORM Code and a reactor reactivity calculation was performed with the AIM-5 Code to determine the change in reactivity. Dimensional changes of the fuel were not considered since they will not have an appreciable effect on reactivity. The results of these calculations are shown in Table II(6)-2.

TABLE II(b)-2

REACTIVITY CALCULATIONS

T (°F)	k_{eff}	Δk	$\Delta k/\Delta T$ ($\times 10^5/°F$)	Average At Operating $\Delta k/\Delta T$
300	1.10016			
		0.00558	-1.86	
600	1.09458			
				$-1.850 \times 10^{-5}/°F$
		0.00534	-1.84	
890	1.08924			
		0.00851	-1.850	
1350	1.08073			

The average over this temperature range is then $-1.850 \times 10^{-5}/°F$.

B. DETERMINATION OF THE COOLANT COEFFICIENT

The average coolant coefficient was evaluated by varying the average temperature of the coolant between 300°F and 1350°F, taking into account the correponding changes in coolant density while maintaining the fuel and moderator temperatures at their operating values. The core reactivity was computed for each temperature by re-evaluating the epithermal nuclear constants with FORM, the thermal cell flux distribution with the S_n Code, and the reactor eigenvalues with AIM-5. In this manner the change in neutron absorption due to the changes in coolant density were accounted for.

The coolant density at operating temperatures is 0.845 gms/cc. The volumetric coefficient of expansion for sodium is $2.66 \times 10^{-4}/°C$ or $1.48 \times 10^{-4}/°F$.

Table II(b)-3 contains the coolant temperatures and corresponding k_{eff}'s.

TABLE II(b)-3
COOLANT TEMPERATURE AND k_{eff}

T (°F)	k_{eff}	Δk	$\Delta k/\Delta T$ (x 10^5/°F)	Average At Operating $\Delta k/\Delta T$
300	1.08449			
		0.00210	+ 0.700	
600	1.08659			$+ 0.55 \times 10^{-5}/°F$
		0.00165	+ 0.568	
890	1.08824			
		0.00227	+ 0.493	
1350	1.09051			

The average coolant temperature coefficient is then $+ 0.55 \times 10^{-5}/°F$.

C. DETERMINATION OF MODERATOR COEFFICIENT

Changes in moderator temperature affect reactivity by changing the effective thermal neutron temperature, which in turn varies the thermal flux cell distribution and the relative thermal neutron reaction cross sections of the cell materials. For the clean critical core, the thermal neutron flux distributions and relative reaction rates for the core materials were evaluated for moderator temperatures between 480°F and 1200°F. Thermal cell flux calculations were performed with the S_n Code and over-all reactivities were determined with the AIM-5 Code; results are presented in Table II(b)-4. The moderator coefficient is seen to vary as a function of the moderator temperature and at operating temperatures is equal to $1.55 \times 10^{-5}/°F$.

TABLE II(b)-4

THERMAL CELL FLUX CALCULATIONS

T (°F)	k_{eff}	Δk	$\Delta k/\Delta T$ (x 10^5/°F)	Average At Operating $\Delta k/\Delta T$
480	1.08132			
		0.00405	2.25	
660	1.08537			+ 1.55×10^{-5}/°F
		0.00387	2.15	
840	1.08924			
		0.00271	1.51	
1020	1.09195			
		0.00238	1.32	
1200	1.09433			

D. FACTORS INFLUENCING TIME CONSTANTS

Time constants for the temperature coefficients (Page 4-31 of the parent report) are based on a heat transfer analysis of the component concerned. The temperature response of a component to a step change in reactor power is made up of the steady-state temperature corresponding to the new power plus a transient term which decays exponentially: The time constant of the transient temperature behavior is mathematically analogous to the mean life of a radioactive source.

Heat transfer from fuel to sodium coolant is a function of the fuel rod geometry and physical properties. Specific factors include thermal resistances in fuel, bond, cladding, and sodium film outside the rod; and heat capacities of each component of a fuel rod.

Factors which enter into calculation of the moderator time constant are the thermal resistance between moderator and coolant and the heat capacity of the moderator. The sodium time constant depends upon the heat capacity of the sodium and the rate at which heat is removed from a given region by sodium flow. It is assumed that due to flow mixing and good thermal conductivity, all sodium at a given height in a process channel is isothermal.

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Question II(c): What is the minimum estimated reactivity insertion that would lead to formation of a shock wave in the reactor? What consideration has been given to the ability of the system to absorb shock wave energy?

Answer: Minimum estimated step reactivity insertion to form in-core shock waves is 6.8% $\Delta k/k$.

The actual formation of shock waves is not considered possible for HNPF. Only qualitative consideration has been given to the ability of the system to absorb shock wave energy.

In order to generate shock waves in a reactor, there must exist a mechanism for very rapid release of a large amount of energy. The only means of such an energy release in the HNPF reactor is rapid vaporization of a significant amount of fuel, with the resulting expansion rapid enough to create supersonic flow in surrounding fluids. The shock waves considered in HNPF are those in the fuel itself and those in various mixtures of fluids within a process tube.

For the current studies it is assumed that the boiling point of U - 10 Mo is 7000°F, independent of local pressure. If the fuel heat generation rate were increasing slowly during a transient which resulted in fuel melting (U-Mo melting point ~2100°F), the 0.010-in. stainless steel fuel cladding would be ruptured within seconds allowing the molten fuel to fall to a subcritical geometry (section 4.2.2 of the parent report). Thus, in order to cause vaporization of fuel, the reactor power must rise so rapidly that fuel melting does not result in fuel relocation. This requires a large step insertion of positive reactivity. The Doppler fuel temperature coefficient is calculated to be -1.6×10^{-5} $(\Delta k/k)/°F$ at normal operating temperatures. For an assumed $1/\sqrt{T}$ variation of the Doppler coefficient,* the amount of reactivity to override this is 5.8% $\Delta k/k$ in going from full power fuel temperatures to fuel vaporization. An additional 0.7% $\Delta k/k$ is required to overcome the delayed neutron fraction to keep the reactor prompt critical so the Doppler plus β "override" is 6.5% $\Delta k/k$.

The basic assumptions made in studying transients which could lead to in-core shock waves include:

1) Initial conditions of reactor at nominal full power (240 Mwt) with fuel at 900°F average temperature.

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*Nuclear Sci. and Eng., 8, 497-506 (1960).

2) Any uncontrolled transient will be terminated 0.5 sec or sooner by the dropping of molten fuel out of the center of the core. Fuel will fall a distance of 4 ft under acceleration of gravity or 2 ft under one-half gravity acceleration in 0.5 sec.

It is reasonable to assume that when the energy density in the fuel is equal to its Young's Modulus, the fuel will physically disrupt itself. For uncooled U - 10 Mo fuel in HNPF, this energy density will be attained in 0.5 sec if average power is $\sim$350 times full power. The step reactivity required to generate a transient with such a power level is 0.4% $\Delta k/k$ above the "override" for Doppler plus β, or a total of 6.9% $\Delta k/k$.

In order to maintain a shock wave, the energy must be generated faster than it can be removed by expansion and/or heat transfer. This requires a reactor period on the order of $\gamma = L/C$ or shorter,* where L is a characteristic length and C is the speed of sound in the medium. A characteristic length of a fuel rod would be one-half the total rod length, or $\sim$6.5 ft. For a shock wave in metallic fuel, a period $\leq$ 0.6 ms is required. This corresponds to a step reactivity insertion of "override" plus 47%. If this same criterion is applied to liquid-vapor systems within a process tube, lower reactivity insertions are calculated. The characteristic length is taken as 10 ft or approximately one-half the total process channel length. If the process tube is filled with uranium vapor, with a calculated C = 1630 ft/sec, a reactor period of 6 ms would be required to meet the $\gamma = L/C$ criterion. The step reactivity insertion to create this condition is "override" plus 4.6%, or a total of 11.1% $\Delta k/k$. The minimum speed of sound within the process channel is $\sim$112 ft/sec in a two-phase mixture of sodium liquid and vapor. The reactor period equal to L/C in this fluid is 89 ms, corresponding to a step reactivity of "override" plus 0.3%, or a total of 6.8% $\Delta k/k$.

Three conceivable means of adding reactivity to the core are by movement of fuel, movement of control rods, and removal of core sodium. Fuel movement is restricted by the moderator elements and fuel process tubes which dictate that any fuel movement must be vertical. The physical connection between

*Bump and Seidensticker, "Analysis of Temperatures and Expansions Resulting from Exponential Power Changes in a Reactor," Nuclear Sci. and Eng., 4: 44-64(1958).

the fuel bundle and its shield plug, and the fact that the latter is securely held in the loading face shield, preclude upward motion of the fuel. Gross downward motion of fuel, due to either melting or physical disconnecting of the element from its hanger tube, will decrease reactivity by increasing neutron leakage and moving fuel to a position of lower statistical weight.

Normal control rod travel is at reactivity rates $\leq 0.03\%\ (\Delta k/k)/\mathrm{sec}$. These are much too slow to initiate shock wave transients. The poison columns move in a gas atmosphere within thimbles, and are connected to drives and actuators outside the reactor.

Poison columns are suspended from above by pull rods. The poison cannot fall more than a few inches below its full-down position even if pull rods shear because the thimble will catch and retain it.

Any force trying to eject a control rod upward from the core would also have to lift the drive and actuator, breaking the bolts connecting the actuator to the loading face shield and to the control rod support structure. No forces of this magnitude exist within the reactor under all transients studied. Thus, there appears to be no way in which control rod movement can introduce large steps of reactivity.

Since the sodium coolant acts as a neutron absorbing poison, its removal from the core will add reactivity. The worth of all core sodium is estimated to be $5\%\ \Delta k/k$. During uncontrolled transients, the fuel coolant inside the process tubes could be expelled by the sodium vapor formed at regions of local boiling. Approximately one third of the core sodium is external to the process tubes as moderator coolant, where it is exposed to considerably higher flux than the fuel coolant. The short fuel meltdown transient would be terminated before moderator coolant could be expelled from the core. Therefore, the reactivity addition due to rapid sodium removal from the core is limited to the fuel coolant worth, estimated to be $\leq 2.5\%\ \Delta k/k$.

Assuming shock waves are generated within the core by some means, the effect on reactor containment can be considered. The free-surface sodium pool above the core will act as a surge volume to decrease maximum pressure generated.* In any event, the 4-ft gas space (1150 ft^3) between the sodium and the

*"Chemical Considerations in the Sodium-Cooled, D_2O Moderated Reactor (SDR)," NDA 84-6, 4-30-60.

loading face shield will prevent transmittal of the shock wave to the shield. The shock wave, if large enough, could possibly cause rupture in the lower plenum region of the core tank or somewhere in the primary piping. If the tank should rupture, the contents of the primary system would be contained in the reactor outer vessel and/or the reactor cavity. If the primary piping should rupture, the sodium would be confined by the guard pipes and/or the primary pipe tunnel or the IHX cells. Thus, this type of failure would not result in release of core fission products to the high bay area.

It is concluded that the conceivable means of reactivity insertions into the core limit the uncontrolled reactor transients to fuel meltdown without shock wave formation. There is no conceivable way of adding enough reactivity to generate shock waves. However, assuming shock waves were to occur, they would probably not violate the primary system containment because of the pressure relief at the free surface at the top of the sodium pool above the core. Shock wave damage to the primary system would not, in any event, result in release of core fission products and radiation hazard to personnel.

Question II(d): Could pressurization of the lower reactor plenum, such as would follow a large transient, result in upward displacement of the grid plate?

Answer: The pressurization required to displace the grid plate upwards ($\sim$300 psi) is greatly in excess of that which is estimated to follow even a transient larger than reasonably expected.

Following the shock wave studies [Question II(c)], the effect of boiling without shock waves was considered. If a large number of channels become choked with vapor at the same time, the lower plenum pressure would increase to an upper limit near the pump static head, or $\sim$80 psi.

As shown in Figure 2.1 of the parent report, the grid plate, and the moderator elements resting on the grid plate, are supported by an array of 141 solid stainless steel staybolt support columns, each 2 in. diameter by 44-5/8 in. long. These columns are welded to the bottom head. A pair of screwed collars at the top of the column support the grid plate and restrain it from upward movement. The screwed collars are tack welded after assembly to prevent loosening. Thus, the grid plate and lower head form a stayed structure.

A conservative calculation, including effects of stress concentration in the threads, indicates that a pressure of approximately 300 psi is required to deform and possibly fail the joint between the support column and collar at operating temperature of 600°F. The support column itself, and the bottom weld are as strong or stronger.

Since the pressure calculated, resulting from a large transient with fuel vaporization in several of the channels, is on the order of 80 psi, no failure of the staybolts is expected, and thus no upward movement of the grid plate will occur. Deflection of the 1-1/2-in. thick grid plate between staybolts is negligible (on the order of a few hundredths of an inch).

It should also be noted that pressures of the magnitudes discussed above, acting on the 4-in.-diameter lower end of a core component (such as a fuel element) will not exert sufficient upward force to break the shield plug hold-down lugs (Reference Par. 4.1.1.2 of the parent report). Control rod actuator hold-down bolts (on one rod) are estimated to hold down 30,000-lb force before failure, equivalent to upward pressure of 2400 psi against the bottom of a control rod thimble.

It is, therefore, concluded that the pressures developed in the lower reactor plenum will not displace the grid plate, the fuel, or control rods. The reactor confinement capability will not be breached.

Question V(a): What is the capability to detect sodium leaks in the primary system tunnels and cells should the leaks be undetected by the installed leak detectors? What minimum primary sodium leak is detectable? How can leaks of secondary sodium in the IHX cells be detected?

Answer: If leaks in the primary sodium system tunnels and cells are not detected by leak detectors installed in the piping pre-heat furnaces surrounding the piping and in valve bonnets, they would be revealed indirectly by other means. These indirect methods are as follows:

1) Airborne radiation from the oxide of Na^{24} would be sensed by the radiation detectors installed on the nitrogen duct work.

2) Sodium oxide smoke in the IHX cells from either primary or secondary sodium would be sensed by smoke detectors to be installed in the nitrogen duct work.

3) If large quantities of sodium are lost and remain undetected, the resultant lowering of sodium volumes in the system would be detected by liquid level instruments.

Leaks as small as a few drops of sodium would be detected if the leakage is directly on a leak detector. However, it would be reasonable to expect that something on the order of one to eight ounces of sodium would be required to leak out at an arbitrary point in the piping and run down inside of the pipe reflectors to the nearest leak detector.

The radiation detectors installed on the nitrogen ducts will detect about 10^{-10} gm primary Na/cm^3 N_2 at equilibrium Na activity, equivalent to 0.1 gm of airborne sodium oxide. This is equivalent to combining 2×10^{-6} percent of the atmosphere as oxygen with sodium. Since the oxygen content of the cells is expected to vary between 0.1 and 1 percent, adequate oxygen will be present to oxidize a small amount of sodium. Spills into the cell atmosphere of a few ounces of primary sodium will, therefore, be detected by the monitors. The location of the leak will determine the amount of leakage before sodium spills into the cell and becomes airborne. If the leak is into normal pipe insulation, several pounds will probably leak into the insulation and freeze before a small amount leaks into the nitrogen atmosphere.

Smoke detectors are being installed in the nitrogen cooler ducts. These detectors will also detect leaks involving a very small amount of sodium in the atmosphere. The smoke detectors will detect about 2×10^{-3} percent of the atmosphere as sodium oxide. Adequate oxygen is expected to be present in the cells for this means of detection, thereby providing backup to the installed leak detectors for secondary sodium and secondary backup for primary sodium.

The third suggested indirect means will require the loss of $\sim 75 \text{ f}^3$ or 3750 lb of primary sodium before the level drops in the reactor vessel sufficiently to expect the operator to observe a change in level ($\sim$3-in. drop).

Leak detectors in the IHX ovens will detect secondary as well as primary sodium leakage from the intermediate heat exchangers. The secondary backup in the secondary system is loss of sodium level in an expansion tank. About 8 ft^3 or 400 lb would probably be noted by the operator, and between 1,000 and 3,000 lb, depending on operating temperature, would cause an alarm annunciation.

Question V(b): Provide a list showing locations of all sodium leak detectors installed in the plant.

Answer: Attached is a list of sodium leak detectors, by number and location. All are in the primary system, except numbers BD-114 through 129, which are on the shell sides of the intermediate heat exchangers. The leak detectors are described on page 2-51 of the parent report.

SODIUM LEAK DETECTORS

Leak Detector Number	Location
BD-1	Reactor Containment Tank Inside at Bottom
BD-2	Reactor Containment Tank Inside at Bottom
BD-3	Reactor Cavity Liner
BD-4	Reactor Cavity Liner
BD-5	Reactor Containment Tank Inside
BD-6	Reactor Containment Tank Inside
BD-7	Reactor Containment Tank Inside
BD-8	Reactor Containment Tank Inside
BD-101	Na Throttle Valve Loop No. 1
BD-102	Na Throttle Valve Loop No. 2
BD-103	Na Throttle Valve Loop No. 3
BD-104	Na Block Valve Loop No. 1
BD-105	Na Block Valve Loop No. 2
BD-106	Na Block Valve Loop No. 3
BD-107	Moderator Coolant Block Valve
BD-108	Moderator Coolant Control Valve
BD-109	Na Pump Loop No. 1 Balancing Leg
BD-110	Na Pump Loop No. 2 Balancing Leg
BD-111	Na Pump Loop No. 3 Balancing Leg
BD-114	IHX 1A Expansion Bellows
BD-115	IHX 1B Expansion Bellows
BD-116	IHX 1A Outer Jacket
BD-117	IHX 1B Outer Jacket
BD-120	IHX 2A Expansion Bellows
BD-121	IHX 2B Expansion Bellows

Leak Detector Number	Location
BD-122	IHX 2A Outer Jacket
BD-123	IHX 2B Outer Jacket
BD-126	IHX 3A Expansion Bellows
BD-127	IHX 3B Expansion Bellows
BD-128	IHX 3A Outer Jacket
BD-129	IHX 3B Outer Jacket
BD-130	Na Fill Tank No. 1
BD-131	Na Fill Tank No. 2
BD-132	Na Fill Tank No. 3
BD-133	Na Fill Tank No. 4
BD-134	Na Fill Tank No. 5
BD-135	Na Service Drain Tank
BD-136	Containment Drain Tank
BD-137	Na Plugging Meter No. 1 Assembly
BD-138	Na Plugging Meter No. 2 Assembly
BD-139	Na Cold Trap No. 1
BD-140	Na Cold Trap No. 2
BD-141	Pri Na Pump No. 1
BD-142	Pri Na Pump No. 2
BD-143	Pri Na Pump No. 3
BD-151	Service Pump No. 1 Loop Side Bellows Seal Valve
BD-152	Service Pump No. 1 Fill Tank Side Bellows Seal Valve
BD-153	Service Pump No. 2 Loop Side Bellows Seal Valve
BD-154	Service Pump No. 2 Fill Tank Side Bellows Seal Valve
BD-155	Na Plugging Meter Assembly No. 1 Inlet Bellows Seal Valve
BD-156	Na Plugging Meter Assembly No. 1 Outlet Bellows Seal Valve
BD-157	Na Plugging Meter Assesmbly No. 2 Inlet Bellows Seal Valve

SODIUM LEAK DETECTORS (Continued)

Leak Detector Number	Location
BD-158	Na Plugging Meter Assembly No. 2 Outlet Bellows Seal Valve
BD-159	Na Cold Trap No. 1 Inlet Bellows Seal Valve
BD-160	Na Cold Trap No. 1 Outlet Bellows Seal Valve
BD-161	Na Cold Trap No. 2 Inlet Bellows Seal Valve
BD-162	Na Cold Trap No. 2 Outlet Bellows Seal Valve
BD-163	Na Fill and Drain Cross Over Bellows Seal Valve
BD-164	Fill Tank Supply Header Bellows Seal Valve
BD-165	Fill Tank Bypass Header Bellows Seal Valve
BD-166	Na Fill Tank Vent Line Drain Header Bellows Seal Valve
BD-167	Na Service Drain Tank Outlet Bellows Seal Valve
BD-168	Na Service Piping Drain Bellows Seal Valve
BD-169	Moderator Coolant Line Cont. Drain Tank Drain Bellows Seal Valve
BD-170	Na Irradiation Facility Supply Bellows Seal Valve
BD-171	Na Irradiation Facility Return Bellows Seal Valve
BD-172	Na Irradiation Facility Supply Drain Bellows Seal Valve
BD-173	Na Irradiation Facility Return Drain Bellows Seal Valve
BD-174	Na Carbon Trap Supply Bellows Seal Valve
BD-175	Na Carbon Trap Return Bellows Seal Valve
BD-176	Na Supply Header Isolation Bellows Seal Valve

SODIUM LEAK DETECTORS (Continued)

Leak Detector Number	Location
BD-177	Na Return Header Isolation Bellows Seal Valve
BD-178	Na Fill Tanks Fill Line Bellows Seal Valve
BD-179	Na Fill Tank No. 1 Inlet Bellows Seal Valve
BD-180	Na Fill Tank No. 1 Outlet Bellows Seal Valve
BD-181	Na Fill Tank No. 1 Vent Bellows Seal Valve
BD-182	Na Fill Tank No. 2 Inlet Bellows Seal Valve
BD-183	Na Fill Tank No. 2 Outlet Bellows Seal Valve
BD-184	Na Fill Tank No. 2 Vent Bellows Seal Valve
BD-185	Na Fill Tank No. 3 Inlet Bellows Seal Valve
BD-186	Na Fill Tank No. 3 Outlet Bellows Seal Valve
BD-187	Na Fill Tank No. 3 Vent Bellows Seal Valve
BD-188	Na Fill Tank No. 4 Inlet Bellows Seal Valve
BD-189	Na Fill Tank No. 4 Outlet Bellows Seal Valve
BD-190	Na Fill Tank No. 4 Vent Bellows Seal Valve
BD-191	Na Fill Tank No. 5 Inlet Bellows Seal Valve
BD-192	Na Fill Tank No. 5 Outlet Bellows Seal Valve
BD-193	Na Fill Tank No. 5 Vent Bellows Seal Valve
BD-194	Containment Drain Tank Inlet Bellows Seal Valve
BD-195	Reactor Vent Bellows Seal Valve
BD-196	Reactor Na Supply Vent Drain Bellows Seal Valve

SODIUM LEAK DETECTORS (Continued)

Leak Detector Number	Location
BD-197	Pri Na Pump No. 1 Vent Bellows Seal Valve
BD-198	Pri Na Pump No. 2 Vent Bellows Seal Valve
BD-199	Pri Na Pump No. 3 Vent Bellows Seal Valve
BD-200	Pri Na Cold Trap No. 1 Flange Connection
BD-201	Pri Na Cold Trap No. 2 Flange Connection

Question V(c): What maximum overpressure can exist in the tunnels and cells without causing rupture to the associated nitrogen cooling ducts and coolers?

Answer: The nitrogen cooling duct system and cooling units will be pressure tested at 6 psig. This is the maximum pressure that can be contained within the primary tunnel and IHX cells since the concrete shield plugs will lift at about this pressure (see sections 4. 1. 1. 5 and 4. 1. 1. 6 of the parent report). The operating pressure in the tunnel and cells varies between 1 and 6 in. water. Relief valves open at 2 psig and rupture discs between cells and the tunnel are installed for flow in both directions at 5 psig.

The cell and tunnel pressures resulting from a large spill of hot sodium have been calculated as presented in Question V(d). If all pressure control and relief valves were to fail, the IHX cell pressure or the pipe tunnel pressure would rise to about 1.0 psi, depending on location of the spill. This is well within the tested capability of the nitrogen ducting.

Question V(d): What is the precision and accuracy of the leak test proposed for these enclosures? Include a brief description of the test procedures and equipment.

Answer: Primary sodium is enclosed in the reactor structure area by a helium filled reactor cavity and in primary coolant areas by nitrogen filled cells. The reactor cavity will not be accessible following power operation, and consequently has been designed and constructed to lower leak rate criteria than the other cells. The answer to the question is subdivided into two corresponding parts and organized within the parts by a description of the test, precision of test equipment, accuracy of the test and criteria for accepting the cell. Since criteria for judging the safeguards aspects of the primary coolant cells are new, a separate discussion is attached following the statement of criteria.

1. Reactor Cavity

a. Description

A differential pressure instrument between the reactor cavity and a temperature compensating chamber which is built into the cavity is used to measure cavity leakage. The test arrangement, including instrumentation, is shown schematically in Figure V(d)-1. The test consists of pressurizing the cavity and chamber with helium to 2-1/2 psig, holding the pressure until an equilibrium temperature between cavity and chamber is reached, connecting a sensitive differential manometer between the cavity and chamber, and recording manometer readings at 30-min intervals for a 3-day period. Temperature and pressure readings are observed to correct the pressure difference data if required.

The compensating chamber is a 3-in. diameter pipe, 24-ft long, located in the north vertical pipe chase so that the bottom of the compensating chamber is at 1394 ft-7 in. elevation. This chamber was inspected for leaks by helium mass spectrometer means before installation. The chamber is connected by tubing to a U-tube manometer filled with 0.79 specific gravity kerosene and 0.83 specific gravity alcohol and water. The other leg of the manometer is connected to the cavity by another tube permanently installed in the cavity.

Leakage from the cavity is determined from the following basic equation and the accuracy of this determination depends on the accuracy of the various terms involved in the equation.

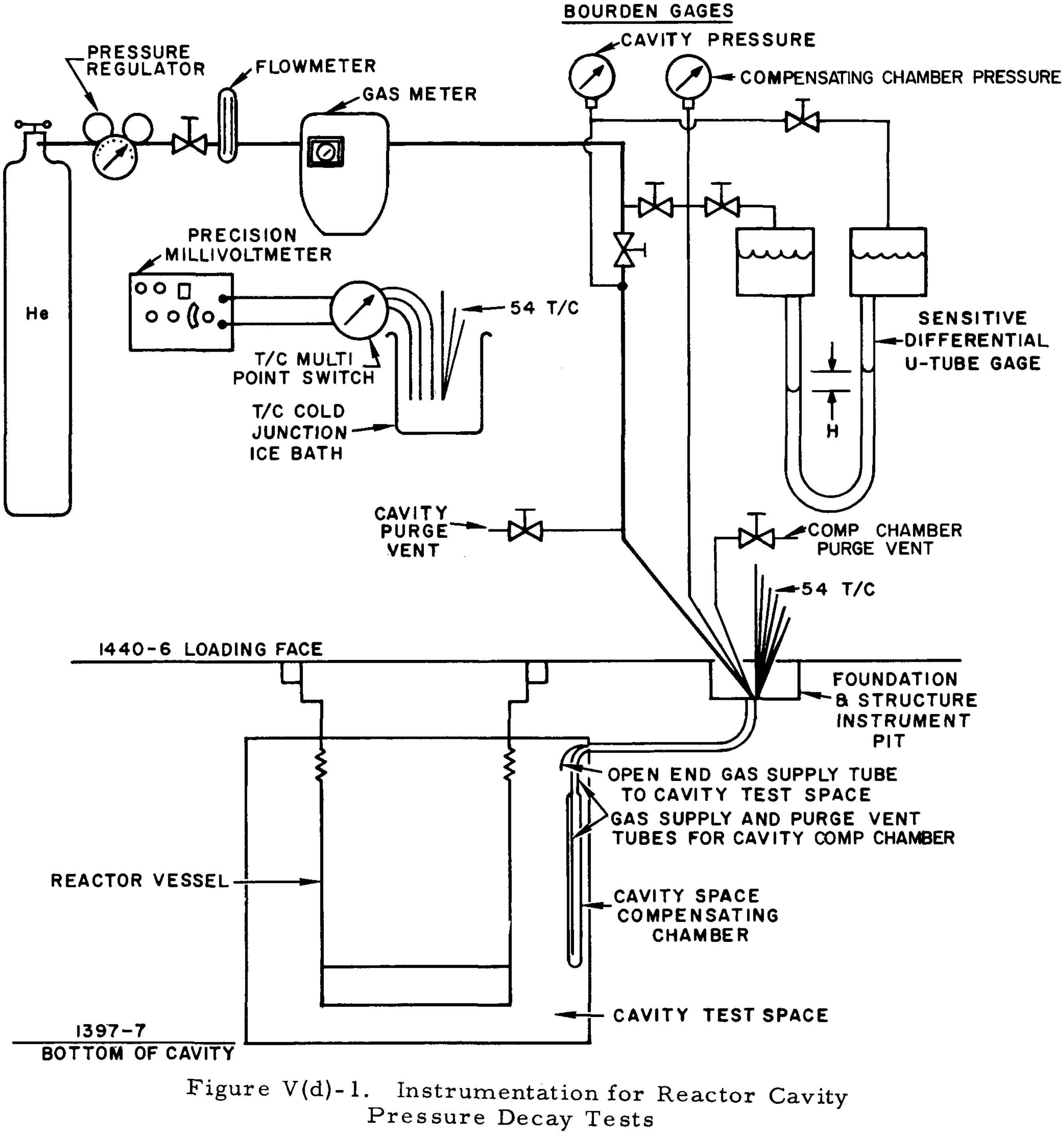

Figure V(d)-1. Instrumentation for Reactor Cavity Pressure Decay Tests

$$\Delta V = V \frac{T_s}{P_s} \frac{P_1}{T_1} \left[1 - \frac{1 + \alpha}{1 + \beta} + \frac{1}{1 + \beta} \frac{\delta}{P_1} \right]$$

where

ΔV = leakage in scf (scf of He)

V = volume of the cavity (ft^3)

T_s = standard temperature (°R)

P_s = standard pressure (psia)

P_1 = cavity initial absolute pressure (psia)

T_1 = cavity initial absolute average temperature (°R)

δ = pressure difference between the compensating chamber within the cavity and the cavity pressure at the end of the time increment for which the leakage occurs (psi)

$$\alpha = \frac{\Delta t}{t_1} = \frac{\text{positive temperature change of gas in compensating chamber (°R)}}{\text{initial absolute average temperature of chamber gas (°R)}}$$

$$\beta = \frac{\Delta T}{T_1} = \frac{\text{positive temperature change of gas in cavity (°R)}}{\text{initial absolute average temperature of cavity gas (°R)}}$$

b. Precision and Accuracy

The pressure difference measurement, δ, is made with the sensitive differential manometer, using two fluids of slightly different specific gravities. The specific gravities of the fluids are certified. The position of the fluid level in each of the two legs of the manometer can be read to within 1/32 of an inch so the differential of position can be read to 1/16 of an inch. This is equivalent to 0.000031 psi and 0.03 scf in terms of an apparent indication of leakage.

The volume of the gas in the cavity, V, is not known precisely. However, a detailed geometric study of the cavity has given a calculated gas colume of 13,900 ft^3. It is estimated that this could be in error by ±3%. This uncertainty reflects directly in a 3% error in leakage.

Thermocouples in various locations in the cavity and compensating chamber indicate temperatures. The absolute temperature of these thermocouples can be measured to about ±5°R or ~1% accuracy. The average cavity temperature, T_1, in the equation for leakage, which is obtained by properly weighting the temperature indicated by each thermocouple according to associated mass of gas, could be in error by as much as 4°R or 0.75% due to temperature gradients observed in the cavity. The total probable error in T_1 is about 1.75% which is also reflected directly in a 1.75% error in leakage.

The cavity pressure is measured with a sensitive Bourdon gage with an accuracy of 0.25% of 10 psig full scale. Thus, with a pressure of 2.5 psig, the accuracy in measuring absolute pressure, P_1, in the equation for leakage is about 0.025/16.6 = 0.15% with a corresponding insignificant error in leakage.

Although the test will be made during a period when there are no apparent thermal distrubances within the cavity, the temperature of the cavity and compensating chamber gases might change within our ability to detect temperature change. With the sensitive potentiometer used it is possible to indicate temperature changes of a particular thermocouple to 0.1°R. Assuming that the error in average change in temperature of the cavity (ΔT) and compensating chamber (Δt) are both 0.1°R based on the readings of all thermocouples, the error in calculated leakage is about 5.93 scf. Accounting for the errors due to V, P_1, and T_1 discussed above gives a total error of about 6.22 scf at zero leakage and increases slightly with leakage to 6.40 scf at 3.5 ft^3/day leakage and 6.82 scf at 12 ft^3/day. (See criteria below.)

c. Criteria

The parent report states that the design criteria leak rate is 1.37 ft^3/day at expected operating conditions. This quantity is equivalent to about 1.05 ft^3/day at standard temperature and pressure (STP). The equivalent leak rate at test conditions depends on shape of the leak path. The potential leak sources are the welds, eleven epoxy seals around condutis, and seventeen pipe hangers. For the minute leaks missed by helium mass spectrometer testing, leaks through the welds and epoxy seals are expected to be through long paths as compared to the diameter of opening. For this type leak the ratio of leakage at test (69 in. H_2O) to operating (6 in. H_2O) pressure will vary directly as the first power of the ratio of these pressures, or as 69/6 = 11.5. The possible leaks through the hangers may vary as the square root of ratio of gage pressures, or as $\sqrt{69/6}$ = 3.4, since a crack in a bellows could behave like an orifice opening. The equivalent leak at test pressure is therefore 1.05 x 3.4 = 3.5 scf/day for orifice type leaks or 1.05 x 11.5 = 12 scf/day for channel type leaks.

The acceptable criteria for leakage from the reactor cavity at test pressure, assuming the square root (orifice type) leak, is: Leakage = (3.5 scf/day) by (days of test) - 6.40 scf. At 3 days, this is 4.1 scf as illustrated on Figure V(d)-2. If measured leakage is below 4.1 scf, the cavity is acceptable and the test will be terminated.

The linear (channel type) leak behavior would allow a permissible leakage of: Leakage = (12 scf/day) by (days of test) - 6.82 scf. At 3 days, this is 29.2 scf. Hence, leakage would be less than the criteria leak rate (1.37 ft^3/day) at operating conditions if 4.1 scf or less is measured and below this criteria leak rate if 29.2 scf or less is measured and it is known that leakage varies as the fir first power of the pressure difference ratio. The location of probable "square root type" leaks are accessible. If the measured leak rate is between 4.1 scf and 29.2 scf, helium mass spectrometer testing will be used to recheck the sources of potential leakage. Any leaks found by this means will be repaired. The leakage test will be repeated and the limit of a measured 29.2 scf for the 3-day test will be used for the test criteria. In the unlikely event of a measured leakage greater than 29.2 scf, the leaks will be assumed to be somewhere in the inaccessible welding. In this case, an operational limit will be placed on reactor cover gas activity at

$$\left[630 \frac{\mu c}{cc}\right] \times \left[\frac{29.2\ ft^3/3\ days}{\text{Measured}\ ft^3/3\ days}\right] \quad \text{in test conditions.}$$

Activity of this level will require purging of the cover gas.

2. Primary Coolant Cells

a. Description

The primary system enclosures are separately enclosed cells, each of which is provided with two water-cooled nitrogen blower units, one normally operating and one on standby. The standby unit is controlled to start up automatically when the outlet temperature from the cell approaches 150°F. Each cell is pressure regulated (from 1 to 6 in. of water) and is connected to a relief system (to the radioactive vent) which operates at 2.0 psig. Further backup is provided by rupture discs between cells that rupture at about 5 psig.

Prior to testing individual cells nitrogen or air will be circulated at 2.5 psig through the enclosure until thermal equilibrium is reached. Then the

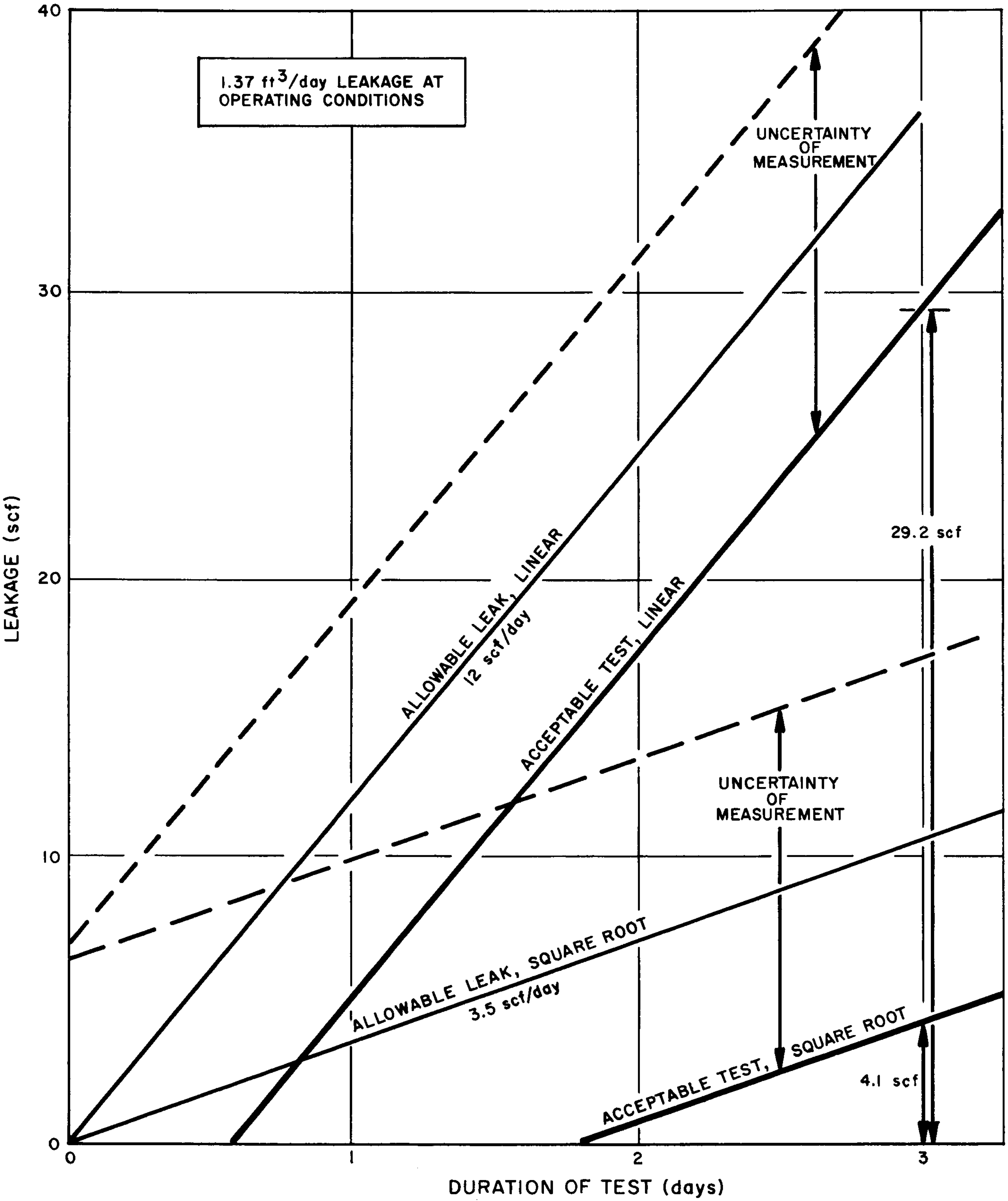

Figure V(d)-2. Reactor Cavity Leakage

nitrogen or air supply will be isolated from the cell. Cell pressure, barometric pressure, and all temperatures will be recorded hourly for a 1-day period.* Special test equipment includes a dual pen pressure recorder to record static pressure and differential pressure, a portable potentiometer calibrated for iron-constantan thermocouples with a range of zero to 400°F, and a mercury barometer.

b. Precision and Accuracy

Leakage from a cell is calculated by

$$\text{Percent leak rate} = \left\{\left[\frac{(P_1 + P_{b1}) - (P_2 + P_{b2})}{\text{Avg. barometric pressure} + P_{\text{cell avg.}}}\right] + \left[\frac{T_2 - T_1}{T_{\text{cell avg.}}}\right]\right\} \times 100$$

P_1 = test pressure = 2-1/2 psig = 69.4 in. H_2O

P_2 = gage pressure in enclosure at end of 24-hr test (psig)

Average barometric pressure = in. Hg times 0.491 = psi

$P_{\text{cell avg.}}$ = average gage pressure inside cell over test period (psig)

P_{b1} = barometric pressure at start of test (psi)

P_{b2} = barometric pressure at end of test (psi)

T_1 = cell temperature at start of test (°R)

T_2 = cell temperature at end of 24-hr test (°R)

$T_{\text{cell avg.}}$ = average temperature inside cell over test period (°R)

Inaccuracies in reading temperature and pressure are insignificant in the measurement of cell leakages which are the order of 1%/day for the proposed tests. Potential instrument error totals about 0.05% of cell volume. Estimated instrument precision and potential error are as shown in Table V(d)-1.

TABLE V(d)-1

	Instrument Precision	Potential Error
Pressure, Static	±1 in. H_2O	0.2% leakage
Pressure, Differential	±0.1 in. H_2O	0.02% of cell volume
Temperature, Absolute	±0.5 °R	0.1% leakage
Temperature, Differential	±0.1 °R	0.02% of cell volume

*Test has been revised from the gas flow test proposed in the parent report.

c. Test Criteria

The criterion for cell leakage was originally determined to limit nitrogen consumption (Table III, NAA-SR-MEMO-4067, p 19; NAA-SR-5700, p 4-72) below 15,000 scf per month. This requirement was more than satisfactory for the postulated accidents discussed in these reports. Additional calculations have been made to establish new acceptable leak rates which still maintain conservative safeguards standards. The basis for establishing these "Safeguards Tests" follows. These "Safeguards Test" leak rates are: 4.2%/day at 1.0 psig for the IHX cells, 2.1%/day at 1.0 psig for the primary pipe tunnel, 5.4%/day at 2.0 psig for the service cells, and 2.5%/day* at 2.5 psig* for the fill tank cells. Each cell will also be tested at the Construction Test Pressure (2.5 psig). The cells will be accepted when leak repair techniques now being employed no longer reduce leakage, provided the leakage rates are below the safeguards limits.

PRIMARY COOLANT CELLS SAFEGUARDS CRITERIA

I. DESCRIPTION

Primary system enclosures are separately enclosed cells; one for each of three primary-to-secondary sodium heat exchangers, associated piping and primary sodium pump; a primary sodium piping tunnel containing connecting piping from reactor to the three cells; a primary fill tank cell containing the primary fill tanks (normally non-radioactive) with associated piping; and the primary sodium cell containing carbon traps, cold traps, pumps and associated piping and fittings.

Each cell is encased in concrete, thus affording lateral biological shielding. Vertical shielding is provided by removable concrete shield blocks sealed to the side shielding by neoprene gaskets.

Primary system enclosures are provided with two water-cooled nitrogen blower units, one normally operating and one standby. The standby unit is controlled for automatic startup when nitrogen exit temperature from the cell approaches 150°F. Nitrogen pressure in the cell is regulated to between 1 and 6 in. of water. Nitrogen is obtained from a liquid nitrogen system, and is essentially water free. A relief system, connected to the radioactive vent gas system,

*Reduced from calculated values to avoid potential deformation of the wall between this cell and the adjacent cell.

operates at 2.0 psig. In event of failure of this system, and failure to detect cell pressure rise on installed measuring and alarm instruments, rupture discs operating at 5 psig relieve over-pressure to adjacent cell(s). Each cell is provided with a sheet steel liner to minimize gas leakage from the cell in normal operation and to contain any spill of radioactive sodium within the cell. The steel lining is complete, interrupted only by the sealing gaskets for the top shield plugs and penetrations for piping and electrical connections. These latter penetrations are individually sealed by bellows or direct welds in the case of piping, and epoxy seals around electrical conduits. Potential gas leak paths through the steel containment are through epoxy conduit seals, pipe hangers, and through leaks in the welds joining the steel plates forming the gas barrier.

The gas leakage rate acceptable from each cell from the standpoint of health hazard in event of a sodium spill within the cell is the pertinent factor required to evaluate the cells for "normal" operation. A total integrated dose for an infinite time of exposure in the reactor high bay area (immediately adjacent to the cells) of 3 rem is assumed to be tolerable. Maximum leakage tolerable was calculated from a "normal" reactor operating condition as follows:

Na^{24} activity saturated at 0.21 curie/gm

Fission products in sodium amounting to 0.05% release from 1% of fuel rods in the reactor. Any halogen fission products released from fuel are retained in the sodium as halides.

Reactor thermal power of 254 Mw, with average fuel exposure of 8,000 Mwd/T

Complete reaction of all oxygen in cell with sodium (1% oxygen concentration assumed).

It is further necessary to ensure that the cell leakage rate is acceptable in case of core failure and subsequent cladding damage. For this condition a total integrated dose of 3r resulting from one-half hour exposure after a simultaneous core meltdown and major sodium spill is assumed to be tolerable. "Casualty" conditions are assumed as follows:

Na^{24} activity saturated at 0.21 curie/gm

Fission products in sodium amounting to 25% halogen release and 50% rare gas release from 33-1/2% of the core inventory. Halogens are retained in the sodium as halides.

Reactor thermal power of 254 Mw, with fuel average exposure of 8,000 Mwd/T

Complete reaction of all oxygen in cell with sodium.

Permissible cell leak rates were first calculated for "normal" operation, then checked for acceptability under "casualty" conditions.

II. PRELIMINARY CALCULATIONS

Cell gas leakage maxima established in preliminary design (Table III, NAA-SR-MEMO-4067, p 19; NAA-SR-5700, p 4-72) were set on the basis of an arbitrary consumption limit of 15,000 scf of nitrogen per month from economic considerations. Evaluation of the dose rates resulting from this leakage from each cell, under both normal and casualty conditions, showed conclusively that no unacceptable hazard was involved.

Physical constraints on construction, stemming from minute weld imperfections, diffusion through electrical insulation, gas penetration through electric conduit seals, and gas diffusion through minute metal flaws have demonstrated that adherence to total nitrogen consumption criteria alone is impractical. Therefore, permissible gas leakage rates have been recalculated on the basis of allowable radiation tolerances.

III. ALLOWABLE LEAKAGE EVALUATION METHOD

A. ASSUMED ACCIDENTS

It has been assumed that massive sodium leakage occurs into one cell under each of two conditions:

1) After steady full power operation with 1% of the fuel rods releasing 0.05% of their fission products to the coolant sodium. Fission product release is conservatively estimated on the basis of recoils from the fuel material surface, and further assumes complete mixing of coolant sodium and that contained in the bond between fuel material and cladding.

2) After steady full power operation followed by fuel meltdown in which 25% of the halogen and 50% of the rare gas fission product inventory is released from 33-1/3% of the fuel rods. Disassembly and melting of one-third of the fuel elements will render the core subcritical (p 4-82

of the parent report) and therefore not subject to further power generation and consequent destruction of fuel elements.

B. SODIUM SPILL CRITERIA

The amount of sodium spilled in each of the five potential radioactive spill areas is assumed to be the maximum available and spilled instantaneously. This is considered to be conservative as protective valves, operated remotely, are available for limiting the sodium flow from a spill. Spill indications are available in pipe leak detectors, level alarms, and temperature indicators. In addition, complete severance of piping (or vessels) is assumed, taking no credit for the flow-retarding effects of thermal insulation, or freezing of sodium upon reaching the cell atmosphere which is maintained at least 58°F below the freezing point of sodium. Assumed sodium spill volumes are:

1) Intermediate heat exchanger cells and primary pipe tunnel: 160,000 lb of sodium at 945°F. This amount is the total available in the reactor upper plenum above the outlet nozzles, plus the contents of one primary loop.

2) Fill tank cell: 21,000 lb of sodium at 775°F. This is the entire quantity contained in one primary loop at the average temperature of the loop. Stop valves normally isolate the fill tanks from the primary loop; sodium volume is that from one loop which has been drained into the fill tanks. A subsequent rupture of the fill tanks or connecting piping has been assumed.

3) Service cell: 7,000 lb of sodium at 945°F. This is the amount of sodium discharged into the service cell after 10 min of pumping with the 100 gpm sodium service pump.

C. CELL PRESSURE RISE CRITERIA

Upon admission of the postulated amounts of sodium noted in (B) above, it is expected that cell pressure will rise due to heating of the nitrogen atmosphere, release of energy by gamma heating from Na^{24}, and energy of reaction between sodium and air or water vapor. At the same time heat may be removed by conduction to the surrounding concrete cooler operation. The criteria used for thermodynamic calculations were:

1) Normal cell venting and relief systems do not operate.

2) Heat capacity of the steel cell lining is included.

3) No heat transfer from the steel cell lining to surrounding concrete is included.

4) Water flow at 83°F is maintained to the coolers.

5) All beta decay energy, and one-half the gamma decay energy, remains with the sodium. Sodium activity assumed saturated initially at 0.21 curie/gm, decaying with a 15-hr half-life.

6) A heat transfer coefficient of 2.0 Btu/hr-ft^2-°F from sodium pool to nitrogen gas was assumed. This is conservative in that it neglects the insulating effects of the oxide layer which must be formed on the sodium surface to at least some degree.

7) The heat generated by sodium-oxygen and sodium-water (air moisture) reactions have been neglected, as the sensible heat released by this mechanism was less than 1% of that contained in the sodium.

D. CALCULATION OF CELL PRESSURE RISE

Using the assumptions in (C) above, calculations and analog simulations of pressure and temperature rise were made. Cell parameters and results are given in Table V(d)-2, and a typical analog simulation is displayed in Figure V(d)-3.

TABLE V(d)-2

MAJOR SODIUM SPILL IN PRIMARY COOLANT CELLS

	Primary Pipe Tunnel — Moderator Coolant Pump Cell	IHX Cell	Fill Tank Cell	Service Cells
Net volume (ft^3)	54,500	27,400	29,800	21,600
Floor area (ft^2)	2,060	1,045	2,530	1,020
Sodium spilled (lb)	160,000	160,000	21,000	7,000
Temperature of Na (°F)	945	945	775	945
O_2 in cell (%)	1.0	1.0	1.0	1.0
Normal heat removal (Btu/hr)	1,000,000	450,000	155,000	93,000
Max. temperature of N_2 (°F)	170	174	297	216
Max. pressure of N_2 (psig)	1.0	1.0	3.9	2.0

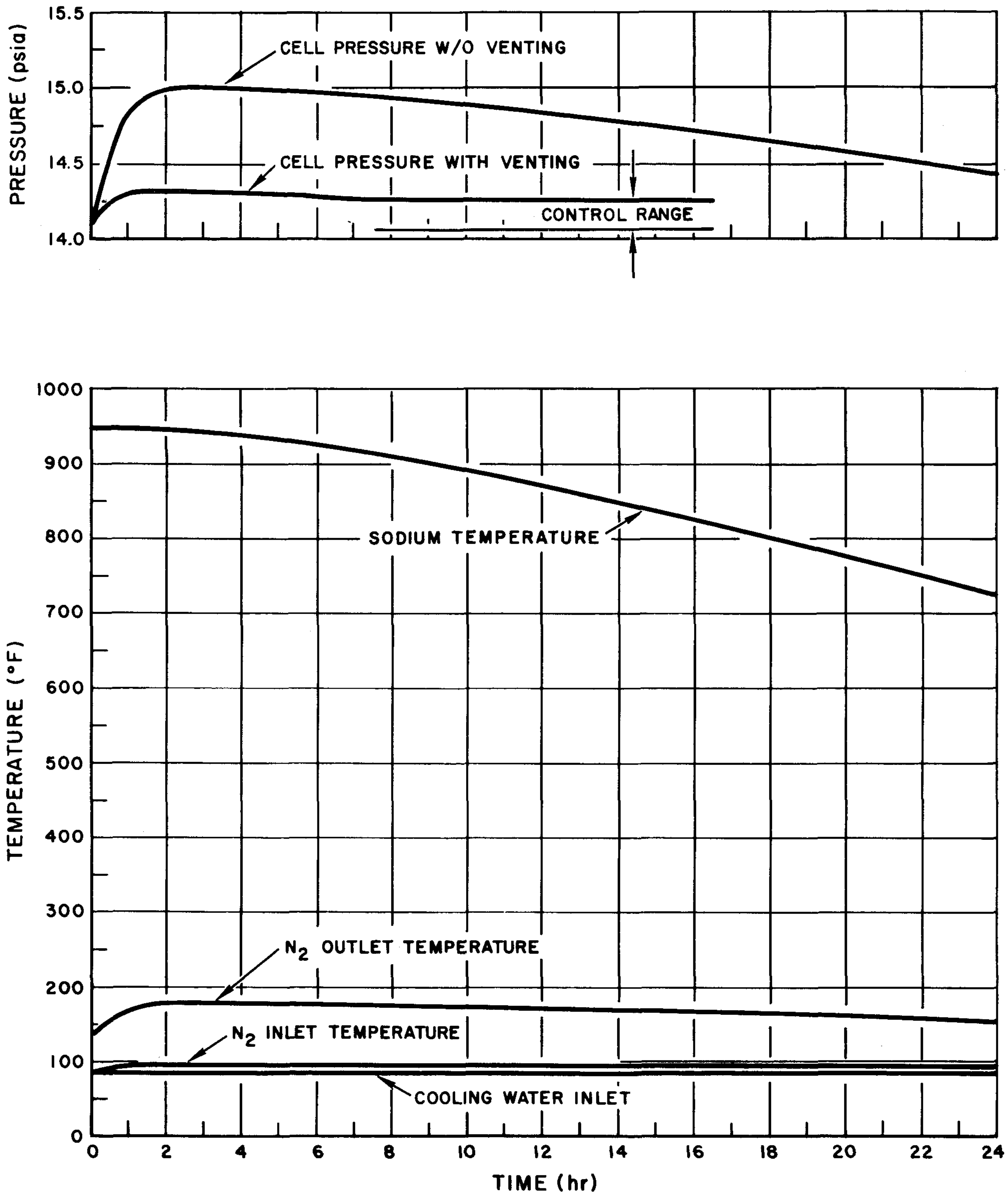

Figure V(d)-3. Sodium Spill Simulation

E. DOSE SOURCE EVALUATION

Pressure calculations described in (D) above have been utilized to calculate dose rate as a function of leakage. A further measure of conservatism has been incorporated by assuming that the peak pressure calculated is maintained throughout the period of leakage, whereas Figure V(d)-3 demonstrates that the pressure will decline as heat is dissipated to the coolers and cell walls, and as sodium activity decays.

Two sources of airborne radiological hazard are present in event of sodium spillage in the cells and consequent leakage. First is the Na^{24} activity contained in sodium oxide smoke, and second are the xenon and krypton gases, which may be expected to be released as a result of iodine and bromine decay in the sodium. The halides in the sodium will remain in solution.

1. Sodium Oxide Source

The maximum potential sodium oxide source was derived by assuming a 1% oxygen content in the nitrogen atmosphere of the cell, and assuming complete reaction of oxygen with sodium. This reaction is: $2\ Na + 1/2\ O_2 \rightarrow Na_2O$, 2.9 lb of sodium will react with each pound of oxygen present. A small amount of sodium hydroxide may be formed by water vapor present in the cell. This hydroxide will increase the total potential airborne sodium source by about 7% that of the oxide. The maximum potential sodium oxide and hydroxide generated was assumed to mix uniformly with cell nitrogen, and leak from the cell at the same rate as the nitrogen. This assumption is, of course, conservative as a substantial fraction of the oxide will remain on the surface of the sodium; significant quantities of the airborne particles would fall out in the cell prior to entering the leakage nitrogen stream, and the sodium oxide, being particulate, can be expected to undergo a filtering action in the minute leakage paths contributing to leakage of pure gas in cell pressure tests. The Na^{24} activity in the primary coolant is assumed saturated at 0.21 curie/gm. Airborne Na^{24} in the cells, calculated according to the foregoing assumption, is shown in Table V(d)-3.

2. Xe and Kr Source

The potential rare gas source in a cell atmosphere was based on the assumed carry-over of the halogen contamination by the spilled sodium. (Any rare gas released directly from the fuel would escape to the core cover gas volume.)

TABLE V(d)-3

POTENTIAL AIRBORNE SODIUM-24 SOURCE

	Cell Volume (ft^3)	Maximum Na^{24} Source (airborne) (curies)
Primary pipe tunnel	5.45×10^4	9.9×10^3
IHX cell	2.74×10^4	5.0×10^3
Primary fill tank	2.98×10^4	5.4×10^3
Service cell	2.16×10^4	3.9×10^3

Halogens with half-lives less than 90 sec and those decaying to stable gases were neglected. The parent-daughter halogen-rare gas chains of significance are:

$$Br^{83} \rightarrow Kr^{83m}$$
$$Br^{85} \rightarrow Kr^{85m}$$
$$I^{131} \rightarrow Xe^{131m}$$
$$I^{131} \rightarrow Xe^{133m},\ Xe^{133}$$
$$I^{135} \rightarrow Xe^{135m},\ Xe^{135}$$

The initial source of each of these halogen isotopes that could exist in a cell as a result of a sodium spill was calculated by,

$$H_{(c,i)}(o) = Q_i \cdot S_i \cdot F_c$$

where

$H_{(c,i)}(o)$ = total amount of halogen isotope i in cell c at t = 0, curies

Q_i = total core inventory of halogen isotope i (saturated), curies

S_i = fraction of halogen isotope i contained in primary sodium

F_c = fraction of total primary sodium spilled in cell c.

Q_i was based on the yield of the isotope and the nominal thermal power of the reactor (254 Mw). S_i was taken as 5.0×10^{-6} (0.05% release from 1% of the fuel rods) in Case 1 and as 8.3×10^{-2} in Case 2 (25% release from 33% of the core inventory). F_c was assumed to be 0.40 for the primary pipe tunnel and IHX cell, 0.053 for the primary fill tank cell and 0.018 for the service cell. These values represent sodium spills of 160,000, 21,000, and 7,000 lb, respectively.

The buildup of the Xe and Kr isotopes in the cell atmosphere as a function of time after the sodium spill was calculated by the appropriate chain decay formulae for each halogen isotope.

3. Dose Calculations

Since the leakage criteria are based on a total integrated dose, the calculations were made as follows:

For Na^{24} two modes of exposure occur: The internal dose from the inhalation and body retention of the sodium oxide and the external dose from the surrounding atmosphere of radioactivity. Both of these exposures may be related to a concentration-time integral. The internal dose was based on a total dose of 1.04×10^{-3} r/μc inhaled for Na^{24} in a soluble form.*

Then assuming a breathing rate of 347 cc/sec,

$$D_I = 1.04 \times 10^{-3} \text{ rem}/\mu\text{c} \cdot 347 \text{ cc/sec} \cdot E_T \frac{\mu\text{c-sec}}{\text{cc}}$$

$$D_I = 0.361 \times E_T$$

where

D_I = total dose (whole body) from inhalation of Na^{24}, rem

E_T = total exposure, $\frac{\mu\text{c-sec}}{\text{cc}}$.

The external whole body dose was calculated by assuming a person was located at the bottom center of a hemisphere of a valume equal to the high bay area volume and with an airborne concentration of Na^{24} of 1 μc/cc. This dose was determined as

$$D_E = 0.087 \times E_T$$

Therefore, the total dose is simply the sum of these two sources, or

$$D_T = 0.448 \times E_T$$

The total integrated concentration-time factor, E_T, was calculated by

$$E_T(\infty) \int_o^\infty C_{H.B.}^{(t)} dt = \int_o^\infty \frac{L_c \cdot Q_c(o)}{R_2} \left[e^{-(L_c + \;)t} - e^{-(L_{H.B.} + \lambda)t} \right] dt^{\dagger}$$

*Calculated from tables by Morgan, Snyder, and Ford, "Maximum Permissible Concentrations of Radioisotopes in Air and Water for Short Period Exposures," Geneva Conference Paper, 1958.

†This equation has been simplified by assuming that the cell leakage rates will be quite small relative to the high bay area ventilation rate.

where

$C_{H.B.}^{(t)}$ = Na^{24} airborne concentration in high bay area at time t ($\mu c/cc$)

L_c = leakage rate of cell c (volume fraction per sec)

Q_c = Na^{24} source in cell c at t = o (μc)

R_2 = ventilation rate of high bay area (cc/sec)

λ = radioactive decay constant of Na^{24} (sec^{-1})

$L_{H.B.}$ = ventilation rate of high bay area (volume fraction per sec).

This integral equation was solved for L_c for the different cells with the value of

$$E_T(\infty) \leqslant \frac{3}{0.448} \leqslant 6.7 \frac{\mu c\text{-}sec}{cc}$$

and therefore,

$$D_T(\infty) \leqslant 3 \text{ rem}.$$

With this condition, the acceptable leakage rates were determined as 2.1, 4.2, 3.9, and 5.4 vol %/24 hr for the primary pipe tunnel, IHX, primary fill tank, and service cells, respectively.

For the Xe and Kr only the external cloud exposure is significant. The dose rate in the high bay area was related to the air concentration by conservatively assuming that exposure to the occupational MPC for each isotope was equivalent to 100 mrem/40 hr or 2.5 mrem/hr. The MPC's used are listed in Table V(d)-4.

TABLE V(d)-4

MPC_{air} FOR Xe AND Kr ISOTOPES OF INTEREST

Isotope	MPC_{air}* ($\mu c/cc$)
Kr^{83m}	6×10^{-5}†
Kr^{85m}	6×10^{-6}
Xe^{131m}	2×10^{-5}
Xe^{133m}	1×10^{-5}†
Xe^{133}	1×10^{-5}
Xe^{135m}	5×10^{-6}†
Xe^{135}	4×10^{-6}

*Based on 2.5 mrem/hr (100 mrem/40 hr) in an infinite hemispherical cloud.
†These values were calculated by methods given in "Report of Committee II on Permissible Dose for Internal Radiation (1959)," Health Physics, Vol 3, June 1960, p 22.

Using the acceptable leak rates determined previously with the Na^{24} source, the concentration of each Xe and Kr isotope listed in Table V(d)-4 was calculated for the high bay area as follows:

The source term for each isotope in a cell atmosphere was plotted as a function of time after the sodium spill. These curves were then used as the source bases for the high bay area concentrations in a manner similar to that previously described for the Na^{24} source. The resulting high bay area concentrations were converted to dose rates by the equivalency discussed above. These dose rates were then summed to obtain total dose rates as a function of leakage time. By numerically integrating these results, cumulative doses as a function of leakage time were derived. From these results it was found that with a sodium spill under normal reactor conditions, the acceptable Na^{24} source leakage rates held the total dose from the rare gases in the high bay area to a negligible quantity (< 10 mr). With the "casualty" condition a rare gas source increase of a factor of about 1.68×10^4 occurs. In this case, the rare gas source becomes the dominant dose factor for leakage from the primary pipe tunnel and IHX cells, while both the Na^{24} and rare gas sources must be considered with the primary fill tank and service cells. However, the leakage rate criteria established for the normal case are still more than adequate to meet the requirement of a 3-rem dose in the high bay area in the first 1/2 hr.

The additional site boundary dose criteria of a total integrated dose of less than 25 rem at the site boundary or beyond is met in Case 1 for three reasons. First, the high bay gases are diluted in the stack by a factor of about 0.7, then effluent gases are diluted by at least 7×10^{-4},* and finally, the normal filters in the ventilation system will remove most of the oxide. The inhalation dose will be diluted directly by these factors. For the external portion of the total dose at the site boundary, the reduction is less due to the larger source radius to be considered. The net result is about 0.008 rem if the sodium oxide is not filtered and insignificant if filtered. With Case 2, the sodium oxide contributions are unchanged since the leak rates established by Case 1 are used. The noble gas TID_{∞} at the site boundary was estimated from the high bay dose calculations for the first day following the spill and using the 0.7 and 7×10^{-4} factors. The results of the above mentioned calculations are shown in Table V(d)-5 and are well below the criteria in all cases.

*Figure C.1 of the parent report.

TABLE V(d)-5

SUMMARY OF DOSES BASED ON SAFEGUARDS CELL LEAKAGE RATES

Dose Criteria	Primary Pipe Tunnel	IHX	Primary Fill Tank	Service Cell
Safeguards leak rate (vol %/24 hr)	2.1	4.2	3.9	5.4
<u>Case 1</u>				
TID in high bay area				
a) Na^{24}	3.0 rem	3.0 rem	3.0 rem	3.0 rem
b) Xe and Kr	neg.	neg.	neg.	neg.
TID at site boundary				
a) Na^{24}	neg.*	neg.*	neg.*	neg.*
b) Xe and Kr	neg.	neg.	neg.	neg.
<u>Case 2</u>				
Exposure time necessary in high bay area to receive 3-rem dose a) Na^{24} b) Xe and Kr	5.0 hr	3.1 hr	11 hr	14 hr
TID at site boundary				
a) Na^{24}	neg.*	neg.*	neg.*	neg.*
b) Xe and Kr	~0.04 rem	~0.09 rem	~0.01 rem	~0.01 rem

*If no credit is taken for sodium oxide filtering, a site boundary TID of about 0.008 rem would occur.

IV. CONCLUSION

The calculated dose rates stemming from leakages noted in Table V(d)-5 contain these conservative factors:

a) The maximum volume of sodium available for reaction is assumed to be spilled instantaneously, therefore, making its entire energy available for cell heating. Even the most sanguine postulation of pipe leakage cannot provide sodium instantaneously.

b) All reaction products of sodium and oxygen or water vapor are assumed to be available for uniform mixing with the leaking gas. It is known, however, that a fraction of the sodium oxide particles are large enough to settle out of the gaseous atmosphere.

c) It is assumed that all sodium oxide particles in the leakage gas will escape with the gas. In view of the minute nak ure of the leaks, some "filtering" action may certainly be expected.

d) Peak cell pressures are calculated using heat transfer coefficients which neglect any insulating effect of sodium oxide on the sodium surface.

e) Peak cell pressures are assumed to be maintained throughout the leakage period, whereas analog studies (Figure V(d)-1) indicate they decline.

f) No venting of the primary cells is assumed for the radiological calculations. Cells are intact, connected to the radioactive waste gas collection and storage system, so that venting may be expected to quickly limit and reduce pressure. This is illustrated on Figure V(d)-1 for an IHX spill.

Tests proposed to verify the cell leakage rates calculated to be safe have been shown to be accurate to less than 0.05% of cell volume. Therefore, it is considered that the tests proposed, and the tests proposed, and the leakage rates specified in Table V(d)6 below are completely adequate to protect reactor operating personnel from excessive exposure under the most unfavorable conditions of sodium spillage into shielded primary cells, and much more than adequate to protect the public beyond the site boundaries.

TABLE V(d)-6

CELL LEAKAGE TESTS

		Primary Pipe Tunnel	IHX	Primary Fill Tank	Service Cell
Construction Test					
Pressure	psig	2-1/2	2-1/2	2-1/2	2-1/2
Leak Rate Criteria	3/4 of 1% std. vol. per 24 hr at 2-1/2 psig				
Safeguards Test					
Pressure	psig	1	1	2-1/2*	2
Leak Rate Criteria	% std. vol. in 24 hr	2.1	4.2	2.5	5.4

*Reduced from calculated value in Table V(d)-2 to avoid potential deformation of wall between cells

Question V(e): Results of all enclosure cell leak rate tests should be furnished when available.

Answer: The results of the cell enclosure tests described in Question V(d) will be evaluated and transmitted to the Test and Power Reactor Safety Branch of the Division of Licensing and Regulation in letter form at the conclusion of each test. Testing is currently underway, including finding and plugging the small leaks which exist.

Question V(f): What is the precision and accuracy of the continuous leakage measurement system for the reactor cover gas?

Answer: The continuous leakage measurement system was installed to detect gross helium leakage in the absence of cover gas activity. The system will provide an indication of leakage if the leakage in standard cubic feet per day is greater than

$$\left[6.1 \text{ scf/day}\right] + \left[\frac{180 \text{ scf}}{\text{number of days of test}}\right]$$

The first term represents the expected average use of helium with no leakage (50 scf/day) times the precision of measurement of this flow through the two meters (12.4 percent). The second term represents the potential unmeasured accumulation of gas in the system. A leak of 42 scf/day should indicate as a positive leak in about 4.3 days of power operation.

Discussion:

Gas leakage from a fuel element shield plug with seals missing was analyzed (for Question 2 in MAA-SR-Memo-4067) with the conclusion that core melt-down, combined with a leak rate of 120 ft^3 per day helium, (42 scf/day), would result in about 0.4 R TID at the site boundary. The area radiation monitors are effective in detecting leaks of about 120 ft^3/day when the cover gas activity is greater than 0.6 μc per cm^3 and effective in detecting leaks of about 0.11 ft^3/day with 630 μc per cm^3 activity.

Helium gas flow into the reactor is measured on a positive displacement flowmeter with an accuracy of about 1/4% of the flow. Temperature is estimated to remain within a ± 10°F range, introducing a possible error of about ± 1.9% in the measurement. Barometric pressure swings will be within about ± 3.0%. Thus, gas flow into the reactor will be measured within about ± 5.15% by the flowmeter.

Helium gas flow out of the reactor complex will be measured on a second positive displacement flowmeter of 1/4% accuracy. Temperature is estimated to remain within a ± 20°F range, for an accuracy of about ± 3.8%. Again, the pressure swing will be within about ± 3.0%, for a maximum accuracy of volume measurement of about ± 7.05%. The maximum error for gas that flows through both flowmeters is the sum of these limits, or 12.2%.

Thermal swings of both sodium temperature and helium temperature are expected to be within the dead band of pressure regulation when the reactor is operating at power and not cause gas flow. Barometric pressure change is estimated to pump an average of 50 ft^3/day of helium. Using the maximum error of gas flow (12.4%) results in an average uncertainty of flow measurement of 6.2 ft^3/day.

Capability of arriving at a realistic leak rate from the reactor, based on the above flow measurements, is dependent on knowing the absolute quantity of gas within the reactor. The nominal volume of gas in the system is ~9500 ft^3, with about 1350 ft^3 in the reactor, and 8150 ft^3 in the storage tanks. This is equivalent to about 5720 scf with the reactor blanket at 945°F and the fill tank blanket at 350°F.

Estimated inaccuracies of system gas volume at a specific time due to various factors are as follows:

Variations due to inaccuracies in temperature and pressure measurements	25 scf
Variations due to changes in sodium temperature (volume)	13 scf
Variations due to gas temperature swings	142 scf
Total uncertainty in measuring gas in system	180 scf

The general equations for calculated leakage and proven leakage are:

$$L_c = \frac{F_i - F_o - \left(\frac{p_i - p_f}{p_{avg}}\right) 5720}{D} + 6.2$$

$$L_p = L_c - \frac{180}{D}$$

where:

F_i = helium flow into reactor (scf)

F_o = helium flow out of reactor (scf)

$\frac{p_i - p_f}{p_{avg}}$ = Fraction change in absolute pressure of the reactor cover gas during a test run

D = Duration of a test run (days)

L_c = Calculated leakage (scf/day)

L_p = Proven leakage (scf/day)

If L is positive, it can be assumed that there is a leak in the system.

If L_p is zero, or is negative, then a leak, if present, is less than $\frac{180}{D}$ (scf/day).

Continuous helium leakage tests will be conducted when the reactor is operating near steady state thermal conditions (over 15% full power). Flow, system pressure, and ambient temperature, will be tabulated and evaluated once a day to estimate system leakage. The calculated leakage is the measured inlet flow (± 5.15%) minus the measured outlet flow (± 7.05%) corrected by the system pressure change divided by the time, all measured from the start of the test. The proven leakage value is the calculated value minus the uncertainty described on the equation above. A proven leak of 42 scf/day will therefore calculate as a positive value at 4.3 days. After a 10-day test run, leakage above about 18 ± 6.2 scf/day will be detected.

If gross leakage is indicated by daily evaluation of flowmeter readings, but in the range where the value is masked by the uncertainty, further leak rate testing will be initiated by careful control of conditions during a test period. By temperature, pressure flow and level measurement at the time of a questionable leak, the accuracy of the test can be improved. If the leak rate is shown to be high, helium mass spectrometer will be instituted to locate the source of the leak and repair action will follow.

Question V(g): What would be the probable effect of a large hot sodium spill on the steel liners of the IHX cells and pipe tunnels. Could such an incident result in loss of integrity of the liner?

Answer: The probable effect is to buckle the liner between supports. No loss of integrity of the liner is anticipated. A small sodium spill on the steel liner will result in a localized hot spot. Assuming the temperature of the sodium spill on the liner to be 1000°F, the maximum possible stress due to this hot spot is ~81,000 psi. Based on fatigue criteria, the steel liners should be able to withstand over 1000 such cycles.

If a large spill occurs, the liners, where not restrained by the "T" supports, will move away from the concrete. The two cases considered were: (a) a linear gradient through the liner, and (b) a uniform temperature through the liner.

For a linear temperature gradient through the liner, the following equation describes the deflection:

$$d = \frac{\alpha r^2 \Delta T}{2t}$$

Where:

d = Deflection

α = Coefficient of thermal expansion

ΔT = Temperature gradient through the liner

t = Liner thickness

r = Radius of sodium spill.

For example, with a sodium spill 5 ft in diameter with a 900°F linear ΔT through the 1/4 in. floor liner, the liner will experience a maximum deflection of ~11 in. This indicates the magnitude of displacements which can be expected.

If the liners are uniformly heated through their thickness, a rise in the average liner temperature of 240°F will buckle the 1/4 in. liner plate between the "T" supports which are 6 ft on centers. Higher average temperatures will simply increase the size of the buckle.

All of the above hypothesized conditions result in displacement type stress induced by the thermal conditions. Considering the weld which joins the liner plates to have a ductility greater than, or equal to, 90 % of the liner material, the liners should be capable of withstanding more than 1000 cycles such as those created by a large hot sodium spill.

At corners where the wall liners are joined to the floor liner, a much more rigid structure exists; consequently, buckling or other thermal generated movement is not expected.

The maximum thermal stress which can be induced at the corner is equal to $E\alpha\Delta T$ or 135,000 psi for a ΔT equal to 900°F. The liners, considering the weld to be 90 percent as ductile as the parent material, can withstand over 100 such hot sodium spill cycles.

Question VI(c): Information regarding design and testing of the scrubber should be provided when available.

Answer: The air scrubber is an Amertherm collector as produced by the American Air Filter Company specifically for the Hallam application. Pertinent specifications are as follows:

a) The scrubber may be started manually from the control room on either normal or emergency power. The system starts automatically on loss of main building power. This feature is to insure negative pressure in the building on loss of normal power (irrespective of a concurrent sodium fire in the area).

b) The system is designed for an air flow rate of 7500 cfm of air at temperatures up to 500°F. This flow rate is estimated sufficient to maintain negative pressure in the reactor high bay under assumed casualty conditions of a sodium fire.

c) The system will remove a minimum of 98.5% of all predicted sodium oxide from the flowing air stream, and will operate for a minimum two-week unattended operating period. System contains four units such that one unit is being regenerated while the other three units continue in service. Waste pre-coat material and sodium oxide are discharged automatically to an underground hopper sufficient to contain all material for a two-week unattended run.

d) The filter medium is a glass fiber bag-type filter unit pre-coated with asbestos fiber. The automatic regeneration equipment includes a pre-coat medium hopper; a feeder; and blowers of sufficient capacity for the specified two-week unattended operating cycle.

e) Maximum sodium oxide loading of the air feeding the filter is estimated at 1000 lb/hr, with the following estimated gradation:

6	micron	47%
4	micron	26%
2 1/2	micron	5%
1 1/2	micron	18%
3/4	micron	4%

In operation, the scrubber system is normally inactive. Standard plant operating procedure shall be for the equipment to be tested a minimum of once a month. Testing shall consist of starting all blowers to ensure that they are operable; and that controls are functioning satisfactorily; and shall include cleaning and reseeding of one filter section to ensure that the automatic reseeding mechanism is in satisfactory working order.

Preliminary bench scale tests were run on the proposed filter medium using a sodium smoke from a test generator. Using coated filters, filter efficiencies of from 90 to 100% were found. On the basis of these tests, a high efficiency type filter was selected for further tests. (See American Air Filter Test Report, Project 1544, MSC, dated June 9, 1961)

Final tests were run on a model glass fiber bag-type filter of the type being installed at Hallam. Results are reported in a report from American Air Filter Company, Project 1544, MSC, dated June 27, 1961. Results of this test indicated that the filter is capable of 100% removal of NaK smoke from a flowing air stream under conditions approximating the predicted casualty condition of a sodium fire at Hallam. NaK smoke could not be detected in the effluent air, either by sight, odor, or refiltering through a high efficiency filter and measuring change in weight of this filter. Operation of the test rig for cleaning and recoating the filter bags indicated that the bags could be cleaned completely and recoated efficiently by the methods to be used at Hallam, without interruption of the filtering action or air flow (except in the filter section actually being regenerated).

The system as designed is shown diagrammatically on attached sketch, Figure VI (c)-1. In normal operation, the scrubber system is inactive, and is valved off by valves V-2, V-3, and V-5. Normal building ventilation flow is through valve V-1, with the main building filter system and main exhaust blower in operation.

Under emergency conditions V-1, V-3, and the main exhaust fan valves would be closed. A ventilating air flow of approximately 7500 CFM would flow through V-2, through the four scrubber units in parallel, and exhaust through the primary scrubber blower and V-5 to the stack.

As differential pressure across each filter unit increases above a pre-set point (about 4 in. water gauge) the unit is automatically cut off the line and regenerated. Regeneration is accomplished in two steps. First, the bags are collapsed using the reverse air blower and solenoid valves, for approximately four to six short cycles.

In this process, the pre-coat, together with entrained sodium oxide, drops to the bottom of the hopper, and is discharged into the storage bin through the rotary lock.

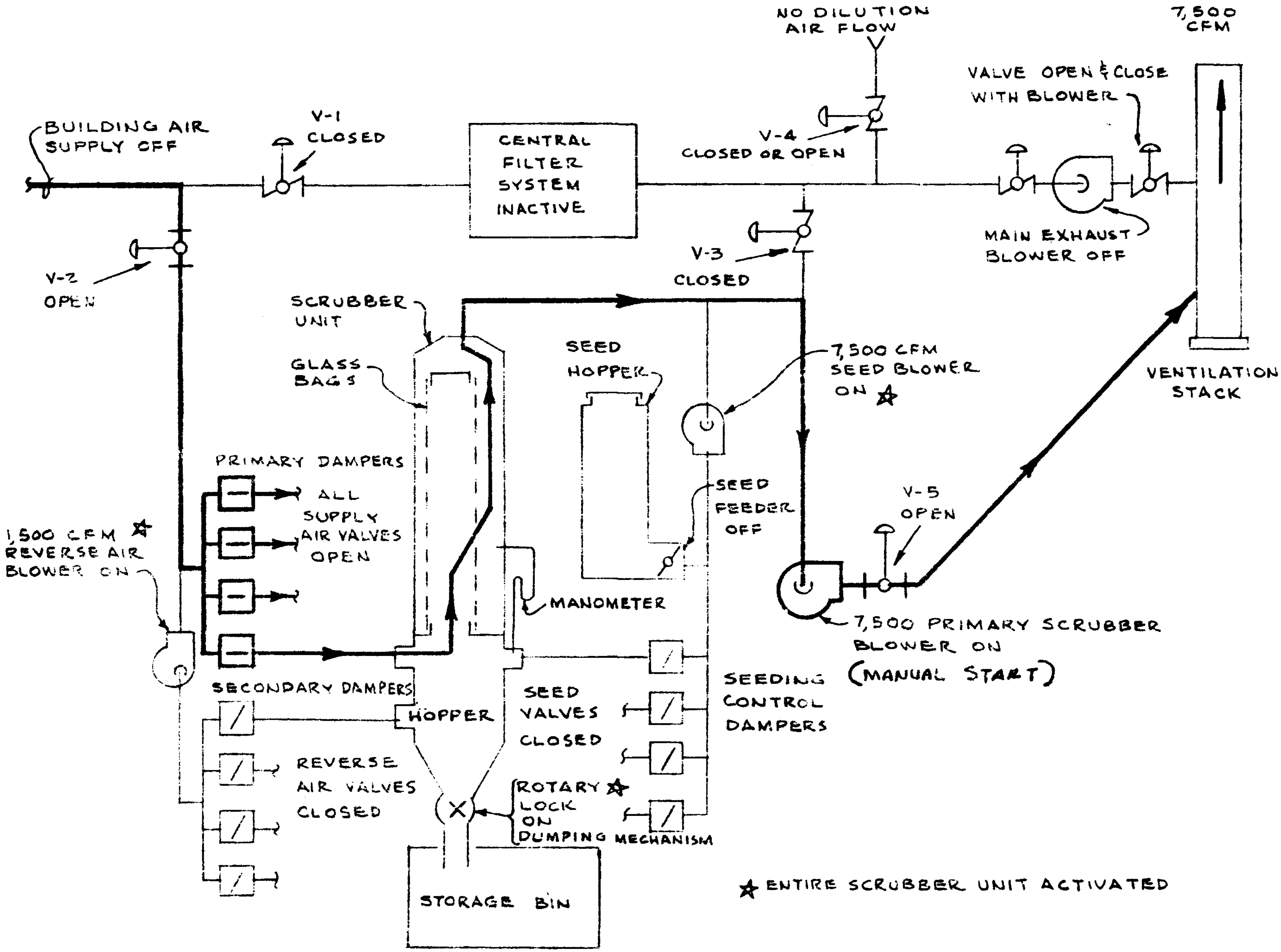

Figure VI (c)-1. Fume Collection During Scrubber Operation

The second step consists of pre-coating the filter. The seed blower comes on automatically, and the filter medium (finely ground asbestos fiber) is fed into the airstream in carefully regulated amounts, until pressure drop across the filter reaches 1 in. water gauge. At this point, the regeneration cycle is complete, and the unit returns to service automatically. An automatic interlock prevents regeneration of more than one filter unit at a time. Total regeneration time for one unit is about four minutes.

The primary scrubber blower serves an alternate purpose of supplying building ventilation during loss of power. During a complete power outage, the Diesel generator is not sufficiently large to drive the main building exhaust blowers. Thus, on loss of power, the main building exhaust blowers go off, and the primary scrubber blower comes on automatically, maintaining a slight negative pressure in the reactor building. Normal flow under these conditions would be through V-1; through the central filter system; through V-3; through the primary scrubber blower; and through V-5 to the stack. The scrubber system is not in service under these conditions, since no fire is present, and the normal building vent filters are adequate.

Following installation, the mechanical components of the scrubber system will be acceptance tested to assure proper operation.

Question VI(e): Further substantiation of the adequacy of the stack height should be provided. With respect to the stack height, it does not appear that consideration has been given to the effects of the adjacent steam plant structure, to velocities anticipated under various ventilation conditions.

Answer: The top of the HNPF stack is 100 ft above grade. The HNPF building height is 70 ft, and the tallest steam plant building, 480 ft away from the stack, is ~108 ft tall. A small extension above the steam plant building extends 143 ft above grade. Ventilation air intakes in the steam plant are 10 to 40 ft above grade. Relative heights of these buildings are shown in Figure VI(e)-1.

For airborne dispersal purposes the effective height of the stack is a function of wind velocity and stack exit velocity (temperature effects have been neglected for HNPF). Normally, the HNPF exhaust rate is held constant at ~80,000 scfm, which gives exhaust velocity of ~45 mph and results in the effective stack heights* listed in Table VI(e)-1. This table shows that for wind velocities below 8 mph the effective height of the stack is greater than that of the extension above the steam plant building. Only under sodium fire or loss-of-power conditions is the stack flow rate reduced significantly below normal (to 7500 cfm).

TABLE VI(e)-1

EFFECTIVE STACK HEIGHT

Wind Speed (mph)	Effective Stack Height (ft)
2	279
5	172
8	145
10	135
20	118
30	112
40	109

*This and other meteorological considerations with regard to the stack have been corroborated by Meteorology Research, Inc., Altadena, Calif.

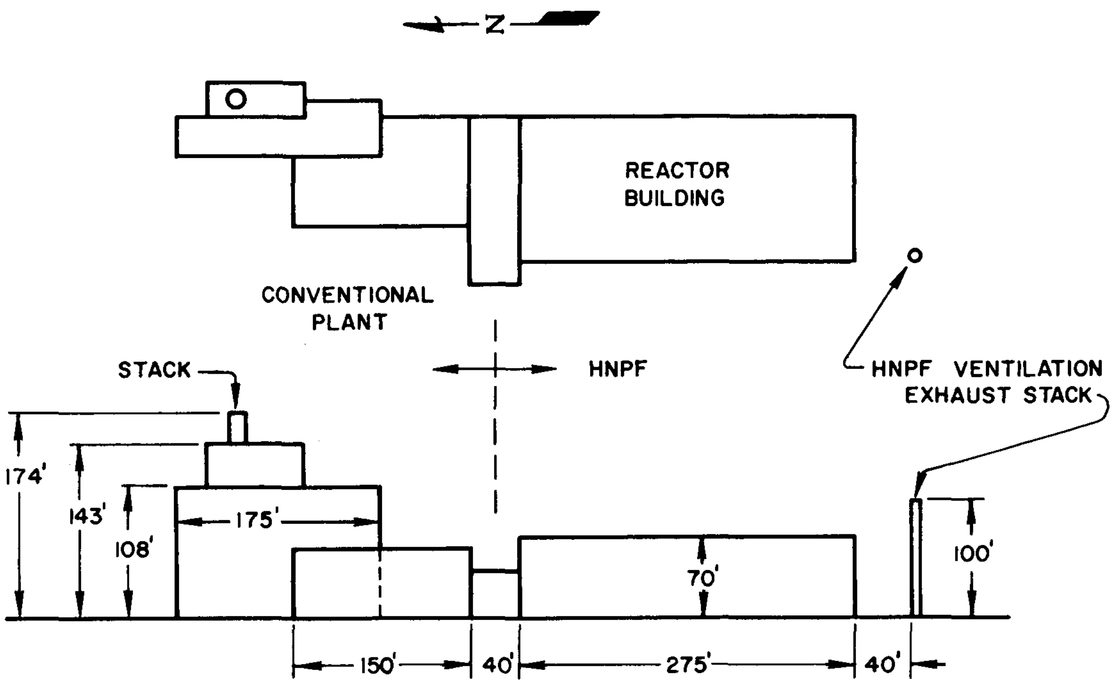

Figure VI(e)-1. Sheldon Station Layout

Interference or down-wash effects of the various building on the gases released from the stack are possible only under conditions of wind direction at the stack being directly towards or away from the adjacent buildings and relatively high wind velocities. When wind is from the building toward the stack, the 30 ft of stack above the nearest (reactor) building plus the high exhaust velocity will tend to make down-wash insignificant. Moreover, there are no on-site buildings downwind from the stack. The site boundaries are sufficiently far from the stack ($\geq$ 1/4 mile) that the minimum dilution factor of 1000 as shown in the parent report will not be significantly changed for off-site concentration of stack effluent even if some local down-wash effects do occur.

If the wind blows from the stack toward the steam plant structure, interference effects might be more pronounced. As mentioned above, with normal stack flow of 80,000 scfm wind velocities below 8 mph will result in effective stack height greater than the height of the tallest steam plant building, 480 ft from the stack. For wind velocity greater than 15 mph the turbulence will be high enough to render any interference effects unimportant. For wind velocities between 8 and 15 mph, the plume from the stack may directly strike the steam plant building.

The primary purpose of the HNPF stack is to discharge building ventilation air to the atmosphere. This air is necessarily at radiation levels below MPC. In this capacity the basic requirement is that the stack be somewhat removed from ventilation intakes, which are located near ground level. Since the stack is at least 30 ft taller than all buildings within 400 ft, there is no problem with regard to normal ventilation, irrespective of possible downwash.

From time to time, stored gaseous fission products will be released through the ventilation stack under strictly controlled condition. Factors limiting release of gases include:

1) Building ventilation rate must be normal, ~80,000 cfm.

2) Maximum concentration of stack effluent is limited to 1000 times Table II of 10 CFR 20. (The minimum dilution from stack to site boundary is 1000, including possible downwash effects.) Averaging of stack effluent concentration, as outlined in Question VI(b) of Supplement 2 to the parent report, is acceptable for this purpose.

3) The concentration of stack effluent will not be greater than 20 times Table I of 10 CFR 20, whenever the wind direction at the stack is directly toward the steam plant. (The minimum dilution from stack to steam plant is 20.)

The above limitation on controlled gaseous waste release make it possible to safely use the ventilation stack for this purpose.

During some of the accidents studied in the parent report the high bay atmosphere becomes contaminated with gaseous radioactivity. These accidents are tabulated in Table VI(e)-2. Many of these accidents can be terminated or at least reduced in severity by prompt operator action. It is possible for the high bay activity to be above MPC during some accidents, and this may require evacuation of the reactor building. The building ventilation system will continue to operate, releasing gaseous activity through the ventilation stack. Under the most unfavorable wind conditions, the minimum dilution factor from the high bay to the air outside the tallest portion of the conventional plant is calculated to be 26 with normal ventilation and 5 with emergency ventilation (7500 cfm). Normal ventilation calculations include dilution of 1.33 between the high bay and the stack and use neutral weather parameters and horizontal downwind diffusion. Emergency ventilation calculations assume horizontal downwind diffusion based on strong inversion parameters. This dilution will ensure that the personnel hazard at the conventional plant will be significantly lower than in the high bay area during these accidents. Should it become necessary to evacuate the reactor building because of airborne radioactivity, it may also be necessary to evacuate the conventional plant or at least its roof. Any such evacuation instructions will come from the AI Shift Leader [see Question IX(d)], whose decision will depend upon the actual concentrations released, ventilation rate, and wind conditions. The site boundary dose rates given in the parent report for the above accidents remain valid even when downwash effects are considered. The undesirable effect of downwash (reducing the advantage of an aboveground release) is partially compensated by the fact that downwash can occur only with relatively high wind velocities which add to dispersion and dilution of airborne activity.

Thus, it is concluded that the 100-ft stack height is adequate for all operating conditions. During planned operation, there is no radiation hazard from stack effluent. Should certain accidents occur, resulting in high concentrations of radioactive gas within the reactor building itself, evacuation of all on-site personnel might be required to avoid exposures above MPC.

TABLE VI(e)-2

ACCIDENTS LEADING TO STACK RELEASE OF RADIOACTIVITY

Postulated Accident	Average High Bay Concentration at Time of Maximum Concentration
(1) Fuel element meltdown in fuel handling machine	$3.0 \times 10^{-4}\ \mu c/cm^3$ (N)
(2) Dropped fuel slug	$6 \times 10^{-7}\ \mu c/cm^3$ (N) $6.8 \times 10^{-7}\ \mu c/cm^3$ (I)
(3) Dropped fuel element	$5 \times 10^{-5}\ \mu c/cm^3$ (N) $5.5 \times 10^{-5}\ \mu c/cm^3$ (I)
(4) Instantaneous oxidation of dropped fuel element	$1.5\ \mu c/cm^3$ (N) $1.6\ \mu c/cm^3$ (I)
(5) Primary sodium spill following core meltdown	160 x MPC (N) for 1% cell leakage in 24 hours
(6) Large fire of primary sodium	High (Na^{24}), requires evacuation
(7) Loading face shield gas leakage	$2.5 \times 10^{-5}\ \mu c/cm^3$ (N)
(8) Loading face shield gas leakage following core meltdown	$0.8\ \mu c/cm^3$ (N)
(9) Primary pump leakage	$1 \times 10^{-2}\ \mu c/cm^3$ (N)

(N) = noble gases, MPC = 10^{-5} to $10^{-6}\ \mu c/cm^3$

(I) = iodine or halogens, MPC of I^{131} is $9 \times 10^{-9}\ \mu c/cm^3$

MPC's from Table I, 10 CFR 20

Question VII(a): What provisions have been made to minimize the possibility of air/oil explosion in compressed air systems serving instrumentation and component operators?

Answer: The instrument air compressors are non-lubricated carbon ring type, with suction air filters. Even should oil enter the system by some unknown means, the oil would be trapped out in a series of removal facilities including after-coolers, separators, receivers, pre-filters and dryers. Thus, it is nearly impossible to get oil into the instrument air system within the nuclear facility. The instrument air compressors are located in the non-nuclear facility, remote from any nuclear hazard.

The service air supply to the nuclear facility is provided from a single oil-lubricated compressor. Normal provision is made for removal of the excess oil from the air. Oil is fed to the compressor on the suction side in measured amounts for compressor lubrication. Excess oil is removed in the intercooler and trap between stages, and in the after-cooler and trap prior to entering the service air receiver. If oil vapor should manage to pass the coolers uncondensed, it would tend to precipitate out in the receiver. The compressor is located in the conventional plant, remote from any nuclear components. It is therefore not reasonable to assume that oil of combustible concentration would pass to the nuclear facility.

The only area where an explosion is remotely possible is in the cylinders themselves, through a series of malfunctions such as excess oil feed together with loss of coolant water. If this should happen, any explosion would take place at the compressors, and would be relieved either from the relief valve on the receiver, or by failure in the compressor itself. No damage or distress would be reflected in the nuclear plant. Service to the nuclear plant would continue automatically from backup lines. Both instrument air and nitrogen supply the valve operators through lines isolated from the service air compressor by check valves. A manual cross-tie from the soot blower compressors in the conventional plant can also be used to resume normal service air supply.

Question VII(b): What would be the effect of a major fire and/or explosion in a steam generator room on the capability to control or shutdown the nuclear plant? What means are available to combat or control fires in the steam generator rooms should direct access be impossible?

Answer: A fire, even a major one, in a steam generator cell would not seriously affect the ability either to control or shut down the plant. Complete shutdown of the plant and/or isolation of a loop can be accomplished from the control room. Access to the steam generator room is not required.

The fire resistive ratings of the cell structure are listed below:

Item	Construction	Fire Rating (Uniform Building Code)
1) Wall between cells and control room, cable room, and switch-gear rooms (Column Line 13)	Reinforced concrete	4 hours
2) Ceiling	Reinforced concrete	3 hours
3) Walls between cells (Column Lines A, Ay, By, Cy)	Plaster on metal lath	1 hour
4) Wall separating cells from High Bay (Column Line 15.5)	Plastered concrete block	4 hours
5) Doors to switchgear room (3)	U. L. Class B swinging door, with Class A sliding door installed on south side of wall	Class B, 1-1/2 hr Class A, 30 min
6) Doors to High Bay (3)	U. L. Class B	1-1/2 hr

No control wiring runs through a steam generator cell except that directly concerned with items in that particular cell. Reactor control wiring is routed around the cells. The firedoors, one of which (the Class A door) is automatically closed in case of fire, adequately protect the switchgear room.

As described in Table 2.6 of the parent report, the "Pyr-A-Larms" installed in each cell will not only indicate the fire location on the fire protection panel in the control room, but will also shut off the air supply fan to that cell. Ample supplies of fire-fighting material will be on hand.

In the event of a fire of such proportions that direct access to the cell is impossible, the following means of fighting the fire are available:

1) The secondary pump will be shut down, and the helium supply to the loop shut off, to reduce the driving force on the sodium leak.
2) Nitrogen from the bulk supply (125,000 ft^3 minimum) can be fed into the cell. (Since the parent report was written, plans have been made to install line accessible from the outside. Connection between the line and the bulk supply will be by hose, to avoid the problem of inadvertantly getting a high nitrogen concentration during normal operation, and possible operator asphyxiation.)
3) Sodium can be drained into the secondary fill tank.
4) The feedwater supply valve will be shut, and steam will be blown off from the steam generator to minimize the available water in the room. The steam can be blown off by operating the electromatic relief valve from the control board. The feedwater can be cut off from a valve located above the steam generator roof, accessible from the control room roof.

The question postulates the inaccessibility of the cell during a fire. However, since three routes of access are available – from the electrical room, from the IHX cell roof, and by lifting a roof slab – it is believed that the cell will be accessible in all fires that may actually occur.

In the event of a moderate pressure excursion in a cell (assumed to be due to a sodium-water reaction), the steam generator roof slabs may lift, or the side walls of the cell may fail, but the control room wall will hold. The operator will proceed as in the case of a fire.

If a large sodium-water reaction should occur (external to the steam generator or other equipment), with a great deal of hydrogen evolved, an explosion could follow in the steam generator room. With near-stoichiometric mixtures of air and hydrogen, explosion and detonation pressures ranging from 6.9 atmospheres[1] to 18 atmospheres[2] are possible. Such explosions could be sufficient to breach the wall between the steam generator room and the control room and could perhaps injure the control room operators. The primary system

(1) B. Lewis and G. von Elbe, Combustion, Flames and Explosions of Gases; (Academic Press, New York, 1951)
(2) B. Lewis and J. B. Friauf, J. Am Chem Soc 52, 3950 (1930).

and the reactor will be protected by the massive structure of the IHX cells. (See Figure 2.40 of the parent report.)

In all probability, the reactor will have been scrammed prior to the hypothesized explosion, since a major fire in the plant is reason for the shift leader to shut the reactor down. If the reactor were still operating at the time of the explosion, scram would occur through one of the following means:

a) Shock to the protective panel. The dropout of any scram relay, or the breaking of any lead, causes a scram. An explosion violent enough to breach the control room wall will shake the protective panel severely.
b) Normal protective system action, such as flow ratio scram.
c) Operator action.

Operating personnel in other parts of the plant would not be injured, and could proceed to organize the fire fighting and recovery operations. The reactor can be scrammed, if necessary, from the control rod carriage; the pumps can be shut down from the local pump panels; the main valves can be operated manually; the helium and nitrogen systems are operable from the local gage-boards; and the reactor temperatures can be monitored with temporary instruments that can be connected in the preamp room or the pullboxes near the loading face. In the worst case, if all three steam generators were damaged, the reactor heat sink would be lost. As pointed out in section 4.2.17 of the parent report, if the reactor has afterglow energy following a full year's operation, some 58 hours could elapse before sodium boiling could commence. This period would be sufficient either to get one of the steam systems back in service, or arrange emergency cooling of the reactor as discussed in 4.2.17. There is no indication that any radioactivity would be released.

Since the hypothetical accident discussed requires the mixing of quantities of sodium and water, the prevention of such mixing is important. Prompt action in fighting small fires will prevent them from becoming large ones. Since the writing of the parent report, plans have been made to construct a curb across each of the steam generator cells slightly south of column line 14. (See Figure VII(b)-1). Since virtually all the sodium system is south of this line, and all the water is north, such a curb will separate any pools of sodium and water on the floor. A light metal partial partition will be erected across each of the cells to confine water spray from any broken water line. As discussed above, the water to a cell containing a major fire will be cut off.

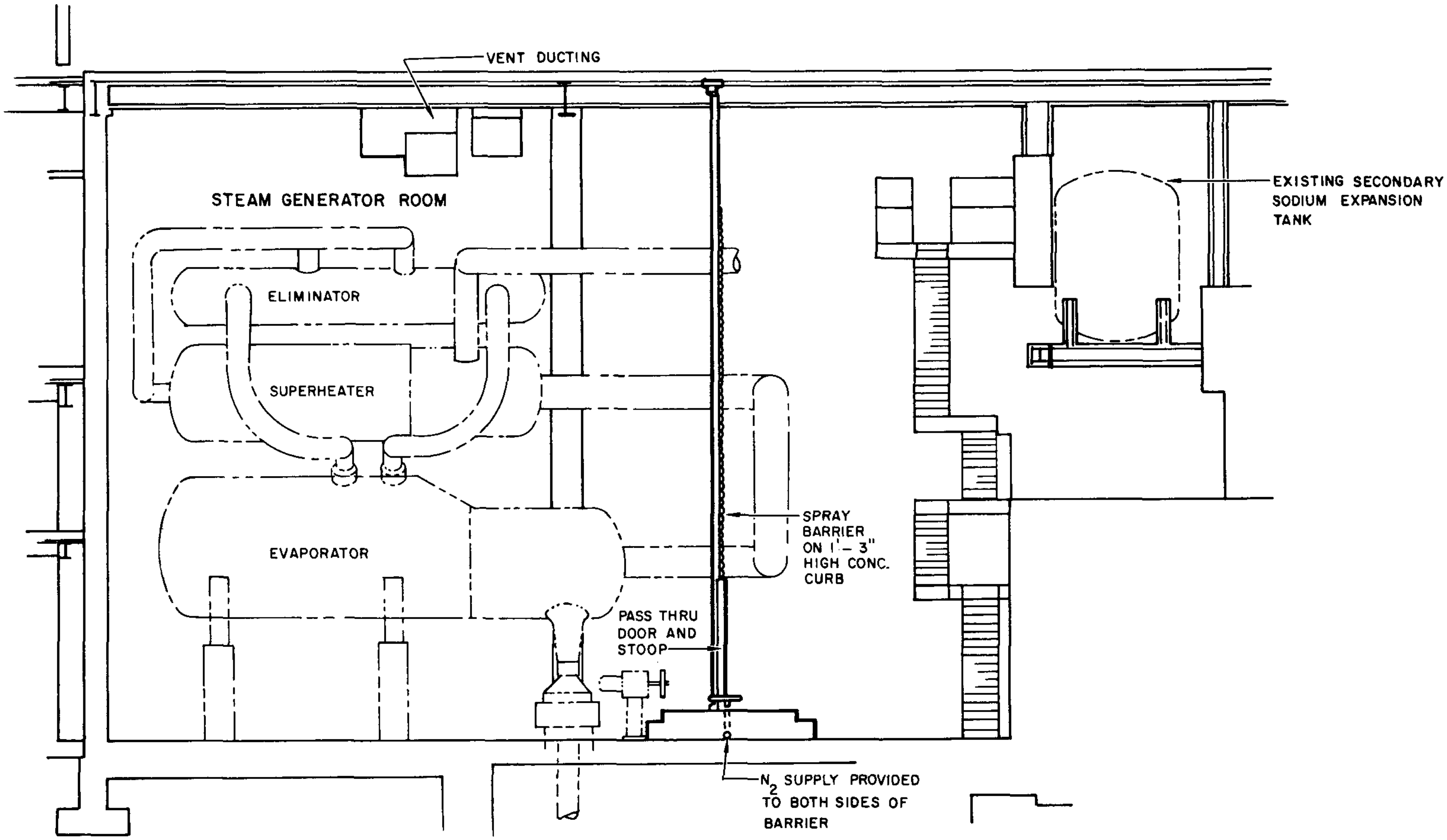

Figure VII(b)-1. Steam Generator Cells

Question VII(c): Several plant areas contain sodium and water lines in close proximity. An analysis should be performed to determine if sodium-water reactions in these areas could have unsafe effects on plant operation, and whether smoke detection and fire-fighting equipment are required.

Answer: Points or areas of the plant having sodium and water in close proximity to each other (in addition to the steam generator area) are as follows:

Area	Water Source	Sodium Source
1) Pipe chase behind wash cell, valve and gage boards.	Cooling water piping.	Sodium fill lines.
2) Pipe tunnel under steam generators. (Cell cooling equipment rooms)	Cooling water piping.	Secondary sodium piping.

Item 1 area contains only the main sodium fill line to the primary and secondary sodium systems. This line was used for initial fill of the systems. The line was then drained and valved off. Hereafter, the line will only be used for make-up after an unanticipated major loss of sodium. In this event special precautions will be taken to minimize danger of fire during its use, including the following:

1) A fireman will be standing by on the pad outside of the building with full equipment during pumping operations.
2) All joints will be inspected at start of pumping operations for signs of leakage.
3) Pyr-A-Larm alarms are installed in the area and will be operating during the sodium unloading process.
4) Sodium will not be pumped if water is present on the floor of the area.

These precautions are equal to, or in excess of similar precautions taken in commercial sodium handling installations. Since it is unlikely that the transfer will be made again and, if done, close surveillance will be exercised, further precautions are not considered warranted.

Item 2 area contains the main secondary sodium piping; sodium service piping; secondary sodium service facilities; and nitrogen cooling units, together with their water supply and return piping. Protective measures available are as follows:

1) Fire protective devices in the area are:

a) A Pyr-A-Larm is installed in each nitrogen cooler enclosure.

b) A Pyr-A-Larm is installed in stairwell No. 1 leading into the area.

c) Two Pyr-A-Larms are installed in the secondary sodium pipeway under the steam generator rooms.

d) Four HAD rate of rise and fixed temperature alarms are installed in the secondary sodium storage room.

e) A smoke detector is provided in the vent duct leading from the area.

f) Doors are provided on the stairwell to control flow of air to the area in event of fire.

g) Fire fighting equipment is provided in the area as follows:

Area	Equipment
Nitrogen Cooling Unit Area, EL 1420	1 20-lb CO_2 Extinguisher
	1 350-lb MET-L-X Wheeled Cart
	1 1500-lb Ca CO_3 Cart
	2 Sets Turnout Clothing
	2 Breathing Apparatus
	1 Emergency Locker
Secondary Fill Tank Area	2 30-lb MET-L-X Hand Extinguishers
	1 25-lb ABC Hand Extinguisher
Stairway No. 1, EL 1420	1 20-lb CO_2 Hand Extinguisher
	1 150-lb MET-L-X Wheeled Extinguisher

The area is, in general, reinforced concrete construction, with a four (4) hour fire rating. Should a sodium fire start in the area, its presence would be signalled almost immediately by the Pyr-A-Larm system, or by smoke detectors in the ventilation ducts.

In addition to the above-mentioned design features, the following modifications are being made to further minimize and limit any potential fire hazard:

a) All water lines and valves in the sodium containing areas will be provided with steel or concrete secondary containment enclosures (probably four), meeting the following criteria:

Sodium leakage, even a large high pressure leak, cannot spray onto water piping. Water leakage cannot spray onto sodium piping or insulation.

Any water leakage would be contained within the secondary water enclosure barriers, and would not spread on the floors in areas containing sodium piping.

Liquid water in the enclosure would actuate alarms and automatic shut-off [See (b) below], thus limiting quantity of water that can collect in the enclosures.

b) Automatic shut-off valves are being provided in the main water supply and return lines in the stairwell. In event of water leakage these valves would close and would prevent further water flow into the area. It is estimated that a maximum of 200 gal of water would be available in one of the enclosures after valve closure, for potential contact with leaking sodium.

c) Curtain walls at column line C_y, and at column line 15, will be modified so that solid concrete basins are formed of the IHX cooler area; the secondary sodium storage area; and stairwell No. 1 area. Curtain walls will be made splashproof. Thus, any potential sodium leakage will be confined to one of the first two areas, and any water leakage will be confined to the stairwell.

d) Atmospheric cooler 11 will be removed from the sodium storage tank area and will be reinstalled in the stairwell south of column line 15.

The extent and ramifications of a sodium-water reaction in this area will depend on circumstances, and must be discussed in this light.

a) The most likely occurrence will be a water leak, through valve packing, etc. In all probability, such a leak would be small enough that no damage will ensue, and repairs will be made without interrupting plant operations. Routine operator patrol (one per shift) will provide ample warning for small leaks. Detectors in each basin will cause automatic water shut-off and alarm for large leaks.

b) A less likely occurrence would be a sodium leak. The sodium would oxidize on exposure to air, and sodium oxide fumes would signal trouble through the Pyr-A-Larm system. The leak would be fought as a sodium fire, utilizing equipment in this area. In all probability, the affected loop would be shut down and drained until repairs could be made. Reactor shut-down probably would not be mandatory.

c) A second order probability would be a sodium leak coincident with a water leak. The water supply would be shut off automatically, and the operator would receive a signal that the valve was closed. Water would be contained in the secondary water piping enclosure, and sodium would be contained in one of the two sodium containing rooms. The sodium would catch fire and would be treated as a sodium fire. No major consequences would result, other than minor damage to insulation. The leaking sodium loop would be shut down. If water supply to the other coolers could not be restarted immediately, the reactor would be shut down and started again when cooling is available.

d) In event of concurrent sodium and water leakage, during which the two fluids contact each other, a violent chemical reaction may take place in which hydrogen is evolved. Heat of reaction will provide a source of ignition, and the hydrogen will burn in a crackling manner with a series of small localized explosions. As long as the sodium or water feed to the fire is limited in rate, and as long as air is available for oxidation of the hydrogen, no great damage should take place. The fire will be fought as a sodium fire. Every effort will be made to shut off flow of sodium, or water, or both, to limit spread of the fire. Local electrical and control wires may be damaged, necessitating shut-down for inspection and repair. The leaking sodium loop will be shut down and drained as soon as possible to minimize further sodium leakage. Except in the case of a very limited fire, the reactor will be shut down and secured.

During a sodium fire it is possible that the barrier with the IHX cells would be ruptured, either by failure of the nitrogen coolers, or by failure of the pipe-way seals, or both. It is not expected that any radioactivity would be released. Hydrogen build-up or explosion is not possible in the IHX cells due to the presence of nitrogen in the cells, coupled with the continuous make-up supply of nitrogen to the cell. In event of such a fire, it is unlikely that the control room or reactor controls would be damaged, and operation could be resumed as soon as repairs were made to the affected components.

From the above discussion, it is seen that appreciable damage cannot ensue from leakage of either water or sodium. Major leakage of either fluid will be signalled in the control room almost immediately. Minor leakage will be detected by routing operator patrol. A simultaneous third order failure of sodium lines, water lines, and secondary water containment barriers is required before a sodium-water reaction can take place. In event of such failure, the extent of the conflagration will be limited by automatic valves which limit the possible amount of water present. The secondary containment barrier of the IHX cells would conceivably be breached in a fire. However, radioactive material would not be released unless another, or fourth order, failure is postulated.

Question VII(d): What are the consequences of a hydrogen explosion in the wash cell decay tank?

Answer: The range of consequences of such a hydrogen-oxygen explosion are listed below:

Oxygen in Tank	lb Na Reacted With H_2O in Wash Cell	Pressure		Consequence
		Peak	Reinforced	
< 5.4%	Any	No explosion		-
≥ 8.6%	20 lb	460 psi	920 psi	Marginal; tank may not burst.
~ 21%	≥ 47 lb	834 psi	1670 psi	Tank bursts.

If the tank bursts, it is possible that it may damage piping on the adjacent No. 1 gas decay tank. It is unlikely that the other two gas decay tanks or the liquid waste storage tanks would be damaged because of the physical arrangement of the tanks in the vault (see Figure 2.44 in the parent report). It is doubtful that the vault would be breached because of its massive construction and the low pressure anticipated. The consequences of gas release from the decay tanks, either into the vault or the upper floor of the waste building, are discussed in the answer to Question VI(d).

Discussion:

Limits for combustion and detonation are as follows (Reference, Lewis and Von Elbe, "Combustions, Flames, and Explosions of Gases"):

	Percent Fuel		Corresponding Percent Oxygen	
	Lower	Upper	Upper	Lower
Limits of Flammability:				
H_2 - O_2	4.65	93.9	95.35	6.1
H_2 - Air	4	74.2	20.16	5.41
Limits of Detonation:				
H_2 - O_2	15	90	85	10
H_2 - Air	18.3	59	17.2	8.61

Note: The above figures are based on dry air. Presence of moisture tends to suppress the reaction and narrow the limits shown.

From the above table, it is seen that the oxygen content of the gas must exceed ~5.4% before combustion of any kind can take place. This means that the tank must contain approximately 25% air, along with the hydrogen and inert gases present, before combustion could possibly take place. The mixture must contain approximately 8.61% oxygen or approximately 41% air, before a detonation type reaction is possible. This quantity of air should not be present in the system, since standard procedure requires that the entire wash cell vent system be purged down to 1% oxygen, both before and after use.

Even if the reactants are present, the chances of an explosion are still small, since there is no source of spark or ignition present. Due to the high moisture content of the gas, generation of a spark by electrostatic mechanism is not a likely possibility.

In order to study the effects of a postulated explosion, two cases were analyzed:

Case 1 Twenty fuel elements, each bearing one pound of sodium, are washed, and the effluent gases, together with an average of 30 ft^3 inert gas per element, are introduced into the storage tank. The tank had been improperly purged, and a minimum of 8.6% oxygen had been left in the tank, providing sufficient oxygen to react with the hydrogen. A source of ignition was assumed.

Case 2 The decay tank had not been purged at all, but was initially full of air. Hydrogen equivalent to 47 lb of sodium is introduced without any inert gas. A source of ignition was again assumed.

Under Case 1, the peak pressure wave was calculated to be 460 psig. Under Case 2, the peak pressure wave was calculated to be 834 psig. In either of the above cases, there is a possibility of a reinforced pressure wave at some point in the system. It is estimated that this reinforced wave could be approximately 2 times the calculated pressure wave, or 920 psig in Case 1, and 1670 psig in Case 2. This estimate is based on a parallel plate analogy of a reinforced wave.

The tank is designed for 175 psig, with an estimated bursting strength of 715 psig. Under instantaneous shock conditions, the tank would probably take approximately 35% more than this pressure, or approximately 965 psig.* Whether or not the reinforced shock waves would rupture the tank is problematical, since these reinforced waves will be localized, and will not be applied to the entire vessel, or an appreciable portion thereof, simultaneously.

*AECU-3572, "Transient Loading of Thin Walled Cylinders," by Mosier and Luker. Also, see "Strength of Materials," Part II, Third Edition, by Timoshenko.

Thus, an explosion could not take place in the tank containing less than 5% oxygen initially. If the tank contained approximately 8.6% oxygen initially, an explosion could take place within the tank, but would probably not rupture the tank. If the tank were not purged at all, but contained an initial 21% oxygen (pure air), an explosion could take place. Results of the explosion may or may not rupture the tank.

Note that it was assumed that all hydrogen necessary for complete reaction with the available oxygen was released prior to combustion. This is not entirely reasonable. In presence of a source of ignition, it is probable that a partial combustion would take place soon after reaching flammable concentrations, thus reducing the probability and intensity of a subsequent detonation. Particularly in Case 2, it was assumed that hydrogen reached approximately 29.5% prior to detonation. If a source of ignition had been present, it is probable that a series of smaller detonations would have taken place at lower hydrogen concentrations, thus reducing oxygen content and the potentiality of additional explosions.

On the assumption that conditions were right for the "ultimate explosion," as outlined in Case 2, and on the assumption that the tank ruptures, it is improbable that the vault would also rupture, since the energy of the instantaneous detonation would be partially expended in rupturing the tank. A likely point for tank rupture would be along one of the longitudinal seams, or in the vicinity of the manhole nozzle-to-shell junction.

Secondary damage could conceivably include displacement of the adjacent tank sufficiently to rupture piping connections. It is improbable that the adjacent tank would be damaged appreciably. By code calculations, the tank is suitable for a working design external pressure of about 58 psia. Assuming a 4:1 safety factor, the adjacent tank should not fail since a pressure of about 230 psi would be required to precipitate such failure. It is highly unlikely that the other tanks or piping in the vault would be damaged due to their location. In particular, the liquid waste tanks are partially shielded by the shielding walls surrounding the valve room.

Assuming that both the wash cell decay tank and piping on the first adjacent gas decay tank were damaged, releasing their content to the vault, the pressure of the combustion products together with the content of an adjacent is approximately design vault pressure (10 psi).

No Na_2O dust would be released since wash cell effluent is water washed twice before arriving in tanks, and is also filtered. Dry radioactive vent gases are also filtered, in absolute filter media, prior to comparison. Therefore, sodium oxide activity, either from the wash cell decay tank, or released from adjacent tanks should be minimal. Their content would be contained within the vault and would escape slowly to the main building stack [see Question VI(d)].

Question IX(a): Discuss the diesel generator with respect to starting reliability and its capability to operate unattended for an extended period of time such as might be required if the station were evacuated?

Answer: The starting reliability of the diesel generator is strongly influenced by (1) the selection and arrangement of the equipment, and (2) the maintenance and operating procedures followed. The latter is particularly important with standby equipment such as the HNPF diesel.

Key points in the equipment selection and arrangement are:

1) The diesel-generator selected is a quality product, with years of experience behind the design.

2) Provisions have been made to keep the unit warm during idle periods by means of thermostatically-controlled electric heaters (a low-temperature alarm in the control room will alert the operator should the heat go off for any reason).

3) An elevated day-tank is provided to ensure that the injectors are properly flooded.

4) Dual starting equipment is provided (its operation is discussed below).

5) The diesel is located inside, protected from the weather.

The diesel engine is started by means of compressed air. The unit is supplied with two air receivers, one motor-driven air compressor, one gasoline-driven air compressor and the necessary auxiliary equipment. The motor-driven air compressor is automatically controlled to maintain air pressure. The gasoline-driven air compressor is manually controlled. Each air receiver has capacity for five starts without recharging. The automatic control signals the engine to start and the automatic start system makes at least five successive tries before it shuts down.

Since the time of its installation and checkout, the diesel has been started about 30 times. Three failures to start were observed, and each was found to be due to low cylinder block temperature. The cylinder block thermostat was reset, and provisions made to equip the system with a low temperature alarm to alert the control room operator. On at least two occasions recently when the station reserve power supply went off (due to lightning) the diesel started up automatically and picked up the load in $\sim$7 sec.

In regard to maintenance and operating procedure, the key item is that the diesel will be started and run sufficiently to warm it up at least once each week. In addition, if there is any expectation of needing emergency power (for example, see Question IX(e) re tornado alert) the diesel will be checked to see that starting air is at pressure, and the day tank is full.

Once the diesel is started it will run until (a) return of normal power and manual operation of the main breakers signals the unit to shut down; (b) it is manually shut down; (c) it runs out of fuel; or (d) malfunction devices trip the unit off the line. Items (a) and (b) are, of course, normal operation; (c) and (d) are most important in the consideration of ability to run during evacuation periods.

The day tank has a capacity of 150 gal of fuel oil, kept filled by an automatic pump with a capacity of about 450 gph. The outside storage tank has a capacity of 100 ft^3, or ~750 gal. It is estimated that a total minimum of about 750 gal fuel oil will be on hand at all times. Consumption of the diesel is estimated at 32 gph for the first five hours after shutdown (100% capacity) and about 20 gph thereafter as the operator programs emergency load. On this basis, there is sufficient fuel to last about 35 hours. If the station is evacuated, the emergency power system will be left with the minimum load corresponding to 20 gph fuel consumption, thus there is sufficient fuel to last about 38 hours.

The automatic shutdown devices supplied with the diesel include

a) Coolant high temperature	- switch closes at 205°F (normal is 165 - 185°F)
b) Low lube oil pressure	- switch closes at 10 psi (normal is >20 psi)
c) Engine overspeed	- switch closes at 1010 rpm (normal is 900 rpm)
d) Generator short circuit	- 1600 amp circuit breaker trips at 800 amp, with inverse time-current characteristics set at maximum time delay (normal full load is 600 amp)

These shutdown devices are set sufficiently high that the diesel will lug along under adverse conditions. Anything less than a direct short circuit will not trip out the circuit breaker on overcurrent. The emergency loads are primarily steady loads, with the exception of some dry scrubber auxiliaries (about 25 - 30 hp), the emergency air compressor (about 40 hp) and the diesel fuel transfer pump (about 1/8 hp), so that starting transients are well within the capacity of the diesel generator. The total load under emergency (evacuated) conditions is predicted to about 100 kva, compared to the diesel's capacity of 500 kva continuously.

Question IX(b): What provisions have been made to insure the uninterrupted operation of vital components served by the emergency bus, should access to switch gear be impossible?

Answer: Uninterrupted operation of vital components is assured through a combination of (a) automatic switching; (b) maintained-closed switches; and (c) remote switching, such that direct access to the switchgear is not required. Even if access to the entire station is impossible, the required loads will remain in service. In order to clarify the foregoing statements, three cases will be discussed:

Case 1: Normal (4160 volt) power lost, entire plant accessible (provides the background for understanding of the system)

Case 2: Normal power lost, switchgear inaccessible

Case 3: Entire plant evacuated.

Case 1: Normal power lost, plant accessible

a) Automatic Operation

1) Main pumps trip off, reactor scrams, convection control initiated

2) All 480-volt circuit breakers trip off on under voltage except the breaker feeding control center A3 (as shown in Figure 2.31 of the parent report) and the breaker feeding instrument air compressors (if previously closed).

3) 480-volt circuit breakers feeding equipment heating control centers are locked out.

4) Diesel generator starts, and its circuit breaker closes when the generator voltage and frequency comes up to operating levels.

5) Critical panels 1 and 2 (non-noise-sensitive) transfer to the noise-sensitive bus which is fed through the batteries and M-G sets. When the diesel comes on the line, these panels automatically transfer back.

6) The air scrubber system starts, providing 7500-cfm building ventilation. The main building fans are locked out.

b) Maintained-Closed Switches

1) The circuit breaker feeding control center A-3 (the "emergency bus") remains closed, and will not trip except on short circuit.

2) All circuit breakers in control center A-3 remain closed. However, starters (e.g., R/A vent fan No. 1) will drop out and not restart, except the starter feeding the steam generator emergency feed pump.

3) If the circuit breaker feeding the emergency power to the air compressors is closed, it will not trip off on undervoltage.

c) Remote Switching

The operators will manually energize certain equipment in order to maintain conditions as nearly normal as possible. Since the problem is principally one of maintaining cell and component temperatures, the operator will have considerable room for freedom because of the large time constants involved with the large masses of concrete. A typical order of energizing equipment, and location of switching, is shown below. Large motors are started first in order to minimize voltage transients on the diesel generator.

Item	Starting Location
1) A-B tie breaker, if any equipment on B bus is required. Any item shown below is on the A bus. Where several numbers are listed, half of them refer to spare motors on the B bus.	Main board
2) Cooling tower water pump 1 or 2	Main board
3) Nitrogen cooler 5 or 6	Gage board 105 or 106
4) Instrument air compressors	Main board and CPPD facility
5) Control center A4 and A1 or B1	Main board
6) Loading face shield coolant compressor 1 or 2	Gage board 31
7) Wash cell & storage pit pump 1 or 2	Gage board 6
8) Steam generator block valves	Main board

If the outage continues it may also be necessary to add

9)	Nitrogen coolers 1, 2, 3, 4, 7, 8A or B	Gage boards 101 through 108
10)	Tank room sump pump	Gage board 45
11)	Secondary sodium throttle valve cooler 1 or 2	Gage board 37
12)	Steam generator emergency feed pump	Main board

Case 2: Normal Power Lost; Switchgear Inaccessible

A review of the previous case shows that priority items can be energized even though the switchgear room itself is not accessible. In the event that the other switching areas are not accessible, the items concerned cannot be started. These specifically include items 3, 6, 7, 9, 10, and 11 listed under Case 1 above. In the worst case, the whole facility is evacuated, and is discussed next.

Case 3: Entire Plant Evacuated

When the plant is evacuated, the following loads will be specifically checked to ensure that they are energized (see Question IX(d) in Supplement II to the parent report).

a) Steam Generator Emergency Feed Pump

b) Emergency Feed to Instrument Air

c) Scrubber (if there is a fire).

Unless power from the reserve station transformer is off at the time the building is evacuated, those items left running will continue to run under automatic control (nitrogen coolers, loading face coolant compressor, etc.) If 4160-volt power is lost, the diesel will start automatically. The instrument air and scrubber feeds will supply power to these units as soon as the generator comes up to voltage and frequency, as there is no undervoltage trip on the feed breakers. Other important feeds (battery chargers, critical transformers, and diesel generator auxiliaries) will automatically receive power from the vital bus (CCA-3). The steam generator will continue to run since it has a maintained-contact start switch.

MAIN HEAT TRANSFER SYSTEM, AI-P-1103

Section 1

Test Purpose:

The purpose of this test is:

a) To make a final verification that the main sodium heat transfer systems, after preheating are in proper condition to receive sodium.

b) To verify that no binding, insulation damage, or other malfunction of the primary and secondary sodium systems have resulted from preheating.

Test Method: The Preheating System, AI-P-1116, preoperational test must have been completed. The main sodium heat transfer system will be visually inspected after preheating to 350°F. The piping, pipe hangers and valve operators will be checked for binding and interferences. Any gaps in the pipe insulation will be noted. Proper valve operation will be verified by opening and closing each valve from its normal control point. Where remote valve operators are involved, the proper valve will be observed to verify that it is, in fact, actuated by its corresponding valve operator.

Precautions:

a) Do not step on thermocouple or heater wiring.

b) A vessel entry permit must be obtained before entering any of the primary sodium pipe galleries.

c) Do not operate the sodium pumps without sodium in the system.

Test Completion Criteria:

a) No binding or interference of pipes, pipe hangers and valve operators exists at preheat temperatures.

Question XI(a): For each facility system test provide the criteria which have been established for ascertaining acceptable completion of the test.

Answer: Summaries of preoperational tests, including a statement of test purpose, test method, precautions, and test completion criteria, are attached. The preoperational tests are mainly concerned with checking the preparedness for operation of completed systems. These tests determine the operability of a completed system and include facility systems tests, such as those for process systems and instrument systems. For convenience of planning and test manual preparation, the nuclear tests of dry and wet criticality have been included in the preoperational test program.

Question IX(c): What communication services are available in the event of a power failure?

Answer: In the event of a power failure the following means of communication are available:

1) The public address system. Power for this system is obtained from the Critical Control Panel which is fed by the motor generator set and batteries or by the diesel generator.
2) The plug-in telephone system. Power for this system is also obtained from the Critical Control Panel.
3) Two land lines to Lincoln, Bell system
4) One microwave setup to Lincoln, Bell system
5) One short-wave setup to Lincoln
6) One power line carrier system to dispatcher in Hastings, Nebraska
7) Two short-wave hand sets with limited range; will reach Lincoln

The Bell systems have conventional Bell power supplies with no backup. The short-wave and microwave systems operate from building power with no backup. The carrier system has a d-c backup from the CPPD batteries, and the hand sets are battery powered. Additional carrier lines are going to be installed according to CPPD so there is more than one alternate route to Lincoln if 3, 4, and 5 above fail.

2. <u>Auxiliary Bay Crane</u>

   a) Move the crane to the south end of the building against the provided rail stops and install clamp-on rail stops against the wheels on the north side of the crane.

   b) Move the trolley of the crane to the west end rail stops and install clamp-on rail stops against the wheels at the east side of the trolley.

   c) Raise the hook to the up-limit stop.

   d) Tie the pendant control cables to the building structure to prevent whipping.

   e) Open the circuit breaker to the power supply.

3. <u>Fuel Handling Machine</u>

   a) Discharge any component from the fuel handling machine.

   b) Move the fuel handling gantry against the steps at the south end of the building and install temporary clamp-on rail stops against the wheels on the north side of the gantry.

   c) Move the fuel handling machine trolley to either end rail stop and install clamp-on rail stops against the wheels at the free end.

4. <u>Radioactive Waste Facility</u>

   a) Shut down the liquid waste evaporator and pull power supply breaker.

   b) Stop all discharge of liquid waste to the leaching field.

   c) Stop transfer operation of liquid waste from hold-up tanks to storage tanks.

   d) Stop any release of radioactive gas from decay tanks to atmosphere.

   e) Stop all non-critical operations which are generating excessive gas storage requirements such as purging, fuel washing, etc.

5. <u>Cooling Water System</u>

   Stop non-essential cooling loads, such as cold trap cooling (by securing cold trap operations), and personnel area coolers.

6. Auxiliary Power Equipment

Inspect the diesel equipment to ascertain that starting air is at pressure, the day tank is full, and the block is warm.

7. Observers

Post one or more observers in position to view the surroundings to advise the shift leaders of an impending tornado strike. Brief each observer with regard to the key terrain points at approximately one mile from the site. (Each observer will have direct communication to the control room.)

8. Communications

Maintain intermittent, but frequent, communication with the power dispatching authority and the weather bureau to follow the progress of the suspected tornado.

B. TORNADO SIGHTING

If the tornado comes within view, the observer(s) will inform the control room. The shift leader will announce same over the public address system, and instruct all persons, except those required at the control board or for securing the plant as described above, to take cover.

C. IMPENDING STRIKE

If the tornado comes within about one mile of the site (as identified by the preselected terrain points), and is still moving toward the plant, the observer(s) will inform the control room, then take cover. The shift leader will announce same, and instruct all persons to take cover. The shift leader (or persons designated by him) will scram the reactor, and the control room personnel will take cover.

D. DISCUSSION

The selection of one mile from the site as the scram point is based upon an average forward speed for a tornado of 40 miles per hour.* This provides about 1-1/2 min for the control room operator to scram the reactor and take cover. Even at the maximum speed listed in the reference (68 mph), the operator has nearly a minute for shutdown action.

*U.S. Weather Bureau, "Tornado Occurrences in the U.S.," Technical Paper No. 20, Revised 1960, Washington, D.C.

Question IX(e): What provisions have been made to inform operating personnel of possible tornado presence in the Hallam area? What plans have been established with respect to securing the reactor if it appears likely that a tornado will pass through or near the plant area.

Answer: The Lincoln Weather Bureau notifies Sheldon Station* of all changing weather conditions that are excessive in nature including damaging winds and large thunder storms, in addition to tornado conditions. The Strategic Air Command radar facilities are available to and used by the Lincoln station in tracking tornadoes. Lincoln also has numerous observation points reporting conditions to it.

The power dispatching authority is located in Hastings, Nebraska, and authorizes the specific power output of Sheldon Station and other power stations on a 24-hour basis. In addition to weather information supplied to him by the Weather Bureau, the dispatcher has numerous points, such as the Sheldon Station which inform him of weather conditions.

Weather instruments on site include an anemometer and recorder, barometer, relative humidity and maximum and minimum temperature gear.

The following plans have been made for securing the reactor:

A. TORNADO ALERT PROCEDURE

Upon receipt of a specific tornado warning from the weather bureau, the AI shift leader will announce same over the public address system and arrange to accomplish the following steps to minimize damage to reactor facilities:

1. High-Bay Area Crane

   a) Move the crane to the south end of the building. The crane should be parked against the rail stops provided, and temporary clamp-on rail stops installed against the wheels on the north side of the crane. This will require that the crane-gantry interlock be bypassed.

   b) Move the trolley of the crand to west end rail stops and install clamp on rail stops against the wheels at the east side of the trolley.

   c) Raise the main and auxiliary hooks into the up-limit stop.

   d) Open the ciruit breaker to the power supply.

*CPPD's combined conventional facility and nuclear facility at Hallam is called "Sheldon Station."

b) No gaps in piping insulation exist at preheat temperatures.

c) All valves operate freely in accordance with plant design at preheat temperatures."

Section 2

Test Purpose:

The purpose of this test is:

1) To verify that sodium can be circulated through the primary and secondary heat transfer systems at preheat temperature; and at design flow rates, on a continuous basis without difficulty or malfunction.

b) To reveal any obvious defect in workmanship or material that can be revealed at preheat temperature prior to operation at higher temperature with nuclear heating.

c) To determine system hydraulic characteristics at preheat temperatures for later comparative purposes.

d) To train and familiarize operators with the operation of the main sodium heat transfer system under non-nuclear preheat conditions prior to operation under more exacting conditions at a later date.

e) To verify and test the capabilities of flow and pressure controls at preheat conditions.

Test Method:

a) Dry Critical, AI-P-1155.

b) Dry Excess Loading, AI-P-1163.

c) Fill of the Secondary Sodium System, AI-P-1156.

d) Fill of the Primary Sodium System, AI-P-1157.

The basic test method will consist of the following general steps:

a) Controls will be checked and set such that those controls which are not pertinent to this test will be locked out of the system to permit the test to proceed.

b) Flow will be at a nominal 15 percent of rating at the start of the test and will then be increased to approximately 50 percent of full flow (approximately 4000 gpm). At these flows an examination will be made to determine that the entire system is operating satisfactorily. Discrepancies will be noted and, if necessary, corrected before proceeding. Flow will then be raised while adjusting the primary throttle valves as required to create the pressure drop normally resulting from the fully loaded reactor core. It is estimated that the pressure drop in the primary heat transfer system with the reactor core fully loaded and the primary pumps running at approximately 675 rpm will be about 38.4 psig. The normal heat transfer conditions will not prevail since the fuel channels will contain no fuel elements. Therefore, the main throttling valves in the primary systems will be throttled to simulate the normal core and to reach a system pressure drop of approximately 38.4 psig. The system will then be operated with this fixed throttling valve setting for the remainder of the test.

c) After setting the primary sodium throttling valves, the system should hydraulically simulate a normal operating condition of 100 percent of capacity. All instruments will be read and recorded. Flow then will be reduced by steps to the following flows, and at each step the systems will be allowed to come to equilibrium and all instruments will be read and recorded. Any discrepancies noted during the course of the test will be cause for an examination to determine if operation should be stopped and any error corrected. Steps at which equilibrium conditions will be attained and data taken are as follows:

1) 6770 gpm 100%

2) 6000 gpm 89%

3) 5500 gpm 82%

4) 4400 gpm 65%

5) 3300 gpm 49%

6) 2200 gpm 33%

7) 1100 gpm 16%

d) At the end of the data taking process, various miscellaneous tests will be made to prove out the control system, to check out various scrams, and to operate the system for a suitable period of time so that mechanical failure will be detected before wet criticality. Since the entire heat transfer complex will be at approximately preheat temperature throughout the test, full check of the temperature measuring and heat transfer characteristics will not be made.

e) Proof test data as obtained at the Atomics International Nuclear Field Laboratory will be compared with actual results of this test as performance of this procedure progresses. Refer to NAA-SR-TDR-5988, Performance of HNPF Primary Pump.

Precautions:

a) Do not operate the main pumps if either high or low sodium level alarms are signalling.

b) Do not operate main pumps if seal oil systems are not full and in operation.

c) Do not overload main pumps by pumping excessively high sodium flows. Maximum allowable sodium flow under any conditions is approximately 7200 gpm. This flow can conceivably be exceeded by mal-operation during this test, since full back pressure is not available in the core.

d) When going from manual to automatic control or vice versa, correct order of setting instruments must be followed, otherwise a bump will occur in the system.

e) Sodium circulation must be maintained in the superheater and evaporator and preheat steam must be admitted manually to maintain temperatures while sodium is in the secondary system. Otherwise, there is a possibility of freezing sodium in the tubes of the evaporator and superheater. Preheat steam may be admitted manually through the eliminator. High water may be blown down manually. Occasional manual venting of the superheater may be required.

Test Completion Criteria:

a) Each pump motor starts and comes up to rated speed (approx. 870 - 890 rpm) with no load.

b) A bumpless transfer from demand pump speed hand control station to the circuit demand steam flow hand control station can be made for each circuit.

c) Each pump can develop a sodium flow of about 2.8×10^6 lbs/hr in each circuit with a pressure drop of about 38.4 psig in the primary loops.

d) Each circuit can be bumplessly removed from and placed back into service with no adverse effect on the other two operating circuits.

e) The sodium circulation systems can be operated continuously at simulated full power flow conditions and preheat temperature for 48 hours minimum continuous run.

f) A flow decay-time relationship has been obtained for each circuit following a "scram" from simulated full power flow conditions.

g) All flow-ratio computer and controller alarms and annunciators actuate and operate as intended. Set points have been verified.

h) All sodium scram setpoints have been verified to be set as given in the test procedure by actually "scramming" the system.

i) Each pump operates without excessive vibration, noise or other indication of malfunction for extended runs, and can be "scrammed" instantaneously to no-load without distress.

j) The sodium flow, pump speed and pressure drop data requested in the test procedure have been obtained for each loop at different flows.

SODIUM SERVICE SYSTEM AI-P-1104

Section 1

Test Purpose: The purpose of this test is to determine that the Sodium Service System is adequately prepared for purging, filling, and subsequent circulating operations, and that no distress or binding has been caused by pre-heating.

Test Method: Pre-operational Test Procedure, AI-P-1116, Preheating System must have been completed insofar as it affects the primary and secondary sodium service systems.

The Sodium Service System will receive an over-all visual examination after being preheated to 350°F. All valves will be determined to operate properly by opening and closing them from their normal point of operation.

Precautions:

(a) Do not step on thermocouples or heater wiring while examining the sodium piping.

(b) Complete an HNPF vessel entry permit, as required, before going into the Sodium Service System vaults and pipe tunnels.

Test Completion Criteria:

(a) A visual inspection has revealed the absence of breaks in insulation, no binding in piping supports, no interferences exist with valve operators nor reach rods;

(b) all valves can be opened and closed and the proper valve is actuated by the proper valve operator.

Section 2

Test Purpose: The purpose of this test is to purge the Sodium Service System with helium and to verify that the sodium fill tanks and associated purification equipment contain helium atmosphere with less than 1% O_2.

Test Method: The following pre-operational test procedures as they affect the primary and secondary sodium service systems have been completed:

AI-P-1104, Sodium Service System, Section 1 (may be performed concurrently)

AI-P-1116, Preheating System (to 350°F)

AI-P-1108, Helium System

AI-P-1106, Cooling Water System (or adequate cell cooling)

The Sodium Service System will be purged with helium until the O_2 concentration is reduced to less than 1%. The purge is programmed such that, once a vessel or pipeline is purged, it will not again be contaminated.

Precautions:

(a) A continuous purge is necessary for satisfactory completion of this procedure. Care should be taken to ensure that the purge is not interrupted.

Test Completion Criteria: Each sodium line and vessel has been purged with helium until analysis of the effluent gases from each zone indicates an oxygen content of less than 1%.

Section 3

Test Purpose: The purpose of this test is to fill the sodium fill tanks and to verify the operability of the equipment which is used to perform this operation.

Test Method: The following pre-operational test procedures as they affect the primary and secondary sodium service systems **shall have been completed.**

AI-P-1116, Preheating System

AI-P-1108, Helium System

AI-P-1109, Nitrogen System

AI-P-1161, Sodium Pressure Instrumentation

AI-P-1160, Sodium Flow Instrumentation

AI-P-1154, Temperature Instrumentation Using Thermocouple Elements

AI-P-1159, Sodium Level Instrumentation (concurrent performance)

AI-P-1104, Sodium Service System, Sections 1 and 2

The sodium service system will receive an overall visual examination at 350°F in both the unfilled and filled condition.

New sodium will be received in railroad tank cars, each containing about 80,000 lb of sodium. The tank cars will be spotted on the siding next to the sodium melt station, one at a time. The portable oil heating system will be connected to the tank car and hot oil will be circulated through the tank car heating pipes to melt the sodium in the car. About 14 to 16 hours will be required to melt the sodium in one tank car.

The fill tanks and associated piping and valves will be preheated to 350°F. The transfer line will be connected to the tank car and the liquid sodium will be transferred to the fill tanks by the sodium transfer pump assisted by nitrogen pressure in the tank car. Transfer rate is about 100 gpm so that a tank car will be emptied in approximately 2 hours.

The primary fill tanks will be filled and then the secondary fill tanks will be filled. The drum melt stations will be tested by heating a drum of heat transfer oil in each station and extrapolating heat transfer data to the melting of sodium. Sodium will be back-filled into the sodium transfer tank to verify that it can be transferred to the fill tanks.

Precautions:

a) Flame-proofed protective clothing must be worn by personnel involved in the transfer operation.

b) When flexible sodium transfer line connections are being broken, have Met-L-X fire extinguisher ready for immediate use and place pan of calcium carbonate under flanges before breaking.

c) When oil lines to tank car are being broken, have CO_2 fire extinguishers ready for immediate use.

d) Do not break connections on flexible sodium transfer line during rain.

e) Do not energize the sodium transfer pump except when it is full of sodium and after it has been preheated.

f) Do not operate heat transfer oil pump empty.

g) While tank car heating unit is in operation, observe occasionally to ensure that controls are operating; that fuel is available; and that heat transfer fluid level is normal.

Test Completion Criteria:

a) Primary and secondary sodium fill and drain tanks are filled with sodium to pre-specified levels.

b) Sodium service system piping components necessary for the fill function properly.

c) There are no sodium leaks in the affected portions of the service system.

d) Simulation of melting drums of sodium in the sodium drum melt station has been accomplished, verifying proper operability of that unit.

e) Sodium can be transferred to the service systems from the sodium transfer tank.

g) There are no interferences due to expansion on loading the piping and equipment, and no discontinuities in insulation have developed.

Section 4

Test Purpose: The purpose of this test is to verify that the Sodium Service System piping can be filled with sodium from the fill tanks and that sodium can be circulated through the system plugging meters, cold traps, and carbon trap. Also, to determine the operating characteristics of the plugging meters, cold traps, and carbon trap by putting them into service using the service pumps. Sodium will be cold trapped and carbon trapped preparatory to filling the heat transfer systems. The sodium auxiliary systems will be cleaned by flushing with sodium to remove any last traces of contamination.

Test Method: Sections 1, 2, and 3 of AI-P-1104 shall have been completed before beginning Section 4. The Sodium Service System piping will be filled with sodium from the system fill tanks. Circulation will be established in all circuits using the service pumps. The plugging meters, cold traps, and carbon trap will be put into service and operated for a sufficient length of time to verify satisfactory performance, and to clean up the sodium and the Sodium Service System. Since the primary and secondary sodium service systems are completely independent of each other, testing of the two systems may be concurrent, or may be performed at different times, depending on available manpower and equipment.

Precautions:

a) Do not circulate sodium from one fill tank to another.

b) Do not operate the EM pumps at full voltage except under the following conditions:

1) The pumps have been preheated to 350 ± 50 F by racking in the main breakers.

2) Sodium has been allowed to flow to the pumps by gravity, thus priming the pumps.

3) A definite flow pattern has been established by opening appropriate valves, so that the pump will have a minimum cooling flow of about 40 gpm during all periods when the pump is operating at full voltage.

c) Do not turn on carbon trap heater until a flow of about 40 gpm has established through the carbon trap system to ensure continuous cooling of the carbon trap heater. Turn off the heater immediately on loss of flow.

d) When shutting down the carbon trap system after operation, turn off heater and <u>immediately</u> close one valve to prevent flow and excessive thermal shock in the system.

<u>Test Completion Criteria:</u>

a) The primary and secondary sodium service systems have been filled with sodium and the service systems pumps used to circulate sodium through the plugging meters, cold traps, and the primary carbon trap without malfunction.

b) Sodium service pumps operate at design conditions for extended periods of time.

c) Cold traps have been operated for extended periods of time to verify operability of the controls, and to purify sodium to a plugging temperature of less than 300°F, prior to filling the heat transfer systems.

d) The carbon trap system has been demonstrated operable.

e) Plugging meters have operated for extended periods of time. Their capability for measuring a plugging temperature, and for repeatability over extended runs has been demonstrated.

COOLING WATER SYSTEM, AI-P-1106

Section 1

Test Purpose: The purpose of this test is to fill spent fuel storage pit No. 3 and to determine that the portion of the system from the condensate supply head tank to the storage pit will operate satisfactorily.

Test Method: No prerequisite tests for this section. Spent fuel storage pit No. 3 will be filled with condensate from the condensate supply pump and supply head tank. Pump capacity, instrument settings, and rate of flow of condensate out of the head tank will be determined.

Precautions: Storage pits must be clean and free of extraneous material such as ladders, scaffolding, etc. Covers should be in place to minimize contamination. Bottom drains must be closed.

Test Completion Criteria:

a) The condensate supply pump will:

   1) Provide about 25 gpm flow under a 70 ft. head

   2) Start or stop automatically upon receipt of a signal from the condensate supply head tank low or high level switches.

b) Low level switch on condensate supply head tank will actuate pump starter when low level set point is reached and low level alarm will annunciate.

c) High level switch on condensate supply head tank will stop pump when high level set point is reached and high level alarm will annunciate.

d) Water will flow freely from storage tank to fuel storage pit No. 3 at rate of approximately 25 gpm.

e) High and low level alarms in fuel storage pit No. 3 annunciate at respective set points (el 1431 ft. and 1430 ft.)

COOLING WATER SYSTEM, AI-P-1106

Section 2

Test Purpose: The purpose of this test is to determine that the closed system serving the reactor pipe chase and pipe gallery cooling coils and the coolers associated with the Radioactive Waste Transfer System will operate satisfactorily.

Test Method: The cooling water system, Section 1 of AI-1106-must have been completed prior to the running of Section 2 of AI-P-1106.

The method to be employed in performing Section 2 of the Cooling Water System preoperational test consists of pumping water from the spent fuel pit through the system and back to the spent fuel pit. The system will be divided into two sub-systems for discussion purposes. The two sub-systems operate in parallel and must be tested in that manner.

a) Sub-system No. 1 consists of the reactor pipe chase and pipe gallery cooling coils and associated piping.

b) Subsystem No. 2 consists of all radioactive waste system coolers and associated piping.

The proper system water pH will be established.

Precautions: During the filling of the system piping, it is necessary to observe the water level in the spent fuel pit and admit sufficient makeup in the event the level drops below set low level alarm point. Lithium hydroxide should be added as required to raise pH of the closed water cooling system to approximately pH 9 to 10 during water circulating operations performed as a part of this test.

Test Completion Criteria:

a) The storage pit coolant pumps function properly and pump approximately 200 gpm each under full

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system head. Shutoff head has been determined for each pump.

b) Subsystem No. 1 was vented and filled and the flow through each coil in the pipe chase and gallery will be adjusted so that the flow through all the coils is approximately the same in each coil.

c) With coolant pump P-604 on standby and P-603 circulating, it has been verified that P-604 will start automatically when the discharge pressure on P-603 falls below the set point (10 psig) on the low pressure alarm downstream of the pumps. It has been verified that the annunciator windows on the mainboard light when this condition occurs.

d) With wash cell and storage pit coolant pump P-603 on standby and P-604 circulating, it has been verified that P-603 will start automatically when discharge pressure falls below the low pressure alarm set point and that the annunciators on the main board light when this condition occurs.

e) Flow through the demineralizers and filters has been adjusted to 10 gpm.

f) Subsystem No. 2 will be vented and filled and the flow adjusted through the service cells and equipment in the R/A Waste Bldg. to achieve the required specified flow through each. A balanced flow in Subsystem No.1 and Subsystem No. 2 will be achieved to maintain a 190 gpm flow through Subsystem No. 1 and a total flow of 320 gpm.

g) The set point on the low pressure alarm will be adjusted so that the pump on stand-by will come on the line when the system total flow reaches 220 gpm (with flow through the pipe chase and gallery maintained at 191 gpm).

h) It has been verified that the solenoid valves (upstream of the vent compressors in the R/A Waste Bldg) open automatically when the compressor drives start, thus furnishing cooling to the compressors.

i) The service cell cooling coils have been blocked out of the system and blown out with N_2.

j) The pH of the filled and circulating system is established as 9.5 to 10 after 2 or 3 days circulation. Lithium hydroxide has been added in sufficient amounts to water in fuel storage pit No. 3 to achieve approximate pH of 9 to 10.

k) The level of water in the fuel storage pit No. 3 has been adjusted to normal with the service cells blocked out of the system and the system shutdown.

COOLING WATER SYSTEM, AI-P-1106

Section 3

Test Purpose: The purpose of this test is to determine that the portion of the Cooling Water System serving the reactor building heating and ventilating coolers, nitrogen coolers and closed system heat exchangers will operate satisfactorily.

Test Method: No prerequisite tests. The method to be employed in performing Section 3 of the Cooling Water System preoperational test consists of pumping water with the cooling tower water pumps through the system coolers and back to the auxiliary cooling tower return header.

Precautions: Note differential pressure across basket strainers. Check every half hour. Change strainers if differential pressure becomes excessive (greater than 5 psi).

Test Completion Criteria:

a) The cooling tower water pumps, P-601 and P-602 will both function properly and will each pump at least 1725 gpm at 10 psi differential pressure.

b) The system has been vented and filled and flow adjusted through all coolers to obtain the required flow through each cooler.

c) The control valves on the inlet of each nitrogen cooler open when the nitrogen temperature reaches 130# and the solenoid valves on the stand-by cooler inletsopen when the nitrogen reaches 150#.

d) With pump P-601 circulating and P-602 on standby, it has been verified that P-602 starts automatically when the discharge

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pressure falls below the set point (10 psig) on the low pressure alarm. It has also been verified that the "Low Press Tower Pump" annunciator window and the "Auto Start" annunciator window on the main board light when the above condition exists.

e) With pump P-602 circulating and P-601 on stand-by, it has been verified that P-601 starts automatically when the discharge pressure falls below the set point on the low pressure alarm and that the proper annunciator windows light on the main board.

f) With either pump circulating and the other pump on stand-by, it has been verified that if the circulating pump is "tripped out" the pump on stand-by will start automatically and that the "trip" and "Auto Start" annunciator windows on the main board will light when this condition exists.

REACTOR LOADING FACE SHIELD COOLING SYSTEM, AI-P-1107

Test Purpose: The purpose of this test is to purge and fill the system with nitrogen and to determine that the system will circulate nitrogen satisfactorily.

Test Method: Cooling water system, Section 2 of 3 of AI-P-1106, must have been completed prior to preoperational test AI-P-1107. The method to be employed in performing preoperational test AI-P-1107 consists of purging and filling the system with nitrogen and testing the operation of the system at operating pressure. Before this system is purged, individual cooling coils in the reactor loading face shield will be tested with service air to assure that each line is open and that pressure drops through the cooling coils are approximately equal.

Purging will be accomplished by introducing nitrogen to the snubber tank and exhausting the atmosphere through the spare connection down stream of the loading face shield. During this purging the compressors will be started for a short time to purge the compressors and their associated piping. After the system has been purged and brought up to operating pressure, system instrumentation readings will be recorded with compressor No. 1 operating through heat exchanger No. 1, compressor No. 2, operating through heat exchanger No. 1 and compressor No. 2, operating through heat exchanger No. 2. The system will be tested for leakage by closing nitrogen supply valve and operating each compressor for a period of 2 hours, observing pressure drop/loss in the system.

High and low pressure alarms will be checked by increasing or decreasing system pressure.

Precautions:

a) The reactor loading face cooling system is to be tested in the sequence specified herein. No part of this test shall be performed until preceding numbered and lettered steps have been completed.

b) The established equipment clearance and valve tagging procedure must be observed at all times to protect personnel.

c) When purge gas is exhausted to the atmosphere in a room or operating area, adequate ventilation must be provided to insure personnel safety. Nitrogen is a non-toxic gas. However, excessive concentrations can cause asphyxiation by displacing available oxygen.

Test Completion Criteria:

a) Pressure drop through each cooling coil in the loading face shield has been determined and the necessary adjustments made in individual coil throttle valve to assure a balanced flow through the entire system of coils.

b) The system has been purged and nitrogen successfully circulated by each compressor (K-901 and K-902) for a period of five minutes each.

c) Pressure will have been raised in the system until the high pressure alarm (located at the snubber tank) annunciates at 255 psig. It will have been verified that the proper annunciator window lights (on the main board) when the high pressure alarm set point is reached.

d) After the high pressure alarm test the system relief valves will have opened and seated properly and the system checked for leaks.

e) The system will have been bled down until the low pressure alarm (located at the snubber tank) annunciates at 215 psig. It will have been verified that the proper annunciator window on the main board lights when the low pressure set point is reached.

f) The system N_2 supply will have been blocked off behind the pressure control valve and bled down until the low pressure alarm (located upstream of the PCV) annunciates when the pressure drops to 275 psig. It will have been verified that the proper annunciator window on the main board lights when the low pressure set point is reached.

g) Both N_2 compressors (K-901 and K-902) will have been operated individually and will have satisfactorily produced 235 psig at the snubber tank. Each compressor will have been checked for leakage and proper cooling of the compressor jacket in conjunction with the cooling water system.

h) The N_2 supply will have been blocked off and the system checked for leakage with the compressor running.

i) It will have been verified that low differential pressure alarm and low flow alarm will not trip when a compressor is running.

j) It will have been verified that the low differential pressure alarm and low flow alarm trips when the compressor is stopped and that the proper window (for each alarm) lights on the main board when this condition occurs.

HELIUM SYSTEM, AI-P-1108

SECTION 1

Test Purpose: The purpose of this Test is:

1) To helium pressurize the main Helium System header and to pressurize each individual helium control valve station, and

2) To demonstrate that the Helium System controls and instrumentation will function properly.

Test Method: There are no prerequisite tests for Section 1 of this test procedure. The helium cylinders will be connected to the system manifold for helium admittance to the main header to allow pressurizing of each individual helium control valve station.

The main helium gas manifold will be tested to ascertain that control pressures are maintained and that continuous service will be maintained by the "standby" manifold when the "in-service" manifold is spent. Individual helium control valve stations will be tested to ascertain that specified flows and pressures are maintained, with the exception of the Reactor, Reactor Cavity and Primary Sodium Fill Tanks control stations, which will be tested as a system. Control station relief valves will be pressurized above set-point to assure that the valves relieve at the required pressure and reseat tightly. A final check of the control stations will be made to assure that the pressure regulators do not allow helium pressure buildup under "no-flow" conditions.

The Reactor, Reactor Cavity and Primary Fill Tanks pressure controllers, together with their control valves, will be tested as a system to assure that the valves are stroking to the correct position. The Reactor and Primary Fill Tanks helium control stations will be tested for specific flow by simulating reactor or fill tanks pressure with controlled instrument air actuating the pressure controllers. The simulated pressures will be increased and the inlet control valves checked to assure that they are fully closed. Reactor Cavity helium inlet flow and pressure will be checked by controlling the downstream block valve to specified flow. High and low pressure alarms will be checked for proper operation by increasing or decreasing system pressure.

Pertinent data as called for in the test procedure will be recorded on the attached data sheets. Any deficiencies, malfunctions or general comments on the operability of the system or system components will be recorded in the "Remarks" section of the data sheets.

Precautions: When purge gas is exhausted to the atmosphere in a room or operating area, adequate ventilation must be provided to assure personnel safety.

Test Completion Criteria:

a) Helium pressure control valves will control pressure and pass the required flow within the limits prescribed in the test procedure.

b) Helium relief valves will relieve overpressure conditions at the pressures specified in the test procedure.

c) Reactor cavity, reactor atmosphere and primary sodium fill tanks pressure controllers will control pressure by admitting helium or exhausting gases to the R/A Vent System under overpressure conditions within the limits called for in the test procedure.

d) Helium pressure alarms will be actuated at the pressures specified.

HELIUM SYSTEM, AI-P-1108

SECTION 2

Test Purpose: The purpose of the Helium System Preoperational Test AI-P-1108, Section 2, is to evacuate and purge out the Helium Third Fluid Monitoring System of each of the three steam generators to prepare them for operation, and to verify correct operability of the controls and instruments in the system.

Test Method: There are no prerequisite tests for Section 2 of this test. The steam generator third fluid system helium supply cylinders will be connected up to the supply header. The third fluid chambers of each evaporator and superheater will be filled with helium. The third fluid chambers will be pressurized and appropriate valves closed to prepare the steam generator third fluid systems for startup. The high and low pressure alarms will also be checked and set points verified for operation.

Precautions: When helium is vented to the atmosphere in the steam generator vaults, adequate ventilation must be provided to assure personnel safety.

Test Completion Criteria:

a) The third fluid chamber of each component is filled with helium to the pressure specified (300 psi)

b) The third fluid system high and low pressure alarms are set and function properly.

c) System pressure holds constant with supply valves shut-off, indicating no leakage into or out of the system.

NITROGEN SYSTEM, AI-P-1109

Section 1

Test Purpose: The purpose of this test is to purge and fill the nitrogen system (except for the nitrogen filled cells), and to determine that the Nitrogen System components will operate satisfactorily in the system. Section 2 of this procedure tests the Nitrogen System associated with the primary sodium equipment cells.

Test Method: No prerequisite tests are required to perform this section of the test. The method to be employed in performing Section 1 of the Nitrogen System Preoperational Test consists of purging the system with nitrogen and testing the operation of the system components. At the start of the test, all manual valves involved in the nitrogen system are closed. The nitrogen evaporator will be connected and portions of the system will be systematically opened, purged and tested for performance.

Individual nitrogen control stations will be tested to ascertain that specified pressure and flow conditions can be met. Control station relief valves will be pressure lifted to check set pressures, then pressurized below set-points and checked to assure that the valve reseats tightly. A final check of control stations will be made to assure that pressure control valve does not allow pressure buildup under "low flow" conditions. Low pressure alarms will be checked for proper operation and correct set-point by bleeding down pressure in the system.

Precautions:

a) The established equipment clearance and valve tagging procedure must be observed at all times to protect personnel.

b) When purge gas is exhausted to the atmosphere in a room or operating area, adequate ventilation must be provided to assure personnel safety. Nitrogen is a non-toxic gas. However, excessive concentrations can exclude oxygen and cause asphyxiation.

c) Valves in Nitrogen System lines supplying nitrogen to primary sodium equipment cells must remain closed during Section 1 of this test. (Cell nitrogen exhaust systems will not be in operation at this time, and admitting nitrogen could result in an overpressure in the cell.)

Test Completion Criteria:

a) The Nitrogen Distribution System has been purged and filled.

b) All pressure control valves control pressure properly under specified flow conditions.

c) All low pressure alarms (PAL) alarm and reset properly.

d) All relief valves lift and reseat properly.

e) No detectable leaks are present based on short time decay tests.

f) Pressure gauges and flow meters function in accordance with their design parameters.

Section 2

Test Purpose: The purpose of this test is:

1) To purge and fill the primary sodium equipment cells with nitrogen.

2) To determine that the Nitrogen System supplying and venting the primary sodium equipment cells will operate satisfactorily.

Test Method: The following tests must be performed prior to running this section of the nitrogen test so that the cells can be purged and left closed and sealed:

a) Helium System, AI-P-1108 (partial)

b) Preheating System, AI-P-1116 (partial)

c) Lights, Power and Communications System, AI-P-1130 (partial)

d) Radioactive Vent System, AI-P-1110 (partial)

e) Nitrogen System, Section 1, AI-P-1109

f) Hot Sodium Circulation, AI-P-1167

g) Cooling Water System, AI-P-1106 (partial)

The method to be employed in performing Section 2 of the Nitrogen System preoperational test consists of purging the primary sodium equipment cells and pipe tunnel with nitrogen and testing the operation of the cell inlet and vent controls under operating conditions. At the start of this test, all manual valves involved in this section of the test are closed. Nitrogen supply to other parts of the system (tested in Section 1 of this preoperational test) will be temporarily shut off to assure accurate flow control during purging.

Cell relief valves will be pressurized to lift points and then checked to assure that they reseat tightly before nitrogen is admitted to the cell.

Each cell pressure controller together with its control valves will be tested as a system to assure that the valves are stroking to the correct position. Each cell will be purged with nitrogen until the cell atmosphere contains 1% or less of oxygen by volume, as indicated on oxygen analyzer O_2An-900. Cell pressure control systems will then be tested for set-point operation by manually pressurizing and depressurizing the cells using control valve by-passes until proper control valves open or close. Sequential operation of the oxygen analyzer will be tested.

Precautions:

a) During and after purging, no one should be permitted to enter the cells containing a nitrogen atmosphere. If it is necessary to enter a cell, the following precautions must be observed: The cell shall either be purged with air and tested to assure that its oxygen content is sufficient to sustain personnel without additional oxygen supply, or personnel entering the cell shall wear their own breathing apparatus. In any event, an appropriate vessel entry permit must be completed prior to entrance.

b) Nitrogen supplies to all other parts of the Nitrogen System (tested in Section 1 of this preoperational test) are to remain shut off during this test to assure accurate flow control during purging of the cells.

Test Completion Criteria:

a) All primary sodium equipment cells have been purged with nitrogen and contain 1% or less of oxygen by volume as ascertained by test.

b) All cell pressure relief valves lift and reseat properly at design set points.

c) Pressure control valves to each cell maintain the cell pressure between approximately 1 and 6 inches of water, exhausting nitrogen to the R/A vent system when cell pressure increases above 6 inches H_2O and admitting nitrogen when cell pressure falls below 1 inch of water.

d) The oxygen analyzer properly samples the atmosphere of each cell in sequence and an alarm occurs if the oxygen content exceeds 1%.

RADIOACTIVE VENT SYSTEM, AI-P-1110

Section 1

Test Purpose: The purpose of this test is to verify that instruments and controls in the Radioactive Vent System are properly set, and are functioning in accordance with plant design.

Test Method: Electrical power, instrument air, and cooling water must be available to the R/A Vent System prior to performing this test. These services are tested as a part of AI-P-1130, AI-P-1117, and AI-P-1106, respectively.

The R/A vent compressor controls, switches, timers, and starters will be verified operable with the motor leads disconnected.

Pressure instruments will be actuated by instrument air using a pressure regulator, and settings will be determined and adjusted as necessary.

Level alarms will be actuated by using auxiliary apparatus to raise and lower water level in the float cage.

Solenoid valves will be actuated to stroke the air operated control valves in the proper direction.

Pressure regulators will be tested by applying instrument air and checking the downstream pressure setting.

Relief valves will be pressurized and listed to verify that the set points are correct and then tested to assure that they reseat tightly.

Precautions: No special precautions.

Test Completion Criteria:

a) The radioactive vent compressors can be started and stopped manually and automatically, as verified by imposing synthetic signals on all automatic control devices.

b) R/A Vent System alarms actuate at the proper set point.

c) R/A Vent System pressure control valves maintain the required pressures in their respective headers.

d) R/A Vent System relief valves lift at the correct pressures and reseat tightly.

RADIOACTIVE VENT SYSTEM, AI-P-1110

Section 2

Test Purpose: The purpose of this is to verify that the Radioactive Vent System compressors and storage tanks are properly prepared for normal system operation.

Test Method: Electrical power, instrument air, and cooling water must be available to the R/A Vent System prior to performing this test. These services are tested as a part of AI-P-1130, AI-P-1117, and AI-P-1106, respectively.

The dry gas vent system header will be supplied with instrument air through a flow meter. The vent compressors will pump the air into one gas decay tank. One R/A vent fan will be run and the gas decay tank pressure released to the stack.

Vent header air will then be directed through the bypass line to the stack and the fuel handling cells vacuum pump will be operated.

The second R/A vent fan will be operated, then shut off for check of system instruments.

During the system test, all instruments and controls will be tested for proper operation.

Precautions:

a) Compressors No. 1 and 2 must have oil at the proper level in both the crank-case and lubricator prior to operating.

b) Drainer traps DT-1001, DT-1002, DT-1003 and DT-1004 must be primed prior to being put into normal operating service.

Test Completion Criteria:

a) The level switches associated with R/A vent compressors Nos. 3 and 4 actuate at the proper set point.

b) R/A vent compressors Nos. 1, 2, 3, and 4 start and stop automatically at the prescribed pressures.

c) R/A vent compressors will compress gas at their design pressure, temperature, and flow conditions without distress or malfunction.

d) R/A vent fans each control back pressure within design limits at flows up to 100 SCFM.

e) The fuel handling cells vacuum pump operates properly at design pressure/temperature conditions in conjunction with the R/A vent fans.

f) No detectable leaks exist in the system.

RADIOACTIVE VENT SYSTEM, AI-P-1110

Section 3

Test Purpose: The purpose of this test is to verify the preparedness for operation of the wet gas portion of the Radioactive Vent System.

Test Method: Electrical power, instrument air, and cooling water must be available to the R/A Vent System prior to performing this test. These services are tested as a part of AI-P-1130, AI-P-1117, and AI-P-1106, respectively.

The wet gas vent system will be pressurized with instrument air through a flow and pressure control apparatus. The wash cell vent compressor will pump this air through the separator receiver and into the wash cell gas decay tank. When the decay tank pressure reaches 85 psig the compressor will be shut off, and the decay tank air released to the ventilation stack.

During the system test, instruments, equipment and controls will be tested for proper operation.

Precautions:

a) Do not operate the wet-seal gas compressor until proper sealant level has been established.

b) Make-up water must be available to the wet-seal wash cell vent compressor.

Test Completion Criteria:

a) Cooling water to the separator receiver and to the wash cell vent compressor wet-seal is automatically maintained at the proper level and temperature.

b) The wash cell vent compressor is capable of pressurizing the wash cell gas decay tank to 85 psig at its design flow rate.

c) The wash cell vent compressor suction line vacuum breaker relieves at approximately -3 psig.

d) The wash cell vent compressor is capable of discharging to the stack as well as to the gas decay tank.

e) The wash cell vent compressor mechanical seal lubrication system functions satisfactorily in accordance with design parameters.

RADIOACTIVE LIQUID WASTE SYSTEM, AI-P-1111

Test Purpose: The purpose of this test is to verify that the Radioactive Liquid Waste System and its associated components function properly; adequately drain areas that are potential radioactive liquid waste collection points; and are capable of safely storing and discharging the liquid collected.

Test Method: The portions of the following tests that are associated with the Radioactive Liquid Waste System must be complete prior to performing this test:

a) Auxiliary Steam, AI-P-1113

b) Electrical Power, AI-P-1130

c) Instrument Air, AI-P-1117

The method to be employed in testing the Radioactive Liquid Waste System consists of admitting water to each of the liquid waste drains, collecting the liquid from the drains in the liquid waste monitoring tanks and pumping the liquid from these tanks to the radioactive waste facility storage tanks, using the steam jet pumps. System tank level instrumentation will be tested to assure its proper operation prior to admitting water to the tanks. Level transmitters will be actuated by varying the water level in the constant head water leg and testing level indicator and alarm operations. Radioactive Liquid Waste System drains, including floor drains, equipment drains, and cell drains, will be flushed with water to verify that the drains are capable of taking water. The level in the appropriate radioactive waste tank will be observed to ascertain that the drain is connected to the proper tank. After all drains to a given tank have been verified, the steam jet pump will be started and the liquid recirculated to mix in the tank. The liquid in the tank will then be pumped to the radioactive waste facility.

The radioactive waste facility tanks and pump will be operated to accomplish mixing of liquid in the facility tanks. The liquid waste evaporator will be put into service with the evaporator feed and waste transfer pump, and proper functioning verified. Liquid collected from the reactor building drains will ultimately be discharged to the leaching field or the truck loading line to establish that these areas are capable of normal operation.

Precautions: There are no special precautions associated with this test. However, the system may well transfer radioactive liquids under normal operating conditions and, therfore, theperformance of this test should include observing the radiological precautions and procedures to be associated with such conditions.

Test Completion Criteria:

a) All reactor building drains that flow to radioactive liquid waste tanks are free of obstructions and connect to the proper tanks.

b) System instrumentation functions properly; indicating correctly with alarms actuating at the proper set-points.

c) Liquid in the system hold-up tanks and storage tanks can be recirculated to accomplish mixing.

d) The steam transfer jets establish adequate flow and are capable of performing the required liquid transfers.

e) The liquid waste evaporator and associated feed pump function properly at the designed capacity.

f) Liquid can be discharged to the leaching field or the truck loading line as required by liquid activity levels.

AUXILIARY STEAM SYSTEM, AI-P-1113

Test Purpose: The purpose of this test is to activate the auxiliary steam system in the reactor building and the radioactive waste facility and determine that it will operate properly.

Test Method: The facility H & V system must be operable before this test can be run.

This test consists of progressively pressurizing sections of the system with auxiliary steam from the conventional plant until the entire system has been brought up to temperature and pressure. As the system is being warmed up, the performance of the condensate return units will be tested. Pressure control valves, level switches, steam traps and other components will be observed and tested for proper operation after each section of the system has been pressurized.

Precautions: No special precautions.

Test Completion Criteria:

a) The facility heating system was activated and the necessary adjustments were made to acquire the required steam flows through heating unit coils. The control valves and temperature controls of the units operate properly.

b) The system condensate return pumps operate properly. Level switches and floats in the condensate tanks and steam traps throughout the system operate properly.

c) Steam can be introduced to the cerrobend melt-out coils around the loading face shield.

d) Service steam can be introduced to the wash cells, decontamination room, laundry, maintenance cell, R/A waste facility and all the various hold up and transfer tanks.

PREHEATING SYSTEM, AI-P-1116

Test Purpose: The purpose of this test is to determine that the sodium pipe and equipment preheating system will operate satisfactorily. This procedure excludes the reactor vessel preheating system which is tested in Reactor Heaters Preoperational Test AI-P-1164.

Test Method: Prerequisite Tests: Prior to preheating the main heat transfer and sodium service system above 100°F, the following tests must be performed:

- AI-P-1104 Sodium Service System, Section 2
- AI-P-1150 Purge of Secondary Sodium System, Section 1
- AI-P-1151 Purge of Primary Sodium System, Section 1
- AI-P-1106 Cooling Water System, Section 2
- AI-P-1164 Reactor Heaters, Section 2, must be performed in conjunction with this test.

The procedure is divided into parts A and B. Part A tests the sodium service preheat system including all freeze and vapor traps. After determining that all heaters and their associated thermocouples operate correctly, all sodium service piping and equipment is brought up in temperature together at a rate such that no greater than 50°F exists between any temperature readings. Component temperatures which vary abnormally will be investigated. After reaching approximately 100, 250 and 350°F, paralleled thermocouples will be tested to determine that they are sensing the correct temperature. Upon reaching 350°F, the piping and equipment will be sufficiently preheated for filling with sodium.

Part B tests the sodium heat transfer system. After determining that all heaters and their associated thermocouples operate correctly, all sodium transfer piping and components will be brought up in temperature together at a rate not exceeding 2.8 F/Hr. and such that no greater than 50°F exists between any temperature readings. Component temperatures which vary abnormally will be investigated. After reaching approximately 100, 250 and 350F paralleled thermocouples will be tested to determine that they are sensing the correct temperature. Upon reaching 350F, the sodium heat transfer system will be sufficiently preheated for filling with sodium.

Precautions:

a) The nitrogen coolers must be available to control cell atmosphere temperature when preheating above 100F.

b) Surveilance of the rise in temperature of each component must be made to be sure that heating rates are not exceeded.

Test Completion Criteria:

a) The operability of heaters has been demonstrated on the sodium service system, the sodium main heat transfer system, and associated equipment.

b) Heater control circuits operate and maintain setpoint temperatures of associated piping and equipment

c) Sodium service system has been preheated to 350F.

d) The sodium main heat transfer system has been preheated to 350F.

SUMMARY

PLANT CONTROL SYSTEM, AI-P-1120

Test Purpose: The purpose of this test is to place each plant control subsystem into operation and verify that each is operable, prior to the reactor startup.

Test Method: AI-P-1117, Instrument Air System, must be performed prior to this test. With the main board connected to the sensing elements, valves and transmitters; voltages, instrument air supply pressures, gains, setpoints, and proportional, reset and rate bands associated with the plant control system will be adjusted as necessary and proper performance of the instruments will be verified. System calibrations will be run and system response times and instrument stability will be observed. Simulated inputs will be used as required in the above tests.

Precautions: No special precautions required.

Test Completion Criteria:

a) All plant control components calibrated to ± 1% of full scale accuracy.

b) All plant control components correctly connected to their input and output devices.

c) Response time of all subsystems is within 5 sec of actuating signal.

d) All subsystems respond to simulated inputs in a stable manner.

e) All plant control interlocks function correctly.

f) All plant control computing functions being performed to an accuracy of ± 1% of full scale.

g) Correct initial values of proportional band, reset, and rate are set into all system controllers.

PROTECTIVE SYSTEM, AI-P-1121

Test Purpose: The purpose of this test is to check out and align the plant protective panel for operation and then to system test all associated input signal circuitry, all associated alarm, setback and scram circuits and all associated startup interlock circuits.

Test Method: Prerequisite Tests:

a) Sodium Flow Instrumentation Preoperational Test Procedure AI-P-1160

b) Control Rod Drive and Actuator Assemblies. Preoperational Test Procedure AI-P-1145 must be complete before that portion of the Protective System Test dealing with reactor startup interlock circuitry can be performed.

All protective panel voltages, gains and setpoints will be set and the panel aligned for operation. With all system installations completed the protective system will be tested using signals from the sensing elements. Where this is not possible simulated signals at the sensing element will be used. All associated annunciators, setback and scram circuits will be made to operate; remote meters will be checked for accuracy; and start-up interlock circuitry will be made to function.

Precautions: No special precautions.

Test Completion Criteria:

a) Protective panel alignment, and trip point settings for alarms, setbacks and scrams have been accomplished.
b) Correct wiring of protective system sensing elements has been demonstrated.
c) Correct operation of the protective system has been demonstrated by using actual or simulated signals from the sensing elements.
d) Each sodium circuit by-pass switch is operable.
e) Correct wiring and operation of startup interlocks has been demonstrated.

WASH CELLS, AI-P-1133

Section 1

Test Purpose: The purpose of this test is to determine that the services (steam, helium, nitrogen, and vacuum) to the No. 1 Fuel Wash Cell are adequate for washing fuel elements. Also to verify that a fuel element can be installed in, properly washed, and removed from the fuel wash cell without difficulty.

Test Method: The following preoperational tests must be completed before the wash cell preoperational test, AI-P-1133, Section 1 of 4, can be run:

1) AI-P-1109, Nitrogen System, Section 1
2) AI-P-1108, Helium System, Section 1
3) AI-P-1110, R/A Vent, Section 3
4) AI-P-1111, R/A Liquid Waste (portion affecting R/A Waste Transfer tank only)
5) AI-P-1106, Cooling Water, Section 3
6) AI-P-1117, Instrument Air, part affecting wash cells.
7) AI-P-1113, Aux Steam, part affecting wash cells.

The No. 1 Fuel Wash Cell preoperational test will consist of performing a cleaning operation using a mock fuel element contaminated with a nonabrasive scouring powder (such as Bon Ami) to simulate sodium contamination.

Precautions: No special precautions are necessary.

Test Completion Criteria:

a) The wash cell was satisfactorily evacuated to approximately 1 inch Hg absolute pressure and filled with helium to approximately 3 psig.

b) Using the fuel handling machine, the wash cell plug was removed and the mock fuel element inserted in the cell. Helium atmosphere in the cell was maintained during this operation.

c) Superheated steam was satisfactorily injected into the wash cell through the steam cleaning nozzles for a period of ten minutes. It was determined that steam flow was continuous and smooth and that the drain valve did not allow the water level to reach the set point on the high alarm.

d) Demineralized water was satisfactorily added to the steam in the mixing nozzles to rinse the fuel element for a period of 3 minutes. The condensate drain valve was manually blocked and the condensate allowed to reach the high alarm set point where it was verified that the alarm annunciated.

e) Condensate was satisfactorily drained from the wash cell to the R/A Waste transfer tank and nitrogen was admitted into the wash cell until N_2 pressure reached approximately 3 psig.

f) The cell was evacuated of N_2 to approximately 1 inch Hg and filled with helium to approximately 3 psig.

g) The mock fuel element was satisfactorily removed from the wash cell and the wash cell plug inserted in the cell by the fuel handling machine while a helium atmosphere was maintained in the cell.

h) Satisfactory cleaning of the mock fuel element was verified by visual inspection at the pickup cell.

WASH CELLS, AI-P-1133,

Section 2

Test Purpose: The purpose of this test is to determine that the services (steam, helium, nitrogen, and vacuum) to the No. 2 Fuel Wash Cell are adequate for washing fuel elements. Also to verify that a fuel element can be installed in, properly washed, and removed from the fuel wash cell without difficulty.

Test Method: Same as for Wash Cell No. 1.

Precautions: No special precautions are required.

Test Completion Criteria: Same as for No. 1 wash cell.

Section 3

Test Purpose: The purpose of this test is to determine that the services (steam, helium, nitrogen, and vacuum) to the No. 3 Fuel Wash Cell are adequate for washing fuel elements. Also to verify that a fuel element can be installed in, properly washed, and removed from the fuel wash cell without difficulty.

Test Method: Same as for Wash Cell No. 1.

Precautions: No special precautions are required.

Test Completion Criteria: Same as for number 1 wash cell.

Section 4

Test Purpose: The purpose of this test is to determine that the services (steam, helium, nitrogen, and vacuum) to the pump wash cell are adequate for washing elements. Also to verify that a fuel element can be installed in, properly washed, and removed from the pump wash cell without difficulty after the cell has been converted for washing fuel elements.

Test Method: Same as for No. 1 Wash Cell.

Precautions: No special precautions are required.

Test Completion Criteria: Same as for Wash Cell No. 1

PORTABLE PURGE UNIT, AI-P-1137

Test Purpose: The purpose of this test is to determine that the portable purge unit will satisfactorily serve its purpose as a tool in purging storage cells, changing reactor shield plug quad rings, and sampling storage cell gas.

Test Method: The following preoperational tests must be completed before the Portable Purge Unit Test, AI-P-1137, can be run:

1) AI-P-1108, Helium System (partial)
2) AI-P-1110, R/A Vent (partial)

Partial here refers to that part of the listed tests concerning this test. The portable purge unit will be positioned over a storage cell containing a shield plug and dummy fuel element. A fuel handling machine index ring will have been previously set in place. The portable purge unit will be employed to raise the element, change the quad-rings on the shield plug, purge the storage cell, and lower the element back into place. The gas sampling device will be tested as a part of AI-P-1122, Radiation Detection and Monitoring System.

Precautions: During evacuation of the portable purge unit the shield plug in the storage cell must be held in place by extending the grapple head downward against the top of the plug. This will prevent the plug from being raised into the purge unit by the pressure differential across the plug created by evacuating the purge unit.

Test Completion Criteria:

a) The portable purge unit fits on the index ring properly.
b) The hydraulic hand pump operates satisfactorily and the hydraulic grapple engages the storage plug pickup adapter properly.

c) A satisfactory seal is created between the purge unit and the index ring when the load of the plug is placed on the purge unit.

d) The hydraulic pump is capable of lifting the plug to the "up" position.

e) A full vacuum can be established in the purge unit and storage cell.

f) Old quad rings can be removed from the plug and new quad rings satisfactorily installed on the plug.

g) The hydraulic pump satisfactorily lowers the plug into the cell and the grapple automatically disconnects from the pickup adapter when the plug seats in the storage cell.

DECONTAMINATION ROOM, AI-P-1138

Test Purpose: The purpose of this test is to determine that the equipment provided for decontamination of items contaminated with radioactive material functions properly.

Test Method: The following preoperational test must be completed before the Decontamination Room test, AI-P-1138, can be run:

1) AI-P-1110 R/A Vent System (partial)

2) AI-P-1113 Auxiliary Steam (partial)

3) AI-B-1111 R/A Liquid Waste System (partial)

   Partial in this case refers to those portions of preoperational tests of associated systems which deal with the equipment in the decontamination room.

The test method will consist of operating the ultrasonic cleaner, the steam cabinet and the steam cleaner and observing their performance to ensure they function properly. The remaining equipment provided as a part of the decontamination room will be adequately tested as a part of construction acceptance testing. Notation should be made as to the adequacy of supplies necessary for decontamination work.

Precautions: No special precautions.

Test Completion Criteria:

a) Ultrasonic Cleaner — operates satisfactorily as per manufacturer's instructions.

b) Steam Cabinet — operates satisfactorily as per manufacturer's instructions.

c) Steam Cleaner — operates satisfactorily for five minutes and does not leak.

d) All decontamination room equipment is properly installed and available for service.

MAINTENANCE CELL, AI-P-1139

Test Purpose: The purpose of this test is to verify that the normal maintenance cell operations can be performed efficiently and safely, and that the planned operating procedures and the tools and equipment are adequate.

Test Method: The portions of the following preoperational tests which are associated with the Maintenance Cell must be completed before the Maintenance Cell Test, AI-P-1139 can be run:

a) AI-P-1109 Nitrogen, Section 2

b) AI-P-1110 R/A Vent

c) AI-P-1106 Cooling Water System, Section 3

The maintenance cell preoperational testing will include the four following tests:

1. Functional test of all equipment.

All operations on each machine will be perfomred to assure proper response and satisfactory smooth operation. The high and low volume ventilation systems will be tested to assure specified atmospheric conditions and pressure in the cell and specified line pressures to the nitrogen operated machinery in the cell and to the access tunnel door inflatable seal.

2. In-cell fuel handling procedure test.

A mock fuel element will be disassembled and reassembled in the cell. Adequacy of tooling, equipment and method of procedure will be tested.

3. In-cell poison column handling procedure test.

A control rod poison column will be disassembled and reassembled to the control rod pull rod and plug assembly in the cell. Adequacy of tooling, equipment and methods of procedure will be tested.

4. In-cell filter exchange procedure test

All of the filters in the cell will be removed from the racks and replaced in the racks to assure adequate handling procedures and equipment.

Precautions: Care must be exercised in the operation of the specialized tools and equipment and in the handling of components in the cell to prevent damage to the tools and equipment, the cell lights and windows and the components on which the operations are being performed.

Test Completion Criteria:

a) The maintenance cell in-cell area lamps operate properly and provide adequate lighting.

b) Audio System

1. Microphone gains have been balanced
2. Volume controls operate properly
3. It was verified that sounds can be generally located in the cell by selective "on" and "off" manipulation of the microphone switches.

c) High Volume Exhaust

1. Exhaust valve opens when key switch is turned to high volume.
2. Minimum velocity of intake through maintenance cell access tunnel door is 100 fpm.
3. Exhaust valve closes when key switch is turned to low volume.

d) Cell Low Volume System

1. Pressure in cell is about -3/8" H_2O when the low volume system is operating under normal conditions.

2. The high pressure alarm annuciates when pressure reaches about -1/4" H_2O and the position switch, which actuates low volume exhaust valve, operates properly, i.e., actuates the valve thus permitting the R/A vent system to pull pressure in the cell back down to about - 3/8" H_2O. Proper lights on gage board light when annunciator alarms.

3. The low pressure alarm annunciates when pressure in the cell reaches - 1/2" H_2O and the N_2 supply valve opens allowing pressure to return to about - 3/8" H_2O.

4. The cell has been purged and oxygen content held to approximately 2 percent by volume (as per O_2 analysis) for a period of not less than 2 hours.

e) Cell Door

1. The maintenance cell access tunnel door seal was inflated and deflated satisfactorily from the maintenance cell operating area console.

2. The door was successfully retracted to the west end of the tunnel and returned to the east end by the pendant controls located in the tunnel.

3. The door shelf was raised and lowered successfully by controls on the console on the first operating level. The proper lights on the console lighted when the shelf was "up" or "down".

f) The maintenance cell in-cell hoist, crane and trolley were operated satisfactorily from all three consoles.

g) The vertical drive for the manipulator carriage was operated satisfactorily from all three consoles.

h) The vertical drive for the tool positioner carriage was operated satisfactorily from all three consoles.

i) 1. The tool slide was satisfactorily moved from side to side and "in" and "out" from all three consoles.

   2. The gripping tool was opened and closed satisfactorily from all three consoles.

j) The manipulator slide was satisfactorily rotated "CW" and "CCW" and "in" and "out" from all three consoles.

k) The Lee Manipulator has been satisfactorily operated according to the manufacturer's instructions.

l) The AMF Master Slave Manipulators have been satisfactorily operated according to the manufacturer's instructions.

m) In-Cell Fuel Handling

   1. Communications were satisfactorily maintained between the maintenance cell operator and the fuel handling machine operator.

   2. A gastight seal was established and maintained between the fuel handling machine cask and the maintenance cell and the mock fuel element positioned satisfactorily during the in-cell operations.

   3. The fuel element process tube was successfully removed from the fuel handle and stored in the fuel rack.

4) The fuel bundle was removed from the fuel element shield plug and hanger rod assembly, placed in a fuel shipping capsule and the capsule stored in the fuel rack.

5) The fuel bundle was reassembled to the fuel element shield plug and hanger rod assembly and the process tube reinstalled on the fuel bundle thus verifying the adequacy of the tools and equipment for all anticipated maintenance cell fuel handling operations.

n) In-Cell Poison Column Handling

1. The control rod poison column was successfully disassembled from the pull rod and shield plug assembly, placed in a shipping capsule and stored in the fuel rack.

2. The control rod poison column was removed from the capsule and reassembled to the pull rod and shield plug assembly thus verifying that the tools and equipment for control rod poison handling were adequate.

o) In-Cell Filter Change

1. Filters were removed from the filter bank and removed from the cell by way of the maintenance cell access tunnel door.

2. Filters were returned to the maintenance cell by the door and reinstalled in the filter bank thus verifying that the tools and equipment are adequate for filter change in the maintenance cell.

PURGE OF THE SECONDARY SODIUM SYSTEM WITH HELIUM, AI-P-1150

Section 1

Test Purpose: The purpose of this test is to establish a helium atmosphere of less than 3% oxygen content in the secondary sodium loops.

Test Method: Preoperational Test, Section 1, AI-P-1108, Helium System must have been completed before performing this test. The oxygen content of the atmosphere contained in the secondary sodium loops will be reduced to an acceptable level by employing a continuous helium purge equivalent to two loop volumes. The loops will be purged individually. Helium will be introduced at the expansion tank and the loop atmosphere samples will be analyzed periodically to determine the oxygen content.

Precautions:

a) Do not allow the helium supply to be interrupted during the purge. Any break in the continuity of the purge operation will result in additional oxygen-helium diffusion with the subsequent requirements of more helium and greater purge time.

b) There will be helium cover gas supplied to other components that must be maintained during this test. However, the flow to these components must be held to a minimum, as indicated on FI-845 and FI-846.

c) It is necessary to monitor the secondary loop pressure indicators (PI-106, 206 and 306) continuously during this procedure to ensure the loops are not pressurized above 20 psig. If this pressure is reached, immediately close helium tank outlet valve V-830 and stop the purge.

Test Completion Criteria: A helium atmosphere of less than 3% oxygen content is established in the Sodium Heat Transfer System secondary loops.

Section 2

Test Purpose: The purpose of this test is to establish a helium atmosphere of less than ½% oxygen content in the secondary sodium loops with the loops at preheat temperature.

Test Method: Section 1 of this test procedure must have been completed prior to performance of this test Section (2). The oxygen content of the atmosphere contained in the secondary sodium loops will be reduced to an acceptable level by employing a continuous helium purge. The loops will be purged individually. Helium will be introduced at the expansion tank and the loop atmosphere exhausted through the loop low point drain line. Loop atmosphere samples will be analyzed periodically to determine the oxygen content.

Precautions:

a) Do not allow the helium supply to be interruped during the purge. Any break in the continuity of the purge operation will result in additional oxygen-helium diffusion with the subsequent requirements of more helium and greater purge time.

b) There will be helium cover gas supplied to other components that must be maintained during this test. However, the flow to these components must be held to a minimum, as indicated on FI-845 and FI-846.

c) It is necessary to monitor the secondary loop pressure indicators (PI0106, 206, and 306) continously during this procedure to insure the loops are not pressurized above 20 psig. If this pressure is reached, immediately close helium tank outlet valve V-820 and stop the purge.

<u>Test Completion Criteria</u>: A helium atmosphere of less than 1/2% oxygen content is established in the Sodium Heat Transfer System secondary loops.

PURGE OF THE PRIMARY SODIUM SYSTEM AND REACTOR VESSEL WITH HELIUM, AI-P-1151

SECTION 1

Test Purpose: The purpose of this test is to establish a helium atmosphere with less than 3% oxygen content in the primary sodium loops and the reactor vessel,

Test Method: Preoperational test AI-P-1108, Section 1, Helium System, must be completed. The oxygen content will be reduced to 3% or below by employing a continuous helium purge of the loops and the reactor vessel. The three reactor outlet lines will be purged simultaneously. The reactor inlet (throttling) valves will then be opened and the reactor outlet (block) valves closed to purge the reactor inlet lines. Following the purge of the reactor inlet lines the throttling valves will be closed and the block valves reopened to purge the reactor vessel. Loop atmosphere samples will be analyzed to determine the oxygen content of the atmosphere established in the primary system.

Precautions:

a) Do not allow the helium supply to be interrupted during the purge. Any break in continuity of the purge operation will result in additional oxygen-helium diffusion with subsequent requirements of more helium and longer purge time.

b) Monitor the reactor vessel pressure indicator PI-803 (Board No. 29) continuously during this procedure to ensure the vessel is not pressurized above 2 psig. If this pressure is reached, immediately close helium tank outlet valve V-820 and stop the purge.

Test Completion Criteria: A helium atmosphere of less than 3% oxygen content is established in the Sodium Heat Transfer System primary loops and reactor vessel.

PURGE OF THE PRIMARY SODIUM SYSTEM AND REACTOR VESSEL WITH HELIUM, AI-P-1151

SECTION 2

Test Purpose: The purpose of preoperational test AI-P-1151, Section 2, is to establish a helium atmosphere with less than ½% oxygen content in the primary sodium loops and the reactor vessel, preparatory to filling the primary system with sodium.

Test Method: Section 1 of this test procedure must have been completed prior to performing this test, Section 2. The oxygen content will be reduced to less than ½% by employing a continuous helium purge of the loops and the reactor vessel. The three reactor outlet lines and the upper portion of the reactor vessel will be purged simultaneously. The reactor inlet (throttling) valves will then be opened and the reactor outlet (block) valves closed to purge the reactor inlet lines. Following the purge of the reactor inlet lines, the throttling valves will be closed and the block valves reopened to purge the reactor vessel. Helium samples will be analyzed to determine the oxygen content of the atmosphere established in the primary system. The purge method is delineated in the accompanying sketch.

Precautions:

a) Do not allow the helium supply to be interrupted during the purge. Any break in continuity of the purge operation will result in additional oxygen-helium diffusion with subsequent requirements of more helium and longer purge time.

b) It is necessary to monitor the reactor vessel pressure indicator PI-803 (Board No. 29) continuously during this procedure to ensure the vessel is not pressurized above 2 psig. If this pressure is reached, immediately close helium tank outlet valve V-820 and stop the purge.

<u>Test Completion Criteria:</u>

a) A helium atmosphere of less than ½% oxygen content is established in the Sodium Heat Transfer System primary loops and reactor vessel.

b) Reactor system pressure is being maintained in a normal manner at 1 to 6" water by the helium and vent control valves.

HALLAM NUCLEAR POWER FACILITY

PREOPERATIONAL TEST PROCEDURE

SODIUM LEAK DETECTION SYSTEM

AI-P-1152

Test Purpose: The purpose of this test is to verify that the sodium leak detection system is capable of detecting a sodium leak which occurs in the vicinity of the detector.

Test Method: Prerequisite Tests: None

The test circuitry which is provided for the leak detection system provides a very good indication of the condition of each leak detector circuit. Therefore, the procedure followed in this test will be to simulate a leak on one of the detectors by shorting out the 25K resistor mounted in the detector assembly and then adjusting the test resistors to provide the proper indication on the test meter which is common to all the detectors. After these adjustments are made, each of the leak detectors will be switched to the "test" position and the meter reading noted.

Each leak detector alarm circuit will then be tested by temporarily jumpering each detector's 25000 ohm field mounted resistor with a 400 ohm resistor.

Precautions: No special precautions.

Test Completion Criteria:

a) Each leak detector will cause a "Sodium Leak" annunciation when shorted by 400 ohms or less.

b) The leak detector circuit is correctly wired.

DRY CRITICALITY, AI-P-1155

Test Purpose: The purpose of this test is to determine the minimum critical mass of the HNPF core without sodium. The data obtained will subsequently be used to determine the reactivity worth of sodium which is normally present in the core. It is important to know the worth of sodium, so that the shutdown margin can be assured for the unlikely event of complete loss of sodium. The data obtained also will be used to correct pre-critical physics calculations.

Test Method: The following pre-operational tests must be completed prior to starting the dry critical test:

a) Instrument Air, AI-P-1117

b) Emergency Power, AI-P-1118

c) Helium System, Section 1 (Section 1 includes all of the helium system except the steam generator third-fluid systems.), AI-P-1108

d) Radiation Detection and Monitoring System, AI-P-1122

e) Radioactive Vent System, AI-P-1110 (partially completed) - all except that portion connected to the liquid waste system.

f) Electrical (lighting, power and communications systems), AI-P-1130

g) Protective System, AI-P-1121

h) Nuclear Instrumentation, AI-P-1119

i) Neutron Chambers, AI-P-1149

j) Protective System Modifications, AI-P-1166

k) Control Rod Drive and Actuator Assemblies, AI-P-1145

l) Control Rod Actuator Support Structure, AI-P-1146

m) Control Rod Operability, AI-P-1148

n) Temperature Instrumentation Using T/C Elements, AI-P-1154 (Instrumentation connected to reactor and fuel element T/C's must have been tested.

The fuel will be loaded in predetermined amounts and locations. After each group of fuel elements is loaded, the control rods will be withdrawn in increments according to the specified procedure, and counting data will be taken. Measurements will be taken before and after any control rods are moved. Several curves of inverse multiplication vs fuel loaded will be plotted after each loading step, and an extrapolation to criticality will be made before more fuel is loaded. It will then be decided if a change should be made in the loading schedule. Generally, not more than half the amount of fuel required for extrapolated criticality may be loaded at one time. However, if the extrapolated curve indicates that less than two elements are required for criticality, one element will be loaded. Following the second incremental loading the rods-in and rods-out multiplication curves will be extrapolated to predict the worth of control rods at the critical loading in terms of fuel. No subsequent fuel increment will exceed one-half of this estimated control rod worth. The expected dry critical loading is 23 fuel elements.

The counting rates on the in-core instruments (three in-core fission chambers) will be recorded after each step in the loading procedure. The multiplication, defined as the ratio of counting rate with fuel to counting rate with source only, will be calculated. A Plot of the inverse of the average multiplication vs fuel loaded will be maintained, and the loading will proceed according to the extrapolated critical loading that this average plot indicates.

The core contains 19 shim-safety rods (designated as control rods, hereafter). The control rods will be operated in five gangs, consisting of 6,6,3,3, and 1 rods in gangs 1,2,3,4, and 5, respectively. Control rods gangs 1, 2, and 3 will be withdrawn as one gang as long as this is desirable, and inverse multiplication vs fuel loaded will be determined and plotted. After determining that the core is subcritical, control rod gangs 4 and 5 will be withdrawn 50%, observing that the core remains subcritical. Inverse multiplication vs fuel loaded will again be determined and plotted. A check for criticality will again be made. Control rod gang 5 will now be fully withdrawn. Provided that the core remains subcritical, gang 4 will be withdrawn the remaining 50% and the inverse multiplication vs fuel loaded determined and plotted. It it is thought that criticality will be reached on full withdrawal of gang 4, the gang will be withdrawn incrementally, observing all nuclear instrumentation, until criticality is reached. The reactor will become critical on gang 4. This is desirable since it is expected that each rod in gang 4 will be worth less than gang 5, and thus criticality will be approached more slowly.

This test will be performed at ambient temperatures with no sodium in the core.

Precautions:

a) The poison sections of the control rods are in the fully inserted position with holding magnets de-energized during any fuel loading operations. Each time control rods are reconnected, rod operation must be checked by raising rods from 12 to 24 in., observing that snubber return light goes out, dropping all rods and observing that rod down lights indicate properly.

b) During the loading of each fuel element, the count rates of the in-core fission chambers are observed by the control room operator. Communications are maintained between the control room and the fuel handling machine or crane operator. Only personnel concerned with the operation are in the high bay area during fuel transfer.

c) None of the seven central rods (gangs, 3, 4 and 5) are removed from the core for maintenance during these tests.

d) The test is suspended if any of the following conditions occur: Failure of one or more of the control rods; failure of any of the in-core fission chambers; failure of either source range instrument; or failure of any normal or specially installed scram or setback. This test may be resumed with these malfunctions in effect with permission of the Field Superintendent provided that safety is assured and the following minimum requirements are met: at least 6 of the 7 nuclear instruments are working properly; and all prerequisite reactor scrams (except the one nuclear instrument which may not be functioning) are operative. Operations may not be continued if any of the inner 7 control rods is malfunctioning.

e) Following the loading of any group of fuel elements, the corresponding shield plugs in the loading face shield are checked with a helium leak detector to insure against any cover gas leakage.

f) No detectable nuclear heating of fuel is allowed. A 10°F rise in fuel temperature will be considered detectable on the in-fuel thermocouples. The high temperature alarm on the in-fuel thermocouples will be set no higher than 25°F above ambient temperature. This setting will allow for some changes in ambient temperature without changing the alarm setpoint. This alarm calls for immediate reactor shutdown by the operator. A 10° rise in temperature at the hottest part of a fuel element is equivalent to holding that element at 240 watts for 10 minutes.

g) The minimum intentional reactor period is 40 seconds.

h) The maximum intentional flux level will be two decades above "source critical".

i) The fuel loading procedure limits each fuel loading to one half the number of additional fuel elements required for criticality until less than two elements are loaded per step.

Test Completion Criteria:

a) The reactor is made critical with a minimum number of fuel elements in the dry core.

b) A sub-critical calibration of the central control rod has been made.

INITIAL FILL OF SECONDARY SYSTEM WITH SODIUM, AI-P-1156

Test Purpose: The purpose of this test is to fill the three secondary heat transfer loops with sodium and to verify the fill procedure.

Test Method: Prior to starting the fill operation, the system must have been purged with helium (AI-P-1150) and preheated to 350°F (AI-P-1116). Also those portions of the following tests associated with the secondary loops must have been completed.

1) Preoperational Test AI-P-1104 - Sodium Service System
2) Preoperational Test AI-P-1152 - Sodium Leak Detection
3) Preoperational Test AI-P-1154 - Temperature Instrumentation Using Thermocouple Elements
4) Preoperational Test AI-P-1159 - Sodium Level Instrumentation in part. Completion of this test is accomplished concurrently with the fill operation.
5) Preoperational Test AI-P-1160 - Sodium Flow Instrumentation
6) Preoperational Test AI-P-1161 - Sodium Pressure Instrumentation

The loops are filled from the three secondary fill tanks by use of the secondary service pump. As the system fills, the expansion tanks (the high points in the secondary loops) vent through individual vapor traps and associated relief valves to the atmosphere. Each loop is filled to about the one-half level in the expansion tank with sodium at 350°F temperature. This level wets the hydrostatic bearings of the sodium pumps. A helium atmosphere is maintained over the free surface of the expansion tanks and pumps at all times.

Precautions: Steam must be available on a continuous basis to maintain the steam generators at preheat temperatures before, during and after initial fill of the secondary sodium system. No other special precautions are required beyond those necessary for working with and around sodium and sodium components.

Test Completion Criteria:

a) The secondary sodium loops are filled with sodium to 75 ± 2 inches in the loop expansion tank at preheat temperature. (Equivalent to normal sodium charge in the systems.)

b) The sodium level, pressure, temperature, and flow instruments associated with the secondary loops fill operations function properly as evidenced by actual use.

INITIAL FILL OF PRIMARY SYSTEM WITH SODIUM, AI-P-1157

Test Purpose: The purpose of this test is to fill the primary heat transfer system with sodium and to verify the fill procedure.

Test Method: Prior to starting the fill operation the primary sodium system including the reactor vessel must be purged with helium (AI-P-1151) and preheated to 350 + or - 25°F (AI-P-1164 and AI-P-1116) and the sodium in the fill tanks must have been trapped for carbon and oxides. Those portions of the following tests associated with the Sodium Heat Transfer System primary loops and reactor vessel also must have been completed:

1) Preoperational test AI-P-1152 - Sodium Leak Detection Instrumentation.
2) Preoperational test AI-P-1159 - Sodium Level Instrumentation in part. This test to be completed concurrently with the fill operation.
3) Preoperational test AI-P-1160 - Sodium Flow Instrumentation.
4) Preoperational test AI-P-1161 - Sodium Pressure Instrumentation.
5) Preoperational test AI-P-1104 - Sodium Service System.

The system is filled from four of the five primary fill tanks by a combination of gravity flow and the use of the electromagnetic sodium service pumps. The method to be used in filling the system is as follows:

A. The reactor vessel will be filled to low operating level with three blocking and three throttling valves open, and with the vent lines from the reactor, pumps, IHX freeze traps and the fill tanks open. This step will fill the pumps, pump suction lines, and the return lines from the IHX's.

B. The IHX's will be filled by pumping sodium into the return line upstream of the throttling valve with the throttling valves closed and the blocking valves open. During this step, the reactor level indicators will be utilized to determine when the IHX's are full, since the core level will be raised by overflow from the IHX's.

C. To fill the pump discharge lines, the blocking valves will be closed, the pump vent lines will be valved off, and the pump casings will be pressurized with helium while pumping sodium into the loop at the service connection on the reactor outlet line between the blocking valve and the pumps. The helium pressure will be controlled to maintain the sodium level in the pump between the level alarms, and will vary from reactor cover gas pressure to approximately 9 psi higher. Once the line is filled, as indicated by the increased pump discharge pressure and increased level in the pump casing, the R/A vent and helium lines to the freeze traps servicing the high points on the IHX's will be closed to prevent unsealing of the trap, the pump casing pressure will be bled to the same value as the core cover gas, and the blocking and throttling valves will be opened. The main pumps will then be started and the sodium will be circulated to remove any trapped gas.

Precautions:

a) All rules, practices and procedures necessary for working with and around sodium and sodium components are to be adhered to.

b) Since the free surfaces of the main primary pumps are below the high points in the primary loops, it is possible to flood the pumps with sodium. Pump sodium level alarms are to be monitored continuously throughout the procedure to avoid this possibility.

Test Completion Criteria:

a) The reactor vessel and primary loops of the Sodium Heat Transfer System are filled to approximately 1427' 9" elevation as indicated on two reactor core level indicators with the pump discharge piping and IHX's also filled for normal operation.

b) Those sodium level, pressure, temperature, and flow instruments which could be proven operable during the fill procedure function properly.

SUMMARY

WET CRITICAL TEST, AI-P-1158

Test Purpose: The purpose of this test is to load fuel into the reactor core and to determine the reactivity worth of sodium. The minimum critical mass of the sodium filled reactor core will be determined experimentally and compared with the theoretical value. The minimum critical mass is expected to occur with 36 fuel elements present and all control rods at or near the fully withdrawn position. The loading procedure outlined within this test procedure will only be applicable for a maximum of 75 fuel elements. A sub-critical calibration of the central control rod will be performed.

Test Method: The following tests must be performed prior to this test:

a) Dry Critical, AI-P-1155

b) Dry Excess Loading, AI-P-1163

c) Hot Sodium Circulation, AI-P-1167 with sodium in the core at 350F.

Fuel will be loaded in predetermined amounts and locations. After each group of fuel elements is loaded, the control rods will be withdrawn according to the specified procedure, and counting data will be taken. As the loading approaches criticality, the number of elements in each loading will be decreased to one element per step.

Measurements will be taken before and after control rods are moved. Several curves of inverse multiplication vs. fuel loaded will be plotted, and after each step an extrapolation to criticality will be made before more fuel is loaded. It will then be decided if a change should be made in the loading schedule. Generally, not more than half the amount of fuel for extrapolated criticality will be loaded at any one time. However, if the extrapolated curve indicates that less than two elements are needed for criticality, one element will be loaded; if the extrapolated curve indicates that between 2 and 4 elements are needed for criticality, two elements may be loaded. The expected wet critical loading is 36 fuel elements. If the extrapolated critical loading is other than 36 elements, the loading schedule will be revised.

The counting rates as determined by the in-core instruments (three fission chambers), will be recorded after each step in the loading procedure. A plot of the inverse of the average multiplication for the instruments being used will be maintained, and the loading will proceed according to the extrapolated critical loading that this plot indicates.

The core contains 19 control rods. The control rods will be operated as five gangs, consisting of 6, 6, 3, 3, and 1 rods in gangs 1, 2, 3, 4, and 5, respectively. Control rod gangs 1 and 2 will be withdrawn as one gang as long as this is desirable, and inverse count rate vs. fuel loading will be determined and plotted. Control rod gang 3 will be fully withdrawn if it will not make the core critical, and inverse count rate will be determined and plotted. When it is determined that the core is not critical, control rod gangs 4 and 5 will be withdrawn 50%, and inverse count rate will be determined and plotted. Control rod gang 5 will be fully withdrawn, observing that the core remains subcritical. Gang 4 will be fully withdrawn, observing that the core remains subcritical, and the inverse count rate vs. fuel loading will be determined and plotted. If it is thought that criticality will be reached upon withdrawal of control rod gang 4, individual rods of this gang may be withdrawn in 1/2-ft increments, until criticality is reached. Thus the reactor will become critical on one of the control rods of gang 4 which will not be worth as much as gang 5 (C-1), and thus criticality will be attained more slowly.

The sodium worth will be determined by four methods. The first is by adjusting analytical calculations. Analytical calculations will have been made which determined the critical radius and the excess reactivity available with the core fully loaded for the dry and wet conditions. There will be some disagreement between the actual wet and dry critical loadings and the analytically calculated values. Therefore, it will be necessary to bring the analytical values in line with the experimental values by adjusting the nuclear constants in the machine code in some manner.

This will be done by adjusting the value for ν (nu) (neutrons per fission) so as to make the calculated k_{eff} equal to one at the experimentally determined west critical radius. By dividing the previously calculated k_{eff} for this radius into ν a k_{eff} of 1.0 is obtained. Using the new core constants, the k_{eff} for the operational loading of 137 fuel elements with sodium will be recalculated. This adjusted full core k_{eff} with sodium will be compared with the adjusted full core k_{eff} without sodium (dry). The difference in reactivity will then be the estimated reactivity worth of sodium.

The second method is similar to the first in that adjustment of analytical calculations is involved. This also will be done by adjusting the value for ν so as to make the calculated k_{eff} equal to 1.0 at the experimentally determined wet critical radius. Using this adjusted function of k_{eff} vs. radius, k_{eff} will be calculated for the dry critical radius. This k_{eff} minus 1 divided by k_{eff} is the critical radius using the adjusted wet critical function is 0.95, then the worth of the sodium would be

$$\frac{k_{eff}-1}{k_{eff}} = \frac{0.95-1}{0.95} = \frac{-0.05}{0.95} = -5.27\% \ \frac{\Delta k}{k}$$

The third method of determining sodium worth will be from data obtained in the dry excess loading test. An example best explains this method. Assume that the wet critical loading is 36 fuel elements and that criticality is attained with control rod gangs 1, 2, 3 and 5 fully withdrawn (156 in.) and gang 4 withdrawn 150 in. Also, assume that with 36 fuel elements in the core during the dry excess loading, criticality is attained with control rod gangs 1, 2, and 3 fully withdrawn and with gangs 4 and 5 withdrawn 80 inches. The worth of sodium is thus equal to the worth of the difference in critical rod positions. In this example, the reactivity worth of the sodium is equal to the 76 in. of gang 5 in the dry core plus the 70 in. of gang 4 between 80 and 150 in. The reactivity worth of these gangs will be determined from period measurements to be made during the dry excess loading. A correction may have to be made in these calculations in order to compensate for the difference in rod worth for the dry and wet core configurations. The fourth method of determining the reactivity worth of the sodium is to extrapolate the k_{eff} vs. number of fuel elements curve from the dry excess loading to the operational loading and compare this k_{eff} with the k_{eff} of the wet operational loading to be determined during the post-critical tests.

Precautions:

a) Control Rods

To load any fuel into the core, the control rods must be disconnected from their respective drive mechanisms and the control rod support carriage removed. Accordingly, the poison sections of the control rods are in the fully inserted position during any fuel loading operation.

Prescribed scram checks, as indicated in the detailed step-by-step section of this procedure, must be performed each time after control rods are re-connected. These checks will be performed according to SOP 5003.

When withdrawing a control rod, have an operator stationed at the control rod support carriage. Stop withdrawal of all rods at 12 in. and have this operator verify that the snubbers return to the fullup position prior to withdrawing the rod(s) any further.

During the withdrawal of control rods, the three in-core count rate meters must be watched closely at all times for any indication of abnormalties. At intervals, as specified in the detailed procedure, the rod withdrawal should be stopped to allow an instrument check under stable conditions. Control rods are not to be withdrawn without permission of the AI shift leader.

False indications of criticality can be obtained on the period meter during rod withdrawal in the subcritical region and due to spurious electrical transients. If the period goes negative in a short time after halting rod withdrawal, the indication is false.

b) Handling Precautions

During the loading of each fuel element (SOP 5104), the count rates of the in-core fission chambers will be observed by the control room operator and communication maintained with the fuel handling machine operator.

Since fuel elements at ambient temperature will be lowered into 350F sodium, sodium may freeze on the outside of the process tube. To avoid jamming the frozen sodium into the process channel, fuel elements will be lowered into the sodium and allowed to reach equilibrium temperatures before lowering the fuel element into its process channel. The time and method required to accomplish this will be decided at the site prior to the test.

Prior to removing a fuel element from the storage cells, verify that the orifice is full open.

c) Malfunctions

If any of the following conditions occur, the test will be suspended:

1. Failure of one of the control rods.
2. Failure of one of the three in-core fission chambers.
3. Failure of one of the two source range instruments.
4. Failure of any normal or specially installed scrams or setbacks.
5. Failure of one of the two intermediate or three power level nuclear instruments.

The test may be continued with these malfunctions in effect with permission of the Field Superintendent, provided that safety is assured. Under no circumstances can the Field Superintendent approve operation with failure of more than one of the seven nuclear instruments or if any of the inner seven control rods is malfunctioning.

d) Permission for Operation

Permission to start this test must be obtained from the Field Superintendent and the AEC site representative. Manipulation of rods will be performed only after obtaining permission from the shift leader. The number of fuel elements to be loaded at each step will be decided by the Test Engineer.

e) Statistics

To assure normal nuclear statistics of the three in-core fission chambers, the counting statistics will be analyzed prior to starting the test and once each day during the test. This will be done according to SOP 5902, Purity of Nuclear Statistics.

f) Shutdown Margin

Adequate shutdown margin will be assured by rod drop techniques throughout the addition of fuel beyond wet criticality. For this test, "adequate shutdown margin" is defined as a shutdown margin that is equal to or greater than the worth of the sodium plus 1%.

Test Completion Criteria

a) The reactor is made critical with a minimum number of fuel elements in the wet core at 350F.

b) A subcritical calibration of the central control rod has been made.

c) Initial sodium worth has been determined.

SUMMARY

SODIUM LEVEL INSTRUMENTATION - AI-P-1159

Test Purpose: The purpose of this test is to verify proper functioning of all sodium level instrumentation and to check the calibration of each sodium level instrument during the initial sodium filling procedure.

Test Method: Prerequisite Tests: None. The test procedure consists of two basic steps. The first step is to test all of the level instruments before any sodium is present to verify that all components are functioning and that the level indication is appropriate. The second step is to monitor the level instruments during the sodium filling procedure and make the final adjustments necessary for proper calibration of the level instruments.

Precautions: No special precautions.

Test Completion Criteria:

a) All sodium level indicators read correctly to ± 3 inches of sodium level

b) All sodium level alarm units alarm within alarm setting.

c) Proper annunciation occurs on simulated high and actual low sodium levels.

SODIUM FLOW INSTRUMENTATION, AI-P-1160

Test Purpose: The purpose of this test is to verify proper operation of all Hallam Nuclear Power Facility permanent magnet sodium flowmeters and associated readouts.

Test Method: After installation, the flowmeters will be visually inspected to ensure proper electrode location and alignment. The leakage flux density of the 6 and 14-inch flowmeters will be measured and recorded for future reference. Proper operation of the readout equipment will be verified.

Precautions: No special precautions.

Test Completion Criteria:

a) All flowmeters are installed per installation drawings.

b) All flowmeters are free of clinging material.

c) All flowmeters have correct signal circuitry. The signal loop resistances have been measured and recorded.

d) All 6 and 14-inch flowmeters have freely operating and correctly wired flux test coils.

e) Calibration data for all flowmeters have been entered on data sheets.

f) All flowmeter readouts have been calibrated using the calibration data for its associated flowmeter.

SUMMARY

SODIUM PRESSURE INSTRUMENTATION, AI-P-1161

Test Purpose: The purpose of this test is to verify proper functioning of all sodium pressure instrumentation components when connected as complete sodium pressure instruments.

Test Method: Prerequisite Tests: None. All of the sodium pressure instruments will be tested when the transducers are at atmospheric pressure to verify that all components are functioning and that the pressure indication is 0 psig. The high pressure alarms will be checked for proper calibration and the alarm points will be set.

Precautions: No special precautions.

Test Completion Criteria:

a) All sodium pressure transmitters calibrated to $\pm$ 1% of full scale accuracy.

b) All sodium vacuum transmitters calibrated to $\pm$ 2% of full scale accuracy.

c) Proper annunciation occurs on (simulated) high sodium pressures.

DRY EXCESS LOADING, AI-P-1163

Test Purpose: The purpose of this test is to obtain data to be used in determining the reactivity worth of the sodium in the reactor core, to train operators in reactor operation with a core loading of slightly greater than dry criticality, and to obtain information on the dry temperature coefficient of reactivity.

Test Method: The Dry Critical Test, AI-P-1155, must be completed prior to starting this test. Pre-operational Test AI-P-1164, Purge of Primary Sodium System, will be completely performed during this test.

The fuel loading will follow a specified sequence. Not more than 1% reactivity or more than five fuel elements will be loaded in one step. A plot of k_{eff} vs. fuel loading will be kept to assure these criteria. The excess reactivity worth for the particular fuel loading will be determined from period measurements. The total excess reactivity loaded will be determined by summing the incremental worths of each loading. Fuel loading will continue as long as adequate shutdown margin is assured. (The core will not become critical with all rods below their midpoint of travel.)

After each addition of fuel with all rods inserted, the inverse multiplication will be determined to furnish a measure of reactivity and shutdown margin. This will provide an indication of the shutdown capabilities of the control rods and, also, indicate how many additional fuel elements may be loaded. The shutdown margin will also be determined using a rod drop technique.

After each addition of fuel the reactor will be made supercritical to obtain a positive period and thus determine the excess reactivity of each fuel addition. The reactor will be put on approximately a 1-minute period. The periods will be determined using scalers with automatic printouts. The printout will record the total count for 10-sec intervals. The total counts will be plotted on semi-log paper as a function of time. The inverse slope of the stable portion of the curve gives the reactor period directly. Also, the time required for the reactor power to double will be measured at various intervals. This method may be preferred because plotting is not required. Several stopwatches will be used to give continuous data, and show when a stable period has been achieved. This test will be started at ambient temperatures with no sodium in the core. After maximum fuel loading (see precaution (k), below), fuel will be removed to one fuel element above dry criticality, heaters will be installed, and the temperature will be increased slowly to 350F. Data will be taken to determine the dry temperature coefficient of reactivity. The reactor will be brought critical and shut down numerous times for the purpose of operator training.

Precautions:

a) The poison sections of the control rods are in the fully inserted position with holding magnets de-energized during any fuel loading operations. Each time control rods are reconnected, rod operation must be checked by raising rods 12 to 24 in., observing that rod down lights indicate properly.

b) During the loading of each fuel element, the count rates of the in-core fission chambers are observed by the control room operator. Communications are maintained between the control room and the fuel handling machine or crane operator. Only personnel concerned with this operation are in the high bay area during fuel transfer.

c) None of the seven central control rods (gangs 3, 4 and 5) are removed from the core for maintenance during these tests. Any malfunction in these rods must be repaired before continuing the test.

d) The test is suspended if any of the following conditions occur: failure of one or more of the control rods; failure of any of the in-core fission chambers; failure of either source range instrument; failure of either intermediate range instrument; or failure of any normal or specially installed scram or setback. This test may be resumed with these malfunctions in effect with permission of the Field Superintendent provided that safety is assured and the following minimum requirements are met: At least six of the seven nuclear instruments are working properly; and all prerequisite reactor scrams (except one nuclear instrument, if not functioning) are operative.

   It should be pointed out that the scrams on the in-core nuclear instruments are temporary and non-coincidence. Thus, if one instrument channel is inoperative, the corresponding scram circuit is also inoperative.

e) Following the loading of any group of fuel elements, the corresponding shield plugs in the loading face shield are checked with a helium leak detector to insure against any cover gas leakage.

f) No detectable nuclear heating of fuel is allowed. A 10°F rise in fuel temperature will be considered detectable on the in-fuel thermocouples. The high temperature alarm on in-fuel thermocouples will be set no higher than 25°F above ambient temperature. This setting will allow for some changes in ambient temperature without changing the alarm setpoint. This alarm calls for immediate reactor shutdown by the operator. A 10°F rise in temperature at the hottest part of a fuel element is equivalent to holding that element at 240 watts for 10 minutes.

g) The minimum intentional reactor period is 40 seconds.

h) The maximum intentional flux level is three decades above "source critical". Maximum flux is only approached for a very short time at the completion of period measurements, and does not represent a sustained power level.

i) The fuel loading procedure limits excess reactivity per step to one percent, with a maximum number of five fuel elements per step.

j) The core is not loaded such that criticality could be achieved by full withdrawal of one control rod with other rods inserted.

k) The minimum shutdown margin is equal in magnitude to the core excess reactivity. A simple way to assure meeting this requirement is to not allow more fuel than that required to make the reactor critical with all control rods 50 percent withdrawn. Due to the slight asymmetry in control rod worth vs. position, the lower 50 percent of rod travel is worth more than 50 percent of total rod worth. This asymmetry will be experimentally verified. In any event, not more than 75 fuel elements are loaded.

Test Completion Criteria

a) Fuel is loaded into the dry core as limited by Precaution k.

b) From period measurements the total excess reactivity in the dry core is known.

c) A plot of fuel loading vs. excess reactivity for the dry core has been completed.

d) The dry temperature coefficient of reactivity has been determined with the reactor core loaded to one fuel element above dry criticality, over the temperature range from ambient to 350°F. The data necessary to make this determination is obtained during the primary system helium purge and pre-heat. These operations afford valuable operator training in starting up and shutting down the reactor.

REACTOR HEATERS, AI-P-1164

Section 1

Test Purpose: The purpose of this test is to verify the proper rating, location, circuiting, and operation of the core heaters and their associated controllers.

Test Method: Prerequisite Tests: None. The core heater controllers will be tested for correct operation and calibration by simulating temperature signals from thermocouples. The resistance and current flow of each heater will be measured and wattage calculated to assure proper heat input to the core. Heater circuits will be checked from the control center to the heaters.

Precautions:

a) All circuits to the heaters must be off for controller tests. Prior to the preheat operation power should not be applied to the heaters for a period longer than 15 minutes. If this time is not sufficient to allow all necessary measurements to be taken, then power should be turned off and the heat permitted to dissipate for at least 30 minutes before turning on the power again.

b) Heater circuits operate at 480 V.

Test Completion Criteria:

a) Heater control circuits operate properly.

b) Reactor heaters operate properly.

c) Associated thermocouples and their recorder-controller and indicating-controllers are operable and are wired correctly.

Section 2

Test Purpose: The purpose of preoperational test AI-P-1164, Section 2, is to heat the reactor core to 350 F, in conjunction with preheating the primary sodium system, AI-P-1116.

Test Method: Prerequisite Tests:

a) Prior to preheating the reactor above 100 F, Section 1 of the Purge of Primary Sodium System Preoperational Test Procedure AI-P-1151 must be performed.

b) Part B of Preheating System Preoperational Test Procedure AI-P-1116 must be performed in conjunction with this part of this procedure.

The reactor will be heated to a temperature of 350 F using the core heaters and controllers. The operation will be observed closely to determine any malfunction or inadequacies of the equipment involved. The time required to complete the heating of the reactor to 350 F is approximately 200 hours.

Precautions:

a) Throughout the reactor structure there are several areas where temperature differential limitations exist. These areas and the temperature ~~differentials are noted~~ in the detailed procedure. These temperature differentials must not be exceeded if the reactor is to remain within tolerable stress limits during preheat.

b) All heater circuits are energized at nominal 480 volts, 277 volts to ground, and should be handled carefully.

c) Since there is no sodium in the reactor, only one of the bottom heater circuits should be energized at any time to avoid undesirably high thermal stresses in the reactor structure.

d) The reactor atmosphere helium content must not be allowed to drop below 95% during the heating operation or thermal damage to the core clamps may result.

Test Completion Criteria:

a) The reactor core has been heated to 350 F without exceeding specified temperature differences as measured by in-core thermocouples.

b) The core temperature is automatically maintained at or near 350 F.

PROTECTIVE SYSTEM MODIFICATION FOR FUEL LOADING, AI-P-1166

Test Purpose: The purpose of this test is to modify the Protective System for fuel loading for the critical experiment to include additional reactor trip circuits and to verify the operability of the Protective System after the modification. The additional trips will come from incore nuclear instrumentation which will be tested for operability in this procedure, and from period instruments. The purpose of this test is also to install the necessary scram, setback, and startup interlock jumpers necessary for fuel loading for the critical experiment.

Test Method: Channels XI, XII, and XIII will be set up and cabled. Prior to installing the fission chambers in the reactor, each channel will be calibrated and set, and detection of actual neutrons will be made, employing a neutron source.

The fission chambers and associated high-temperature cable will then be installed in the reactor. Neutrons from the incore source will be subsequently detected on each channel.

Finally, with the additional trips wired into the Protective System, each trip will be checked for operability.

Precautions:

a) The neutron source which is used during this procedure is hazardous to personnel. Consult with the Health Physicist regarding handling.

b) Chambers and cable assemblies are delicate instruments and should be handled accordingly.

Test Completion Criteria:

a) Core loading and mapping cabinets 1 and 2 have been installed in the control room. Cables have been connected and checked for insulation resistance (10^{12} ohms minimum for low temperature coaxial cables, 10^{11} ohms minimum for high temperature signal cables, and 10^{10} ohms minimum for high temperature high voltage cables) and noise (no measurable noise with oscilloscope set at 0.005 volts/cm).

b) Each of the three channels XI, XII, and XIII has been checked and calibrated in accordance with the manufacturer's instrucions from the preamp through the Linear Count Rate Meter (LCRM) using a signal generator.

c) With each fission chamber connected to the channel, and using the Pu Be source for obtaining neutrons, the shape of the neutron and alpha pulses at the output of the LCRM has been observed and compared with predicted shapes. The count rate vs discriminator setting data has been recorded. The discriminator setting is locked at an alpha background of 20 cpm.

d) The range selector physical stop on each LCRM has been set so that ranges greater than 10^6 cpm cannot be used.

e) The alarm and scram solid-state relays in each LCRM have been set at 75% and 90% full scale, respectively, using a simulated signal to verify the setting.

f) The three fission chambers have been installed in their thimbles in the reactor core, and the insulation resistance of the high temperature coax cables verified to be greater than 10^{11} ohms for the signal cables and 10^{10} ohms for the high voltage cables.

g) Background count-rate data have been recorded for each of the three channels, with the detector positioned near the core centerline.

h) With the reactor neutron source in the core, it has been verified that each chamber is counting neutrons. Count-rate data have been obtained with each detector at the lower limit of travel, at core centerline, and at upper limit of travel.

i) The following wiring connections have been made:

   1) All three alarm contacts have been connected to the annunciator.

   2) All three scram circuits in the 3 LCRM have been wired in series and connected to terminals TB-A-58 and TB-C-60 in the protective panel (reference drawings 7508-D723304 and 7518-D72315).

   3) Relay contacts K-57-1, K-58-1, and K-59-1 have been wired in series. The lead from K-59-1B to K-1-1A has been been removed.

j) It has been demonstrated that any one of the 3 in-core channels or any one of the 2 period channels, or any one of the two source range channels on high count rate will scram the reactor by putting in a simulated signal at each LCRM and period chassis, and verifying that control rods drop. (Calibration of the period channels is covered under AI-P-1119). Operation of the annunciators by any of the 5 channels has been verified.

k) A jumper has been installed in the main control board from lead 3X3 to 3X15 (Reference Drawing D794552), thereby bypassing all sodium valve interlocks and the Channel I and II source interlocks. It has been verified that the remaining startup interlocks (key, gantry crane, control rods down, scram and setback) are operative by opening each interlock, in turn, and observing that rods cannot be withdrawn.

1) Jumpers have been installed in the nuclear panel to interrupt current to K59 in the protective panel upon high source range count or short period on intermediate range channels (Reference Drawings 7508-D72304 and 104R189).

HOT SODIUM CIRCULATION TEST, AI-P-1167

Test Purpose: The purpose of this test is to achieve the following:

a) Verify that expansion of the primary piping under the temperature conditions of the test will not impair the integrity of the system and will, therefore, be indicative of the performance at actual power operating conditions.

b) Demonstrate mechanical operability of process system components at elevated temperatures.

c) Check alignment of the core and the loading face shield after elevated temperature operation.

d) Verify satisfactory operability of control rods over the range of temperature established during the test.

Test Method: The following are prerequisites for the performance of this test:

a) AI-P-1155, Dry Criticality

b) AI-P-1163, Dry Excess Loading

c) AI-P-1156, Initial Fill of the Secondary Sodium System with Sodium

* d) AI-P-1157, Initial Fill of the Primary Sodium System with Sodium

e) AI-P-1103, Main Heat Transfer System

The necessary controls for flow, power, etc, plus any interlocks which may be involved, will be temporarily altered to permit the test to be performed as outlined in the following paragraphs. Any altered instrumentation will be restored to it original condition after completion of the test.

Before the heat transfer system has been preheated above 100 F, observations and measurements will be made on components and piping to be used as reference points for later observations and measurements.

*All fuel will be removed from the reactor before this test begins.

After the heat transfer systems have been preheated and the primary and secondary sodium systems have been filled with sodium, circulation of sodium will be initiated until isothermal conditions of approximately 350 F are reached. At this time, valves in the primary and secondary sodium systems will be opened and closed, pumps will be stopped and started again, and certain steam system valves will be operated, all to verify functional operability of the various components. Control rods will be maneuvered to verify their freedom and operability at this temperature. Alignment of the core structure and the loading face shield will also be checked.

After the above has been completed, the temperature of the primary and secondary sodium systems will be gradually raised to 525 F utilizing the electrical preheat system and heat input from the primary and secondary circulating pumps; and by introducing conventional plant steam at 850 psi to the steam generators, There will be no liquid water in the unit. This will enable maintenance of the steam generators at 525 F minimum.

The temperature of the primary and secondary loops will then be raised to approximately 575 F. A 50 F differential between the bulk secondary sodium temperature and the steam generator shell temperature must not be exceeded. When isothermal conditions in the primary and secondary sodium heat transfer systems have been achieved, critical pipe movements will be charted and a check for leaks will be made. At this temperature system components will again be checked for operability, as before, The same items previously observed will be rechecked and proper performance verified. Control rods will be operated and scrammed.

The plant will remain at the isothermal temperature of approximately 575 F in the sodium systems with sodium circulation for a period of 48 hours during which time plant conditions will be observed. The actual temperature will fall where it may; limited by the pressure of available steam being introduced into the steam generators from the conventional plant, the design pressure of the equipment, the heat transfer rate from the steam to the shell, and a maximum allowable ΔT of 50 F between the bulk sodium temperature and the shell temperature. This limiting temperature may be as low as 535°F. At the end of the 48-hour period, a final check of the mechanical operability of components will be made to insure that satisfactory performance has not changed.

After all tests and observations have been completed, cool-down operations will be initiated. The steam supply to the steam generators will be shut off. Condensate will be allowed to collect in the evaporator for 12 to 24 hours. Feedwater will then be added to normal operating level.

Dummy or mock-up fuel elements will be inserted and removed in selected core locations to insure operability of the fuel handling machine. The use of conventional plant steam for maintaining the steam generators at temperature will then be discontinued. Cool-down will be accomplished by dumping a small amount of steam to the steam dump system for heat dissipation. Preheat system control temperatures will also be lowered. When the sodium and steam system have reached 350°F preheat temperature, a final set of pipe movement measurements will be made together with a final check for operability of components, as before. All controls will be reconnected and their operation verified. The Hot Sodium Circulation Test will then be complete and the plant will be released for subsequent tests.

Precautions:

a) The temperature difference between secondary sodium and the steam generator shells shall not exceed 50 F.

b) Rates of temperature change in any part of the system shall not exceed 50 F per hour.

c) The heating and ventilating system must be in operation.

d) When vaults or cells must be entered, the atmospheres must be suitable for breathing, and a vessel entry permit must be completed.

Test Completion Criteria:

a) Primary and secondary sodium heat transfer systems have been filled and sodium circulation has been accomplished at the temperatures specified in the test.

b) Operability of the system mechanical components has been demonstrated at the various temperatures specified.

c) Piping movements have been charted at the temperature levels called for in the detailed test procedure. Piping movements have not caused failure, damage, or distress and consequent leakage in the systems.

d) Control rods function properly and without binding over the range of temperatures tested.

e) Alignment of the core and loading face shield has been preserved within design limits.

Question XI(c): Expand on the general plans for postcritical testing with respect to test objectives, methods, and precautions. Outline briefly the anticipated power ascension program. What action will be taken if at any stage a test should indicate deviation from expected performance?

Answer: Postcritical tests are those tests conducted between wet criticality and the completion of the initial full-power run. In general, the objectives of the postcritical tests are:

a) to establish the operational safety and integrity of the plant systems under nuclear conditions;

b) to determine plant nuclear characteristics;

c) to provide personnel with operational training and experience; and

d) to demonstrate safe-power operation of the HNPF.

The postcritical tests are accomplished as part of the planned power ascen-program, described in detail in the attached summary of test AI-PC-830, "Rise to Power." This test provides a specific sequence of testing at each of the power levels in this testing period, and specifies the requirements to be met before power can be raised to a level above that previously attained.

A written test procedure for each of the postcritical tests has been developed. Each test procedure includes step-by-step detailed instructions which must be followed during its performance. Many of these procedures were reviewed and approved by the AI Sodium Reactor Safeguards Committee. These included all nuclear test procedures, all protective system test procedures, all control system test procedures, and the fuel channel orifice adjustment test procedure. These postcritical test procedures have been or are being given a detailed review by the Reactor Engineering Division, Chicago Operation Office. Summaries of test purpose, test method, precautions, and test completion criteria are attached.

Each test is assigned to a responsible test engineer who is responsible to the testing manager for his assigned tests. Each procedure will be given a thorough review by the responsible test engineer before starting the test. The test engineer will modify portions of the procedure and write in detailed instructions as necessary and advisable for performance of the test. This completed product is called "The Master Procedure." The Master Procedure for each test is reviewed by the Test Manager, Assistant Field Superintendent, Operations Manager, Evaluation Manager, Operating Services Manager, Field Superintendent, AEC site representative, and the Shift Leaders. The test procedure, with comments incorporated, is then given a detailed review in a formal meeting normally including the Field Superintendent, AEC site representative, Test Engineer, Assistant Field Superintendent, Test Manager, and Operations Manager. Finally, to start the test, approval signatures from both the Field Superintendent and the AEC site representative are required.

The responsible test engineer follows the test through the shifts as necessary to ascertain that the test is being performed as intended; he also serves as a consultant to Shift Operations. When necessary, the test engineer responsible for a particular test will coordinate all or part of the test for the Shift Leader. The responsible test engineer is the principal evaluator for results of a given test.

If actual results deviate appreciable from anticipated results, the Evaluation Manager and his staff also perform detailed evaluations. Deviations from anticipated results will be corrected or satisfactorily explained before the test is allowed to proceed. Any system or component discrepancies will be corrected. In event the equipment performance criteria for a given test cannot be met in detail, actual performance will be analyzed with regard to its influence on personnel safety and over-all plant operation. If personnel safety will not be compromised and plant operation and output are not adversely affected by less restrictive criteria which are met by the equipment, the test may be declared acceptable by the Field Superintendent subject to approval by the AEC site representative.

The general criteria for ascertaining acceptable completion of each test are given below. Specific criteria are outlined as a part of each test summary.

a) The test must have been run according to a detailed, step-by-step, approved procedure.

b) Each step in the procedure must have been signed off properly.

c) An engineering evaluation of the test results must have shown the adequacy of the system.

d) The completed test must be acceptable to the AEC site representative.

HYDRAULIC TEST OF MAIN HEAT TRANSFER SYSTEM, AI-PC-510

Test Purpose: The purpose of this test is to obtain and correlate performance data on the Main Heat Transfer System during the rise-in-power test (AI-PC-830). Results obtained will serve a two-fold purpose.

a) Satisfactory performance at each power level will be the basis for the next rise-in-power step.

b) The resultant performance data will be plotted and included in the plant operations manual. A permanent record of "new clean" performance will thus be available for future comparative purposes in analyzing plant performance.

Test Method: This test is to be performed as a part of the rise-in-power test (AI-PC-830). When a new power level is reached and steady state conditions are attained, three sets of data will be taken at one-hour intervals. The results of the three sets of data will be averaged and plotted on charts which also show predicted performance. Wide deviation from predicted system performance will be brought to the attention of the test engineer and plant supervision for analysis and evaluation prior to further power increases.

Pressure drops in the IHX's and steam generators will be recorded during testing of the heat exchange equipment (Test AI-PC-530) and will not be repeated here. However, the resulting data will be cross-referenced. Pressure drop across the reactor will be determined as part of this test. It is to be noted that the rise-in-power test is written such that data from one portion of the test will correlate with data from other portions of the test, thus permitting accurate cross-referencing of results.

Precautions: If wide deviations from predicted performance are observed, the facts must be brought to the attention of the test engineer immediately.

Periodically, a visual check should be made of all pumps to determine that (1) bearing lubricating systems are at the proper oil levels, (2) oil temperature is satisfactory, and (3) no unusual noises or vibrations are noted. Abnormal conditions should be reported immediately to the engineer in charge of the rise-in-power test.

Test Completion Criteria: (a) Sufficient operating data has been obtained at power levels, as required by the test, to allow determination of the following information regarding the hydraulic characteristics of the Sodium Heat Transfer System and its associated components at those power levels:

1) System sodium flow rates in pounds per hour.
2) Pressure rise in the system pumps.
3) System pump differential pressure in feet.
4) System hydraulic work factor.
5) System output work in horsepower.
6) System input power.
7) Approximate pump efficiency.
8) System pressure drop as a function of flow.
9) System efficiency as a function of flow.
10) Reactor pressure drop as a function of flow.
11) Pump transmitted horsepower as a function of coupling current.

(b) The system is demonstrated to perform in accordance with plant design parameters.

INTERMEDIATE HEAT EXCHANGER PERFORMANCE, AI-PC-530

Test Purpose: The purpose of this test is to obtain performance data on the intermediate heat exchangers (IHX) during the rise-in-power test. Results obtained will serve a two-fold purpose:

a) Satisfactory performance at each power level will be the basis for the next rise in power.

b) The resultant performance data will be plotted and included in the Plant Operations Manual. A permanent record of "new clean" performance will thus be available for future comparative purposes in analyzing plant performance.

Test Method: This test is to be performed as a part of the rise-in-power test (AI-PC-830). When a new power level is reached and steady state conditions are attained, three sets of data will be taken at one-hour intervals. The results of the three sets of data will be averaged and placed on charts which also show predicted performance. Wide deviation from predicted performance will be brought to the attention of the test engineer and plant supervision for analysis and evaluation prior to further power increases.

Precautions:

a) Where wide deviations from predicted performance are observed, the facts must be brought to the attention of the responsible test engineer immediately.

b) The sodium leak detectors should be monitored periodically to detect possible sodium leakage.

Test Completion Criteria: (a) Sufficient operating data has been obtained at power levels as required by the test to allow determination of the following information regarding the heat exchangers at those power levels:

1) Primary sodium enthalpy loss through the heat exchangers

2) Secondary sodium enthalpy gain through the heat exchangers

3) Calculated reactor thermal output

4) Accuracy of the calibration of the nuclear power computer

5) The mean delta T associated with the heat exchangers

6) System heat transfer coefficient

7) Pressure drop versus flow curves for each primary and secondary sodium loop.

(b) Intermediate heat exchangers perform in accordance with plant design parameters.

MODERATOR COOLANT LOOP, AI-PC-540

Test Purpose: The purpose of this test is to determine the flow rates necessary in the moderator coolant channels to dissipate heat energy stored in moderator element graphite at power levels up to and including full power; to determine the necessary flow controller adjustments at various power levels; and to determine the characteristics of the moderator coolant loop under scram conditions. This information will be used as a base point during subsequent power operations for recognition of changes in moderator element operating parameters.

Test Method: This test will be conducted during the initial rise to power test (AI-PC-830) between 4% and 100% power. Moderator flows will be controlled manually to maintain moderator exit temperatures at the mixed mean outlet temperature ±25 F. Moderator flows, valve positions, main primary pump discharge pressures, and other pertinent data will be recorded hourly when changes are effected. At 150 Mwt, a bumpless transfer from manual to automatic control will be made. The automatic controller will be used during the rise from 150 thermal megawatts to 100% power. Then a bumpless transfer will be made from automatic to manual. Manual control of the moderator flow will then be used for full power operations.

The moderator element operating parameters will be monitored after two scram tests, one from a low power level and another from 100% power.

Precautions: At no time during this test will the moderator coolant exit temperature as measured on TRC-3 be permitted to be more than 50 F higher or more than 30 F lower than the special fuel element core outlet temperatures measured on TRC-2 and TRC-4. This limit is imposed to minimize the possibility of moderator element damage by thermal stresses concomitant with an unequal temperature distribution across the tops of the moderator elements.

Test Completion Criteria:

a) It has been verified that the moderator coolant flow control valve can be adjusted to match moderator coolant temperature with the average fuel element core outlet temperature at various power levels.

b) Sufficient operating data has been obtained, to determine required moderator coolant flow as a function of power to produce the specified coolant outlet temperature.

c) The moderator coolant automatic flow control system maintains moderator coolant outlet temperature within $\pm$ 25F of average fuel coolant outlet temperature during normal loop operating periods.

d) The automatic moderator coolant flow control system controls moderator coolant temperatures satisfactorily following a scram from power.

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EVAPORATOR PERFORMANCE TEST, AI - PC-550

Test Purpose: The purpose of this test is to obtain and correlate performance data on the steam evaporators during the rise-in-power test. Results obtained will serve a two-fold purpose:

a) Satisfactory performance at each power level will be the basis for the next rise-in-power step.

b) The resultant performance data will be plotted and included in the Plant Operations Manual. A permanent record of "new clean" performance will thus be available for future comparative purposes in analysing plant performance.

Test Method: This test will be performed concurrently with, and as specified by, rise-in-power test, AI - PC-830. When each new power level is reached and steady state conditions are attained, three sets of heat balance data will be taken at one-hour intervals. The results of the three sets of data will be averaged and plotted on charts which also show predicted performance. Wide deviation from predicted performance should be brought to the attention of the test engineer and plant supervision for analysis and evaluation prior to further power increases.

Precautions: Where wide deviations from predicted performance are observed, the facts must be brought to the attention of the test engineer immediately. Periodically, a check will be made of the third fluid system to insure that no leakage is taking place. External surveys of the steam generators should be made occasionally to observe possible external leakage, failure of insulation, leaking valves, malfunction of controls, etc. It is recommended that each steam generator be under surveillance during each rise-in-power.

Test Completion Criteria:

a) Recorded data at each power level of the rise-to-power test are evaluated and an overall heat transfer coefficient is determined therefrom and compared to the predicted coefficient.

b) Evaporators are producing saturated steam at the flow and temperature and pressure conditions corresponding to the power level of the particular step of the rise-to-power test.

SUPERHEATER PERFORMANCE TEST, AI-PC-551

Test Purpose: The purpose of this test is to obtain performance data on the steam superheaters during the rise-in-power test. Results obtained will serve a twofold purpose:

a) Satisfactory performance at each power level will be the basis for the next rise in power.

b) The resultant performance data will be plotted and included in the Plant Operations Manual. A permanent record of "new clean" performance will thus be available for future comparative purposes in analyzing plant performance.

Test Method: This test will be performed concurrently with, and as specified by the rise-in-power test, AI-PC-830. When each new power level is reached and steady state conditions are attained, three sets of heat balance data will be taken at one-hour intervals. The results of the three sets of data will be averaged and plotted on charts which show predicted performance. Wide deviation from predicted performance should be brought to the attention of the test engineer and plant supervision for analysis and evaluation prior to further power increases.

Precautions: Where wide deviations from predicted performance are observed, the facts must be brought to the attention of the test engineer immediately. Periodically, a check will be made of the third fluid system to insure that no leakage is taking place. External surveys of the steam generators should be made occasionally to observe possible external leakage, failure of insulation, leaking valves, malfunction of controls, etc.

Test Completion Criteria:

a) The recorded data at each power level of the rise-to-power test are evaluated and an overall heat transfer coefficient is determined therefrom and compared to the predicted coefficient.

b) The superheaters are producing superheated steam at the flow and temperature and pressure conditions corresponding to the power level of the particular step of the rise-to-power test.

STEAM GENERATOR, STEAM PURITY, AI-PC-552

Test Purpose: The purpose of this test is to obtain steam purity performance data on the steam generator during the rise-in-power test. Results obtained will serve a two-fold purpose:

a) Satisfactory performance at each power level will be the basis for the next rise-in-power step.

b) The resultant performance data will be included in the plant operations manual. A permanent record of "new clean" performance will thus be available for future comparative purposes in analyzing plant performance.

Test Method: This test will be performed concurrently with and as specified by the rise-in-power test, AI-PC-830. During the rise-to-power test, in the initial phases, boiler water control will be initiated in an attempt to attain "normal" boiler water condition. These parameters are outlined as follows:

1) Hydrazine will be used for oxygen scavenging. Hydrazine will be introduced between the deaerator and the deaerated feed-water storage tank in concentrations to maintain an excess of 0.02 ppm in the boiler water.

2) Morpholine will be injected into the evaporators for condensate pH control, in quantities to maintain hot well pH between 9.0 and 9.2.

3) Tri-sodium phosphate will be injected into the evaporators to maintain an excess phosphate content of about 5 to 7 ppm (maximum 15 ppm).

4) pH of the boiler water will be maintained at about 10.5 (10.0 to 10.7). Tri-sodium phosphate will be used to effect this, limited by (3) above. Sodium hydroxide will be added as

necessary to supplement the tri-sodium phosphate, with an upper limit of 18 ppm caustic alkalinity.

5) Total dissolved solids will be maintained at about 50 ppm, with an upper limit of 100 ppm, by use of the continuous blowdown.

6) Silica concentration will be limited to 3 ppm.

At reactor powers between 4% and 15% of full power, solids in the boiler water will be increased slowly (25 ppm/hr) by extra addition of sodium phosphate, until total dissolved solids concentration reaches 1500 ppm. During this portion of the test, careful attention will be paid to steam conductivity leaving the separators. When and if conductivity readings on the steam indicate approximately 1 ppm solids or more, solids concentration will be held at that point.

After 1500 ppm TDS (or 1 ppm solids in steam) has been reached, available conductivity instrumentation will be read periodically. During the rise-to-power test at powers of 20%, 60% and 100%, samples of steam and boiler water will be drawn for laboratory analysis. Results of these tests will be correlated with conductivity readings.

When 100% power has been reached, and with 1500 ppm TDS in the evaporators, an evaporated sample will be taken from one of the steam risers off the evaporators. These samples will be correlated with conductivity readings on the installed equipment.

After a period of operation at 100%, which is sufficient to reach equilibrium conditions and draw samples, evaporators will be blown down cautiously a number of times until "normal" boiler water condition is attained. This will conclude the test. Data will be correlated and included in the operations manual for future reference.

Precautions:

a) Solids concentration in the evaporators will be increased slowly

(25 ppm/hr rate of rise). During this time the steam conductivity meters (Larson-Lane analyzers) will be connected to the saturated steam sampling points on the separator outlets. Conductivity readings will be monitored continuously. If conductivity exceeds 3 micromhos, further addition of solids will be discontinued.

b) Readings from the steam conductivity meters will not be entirely accurate until the various sample lines have been in service for about 24 to 48 hours. Sample lines should be steamed out for a period of about 24 to 48 hours during preliminary warm-up. This may be done with warmup steam. As soon as warmup steam is available in the steam drums, the Larson-Lane analyzers should be placed in service for clean-out.

c) In activating a test point, always have condenser water flowing through the coils before steam or boiler water is admitted to the test chamber.

d) Since there are four sample lines connected to each Larson-Lane analyzer, care must be used to flush sample lines before taking readings. The sample through a single sample line will be allowed to flow for a minimum of 1/2 hour, or a minimum of 15 minutes after conductivity reading becomes constant, whichever is later, before readings are taken.

e) While taking samples or readings, do not blow down evaporators and do not take samples while adding excess chemicals. In other words, do not take samples under abnormal [illegible] conditions of chemical feed and blowdown.

f) Do not take samples or readings during conditions of sudden load change. Allow operating conditions to reach equilibrium. In this connection, if sudden load change causes carryover, the Larsen-Lane instrument will record the fact.

Test Completion Criteria: Steam purity at full load conditions, and at anticipated transient conditions, is satisfactory (approximately 0.25% moisture and 1 ppm solids in the steam) for the superheater and turbine.

COOLING SYSTEM TESTS, AI-PC-560

Test Purpose: The purpose of this test is to demonstrate that the cooling water system is adequate for the purpose of providing cooling to the reactor cavity, pipe chases and galleries, spent fuel pit, nitrogen coolers, loading face shield cooling system, and the heating and ventilating equipment. Included in the procedure is the technique for determining the final heat balance of the system.

Test Method: This test must be performed as scheduled in AI-PC-830 and as the prerequisites listed in Section IV of that test apply.

This test will be conducted in conjunction with the rise-to-power test and will be executed as follows:

a) Four hours after a new power level is achieved (or a cooling water flow adjustment is made), the data listed on the attached data sheets will be recorded.

b) These data will be recorded twice more at 8-hour intervals to establish any trends, after which the recording interval will be extended to 24 hours.

c) Shorter intervals than those above may at times become desirable and may be established at the discretion of the test engineer.

d) Changes in cooling line valve positions are to be made as the need becomes apparent; however, due to the great heat capacity of some of the areas being cooled (such as the reactor cavity), it is anticipated that $\sim$4 hours will be required for these temperatures to stabilise sufficiently to obtain a temperature trend. To preserve continuity of information, changes in valve positions will be recorded.

Precautions: Temperatures in all units being cooled will be observed periodically and will be kept below design limits during the rise to power test.

Test Completion Criteria:

a) All components requiring cooling from the cooling water systems are within design temperature limits at all power levels up to full power.

b) Systems requiring balancing (e.g., Cavity Liner) are balanced to maintain essentially uniform temperature throughout the component.

c) Automatic temperature control devices (where provided, as in the cell cooling systems) function properly and maintain essentially constant temperatures in the areas being cooled.

d) Sufficient operating data has been obtained to verify plant design parameters of heat generation, flow, pressure drop and heat transfer for future use in analyzing plant and equipment performance.

LOADING FACE SHIELD COOLING SYSTEM, AI-PC-570

Test Purpose: The purpose of this test procedure is to demonstrate that the reactor loading face shield cooling system will accomplish its function of maintaining the loading face shield at an acceptable temperature. Included in the procedure is the technique for determining the final heat balance.

Test Method: This test will be performed as scheduled by AI-PC-830, "Rise in Power" and as the prerequisites listed in Section IV of that test.

This test will be conducted in the following manner:

a) Four hours after a new power level is achieved or a cooling water flow adjustment is made, operating temperature, pressure and flow data will be recorded.

b) These data will be recorded twice more at eight-hour intervals to establish any trends, after which the recording interval will be extended to 24 hours.

c) Based on indications from thermocouples in the loading face, the proper adjustments are made to individual coil throttle valves to achieve a flat heat gradient across the loading face shield. The necessary adjustments are made to cooling water flow through the system heat exchangers to provide a return N_2 temperature of about 95°F.

Precautions: Nitrogen outlet temperature must not exceed 135°F.

Test Completion Criteria:

a) Mechanical components in the loading face shield coolant system operate satisfactorily in accordance with system design parameters.

b) The loading face shield bottom plate temperature can be maintained within design limits ($\sim 180^\circ$F. maximum).

c) Cooling circuits are balanced to maintain uniform temperature gradients.

HELIUM SYSTEM PERFORMANCE, AI-PC-580

Test Purpose: The purpose of this test is to determine if there are any significant increases in helium leakage rates when the operating temperatures of the reactor and environs are increased. These data will also provide a useful history against which future helium leak rates may be judged.

Test Method: This test will be performed concurrently with, and as specified by the Rise-in-Power test, AI-PC-830. This test will be accomplished by recording and comparing the flow rates in the appropriate branch circuits of the helium system at power levels attained in the Rise-to-Power test AI-PC-830. Any significant increase in helium leakage resulting from the higher operating condition will be thus identified according to the general area in which it is occurring.

Precautions: The established equipment clearance and valve tagging procedure must be observed when locating or correcting helium leaks. Helium leaks from the reactor core and primary loops must be treated as potential radiation hazards and the proper precautions must be taken.

Test Completion Criteria: Sufficient operating data has been obtained, as required by the test, to allow identification of excessive leakage rates. All helium usage will be localized by branch line, and all identified leaks will be repaired.

RADIOACTIVE VENT SYSTEM, AI-PC-600

Test Purpose: The purpose of the Radioactive Vent System post critical test is to complement the preoperational test of the radioactive vent system with an operational test. The wash cell, or "wet" portion of the Radioactive Vent System, has already undergone an operational test in the preoperational test procedures for the wash cells (AI-P-1133) and Radioactive Vent System (AI-P-1110). Therefore, this test procedure is written to operationally test the "dry" portion of the Radioactive Vent System.

Test Method: This test will be performed concurrently with and as specified by the Rise-in-Power Test, AI-PC-830. The method employed in verifying the operability of the "dry" portion of the Radioactive Vent System is to periodically record applicable data on the rise-to-power data sheets while the vent system is in operation, to analyze the data taken and to note any adjustment found necessary through analysis of the data.

Precautions: Adhere to normal vent system operating procedures while operating the system.

Test Completion Criteria:

The dry Radioactive Vent System performs as designed to maintain a slight negative pressure on the vent system collection piping under varying conditions of plant operation. Vent compressors start and stop automatically from preset suction pressure signals, and in accordance with predetermined operating cycles.

Test Purpose: The purpose of this test is to re-align each computer in the protective panel and to check the operability of each setback and scram circuit. The latter will be tested by observing that the control rods drive in on setback and drop on scram when appropriate signals are applied to the sensing elements. Where this is difficult to do, simulated signals will be applied to the protective panel inputs to obtain the setbacks and scrams.

Test Method: Prerequisite Tests: Sections VII-C and VII-D of this procedure shall be run in conjunction with and as specified in post critical test, Rise to Power, AI-PC-830.

The following tests will be performed with the reactor subcritical:

a) Each of the computer chassis of the protective panel will be removed from its cubicle, one at a time, and installed in the the test cubicle, where it will be re-aligned, in accordance with the Plant Protective Panel Maintenance Manual.

b) Each setback and scram circuit will be checked by actuating the sensing element or, where this is impractical, using a simulated signal input and witnessing that the control rods are driven in on a setback or dropped on a scram.

c) While the sodium flow control system is being tested, the sodium flow signals, the reactor inlet temperature signals, and the reactor fuel channel outlet temperature signals which feed the protective panel will be monitored for correct values at the protective panel.

During the Rise-in-Power Post Critical Test (AI-PC-830), the protective panel input signals will again be monitored for correct values at the protective panel. The three neutron flux signals from the nuclear instrumentation channels V, VI, and VII will be calibrated and adjusted for use in the T_o computers.

Precaution: After pulling each poison rod off bottom position, verify that the snubber has returned to the up position by observing that the associated relay in the snubber alarm relay box, located on the control rod carriage, has operated as indicated by the associated relay light being on.

Test Completion Criteria:

a) Re-alignment of the protective panel and resetting (as necessary) all alarm, setback, and scram trips has been accomplished.

b) All setback circuits operate correctly.

c) All scram circuits operate correctly.

d) It has been demonstrated that the following computers and remote indicators correctly compute and indicate at various sodium temperatures, sodium flows, and reactor power levels, according to measured values of inputs.

1) Flow to flow computer and indicators.

2) High Na temperature fuel channel outlet chassis indicators.

3) dT/dt Na temperature fuel channel outlet chassis and indicators.

4) T_o computer and indicators.

Test Purpose: The purpose of this procedure is to test the performance of the plant control system. This will include re-aligning the subsystems of the plant control system and determining their stability when on auto control. It will also include demonstrations of bumpless transfer from manual to auto and from auto to manual. As a final test, the reactor plant will be operated as a load-following plant. This procedure will not raise power but will test subsystems at various power levels attained by AI-PC-830, "Post-Critical Master Schedule Including Rise to Power".

Test Method: This test, except for Section A, will be performed in conjunction with the rise-to-power test (AI-PC-830) and as scheduled therein.

Prior to any appreciable rise in power, with the reactor fully loaded and shut down, the primary and secondary sodium loops will be placed on flow control and slaved together from the division-of-load computer. By manually varying the power demand signal into the division-of-load computer, the cascaded system will be checked out for accuracy and stability.

Stability tests will be performed at the power levels attained in the rise-to-power test (AI-PC-830) from 2.5 MW through full power. Specifically, these power levels are 2.5, 10, 20, 30, 38, 50, 100, 125, 150, 200 and 250 Mwt.

With the reactor's thermal power at about 2.5 MW with full Na flow, the neutron flux controller will be placed on auto and checked for ability to control.

With the reactor's thermal power at about 20 MW with full Na flow, the nuclear power computer, together with the fuel outlet flux compensation controller, will be placed on auto and checked for ability to control. The sodium flow control system will then be operated to demonstrate that the plant will follow a demand flow.

Finally, with the reactor operating at approximately 50 Mwt and with the reactor plant feeding steam to the turbine, the plant power control system will be cascaded with the flow control system and the plant will be operated on load-following control.

Load-following tests will be run at 50 Mwt, 100 Mwt, 150 Mwt, 200 Mwt and 250 Mwt. At that time it will be demonstrated that the plant's load-following capabilities will meet 5 Mwe/min load changes between 15 and 100 percent rated design power.

Precautions: Continuously monitor the plant control system during this test to immediately detect any instabilities, should they occur. If instabilities occur, immediately transfer to manual control and retune the controller causing the disturbance.

Test Completion Criteria:

a) Plant control system is stable at all power levels.

b) Bumpless transfer between automatic and manual control is possible.

c) Plant control system will follow load changes of at least 5 Mwe/min.

d) Feedwater control system controls evaporator level to $\pm$ 2 inches of normal water level at all power levels and during maximum load transients.

e) Plant power control subsystem controls steam pressure to turbine at 800 psig and load swings can be handled with pressure variations of less than $\pm$25 psig.

f) Attemperator control subsystem holds steam temperature at approximately 825 F.

g) Division of Load Computer correctly distributes and programs plant load to primary and secondary heat transfer systems, with primary ratio at 1.1, and secondary ratio at 1.0.

h) Nuclear Power Computer correctly calculates reactor power as a function of load and outlet temperature.

i) Neutron Flux Controller is stable at all power levels and correctly controls neutron flux regardless of the number of rods on automatic control.

EMERGENCY FEEDWATER SYSTEM CAPACITY, AI-PC-640

Test Purpose: The purpose of this test is to prove the operational capability of the turbine-driven emergency feedwater pump and the motor-driven standby emergency feedwater pump.

Test Method: This test will be performed concurrently with and as specified by the Rise-in-Power Test, AI-PC-830. This test will prove the capability of the emergency feedwater system by supplying feedwater to the HNPF steam generators using the emergency pumps.

Precautions: When the pressure difference between evaporator steam pressure and feedwater pressure approaches 100 psi, close attention must be paid to feedwater levels to avoid a low water level scram.

Test Completion Criteria: The turbine driven emergency boiler feed pump supplies water to the steam generators to sustain a power level up to approximately 38 Mwt. This is adequate to remove afterglow heat immediately after shut-down from extended full power operation. The backup motor driven emergency boiler feed pump supplies approximately 12 gpm of feedwater to the steam generators.

EMERGENCY FEEDWATER SYSTEM OPERATION, AI-PC-641

Test Purpose: The purpose of this test is to demonstrate that the emergency feedwater system operates properly. Preliminary operation of the reactor plant will have established equipment capability and control, but will not have tested actual operation of the unit from a trip signal.

Test Method: This test will be performed concurrently with and as specified by the Rise-in-Power Test AI-PC-830. To test the emergency feedwater system it will be necessary to scram the reactor on a loss of power scram (loss of all building power) at some nominal power. The 50-Mw plateau of the rise to power is the recommended level. The normal boiler feed pumps are to be shut down manually so as to check out afterglow heat removal. Heat removal is to be continued until the emergency turbine-driven pump can no longer supply feedwater and the emergency standby 12.5-gpm motor-driven pump has subsequently been started and tested for proper operation.

Precautions: The loss-of-power scram must continue in effect for the duration of the test to insure that the turbine-driven pump is not automatically stopped.

Test Completion Criteria:

a) The turbine-driven emergency feedwater pump automatically starts following loss of house power.

b) The standby emergency feedwater pump can be started when the turbine-driven pump has an insufficient steam supply to keep it in operation. Also, the standby pump develops adequate pump discharge pressure.

c) The feedwater pump turbine can be manually tripped by closing the throttle valve.

RADIATION DETECTION AND MONITORING SYSTEM, AI-PC-650

Test Purpose: The purpose of this procedure is to operationally test the the performance of the Radiation Detection and Monitoring System. This will include realignment of all components as necessary, verification of control and/or alarm action, setting of all control and/or alarm set points, and determination of the effect of reactor operation on area radiation levels.

Test Method: This test is performed in conjunction with rise to power test AI-PC-830, and as scheduled therein. With the reactor fully loaded and generating power, each of the radiation detectors will be calibrated using the detector's internal calibration source.

With the system calibrated, each channel will be made to initiate its control and/or annunciator action by using the internal calibration source and moving the alarm or control set point below the calibration point. Correct alarm and/or control action will be verified by observation of appropriate annunciation, flowmeters, and indicating lights.

All alarm and/or control points will be established and set. Most alarm points will have been established and set previously; this test will verify that all alarm points are returned to the correct setting after performance of these tests.

The reactor will be operated at various power levels to determine the effect on building radiation levels.

Precautions:

A. These tests will not of themselves cause any induced radioactivity. Using the internal calibration sources will cause a high indicated radioactivity during the time required for calibration.

   1) Note on the chart of the recorder which records the channel being calibrated the time and duration of use of the internal calibration source.

2) Maintain continuous surveillance of all indicated radiation levels during these tests. Be prepared to take corrective action if a high radiation level is observed on any channel not being calibrated.

3) Prior to intentional activation of local alarms, notify all personnel in the immediate area to prevent unnecessary apprehension and evacuation.

Test Completion Criteria:

1) All components are calibrated and are operating correctly.

2) Correct control and/or annunciation occurs on (simulated) high radiation.

3) All alarm and/or control points are properly set as required by the determination of associated radiation fields.

4) System is sensitive to radiation levels present in the system.

SCRAM AND LOSS OF LOAD, AI-PC-660

Test Purpose: To verify that the HNPF will function properly and safely during the transient conditions of scram or loss of load.

Test Method: This test will be conducted in accordance with Rise to Power Test, AI-PC-830.

First, the reactor will be scrammed at 10 Mwt, and 30% flow with a Δ T of about 45 F to verify that the convection flow system functions properly. Next, the reactor will be brought to 15% power and 50F Δ T and then the reactor will be scrammed. Finally, the reactor will be scrammed from full Δ T and at several power levels. At the first full Δ T power the scram will be initiated by simulating a loss of electrical power to the nuclear plant. This will be coordinated with the Emergency Feedwater Operation Test, AI-PC-641, to assure proper functioning of the emergency feedwater system under these conditions. After every scram the behavior of the turbine will be observed and noted on the data sheet.

In addition, loss of load tests will be initiated at several power levels with full Δ T by manually tripping the turbine. The resulting behavior of the nuclear plant will be observed.

An oscillographic recording system will be utilized to record temperatures during a scram or loss of load transient. The four channels of this oscillograph will be connected to an in-fuel thermocouple, a fuel channel outlet fast response thermocouple from an "instrumented fuel element," and two thermocouples which measure the moderator can skin temperatures in core position M-3. The thermocouples to be used will be chosen at the time of the test.

Precautions: The operator must assure that the conditions following a scram are proper. SOP 4001, Immediate Action After Manual Scram, lists those conditions to be observed when a scram is initiated. Fill out this procedure each time a scram is initiated. After the conditions of SOP 4001 have been established, the control of the reactor will be governed by schedule requirements.

The transient following a turbine trip is expected to cause the evaporator level to drop. If the level does drop and the associated feedwater throttling valves do not open to the full open position, operator action will be required. Therefore, prior to performing a turbine trip, post an operator so that the hand control stations for the feedwater throttling valves can be switched to manual and full open if this action is necessary.

Test Completion Criteria: The HNPF can be operated at full power with adequate equipment protection during the transient conditions of reactor scram or loss of load (turbine trip) as demonstrated by:

a) The convection flow system operates automatically to maintain core temperature gradient following scram.

b) No excessive temperature transients ($>$ 50F in one hour) occur in the Sodium Heat Transfer System or in the steam generators.

c) Temperature transients in the fuel, in the moderator elements, of in the fuel channel exit sodium are not high enough to result in material damage.

d) The HNPF has been scrammed from full power and the turbine has been tripped from full power but excessive thermal transients do not occur.

FUEL CHANNEL ORIFICE ADJUSTMENT, AI-PC-680

Test Purpose:

a) To initially adjust all fuel channels for constant sodium exit temperature.

b) To verify the operability of the fuel channel orifice and plug assembly under reactor operating conditions.

c) To determine the sensitivity of the orifice and plug assembly. Sensitivity is defined as the change in sodium exit temperature per revolution of the orifice plug (F/rev).

d) To investigate the sodium exit temperatures of the fuel channels as a function of reactor power level.

e) To assure the adequacy of the Fuel Channel Orifice Adjustment Standard Operating Procedure, SOP 5007.

Test Method: The Radial Power Distribution Test, AI-PC-800, will be completed or run concurrently with this test.

Prior to achieving 1 kwt, while subcritical, with full sodium flow, the operability of all orifices in the reactor will be checked by running each orifice plug through its full travel, from full open to full closed to full open. When a 50 F difference between any two fuel channel outlet temperatures is reached during AI-PC-830, Rise to Power, all fuel channel orifices will be adjusted to establish a constant sodium exit temperature for all fuel channels. Then the full reactor ΔT will be established and the reactor power will be increased in steps to full power. After each step in power, all fuel channel exit sodium temperatures will be recorded. This will indicate the relative sodium exit temperatures of the fuel channels as a function of reactor power level. After orifices are initially adjusted, they will not be adjusted again during the rise to full power unless the difference in any two fuel channel exit temperatures exceeds 50 F. At 50% and 100% power, the orifice plug sensitivity will be determined for the fuel channel producing the greatest power and for the fuel channel producing the least power. This will be performed over as wide a temperature range as possible for these two channels.

Precautions:

a) Maintain all fuel channel exit sodium temperatures within 50 F of each other.

b) Precautions associated with orifice adjustment are listed in SOP 5005 and 5007.

c) Once the orifices are set initially, with a full reactor Δ T, the orifices will not be adjusted during the approach to full power unless the difference in temperature between any two channels exceeds 50 F.

Test Completion Criteria:

a) The fuel channel orifice and plug assemblies of all fuel elements in the core operate under reactor operating conditions.

b) All fuel channels are adjusted to obtain a uniform sodium exit temperature at full power.

c) The temperature sensitivity of the fuel coolant to orifice plug adjustment has been determined.

d) The standard operating procedure for adjusting fuel channel orifices has been verified.

e) Position of all orifices has been measured and recorded.

RADIATION SHIELDING, AI-PC-690

Test Purpose: The purpose of the radiation shielding test, AI-PC-690, is to determine the adequacy of the shielding around the reactor and process systems.

Test Method: This test is conducted in conjunction with the Rise-to-Power Test, AI-PC-830. The test method will consist of measuring and recording the gamma and neutron levels over the reactor top shield and the primary coolant loops. This will be effected during the rise to power tests and completed at full power. Detailed surveys (gamma and neutron) will be performed at each increase in power and recorded in the radiation survey log.

Precautions:

a) All normal precautions observed while making any radiation survey should be observed during this test. (See Health and Safety Section, Volume I, HNPF Operations Manual, NAA-SR-6500.)

b) If any unexpected increase in radiation levels is encountered during the performance of this test, immediately convey the information to the AI shift leader for further action.

Test Completion Criteria:

a) Detailed gamma and neutron surveys have been made for each steady state power level from one kilowatt to full power.

b) All sources of neutron or gamma streaming have been attenuated to MPL by appropriate shielding, or access to these areas has been restricted or limited.

c) All areas that contain detectable radiation levels are identified with appropriate posting.

WET EXCESS FUEL LOADING, AI-PC-710

Test Purpose: The purposes of the excess fuelloading are:

a) To provide an operational loading of fuel.

b) To maintain a shutdown margin equal to or greater than the reactivity worth of the sodium + 1%. (At least 6% shutdown margin for the fully loaded core.)

c) To determine the excess reactivity of the fully loaded core.

Test Method: The pre-operational test program must be completed prior to performing this test, except that the following tests need not be completed specifically for wet excess level loading:

a) R/A Liquid Waste, AI-P-1111

b) Auxiliary Steam, AI-P-1113

c) Maintenance Cell, AI-P-1139

d) Sodium Flow Instrumentation, AI-P-1160

e) Sodium Pressure Instrumentation, AI-P-1161

With sodium in the core at 350 F, the fuelloading will continue following a specified sequence. Not more than 1% reactivity or more than fifteen fuel elements will be loaded in one step. A plot of K_{eff} vs fuel will be kept to assure these criteria.

The excess reactivity worth for the particular fuel loading will be determined from period measurements. The total excess reactivity loaded will be determined by summing the incremental worths of each loading. Fuel loading will continue as long as adequate shutdown margin is assured.

After each addition of fuel with all rods inserted, the inverse multiplication will be determined to furnish a measure of reactivity and shutdown margin. This will provide an indication of the shutdown capabilities of the control rods and, also, indicate how many additional fuel elements may be loaded. The shutdown margin will also be determined using a rod drop technique.

After each addition of fuel, the reactor will be made supercritical to obtain a positive period and thus determine the excess reactivity of each fuel addition. The reactor will be put on approximately a 1-minute period; the periods will be determined using scalars with automatic printouts. The printout will record the total count for 10-sec intervals. The total counts will be plotted on semi-log paper as a function of time. The inverse slope of the stable portion of the curve gives the reactor period directly. Also, the time required for the reactor power to double will be measured at various intervals. This method may be preferred because plotting is not required. Several stopwatches will be used to give continuous data, and show when a stable period has been achieved.

Precautions:

a) Control Rods To load any fuel into the core, the control rods must be disconnected from their respective drive mechanisms and the control rod support carriage removed. Accordingly, the poison sections of the control rods are in the fully inserted position during any fuel loading operation.

   Prescribed scram checks, as indicated in the detailed step-by-step section of this procedure, must be performed. These checks will be performed according to SOP 5003.

During the withdrawal of control rods, the three in-core count rate meters must be watched closely at all times for any indication of abnormalities. At intervals, as specified in the detailed procedure, the rod withdrawal should be stopped to allow an instrument check under stable conditions. Control rods are not to be withdrawn without permission of the AI shift leader.

False indications of criticality can be obtained on the period meter during rod withdrawal in the sub-critical region and due to spurious electrical transients. If the period goes negative in a short time after halting rod withdrawal, the indication is false.

b) Handling Precautions During the loading of each fuel element (SOP 5104), the count rates of the in-core fission chambers will be observed by the control room operator and communication maintained with the fuel handling machine operator.

Prior to removing a fuel element from the storage cells, verify that the orifice is full open.

c) Malfunctions If any of the following conditions occur, the test will be suspended:

1. Failure of one of the control rods.
2. Failure of one of the three in-core fission chambers.
3. Failure of one of the two source range instruments.
4. Failure of any normal or specially installed scrams or setbacks.
5. Failure of both intermediate or three power level nuclear instruments.

The test may be continued with these malfunctions in effect with permission of the Field Superintendent provided that safety is assured. Under no circumstances can the Field Superintendent approve operation with failure of any normally installed scram, and with less than two in-core instruments, 18 control rods, and 1 source range instrument.

d) Permission for Operation Permission to start this test must be obtained from the Field Superintendent and AEC site representative. Manipulation of rods will be performed only after obtaining permission from the shift leader. The number of fuel elements to be loaded at each step will be decided by the Test Engineer.

e) Statistics To assure normal nuclear statistics of the three in-core fission chambers, the counting statistics will be analyzed prior to starting the test and once each day during the test. This will be done according to SOP 5902, Purity of Nuclear Statistics.

f) Shutdown Margin Adequate shutdown margin will be assured by rod drop techniques throughout the addition of fuel beyond wet criticality. For this test, "adequate shutdown margin" is defined as a shutdown margin that is equal to or greater than the worth of the sodium plus 1%.

Test Completion Criteria:

a) An operational fuel loading is in the core.

b) The shutdown margin in the core is equal to or greater than the worth of sodium plus 1%.

c) The excess reactivity in the core is known.

d) The reactor cannot be made critical by fully withdrawing any one control rod with the remaining control rods fully inserted.

AXIAL POWER DISTRIBUTIONS, AI-PC-740

Test Purpose: To obtain relative axial power distributions in representative fuel channels throughout the reactor core and to determine the changes in these power distributions due to different control rod patterns.

Test Method: This test will be run as scheduled in AI-PC-830, Rise to Power test. The in-core loading and flux mapping fission chambers will be placed successively in nine different fuel channels throughout the core. The nine channels selected are located at nine different core radii and are considered to be representative of the core fuel channel locations.

The three fission chambers will each be used to traverse three different fuel channels. With the reactor operating at a specified power level (1 kwt), the fission chambers will be traversed up and down each channel from the bottom of the fission chamber thimble to the upper sodium pool in 6 to 18 inch steps and a readout will be recorded at each step. Traverses will be made in each channel for five specified control rod patterns which may be desirable. Also, one channel will be traversed with the adjacent control rod at two different withdrawal positions, one-quarter and three-quarters withdrawn. The chamber readout, which is porportional to the power developed in the channel, is plotted vs the axial length of the fuel channel to give the relative axial power distribution in each channel for the different control rod patterns. In order to make it possible to correlate the data obtained from the three different fission chambers, one traverse will be duplicated with the chambers rotated so that No. 1 is where No. 2 was, No. 2 is where No. 3 was, and No. 3 is where No. 1 was.

Precautions:

a) Fission chambers are delicate pieces of equipment; therefore, handling of them should be kept to a minimum and any required movement should be done with great care.

b) The signal cables from the fission chambers should be routed in such a way that they are isolated from other electrical lines as much as possible. Also, they should be placed to avoid surface traffic over them, especially during the data traverses.

c) The fission chamber cables should be monitored for radioactivity and checked for contamination and appropriate protective measures taken when they are withdrawn from the core.

d) Whenever the fission chambers are raised or lowered with the reactor critical, the reactor operator should be alerted so that he may observe the reactor control instrumentation closely.

e) The in-core fission chambers may become radioactive due to accumulation of fission products in the chamber. Fission chambers should be monitored during removal so that over exposure of personnel will not take place.

Test Completion Criteria:

a) The relative axial power distribution in the core has been determined in representative process channels.

b) The attenuation of neutrons from the active core through the upper sodium pool has been determined.

c) From data obtained, the range of vertical control rod travel that permits the minimum peak-to-average axial power distribution has been determined.

CONTROL ROD CALIBRATION, AI-PC-750

Test Purpose:

a) Measure the integral and differential reactivity worths of the several control rod gangs.

b) Determine the effect of shadowing.

c) Determine the equivalent critical settings for various control rod gang configurations.

d) Determine the integral and differential worths of two control rods.

Test Method: This test will be performed as scheduled in post critical test procedure AI-PC-830.

The reactor will be made critical and raised to a power level such that the contribution of neutrons by the source is negligible. With the power steady at this level the control rod positions will be recorded. Next the power level will be reduced in steps. At each step and with the same control rod pattern as recorded initially, the power trace will be observed for at least 5 minutes.

The level at which the source neutrons begin to affect the dynamic behavior of the reactor will be indicated when a slow power rise is detected during the 5 minute observation period. When this power level is found, the power will be increased in a stepwise fashion and, again, at each level and with the same control rod pattern as recorded initially, the power trace will be observed. The lowest steady power level (constant power trace for 5 minutes) will be considered as the beginning of the source independent operating range and all subsequent period measurements will be performed at or above this power level.

The reactor power will be increased to a level about 10 times above the lowest source independent power level. While holding the power steady at this level, control rod positions will be changed and many different equivalent critical control rod patterns will be recorded.

Starting each time with the reactor power level at the lowest source independent level and with a "banked" control rod pattern (all control rods withdrawn the same distance), each of the 5 control rod gangs will be withdrawn twice to place the reactor on periods of about 130 and 40 seconds, while holding the other rod gangs fixed. The differences in withdrawal distances required for each gang to place the reactor on the 130 and 40 second periods will be indicative of the relative or statistical worth of each gang.

With the reactor power level in the source independent range, each control rod gang will be accurately calibrated using the rod withdrawal-period measurement technique. Using this method, the gang to be calibrated is withdrawn from its critical position to place the reactor on a measurable positive period (usually between 30 and 200 sec period). After a waiting time about equal to the reactor period (necessary to allow transient effects to die out), power level data is collected while the reactor power is allowed to increase about 2 decades. Either the gang being calibrated or another gang may be used to terminate the power rise and return the power level to its original value.

The power rise data is analyzed to determine accurately the reactor asymptotic period, which is then converted to reactivity in cents. This is then the reactivity worth associated with the movement of the gang over the region and distance used in the measurement. If this reactivity value is divided by the distance of gang travel, the resulting quantity is the "differential reactivity worth" of the gang for this region of travel.

Such measurements of differential reactivity worth may be made at a number of points along the travel of the gang and plotted versus average gang position for each measurement. By drawing a best-fit smooth curve through the plotted points and integrating the area under the curve, one may obtain the total or "integral reactivity worth" of the gang.

A variation of the above method is often used since the above assumes that the reactor is just critical prior to withdrawal of the control rod gang to establish a period. Determination of the "just critical" condition is sometimes difficult and may introduce significant errors in the determination of differential reactivity worth. In such cases an improvement in accuracy can usually be effected by placing the reactor successively on two periods - one long and one relatively short. The difference in reactivity for these two gang settings is then used to calculate the differential reactivity worth for the region between the two gang positions used for these measurements.

In order to gain some information about the effect of control rod shadowing by other rods, the worth of one rod in each gang will be compared with the total worth of the gang. The reactor will be raised to a power level about 200 times above the lowest source independent level and held steady with a banked control rod pattern. The worth of one rod in each gang and the worth of each gang will then be determined using the rod drop method. The shadowing effect will be indicated by the difference between the sum of the worths of the individual rods in a gang and the total worth of the gang as a whole.

In the rod drop method, the reactor is made critical at a moderate power level and the power level is carefully monitored. One or more rods are then simultaneously dropped either from the fully withdrawn position or some intermediate position, and the power level decay is carefully followed. Power level may be monitored either with a neutron level recorder or with a scaler-printer system arranged to cycle on and off periodically. Data from either of these systems may be used in conjunction with a curve of negative reactivity versus flux ratio to estimate the total negative reactivity insertion associated with dropping the rod or gang. The average negative reactivity, as obtained by using the curve for several delay intervals after rod drop, should be used. If reactor power level drops to the point where the source contribution becomes significant, these data points should be discarded.

A simpler and faster technique is possible if a scaler-printer combination is arranged to operate cyclically with a count period of 15 sec and an "off" period of 1 sec (printing and resetting occur during the 1 sec "off" period). In this case several cycles are recorded while the reactor is at a stable power level. Rods are dropped coincident with the end of a counting period. The counts accumulated during the 15 sec interval immediately after rod insertion are recorded, and the reactivity change in dollars is computed from the following:

$$P \text{ (dollars)} = C (R - 1)$$

where $$R = \frac{\text{counts per 15 sec while reactor is critical}}{\text{counts during 15 sec immediately after rods inserted}}$$

The value of C to be used in the above expression is determined by the value of R and is about 0.39.

To complete the test, two or more rods will be calibrated accurately over their entire travel using the rod withdrawal-period measurement method for determining rod worths. This part of the test will be done with the reactor operating at or above the lowest source independent level and with a control rod pattern (except for the rods being calibrated) that might be used during reactor operation at or near full power.

Precautions:

a) Control Rods: As in all cases of control rod manipulation, ALL on-scale nuclear instruments must be monitored AT ALL TIMES, regardless of whether the control rods are being withdrawn or inserted.

By their very nature, control rod calibrations involving period determinations involve frequent and rapid changes in reactor power, and frequent manipulation of rods over a wide range of travel and hence, over a wide range of differential worths. IT IS IMPERATIVE that those performing these tests be THOROUGHLY FAMILIAR with these procedures BEFORE undertaking the measurements. It is also important that all persons performing these tests appreciate that it will take a significant period of time to increase or reduce core reactivity by rod motion. Thus, to prevent unnecessary scrams when power is rising on a relatively short period, and to prevent the power from falling too low into the source region when power is being reduced, the delay in effectiveness of corrective action should be anticipated and provided for. This will be particularly important when the rod or gang to be moved is near one end of its travel and when a large amount of negative reactivity has been inserted to accelerate power decay following a period run.

At times it will be desirable to reproduce a previous rod or gang position with maximum accuracy. Since backlash in the rod drive and position-indicating mechanisms could introduce errors in this setting, the position should always be approached from the same direction. For these experiments for the purpose of reproducing rod positions, the position indication that is obtained when a rod is withdrawn will be used as a standard.

b) Source Range Instruments: It is anticipated that most of the measurements to be performed in this procedure will be done at a power level near or above the upper sensitivity limit of the source range channels when they are fully inserted. Since a scaler will also be connected to Channel I, it will be necessary at times to partially retract the source range detector in this channel in order to obtain the proper counting rates. If this detector is retracted, care should be exercised when power level is reduced to insure that the chamber is re-inserted so as to maintain power level indications on-scale. The rate of re-insertion should be carefully controlled to prevent spurious "short period" alarms. It is an operating practice to retract or insert the source range BF_3S as necessary to keep them on scale at all times.

c) Malfunctions: If any of the following conditions occur, the test will be suspended and the condition reported to the Field Superintendent.

1) Failure of any of the control rods or their drives and/or position-indicating units.
2) Failure of any normal scrams or setbacks.
3) Failure of either of the source range channels.

d) **Data Recording:** All pertinent data will be recorded as indicated in the step-by-step procedure. A laboratory notebook should be obtained and used to chronologically record the test activities and pertinent supplementary data.

If at any time there is reason to question the potential safety or propriety of the procedures outlined herein, the tests should be suspended and the question referred to the Operations Manager for resolution.

If at any time the necessity or desirability of a deviation from these procedures is indicated, the matter should be referred to the Test Manager.

No deviation will be permitted without approval of the Field Superintendent.

Test Completion Criteria:

a) Reactivity worths have been determined for each rod or group of rods specified in the procedure.

b) The effect of shadowing on control rod worths has been determined.

TEMPERATURE COEFFICIENT OF REACTIVITY, AI-PC-760

Test Purpose:

The purpose of this test is to measure the changes in reactivity associated with changes in the isothermal temperature of the core.

Test Method: This test will be performed as scheduled in post-critical test procedure AI-PC-830, Master Schedule including rise-to-power.

With the reactor critical and at approximately zero power, the isothermal temperature coefficient measurement will be accomplished by first lowering the primary system temperature to about 300°F, followed by increasing the temperature in steps to about 500°F. Prior to each temperature increment, the reactor period associated with a given rod configuration will be determined. The reactor power will be maintained level during the temperature increase. Following the change in system temperature, the previous rod configuration will be duplicated as nearly as possible and the new period measured. The change in reactivity as indicated by the change in reactor period and rod motion, if required, will provide a measure of the isothermal temperature coefficient over this temperature interval.

Precautions:

a) Termination of Reactor Periods

In this procedure, reactor periods will normally be terminated by insertion of a fully-withdrawn control rod. Although all periods are expected to be relatively long (greater than 50 sec), the initial lack of effectiveness of a fully-withdrawn control rod must be appreciated and taken into account when terminating periods.

b) Malfunctions

Upon occurrence of any malfunction of the plant control system or other plant system, the matter should be referred to the shift leader and the experiment suspended until approval to proceed is received from the shift leader. Because of the time involved in changing system temperature, controllers may be set to maintain the existing temperature at the time of test suspension. During test suspension, the reactor may either be shut down or, if the malfunction appears to be amenable to quick repair, the reactor may be maintained critical.

Test Completion Criteria:

The reactor isothermal temperature coefficient of reactivity has been determined in the range of 300 - 500°F.

POWER COEFFICIENT, AI-PC-770

Test Purpose: The purpose of Power Coefficient Test Procedure, AI-PC-770, is to determine the reactivity change in the HNPF produced by incremental changes in power level under specified conditions. One of these specified conditions is that inlet and outlet sodium temperatures are to be held constant. Another is that flow is held constant as power level changes. A knowledge of these relationships will be of future usefulness in predicting the behavior of the reactor from a nuclear standpoint during scram recovery and other necessary power changes.

Test Method: This test will run concurrent with the HNPF Post Critical Master Schedule Including Rise to Power, AI-PC-830. The reactivity changes effected by changes in the reactor power output can be considered as composite effects:

1) Temperature changes in the reactor structure induce expansion or contraction, thus changing buckling and accompanying neutron leakage probability.

2) Thermally induced changes in sodium density alter the number of sodium nuclei per unit volume, with accompanying changes in sodium macroscopic absorption cross section (Σ a).

3) Changes in moderator temperature result in shifts in the thermal neutron energy spectrum, with accompanying changes in the nuclear properties of the fuel, moderator and absorber.

4) Xenon poison concentration, xenon poison buildup and xenon poison burnup are all functions of reactor flux level. At equilibrium conditions the buildup rate is the same as the

burnup rate, and a constant xenon concentration and constant negative reactivity worth is experienced. A significant change in reactor power, or flux level upsets this equilibrium, causing transient poison concentration effects while the xenon concentration seeks its new equilibrium level. These transient effects can rightly be factored into the power coefficient, since they affect the control stability of the reactor during power level changes.

5) Power increases and accompanying temperature increases in the reactor core, result in an increase in thermal energy of neutrons due to increased random motion of target nuclei. If the new neutron energy level extends into the resonance band for the fissionable material, the temperature portion of the power coefficient is negative. Both fuel and poison "Doppler Effect" contributions are expected here.

The chief method of determining the reactivity changes associated with incremental power changes will be to note the positions of the control rods at just critical position before and after changes in power level are effected, and convert the changes in rod withdrawal to changes in reactivity worth.

At low powers and where possible at higher powers, the power coefficient under constant flow conditions will be investigated. Also at higher powers (after taking the steam dump system out of service), the power coefficient under constant ΔT conditions will be investigated; where possible, identical control configurations will be used throughout the test. The closely calibrated rod in C-36 will be used to maintain criticality and its compensation will be used as a measure of reactivity change.

At power levels where other post-critical tests necessitate an extended stay, long-term xenon effects will be investigated. At other power levels a 1-hr wait for structural heating will be sufficient before concluding that portion of the test.

Precautions:

a) Normal precautions outlined in the Rise to Power test will be observed in the execution of this procedure.

b) The initial control rod configuration set up at the beginning of the test will be used, if possible, throughout the entire test. If a deviation from the pattern is found necessary due to bad temperature distribution or any other cause, consult the engineer in charge of the test on the new configuration to be set up.

Test Completion Criteria: Steady-state power coefficients of reactivity have been determined over a wide power range for constant flow conditions and again for constant reactor Δ T conditions

RADIAL POWER DISTRIBUTION, AI-PC-800

Test Purpose:

a) To determine core radial power distributions and corresponding radial peak-to-average power ratios for different control rod patterns and different reactor power levels.

b) To obtain data that will enable selection of allowable, full power, operating control rod patterns, taking into consideration both axial and radial power distributions.

c) To investigate the possibility of control rod shadowing of any of the nuclear instrumentation channel neutron detectors.

d) To obtain information that might lead to the restriction of control rod positions or patterns at full power. These would be rod positions or patterns that result in any of the following conditions:

1) Appreciable shadowing of neutron detectors.

2) An axial power distribution in any fuel channel with a peak-to-average ratio greater than 1.65.

3) A radial power distribution with a peak-to-average ratio greater than 1.5.

4) A difference greater than 50 F between any two fuel channel sodium exit temperatures.

Test Method: The reactor power level will be about 10 Mwt per AI-PC-830, with all of the fuel channel orifices open, the central control rods (7) essentially inserted, the outermost control rods (12) essentially withdrawn, and the sodium flow rate at 30%. This should result in a sodium temperature difference through the core of about 45F. With reactor power constant, the sodium flow will be increased to 100% of full flow, and a representative number of the fuel channel sodium exit temperatures will be recorded at 30, 50, 70, 85 and 100% of rated sodium flow. On the basis of this temperature data, a sodium flow rate which results in a reasonable sodium exit temperature distribution will be selected as the test flow rate. At the selected sodium flow rate, all of the sodium exit temperatures will be recorded. Assuming that the flow through each channel is the same with the orifices open, and that the channel sodium inlet temperatures are the same for all channels, the sodium exit temperatures will be indicative of the radial power distribution and the difference in the sodium exit and inlet temperatures for each channel will be indicative of the power developed in each channel.

With the reactor operating as above, the control rod pattern will be changed in a stepwise fashion from a symmetrical pattern to a pattern in which the rods on one side are essentially inserted and the ones on the other side are essentially withdrawn. This asymmetrical rod pattern should produce the greatest possible readout differences between the nuclear instruments on one side of the core and those on the other.

With the reactor still operating as above (reactor power at 10 Mwt, all orifices open and the sodium flow at the selected rate), the sodium exit temperatures will be recorded with several different, symmetrical control rod patterns. The radial peak-to-average power ratio will be calculated for each rod pattern, using the sodium channel exit temperatures and the sodium inlet temperature. One or more of these control rod patterns will be selected at 10 Mwt (all orifices open) as possible full power operating rod patterns. The sodium exit temperatures also will be recorded and the radial peak-to-average power ratios determined for these patterns at 20 Mwt, 38 Mwt, 50 Mwt, and at each 50 Mwt increase in power thereafter, until it becomes necessary to adjust the fuel channel orifices. (The maximum temperature difference between any two fuel channel sodium exit temperatures should not exceed 50 F.)

Precautions:

a) The precautions listed in AI-PC-830, Rise to Power, are applicable to this test, also.

b) An operator should observe the fuel channel sodium exit temperatures closely while the control rod pattern is being changed to an asymmetrical pattern with reactor at power.

Test Completion Criteria:

a) Control rod positions or patterns that would produce either a peak-to-average axial power ratio greater than 1.65 or a peak-to-average radial power ratio greater than 1.5 in any fuel channel have been identified and are restricted for power operation.

b) An operating control rod pattern has been chosen for power operation which results in the best compromise between axial and radial power distributions.

c) The shadowing of neutron detectors by control rods has been investigated.

d) A core radial power distribution has been determined for the preferred operating control rod pattern.

RADIAL STATISTICAL WEIGHT TEST, AI-PC-810

Test Purpose: To determine a radial statistical weight curve for the reactor core.

Test Method: This test will be conducted in conjunction with Rise to Power Test, AI-PC-830.

A dummy element will be successively interchanged with the fuel elements in 9, and possibly 10, different core channels. Eight of these channels are located along a radial line from the core center. The other two channels are located symmetrically with respect to one of the above eight channels and are being used primarily to check the flux symmetry in the core. If identical reactivity measurements are noted in two of the three symmetrical channels, it will be assumed that the flux pattern is symmetrical and measurements in the third symmetrical channel will be eliminated.

A reactivity base will be established by operating the reactor at about 1 kwt (with a specified sodium flow rate and temperature difference through the core) and recording the control rod positions. The reactivity change due to interchanging the dummy element with a fuel element will then correspond to the change in rod position required to bring the reactor back to the same operating conditions under which the reactivity base was established. The change in reactivity resulting from a local perturbation in a reactor is proportional to the change in the local nuclear constants and the statistical weight at the location of the perturbation. Therefore, if the same physical change is made at different points in a reactor, the resulting reactivity changes will be indicative of the statistical weight at the locations of the physical change. Therefore, since in this test the same dummy element will be interchanged with the fuel elements in eight core channels along a core radius, a radial statistical weight curve can be correlated with the resulting reactivity change.

Precautions: No special precautions.

Test Completion Criteria: A radial statistical weight curve has been determined for the reactor core.

MASTER SCHEDULE INCLUDING RISE-TO-POWER, AI-PC-830

Test Purpose:

a) To raise the HNPF to the nominal full power level, 710,000 lb/hr steam flow, estimated at, and henceforth referred to in this procedure as, 240 Mwt (this value may be as high as 256 Mwt).

b) To obtain operating data as required to demonstrate the capability and safety of operation of the plant.

c) To give the sequence for performing the concurrent post critical tests.

Test Method: All preoperational tests must be completed satisfactorily prior to beginning this test. In addition, the Wet Excess Fuel Loading, AI-PC-710, step 1 of the Fuel Channel Orifice Adjustment, AI-PC-680, Part VII-A of Plant Control System, AI-PC-630, and Part VII-A and B of Plant Protective System, AI-PC-620, post critical tests, must be completed prior to beginning this test.

The reactor will be made critical using the procedure outlined in AI-PC-750, Control Rod Calibration, and the following tests or portions of tests will be conducted in the order named:

AI-PC-620, Plant Protective System
AI-PC-750, Control Rod Calibration
AI-PC-760, Isothermal Temperature Coefficient

After the aforementioned tests are completed, the reactor will again be made critical and the power raised to about 1 kwt as indicated on nuclear channels III and IV. At this level the following tests or portions of tests will be conducted in the order named:

AI-PC-620, Plant Protective System
AI-PC-810, Radial Statistical Weight
*AI-PC-740, Axial Power Distribution

*Prior to conducting this test, the in-core fission chamber outputs will have been converted from count rate to current so that they can measure the higher flux levels.

Also at this level the in-core fission chamber outputs will be compared with the outputs measured on nuclear channels III and IV to obtain a better idea of outputs of regular startup channels as a function of power level.

After these tests are conducted, the reactor will be shut down for removal of the in-core fission chambers, then made critical again with a rod configuration found in the control rod calibration and axial power distribution tests. The main loop sodium flows will then be increased to 20% and the power level raised to about 1 Mwt as determined by heat balance calculations. The power level will then be raised to approximately 2.5 Mwt as determined by heat balance calculations.

While waiting for the system temperature to rise, basic system data will be collected as specified in the following test procedures (No particular order implied):

AI-PC-510, Hydraulic Test of Main Heat Transfer System

AI-PC-530, Heat Transfer Tests

AI-PC-540, Moderator Coolant Loop Tests

AI-PC-550, Evaporator Performance

AI-PC-551, Superheater Performance

AI-PC-552, Steam Generator Performance - Steam Purity

AI-PC-560, Cooling Systems

AI-PC-570, Loading Face Cooling System

AI-PC-580, Helium System

AI-PC-600, R/A Vent System

AI-PC-650, Radiation Detection and Monitoring System

AI-PC-690, Radiation Shielding

The foregoing group of procedures fall by their nature into the process category and will be referred to hereafter as the "process group of post critical test procedures."

The above general test pattern will be repeated at successively higher power levels until full power operation is reached. Following is the test method presented in tabular form from reactor startup to nominal full power:

	Condition	Action
A.	1. Sub-critical	1. Clear startup interlocks and go critical as specified in AI-PC-750, Control Rod Calibration.
	2. Avg Na system temps 350°F	2. Do AI-PC-750, Control Rod Calibration.
	3. On emergency feedwater	3. Do portion of AI-PC-620, Plant Protective System.
	4. Reactor loaded to operational loading per AI-PC-710, Wet Excess Fuel Loading	4. Do AI-PC-760, Isothermal Temperature Coefficient, Part A (to 500°F).
	5. Sodium Flow at 20%	5. Change in-core fission chamber outputs from count rate to current.
		6. Go to about 1 Kwt as indicated on nuclear channels III & IV.
		7. Compare outputs of in-core fission chambers with outputs of channels III and IV.
B.	1. Avg Na system temp 500°F.	1. Do portion of AI-PC-620, Plant Protective System.
	2. Sub critical to ~1 Kwt.	2. Do AI-PC-810, Radial Statistical Weight.
	3. On emergency feedwater.	3. Do AI-PC-740, Axial Power Distribution.
	4. Sodium flow at 20%	4. Shut down and remove in-core fission chambers and replace with scheduled fuel or dummy elements.

Condition	Action
	5. Set sodium flows at 20%.
	6. Go critical again and raise power to about 1 Mwt as obtained by heat balance calculations.
	7. Check flows and Δ T's for "ball park" estimate of power levels.
	8. Obtain power coefficient data.
C. 1. ~1 Mwt power level.	1. Obtain power coefficient data.
2. 500-600°F avg Na system temps.	2 Place steam dump system into operation and take warmup steam away from steam generators.
3. On emergency feedwater.	
4. 20% Na flow in all loops.	3. Raise power to 2.5 Mwt as indicated by core heat balance calculations.
5. Reactor ΔT between 6 & 7 F.	
D. 1. ~2.5 Mwt power level.	1. Obtain power coefficient
2. 500-600°F avg Na system temp.	2. Check flows and ΔT's for power level estimate.
3. 20% Na flow.	
4. Reactor Δ T about 17 F.	3. Do portion of AI-PC-620, Plant Protective System test.
5. Steam dump system operating.	4. Collect and analyze data on process group of post critical test procedures.
	5. Do portion of AI-PC-630, Plant Control System test.
	6. Obtain OK for power rise from engineers in charge of individual post critical tests.
	7. Obtain constant flow power coefficient data for flows of 20%, 30%, 50% and 100%.
	8. Withdraw BF3 chambers far enough to keep them indicating on scale when power level is increased by a factor of 4.

Condition	Action
	9. Set sodium flows at 30%.
	10. Raise power to 10 Mwt as indicated by core heat balance calculations.
E. 1. Power level 10 Mwt.	1. Obtain power coefficient data (constant flow at 30%, 50% and 100% flows).
2. Avg system temperatures above 600°F.	2. Make power calculations based on heat balance.
3. Emergency feedwater.	3. Do portion of AI-PC-620, Plant Protective System test.
4. 30% sodium flow in all main loops.	4. Do portion of AI-PC-630, Plant Control System.
5. Reactor ΔT about 45°F.	5. Do process group of tests, analyze data and get OK for next power increase.
6. Steam dump system operating.	6. Do AI-PC-800, Radial Power Distribution.
	7. Adjust orifices as necessary then reduce sodium flow to 20% to get constant flow power coefficient data.
	8. Execute scram and recovery as per AI-PC-660, Scram & Loss of Load, Part A. Check on convection flow control system.
	9. With 20% sodium flow, increase power to 15 Mwt as indicated by heat balance, then increase power to 20 Mwt.
	10. Withdraw start up channels as necessary to keep them indicating on scale.

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	Condition	Action
F.	1. Power level 20 Mwt. 2. Avg Na system temperatures above 700°F. 3. On emergency feedwater. 4. 20% Na flow in all main loops. 5. Reactor ΔT about 134°F. 6. Steam dump system operating.	1. Obtain power coefficient data at 20%, 50% and 100% flow. 2. Make power calculations based on heat balance. 3. Do portion of AI-PC-620, Plant Protective System. 4. Do portion of AI-PC-630, Plant Control System. 5. Increase power level in ~1 Mwt increments as specified in AI-PC-640, Emergency Feedwater System capacity until evaporator level starts to drop, then decrease power to 20 Mwt. Obtain power coefficient data at each increment. 6. Change from emergency feedwater to normal feedwater (3-element control). 7. Do process group of tests, analyze data and get OK for next power increase. 8. Withdraw startup channels as necessary to keep them indicating on scale, then increase power to 30 Mwt as indicated by heat balance calculations.
G.	1. Power level 30 Mwt. 2. Avg sodium system temperature above 700°F. 3. Normal 3 element feedwater control from here to full power. 4. 20% sodium flow in all main loops. 5. Reactor ΔT about 200°F. 6. Steam dump system operating.	1. Obtain power coefficient data at 20%, 50% and 100% flows. 2. Do portion of AI-PC-620, Plant Protective System. 3. Do portion of AI-PC-630, Plant Control System. 4. Calculate power level from heat balance. Match nuclear indications with calculated power. 5. Do process group of tests, analyze data, and get OK for next power increase.

Condition	Action
	6. Increase power to 38 Mwt as indicated on nuclear instruments.
H. 1. Power level 38 Mwt.	1. Obtain power coefficient data
2. Avg reactor sodium outlet temperatures above 900°F	2. Do portion of AI-PC-620, Plant Protective System.
3. 20% sodium flow in main loops, initially.	3. Do portion of AI-PC-630, Plant Control System.
4. Reactor ΔT about 250 F, initially.	4. Do process group of tests, analyze data and get OK for next power increase.
5. Steam dump system operating.	5. Scram and recover as per AI-PC-660.
	6. Do AI-PC-641, Emergency Feedwater System Operation Test.
	7. Increase system temperature and reduce flows until quality (825°F-800 psi) steam is being produced, then warm up main header and admit steam to turbine. Work closely with conventional side in taking over portion of electrical load.
	8. Place steam dump system on standby.
	9. Do heat balance calculations and adjust nuclear instruments to indicate calculated power.
	10. On fully automatic control, increase power to 50 Mwt.
	11. Obtain power coefficient data.
I. 1. Power level 50 Mwt.	1. Obtain power coefficient data.
2. Superheater average sodium temperatures above 850 F.	2. Do portion of AI-PC-620, Plant Protective System.
3. Sodium flows adjusted to keep reactor outlet at or less than 945°F.	3. Do portion of AI-PC-630, Plant Control System.
4. Feedwater on normal (3-element) control.	4. Record orifice data as per AI-PC-680, Fuel Channel Orifice Adjustment.

Condition	Action
	5. Record data as specified in process group of tests.
	6. Adjust orifices as necessary to keep temperature differences between fuel channel sodium exit temperatures less than 50 F.
	7. Record and analyze data from process group of tests. Get OK for power increase.
	8. On fully automatic control, increase power to 75 Mwt.
	9. Obtain power coefficient data.
	10. Set up and obtain power coefficient data under constant flow conditions by first reducing flow and holding power level steady semi-automatically, then ramping power down and back up to initial conditions (as specified in AI-PC-770, Power Coefficient Test).
	11. Obtain power coefficient data and raise power on fully automatic to 100 Mwt.
J. 1. Power level 100 Mwt.	1. Do portion of AI-PC-620, Plant Protective System.
2. Superheater average sodium temperatures above 850°F.	2. Do portion of AI-PC-630, Plant Control System.
3. Na flows adjusted to keep reactor outlet at or less than 945°F.	3. Record orifice data per AI-PC-680, Fuel Channel Orifice Adjustment, Section VII-5.
4. Feedwater on normal (3-element) control.	4. Record data as specified in process group of tests.
	5. Adjust orifices as necessary to keep temperature differences between fuel channel sodium exit temperatures less than 50 F.

Action

J.

6. Record and analyze data from process group of tests. Get OK for power increase.

7. On fully automatic control, increase power to 125 Mwt.

8. Obtain data for constant (T_o) Power Coefficient.

9. Set up and obtain power coefficient data under constant flow conditions by first reducing flow and holding power level steady manually, then ramping power down and back up to initial conditions (as specified in AI-PC-770, Power Coefficient Test).

Conditions	Action
K. 1. Power level 125 Mwt.	1. Do portion of AI-PC-620, Plant Protective System.
2. Superheater average sodium temperature above 850°F.	2. Do portion of AI-PC-630, Plant Control System.
3. Sodium flows adjusted to keep reactor outlet at or less than 945°F.	3. Record orifice data per AI-PC-680, Fuel Channel Orifice Adjustment, Section VII-5.
4. Feedwater on normal (3 element) control.	4. Record data as specified in process group of tests.
	5. Do heat balance calculations and adjust nuclear instruments to match heat balance.
	6. Adjust orifices as necessary to keep temperature differences between fuel channel sodium exit temperatures less than 50 F.
	7. Do orifice sensitivity check as specified in AI-PC-680, section VII-7.
	8. Record and analyze data from process group of tests, get OK for power increase.
	9. On fully automatic control, increase power to 150 Mwt.
	10. Obtain data for power coefficient at constant Na outlet temperature
	11. Set up and obtain power coefficient data under constant flow conditions by first reducing flow and holding power level steady manually then ramping power down and backup to initial conditions (as specified in AI-PC-770, Power Coefficient Test).

Conditions	Action
L. 1. Power level 150 Mwt	1. Do portion of AI-PC-620, Plant Protective System.
2. Superheater average Na temperatures above 850°F	2. Do portion of AI-PC-630, Plant Control System.
3. Sodium flows adjusted to keep reactor outlet at or less than 945°F	3. Record orifice data per AI-PC-680, Fuel Channel Orifice Adjustment, Section VII-8.
4. Feedwater on normal (3 element) control	4. Record data as specified in process group of tests.
	5. Adjust orifices as necessary to keep temperature differences between fuel channel sodium exit temperatures less than 50 F.
	6. Transfer feedwater supply pump 1-A from conventional boiler to nuclear steam generator using method specified in Part VII of this test.
	7. On full automatic control increase power to 200 Mwt.
	8. Obtain data for power coefficient at constant sodium outlet temperature.
	9. Set up and obtain power coefficient data under constant flow conditions. by first reducing flow and holding power level steady manually, then ramping power down and back up to initial conditions (as specified in AI-PC-770. Power Coefficient Test).
	10. Do Turbine trip and reactor scram per AI-PC-660, Scram and Loss of Load. Recover to 150 Mwt.
M. 1. Power level 200 Mwt.	1. Do portion of AI-PC-620, Plant Protective System.
2. Superheater average sodium temperatures above 850°F.	2. Do portion of AI-PC-630, Plant Control System.

Conditions	Action
3. Sodium flows adjusted to keep reactor outlet at or less than 945°F.	3. Record orifice data per AI-PC-680, Fuel Channel Orifice Adjustment, Section VII-10.
4. Feedwater on normal (3 element) control	4. Record data as specified in process group of tests
	5. Adjust orifices as necessary to keep temperature differences between fuel channel sodium exit temperatures less than 50 F.
	6. Record and analyze data from process group of tests. Get OK for power increase and on fully automatic control, increase power to full power.
	7. Obtain data for power coefficient at constant sodium outlet temperature.
	8. Set up and obtain power coefficient data under constant flow conditions by first reducing flow and holding power level steady manually, then ramping power down and back up to initial conditions (as specified in AI-PC-770, Power Coefficient Test).
N. 1. Power level at nominal full power ($\sim$ 240 Mwt).	1. Finish AI-PC-620, Plant Protective System.
2. Superheater average Na temperatures above 850°F.	2. Finish AI-PC-630, Plant Control System.

Condition	Action
3. Sodium flows adjusted to keep reactor outlet at or less than 945°F.	3. Record orifice data per AI-PC-680, Fuel Channel Orifice Adjustment, Section VII-10.
4. Feedwater on normal (3-element) control.	4. Record data as specified in process group of tests.
	5. Adjust orifices as necessary to keep temperature differences between fuel channel sodium exit temperatures less than 50 F.
	6. Do full power portion of AI-PC-840, Xenon Test.
	7. Scram as specified in AI-PC-660, Scram and Loss of Load Test.
	8. Do shutdown portion of AI-PC-840, Xenon Test.

Precautions: During the performance of this test, no design operating limits will be exceeded.

1) Primary sodium outlet temperature as measured on TI-1, TI-2, TI-3 will not exceed 950°F.

2) Maximum U-Mo fuel temperatures, as measured on TR-12 and TR-13, will not exceed 1250 F during steady state and will not be subjected to transient temperature conditions beyond 1450 F.

3) No planned positive period shorter than 60 sec will be effected during the initial startup. This precaution is not due to design limits, but will insure against short period scrams until the nuclear instrumentation characteristics are established, and until the "feel" of operation is acquired through experience.

4) The maximum rate of change of fuel sodium exit temperatures measured on TR-3A and TR-3B will not exceed 50 F/hr.

5) The first six times that the control rods are withdrawn from the fully inserted position, withdrawal must be stopped at 12 inches up from the bottom position until an operator verifies visually that the snubbers are up.

6) The spread (high to low) between any two fuel channel sodium exit temperatures will never exceed 50 F.

Test Completion Criteria:

a) The post critical tests have been completed satisfactorily.

b) Safe operations of the plant at power levels up to and including nominal full power (240 Mwt) has been demonstrated.

c) All components performance tested during the rise-to-power tests have performed satisfactorily in accordance with plant design parameters. Otherwise, actual performance has been measured and deemed acceptable or suitable rectifying action has been taken.

XENON POISONING, AI-PC-840

Test Purpose: To determine the xenon characteristics of the HNPF reactor during and after operation at full power. The following information will be obtained:

a) The buildup of xenon as a function of time.

b) The time required to reach equilibrium xenon.

c) The buildup and subsequent decay of xenon after reactor shutdown as a function of time.

d) The time required to reach peak shutdown xenon.

Test Method: The Rise in Power Test, AI-PC-830, must have been completed.

The reactor power will be raised to full power (a control configuration deemed as optimum for reactivity measurements will be set up). Then, while holding the reactor power steady for about 3 days, the control rod positions will be recorded at 1-hr intervals. During these 3 days, the change in the control rod positions required to hold the power steady will be primarily due to xenon poison buildup. Therefore, the changes in rod positions are indicative of the xenon concentration in the reactor, and by converting these rod position changes to reactivity, a graph of negative reactivity due to xenon poison buildup vs time can be obtained. This graph will indicate the equilibrium xenon reactivity and the time required to reach equilibrium with the reactor operating at full power.

After a period of full power reactor operation, of sufficient length to achieve xenon equilibrium conditions, the reactor will be shut down as quickly as possible. Following the shutdown, the reactor will be raised to a low power level (about 1 Kwt) as soon as is practicable and held steady at this power level for about one day. During this time, the control rod positions will once again be recorded at 1-hr intervals. As before, by converting the changes in rod positions to reactivity a curve of xenon buildup can be plotted.

Precautions: No special precautions required. However, the operators should be aware of the normal precautions applicable to both full power reactor operations and shutdown.

Test Completion Criteria:

a) The time required to reach equilibrium xenon after reaching full power has been determined.

b) The equilibrium xenon reactivity has been determined.

c) An accurate graph of negative reactivity due to xenon poison buildup *vs* time has been made.

d) An accurate graph of negative reactivity due to xenon buildup after reactor shutdown *vs* time has been made.

e) Peak xenon reactivity has been determined.

f) The time required to reach peak xenon after reactor shutdown has been determined.

Question XI(d): The preoperational testing program proposed does not provide for testing at operational temperatures until such time as nuclear heat is available to raise system temperatures to operating values. It is considered necessary to provide, insofar as feasible, hot gas and/or hot sodium testing at temperatures at or approaching operating values prior to initial criticality. Such testing would serve the following purposes:

1) Verify anticipated expansion effects in piping and components, particularly in areas that will be inaccessible following operation.

2) Determine the operability and leak-tightness of sodium system components, i.e., pumps, freeze seals, valves, etc.

3) Check alignment of reactor components at operating temperatures, particularly alignment of core and loading face shield.

4) Test the operability of control rods over the range of operating temperatures.

Answer: A preoperational test procedure "Hot Sodium Circulation Test," AI-P-1167, has been prepared. A summary of this test is attached, and has also been included with the other test summaries. (See answer to Question XI(a) and (c). This test will verify the items listed in the question at temperatures of approximately 585°F.

This test will be conducted after dry criticality and dry excess tests but prior to wet critical tests. There will be no fuel in the reactor.

A. DISCUSSION

A review of the sodium system has been made to determine the allowable test temperature. The test must necessarily be done at near-isothermal conditions since the heat source (electric heaters on the pipes and equipment, together with operation of the main sodium pumps) is not great enough to establish a temperature gradient between the hot and cold legs. The review indicates the following limits:

1. Reactor

1050°F. However, it is not desirable, in our opinion, to heat the bottom ends of the moderator elements much above their normal operating temperature of 610°F since such heating tends to form buckles in the scallops of the moderator cladding. Moderator elements heated isothermally in the second LCTL test (see Appendix A-2, page A-12, of the parent report) showed some insignificant, though observable, buckles at the bottom near the head.

There is no question that isothermal heating of the moderator elements to 1000°F can be done; however, we believe any gains in high temperature testing are not worth the risk, small though it be.

2. Pumps, Valves, Secondary Expansion Tank

1000°F.

3. Piping

1000°F. Note in the answer to Question III(a) that the thermal expansion stresses in the cold leg (at 1000°F) are increased 60% above their normal operating levels, but still not above the allowable value.

4. Intermediate Heat Exchanger

1000°F.

5. Steam Generator

Approximately 585°F. The limiting item here is the stress in the shell weld at the evaporator helium-to-water tubesheet (see Figure 2.14 of the parent report). The temperature difference across this weld should be limited to 60°F in order to keep stresses below the yield point. The shell of the steam generator is designed for 550°F maximum. Using superheated steam from the conventional facility at 850 psig, it is believed that the shell of the evaporator can be raised to about 525°F. Therefore, with a maximum temperature differential across the weld of 60°, the maximum sodium temperature is 585°F. (Under normal operating conditions the sodium temperature in this region does not exceed 557°F.)

With the above limits in mind, a temperature of 585°F was selected as a target for the hot sodium test. The limit of 60°F on the steam generator shell weld discussed above is the key item. Depending on the measured temperatures on the steam generator, the final sodium temperature may be slightly above or below 585°F.

The test objectives, as listed in the question, can all be met at this temperature, except for number four, which cannot be fully met by any non-nuclear test.

a) The expansion of the piping will be measured as the system is heated from 100 to 585°F. Extrapolation to 1000°F is straightforward, and close clearances will be checked to be sure that clearance will be adequate at the high temperature.

b) Leak tightness will be demonstrated by 585°F operation. Note that one of the primary pumps has already been operated at 975°F in its proof test (see Section D.2 of the parent report).

c) Alignment of reactor components with the loading face shield is most important at about 600°F, since this will be the refueling temperature. The reactor is designed so that the moderator elements and process channels lean slightly outward at room temperature; are lined up with the loading face at refueling temperature; and lean slightly outward at full temperature and gradient. The proposed alignment check at 585°F is consistent with the refueling temperature listed above.

d) Operability of control rods can be verified. Since the control rods are designed to operated at temperatures up to 1800°F on the retainer tube, no non-nuclear test can fully establish operating temperatures. Note that one of the control rods has already been proof tested at temperatures from 600 to 1000°F (Appendix D.1 of the parent report).

Question XI(e): How is it proposed to verify that protective system circuitry will respond properly to real abnormal signals ?

Answer: At each major planned shutdown, the protective system will be checked for proper response and setpoints. Both actual signals and simulated signals will be used as shown in Table XI(e)-1. Where simulated signals are used, the

TABLE XI(e)-1

SIGNAL CHECKS

Scram or Setback	Real Signals	Simulated Signals
1. Pump trip	X	
2. Manual scram	X	
3. To computer		
a. Flux		X
b. Temperature		X
c. Flow	X (if reactor is isothermal)	X (if reactor has greater than 50° ΔT)
4. dT/dt computer		X
5. Flow/flow computer	X (if reactor is isothermal)	X (if reactor has greater than 50° ΔT)
6. Reactor sodium level		X
7. Evaporator level		X
8. Period		X
9. Manual setback	X	

signal will be introduced at the nearest convenient connection to the sensor. For example, the reactor outlet temperature will be simulated by a precise millivolt signal applied at the connector on the loading face shield. Values for the simulated signals will be taken from the calibration curves for the sensors concerned. The sensor itself will be checked for continuity, and observed for proper reading at isothermal conditions. It will be verified that the scram relays are de-energized when any of the real or simulated signals reach the scram setting. It will also be verified at least once that the control rods actually drop when the scram relay is de-energized.