Investigation of Releases From Santa Susana Sodium Reactor Experiment in July, 1959

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1. Introduction

1.1. Background

At the request of counsel for defense in the matter of "Lawrence O'Connor, et. al. vs. Boeing North America, Inc.", I have undertaken an investigation to determine whether there was a release of radionuclides form the Sodium Reactor Experiment (SRE) in July 1959, and if so, to determine the quantity and identify the specific radionuclides released.

I have over 30 years experience in nuclear engineering. I graduated from the University of Florida in 1972 with a Bachelor of Science degree in Nuclear Engineering Sciences. My primary areas of expertise are nuclear power plant safety analysis and fission product transport and behavior.

I began my career at United Engineers and Constructors, in Philadelphia, performing nuclear safety analyses for the licensing of nuclear power plants under contract to UE&C. I worked on the Seabrook Nuclear Station, Washington Public Power Supply Systems Units 1 and 4, and Brunswick Power Station. I performed containment pressure-temperature analyses and various analyses for postulated nuclear accidents, including release from fuel and off-site dose analyses. As part of my work, I developed computer codes to assess the performance of safety systems designed to mitigate the consequences of upset and emergency conditions in the plant. I also performed safety assessments of foreign reactor systems, including the Canadian heavy water reactors and the French Phoenix and Super-Phoenix sodium-cooled breeder reactors, as part of a study conducted by the Department of Energy.

In 1978, I joined General Public Utilities, and was at the Three Mile Island facility from April, 1979 until May, 1984, as technical advisor to the Vice President of GPU Nuclear, and as Manager of the TMI-2 Radiological Analyses Group. I was responsible for the assessments of radiological safety for all decontamination operations in the plant and to provide technical support to the recovery team. I also supervised the equipment design and collection of highly radioactive samples to assist the recovery effort.

I joined Science Applications, Inc. while at TMI in 1982, and became the TMI site manager for SAI. I was responsible for the laboratory's contribution to recovery

operations, under contract to Department of Energy. I developed a systematic approach for documenting the fission product release from the fuel and transport through the primary and auxiliary systems, utilizing a combination of specialized sampling, computer modeling and in-situ gamma spectrometer measurements. I made several entries into the TMI reactor building to perform scientific measurements related to the accident sequence and to measure the effectiveness of decontamination efforts.

I founded Daniel & Associates, Inc. in 1984, and now consult to various utilities and governmental agencies. I was the Principal Investigator for an NRC project that developed a correlation between radioactive iodine resuspension and ambient temperature, published as NUREG/CR-4953.

I have written several computer codes, including RADTRAN, a fission product transport code. The RADTRAN code, after review by the NRC, was used to calculate the allowable release rate of radioactive gases from the TMI-2 reactor building during the venting of gases during 1980. I revised a version of the ORIGEN2 radioactive isotopic generation/depletion code and the Oak Ridge National Laboratory (ORNL) shielding code ANISN to operate under the Windows® operating system. I have also written numerous small codes related to dose calculations, gamma-ray shielding, and two phase flow in piping systems.

A list of publications and papers is attached as Appendix A.

1.1.1 Prior Testimony

I have not participated in any litigation in the past four years.

1.1.2 Compensation

My compensation in this case is $ 120/hr.

1.2. Approach

Investigation of the SRE fuel damage incident lends itself to an approach known as "forensic reconstruction", a technique commonly used by investigators of aircraft and other industrial incidents to determine the cause and significance of events leading up to

the failure being investigated. In this technique, mathematical calculations, called "models", are commonly used to compare calculations to data recorded during the incident, and to fill in missing data or gaps in the data that may be relevant to the incident. The "models" use the laws of physics to predict and quantify the outcome of events. On occasion, it may be necessary to make assumptions, due to lack of data, for example. In these instances, the investigator should pursue the validity of the assumption, and then test the generated results against available data. Taken in total, the "reconstruction" must satisfy all data recorded and available from the event, and also must not violate the physical constraints imposed by the plant piping and equipment arrangement.

The approach taken in this investigation was accomplished in two phases. Phase I involved determining an accurate inventory of fission products in the SRE reactor core. Phase II involved determining the inventory of fission products remaining in the reactor core and plant systems after the incident occurred. The difference in inventories, i.e., before and after the incident, provides the best estimate of what may have been released during the incident.

In order to accomplish the forensic "reconstruction", certain system descriptions and operating data were examined and reviewed in detail. Relevant descriptions and data are contained in this report to provide a clear understanding of individual systems and their role in the events that occurred in July, 1959 with respect to release from the SRE. Whenever possible, comparison to SRE data will be made and presented to verify the validity of calculations. Using this generally accepted scientific methodology will result in determining whether a release occurred, and If so, the magnitude of that release.

1.3. Organization of Report

Section 2 of this report is a description of the SRE design features that are necessary to understand the operation and pathways available for release to the environment to determine whether or not a release occurred. Descriptions are taken from SRE design documents and drawings. Every attempt has been made to verify that the drawings and descriptions are an accurate representation of the facility during the period in question. This section thus describes the equipment and systems at the time of the fuel damage

incident. As a general rule, only the systems and equipment necessary for an understanding of the events of the incident are discussed. The exceptions to the rule are that descriptions may also be given for systems and pathways identified by plaintiff's expert to fully understand the allegations presented.

Section 3 describes the operating and irradiation history of the SRE fuel, in order to provide a background for determination of fission product inventory in the reactor core during the period July 12 through July 26, 1959. The information in Section 3 was primarily taken from the reports describing the fuel damage incident. [Ref. 1-4]. Section 3 provides the operational history for power runs up to run 14, and provides the background for the problems that developed in Run 14.

Section 4 is a description of Run 14 and presents the thermodynamic aspect of the fuel damage. This section also presents the results of experiments undertaken to determine the root cause of the fuel damage.

Section 5 deals with the radiological aspect of the fuel damage incident, and introduces the calculated fission product inventory and analyses performed to determine the release from the fuel. Comparisons to sample data are used as benchmarks to compute the source term for release. Section 5 contains the details of the release to the environment from the event. This section also presents the results of experiments undertaken by Atomics International to determine the fate of fission products released from the fuel matrix.

Section 6 discusses specific allegations made by plaintiff's expert, contained in the report "Iodine-131 Releases from the July 1959 Accident at the Atomics International Sodium Reactor Experiment", dated January 1959 [Ref. 32].

Section 7 lists the references relied upon to prepare this report.

1.4. Summary and Conclusions

1.4.1. Summary of Fuel Failure Incident

During Power Run 14 of the SRE which began July 12, 1959, fluctuations of the core exit sodium temperature were noticed at various fuel channels. Some of these fluctuations were severe, and caused swelling of the fuel such that the stainless steel cladding was

breached, exposing the fuel to the sodium coolant, and releasing radioactive fission products to the coolant and reactor cover gas.

On one occasion, early in the run, the reactor's neutron behavior was not within normal operating parameters. The run was terminated on July 26. A series of fuel element inspections were begun which revealed that 13 of the 43 fuel elements had sustained damage. Investigations into the causes of the event revealed that partial flow blockage in certain coolant flow channels was caused by the decomposition products of an organic coolant, tetralin. The flow blockage was responsible for the thermal cycling that eventually caused cladding failure and partial melting of the uranium fuel in the 13 affected fuel elements. Even though the cladding was breached in 13 of the fuel elements, with subsequent release of fission products to the coolant, no radiological hazard was presented to the reactor building. Recovery operation were initiated and conducted within standard AEC regulations on radiation exposures.

1.4.2. Conclusions Regarding the Fuel Failure Incident

This investigation, performed independently from the Atomics International (AI) investigation, relied upon descriptions of events and data contained in the reports generated by AI, as well as experimental data obtained by AI and other industry groups as it related to fuel behavior during the events of July, 1959. The curie balance for Kr^{85}, Xe^{133} and I^{131} as obtained by this investigation are summarized in Figure 1.1.

The basic conclusions reached in this investigation were as follows:

- The AI conclusions that fuel damage was caused by thermal cycling and fuel swelling is supported by industry experience with natural uranium fuel.
- Industry experimental data and plant operational data support the conclusion that radioactive iodine was either retained in the fuel or in the coolant.
- The AI conclusion that no radioactive iodine was released from the site is supported by this investigation and data collected at the time of the incident.
- The majority of fuel damage occurred between July 22 and 24, which is after the date that the vent header alignment was confirmed to be to the holdup tanks.

- The release fraction for noble gases was determined to be 1.5% of the core inventory.

- The releases of radioactive gas were retained in the holdup tanks for decay prior to discharge to the environment.

- The increase in radiation levels noted on July 12 in the high bay area were most likely due to activation of the nitrogen injected into the cover gas between Runs 13 and 14; not as a result of fuel cladding ruptures as postulated by AI in their Interim Report.

- The source of the stack monitor increase on July 12 and 15 was most likely due to an inadvertent valve alignment of the fuel handling cask vent, which was limited in duration. Only noble gases were released by this pathway.

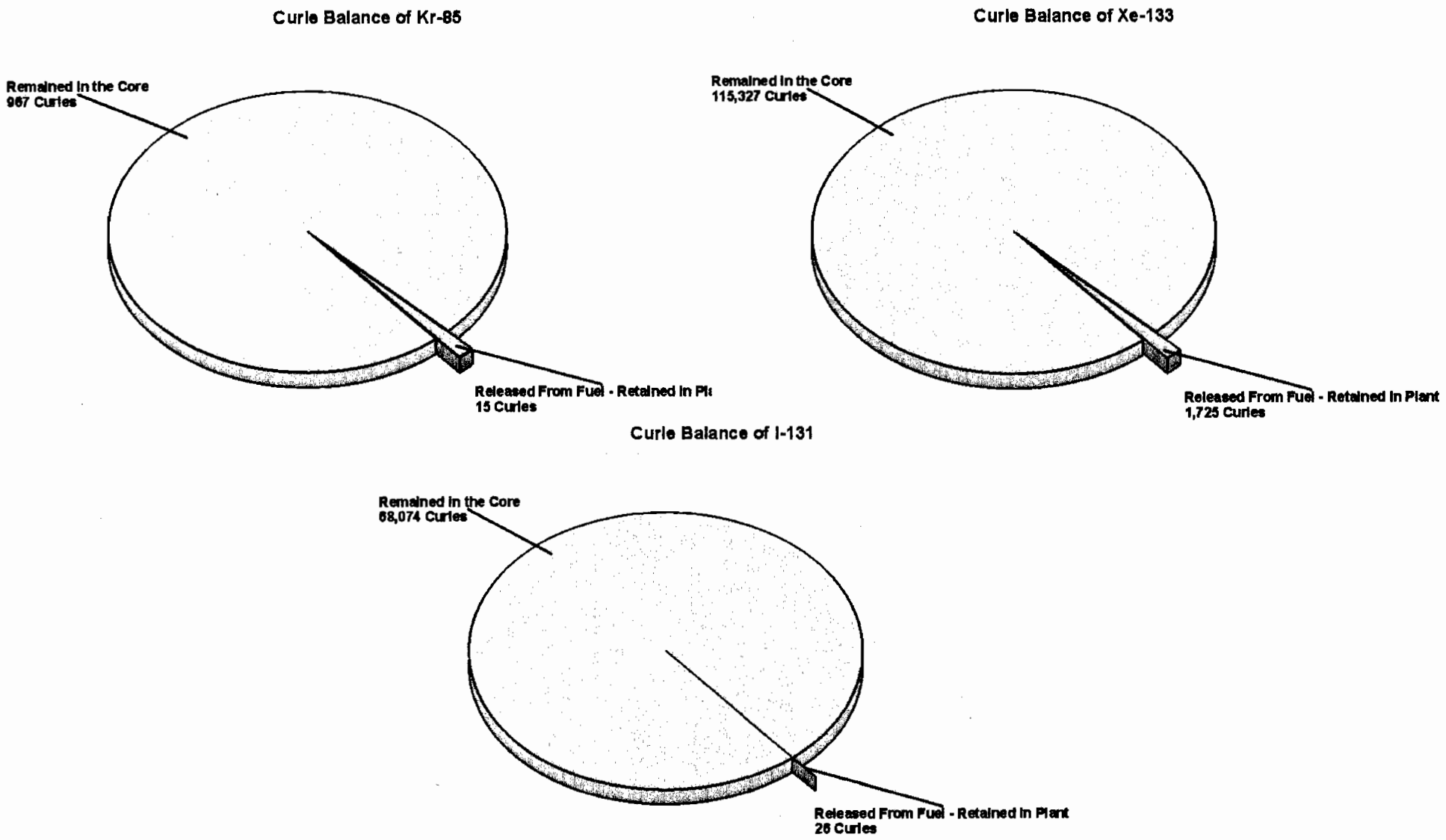


Figure 1.1 Curie Balance for Kr^{85}, Xe^{133} and I^{131}

2. Design of the Sodium Reactor Experiment

2.1. Overview

The purpose of the Sodium Reactor Experiment (SRE) was to generally demonstrate the feasibility of a sodium-cooled reactor as a heat source for a commercial power reactor to produce electricity. A secondary objective was to obtain operational data on slightly enriched uranium metallic fuel and uranium thorium fuel mixtures. The SRE was operated from April, 1957 until February, 1964, and provided the basis for commercial nuclear power development. The Sodium Reactor Experiment site is located 30 miles northwest of downtown Los Angeles, approximately 6 miles west of Chatsworth and 3 miles south of Santa Susana, California. The site is situated in the Simi Hills, and was part of a larger research complex built as a joint effort of Atomics International, Rocketdyne, and the Atomic Energy Commission. [Ref. 5, pp 1]

Figure 2.1 shows the principal buildings of the SRE and their relationship to site north. The reactor building is identified as Building 143, and the other buildings relevant to this study include the liquid and gaseous waste holdup systems in Building 653, and the primary sodium fill tank and system in Building 753.

The elevation of the reactor building is approximately 1850 ft. above sea level. The maximum elevation of the Simi Hills is about 2400 ft, while the San Fernando Valley floor and the Simi Valley floor is approximately 900 ft.

2.2. Reactor Building

The Reactor Building is the main building of the SRE, consisting of a high bay area, a side bay area and hot cell facility. Figure 2.2 shows the reactor building and some of its internal features. The side bay contains the control room, administrative offices, electrical shop, and air conditioning equipment. The high bay area, so-called because of the elevated roof, contains the reactor and its primary and auxiliary coolant systems, new fuel storage, irradiated fuel storage area, a fuel handling machine, and a second moderator handling machine slightly larger than the fuel handling machine which handles the larger graphite moderator cans.

Figure 2.3 shows the floor plan of the reactor building. The reactor, primary sodium coolant loops and irradiated fuel storage area are located below ground. The reactor building floor is mostly clear except for control rod drive mechanisms over the reactor and pump motors over the coolant galleries. A 75 ton bridge crane spans the high bay area to move and position the fuel handling machine and other heavy equipment related to the reactor. The high bay area is normally kept at a negative pressure, to avoid potential air leakage out of the building. The high bay area is windowless with weather-stripped or gasketed access doors to inhibit air leakage into the high bay area from other building spaces.

A side bay area, adjacent to the high bay area, houses reactor experimental facilities, the reactor control room, electrical distribution equipment, instrument storage and maintenance facilities, inert atmosphere systems, and the administration offices.

The hot cell area is an underground area located adjacent to the high bay area. The hot cells are used for examination, dismantling and preparation of irradiated fuel elements for final processing.

2.2.1. Reactor System

The SRE reactor was designed as a low pressure sodium-cooled reactor using slightly enriched uranium fuel. The reactor, shown in Figure 2.4, contains the active core approximately 18 feet below the surface of the reactor building floor. Liquid sodium fills the reactor vessel and passes from the bottom of the reactor vessel to the top sodium pool during power operation. The reactor has heaters in the bottom of the vessel to keep the sodium in a liquid state when the reactor is not in operation. Above the core, the liquid sodium collects in an open volume called the "pool". The pool allows mixing of the sodium that has passed through the fuel assemblies and the moderator cans so that a more uniform temperature exist prior to passing through the heat exchangers in the cooling circuits.

As the sodium passes through the core, some of the sodium atoms absorb neutrons and become radioactive. This process is called "coolant activation". The principal radioactive sodium isotope is Na^{24}, with a half-life of 15.02 hours. In order to keep the radioactive material contained in shielded areas away from plant personnel, the cooling loops are

divided into two systems – the primary sodium loop and the secondary sodium loop. The primary sodium loop passes through the core where it absorbs heat from the reactor, and then to a heat transfer component called an intermediate heat exchanger, where it transfers its heat to the secondary sodium loop. The cooling systems are shown in the schematic diagram, Figure 2.5. The secondary sodium loop, which does not contain radioactive sodium, transfers the heat to the steam generator to produce steam for the turbine, or transfers the heat to the main airblast heat exchanger, where large fans cool the coolant before it goes back to the heat exchanger to pick up more heat.

An auxiliary system is provided for low-power operation, in which a smaller primary sodium loop transfers heat to an auxiliary intermediate heat exchanger, and a non-radioactive sodium loop then transfers the heat to an airblast heat exchanger located on the roof of the reactor building.

Piping between the reactor and heat exchangers inside the reactor building are contained in concrete passages called "galleries" which are filled with an inert nitrogen gas system. This was a precaution to prevent oxygen contamination of the liquid sodium, and to prevent possible fires from water vapor reacting with the sodium in the event of a leak.

During the performance of this study, it was convenient to refer to the design parameters of the SRE. Table 2.1 contains the relevant data of the SRE reactor design. [Refs. 2,3,36,43]

Table 2.1

SRE Reactor and Fuel Design Parameters

Reactor	
Design Heat Output (Mw_t)	20
Coolant Inlet Temperature (°F)	500
Coolant Outlet Temperature (°F)	960
Core Operating Pressure (psig)	3
Reactor Vessel Mechanical Design	
Liner Diameter (Ft-In)	13- 9
Liner Height (ft)	11'
Liner Thickness (in)	0.25
Material	304 SS
Reactor Core	
Fuel Loading (kg U-235)	83.4
Fuel Loading (kg U-238)	2190
Average Neutron Flux (20 Mw_t)(n/cm^2-sec)	5.02×10^{12}
Enrichment (wt % U-235)	2.77
Size (Dia. x Height)	6' x 6'

Table 2.1 (Cont'd)

Control and Safety Rods	
Number	
Shim	2
Regulating	2
Safety	4
Poison	B-Ni
Total Rod Worth	~9.6 Δ *k/k*
Sodium Primary Coolant	
Mass (Kg)	2.55×10^4
Flow Rate (ft/sec)	5
Sodium Melting Point (°F)	208
Sodium Boiling Point (°F)	1618
Fuel Elements	
Fuel Slug Dia. (In)	0.75
Fuel Slug Length (In.)	6.0
Fuel Slugs per Rod	12
Rods/Element	7
Total Number of Slugs	3612
Total Number Fuel Assemblies	43
Cladding Material	304 SS
Cladding Thickness	0.010
Cladding Dia. (In)	0.79

Table 2.1 (Cont'd)

Cladding Gap Material	NaK
NaK Melting Point (°F)	12
NaK Boiling Point (°F)	1445
Cover Gas Volume	
Core Tank Cover Gas (ft^3)	253.5
Primary Fill Tank (ft^3)	1182
Reactor High Bay Volume (ft^3)	$1.4x10^6$
Gaseous Waste Holdup Tanks	
Number	4
Volume (ft^3)	350
Capacity (ft^3 @ 100 psia)	2700

2.2.1.1. SRE Reactor Structure

The SRE was an experimental reactor built in place, rather than a fabricated steel reactor vessel found in today's power reactors. Instead, the SRE reactor was built in place below grade and was designed such that the outer tank would contain the liquid sodium in the unlikely event of catastrophic failure of the core tank. The design utilized sodium as a coolant, so that high pressure water systems would not be required. With low pressure systems, there was no need for a containment vessel to contain the energy released in a hypothetical accident.

Figure 2.6 shows the cross-sectional view of the reactor, and following the order of construction, i.e., from the outside in toward the reactor fuel, the reactor structure consists of three concentric tanks,

- the cavity liner,
- the outer tank , and
- the core tank.

Each of the main components of the reactor is described below. There are descriptions of the top shield plug and the core cover gas region, which are relevant to the events that occurred during Run 14 at the SRE.

2.2.1.1.1. Cavity Liner

The *cavity liner* is the outermost containment vessel in the reactor structure. The tank is constructed of ¼ inch thick carbon steel. The tank is 23 feet high and 14 feet 8 inches in diameter. Around the outer edge of the cavity liner and across the bottom of the cavity liner are 28 evenly spaced 1 inch diameter cooling circuits. The cavity liner is attached to 4 ft thick high density concrete by anchor bolts embedded in the concrete. The concrete forms a biological shield and forms the major support structure for the reactor structure. The cavity liner extends to the reactor operating floor, and is stepped to accommodate the reactor loading face shield assembly. [Refs. 6,7]

2.2.1.1.2. Outer Tank

The *outer tank* is made of ¼ inch thick low alloy steel, and is 18 feet 11 inches high and is 12 feet 6 inches in diameter. It rest on four concentric rings 21 ¼ inches high, which are supported by cavity liner support plates. The outer tank is sealed to the cavity liner at an elevation near the top of the outer tank by a welded bellows seal, shown as an "accordion" shaped object 11 inches high just below the 13 reflector plates underneath the top shield plug. The bellows seal is constructed of Type 347 stainless steel and is 12 feet 6 inches in diameter.

The cavity liner and the outer tank serve as a secondary sodium containment vessel. Insulating blocks are placed in the annulus between the cavity liner and the outer tank to provide insulation for the reactor. The annulus space between the cavity liner and the outer tank is called the "insulation cavity", and contains an inert nitrogen cover gas to prevent the sodium from coming into contact with oxygen in event of a sodium leak into the volume. The insulation cavity is sized such that in the event of a sodium leak that breaches the core tank and the outer tank, the sodium level will remain above the active core region. [Refs. 6,7]

2.2.1.1.3. Core Tank

The inner tank, referred to as the *core tank,* is 1 ½ inches thick and constructed of Type 304 stainless steel. The core tank is 11 feet 3 inches in diameter.

The annulus between the core tank and the outer tank contains the side thermal shield, which consists of seven 5 ½ inch thick interlocking rings. The overall height of the interlocking rings is 19 feet 11 inches. The inner diameter of the rings is 11 feet 6 inches.

The core tank rests on the bottom of the outer tank. The core tank is sealed to the cavity tank at the top by a bellows seal, just as the outer tank was sealed to the cavity liner.

The annulus between the core tank and the outer tank is called the "core cavity". The core cavity contains stagnant sodium coolant adjacent to the core tank wall, and thereby relieves thermal stresses caused by thermal expansion or uneven heating. There is an open space above the sodium in the annulus to allow for thermal expansion of the sodium. The space is filled with an inert helium atmosphere just as inside the core tank.

The helium pressure is maintained at 3 psig to reduce stresses. There is some activation of the sodium in the core cavity, and therefore the vents for this volume are considered contaminated. The core cavity volume does not communicate with the cover gas in the core tank above the reactor core.

The main and auxiliary cooling piping and moderator cooling piping pass through the core tank wall at an elevation just above the top of the moderator elements. Some of the outer moderator elements are modified to allow the coolant piping to be routed down to the lower plenum. The coolant inlet and outlet piping is contained within double walled piping sleeves sealed with a bellows at the cavity liner and extending into the core tank. The double walled piping provides for thermal expansion of the sodium coolant piping, while maintaining the barriers of the outer tank and core tank. The double walled piping also acts as a barrier to keep the "cold" sodium from mixing with the "hot" core sodium in event of a coolant pipe leak. At the same time, the double walled pipe provides a thermal barrier between the "cold" coolant and the "hot" sodium in the core tank. A helium atmosphere is maintained in the double walled pipe, also. [Refs. 6,7]

2.2.1.1.4. Top Shield Plug

One of the unique features of the SRE reactor design is the rotatable shield plug at the top of the reactor. The shield plug fits inside the "ring shield" as shown in Figure 2.7. The ring shield is 71-5/8 inches high, with a 16 ft. outside diameter and 12 ft. inside diameter at the top. The inside section is stepped to prevent radiation streaming from the core region, and to provide physical support for the shield plug. The shield plug is 140 inches in diameter at the top and is made of high density concrete. The plug weighs approximately 75 tons. The plug has Type 304 stainless steel casing at all surfaces that may come into contact with sodium or the helium cover gas. There are 81 small plugs that extend through the shield plug. In addition, there are two 40 inch plugs and one 20 inch plug. These plugs are arranged so that by rotating the top shield plug, any moderator assembly may be removed from the core tank.

The openings for the 81 small plugs have stainless steel sleeves that are seal welded to the bottom shield plug plate for cover gas containment, and to the loading face shield at the top. Access to any one of the moderator cans with a central channel containing fuel

assemblies or instrumentation can be made through the small plug openings by using the fuel handling cask.

Thermal protection is provided on the bottom of the shield plug by 13 reflector plates attached to the bottom of the top shield plug. Cooling circuits are built into the concrete using kerosene as a coolant to aid in temperature control.

All plugs are sealed by two O-rings compressed between the plug and the casing near the top of the shield plug. Another gasket, compressed by a retaining ring, is incorporated in the top lip of each shield plug. The retaining ring may be removed and the gasket serviced without removal of the shield plug.

2.2.1.1.5. Grid Plate

Mounted 18 inches above the bottom of the core tank is the grid plate which has holes drilled into it at each fuel assembly location. It also has indentations to position the moderator cans. The grid plate is 135 inches in diameter and is made of type 304 stainless steel. The purpose of the grid plate is to support the active core region and to provide flow distribution for the coolant that passes up through the active core past the fuel assemblies, and to provide a barrier between the moderator coolant and the primary coolant.

The primary coolant flow is directed to the bottom of the core tank, and then passes up through the active core via holes drilled into the grid plate as shown in Figure 2.8. The moderator coolant travels between the top of the grid plate and the bottom of the moderators assemblies and then up through the moderator cans through the moderator cooling channels. Figure 2.8 does not show all of the coolant channels, but is intended to show the flow path for the moderator and primary sodium coolant. The "outlet plenum" is simply a pool of sodium at the top of the active core where the primary sodium and moderator coolant collect before passing out of the core tank through the coolant outlet pipes. [Refs. 6,7]

2.2.1.1.6. Cover Gas Volume

The space above the active core and sodium pool inside the core tank and below the shield plug is a helium filled region known as the "*cover gas*". This gas volume allows

the sodium to expand and contract with changes in temperature during reactor operation. Helium was chosen as a cover gas because it does not become activated by neutrons from the core. Helium is also used as a cover gas over the primary sodium fill tank. The sodium fill tank cover gas volume and reactor cover gas volume are considered as the primary cover gas system. The fill tank and reactor cover gas volumes are connected by a 2 inch pipe, shown in Figure 2.6 as the "crook neck" pipe identified as "Fill Tank Vent Line". The cover gas is normally maintained at 2-3 psig pressure in order to prevent primary sodium pump cavitation. [Ref. 8,9,10]

When a decrease in pressure is required to the cover gas, the reactor cover gas may be vented to the waste gas decay tanks located north of the reactor building. Makeup gas and helium used to increase pressure of the cover gas are provided by the helium supply system, stored in steel bottles located outside the reactor building.

2.2.1.2. Active Core

As shown in Figure 2.6, the reactor core is located inside the core tank, and is made up of a matrix of hexagonally shaped moderator cans which contain control elements, uranium fuel assemblies, and temperature and sodium level measurement devices. Figure 2.9 shows a top view of the core, and identifies the location of fuel elements in the core.

The primary structural elements of the active core are the moderator assemblies, called *moderator cans*, which contain the graphite moderator. The purpose of the moderator in a nuclear reactor is to slow down the neutrons given off in the fission process to a speed where they can be absorbed by other uranium atoms, and thereby get better "mileage" out of the uranium fuel. At the outer edges of the core are "*reflector cans*", whose purpose is to bounce back the neutrons that reach the outer edge of the core before the neutrons escape the core region. The reflector cans do not have coolant channels through the center of the can, as do the moderator cans.

The moderator/reflector cans are slightly less than 11 inches across the flat part of the hexagon, and the spacing between cans is 11 inches on center. This spacing is sufficient to provide a gap of approximately 0.17 inches between cans to allow coolant to flow between the cans to remove heat generated in the graphite.

There are a total of 119 cans in the reactor. A diagram of the construction of a moderator can is shown in Figure 2.10. The moderator cans are approximately 10 feet in height, and are wrapped with 0.035 inch thick zirconium metal sheets that have "dimples" to help maintain clearance between cans. The top and bottom use 0.10 inch thick zirconium metal sheets. The "wrapper" of zirconium is also referred to as "moderator cladding" as opposed to the stainless steel wrapper around the uranium fuel, which is called "fuel rod cladding". Sometimes the fuel wrapper is referred to as "cladding".

The purpose of the cladding is to protect the contents within the wrapper. In the moderator, it is necessary to prevent sodium from penetrating into the graphite. In the case of the fuel, the cladding acts as a barrier to prevent any radioactive fission products from getting into the sodium coolant.

Each moderator can has a central channel to allow coolant to flow up through the channel to remove heat generated by the nuclear reactions taking place in the uranium fuel. To prevent contact between the sodium coolant and the graphite, a 0.035 inch thick zirconium tube that is 2.8 inches (inside diameter) is welded to the top and bottom of the moderator can.

When the moderator cans were first built and tested prior to installation in the core, it was discovered that gases were given off by the graphite due to impurities in the graphite. As the temperature was increased, the gases expanded, creating internal stresses on the moderator can side walls which could possibly damage the moderator can, or cause it to distort its shape and to become stuck in place. [Ref. 39] To alleviate this problem, a zirconium "vent tube" was built into each moderator can. A zirconium "snorkel tube" is attached to the can tops and extends into the cover gas region above the top sodium pool. When the moderator cans were manufactured, the snorkel tube and vent tubes were sealed with a fusible plug after the can was evacuated and filled with an inert gas. This provided gas-tight integrity of the cans until they were installed into an inert atmosphere in the reactor. The fusible plug was designed to melt at a temperature of about 450 °F and be collected in the "condenser cup" shown in Figure 2.10. The purpose of the cup is to collect any sodium vapor which might pass down the snorkel tube. By this design, any pressure built up inside the moderator can during operation in the reactor could relieve

itself to the cover gas region and prevent damage to the moderator can. The tube is only open at the bottom of the moderator can and at the end of the snorkel tube in the cover gas.

The bottom of the moderator cans are equipped with a socket adapter that mates with the grid plate located in the bottom of the reactor. A spherical step on the socket fits into a tapered dimple in the grid plate and thereby restricts moderator coolant from passing through the grid plate. The adapter contains slots to allow the coolant in the moderator plenum to travel up the moderator coolant channels. Similarly, the fuel channel adapters are open at the bottom, and sealed at the moderator plenum section, thereby allow only sodium coolant to pass into the fuel channel.[Refs. 2,6,7]

2.2.1.2.1. Fuel Elements

Core I of the SRE contained a total of 43 fuel elements, also referred to as *assemblies*. A typical fuel assembly is shown in Figure 2.11. A basic fuel assembly is made up of the fuel rods, a hanger rod with hold-down tube, an orifice plate, and retaining hardware with guide vanes. The fuel assembly is attached to a shield plug, lowered through openings in the reactor top shield, and inserted into the core region of the reactor.

Each fuel element consists of 7 rods containing 12 uranium fuel slugs, making up the fuel element. The walls of fuel rods are 0.10 inch wall thick, and are also referred to as "*fuel cladding*". The 12 fuel slugs are ¾ inches in diameter, six inches long, and are held in place by a sodium-potassium compound called "NaK". (An acronym made up of the chemical symbols for the two elements, which come from their latin names – sodium [**Na**trium] and potassium [**K**alium].) The NaK fills the annulus between the uranium fuel and the stainless steel cladding. The six outside rods of the fuel element are spirally wrapped with stainless steel wire to prevent the rods from touching each other or the coolant tube in the moderator can. [Ref. 12]

At the top of each fuel element is a gas space of 110 cubic centimeters (cc) for thermal expansion of the NaK, and for collecting radioactive fission gases which may not be retained in the fuel during irradiation in the reactor. Fission gases collected in the annulus or gas space of each fuel rod are also referred to as the "*gap activity*".

Each fuel element has an adapter on the bottom end which serves as a guide for placing the assembly and also has an orifice plate to limit the sodium flow through that particular channel. The purpose of the orifice plate is to adjust the flow through each coolant channel to provide an even temperature distribution across the width of the core. The center of the core has the highest neutron population, and therefore the highest heating rate. By adjusting the flow rate through individual fuel channels, it is possible to change the amount of heat removed in each individual fuel channel, and thereby attain a more uniform exit temperature across the diameter of the core. The flow rate through the center of the core is designed to be 5 feet per second in order to remove a large quantity of heat, while at the outside edge flow rate might be much less, in order to keep the temperature at the edge of the core closer to that at the center. [Refs. 2,6,7]

2.2.1.2.2. Safety Control Rods

There are 4 safety control rods that perform a safety shutdown function called a reactor "*scram*". A scram is an automatic shutdown of the reactor by dropping the safety control rods into the core. Normally, the automatic control circuits initiate a reactor scram, but the operator may also initiate a scram from the reactor control console. The safety control rods contain an alloy of boron and nickel. Boron is an excellent absorber of neutrons, and is used in reactors to provide a neutron "sponge" function. The safety rods are contained in a stainless steel pull tube contained within a thimble assembly that extends from the top service area of the rotatable shield plug to just below the active region of the core. The pull tube is raised or lowered from the reactor core by means of a screw and-nut arrangement. The nut, attached to the pull tube, is prevented from moving by guides that move in flutes machined into the wall of the thimble assembly above the core. A control rod drive motor located on top of the shield plug turns a screw which rotates through the nut, raising or lowering the pull tube.

The safety rods are operated by a high-speed drive that retracts the pull tube at 3.75 ft/min. A "hold" magnet is built into the thimble which can be released at any time during retraction from the core or after full withdrawal. When the magnet is disengaged, such as by a scram signal, the safety rods fall into the core by gravity.

The safety rods are fully withdrawn prior to reactor startup. [Ref. 2]

2.2.1.2.3. Shim and Regulating Control Rods

There are 2 control rods that provide a "*shim* "or very fine adjustment to the neutron population in the core. Two other control rods provide "*regulating*" adjustments. The shim rods and regulating rods also contain boron-nickel alloy.

The shim rods have dual speed drive motors that produce 0.29 ft/min for fine control rod adjustments. The regulating rods have "high speed" travel at 3.75 ft/min. The high speed drives are prevented from traveling more than 7 inches in either direction by the automatic control system. The operator may move the rods in or out in a shutdown or startup operation by pressing a switch on the control console. The shim rod and regulating rods do not provide a scram function.

The shim rods and regulating rods are the main control mechanism for bringing the reactor to initial criticality during startup. Increase or decrease of their position in the core during power operation regulates the neutron population in the active core. The shim rods and regulating rods are never fully withdrawn from the core, as are the safety control rods. [Ref. 2]

2.3. Support Systems and Equipment

2.3.1. Fuel Handling Cask

Provisions are made to remove a fuel assembly for examination and to insert experimental assemblies into the active core region when the reactor is shut down by use of a machine called the *fuel handling cask*. The SRE documents and drawings may also refer to the cask as the Fuel Handling Machine (FHM) or "coffin".

The fuel handling cask is 35 feet high, weighs 50 tons, and is moved around the reactor high bay area by the overhead 75 ton capacity crane. It is a lead-lined shielded device, which contains hoisting devices used for fuel assemblies and other reactor components. Difficulty in fuel assembly removal is compounded by the fact that an inert atmosphere must be established and maintained during all fuel handling operations. This requires that the cask form a gas-tight seal to the top shield to prevent oxygen from reacting with the sodium coolant, and as a precaution to prevent a sodium fire in the reactor.

A schematic diagram of the operational configuration of the fuel handling cask is shown in Figure 2.12. Access to the core is made by first removing the retainer ring and gasket and any electrical or instrument connections on the particular shield plug. The fuel handling cask is then positioned over the shield plug. The cask is positioned by marks on the reactor loading face to provide an accurate positioning. A pneumatic ram inside the cask pushes down on the shield plug to seat the O-rings tightly. Then a lead shield skirt is pneumatically lowered down to the top shield surface. A gas lock at the lower end of the cask, where the seal has been made to the shield casing, is then evacuated by means of a vacuum pump. After verification of a gas-tight seal, helium is admitted into the cask, and the cask is pressurized to 3 psig. The exhaust of the vacuum pump is connected to header 506 in the reactor building through a flexible hose connection. Normally, header 506 is connected to the suction tank, and any gases collected during operation of the fuel handling cask are drawn into the holdup tanks by the waste gas compressors.

Once the pressure is equalized between the reactor cover gas and the fuel handling cask, the planned operation is ready to be performed. If a fuel assembly is to be removed, for example, one of two grappling devices is lowered and attaches to the fuel assembly's shield plug. After latching onto the shield plug, the plug and assembly are lifted up into

the cask. A rotating device in the cask turns 180 degrees to store the old assembly while simultaneously placing the new assembly in position for insertion into the core. Each assembly is attached to its own individual shield plug, and records are kept so that each assembly's location in the core may be checked by referring to the shield plug number assigned to that assembly. After the new assembly is placed in the core, the grappling mechanism is retracted into the cask, and the pneumatic latching device then presses down on the O-ring seal, and evacuates the cask again to purge any radioactive gases. Fresh helium is drawn into the gas lock. The lead skirt is raised, and the gas lock seal is broken to the shield casing. The cask is now free to transport the assembly to the fuel storage area, or to move to a new position. [Ref. 6, 11]

2.3.2. Radioactive Vent Systems

Gases used for cover gas in sodium systems, such as the reactor and fuel assembly wash cells, are potentially radioactive. The design of the SRE was to collect all such gases in a tank, compress the gases and put them in a gas holdup tank until they had decayed sufficiently to allow discharge to the environment. The radioactive vent system is designed to collect such gases, and collects all potentially radioactive gases from the plant.

A schematic diagram of the Vent Gas System is shown in Figure 2.13. The Vent Gas System, actually consists of two separate systems upstream of the suction tank. Helium cover gas is used for all areas that might be directly exposed to a free sodium surface. A nitrogen cover gas is used for all areas that required a cover gas, but is not in direct contact with sodium. [Ref. 15, 2]

Figure 2.13 is a simplified diagram of the vent system as it existed in July, 1959, as could best be determined from available records. Appendix B contains the main figures from which Figure 2.13 was derived, together with excerpts from operating procedures and valve lists for the vent system and documents that provided abbreviated diagrams of the system.

The helium supply system supplies helium to all components and systems that contain sodium or can potentially come into contact with sodium used in the reactor. The nitrogen supply system provides nitrogen to all other components and systems that must

have an inert environment. This includes vaults, which are sealed equipment compartments.

The waste gas vent system collects, directs and removes gas from two sources – those that are "probably" contaminated, and those which are "possibly" contaminated. The vent gas headers are separated according to their potential contamination. The gas sources were described as follows: [Ref. 15]

A. Normally Containing Radioactivity

- Reactor and primary fill tank cover gas
- Primary system cold traps and vents
- Cleaning cell vents
- Fuel Handling Cask service connections
- Radioactive liquid waste holdup tank vents
- Hot cell radioactive gas service connections
- Hot trap vents

B. Normally Non-radioactive

- Primary system pump vents
- Reactor Drain line freeze trap vents
- Helium supply system relief valves for following:
  - Block valves
  - Double walled pipes
  - Control & safety rods
  - Disposable cold traps
  - Core tank cavity
  - Main & aux piping and equipment
- Nitrogen supply system relief valves as follows:

- Main & auxiliary primary system galleries
- Fill tank vault
- Disposable cold trap vault
- Insulation cavity

- Nitrogen System Gallery atmosphere discharge
- Insulation cavity vent (normally closed)
- Core tank cavity vent (normally closed)
- Block valve vents (normally closed)
- Double walled pipe vents (normally closed)
- New fuel storage cells

The Vent Gas System consists of five major headers, the vent compressor area, and the waste gas decay tanks. Each portion of the system is described below. It is suggested to refer to Figure 2.13 for an understanding of the vent gas system.

2.3.2.1. Header 492

Header 492 is the relief valve collection header for all primary systems using helium in the pressure control stations.[Ref. 6] Relief valves are provided to prevent overfilling of the blanket gas in various components and areas. Header 492 is connected to the following:

- reactor insulation cavity relief valves,
- core cavity relief valves,
- primary cold trap relief valve,
- auxiliary double wall pipe relief valves,
- safety rod relief valve,
- shim rod relief valve,
- main double wall pipe relief valve,
- main gallery relief valve,

- primary fill tank vault relief valve,
- sodium service vault relief valve,
- auxiliary gallery relief valve,
- NaK system relief valve.

Some system vents may also be opened to Header 492 via manually operated valves during initial startup operations. These valves are normally closed during power operation. When the vent is opened, there is helium flow through the supply system. When the vents are closed, there is no longer any gas flow through the system, and the system is confirmed closed. Header 492 also is used to vent the following components via manually operated valves:

- reactor insulation cavity vent,
- core tank cavity vent,
- Valve V104 vent,
- auxiliary double wall pipe vent,
- main double wall pipe vent,
- hot trap A&B freeze trap vents,
- primary cold trap vent.

Header 492 is divided into two sections in the reactor building – the high bay branch header (490) and the secondary branch header (489). These two headers join Header 492 at the northwest corner of the reactor building which continues underground to the compressor suction tank. North of the primary fill tank vault, branch header 375 (gallery and primary fill tank vault relief valves) joins Header 492.

A main block valve, V492C, is installed in Header 492 before the suction tank and is located in the valve pit south of the compressor vault. The block valve is used to isolate Header 492 from the suction tank when the reactor is not operating. Normally, the block valve is open when the reactor is at power.

Header 492 normally does not have flow through the pipe, and is provided as a pathway for the relief valves to vent to the suction tank when the reactor is at power.

2.3.2.2. Header 497

Header 497 is the header that services the reactor cover gas and primary fill tank atmospheres. Vent line 496 connects the primary fill tank to the reactor cover gas volume through a vapor trap, and contains isolation valve 496. Valve 496 and valve 137 are normally open during power operation. [Ref. 16, 6]

Header 497 is connected to the following:

- Reactor and primary fill tank cover gas
- Primary fill tank relief valves and vents
- Primary system cold traps and vents
- Hot trap vents
- Disposable cold traps and vents

2.3.2.3. Header 506

Header 506 serves as a vent for the liquid waste system and also the reactor high bay area four vent connections for the fuel handling machine. The four vent connections are located on the east and west side of the loading face, west of the storage cells, and south of the wash cells. From the wash cell area, the header goes underground to the radioactive sump tank, and terminates at the compressor suction tank. It is connected to the bottom of the compressor suction tank, since it serves as a drain for the suction tank and compressor system. [Ref. 6]

Header 506 is connected to the following:

- Fuel Handling Cask service connections
- Liquid waste vents
- New fuel storage cells

- Cleaning cell vents
- Radioactive liquid sump tank vent
- Compressor suction tank drain (normally closed)

Header 506 has a bypass around the waste gas suction tank such that vent gases may go directly to the stack. An inline radiation detector monitors for radioactivity, and a solenoid valve diverts any radioactive gases to the suction tank and subsequently to the waste gas holdup tanks if the gases exceed a preset limit.

2.3.2.4. Header 520

Vent Header 520 is the vent header for the disposal of radioactive gases resulting from hot cell operation. This header is closed unless the hot cells are in operation. Opening and operation of the hot cells requires the approval of the shift supervisor, and is controlled by procedure SRE-415.

Header 520 had no impact or involvement with the fuel damage incident of Run 14.

2.3.2.5. Header 526

Header 526 is the vent header for the helium supply system. It provides for relief of the main primary piping helium pressure control stations and the auxiliary helium pressure control stations.

Header 526 was not involved in the events associated with Run 14 of the SRE.

2.3.3. Waste Gas Compressors and Decay Tanks

Gases that are considered potentially radioactive, e.g., headers 492 and 497 are routed to a "*suction tank*" located approximately 170 feet north of the reactor building in a shielded pit.[Ref. 11] As shown in Figure 2.13, two compressors take suction on the suction tank and force the gas into one of four decay tanks. The suction tank is maintained at a negative pressure of -1.5 to -4 psig, so that it provides sufficient vacuum to always induce flow from the connected headers. The negative pressure also ensures that any leakage is into the vent system. Should the compressor tank start to lose vacuum, one or both compressors automatically start. If both compressors cannot maintain vacuum, and the

pressure reaches a pressure of -0.3 psig, an alarm sounds (BHPA602) in the control room. [See Dwgs. Appendix B]

The compressors operate individually, but are set to switch duty so that each compressor operates an equal amount of time. By selection of valves, the operators may choose which holdup tank the compressors will fill. Each holdup tank holds 2700 standard cubic feet of gas at 100 psig. The tanks are monitored by health physics personnel, who sample the contents, and when the radioactive gases have decayed to a sufficient level, a calculation is done to determine the release rate for the tank, and the tank is then vented to the atmosphere via the plant stack.

2.3.4. Plant Stack

The plant stack is the release point for the vent system, including the decay tanks. The stack is 3 feet in diameter. When venting a decay tank is required, the tank is vented through line 537, as shown in Figure 2.13. The rate of discharge is monitored and a 25,000 cfm dilution fan further reduce the concentration of gases released to the environment. Records indicate that the normal release rate from a decay tank is 145 cubic feet per hour (approximately 2.5 cfm).[Ref. 23] Lower flow rates may be used if the activity is higher.

The gas in the stack is also monitored for radioactivity by a radiation detector which draws a small flow out of the stack and checks the level of radioactivity. Preset limits are built into the detection system, and if the limits are exceeded, the system closes solenoid valve SV-604, which terminates the discharge from the decay tank. The limit for the stack monitor is 5.0×10^{-7} μCi/cc [Ref. 23] The monitoring point is 12 feet above the vent line entry point.

The original design of the SRE vent system included the capability for individual vent lines to go directly to the stack, and radiation detectors in the individual lines would automatically divert the flow from the stack to the suction tank if radioactivity was detected in the vent. The problem with this design was that when such a signal was received, the solenoid valve required to open to the suction tank would stick closed, and the venting would be terminated. Relief valves were then added to the vent system, to prevent overpressurization of a vent line when the vent line was being filled by operation

of the helium/nitrogen supply system and the solenoid valve failed to open. In December, 1957, recommendations were made to install a bypass line around the radiation monitors. With the bypass open, the alignment of the vent lines in "Category A" discussed in Section 2.2.2. were to the suction tank.[Ref. 11] The problems with the automated vent system were thus solved by the administrative directive that all vent headers were to be aligned to the suction tank for normal operation. [Refs. 5,7,8,9,11,13,14,16, 28]

2.3.5. Reactor Building Ventilation

The reactor building ventilation system is designed to move air toward potentially contaminated areas. This is accomplished by keeping the hot cell area and the reactor high bay area pressure more negative than the contiguous air spaces, such as the conference room, control room and administrative offices. Makeup air is brought in from the outside, combined with recirculated air in the offices and control room to maintain a positive pressure with respect to the potentially contaminated areas. Independent of the makeup air to the administrative areas, 5 air changes an hour are brought into the high bay area through roughing filters in the low bay roof and is discharged into the high bay volume. The two high bay supply fans are rated at 7500 scfm and are equipped with dampers to maintain the negative pressure required. Two Exhaust fans in the high bay area are located on the reactor building roof, and draw air out of the building at 12,000 scfm each. Interlocks in the ventilation system prevent the supply fans from starting unless the exhaust fans are already running, in order to prevent pressurization of the reactor building.

The original design included filters on the reactor building roof to remove particulate material, but the system was continually compromised by water vapor collecting in the filters. The filter system was eventually bypassed entirely.

The high bay area ventilation exhausts directly to the environment, and not through the plant stack.

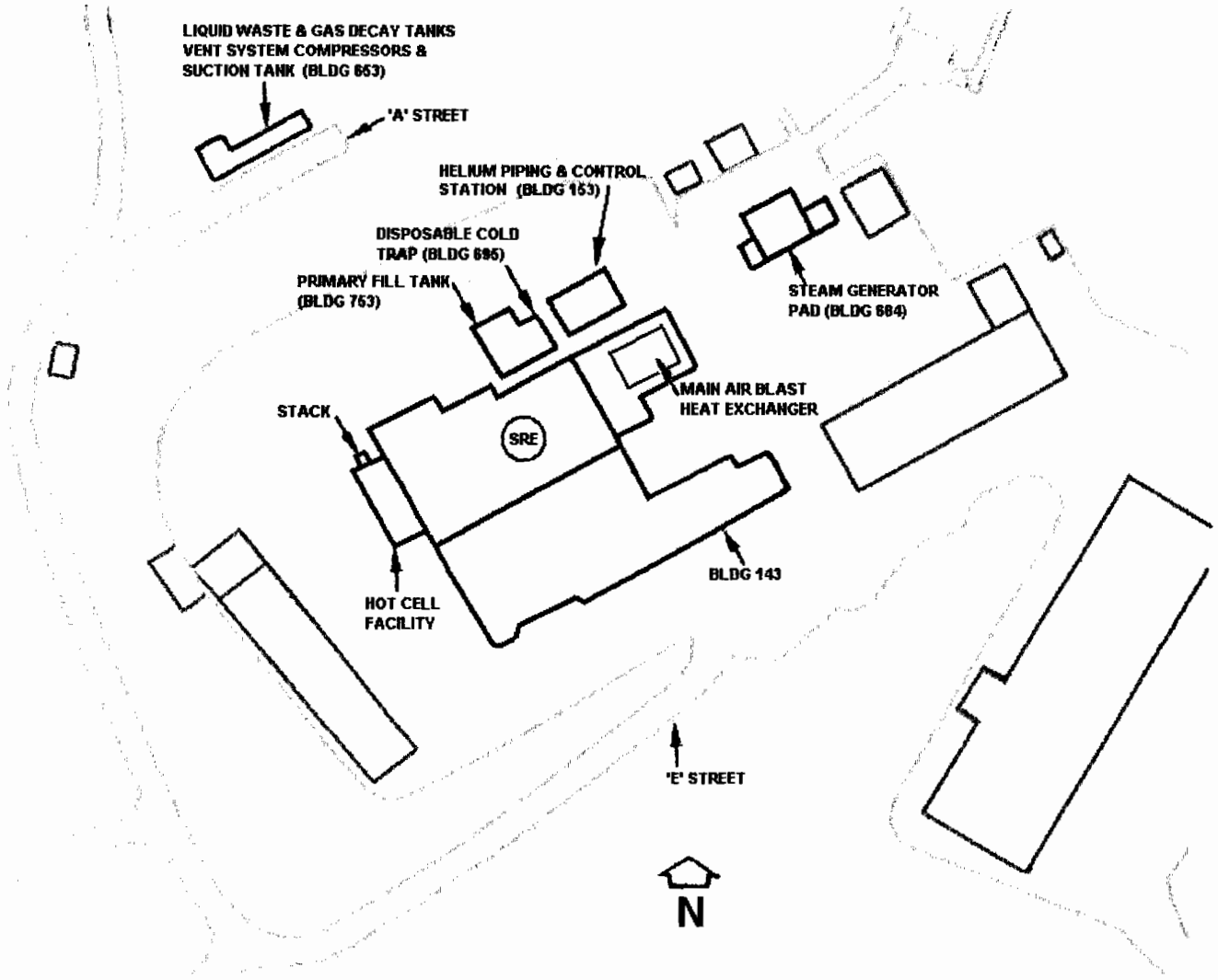


Figure 2.1 Sodium Reactor Experiment Site Plan

2 - 25

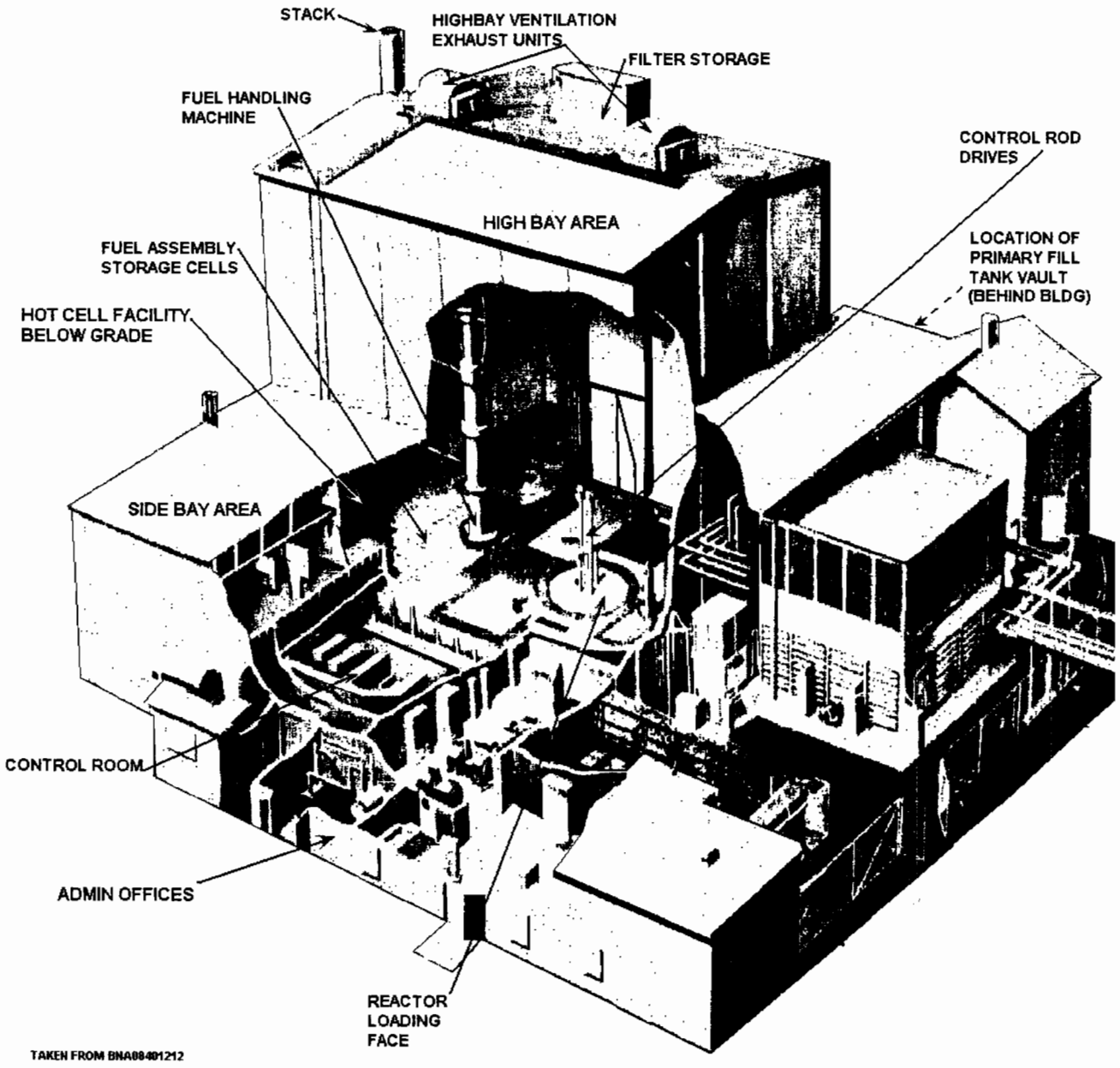


Figure 2.2 SRE Reactor Building View

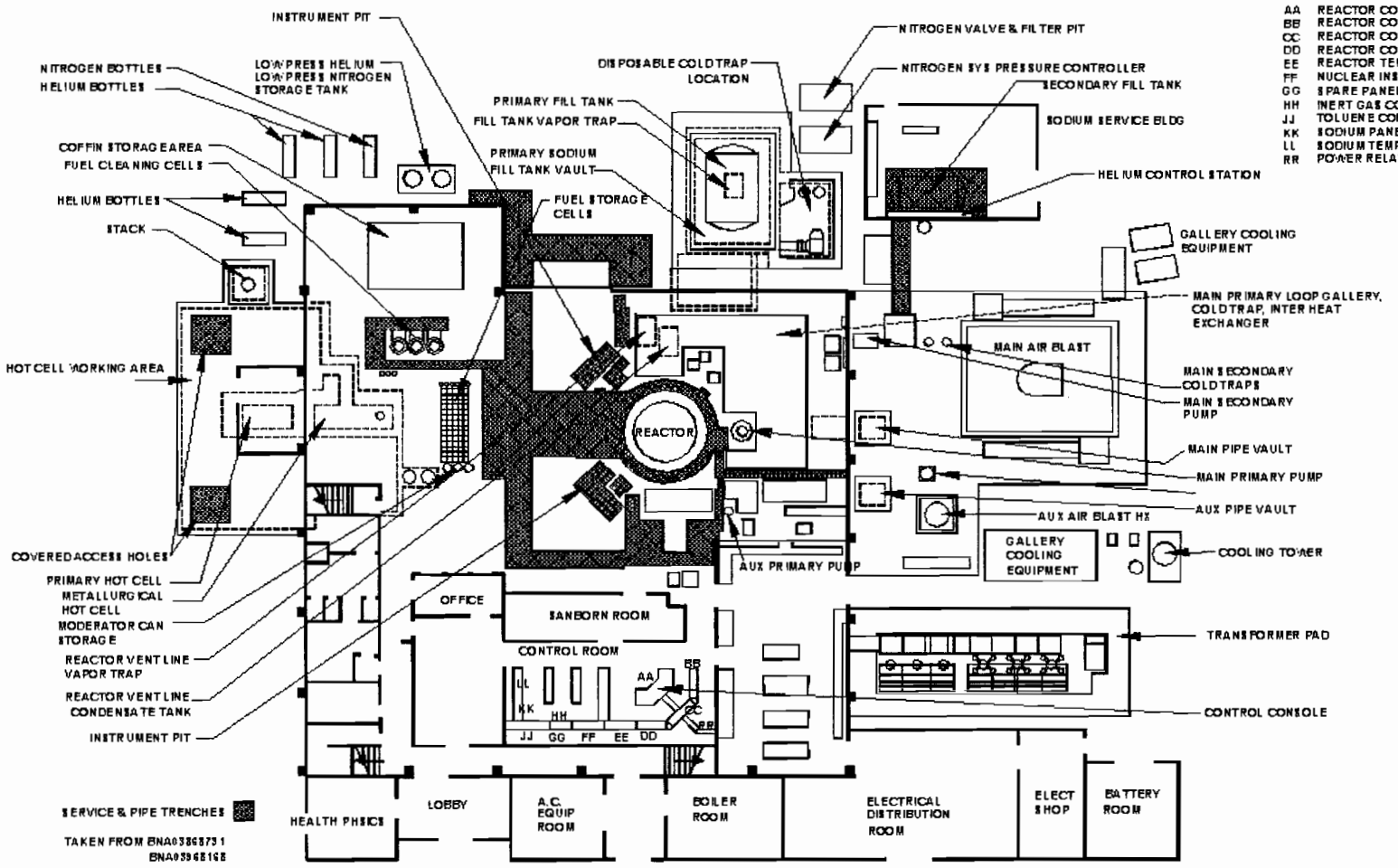


Figure 2.3 SRE Reactor Building Plan

2 - 27

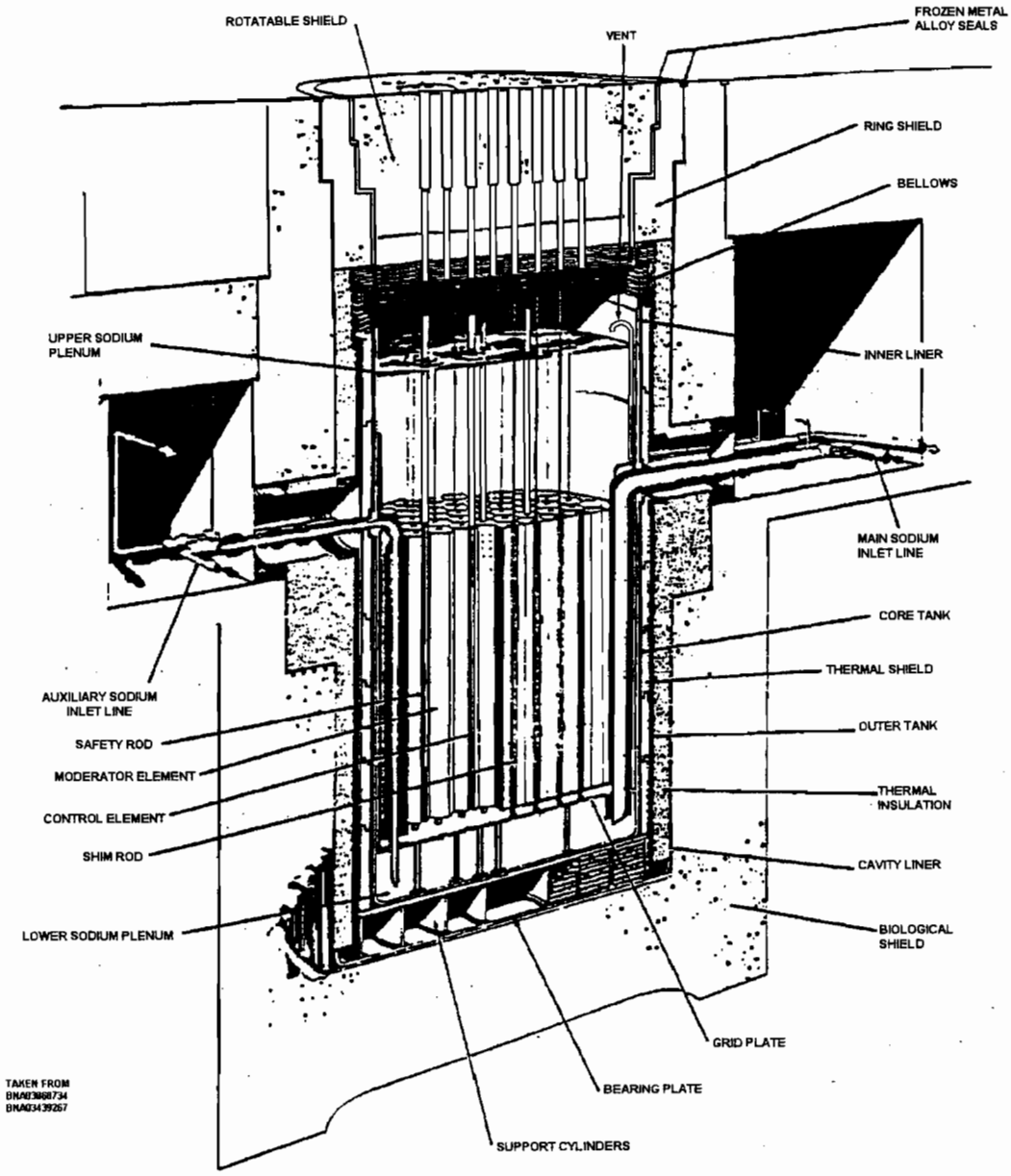


Figure 2.4 Cutaway View of SRE Reactor

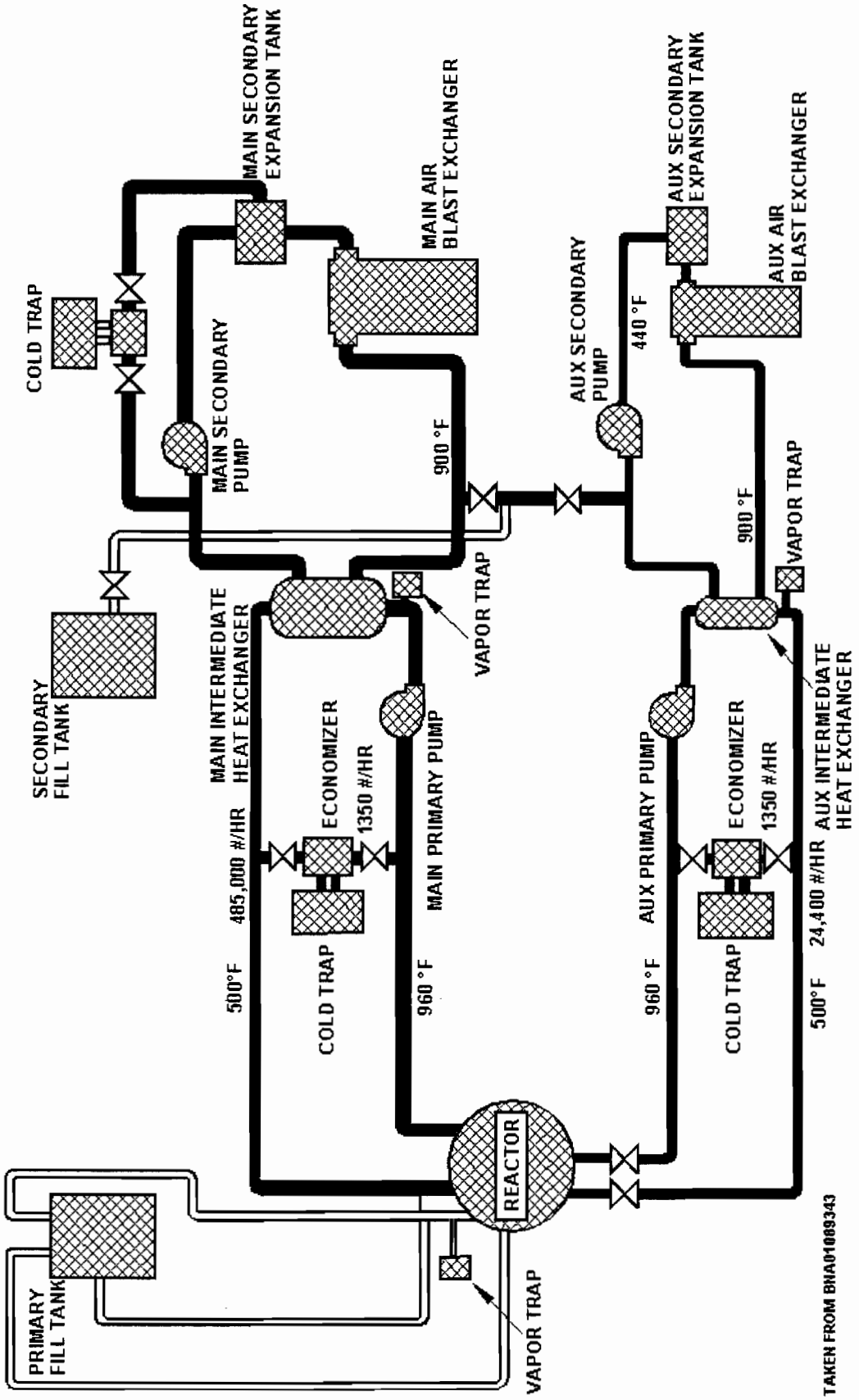


Figure 2.5 SRE Sodium Cooling Systems

2 - 29

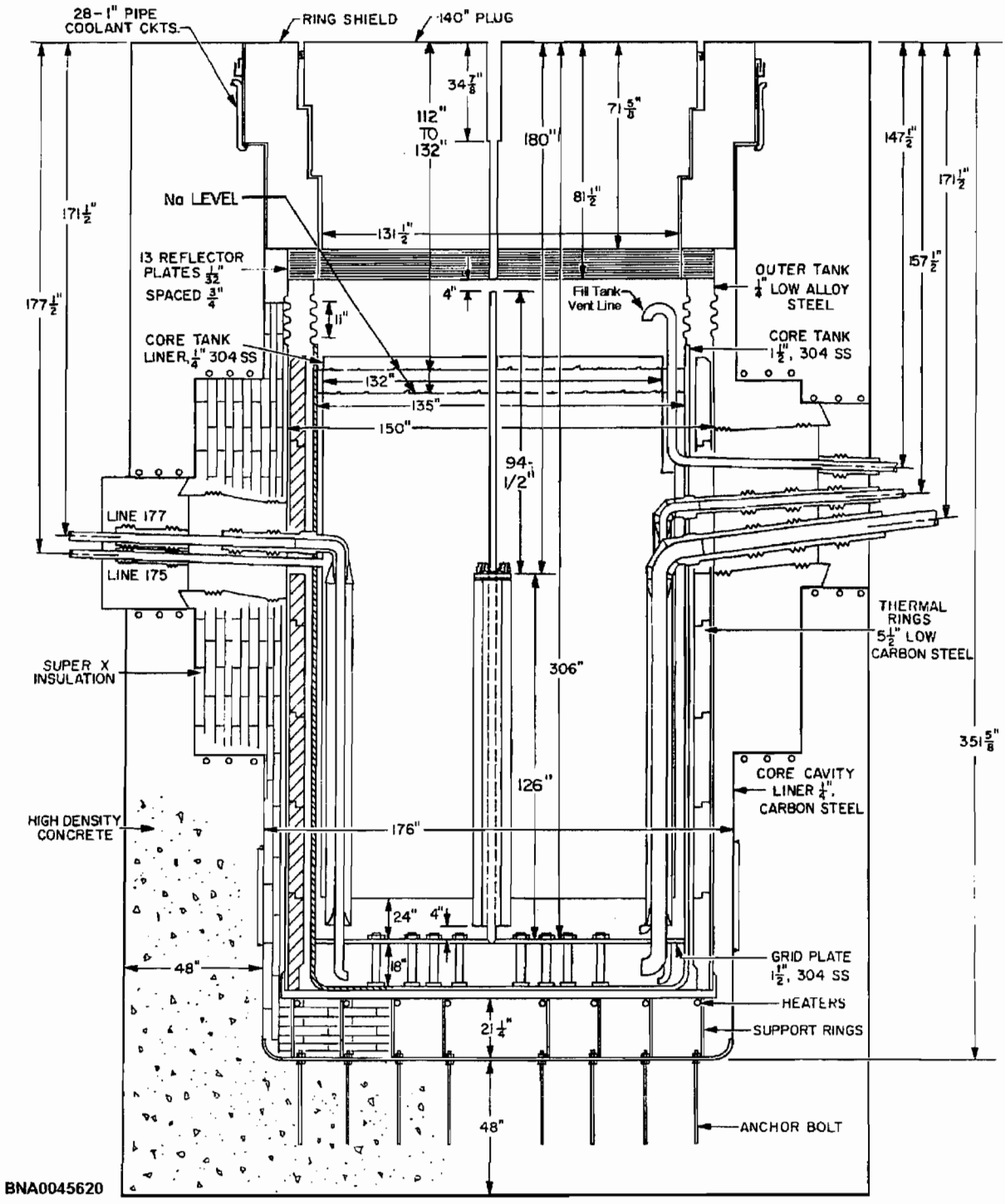


Figure 2.6 SRE Reactor Cross-Sectional View

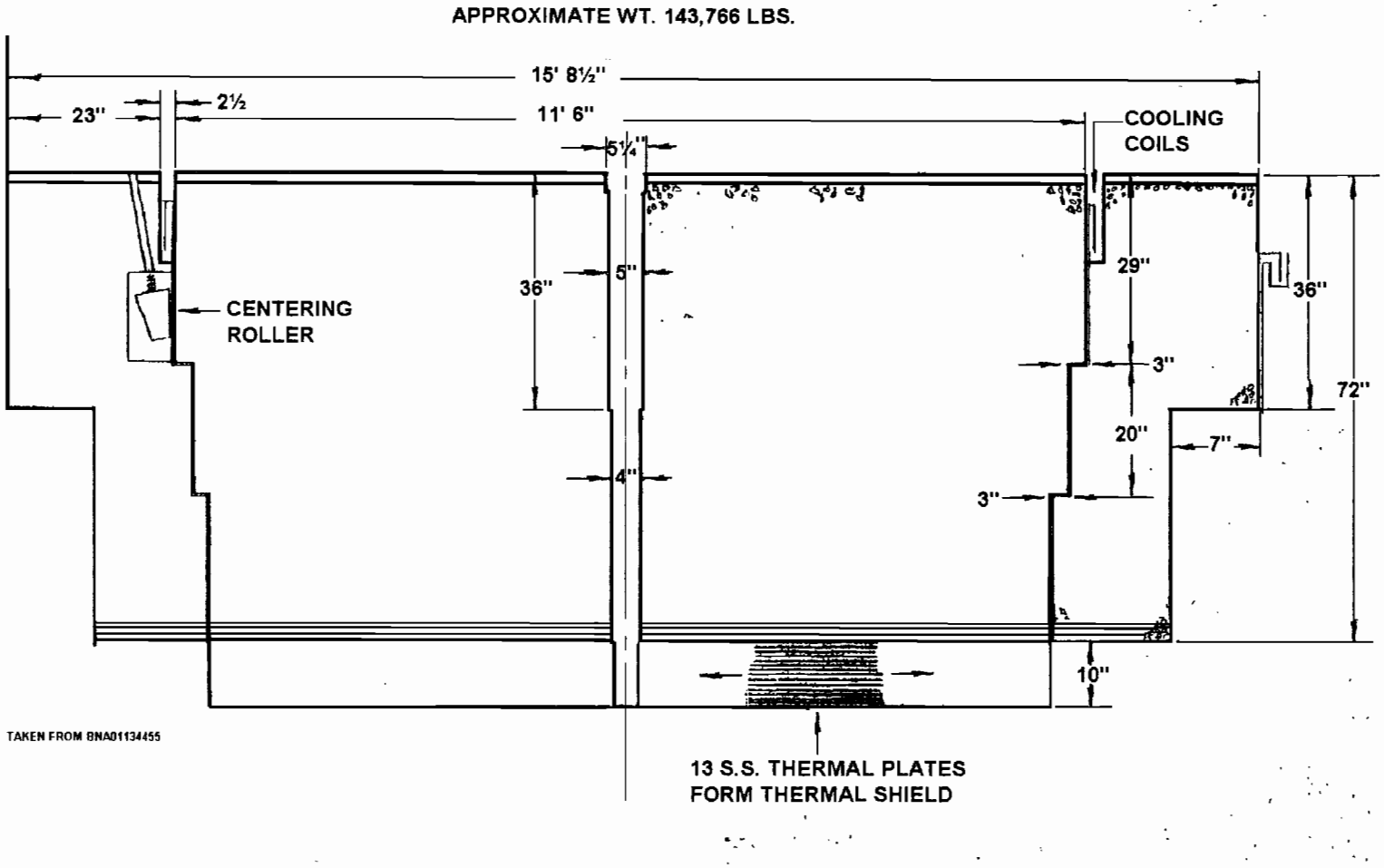


Figure 2.7 SRE Top Shield Plug

2 - 31

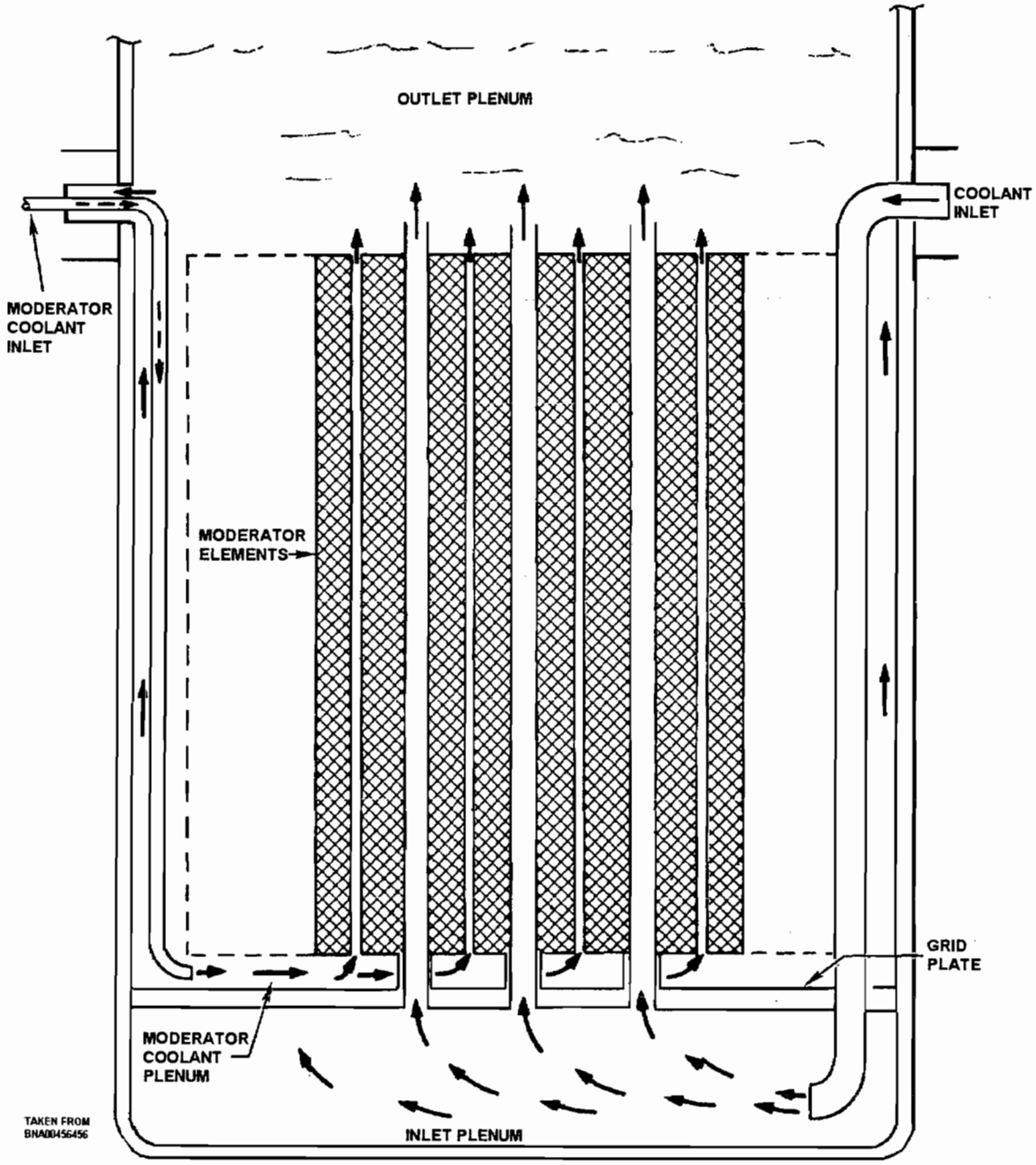


Figure 2.8 SRE Primary Coolant and Moderator Coolant Flow Paths

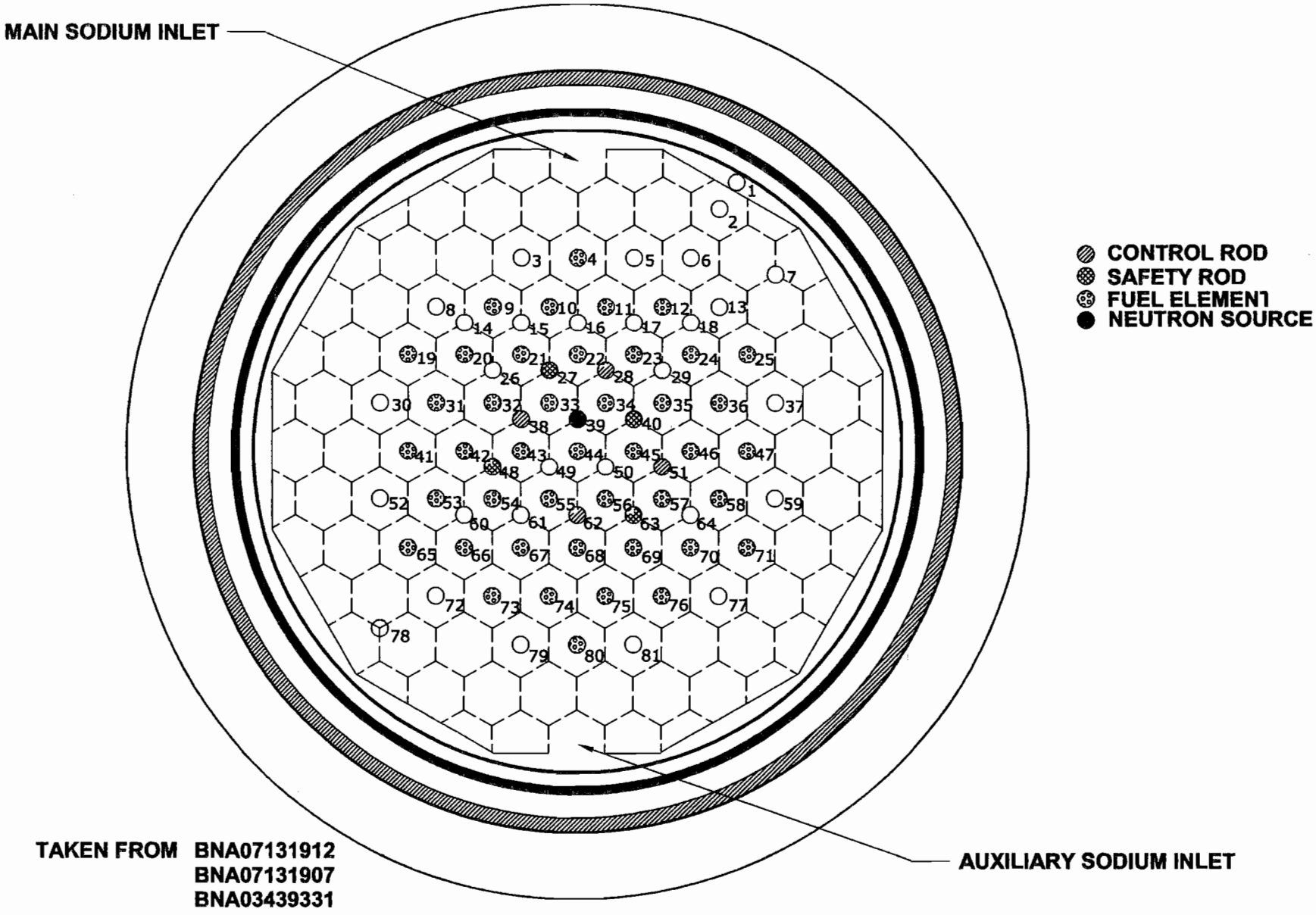


Figure 2.9 SRE Core I Fuel & Control Rod Locations

2 - 33

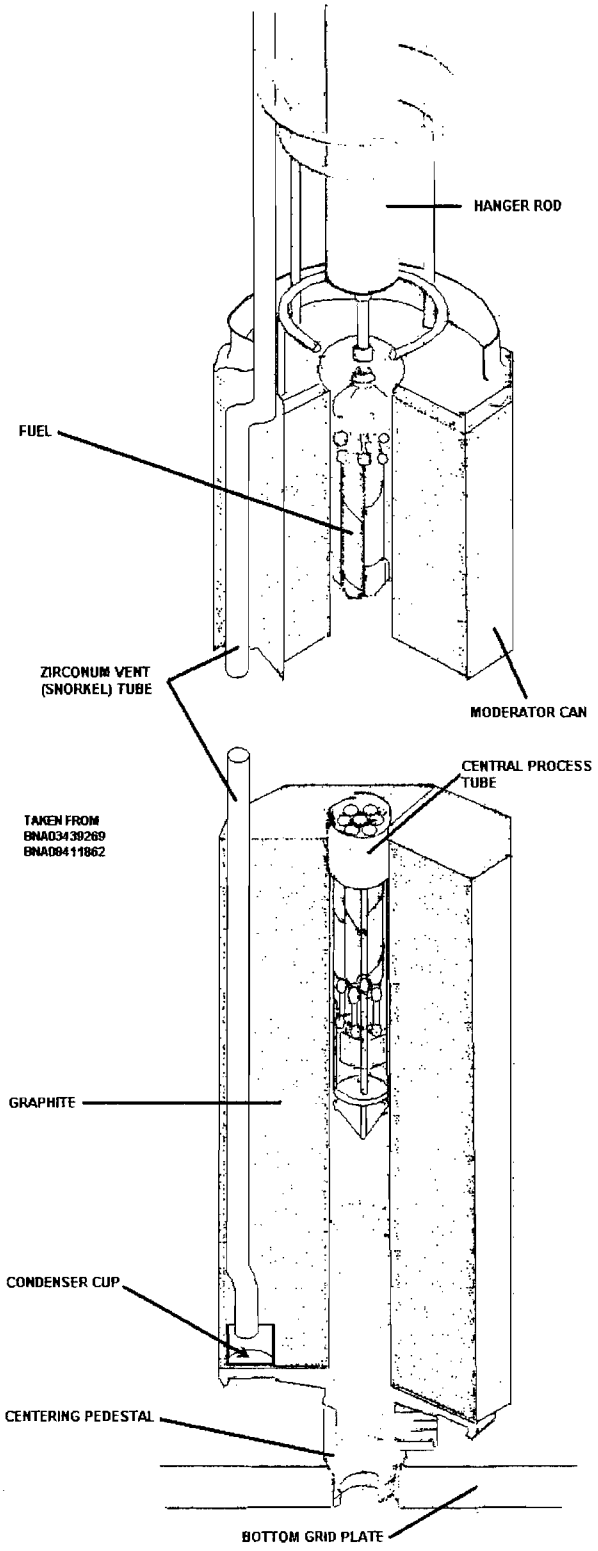


Figure 2.10 SRE Moderator Can Design

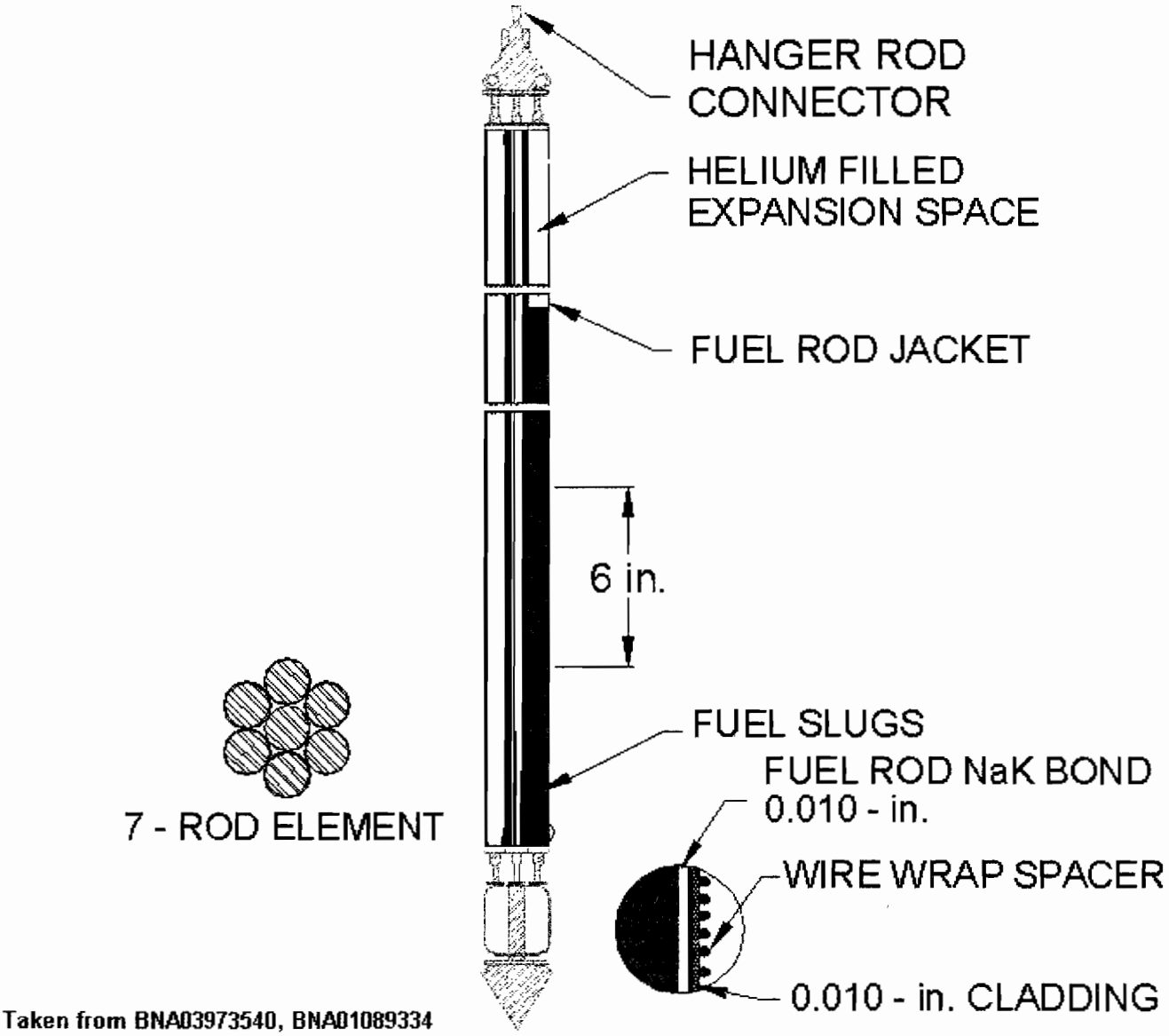


Figure 211 SRE Core I Fuel Assembly.

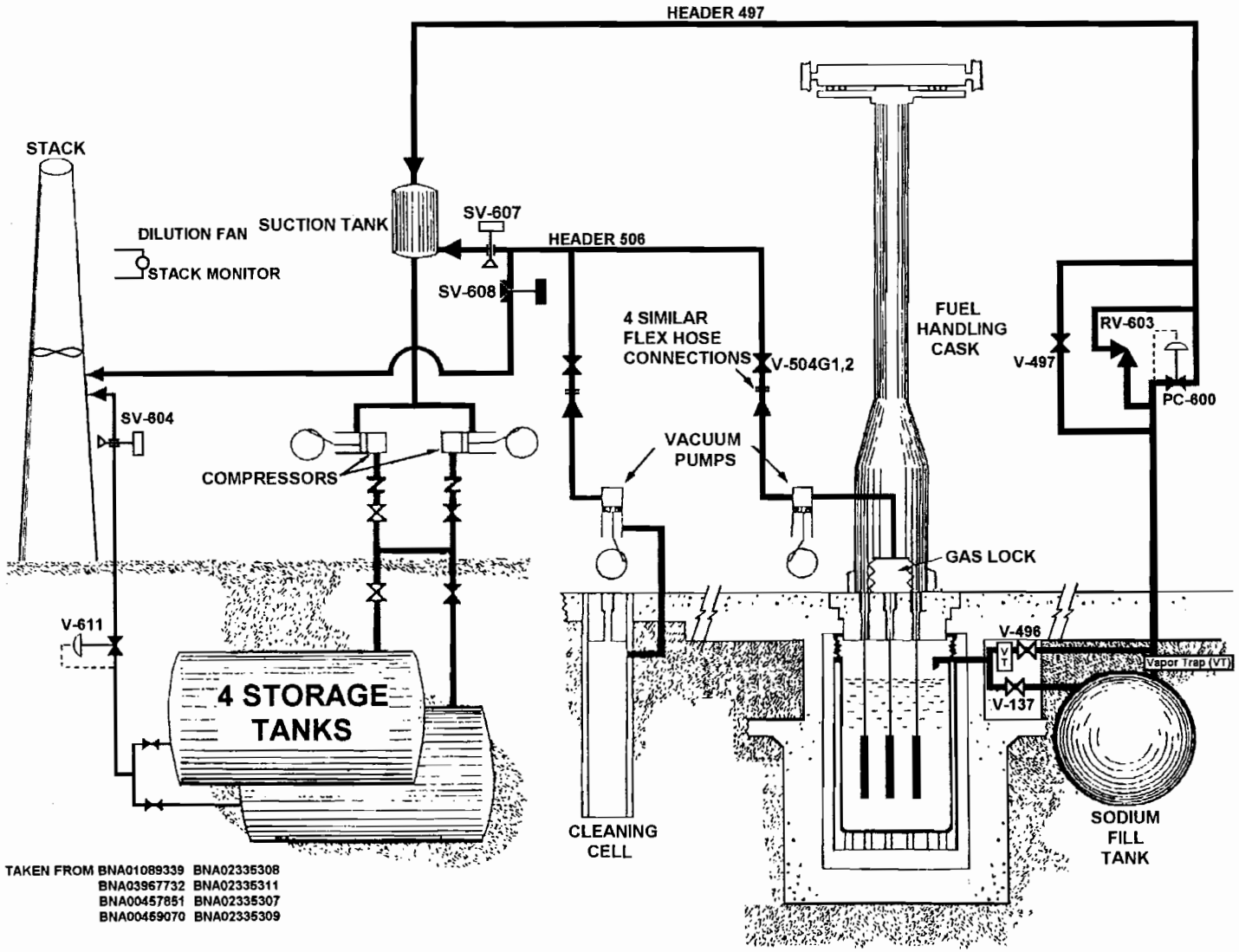


Figure 2.12 Fuel Handling Machine Vent Alignment-(Normal)

2 - 36

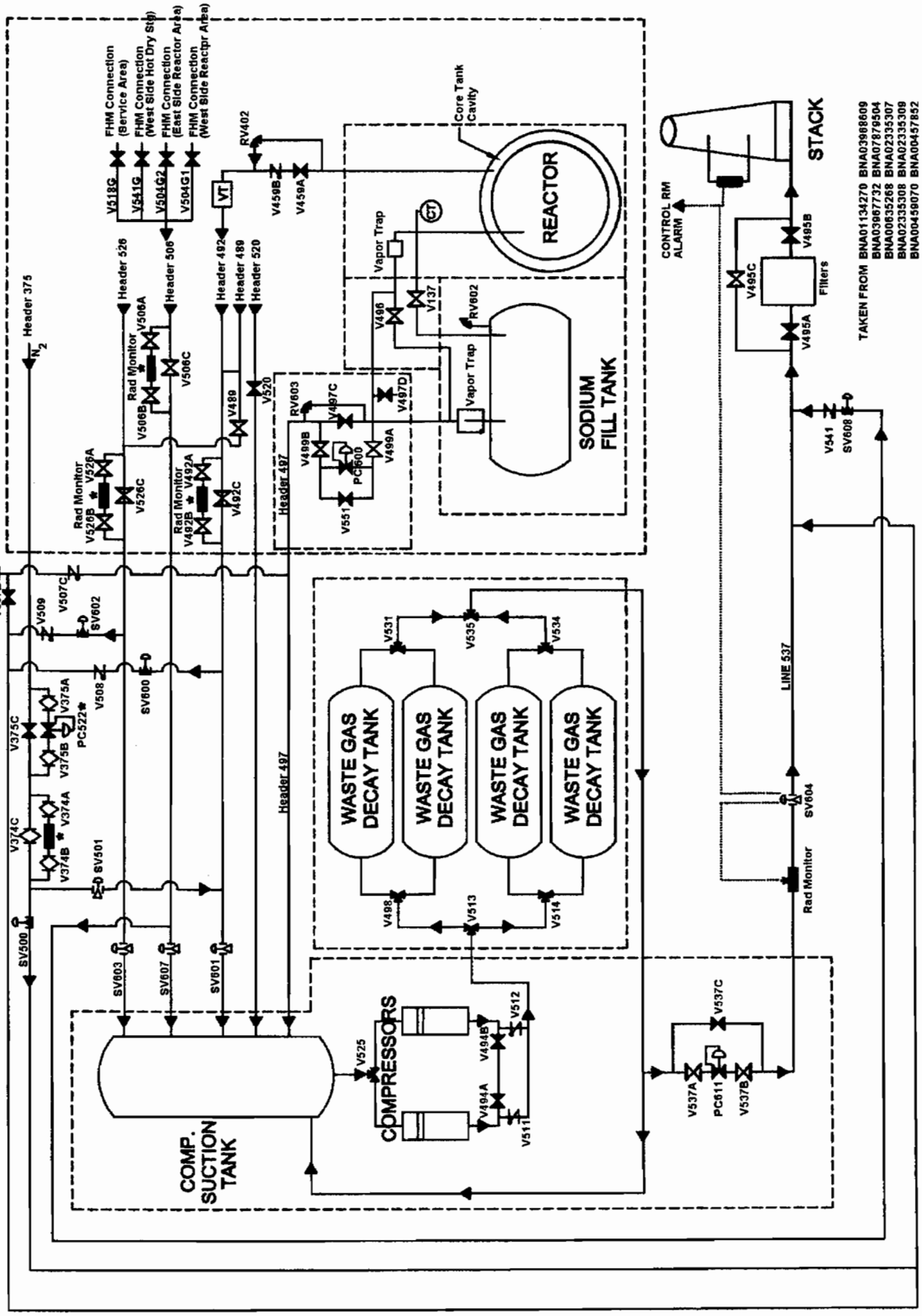


Figure 2.13 Radioactive Gas Vent System

2 - 37

3. Sequence of Operating Events Prior to Run 14

Operation of the SRE has been described by specific periods of operation of the reactor called "Runs", which were planned experiments with defined objectives. The overall objective of the SRE was to demonstrate the feasibility of a sodium-cooled reactor as a heat source for generation of electricity. Another objective was to gain experimental data and operating experience on natural uranium fuel and various uranium alloys. Table 3.1 shows a summary of the SRE power history, including dates of operation and irradiation history of the core.

A reactor is said to be "*critical*" when the uranium fission process in the core produces as many neutrons as it loses by various processes, and the fission process becomes self-sustaining. This milestone for the SRE was reached for the first time on April 27, 1957. At this point in the operational history, the reactor was not producing any power, but experiments and startup testing of the instrumentation and calibration of control rods was begun.[Ref. 2]

3.1. Power Runs 1 through 7

Power Run 1 has been defined as the period July 9, 1957 to July 15, 1957. During this period, various reactor measurements were made, including measurements of the effect of temperature changes on the reactivity of the fuel. Testing of the emergency shutdown procedures were also conducted, called "scram" tests.

Power Run 2 (July 15, 1957 – July 26, 1957) dealt with filling the secondary sodium loop and connecting the electrical plant to the reactor plant. The SRE had a turbine-generator that could be operated by steam produced by heat from the reactor. The steam plant was operated by Southern California Edison, and first produced electricity on July 12, 1957. Following this experiment, the reactor was shut down until November, 1957. Modifications were made to the systems during this time. [Ref. 2]

Table 3.1 Summary of SRE Power Operation

Power Run	Time Interval	Operating Days	Average Thermal Power (MW)	Total Irradiation (MWD)	Total Irradiation Accumulated since Startup (MWD)
1	07/09/57 – 07/15/57				
2	07/15/57 – 07/26/57	5.7	3.93	22.6	22.6
3	11/07/57 – 11/29/57	12.6	6.20	78.2	100.8
4	05/21/58 – 05/28/58	13.3	8.73	116.2	217.0
5	07/18/58 – 08/04/58	11.5	17.7	203.8	420.8
6	08/08/58 – 09/01/58	22.0	17.9	394.0	814.8
7	09/08/58 – 09/25/58	17.2	17.8	306.0	1120.8
8	11/29/58 – 01/29/59	37.0	16.15	597.8	1718.6
9	02/14/58 – 02/26/59	11.5	11.0	126.5	1845.1
10	03/01/59 – 03/07/59	0.6	5.3	3.1	1848.2
11	03/13/59 – 04/06/59	23.6	12.4	293.5	2141.1
12	05/14/59 – 05/24/59	9.7	15.9	154.3	2295.4
13	05/27/59 – 06/03/59	6.6	17.3	114.3	2409.7
14	07/12/59 – 07/26/59	14.2	1.1	16.1	2425.8

Power Runs 3 -7 also dealt with operation of the reactor/steam turbine operation. Electricity was produced during July and November 1957, and May 1958. Much of the time spent during Runs 3 – 7

involved reactor physics measurements. At the end of Run 7, September 25, 1958 , the reactor had accumulated a total irradiation of 1120.8 MWD. [Ref. 2] (A MWD is equivalent to the fissioning, or burnup, of 1 gram of Uranium 235.)

3.2. Power Runs 8 through 12

Run 8 (11/29/58 – 01/29/59) was marked by problems with oxide content in the primary sodium loop. During the time interval between Run 7 and the beginning of Run 8, the primary sodium was pumped back and forth several times between the primary loop and the primary fill tank. The fill tank was known to contain large amounts of sodium oxide. [Ref. 2 pg.III-1] These pumping and mixing operations resulted in the introduction of sodium oxide into the primary system. Sodium oxide is a very fine whitish-greyish solid, which is twice as dense as pure sodium. Eighteen fuel elements were removed from the reactor, examined in the SRE hot cell, and reinstalled in the reactor for continued operation. [Ref. 2 pg.III-1]

Normally, the difference in temperature between core inlet and outlet primary sodium is around 100°F. At the beginning of run 8, some of the exit temperatures were abnormally high as compared to the core inlet temperature, which was around 460 °F. The expected core exit temperature should have been around 560 °F; instead, some were recorded as high as 800 °F and as low as 415 °F, which was below the core inlet temperature. [Ref. 2]

The fact that the core exit temperature thermocouples recorded a temperature at the core exit that was lower than the core inlet is an indication that the exit temperature recorders were not working correctly, possibly due to oxide buildup.

The operators attributed the wide spread in temperature differences to flow blockages in the coolant channels caused by the high oxide content of the sodium. The reactor was shut down, and the sodium coolant was passed through a fine mesh filter called a "cold trap" to reduce the sodium oxide content. On December 12, 1958, the fuel assemblies from channels 9 and 10 (which had been running hot) were removed from the reactor and washed. It was noted that both assemblies had a black residue on their surface before washing. [Ref. 2] Washing the fuel elements proved to be a successful method for improving the temperature problems that they were experiencing.

Reactor operations were resumed at low power (1-2 MW) until December 18. The operators found that raising the reactor power level, and thereby the inlet temperature, was also effective in bringing the temperature differences back into an expected range. This was due to the sodium oxide going back into solution at the elevated temperatures.

The operators also found that moving the fuel assemblies up and down about 1 inch or less in the process tubes also had a beneficial effect on the outlet temperature. Movement of the fuel channel apparently had the effect of dislodging any foreign material from the fuel element and the orifice plate of the fuel assembly.

On December 18, reactor power was increased to 12 MW. Primary sodium flow rate was increased to help maintain more even outlet temperatures across the reactor outlet. On December 19, reactor power was increased to 14 MW, and continued at that level for several days. The reactor was shutdown on December 23, to inspect fuel elements. [Ref. 2]

Reactor operations were resumed on December 27, after washing 15 fuel elements. A power level of 20 MW was attained on December 28. The operators continued the practice of moving the fuel elements to dislodge particles causing flow blockage.

The fuel channel exit temperature problem continued to improve through the remainder of Run 8, until the planned exposure of 600 MWD was achieved. On January 7, a core cover gas sample was obtained and analyzed. It was at this time that the first evidence of tetraline in the primary sodium loop was found. The account of this discovery was reported as:

> "On January 7, a sample of the core cover gas was bubbled through cyclohexane and the solution analyzed. Napthalene was identified, indicating that tetralin had entered the primary sodium at some earlier time. Prior to this analysis, the presence of tetralin in the primary system was not suspected. The only known tetralin leak prior to this occurred in June 1958 when a crack was found in the bearing housing casting of the main primary pump. It is not known if any tetralin had entered the primary sodium at that time." [Ref. 2 pg. III-3]

Power Run 8 can be summarized as being a period when reactor operations did not go as expected. The introduction of sodium oxide into the primary loop with some introduction of tetralin into the primary system caused outlet temperatures at the reactor core to vary, indicating some interference with heat transfer was taking place. The flow blockage was not sufficient to cause automatic reactor scram, or any apparent fuel damage.

Following Run 8, more fuel elements were removed and washed, and the primary sodium was processed through cold traps to get the oxygen level down to less than 5 ppm.

Run 9 (February 14 – February 26, 1959) was marked by core exit temperatures being higher than expected. Run 9 was carried out at power levels of 18-20 MW. Reactor shutdowns were done on February 16 and 18 to wash some fuel assemblies. The expected burnup of 125 MWD was achieved and the run was terminated on February 29.

At the end of Run 9, the fuel assembly in core channel 56 was removed and examined. The orifice plate had a thin black deposit, indicating the presence of an oxide.

Run 10 (March 6 – March 7, 1959) was conducted as part of a test on a uranium oxide fuel assembly. UO_2 fuel later became the standard fuel design for light water reactors in the United States. There were no unusual circumstances noted in this brief run.

Run 11 (March 16 to April 6, 1959) still showed some problems with core exit temperatures during the period March 16 through March 20, while the reactor was at low power levels. Beginning March 20 through March 23, the reactor power was increased to 20 MW, and continued at that level until problems with the primary sodium flow caused several reactor scrams. The problem with the sodium flow was traced to introduction of helium into the primary sodium coolant, causing the primary sodium pumps to cavitate. [Ref. 2 pp III-4] The reactor was brought back to 18 MW on March 28.

A reduction in the temperature differences was noted toward the end of Run 11. During the shutdown following Run 11, 21 fuel assemblies were visually examined with a television camera mounted in the fuel handling cask. All 21 assemblies were found to be in good condition and were returned to the reactor.

Radiation levels in the main sodium galleries had increased during Run 11, but were not so high as to prohibit maintenance work from proceeding. [Ref. 2 pg. III-5] The increase in

radiation levels was due to buildup of fission products in the cold traps, which are very fine mesh filters used to remove particulate material from the sodium coolant. The increase in primary sodium coolant activity was not considered out of the ordinary after operating the reactor for the length of time that Run 11 consumed.

Run 12 (May 14 – May 24) was completed without incident. On May 22, a high temperature run was conducted in which the reactor outlet temperature was increased to 1065 °F for a period of about 1 hour with a power level of 6 MW. Following Run 12, a core cover gas sample was taken and was analyzed to be 1.7×10^{-3} μCi/cc. Xenon activity had been noticed after lengthy runs and was not considered abnormal, but due to pinhole leaks in the fuel cladding, [Ref. 2] or possibly "tramp" uranium in the primary sodium.

3.3. Power Run 13

Although the fuel damage incident occurred during Run 14, certain events took place during Run 13 which were directly linked with the operational problems that occurred during Run 14.

3.3.1. Temperature Effects

Power Run 13 covered the time period May 27 to June 3, 1959. Following the end of Run 13, there was a shutdown of approximately a 39 day period before the start of Run 14, which began on July 12, 1959. Power Run 13 was planned to follow the same power history as Run 12, and achieve a burnup of 150 MWD. The run proceeded without incident until near the end of the run, when on May 29, a scram occurred due to an abnormal sodium flow rate. The reactor was restarted and returned to normal operating conditions until 9:00 AM on May 30. At that point, several deviations from normal operation started to occur. They were identified as:

> "a) The reactor inlet temperature started a slow rise from 545 °F to 580 °F . The rise was very slow, extending over a period of about 3 days.
>
> b) The log mean temperature difference (LMTD) across the intermediate heat exchanger started to increase which indicated impaired heat transfer characteristics. A rather sharp increase in LMTD occurred on June 1.

c) A thermocouple located in a fuel slug in the element in core channel 67 showed an increase from 860 °F to 945 °F. This change started at 0840 and ended at 0900 on May 30. A similar thermocouple in the fuel clement in core channel 36 did not show a corresponding increase.

d) Some of the fuel-channel exit temperatures showed a slight temperature increase of about 10 °F.

e) The moderator delta T chart shows an abrupt jump of about 30°F at 2230 on May 30. The chart shows fluctuations of about 18°F for the 4 hours immediately proceeding. Prior to this, it had been quite stable.

f) The temperature indicated by a thermocouple in a probe located in corner channel 16 showed fluctuations of about 30°F. A few hours later, this temperature settled down to a steady value.

g) Although it was not noted at the time because the reactor was on automatic control, an examination of the record of shim-rod position (made after run 14) showed that a shim-rod motion corresponding to a reactivity increase of about 0.3% had occurred. This change in reactivity was gradual and extended over a period of about 6 hours. Following this the reactivity showed a steady increase of about 0.1% over the next three days of operation. " [Ref. 2 pp. III-7]

The *log mean temperature difference* (LMTD) is a term used to provide an indication of the thermodynamic performance of the heat exchanger. This difference is normally about 75-80 °F [Ref. 2] for the intermediate sodium heat exchanger, and an increase in this difference is an indication that something is interfering with heat transfer. The implication is that the same problem might be occurring in the fuel channels as well. Poor heat transfer in the core may lead to excessive fuel heating and possible damage to the fuel. Figure 3.1 shows the log mean temperature difference as recorded for the period May 30 through June 2.

Figure 3.2 shows the fuel channel 54 exit temperature for the period May 27 through June 2. The design temperature outlet temperature is 1000 °F for the SRE. The outlet

temperature of the fuel channels is measured at the top of the fuel channel. Since the temperature of the coolant is always less than the fuel temperature when the reactor is operating, we can infer that the fuel temperature must be somewhere around 80 to 100 °F higher than the coolant. Although the temperature was elevated beyond normal levels, no evidence was found that fuel damage actually occurred during Run 13. Inspection of the decay tank samples after Run 13 showed activity on the order of 10^{-4} μCi/cc, which was not considered out of the ordinary.

3.3.2. Tetralin Contamination

Previous experience with tetralin leaks in Run 8 helped identify the problem with heat transfer that was identified in Run 13. Tetralin is a clear colorless liquid that is a very good solvent. Tetralin has a boiling point between 405 and 420 °F.[Ref. 27] Tetralin decomposes at temperatures in the range of 800-850 °F. Decomposition products include hydrogen, napthalene, and carbon.[Ref.2 pp IV-B-1] Carbon may go into solution up to its limit of solubility, and then exist in small particles that are suspended in the sodium coolant.

The best estimate of the time that the tetralin leaked into the system was from May 30 until June 3, the time of reactor shutdown for Run 13. It was estimated that

> "the total tetralin leakage during and following run 13 was 1 to 10 gallons corresponding to 8 to 80 lb. Since tetralin is 90.9% carbon and 9.1% hydrogen, this corresponds to a range of about 7 - 70 lb of carbon and 0.7 - 7 lb of hydrogen."[Ref. 3]

The operators recognized the problem of tetralin leakage, partly from their experience gained from the leakage that occurred during Run 8, and also to the odor of tetralin detected in the pump casing of the main primary sodium pump. Run 13 was terminated on June 3, 1959.

On June 12, after a period to allow Na^{24} to decay, the primary pump was removed. A small leak was discovered in the wall of the thermocouple well of the freeze gland seal of the pump. [Ref. 3] Plans were then made to remove the tetralin to restore the heat transfer capability to design conditions.

3.3.3. Wash Cell Incident

Following the shutdown of the reactor after Run 13, 17 fuel elements were removed from the reactor and were visually examined with a television camera mounted in the fuel handling cask. The assemblies were found to be "dirty" but otherwise in good shape. The fuel assembly from core channel R-56 was removed and placed in wash cell "B" for cleaning and examination. During this process on June 4, the wash cell was evacuated and approximately 18 gallons of water were introduced into the cell. The fuel handling cask had been moved to its next position in the storage cell area, and the hold-down clips on the shield plug in wash cell B had not been installed.

A sodium-water reaction occurred, which caused a pressure surge, which severed the hanger tube and lifted the shield plug and hanger tube approximately 18 feet into the air. The shield plug/hanger tube came to rest between wash cells "B" and "C". Operations personnel turned off the supply and exhaust fans in the high bay area while a survey was being made. The fans were turned back on a few minutes later, and the high bay filters were placed in service.[Ref. 17]

The wash cell vent line was closed at the holdup tank storage area, since the ventilation in the wash cell was pulling air through the open shield plug opening at the top of the wash cell. An air sampler was placed in the vicinity of the wash cells in the high bay area, and an activity of 3.0×10^{-9} μCi/cc was measured. Plastic sheets were placed on the floor of the high bay area to prevent recontamination of the floor during cleanup operations.

No further washings of fuel assemblies were conducted. The wash cell incident was essentially over by June 8. It was determined that hydrocarbon deposits (from the breakdown of tetralin) clogged the drain holes on the hanger rod, preventing the sodium to drain out of the tube when it was removed from the reactor. [Refs, 17,18] The hanger rod is located at the top of the fuel assembly, as shown in figure 2.10.

3.3.4. Removal of Tetralin

Plans were made to bubble nitrogen gas through the primary sodium in order to remove the tetralin. The same procedure had been used during Run 8 to clean the secondary sodium fill tank. In preparation for nitrogen gas purging of the primary sodium, the

helium cover gas had to be replaced in order to remove any radioactive gases. Sample data from the holdup tanks indicate that the cover gas was vented to the holdup tanks on June 2, 3, and 4th .

The Interim Report [Ref. 2] identified the pathway for nitrogen injection as follows:

> "Seventeen elements were removed from the reactor and the sodium level in the upper pool was lowered to about 6 inches above the moderator cans. The nitrogen stripping operation began on June 17 and continued until July 5 with 26 fuel elements remaining in the reactor. The sodium temperature was 350 °F at the start and was raised to 425 °F by the end to enhance the removal of tetralin. Nitrogen was admitted to the system through the primary pump casing, passing through the heat exchanger, and then into the bottom of the reactor; 400,000 ft^3 was used."

There is no mention in any of the available reports as to how the nitrogen was collected, other than it was discharged from the reactor through the vent system. Operating procedures were reviewed, and an assessment was made as to the most probable method of venting the nitrogen, given the fact that some of the fuel assemblies were removed in order to facilitate bubbling up through the core. A relevant fact is that the holdup tanks show ventings to the environment during at least a portion of the period when nitrogen purging of the reactor was taking place. Samples were taken of the holdup tanks from June 17 through June 24, indicating that the pathway was from the reactor to the holdup tanks, and most likely through the normal vent header 497. Following June 24, there is no sample from the holdup tanks until July 1. From July 1 until the end of the nitrogen purge on July 5, an additional 3 ventings took place, indicating that the valve alignment at that time was to the holdup tanks. The sample data for the holdup tanks are provided in Appendix C.

The added explanation in the Interim Report was

> "The stripping process was terminated when no more impurities were being removed. The system was then purged for 10 hours with 4700 ft^3 of helium and argon."

There were two holdup tank volumes collected on July 1, which would account for the purging of the nitrogen with helium and argon. The above quote also gives an idea of the purging rate for that particular alignment, which was indicated as 470 ft^3 per hour, or about 7 cfm. If we assume that the stripping went directly to the environment for the period June 24 until July 1, then 380,000 ft^3 must have been vented at a flow rate of approximately 50 cfm.

Another possible scenario would be to vent to the holdup tanks only periodically, as it would be prudent to assume that sampling of the release would be a requirement as per procedure. In this case, the venting would be aligned such that samples could be taken more frequently at first, since the concentration of radioactive gases in the exhaust stream would be higher, and less frequently later on during the venting, after it was established that releases were not exceeding limits for release to the environment. In this case, the 400,000 ft^3 would be vented intermittently to the holdup tanks over a 13 day period. This equates to a flow rate of 20-25 cfm.

As mentioned above, 17 of the 43 fuel assemblies were removed for the nitrogen stripping operation. This was done to enhance nitrogen flow through the core. The only direct access to the cover gas over the core is via the fuel handling cask with a shield plug removed.

While it cannot be determined with certainty the exact route that the vent gas followed during the stripping operation, it is most likely that the fuel handling cask was situated over a shield plug opening and the vacuum pump of the fuel handling cask was used to transfer the nitrogen gas to the vent system. The fuel handling cask vents to header 506, as shown in Figure 2.12. This pathway requires use of header 506 bypass when it is directed to the environment via the stack. This can be accomplished by closing solenoid valve SV-607 and opening SV-608. This provides a short and direct pathway to the stack, and allows periodic sampling of the nitrogen gas by diverting the flow back to the holdup tanks. No other system valving needed to be changed to accomplish the venting.

After termination of the venting and insertion of the shield plug into the face shield, as per the procedure for operating the fuel handling cask, the vent header valve (504G-1, west side of face shield, or 504G-2, east side of face shield) was closed in the high bay

area, and the flexible hoses to the fuel handling cask were removed. It is possible that the solenoid valves SV-607 and SV-608 were left in the bypass position, since the arrangement described above was not a normal operational alignment. Normally, vent header 506 is aligned to the decay tanks via the suction tank.

3.3.5. Nitrogen Contamination

On July 30, 1959, the minutes of the Ad Hoc Review Committee contained the following statements:

> "...The nitrogen was introduced by putting it through the [primary] pump and forcing it into the bottom of the reactor. The nitrogen content was not measured. Calculations were made, subsequently, so that enough helium would be bubbled through the core to purge and reduce the nitrogen concentration to 4%. This is considered acceptable for continued operation." [Ref.19]

Both the Interim Report and the Final Report [Refs.2, 3] discuss the metallurgical effects of the introduction of nitrogen into the primary sodium system, and do not mention a very important aspect of the nitrogen stripping operation – the production of N^{16} through activation by neutron absorption. The half-life of N^{16} is very short, on the order of 7 seconds, and is not important in normal circumstances when there is no access above the reactor vessel during power operation. However, even with the reactor at low power, any leakage of gas from the 81 shield plugs at the loading face would cause the background radiation levels to increasesignificantly. N^{16} decays to O^{16}, which would react with the sodium coolant to produce sodium oxide - the problem experienced with heat transfer and flow blockage during Run 14. The neutron flux in the cover gas was estimated to be high enough to cause activation of N^{15} to N^{16}. [Refs. 41, 44]

Apparently, at the time of the SRE, the N^{16} activation problem had not yet been discovered, or was not well known. A few years later, when experiments were being conducted on light water reactors, the discovery was made of N^{16} generation. In light water reactors, as reactor coolant travels through the core, there is some radiolytic decomposition of water, producing O^{16} and free hydrogen. N^{16} is produced by the

$O^{16}(n,p)N^{16}$ reaction in water with resultant beta decay, emitting very strong gammas 78 percent of the time. The gamma energies are 6.13 and 7.11 Mev.[Ref. 20].

3.3.6. Purging Cover Gas

Following the nitrogen purge, the cover gas had to be recharged with helium. Helium and argon were used to purge the system for 10 hours to replace the nitrogen in the system. [Ref. 2, pp III-8] Like nitrogen, argon also undergoes activation by neutrons, and becomes Ar^{41}. Argon-41 has a half-life of 1.83 hours with a gamma photon energy of 1.29 Mev. [Ref. 20] The relative amount of argon that was added was not mentioned in the AI investigation of the incident, but this may also have contributed to the radiation effects that occurred during Run 14 with the activity problems in the high bay.

The holdup tanks are sampled just prior to venting to the environment. The records in Appendix C indicate that the holdup tanks were sampled and discharged to the environment on July 9th, 10th and 11th . Earlier ventings of the holdup tanks took place on July 1st and July 3rd, indicating that the valve lineup of header 497 was to the suction tank.

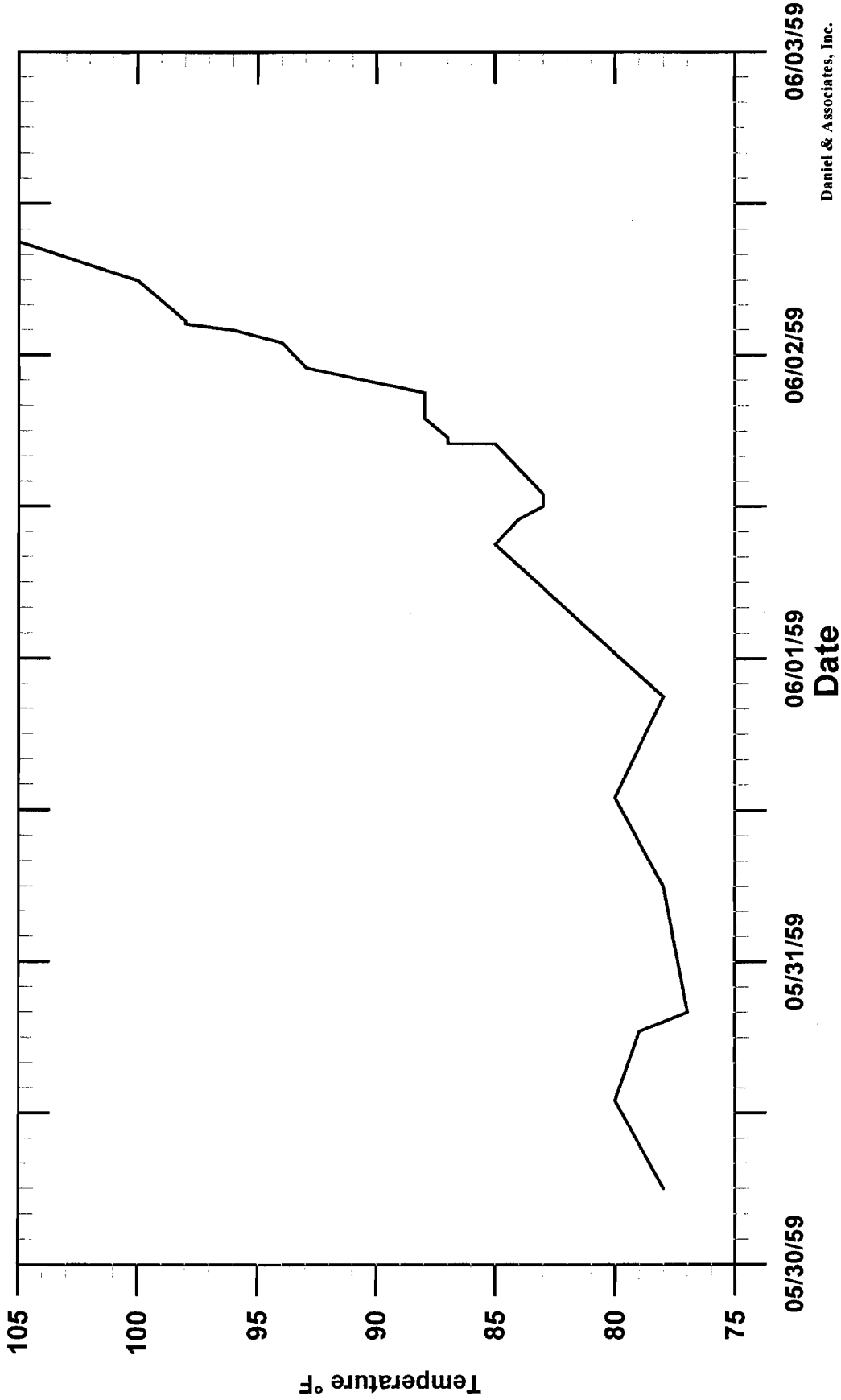


Figure 3.1 Log Mean Temperature Difference- Intermediate Heat Exchanger

3 - 14

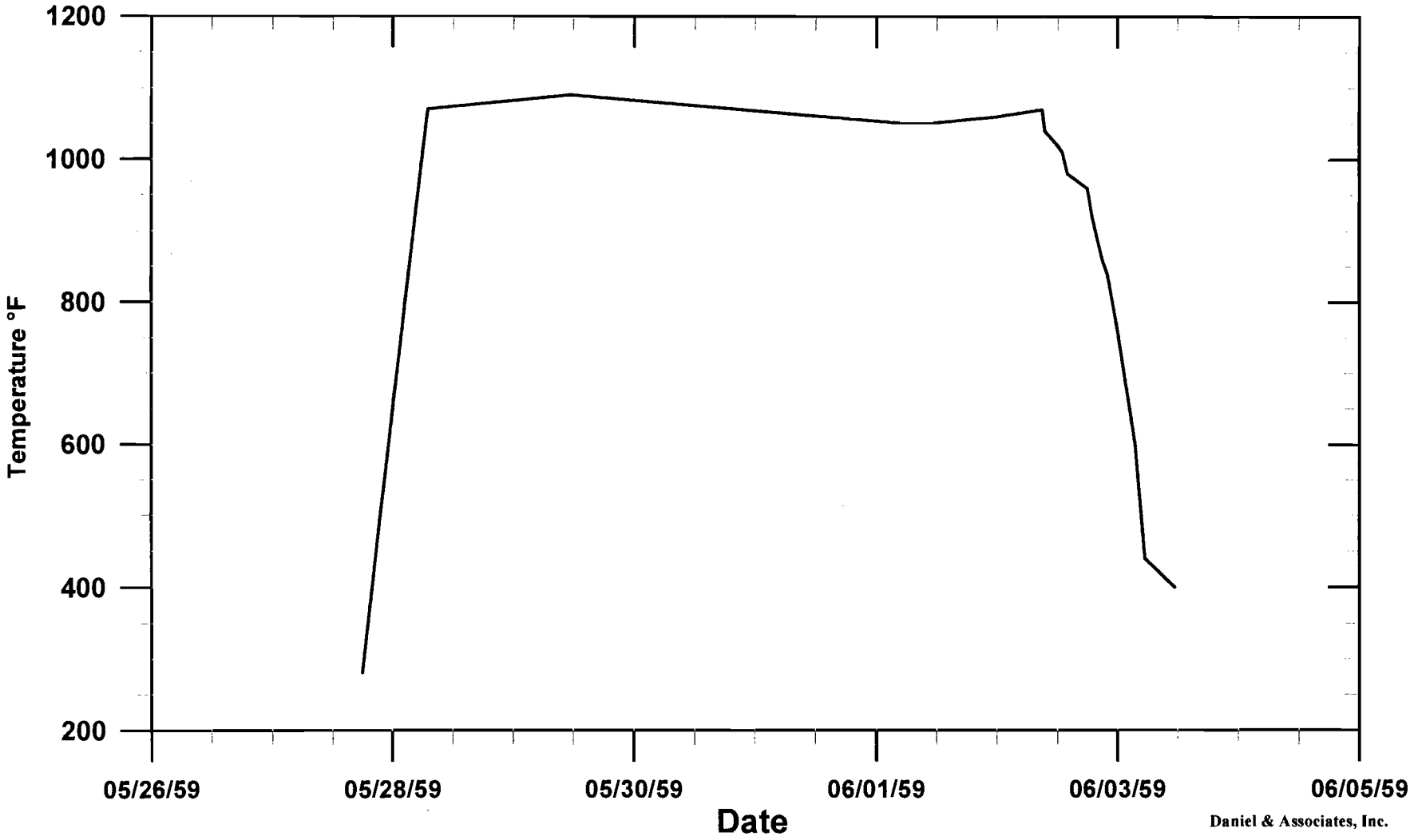


Figure 3.2 Fuel Channel 54 Exit Temperature

3 - 15

4. SRE Fuel Damage Incident (Run 14)

4.1. State of Knowledge in 1959

The design and construction of the SRE was to gain operating experience using uranium fuel in a reactor used to produce electricity. The fuel elements in the SRE design were operating under untried conditions. Fuel design limits were based on theoretical limits, not operational experience. Cladding materials were likewise untested, with little or no operating experience. The Zirconium cladding on the moderator cans was later replaced by a zirconium alloy, called "zircaloy", which is still in use in today's nuclear designs.

4.2. Initial Events (July 12-15)

The reactor was brought to criticality at 6:50 AM on July 12. The startup procedure checklist includes checking that the suction tank compressors are on, and that the compressors are aligned to a holdup tank, and that the holdup tank is holding pressure. This is to ensure that radioactive gases will not be vented to the environment. The startup checklist is provided in Appendix D.

The power increase was described as "cautious", and reached a power level of 500 kilowatts at 8:35 AM. The operators made note of moderator temperature fluctuations of about 10 °F, which were considered out-of-ordinary, but they expected problems with temperature variations because of the tetralin contamination and their experience with Runs 8 and 12. Some of the fuel channel exit temperatures, which should be close to one another, started to diverge. This was an indication of either flow blockage in the fuel channels or uneven heat transfer in some of the fuel channels. Power levels were kept below 1 Mw during the morning until a reactor scram occurred at 11:42 AM. This automatic shutdown was due to loss of auxiliary primary sodium flow. [Ref. 2]

The reactor was restarted and brought to criticality at 12:15 PM. Over the next few hours, the reactor was increased in power level, such that at 5:00 PM, the reactor was at a power level of 2.7 MW.

4.2.1. High Bay Activity

At 3:30 PM, the two air monitors in the high bay area showed an increase in radioactivity levels. The high bay area radiation levels are reproduced as Figure 4.1 for this time

period on July 12. The data in Figure 4.1 was benchmarked to the statement in the Interim Report that

> "At 1620 it was noted that the filter from the air sampler showed an activity level of 160,000 CPM." [Ref. 2, (pg. III-10)]

With this data point established, it is possible to determine the times that other events took place. For example, the air filter on the continuous air monitor was changed at 4:25 PM, and immediately went off scale when it was returned to service at 4:45 PM. It was then removed from service until reactor startup was achieved at 4:40 AM on July 13.

When the continuous air monitor first started to increase at 3:15-3:30 PM on July 12, the operators decreased the reactor cover gas pressure. A radiation survey was conducted and it was determined that the most leakage was from the shield plug at core channel position 7. The sodium level probe was installed in this position. This instrument measures the sodium level in the reactor and displays the level to the operators in the control room.

To reduce the leakage into the high bay area the decision was made to reduce pressure in the reactor. This would take only a minute or two once the decision was made to do so. It is logical to assume that the operators recognized the benefits of decreasing the leak rate before conducting the radiation survey, so it is estimated that reduction of the pressure by venting the cover gas to the holdup tanks took place between 3:30 and 4:00 PM.

Conducting the radiation survey would require a health physics technician to first dress in anti-contamination clothing, put on and check a suitable respirator before entering the high bay area. One of the first tasks for the technician would be to locate the source of increasing radioactivity in the high bay. Once this was done, immediate plans could be made to either stop or control the leak. The fact that the air sampler was changed at 4:25 PM supports the assumption that either additional personnel were already in the high bay area or that they were ready to enter the high bay after the survey identified the leak. The initial reading over the shield plug was 500 mr/hr. [Ref.2 pp III-10]

Faced with the immediate task of stopping the leak at core channel 7, the operators decided to remove the instrument thimble from core channel 7 and replace it with a standard shield plug. Removal of the instrument required use of the fuel handling cask. Core Channel 7 is relatively free of obstructions since it is located near the edge of the top shield, as a shown in Figure 2.9. The fuel handling cask (FHC) can be positioned over this channel without removal of any other equipment on the reactor loading face. When the cask is sealed to the loading face, the internal volume of the gas lock is vented to header 506. A vacuum is drawn in the gas lock and maintained to verify that the seal is complete. If the seal is incomplete, the vacuum is released, the FHC repositioned and the vacuum drawn again. When it is determined that a good seal exists, i.e., no gas can leak out of the chamber, the faulty shield plug can be replaced. After replacing the shield, the cask is vented again before the seal is broken and the cask removed. The radiation level near the FHC would be continuously monitored. At 5:00 PM, the radiation survey was recorded as 25 r/hr at channel 7, which is consistent with removal of the shield plug, since the opening to the cover gas would be unshielded at that point. Once the solid shield plug was installed, the reading should decrease to "normal" background if the leak were terminated.

In addition, at 5:00 PM, the stack monitor showed a "sharp increase" [Ref. 2, pp III-10] to 1.5 x 10^{-4} μCi/cc. This coincides with the operations that were being conducted in the high bay area, and specifically with the probable time of the ventings of the FHC to purge the gas lock. It is likely that the solenoid valve for header 506 was closed to the suction tank, and aligned to the stack at the time. The stack monitor increase occurred approximately one and a half hours after the first venting of the reactor cover gas between 3:30 and 4:00 PM.

It was noted at 5:30 PM that reactor shutdown was in progress, which takes less than 10 minutes. Once the reactor was shutdown, the short-lived N^{16} activity would cease, and relatively normal conditions would exist. Since the reactor did not have sodium level indication with the probe removed from core channel 7, the decision had been made to install a manual level probe in core channel 50, which is near the center of the reactor. Access to core channel 50 requires removal of the control rod drive motors to allow sufficient room for the fuel handling cask to be positioned at this location. Moving the

fuel handling cask from core channel 7 required purging the gas lock volume several more times before the cask could be repositioned over core channel 50. Once in place over core channel 50, additional purgings of the cask would take place to install the new sodium level probe. This process would take several hours.

The notation was made that

> "At 2057 [8:57PM], the reactor was shut down, the drive units removed, and the cask placed in operation." [Ref. 2, pp III-10]

By 10:00 PM, the stack monitor had returned to normal. This is consistent with no more ventings of the FHC, after completion of installing the manual level probe in core channel 50. Startup of the reactor was begun at 4:40 AM on July 13, and no additional problems were encountered at that time with leaks in the shield plug. The FHC was not used again until July 15, when the operators removed the manual level probe from core channel 50 and replaced the repaired standard level probe back into channel 7. This coincides with the second occurrence of stack activity increasing intermittently.

At 9:00 AM on July 14, the high bay activity rose again. This time the problem was traced to core channels 29 and 50. These two channels were repaired by placing seal rings at the top shield, and by taping over locations where leaks were detected. Radioactivity returned to normal by 2:00 PM on the 14^{th}.

The high bay activity continuous air monitors increase was measured at $3x10^{-7}$ μC/cc after 15 minutes decay and $4.5x10^{-8}$ μCi/cc after 90 minutes decay.[Ref. 34 pp 5]. The decay of the measurement indicates extremely short-lived radioisotopes, and is indicative of off-gassing from the filter paper. The fact that activity levels returned to normal indicates the presence of noble gas activity. The presence of "chemically reactive" isotopes, such as cesium or iodine, would cause activity levels to continuously increase, and the high bay area to become contaminated as plateout of radioactive isotopes occurred. No such contamination occurred.

4.2.2. Reactivity Excursion

The SRE had been demonstrated to be a very stable reactor in all operational tests. However, on July 13, the reactor began behaving erratically and fluctuating in power

without the operators intentionally increasing power. To understand this event, it is necessary to understand the physical significance of certain terms as they apply to reactor physics.

When a reactor is in steady-state operation, at any power level, the neutron population remains the same. There are just enough neutrons being absorbed by the U^{235} atoms to cause exactly one more fission by each neutron, thus making the reaction self-sustaining. The number of neutrons, i.e., the neutron population, increases as the power level increases. In order to increase power, the reactor needs more fission events to take place, and the control rods, which absorb neutrons, are withdrawn from the core. This is also called increasing "*reactivity*", or insertion of "*positive reactivity*". Likewise, a decrease of power requires that the neutron population decrease and the rods go into the core to reduce the number of neutrons. This is called "*negative reactivity*". The neutron population is not changing when the reactor is at steady state, although neutrons are being absorbed and new fissions are taking place to replenish the neutrons that have been absorbed or lost from the reactor.

When a reactor is at steady state, and positive reactivity occurs, the neutron population will increase. The rate at which the population increases is measured by what is called the "*reactor period*", which is defined as the time required for the neutron population to increase by a factor of 2.71828. This value is a number that defines the relationship of exponential behavior of events that occur in nature, and has the universal symbol "*e*". Reactivity and reactor period are inversely proportional, that is, an increase in reactivity causes the reactor period to decrease. A reactor in which the power is not changing has zero reactivity, and the reactor period is said to be "*infinite*".

On July 13, at 5:28 PM, the reactor was at 1.2 MW power, and the operators began an increase in reactor power to interface to the turbine to produce electricity. The power level began to increase faster than expected. The control rods were inserted slightly to hold back the increase. At 6:07 PM, a negative period (decrease in power) of about 45 seconds was recorded, and the reactor lost power to about 2.4 MW in 3 minutes. Control rods were withdrawn to bring the power back to 4.2 MW, and the power rose to around 3 MW by 6:21 PM. At this time, the reactor power began to rise more rapidly, even though

the control rods were being inserted. An event inside the reactor was causing a series of positive reactivity insertions. Around 6:24-6:25 PM, a positive transient with a period of 7.5 seconds caused the operators to manually scram the reactor. [Ref. 2 pp III-11]

An investigation of the power excursion was conducted after Run 14 was terminated. The conclusion reached as to the reason for the excursion was that partial plugging of the flow channels in several fuel assemblies caused boiling of the sodium in those channels. The sodium boiling was caused by local overheating in the fuel channel due to partial flow blockage.

One of the significant findings of the investigation was that:

> "Calculations have been made which show that severe plugging in a fuel channel can lead to quite high local temperatures in the fuel due to the thermal insulation provided by the plug. It was found that at a reactor power of 2 Mw the temperature of the hot spot on the surface of the fuel rod that is insulated over 25% of its surface can be about 180 °F above the local sodium temperature." [Ref. 29]

While it cannot be ruled out that some damage occurred to the fuel cladding or to the fuel itself during the power excursion of July 13, the exit temperatures of the standard fuel assemblies that were being recorded during this time period do not support temperatures to cause significant fission product release from fuel.[Refs. 2, 38] However, the temperature records of the experimental fuel assemblies were not being recorded during this time period.

4.3. Temperature Records During Run 14

The fuel damage that occurred during Run 14 was temperature driven, complicated by coolant flow blockage. The flow blockage was caused by degradation products of tetralin in the system. The final report of the fuel damage incident [Ref. 3] identified thermal cycling of the fuel through the uranium $\alpha - \beta$ transformation temperature and subsequent melting and formation of Fe-U alloys as the source of fission product release.

Alloys are made up varying amounts of two or more elements. Their properties relating to the different physical states are described graphically in what is called a "*phase*

diagram". The phase diagram can be used as a "map" from which the phases at any particular temperature of composition can be read for an alloy under equilibrium conditions. Figure 4.2 is the phase diagram for Uranium-Iron alloy. The "eutectic" is a temperature that represents the lowest temperature at which any liquid in the alloy can exist. For the damaged fuel of the SRE, the alloy was approximately 11 wt.% Iron and 89% Uranium. At this mixture, the eutectic temperature is 1337 °F as shown in Figure 4.2. [Ref. 30]

Above the eutectic temperature, a mixture of U_6Fe would exist until the temperature was high enough for a complete liquid phase. Below 1337 °F, a solid mixture of Iron and Uranium exists. When the eutectic temperature is reached, the uranium diffuses into the iron since the uranium has the lower phase change temperature at this composition.

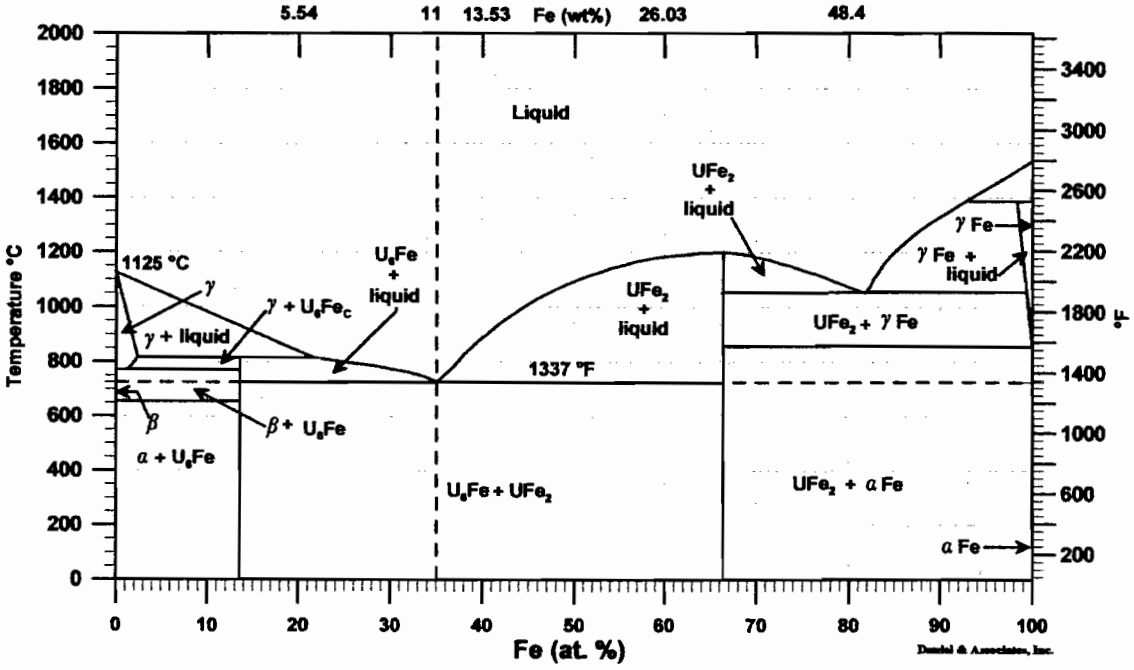


Figure 4.2 Iron-Uranium Phase Diagram

All of the core channels containing fuel assemblies were monitored for sodium exit temperature. In addition to monitoring coolant exit temperature, six of the SRE fuel assemblies were equipped with two thermocouples that provided temperature measurements of the fuel at different elevations. Figure 4.3 shows the fuel thermocouple locations. Of the six standard fuel assemblies with thermocouples, only two were still

operating by Run 14 and were located in core channels 43 and 67. Three experimental assemblies had thermocouples, located in core channels 33, 34, and 55. The standard fuel assemblies record their temperature response in the control room. The recorder for the experimental assemblies was located in the high bay area of the reactor building. [Ref. 2]

The normal temperature for SRE fuel is between 1050 and 1075 °F at 20 MW power. The design temperature for the fuel is 1200 °F, based on some conservatism built into the calculations.

The temperature response of the fuel thermocouples in core channel 55 is shown in Figures 4.4 and 4.5 for TC09 and TC10 respectively. The two thermocouples in the fuel assembly in core channel 55 are both above the midplane of the core. TC09 is located 9 inches above the midplane, and TC10 is 21 inches above the midplane of the core. Figure 4.4 shows that at least a portion of the assembly was above the temperature required to form a eutectic between the fuel and cladding. The recorder ran out of paper on July 20, which is the reason for the missing data during that period. The maximum temperature recorded at the end of the chart when the recorder ran out of paper was 1350 °F. The maximum temperature recorded by TC09 was 1465 °F on July 23. Most of the damage to fuel assemblies occurred approximately 12 inches below the midplane.[Ref.2]

Figure 4.6 shows the temperature oscillations recorded for core channel 55 between 8:00 AM on July 22 and 8:00 AM on July 23rd. The temperature oscillations are severe, ranging 350 degrees from high to low. The thermal cycling eventually led to cladding failure. The coolant temperature was also recorded as it passed out of the coolant channel, and is shown in Figure 4.7. The exit temperature was significantly below the fuel temperature during July 19-24.

The recorder for the experimental assemblies thermocouples were being repaired during July 12-15, so there is not a similar trace available for the period of the reactivity excursion. [Ref. 2]

The reactor continued operation at low power until July 20th. On July 19th and 20th, a test was conducted to try to determine the effects of pressure on reactivity, as part of an investigation into the causes of the power excursion.

Beginning on July 20th, the temperature of the coolant was increased in an attempt to put the solidified particles back into solution. This was apparently due to the operating experience gained during Run 8, and to the fact that the wash cell was unavailable. The problem that was being addressed by the operators was that of poor heat transfer. Although it was unknown at the time, the increase in power only increased the production of N^{16}, thereby providing more oxygen to react with the sodium to form sodium oxide.

The neutron flux of the reactor provides an indication of power level. The relative power level as a percent of full power (indicated by neutron flux) for Run 14 is shown in Figure 4.8. The reactor scrams are shown by the flux dropping to zero. The duration of the shutdown is indicated by the time the power level/neutron flux remains at zero. As demonstrated by Figure 4.8, the reactor experienced several scrams during the period July 16 – 26th, after the reactivity excursion. Reactor power reached a maximum of approximately 5 MW and was sustained there July 22-23rd. [Ref. 2]

On July 24, an attempt was made to dislodge some of the material believed to be interfering with heat transfer by "jiggling" some of the fuel assemblies. At this point, it was noticed that the elements in core channels 10, 12, 35 and 76 were stuck in place. During a similar operation on the evening of July 22, the assembly in core channel 10 was free. [Ref. 2 pp. III-17]

The coolant outlet temperature was gradually reduced to 510 °F around midnight on July 25th, and the cold trap was put into service. The cold trap is used to remove very fine particulate material from the coolant.

The oxygen content of the sodium can be indicated by a device called the "plugging" indicator. This device utilizes the temperature effect of oxide solubility for its operation. A small flow of sodium is diverted to the plugging indicator just upstream of a flow restriction, and the flow is cooled in a small heat exchanger. When the sodium cools, the oxide will precipitate out of solution and plug the flow restriction. The plugging causes a sharp change in the flow rate meter on the plugging indicator. Using a solubility curve for oxygen in sodium, the temperature at which the plugging precipitates gives an indication of the oxide content in the sodium. The temperature at which the sodium

precipitates is referred to as the plugging temperature. The higher the oxide content, the higher the plugging temperature. [Ref. 2, 6]

On July 25th, when the cold trap was put into service, the plugging temperature was 455 °F, which is a relatively high temperature, indicating a high oxide content. Operation of the cold trap reduced the plugging temperature to 350 °F on July 26th at 7:00 AM. [Ref. 2]

The reactor was shut down at 11:20 AM on July 26th to examine the fuel elements that had been running hot. A television camera in the fuel handling cask was used to examine the fuel elements by pulling them up into the cask past the camera. On July 26 at 7:15 PM, the first damaged fuel element, core channel 25, was observed. [Ref. 2]

One of the fuel assemblies that recorded coolant exit temperature was the assembly in core channel 54. A standard fuel assembly fuel temperature thermocouple was functional in core channel 67. These data are shown in Figure 4.10, and serve to show the magnitude of temperatures during the first part of Run 14 as compared to the last part of the run when reactor power had been increased. Although the temperature was elevated for a portion of time during July 15, the longest duration of elevated temperature, and the time of temperature cycling recorded by core channel 55, was during July 22-24 while the reactor was at approximately 5 MW. More of the core would have been near the higher temperatures during July 22-24. This study concludes that most of the fuel damage occurred during the July 22-24 time frame.

4.4. Fuel Assembly Examination

Beginning with the examination of fuel assemblies with the television camera on July 26th, more detailed examination followed over the next few months after shutdown. Figure 4.9 shows the location in the core of the fuel assemblies found to be damaged. Appendix E provides a more detailed description of the damage found, as reported in the Interim Report.[Ref. 2, TableIII-5]

About one-half of the assemblies were broken in two below the midplane of the assembly. Two were broken about one-third up from the bottom of the assembly, and two were stuck in the core, (24 & 76), swollen to the extent that they moved the adjacent moderator cans when pulled up from their position in the core.

The assembly from core channel 24 was removed and examined in the hot cell. There were four distinct features noted in the Supplemental Report [Ref. 3]

1. There was a solid plug of material about 12 inches up from the bottom of the assembly, consisting of an Iron-Uranium alloy of near eutectic composition, which had moved down the assembly from the region of melt-through. The mass of this material was approximately 800 grams.
2. There was a region of Iron-Uranium melt-through about one-third of the way up the assembly, and extending for several inches.
3. There was a region above the melt-through where the cladding had ruptured.
4. There was a region where a black spongy material had formed in the fuel channel below the element, which was porous to sodium.

The spongy material was determined to consist of insoluble carbon and was soluble in alcohol, indicating high sodium content.

Additional tests were conducted on fuel assemblies that were not damaged, and which had been in the core during the high temperature runs of Run 12 and 13. The test found these assemblies to be normal. Unirradiated specimens of uranium, similar to the SRE fuel, were thermally cycled through the eutectic temperature between 900 and 1300 °F. It was found that rapid cycling across the $\alpha - \beta$ phase transformation temperature caused the cladding to fail within 24 hours.

The ultimate conclusion of fuel damage was that there were two factors involved:

1.) thermal cycling across the eutectic temperature which caused the fuel to swell until the cladding ruptured, and

2.) diffusion of uranium into the stainless steel cladding to form low-melting alloys, which may have also caused cladding failure.

Both types of failure were caused by the plugging of coolant passages by decomposition products of tetralin.

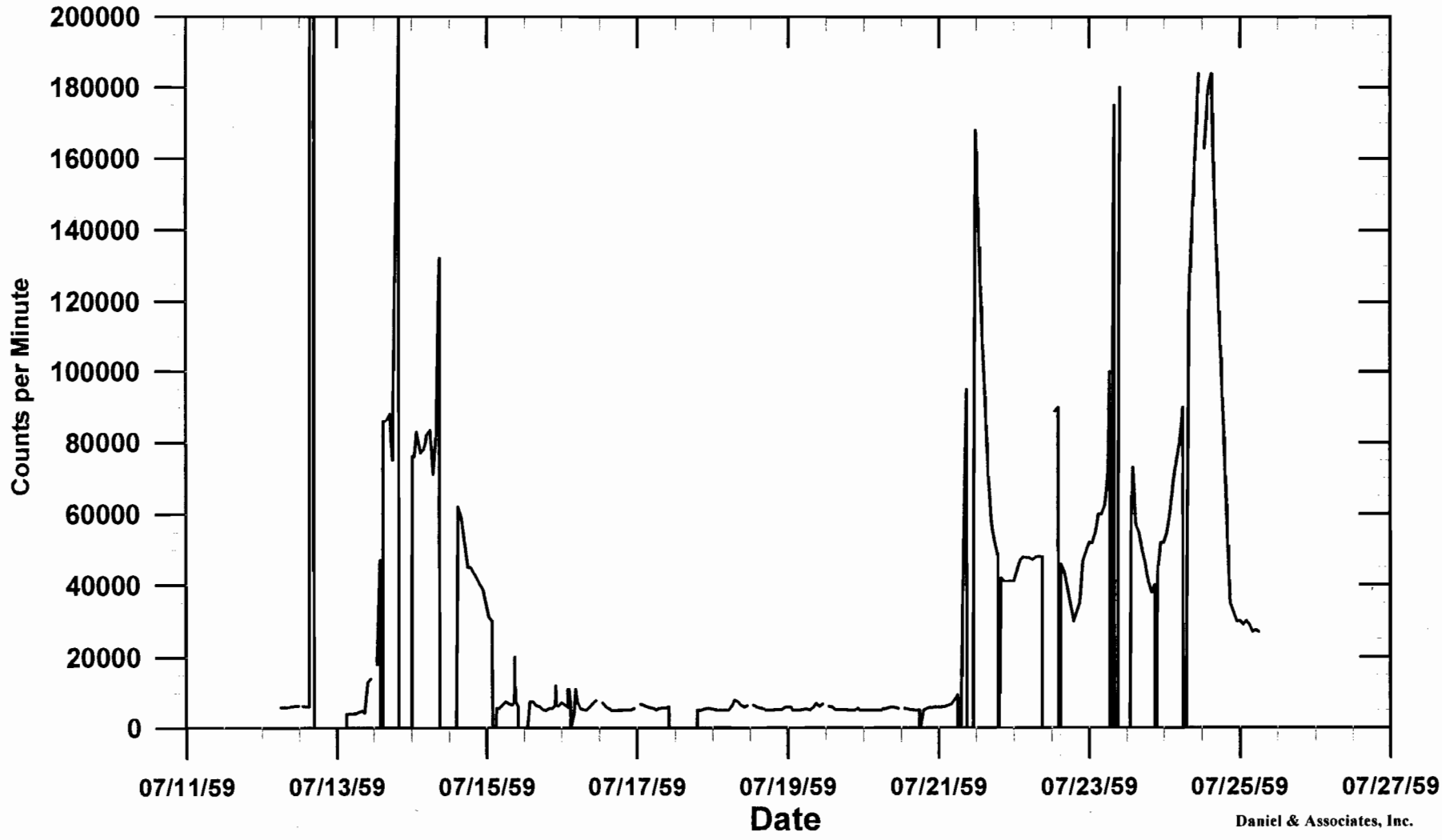
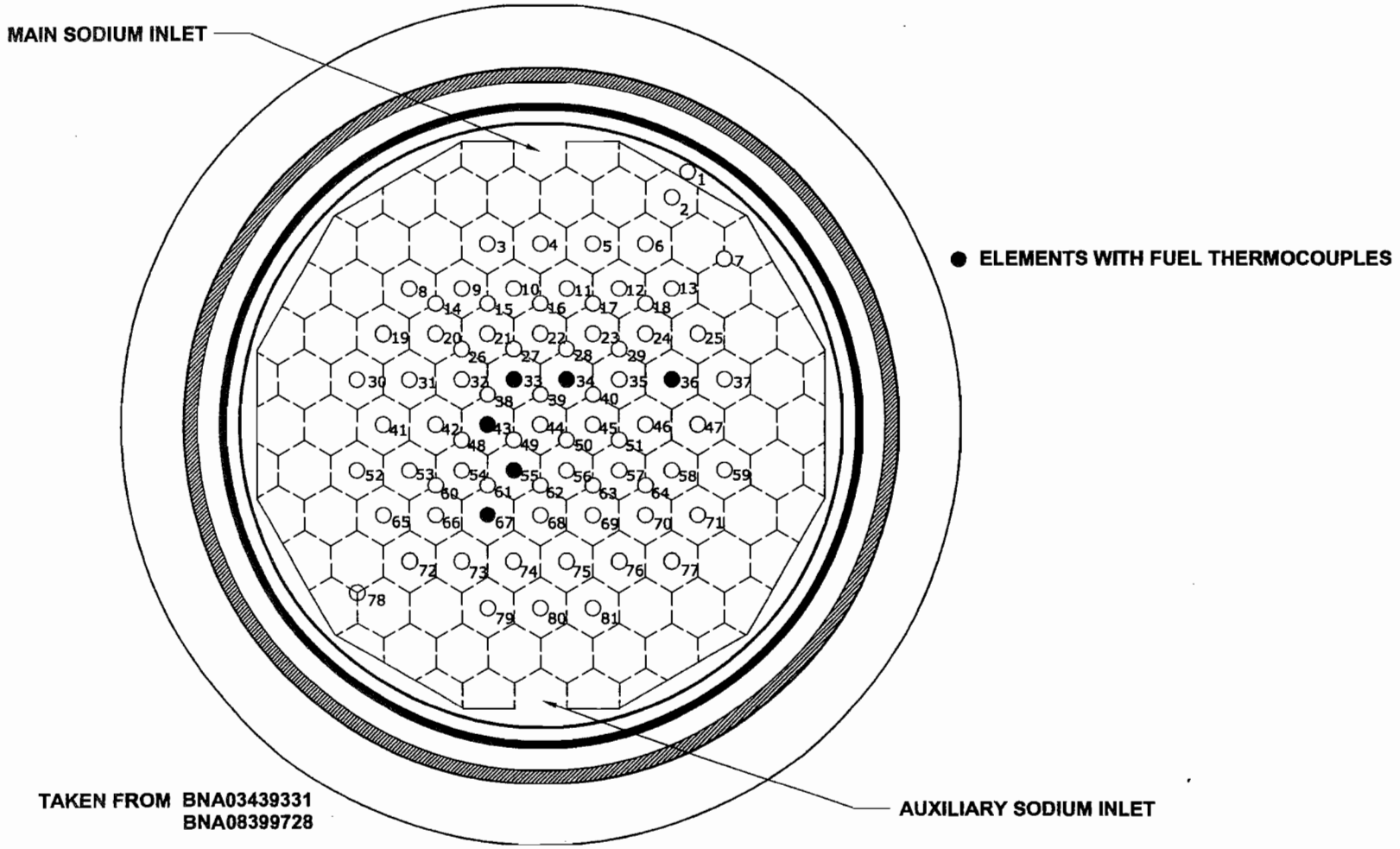


Figure 4.1 High Bay Activity Levels

4 - 12



Fugure 4.3 Location of Fuel Thermocouples

4 - 13

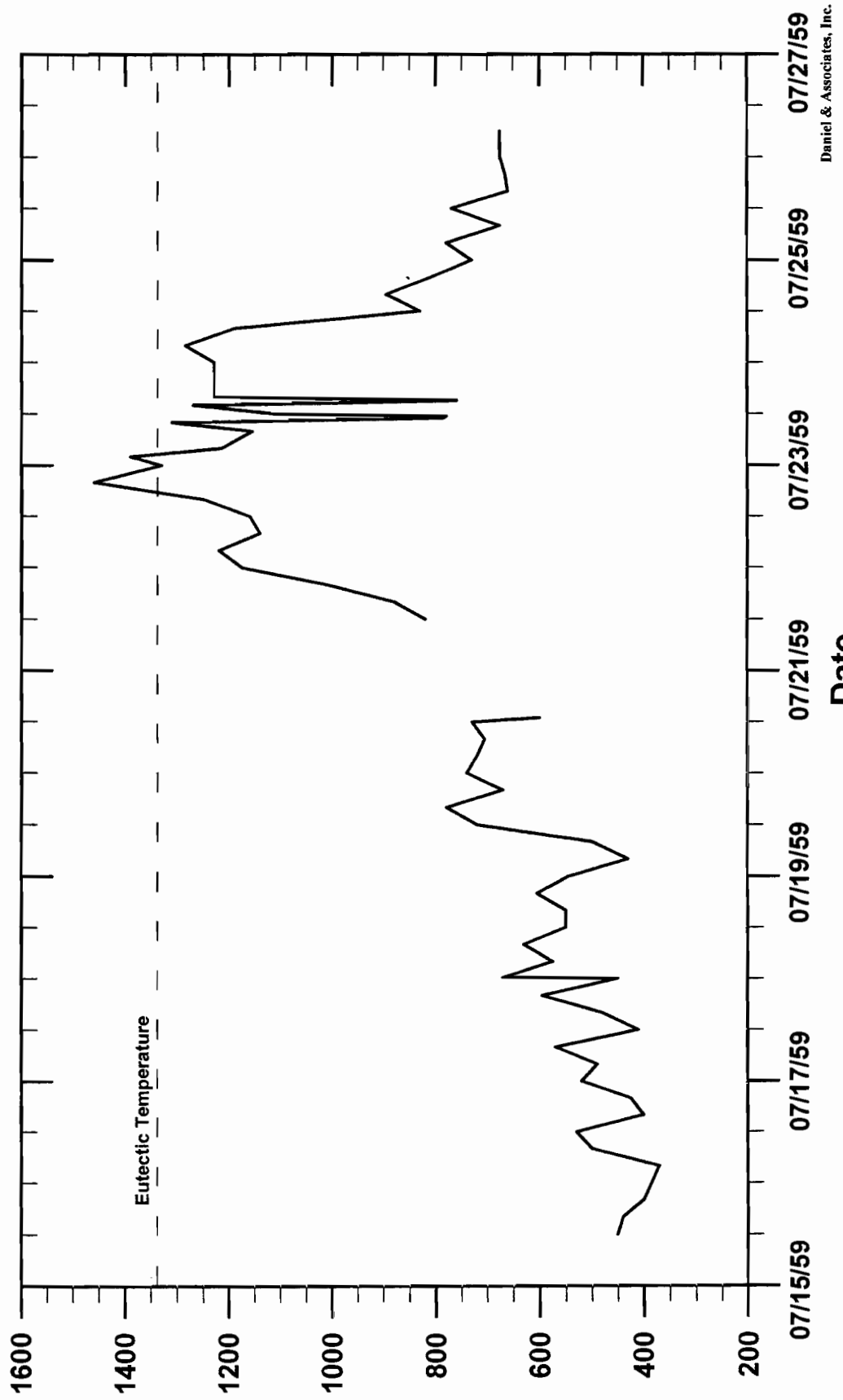


Figure 4.4 Thermocouple Temperature TC09 (9 inches above midplane)

4 - 14

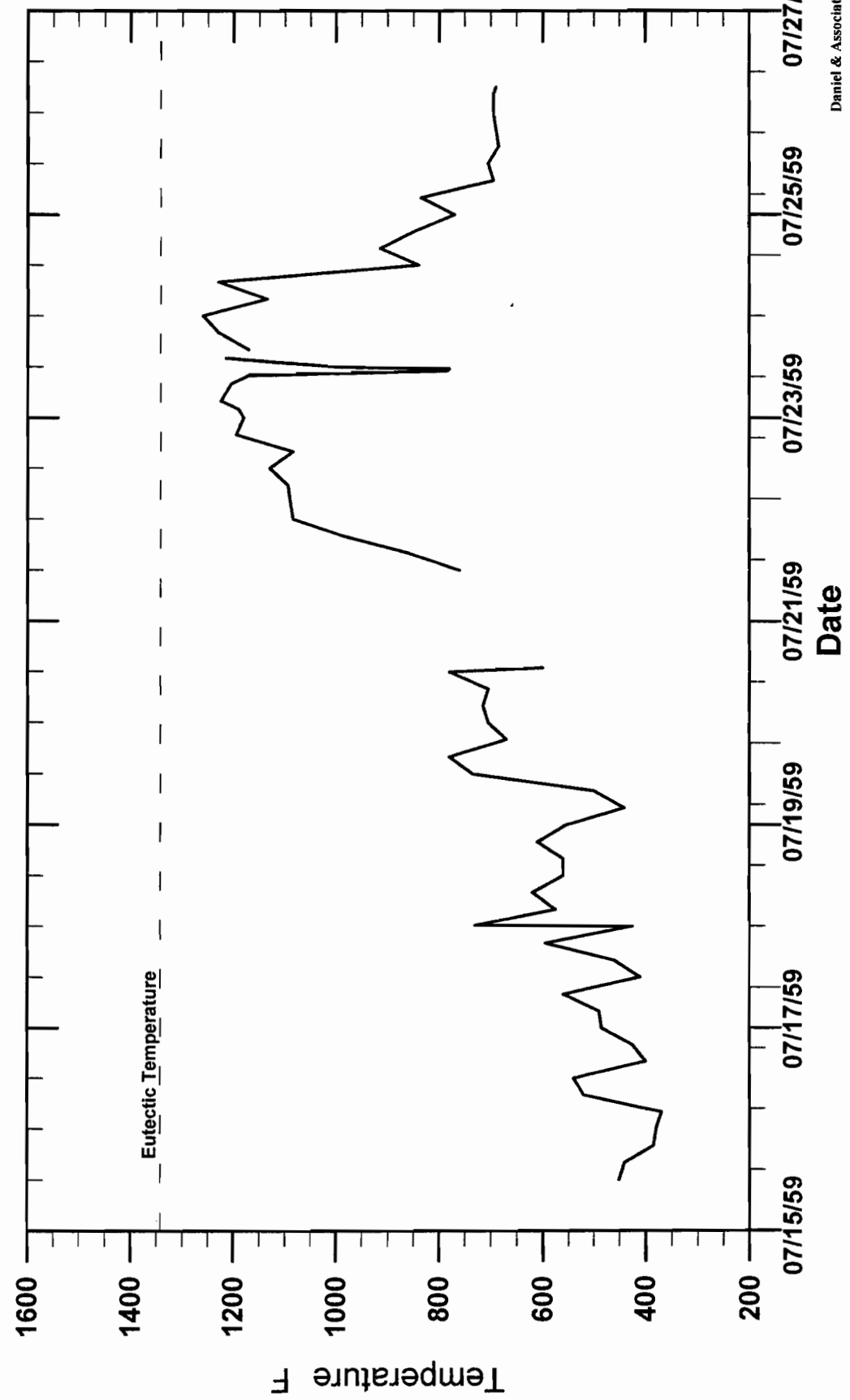


Figure 4.5 Channel 55 Thermocouple Temperature TC10 (21 inches above midplane)

4 - 15

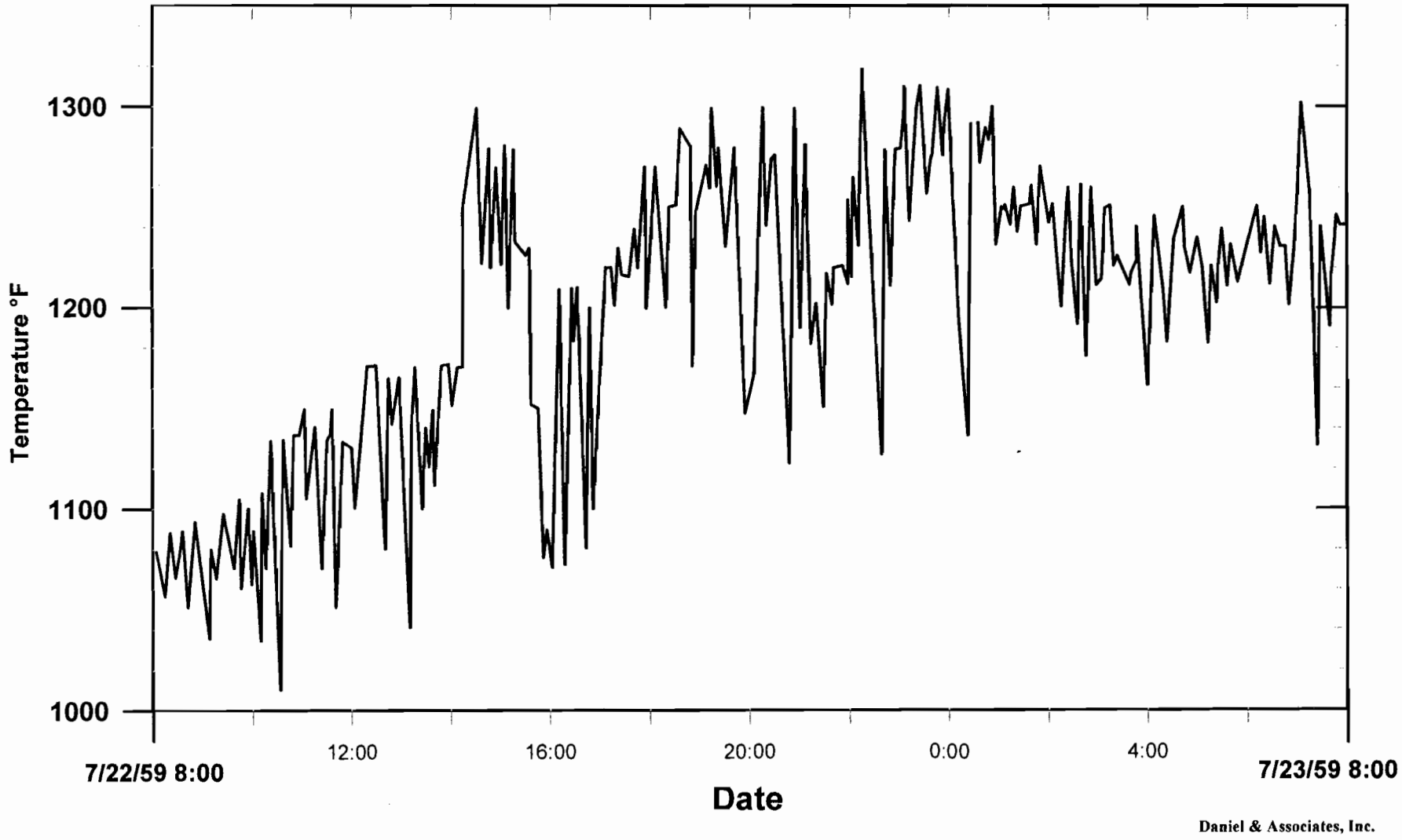


Figure 4.6 Channel 55Temperature Oscillations

4 - 16

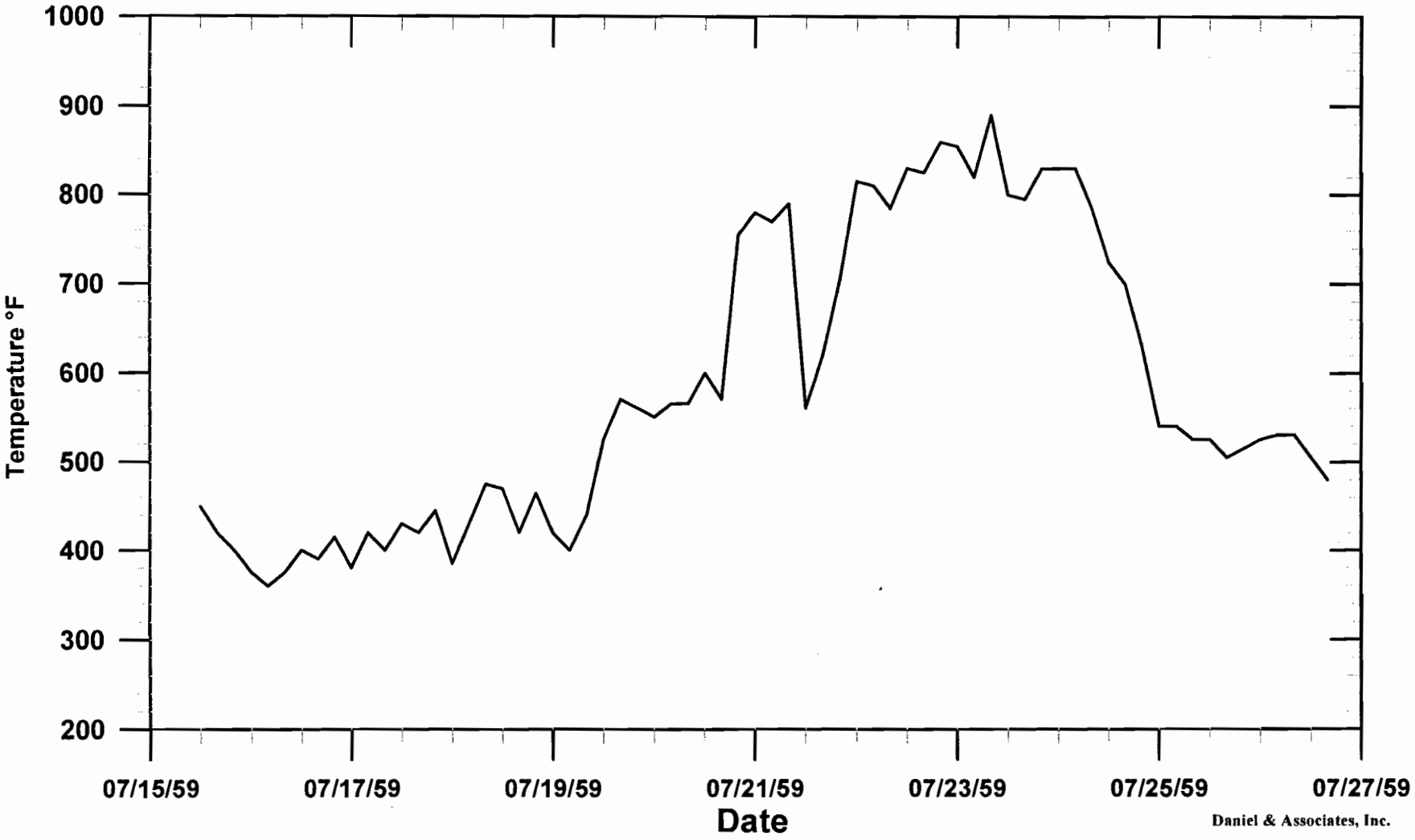


Figure 4.7 Channel 55 Sodium Exit Temperature

4 - 17

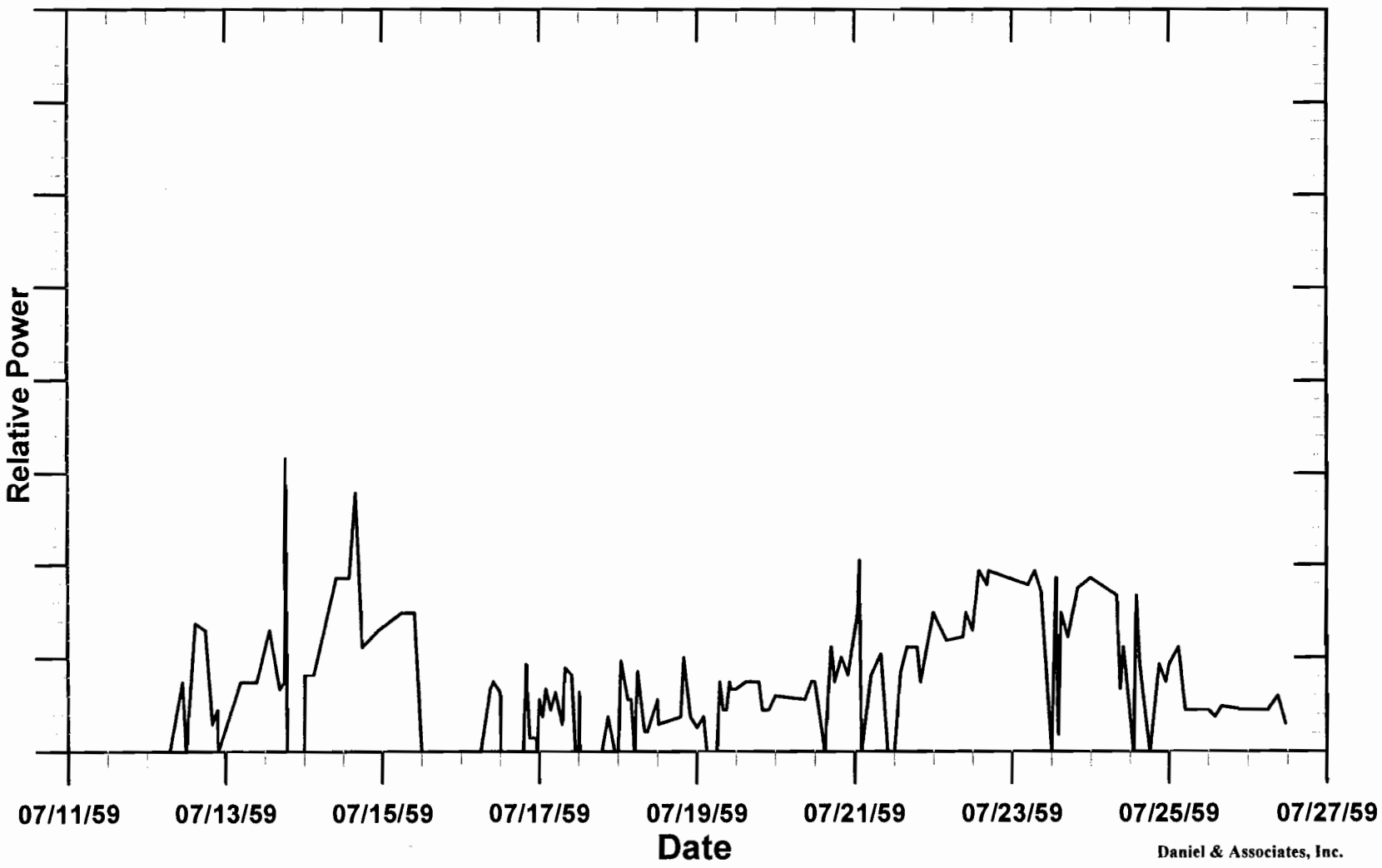


Figure 4.8 Relative Reactor Power Level as Indicated By Log Neutron Flux

4 - 18

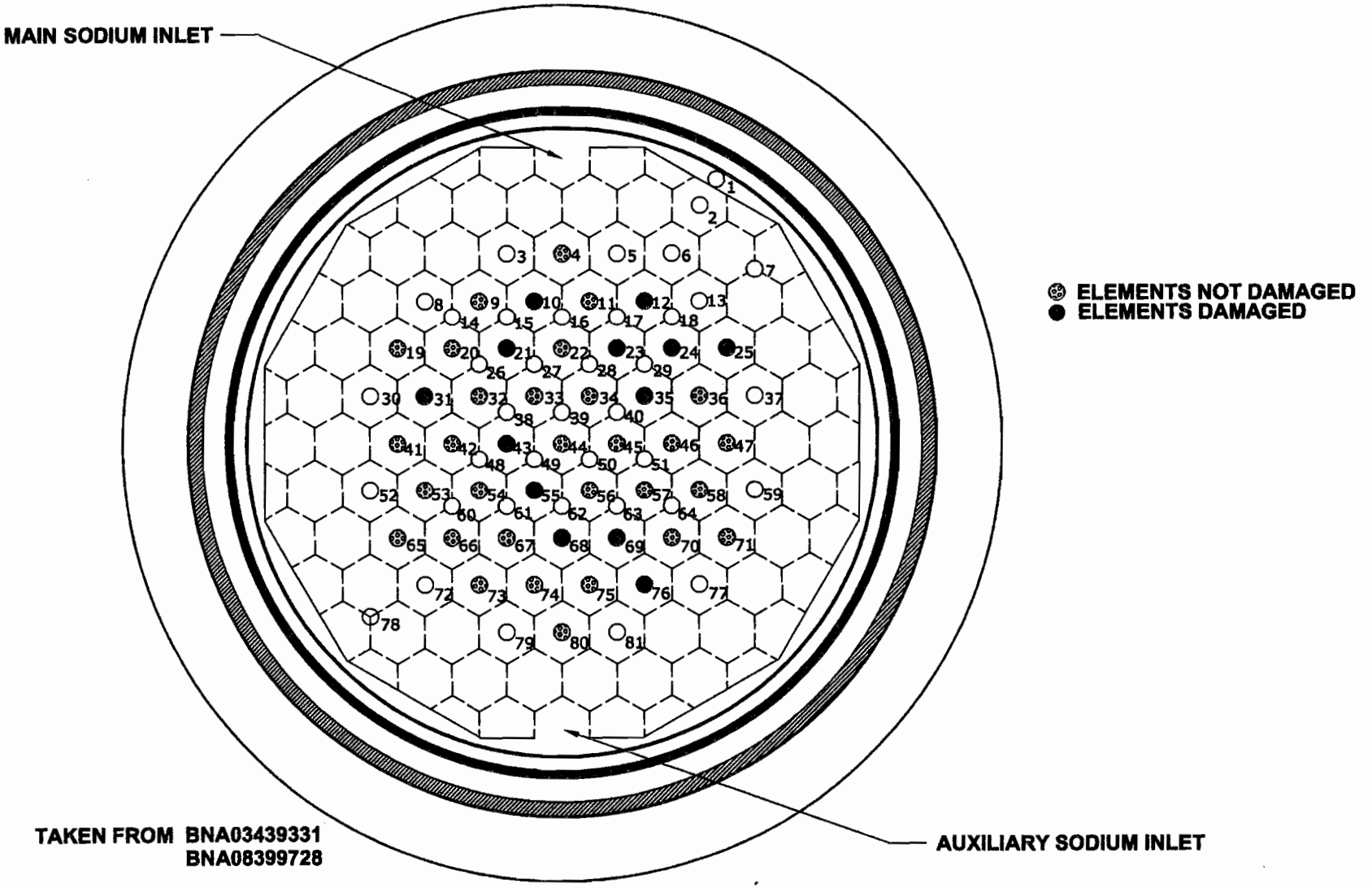


Figure 4.9 Location of Damaged Fuel Elements

4 - 19

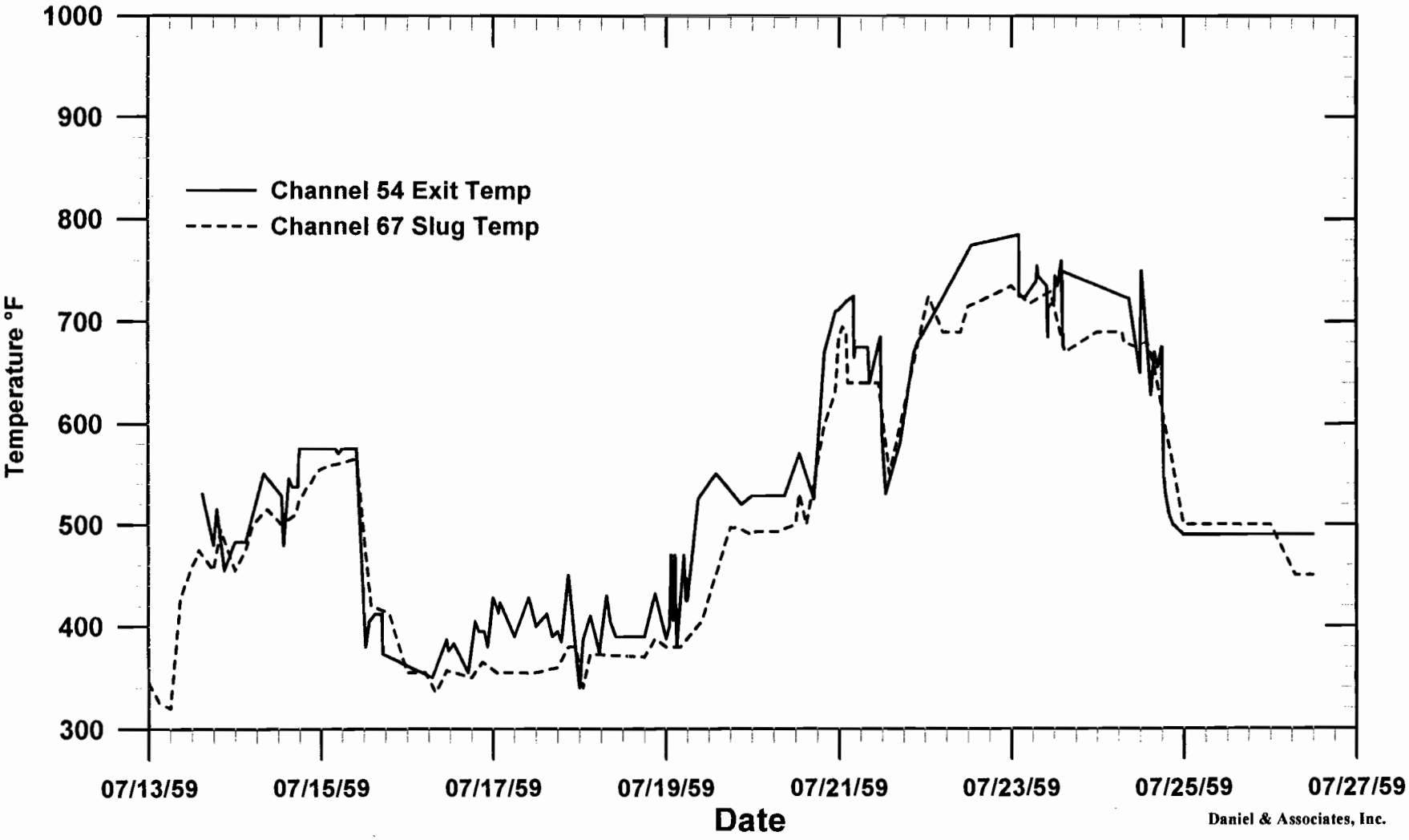


Figure 4.10 – Core Channel 54 Exit Temperature with Core Channel 67 Fuel Slug Temperature

4 - 20

5. Radiological Impact of Fuel Damage at SRE

5.1. SRE Fuel Inventory

A material balance of the fission products released from the fuel must of necessity begin with an accurate inventory of the fission products present in the fuel before fuel damage occurred, or at least at some point in time that can be benchmarked to work done by the original investigators of the incident. The core inventory of the SRE was calculated using the ORIGEN2 computer code.[Ref. 21] These data were compared to the fission product inventory calculated by Hart, [Ref. 4] and are shown in Table 5.1, below.

The fission product inventory was calculated using the power level and burnup history for the SRE. [Ref. 4] Average power levels were used for Runs 1-13, and actual power levels as could be determined from the references for Run 14. The total burnup for Run 14 was compared to the calculations, and adjustments made to the calculations so that short lived fission products would be computed as accurately as possible.

The differences between the two core inventories were investigated. The large difference for Xe-133 was found to be inclusion of precursor radionuclides in the Daniel computation, whereas Hart apparently considered only the single isotope. There were some differences in half-life for some of the isotopes, such as cesium. Some of the differences were attributable to the fission yields used for individual isotopes.[Ref. 4, 37, 40] The values of half-lives and fission yields have been improved since 1959. It is believed that the Xe-133 inventory as shown in this report is the more accurate of what inventory actually was present. Computation of multiple branching decay chains is a very complicated process, and not easily accomplished by hand calculations as was done in 1959.

Table 5.1

SRE Core Inventory Comparison

Isotope	Hart Inventory	Daniel Inventory		% Difference
Cs-134*	2.00 (2)	*f*-		
Cs-137	8.70 (3)	7.89 (3)		9
Sr-89	1.60 (5)	1.29 (5)		19
Sr-90	8.15 (3)	7.80 (3)		4.3
I-131	1.68 (4)	1.92 (4)		14.3
Ce-141	1.27 (5)	1.22 (5)		4
Ce-144	1.69 (5)	1.54 (5)		9
Ru-103	7.52 (4)	7.18 (4)		4.5
Ba(La)-140**	5.61 (4)	5.54 (4)		2.3
Zr(Nb)-95#	5.53 (5)	1.97(5)		9
Kr-85	1.10 (3)	9.82 (2)		10.7
Xe-133	5.08 (4)	9.22 (4)		81
Xe-131M	-	7.15 (2)		
I-133		1.19(4)		
I-135		2.35(3)		

As of shutdown, 7/26/59 11:20 AM

* Estimated

f Not Computed – not fission product

5.2. Damaged Fuel Inventory

A fraction of the inventory in the 13 damaged fuel assemblies is the only amount that could have been released to the coolant and cover gas. The damaged fuel inventory was calculated by proportioning the core inventory by the amount of burnup in the damaged fuel assemblies. The burnup of each fuel assembly was given in references [Ref. 2, 22] and the fraction of total burnup for each fuel assembly in the core calculated. The individual burnup for the damaged fuel assemblies were then summed to determine the fraction of core inventory contained in damaged fuel assemblies. These data are shown in Table 5.2. The percentage of damaged fuel turned out to be 32.4 % of the total core inventory, which is the absolute maximum that was available for release, since the remainder of the assemblies had no cladding failure at all. The actual amount released would not be the entire inventory of the damaged assemblies, but some fraction thereof, depending on the chemical characteristics and properties of the individual fission products. The actual release from fuel occurred only from that fraction that reached the eutectic temperature or the temperature of the $\alpha - \beta$ phase transformation.

The release fractions contained in this report are fractions of *total core* inventory, as is the convention when describing the magnitude of release from fuel.

Table 5.2 Fuel Burnup Data

				Fuel Loading		2.97 MT	
				Total Burnup		2426 MWD	
Experiment #	Reactor Location	Fuel Type	Damaged	Most Recent Inspection Date	Total Burnup Burnup MWD/T	Total Burnup (MWD)	Fraction of Burnup
SU-1-5	R-4	STD		05/03/59	581.100	49.578	0.020
SU-1-6	R-9	STD		07/10/59	440.700	37.600	0.015
SU-1-47	R-10	STD	√	04/26/59	**645.200**	55.047	**0.022**
SU-1-8	R-11	STD		04/25/59	793.700	67.717	0.027
SU-1-9	R-12	STD	√	05/06/59	**747.800**	63.801	**0.026**
SU-7-1	R-19	UO2		05/09/59	462.700	39.477	0.016
SU-1-11	R-20	STD		04/26/59	826.200	70.490	0.028
SU-1-12	R-21	STD	√	04/26/59	**853.000**	72.776	**0.029**
SU-1-13	R-22	STD		04/26/59	809.200	69.039	0.028
SU-1-14	R-23	STD	√	04/26/59	**801.300**	68.365	**0.027**
SU-1-15	R-24	STD	√	02/07/59	**303.500**	25.894	**0.010**
SU-1-40	R-25	STD	√	04/26/59	**708.900**	60.482	**0.024**
SU-1-17	R-31	STD	√	04/26/59	**745.100**	63.571	**0.025**
SU-22-1	R-32	Th-U		02/07/59	447.700	38.197	0.015
SU-2-1	R-33	STD		07/10/59	939.500	80.156	0.032
SU-2-2	R-34	STD		04/26/59	1117.000	95.300	0.038
SU-1-10	R-35	STD	√	04/26/59	**543.200**	46.345	**0.019**
SU-1-27	R-36	STD		04/26/59	472.200	40.287	0.016
SU-1-18	R-41	STD		04/26/59	734.200	62.641	0.025
SU-1-19	R-42	STD		05/03/59	963.900	82.238	0.033
SU-2-3	R-43	mxd	√	04/26/59	**1144.400**	97.638	**0.039**
SU-22-2	R-44	STD		05/09/59	463.300	39.528	0.016
SU-1-20	R-45	Th-U		07/10/59	876.200	74.756	0.030
SU-1-21	R-46	STD		07/10/59	843.500	71.966	0.029
SU-1-22	R-47	STD		07/10/59	266.200	22.712	0.009
SU-1-23	R-53	STD		07/10/59	516.200	44.041	0.018
SU-1-24	R-54	STD		07/10/59	759.500	64.799	0.026
SU-2-5	R-55	mxd	√	02/06/59	**732.800**	62.521	**0.025**
SU-1-44	R-56	STD		07/10/59	593.200	50.611	0.020
SU-12-1	R-57	STD		05/12/59	100.800	8.600	0.003
SU-1-28	R-58	STD		07/10/59	867.800	74.039	0.030
SU-1-29	R-65	STD		07/10/59	620.800	52.966	0.021
SU-1-30	R-66	STD		07/10/59	790.300	67.427	0.027
SU-1-26	R-67	STD		07/10/59	511.700	43.657	0.017
SU-1-31	R-68	STD	√	07/10/59	**870.300**	74.252	**0.030**
SU-1-32	R-69	STD	√	07/10/59	**880.200**	75.097	**0.030**
SU-1-2	R-70	STD		07/10/59	324.100	27.652	0.011
SU-1-33	R-71	STD		07/10/59	671.300	57.274	0.023
SU-1-34	R-73	STD		07/10/59	739.600	63.101	0.025
SU-1-35	R-74	STD		03/07/59	833.500	71.113	0.028
SU-1-36	R-75	STD		02/05/59	812.000	69.278	0.028
SU-1-4	R-76	STD	√	01/25/59	**518.300**	44.220	**0.018**
SU-1-37	R-80	STD		05/08/59	635.900	54.254	0.022
					Damaged Fraction:		**0.324**

5.3. Release From Fuel

Metallic uranium fuel such as that used in the SRE differs from oxide uranium fuel in that there is no oxygen bound with the uranium, and therefore the fuel is much more dense. The lack of oxygen in the metallic fuel affects the types of chemical compounds that can be formed with fission products that are generated in the fuel. For this reason, one must be careful to consider only those mechanisms that are relevant to metallic fuels when computing a release from fuel. The release of fission products from the SRE fuel was temperature driven (as the semi-solid uranium formed an alloy with the cladding) with the release occurring at elevated temperatures relative to normal operation. Following the formation of the alloy, or simultaneously with the formation of the alloy, the cladding was breached along some portion of the length of the fuel rod that formed the alloy. [Ref. 42, 47] In some instances, cladding was ruptured as a result of fuel swelling, caused by cycling through the $\alpha - \beta$ phase transformation temperature. [Ref. 30]

5.3.1. Release to Cover Gas

There are three mechanisms for release of noble gases with respect to the SRE fuel failure. Two of these, *diffusion* out of the fuel and fission *recoil*, were investigated and found to be so small as to be insignificant when compared to the release at the formation of the iron-uranium alloy.

Thermal cycling was identified as a reason for cladding failure, by cycling back and forth across the uranium phase change temperature. [Ref. 3,30] This occurred primarily between July 22 and July 24. If one assumes fuel failure occurred in the middle of this period, (July 23) the core inventories of Kr^{85} and Xe^{133} in the damaged fuel assemblies are 982 curies and 1.15×10^5 curies, respectively. An iterative technique was then used with the core inventory as of July 23 to determine the release fraction from fuel to be applied to all noble gases. This resulted in a release fraction of 0.015 for both Xe^{133} and Kr^{85}. A release fraction of 1.5% is considered conservative for the SRE on the basis of the actual sample data. A curie balance of holdup tank inventories was used to check the release data. These release fractions were compared to those obtained by Hart et. al., as shown below:

Table 5.3 Gaseous Release

Isotope	* Hart* Release Fraction	** Daniel Release Fraction	* Hart Release (Ci)	* Daniel Release (Ci)
Kr-85	0.00018	0.015	0.2	14.3
Xe-133	0.00092	0.015	47	1265

* As of 7/26/59 ** As of 7/23/59

The release fraction is computed by dividing the released amount by the core inventory at the time of release. The apparent disparity between Hart's release fractions and those presented in this report can be partially explained by the fact that Hart determined the release fractions based on cover gas samples taken on August 12, after venting some of the cover gas to the holdup tanks between the time of fuel damage and the actual sample date, and then back-calculated theinventories to July 26. The decrease in power level at the end of Run 14 skewed the inventory of Xe^{133} such that it was less than the inventory at the time of release (July 23), $9.2x10^4$ vs. $1.15x10^5$ curies. Although the fractions removed by venting the cover gas were small as compared to the total volume, (~20-25%), there were additional uncertainties in Hart's fractions. Hart apparently used a different cover gas volume than indicated by the dimensions of the reactor and fill tank volumes. In fact, the volumes associated with the Hart Kr^{85} release does not agree with the Xe^{133} release, although it was stated that a "factor of two [was included] to account for probable losses due to pressure manipulations." [Ref. 4] However, the factor of two was only applied to Xe^{133}. The core inventories also differ between Hart and this report for Kr^{85} and Xe^{133}. It was determined that Hart had not included precursor radionuclides for Xe^{133}.

The release fractions in this report are dated as of the time of release, July 23, and include the amounts transferred to the holdup tanks between July 12 and 20th. The release quantities (last column) from this investigation are decay corrected to July 26, for comparison to Hart's release quantities.

Figure 5.1 shows the release from fuel for noble gases as calculated above compared to cover gas measurement data taken at various dates. The solid line represents the decay of

the sum of Kr^{85} and Xe^{133} released to the cover gas following release from the fuel on July 23rd, in order to compare to the gross activity measurement data. No attempt was made to proportion the activity between cover gas and holdup tanks in this figure – it represents the total release from fuel of gaseous fission products.

Figure 5.2 shows the comparison of calculated Xe^{133} in the cover gas compared to the two samples that were analyzed for specific isotopic content. Figure 5.2 shows relatively good agreement with the Xe^{133} sample data, and also with the gross measurement data. Xe^{133} was the predominant radionuclide until around August 28, when Kr^{85} became predominant. The non-decay decrease is due to transfer of cover gas to holdup tanks on July 25, 27 and 29th. Decay and transfer of cover gas was modeled with the RADTRAN code. A description of the RADTRAN mathematical model is contained in Appendix G.

Figure 5.3 shows the comparison of calculated Kr^{85} to sample data. The comparison is not as good as Xe^{133}, but shows the calculated Kr^{85} to be within 5 curies of the measurement data. A small increase in either the Kr^{85} concentration would produce much better agreement with the 0.015 release fraction. It is not likely that the unaccounted for Kr^{85} was released from the plant, since the xenon would have been transported with it, and the xenon is accounted for in the plant. It is also possible that there are errors in analysis in the measurements of Kr^{85}. The author has experienced such counting errors, which usually are low with respect to actual concentrations.

Figure 5.4 shows the release from fuel in comparison to all cover gas and holdup tank data. This curve represents the total release from fuel, decayed until November 1.

Figure 5.5 demonstrates the results of the iterative analysis to determine release from fuel. The requirement that the release fit the sample data translated into the condition that the crossover of Xe^{133} to Kr^{85} occur on or about August 28, as was recognized by Hart. It is this point that Kr^{85} became the predominant radionuclide in the cover gas, and waiting for additional decay before disposing of the gas would not gain any advantage to the cleanup of the incident. The second requirement that must be met is that the total activity collected in the holdup tanks, when decay corrected to July 26, must be met by the release from fuel also. The iteration progressed by assuming a release rate from fuel, and using the core ratio of Xe/Kr, generate total release from fuel until the two conditions

were met. It was determined that a release of 1.5% met the conditions. Investigation of potential pathways resulted in the determination that there were no unidentified releases to the environment that occurred prior to the first measurements of cover gas activity.

The fact that the Xe^{133} and Kr^{85} activity does not decrease in the cover gas samples between August 12 and September 14 (other than radioactive decay) indicates that no transfer of cover gas was done during this period. However, beginning around 9/14/59, transfer of the cover gas to the holdup tanks began, as indicated by the samples of holdup tanks and by the decrease of Kr^{85} inventory in the cover gas. Figure 5.4 shows the comparison of calculated cover gas inventory to all sample data, including the holdup tank inventories. The effectiveness of Kr^{85} removal shows that there was some difficulty in reducing the cover gas inventory, as indicated by the number of holdup tank volumes generated. This may be due to the fact that the purge line for the cover gas is located between the core tank and primary sodium fill tank. As "contaminated" helium is withdrawn and sent to the holdup tanks, clean helium must be provided as makeup, to prevent drawing a vacuum on the cover gas. This serves to dilute the cover gas, and makes it more difficult to remove the remaining contaminated gas.

This analysis clearly shows that the release of noble gases from the fuel can be accounted for as remaining within the plant with reasonable certainty.

5.3.2 Release to Primary Sodium

The release from the fuel to the primary sodium is relatively straightforward, as no adjustments have to be made for sodium removal during the incident as in the cover gas. The release fractions and release quantities (Curies) at the time of release for selected isotopes are shown in Table 5.4 in comparison to Hart's calculations:

Table 5.4 Non-Gaseous Release (Fraction of Core Inventory)

Isotope	Hart Fraction	Daniel Fraction	Hart Release	Daniel Release
I-131	0.00097	0.0011	16.3	25.8
I-133	(Not Given)	0.0011	(Not Given)	74.9
I-135	(Not Given)	0.0011	(Not Given)	65.3
Sr-90	0.00026	0.0011	21.4	8.6*
Ce-141	0.00076	0.0011	96.5	142
Ce-144	0.00067	0.0011	141.0	170.5
Cs-137	0.00318	0.0040	27.7	31.6

* $Sr^{90}/Y^{90} = 17.2$

Comparison of release fractions may not be meaningful since Hart based his figures on core inventory as of July 26, after additional irradiation and changes in power level had occurred. The most meaningful comparison in this case is the quantity of release, which differ by only few per cent. It is not clear how Hart's Sr^{90} sodium concentration was determined, since techniques for counting and reporting Sr^{90} differ between laboratories. It is common practice (currently) to count the ingrowth of the daughter radionuclide Y^{90} to determine the Sr^{90} content of a sample, and to signify this technique by reporting Sr^{90}/Y^{90} together to avoid confusion. For this reason, the Sr^{90} release reported in this analysis should be multiplied by a factor of 2 to account for the contribution of Y^{90} in order to compare to Hart's Sr^{90} release. This assumes that Hart's reported value for Sr^{90} is actually the Sr^{90}/Y^{90} laboratory result multiplied by the sodium inventory.

Figure 5.6 shows the results of this analysis compared to the gross measurements at different dates and reported in Reference 4. It is interesting to note that sampling to determine gross activity of the primary sodium did not begin until the short lived isotopes such as Na^{24} and I^{135} in the primary sodium had decayed to insignificant levels.

Figure 5.7 shows the comparison to individual radionuclides in the primary sodium as determined by spectroscopic analysis. The counting of Sr^{90} involves chemical separation, but is included in Figure 5.7 also.

Figure 5.7 shows the effect of cold-trapping and/or deposition on metal surfaces to remove particulate material from the primary sodium.[Ref. 45] The soluble fission products I^{131} and Cs^{137} follow their decay curve very closely, indicating that they went into solution and remained in solution, whereas Ce^{141} and Ce^{144} were greatly reduced between the sampling dates of August 2nd and October 31st. The iodine and cesium curves fall very close to their sample points on August 2, which shows very close agreement with the release fraction of 0.11% as determined in this report.

In summary, the fission product release from the fuel to the sodium in the SRE fuel damage incident was small even though approximately one-third of the fuel assemblies were damaged to some extent. The small release is attributable to the characteristics of natural uranium fuel.

5.4 Release to Environment

The available data demonstrate that the radioactive vent system was aligned to the suction tank during Run 14, and therefore there was no direct, continuous pathway for gases to be released to the environment from the reactor cover gas. However, the response of the stack monitor indicates increased activity on two occasions, both of which coincide with operation of the fuel handling cask. By process of elimination of possible pathways for a release, it was determined that the fuel handling cask venting through header 506 was the likely cause of the stack monitor response, both in time of occurrence and in magnitude of the response.

Each venting of the lower gas lock on the fuel handling cask releases approximately one cubic foot of gas into header 506. If one assumes that the fuel handling cask was moved and sealed repeatedly during the 5 hour period between 5:00 PM and 10:00 PM on July 12th, then it becomes a question of how many times a release of 1 ft^3 occurred. The investigation of the cause is aggravated by the fact that there is no record of the stack monitor's response during these two time periods. The activity of the cover gas, as indicated by the samples of the holdup tanks on July 20th, was approximately 5×10^{-2}

μCi/cc. If one assumes that the predominant radionuclide on July 20th was Xe^{133}, then the concentration of Xe^{133} on July 12th would have been 0.14 μCi/cc. Using the concentration of July 20, and decay correcting to July 12 to get a concentration for release from each venting of the fuel handling machine, and adding the short-lived isotopes into the mix yields a concentration of approximately 3.15 μCi/cc on July 12. Therefore, each venting from the fuel handling machine would release 3964 μCi, or approximately 4,000 μCi (0.004 Ci) of Xe^{133}. Adding short lived radionuclides by using the ratio of those radionuclides (such as Xe^{135} and Xe^{135M}) to that of Xe^{133}, the release for each venting would be 25,000 μCi (0.025 Ci). This is an insignificant amount compared to venting a holdup tank. However, the release of 1 ft^3 results in a concentration at the stack of $8.8x10^{-5}$ μCi/cc as compared to the reported stack activity of 1.5x10-4 μCi/cc. This is very good agreement, and reinforces the conclusion that the fuel handling cask was the source of the stack activity.

The releases to the environment are shown in Figure 5.8, and were controlled releases from the holdup tanks according to procedure. Each tank was sampled prior to release, and the release rate was determined for each release so as not to exceed the release rate for the plant. The radionuclides Xe^{133} and Kr^{85} were the predominant radionuclides released.

5.5 Environmental Samples

Environmental air sample data were reviewed to determine if they provided any indication of a release of radioactive material from the SRE in July, 1959. There were no increases above background for the month of July 1959. Of particular interest were the beta-gamma air samples taken from the Santa Susana site, as well as Van Owen. Air sample data for Santa Susana and the Van Owen Facility are included as Appendix F. The environmental air sample data provides total activity above background for all beta and beta-gamma isotopes. Most of the environmental activity is attributable to Be^7 and K^{40}, which are naturally occurring isotopes. Radionuclide fallout from weapons testing reached a peak around 1962, when atmospheric testing was being conducted by both the U.S. and the Soviet Union. In 1959, the activity in the atmosphere was typically 10^{-14} Ci/cc , whereas today the activity is more on the order of 10^{-9} Ci/cc, or about 100,000

times greater.

Figure 5.9 shows the beta-gamma activity for the months of May-September, 1959 for the Santa Susana monitoring station. Figure 5.10 shows the beta-gamma air sample data for roughly the same period for the Van Own facility, located in Canoga Park, in a southeasterly direction.

There is no evidence of uncontrolled release to the environment from the fuel damage incident at the SRE. The release of the accumulated noble gases in the holdup tanks during the period July 20th through September 19th was conducted with a very small release rate, and was not detectable above background levels.

5.6. Experimental Evidence of Fission Product Release

Following the fuel damage incident, the AI investigators began a series of experiments to determine the root causes of the accident, and to determine why the fission product release fractions were not as expected. One experiment involved irradiating a fuel disk with a burst of neutrons until the melting temperature was reached. [Ref. 24]. A 93% enriched uranium foil was irradiated in a stainless steel capsule partially filled with sodium. The stainless steel capsule had a cover gas in the top to simulate reactor conditions. After irradiating the capsule to generate fission products in the fuel, the capsule was removed and the fission products were measured. In this experiment

> "..in which the fuel was melted and dispersed in the sodium, only very volatile fission products were released from the fuel in quantity. This was apparently due to the excellent heat transfer characteristics of sodium which caused rapid refreezing of the fuel and subsequent trapping of the fission products." [Ref. 24]

The above experiment demonstrates the fact that the release from fuel would be a short duration and quickly terminated, because of the heat transfer characteristics of sodium coolant.

A second experiment dealt with the release of iodine from sodium into the cover gas. In this experiment, capsules containing non-radioactive molecular iodine vapor were placed at different depths in sodium to represent iodine evolving from the coolant into the cover

gas. In this experiment, nitrogen gas was in the capsule along with the sodium, and the nitrogen/iodine gas was then released from the capsule. The gaseous bubbles were then allowed to travel up through the sodium to the cover gas region at the top of the experimental column. The experiment was conducted at temperatures ranging from 500 to 1000°F. The investigators reported:

> "It was observed that vaporized iodine released in the form of gradually ascending bubbles (in quantities up to 60 mg to 500°F sodium at a depth of 6 to 10 feet) was largely absorbed by the sodium. A maximum of only 1-½ % escaped to the cover gas." [Ref. 24]

The second experiment described above differs from the SRE actual conditions in that the iodine was released from a capsule into the sodium as molecular iodine directly. Other studies have shown that with low burnup metallic fuels, the fission gases are essentially retained in the fuel for the temperatures that the SRE experienced.[Ref. 47]

5.6.1. The Fermi Fuel Melt Experience

On October 5, 1966 an incident in which a reactor internal flow guide plate became dislodged and blocked the flow of coolant to the core resulted in melting several fuel assemblies. The Fermi reactor was a sodium cooled, graphite moderated reactor larger than the SRE. In this incident, an equivalent of one fuel assembly (140 rods) underwent melting, and flow was estimated to have been reduced to about 3 %.[Ref. 25] Measurements showed that approximately all of the Kr^{85} in the melted fuel was released. Approximately 1 to 10 % of the cesium and iodine was released to the coolant. None of the released iodine was found in the cover gas. [Ref. 26].

5.6.2. Experience with Cladding Failure

Following the fuel damage incident, Atomics International investigated the cladding failure, after noting that the assemblies in channels 12, 21 and 25 were found to have cladding damage (splits and cracks) but no apparent eutectic formation. [Ref. 30 pp 33] Experiments were conducted to determine the mode of failure. It was discovered that by cyclically heating and cooling the uranium/cladding samples across the $\alpha - \beta$ phase transition temperature, the cladding failed within 24 hours, after a total of 275 to 300

cycles. The experiment was performed on unirradiated fuel, whereas the actual fuel in the core had been previously irradiated, and Run 14 was the last run before changing fuel. It is likely that the cladding failed before it reached the same number of cycles as the unirradiated fuel. The experiment reinforced the conclusion of thermal cycling temperature failure of the SRE cladding. [Ref. 30 pp 34]

The fuel assembly in channel 55 recorded a maximum recorded temperature (TC09) of 1465 °F on July 22. [Ref. 2] The boiling point of NaK is 1445 °F. The boiling point of sodium is around 1620 °F. If the thermocouple temperature 9 inches above the midplane was measured at 1465 °F, it is quite possible that the midplane was even higher, and that the NaK in the fuel rod gap exceeded the boiling point. This would severely stress the cladding by internal pressure as the NaK expanded, and may have contributed to the cladding failure, even though the sodium coolant itself had not reached the boiling point.

In studies done following the EBR-I fuel melt, it was reported that

> "The heat of formation of the uranium-iron eutectic provides a heat source in addition to fission. It has been recently measured and found to be ~ 4 calories/gram of eutectic formed. This additional heat would tend to counterbalance heat transfer from the core so that the heat content [of the fuel] remains about the same." [Ref. 31]

Because of the excellent heat transfer capabilities of sodium, this phenomenon of additional heat would be a very short duration event, with rapid refreezing of the uranium and thereby limiting the release from fuel. However, the internal pressure buildup due to NaK boiling in the fuel rod gap provides a mechanism for cladding failure without the sodium coolant reaching the boiling point.

In the case of boiling the NaK as a mechanism of cladding failure, the temperature of the fuel would have exceeded the eutectic temperature prior to the NaK reaching the boiling point. Any radioiodine that would have escaped the fuel would have reacted with the NaK (vapor or liquid) and formed NaI or KI, prior to cladding failure.

The point to understand is that the environment for cladding failure to have occurred is in the July 22-24 time frame. The plaintiff's expert contention that major fuel damage with

releases to the environment occurred between July 12 and July 15 is refuted by the plant temperature recordings. Furthermore, the vent system was aligned such that even if plaintiff's arguments were true, the release would have been contained in the holdup tanks.

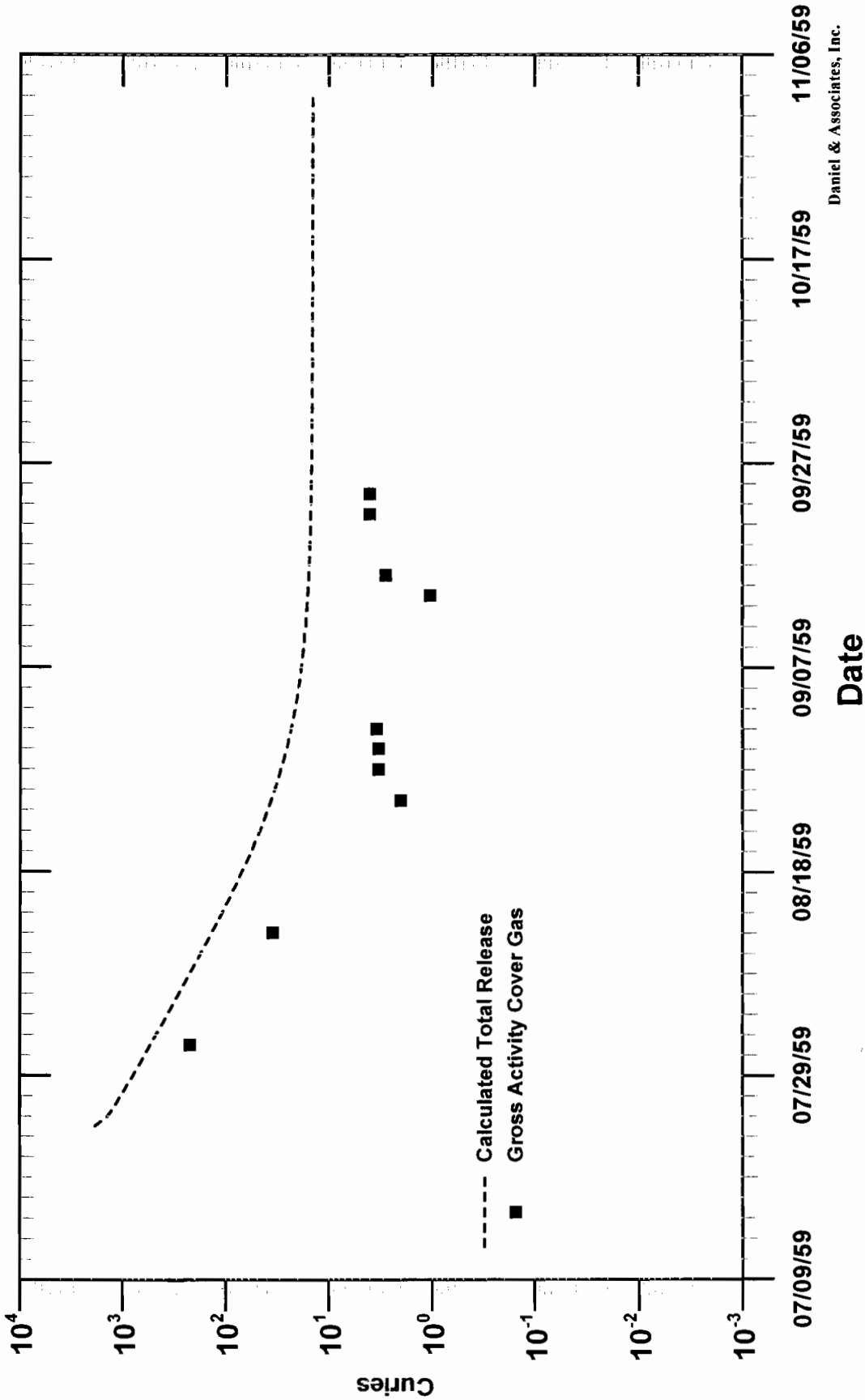


Figure 5.1 Cover Gas Total Activity Decay Curve

5 - 15

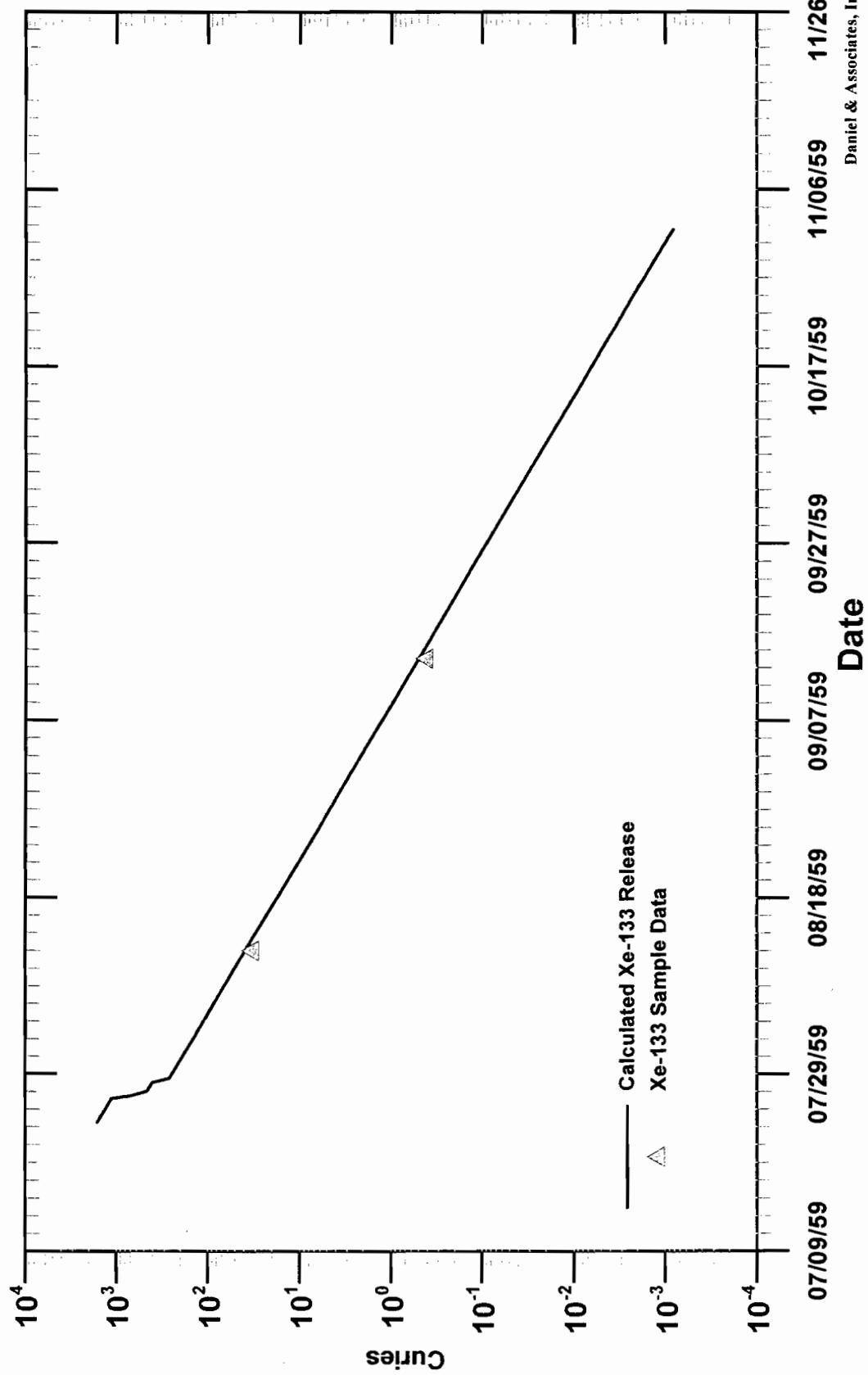


Figure 5.2 Cover Gas Xe-133 vs. Sample Data

5 - 16

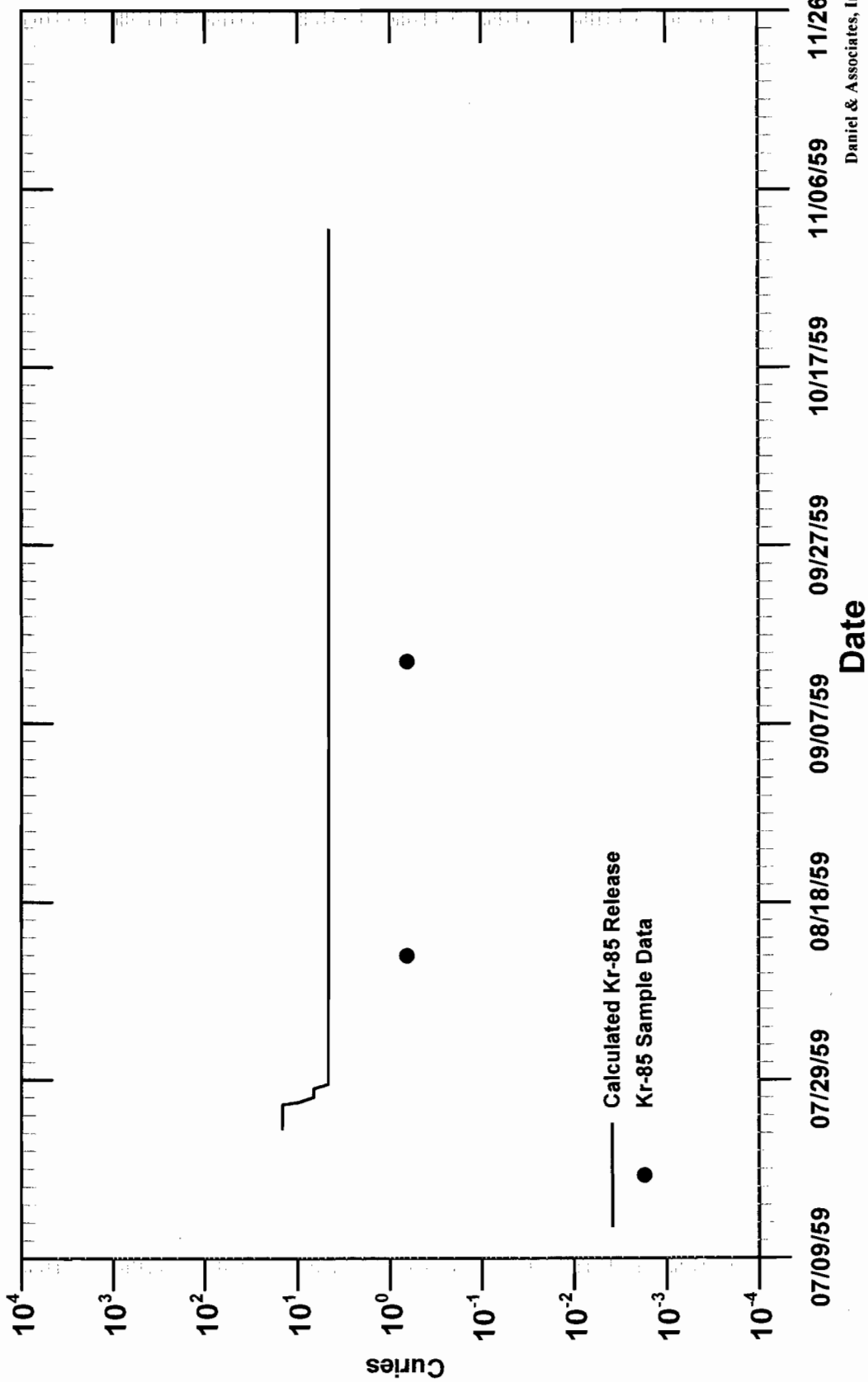


Figure 5.3 Kr-85 Cover Gas vs. Kr-85 Sample Data

5 - 17

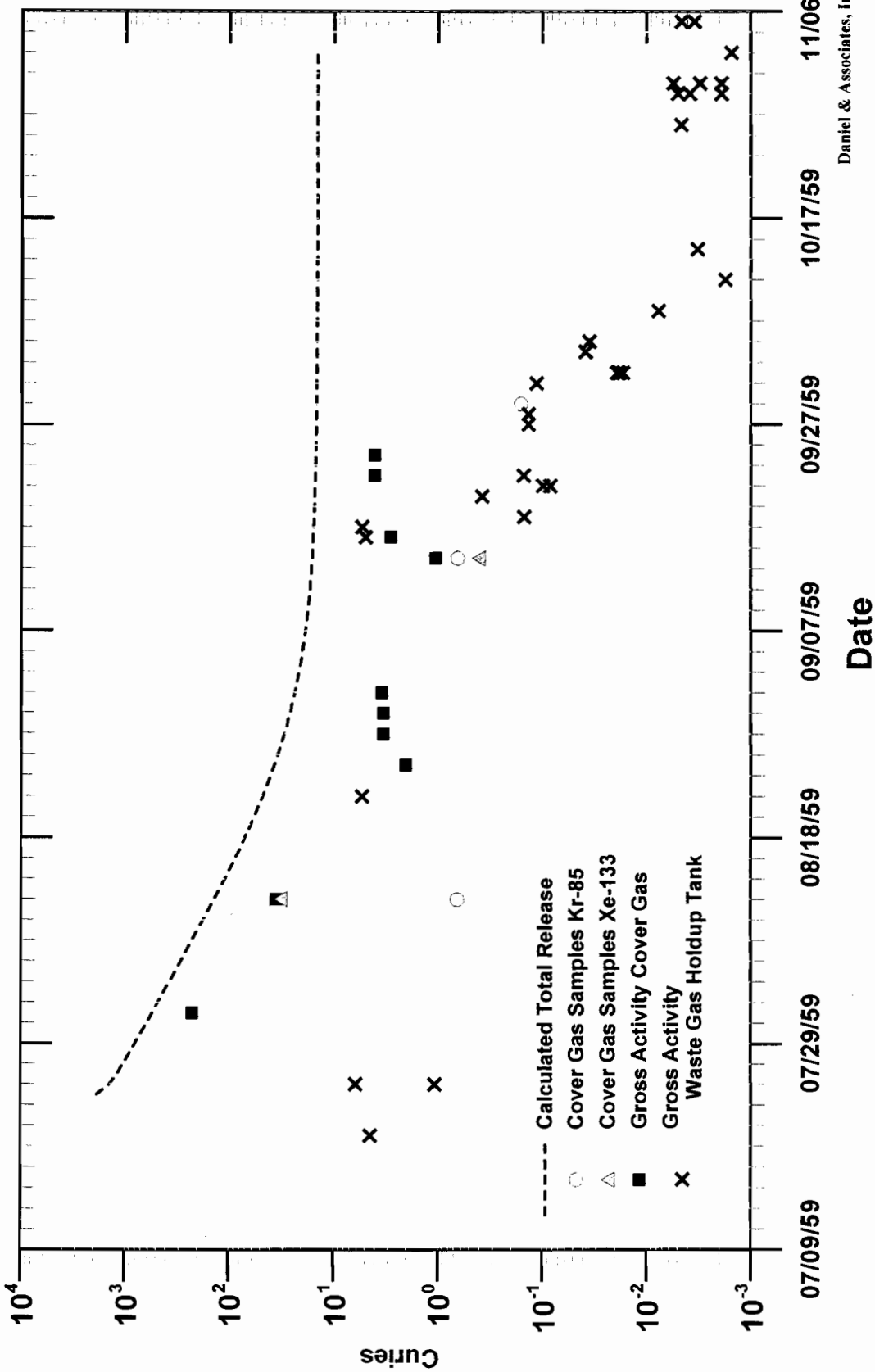


Figure 5.4 Total Cover Gas vs. Holdup Tank Sample Data

5 -18

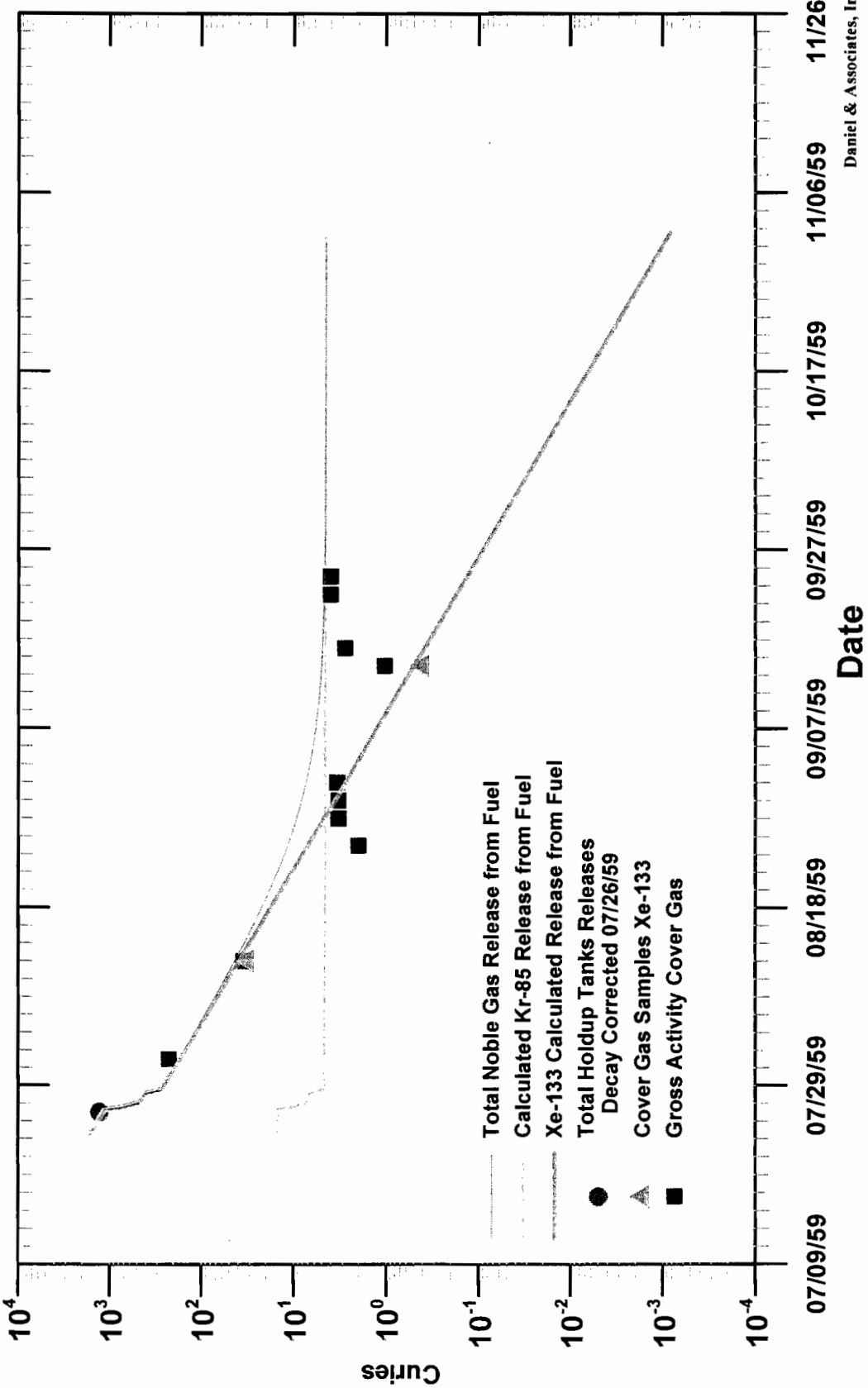


Figure 5.5 Release from Fuel Compared to Sample Data

5 - 19

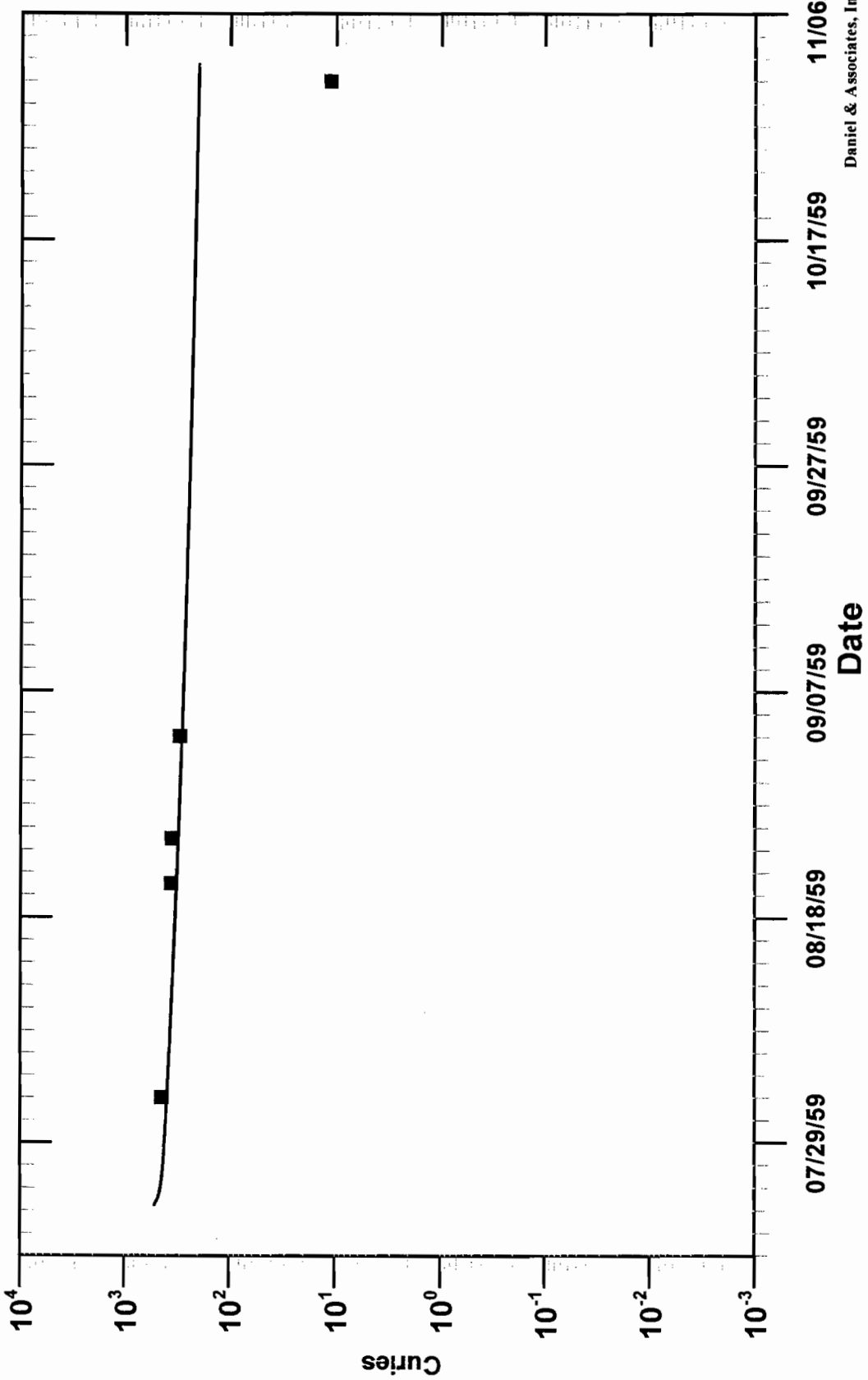


Figure 5.6 Primary Sodium to Gross Sample Data

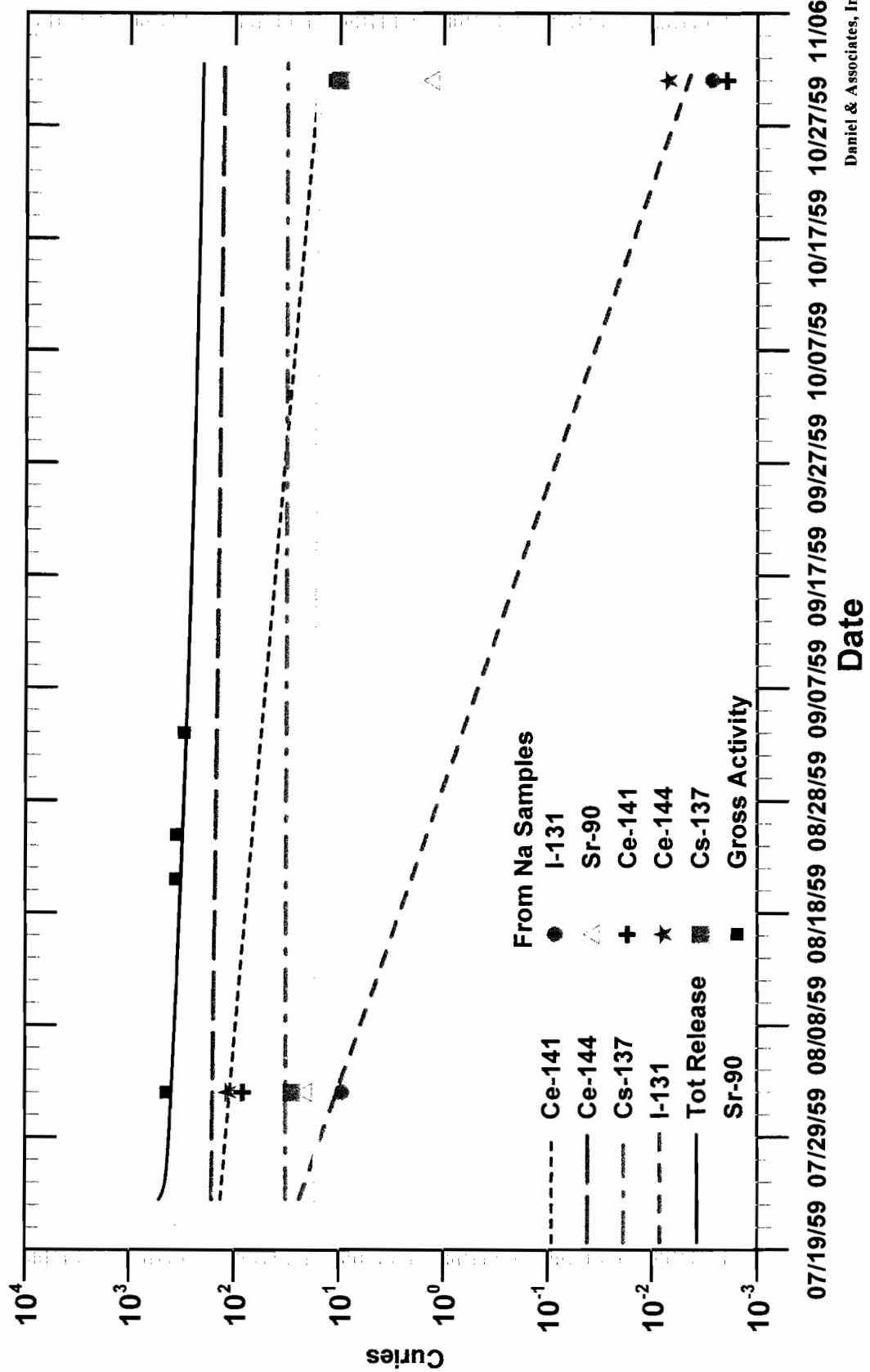


Figure 5.7 Isotopic Primary Sodium Compared to Sample Data

5 - 21

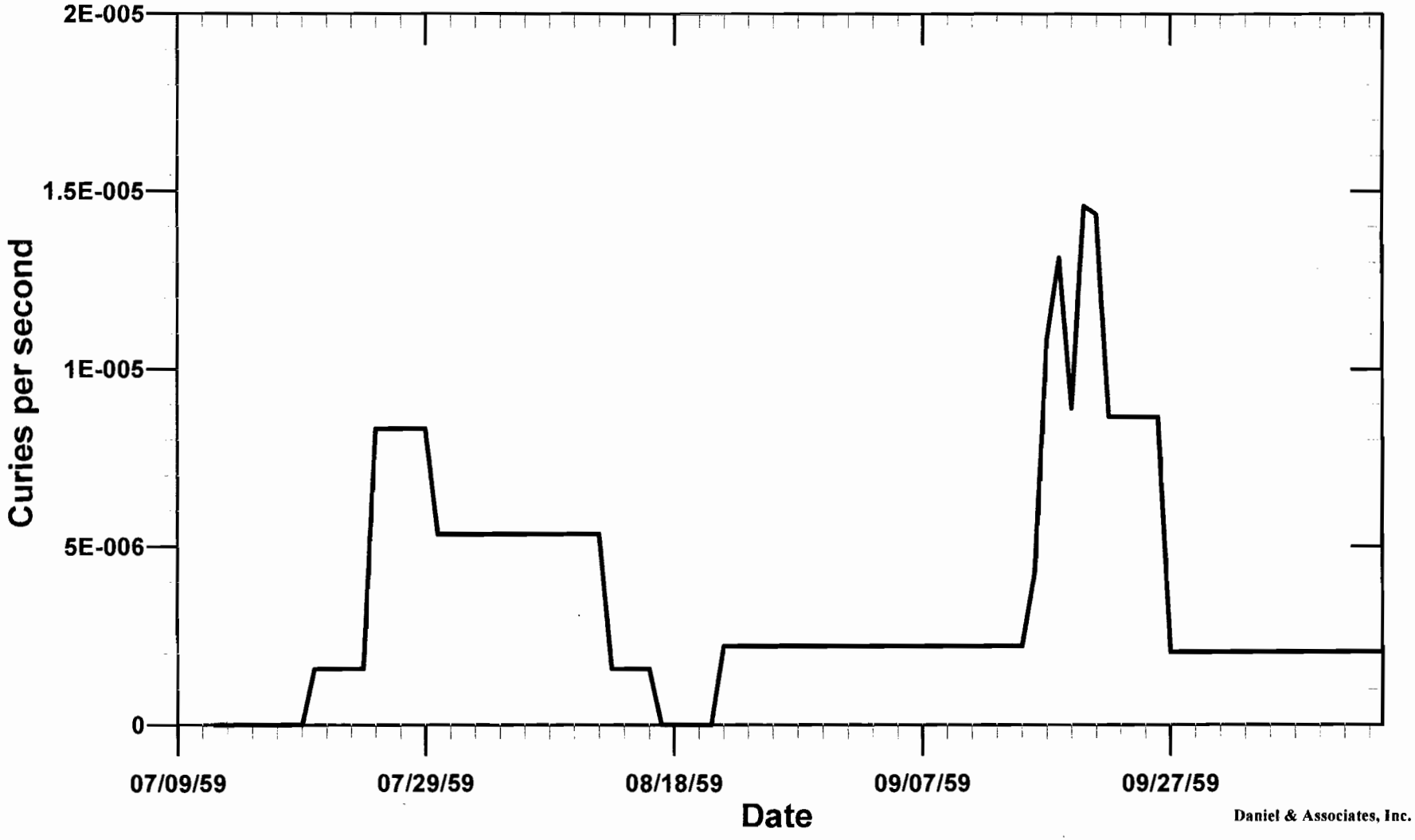


Figure 5.8 Waste Gas Release from Holdup Tanks

5 - 22

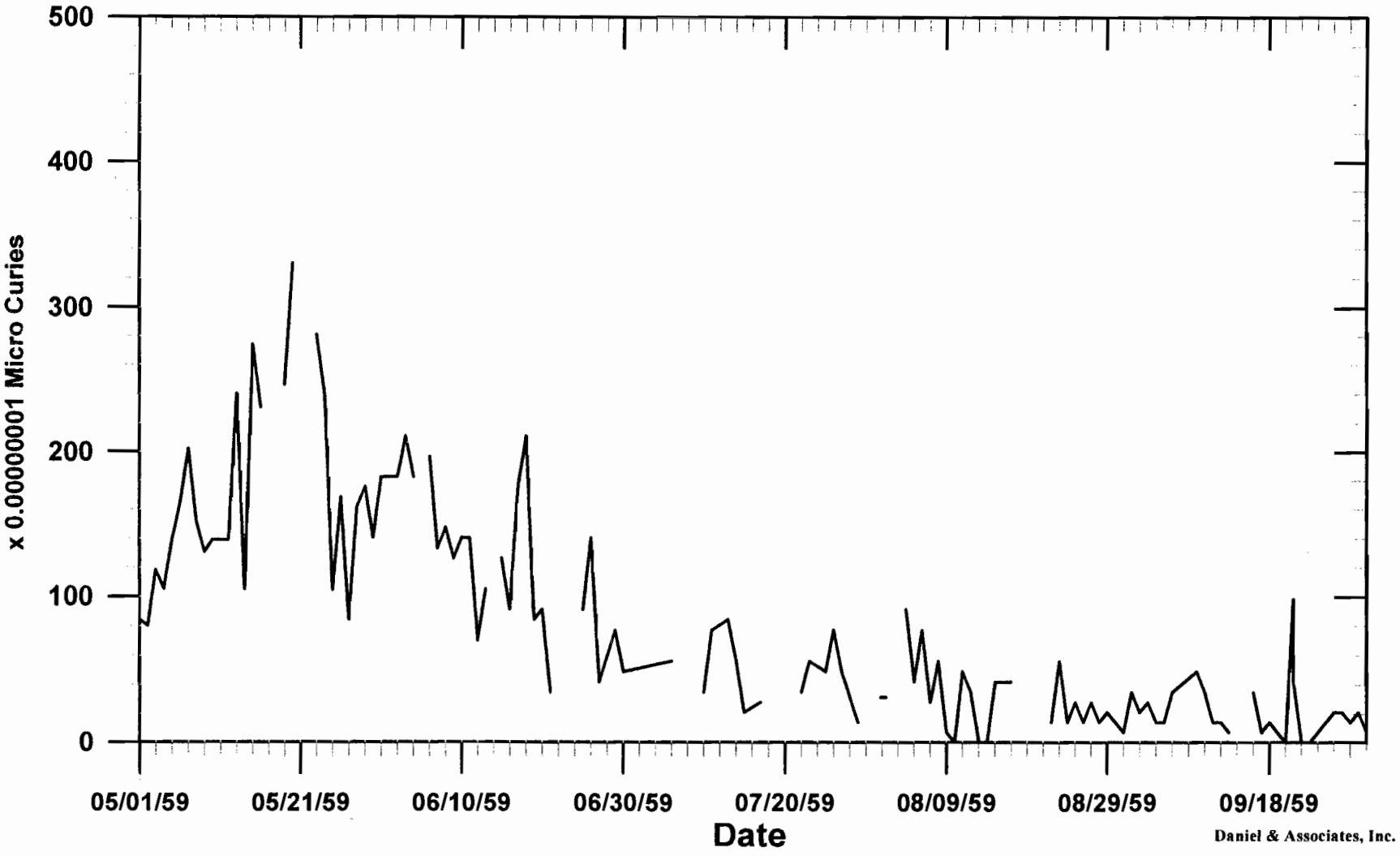


Figure 5.9 Air Sample ($\beta - \gamma$) Activity Santa Susana Station

5 - 23

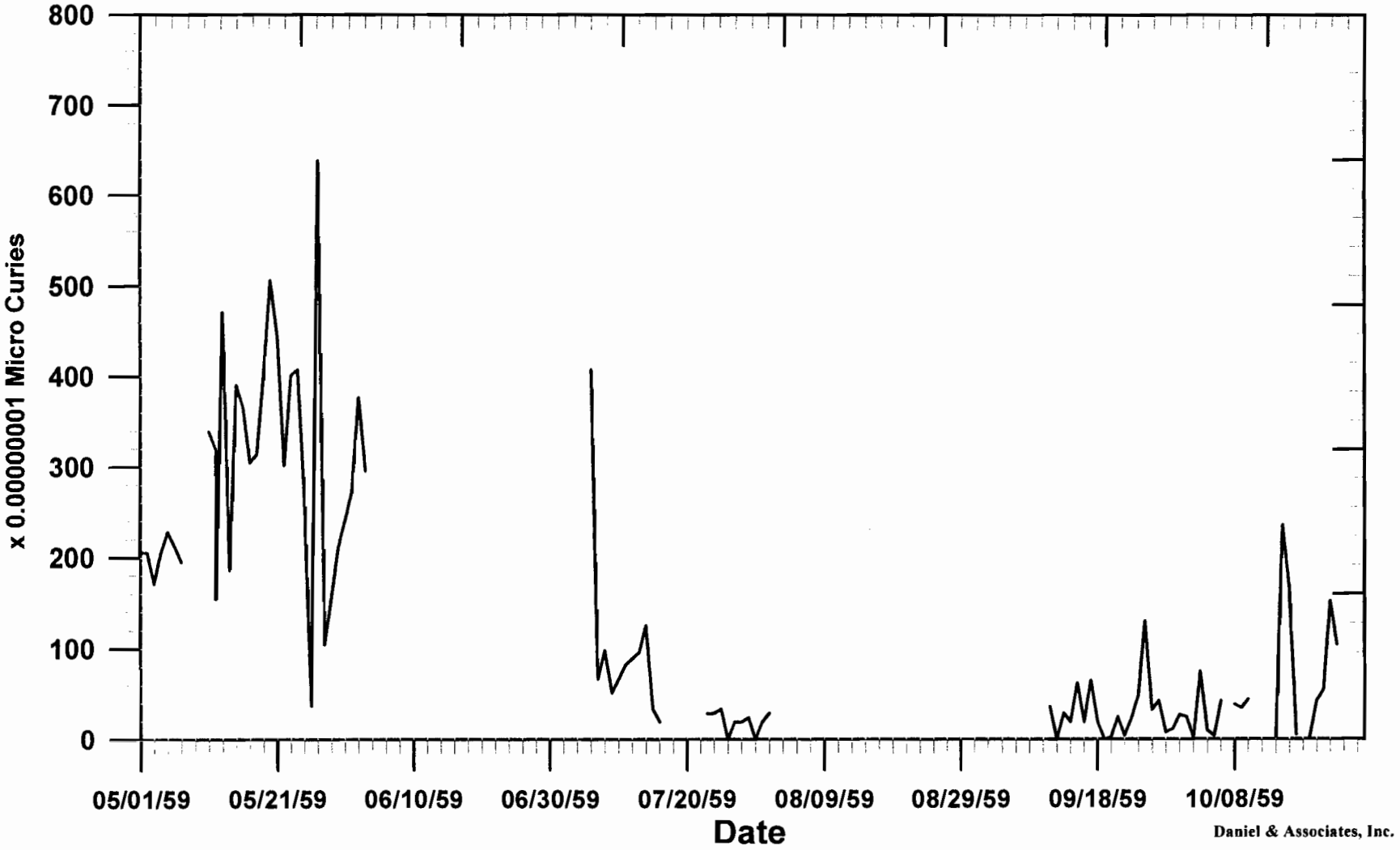


Figure 5.10 Air Sample ($\beta - \gamma$) Activity Van Owen Facility

5 - 24

6. Response to Plaintiff's Allegations

This section address some of the major points of plaintiff's expert contention that 1330 curies of I^{131} were most likely released to the environment during July 12 -15, 1959. (I reserve the right to testify about any of the opinions offered by plaintiff's experts during the trial in this matter.) The details of this hypothetical release are described below, with comments as to the validity of the hypothesis.

6.1. Pathways for Release

6.1.1. Plaintiff's Pathway for Release

Plaintiff's expert, Dr. Arjun Makhijani, maintains that his "best estimate" release, 1330 curies of I-131, was released from the SRE during the period July 12-15. The report prepared by Dr. Makhijani [Ref. 32] did not identify the pathway for release, other than saying

> "Thus it can be concluded that the decay tanks did not hold-up the bulk of the activity flushed from the reactor following the accident, and that **it was therefore most likely to have bypassed the tanks and vented directly out the stack.**" [emphasis added]

When questioned about the release pathway during his deposition, Dr. Makhijani identified the release pathway from the core cavity annulus through a relief valve and through vent header 492 to the suction tank, through the holdup tanks and then out the stack. The diagram used to explain his pathway is shown in Figure 6.1.

The pathway identified by Dr. Makhijani as being the pathway for release did not exist. Release to the environment of radioactive material requires two necessary conditions – (1) radioactive material, and (2) mass flow rate. Neither of these conditions was met by the pathway defined by Dr. Makhijani.

A description of vent header 492 is given in Section 2 of this report. The relief valve identified as "18" on the diagram that Dr. Makhijani traced for his pathway is identified in the legend as "Core Cavity RV-He" (Relief Valve- Helium). This valve is shown on the piping and instrumentation diagram (P&ID), Figure A-IV-3 sheet 2 of 3, Appendix B, as the relief valve around valve 459A, which relieves the annulus volume between the

outer tank and the core tank (the space between the bellows). Valve 459A is normally closed unless there is some reason to vent the core cavity annulus volume, and the relief valve is provided to prevent overpressurization of the core cavity annulus, such as initial filling of the volume with helium.

The description of the core tank and outer tank are also discussed in Section 2. The annulus volume between the outer tank and core tank is sealed by two welded bellows at the cavity liner. Thus, the core cavity annulus is not connected to the cover gas volume at all. The annulus volume does not accumulate fission products, as does the cover gas volume directly over the core.

The helium pressure in the core cavity annulus is 3 psig, and a blocking valve in the vent line (V-459A) is normally closed. A relief valve (RV600, aka #18) relieves at 6 psig. This relieves only upon failure of the helium pressure control station 473 relief valve, (RV402) which is set to relieve at 4.5 psig. Both relief valves vent to header 492 when actuated. [Ref. 11, pp 115]

During Run 14, there is no record of the core cover gas or of any helium cover gas system pressure ever being greater than 3 psig. There is no reason to suspect that the relief valves ever opened, nor any description of an operational event that might have cause the annulus volume to become overpressurized. The relief valves associated with the core cover gas, which is considered a "probably contaminated" radioactive system, vents to header 497.

The pathway is described by plaintiff's witness as passing through vent header 492 to the suction tank. There is a vapor trap between the relief valve and the suction tank, which would serve to remove, or at least reduce, any radioiodine in the gas stream. The compressor suction tank is normally kept at a negative pressure relative to the vent header pressure. The suction tank operating range is between 13.2 psia and 10.7 psia:

> "Each compressor is controlled from a high and low pressure switch on the suction tank. The pressure switches for compressor A will be set to start it when the pressure in the suction tank is 13.2 psia [-1.5 psig] or higher, and to stop it when the pressure drops to 10.7 psia (-4 psig). The switches for compressor B will be set to start it when the pressure in the

suction tank reaches 13.7 psia (-1 psig) with compressor A running and to stop it when the pressure drops to 11.2 psia (-3.5 psig) with compressor A running." .[Ref. 11 pp 105]

The compressors withdraw gas from the suction tank and force the gas into one of four holdup tanks. If the holdup tank were open to the stack, as hypothesized by Dr. Makhijani, the compressor suction tank could not be maintained at negative pressure. Furthermore, the compressors would be running at full speed and would not shut off. This would violate their procedure for startup, and would alert the operators of a problem. The suction tank pressure is indicated by PI 601. If the pressure exceeds the prescribed setting, an alarm is actuated in the reactor control room as BHPA-602. [Ref. 11 pp 105.]

There are measurements of holdup tank activity taken on July 9, 10, and 11, prior to their venting to atmosphere (Appendix D). These measurements indicate that the valve alignment was to the suction tank as required, and not to the stack, assuming it were even possible to vent directly to the stack from the suction tank. There is no reason to believe that this alignment was ever changed, since accumulation of radioactive gas occurred as indicated by samples of the holdup tanks on July 15, and venting of holdup tank on July 20^{th} .

Figure 2.13 shows the vent gas system and headers, including the pathway identified by Dr. Makhijani. As shown in Figure 2.13, solenoid valve 604 is the solenoid valve that is closed by the stack monitor in the event that the activity of the release exceeds the stack monitor setpoint. Thus, if the stack monitor reached the setpoint, such as the event of July 12 when the stack monitor increased, it would close solenoid valve 604. The stack monitor showed an increase in activity again on July 15. The activity that was seen by the stack monitor on July 15^{th}, could not have come from the holdup tanks via line 495, since the solenoid valve should have closed on July 12.

The pathway described by Dr. Makhijani did not exist.

6.1.2. Pathway – Fuel to Cover Gas via Moderator Can Vent Tube

Dr. Makhijani maintains that a pathway for radioiodine release from the fuel to the cover gas was via the moderator can vent. In his report,[Ref. 32 pp. 78] he states:

> "In addition, the damage to the moderator can surrounding core channel 10 during the time of fuel damage provides a direct pathway for the iodine released from this element to reach the cover gas without interacting with the sodium coolant **by flowing through the venting line of this moderator element.** These mechanisms provide plausible means by which large quantities of iodine could have been released from the fuel while only a fairly small amount could have been found in the primary sodium." [Emphasis added]

The moderator tube vent line is installed in the moderator can to relieve pressure that may build up in the can during irradiation. This design feature is discussed in Section 2 of this report. The vent tube is not porous. The vent tube has an opening only at the bottom of the tube. The vent tube is not a pathway for radioiodine to transport to the cover gas.

Section 4.3 discussed the attempts at dislodging some of the material believed to be interfering with heat transfer by jiggling the assemblies. When this was done on July 24, it was noticed that the element in core channel 10 was stuck in place, and had been free on the evening of July 22. Had damage to core channel 10 occurred earlier, of the nature described, it would not have been free to move on July 22. This reinforces the conclusion that core damage occurred between July 22 and July 24.

The moderator can pathway did not exist.

6.2. Mass Transfer Calculations

Plaintiff's expert, Dr. Makhijani, was asked in his deposition if he had performed any flow rate calculations for his release. His answer was no.

A simple calculation whereby the two hypothesized "major periods" of release, i.e., the cover gas ventings on July 12th and 15th would imply a release rate of 1330 curies I^{131} over a period of 10 hours, or approximately 133 curies/hr. The specific activity of the cover gas would be 32 μCi/cc for I^{131} assuming that the radioiodine was uniformly mixed in the cover gas. The volume of gas required to reduce the cover gas pressure from 2 psig to 1 psig is 6% of the mass of helium present at 2 psig. An equivalent mass of iodine would be released along with the helium, (6%) or approximately 80 curies. Thus,

94% of his inventory would be left in the cover gas, at a concentration of 30.7 μCi/cc.

Using the same approach, the ventings of July 15th would reduce the inventory by 7.3% for pressure reduction of 1.8 to 0.6 psig, and 11% for reduction of 3 to 1 psig. Thus, a total reduction of 18.3% of the remaining inventory would amount to only 228 Curies.

6.3. Damaged Fuel Inventory

Dr. Makhijani's inventory in the damaged fuel for I^{131} was reported to be 3900 Curies. If we assume that he considers the entire inventory to be released, then the 3900 curies is the maximum available for release to the environment. However, he clearly states that his upper bound for release to the environment is 2540 curies, [Ref. 32 pp 45] and his "best estimate" is 1330 curies.

By the logic presented in Section 6.2 above, only 24% of the cover gas inventory could have been released via his hypothesis. This implies that the cover gas would have to contain 5542 curies on July 12, when his hypothetical "intermittent" release began. This amount is greater than what he maintains was available for release, by approximately a factor of 1 1/2, and is refuted by the plant data as well.

6.4. July 12 Release

The cover gas venting of July 12 was described by Dr. Makhijani as:

> "Shortly after this purging of the cover gas, at 5:00 pm "a sharp increase in the stack activity to 1.5 x 10^{-4} μc/cm³ was noted ." As will be discussed in section 5 .D, the numerical value reported for the stack activity is not likely to be correct, but the timing of the increase in close correlation with the operators purging the reactor cover gas is significant."

The cover gas was purged at 3:30 PM. The stack monitor increased at 5:00 PM,- one and one-half hours after venting the cover gas. Dr. Makhijani postulates that the two events are related, namely that the venting of cover gas caused the stack monitor to increase. The velocity in the 2-inch diameter vent pipe would be approximately 0.5 ft/sec. The length of pipe from the reactor cover gas to the stack via his pathway is approximately 340 feet. At a velocity of 0.5 ft/sec, the gas would reach the stack in 11 seconds, essentially simultaneous with opening the valve to reduce pressure.

For the pressure venting to be related to the stack monitor, the length of pipe in the vent system would have to be 2700 feet in length. Dr. Makhijani's hypothesis does not correlate with the physical design of the plant.

6.5. Time of Fuel Damage

Dr. Makhijani makes an argument that the fuel was damaged during the "reactivity excursion" of July 13th. He ignores the conclusion and experiments that were performed by the AI investigators as to the reasons for fuel damage, (the cycling across the $\alpha - \beta$ transition temperature until cladding failure) during the time period July 22-24. Specifically, the following items provide indication that fuel failure was in the time period July 22-24 time frame:

- Data of temperature oscillations on a test specimen of fuel after Run 14 indicated cladding failure occurred after temperature oscillation through $\alpha - \beta$ transition temperature, which were experienced by the core during the time frame July 22-24.
- Gross measurement of holdup tank samples taken on July 20 do not indicate gross fuel failure.
- The ratio of Xe/Kr on July 13 in the fuel is less that that on July 23, and measurement data from cover gas is closer to the ratio of July 23 than that of July 13.
- The failure of the moderator can in core channel 10 occurred after July 22, as demonstrated by the fact that it was freely moving when it was "jiggled" on July 22.

The original AI investigators identified conditions that would cause major fuel damage during temperature oscillations between July 22 – 24th, as discussed in Section 4 of this report. Dr. Makhijani thus postulates a release of greater magnitude than the data support, and proposes that it occurred at a totally different time than records suggest.

The feasibility of a July 12 fuel damage release was investigated. The holdup tank volumes sampled and released on July 20 contained activity when decay corrected to July 13th amounted to 5.7 curies Xe^{133} and 2 curies Kr^{85}, according to gross activity measurements. The holdup tank was filled between July 3rd and July 20th , as indicated by the fact that 3 holdup tanks prior to July 20 were released on July 9,10, and 11. One holdup tank has to be aligned at all times to the suction tank according to operating procedures. While it cannot be ruled out that no damage occurred during the 12th-15th time frame, the activity in the holdup tank on July 20 is not of a magnitude to imply gross fuel damage.

Section 6.1 discussed the observations of core channel 10 moderator can.

6.6. Comparison to TMI Release

Dr. Makhijani maintains that:

> "Our best estimate for the amount of iodine-131 released during the July 1959 SRE accident is 80 to 100 times larger than the official estimate for the release of radioiodine to the environment from the 1979 partial core meltdown at Three Mile Island. The full range of our estimates are approximately 20 to 200 times the official estimate for the TMI iodine-131 release ."[Ref. 32 pp 45]

Since the TMI-2 power reactor was considerably larger than the SRE, (2772 MW vs. 20 MW) it is relevant to compare fission product quantities, fuel damage quantities, and fission product release fractions before making statements such as above. A comparison of the TMI fuel inventory vs. SRE and other relevant data are presented in Table 6.1 below:

Table 6.1 Comparison of SRE to TMI-2

Item	TMI-2	SRE*
Fuel Loading (Tons U)	94	3
Fuel Type	UO_2	U

Per cent Damaged Fuel	100	30
Containment	Reinforced concrete	None
I-131 Inventory (grams)	547	0.15
I-131 Inventory (10^6Curies)	65.4	0.0192
Environmental (Ci)	8-14	1330*
Release Fraction (Airborne)	0.00003	0.07*

*According to Dr. Makhijani

According to Dr. Makhijani, the SRE environmental release was 100 times the TMI release, even though the SRE contains approximately 3 % of the fuel that TMI-2 contained, and there was no molten SRE fuel. The inventory of I^{131} in the SRE was 0.03% that of TMI. All parameters that contribute to release from fuel are higher in the TMI column than SRE. All experimental evidence is contrary to Dr. Makhijani's figures. The comparison of "apples to apples" can only be done by comparing release fractions.

Release fractions for airborne I^{131} at TMI ranged from 0.00007 to 0.00003, depending on the analysis selected. The I^{129} figure is the most reliable, since there is no correction for half-life. Table 6.1 contains the release fraction derived from I^{129} samples.

The release fraction for I^{131} for TMI was reported in a letter from Dr. Dave Campbell, ORNL, to Dr. Andrew Hull, BNL, in 1980, using data provided by the author.[Ref. 33]

Dr. Makhijani's "estimated" release fraction for the SRE is 2,000 times higher than the reported release fraction for TMI. He based his estimate on a single sample of primary sodium that he maintains, indicated a greater release of iodine. The knowledge of iodine behavior has certainly expanded since 1959, when the AI investigators commented that they expected iodine to be more volatile. He disregards the gross measurements in the primary sodium and cover gas, since they do not report specific isotopes. He disregards radiation monitor responses. In short, there is no evidence whatever to support his release, and disregards evidence that is contrary to his conclusions.

6.7. Makhijani's Release From Fuel

Table 6.2 contains a summary of Makhijani's damaged fuel inventory. The column "Fuel Release Fraction" is the fraction of the damaged fuel inventory assumed to be released. The "Environmental Release Fraction" column is the fraction of damaged fuel destined to be released to the environment.

Isotope	Inventory (Ci)	Fuel Release Fraction	Environment Release Fraction
I-131	3900	0.45	0.34
Cs-137	2000		
Xe-133	5700	0.45	0.45
Kr-85	270	0.45	0.45
Sr-90	2100		

The xenon environmental release fraction was given as 0.45, exactly the same as the release from fuel. This implies that 5700 curies of Xe^{133} was released from the fuel, and 5700 curies Xe^{133} escaped to the environment. That leaves a total of zero curies Xe^{133} in the plant, following the incident. The holdup tank activities, when decay corrected back to the time of release, amount to 1266 curies, and would be even more if decay corrected to July 13.

If Makhijani's release from fuel and to the environment (via a nonexistent pathway) left zero curies in the plant, there is a discrepancy of 1266 curies that must be resolved. The evidence from the plant must take precedence over a theory that violates the physical laws of nature.

6.8. Diffusion of Fission Products

Dr. Makhijani states in his report [Ref. 32 pp 78] that:

> ".....the fuel in the SRE had been maintained at a high temperature for approximately two weeks prior to run 14 aiding in the diffusion of fission

Actually, the reactor was shut down for a period of approximately 39 days prior to Run 14, and could not have been "maintained at a high temperature for approximately two weeks" as Makhijani claims.

The diffusion of fission products was investigated as a possible means of release. The result of this investigation determined that diffusion from metallic fuels was small in comparison to the eutectic release. Diffusion is a means for release from oxide fuels, but is on the order of 10^{-4} percent of inventory in metallic fuels for a temperature of 800 °F held for a 24 hour period. Diffusion may increase by an order of magnitude for temperatures between 900 and 1400 °F, [Ref. 35] which is still considered insignificant.

6.9. Environmental Sample Data

A discussion on environmental samples taken during July-August is found is Section 5.5. There was no indication of a release of the magnitude that Makhijani claims. Air sample data showed no activity above normal background for the sites monitored. [See Appendix F]

The results of analysis for I^{129} were reviewed from data collected at sites adjacent to the Santa Susana Field Laboratory.[Ref. 48,49,50,51,52,53,54] This sampling program was conducted to determine if chemicals or radionuclides had migrated or had been deposited on two properties located north/northwest of the Santa Susana Field Laboratory known as the Brandeis-Bardin Institute and the Santa Monica Mountains Conservancy. A total of 118 soil samples were analyzed for I^{129} that were collected from the study area. None of the samples analyzed had I^{129} activities above the lower limit of detection. The lower limits of detection varied between 0.03 and 3.3 picoCuries/gram. It is concluded, therefore, that there is no environmental evidence of a release of radioiodine from the SRE of the magnitude claimed by Dr. Makhijani.

6.10. Intermittent vs. Continuous Release

After completion of his report, Dr. Makhijani modified his release "duration" to have occurred assuming

> "...a continuous release between 5:00 p.m. between July 12, when the reactor cover gas venting began, and about noon July 15, 1959, following

> the completion of the second cover gas venting. The actual releases were likely episodic throughout this period, but the number and size of the individual events cannot now be scientifically reliable way."

There is no basis in any of the AI reports for an "episodic" release. In his report, Dr. Makhijani seems to imply that July 13 was the "major" period of fuel failure. He thus contradicts his hypothesis in his amended statement.

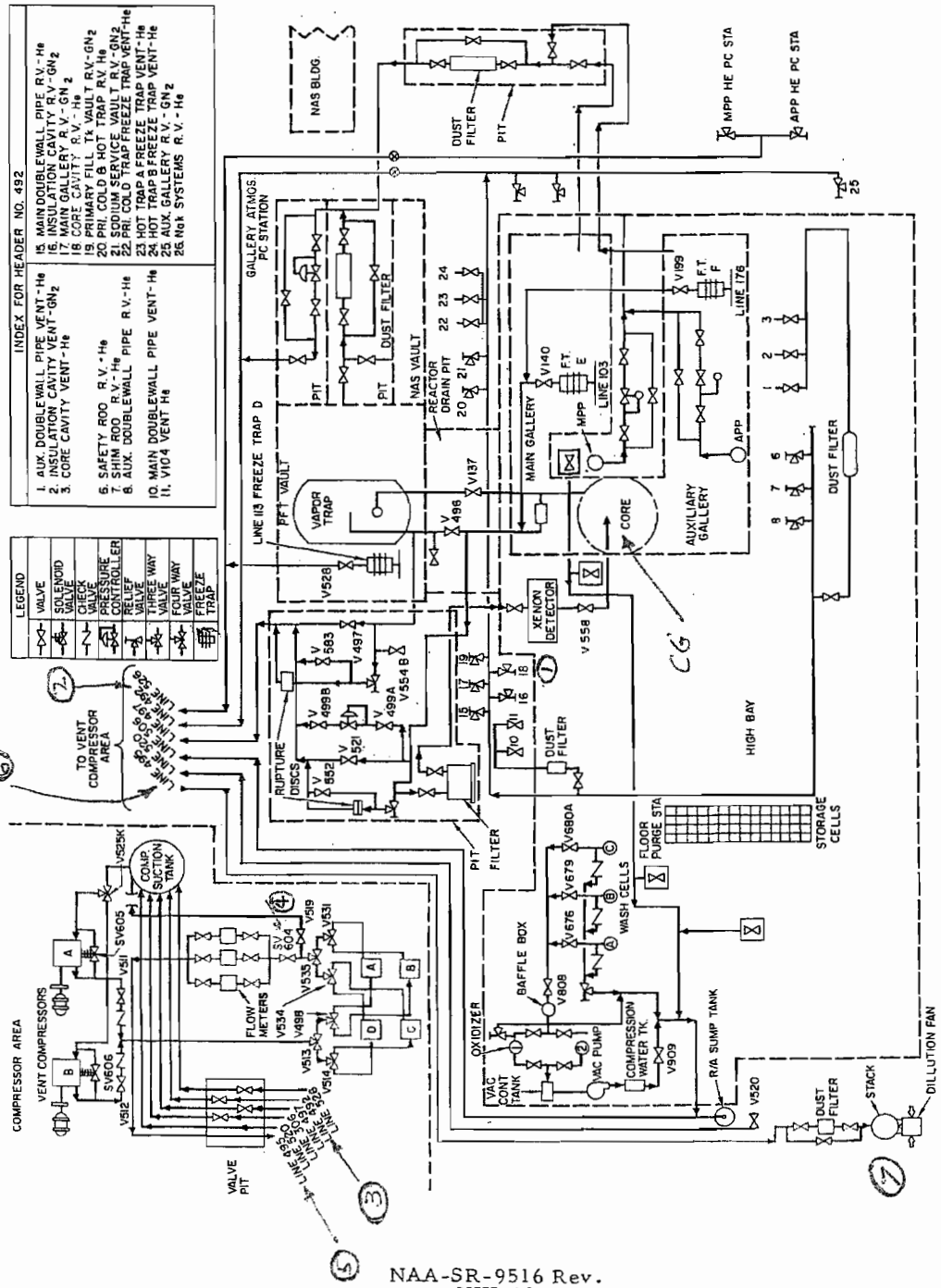


Figure 6.1 Hypothetical Pathway for Iodine Release

Figure39. Radioactive Vent System

6 - 12

The analysis in this report documents the results of my study of whether or not releases of I-131 occurred from the stack during or as a result of the July, 1959 incident. The results indicate conclusively that iodine isotopes were retained within the plant. I reserve the right to testify about any of the opinions offered by plaintiff's experts during the trial in this matter.

5/27/05

Date

John A. Daniel Sr

Signature, John A. Daniel, Sr.

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Appendix B

SRE Piping & Instrumentation Diagrams

Radioactive Waste Systems

BNA00457851

Pages II-C-2 through II-C-4

Pages II-E-1, II-E-3

Page III-A-1

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Pages VII-1 through VII-4

Pages XV-22 and XV-23

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Pages 1 through 3

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V-3 through V-5, V-18, V-20, V-27

BNA03967732

Pages 67, 88, 96, 99, 177, 178, 186-193, 203, 221

BNA03968609

Pages 102-115

BNA07879504

Pages 362-366, 368-371

h) Open fill valves V-156 and reactor sodium inlet valve V-103.

i) Watch the line temperature readouts on LL4, LL5, and LL7 for indications that sodium has passed into the suction side of the main primary pump. When these temperatures level out at sodium temperatures, sodium has passed into the line. Establish pump case freeze seal by opening the vent pressure control bypass valve (V-521C) and increase the pump speed to 500 rpm. After 5 min, close this valve and pressurize the pump case to 0.5 psig.

j) Start auxiliary primary pump and run at minimum speed (450 rpm).

k) When reactor sodium level reaches 130 in. from the loading face as indicated by the level coil, open the following sodium valves: V-101, V-175, and V-177.

l) Monitor auxiliary primary piping temperatures read out on LL114, LL115, LL116, LL117, and LL119. When these temperatures level out at sodium temperatures, the auxiliary primary pump is primed. Vent pump case to establish freeze seal by opening the vent pressure control bypass valve (V-521-C). Then increase the auxiliary primary pump speed to 500 rpm. After 5 min, close this valve and pressurize the pump case to 0.5 psig.

m) When reactor sodium level reaches 120 in. from the loading face as indicated by the level coil, close sodium fill valve V-156. The main and auxiliary primary systems are now filled.

n) If gas voids or other causes prevent establishing flow in fill line 156, a delta P can be established between the primary fill tank and the reactor to provide an initial driving force to get flow started. Maximum allowable pressure on primary fill tank is 15 psig. This can be accomplished by doing the following prior to step g).

    1) Open vent valve (V-497) and vent reactor and primary fill tank to 0.5 psig.

    2) Close vent valves 137 and 496 to isolate primary fill tank from reactor.

3) Pressurize primary fill tank to 4.0 psig by opening pressure control valve V-401. This valve is controlled from panel in the control room.

4) Continue with steps g), h), and i). NOTE: The primary fill tank pressure will drop due to sodium being displaced from the primary fill tank into the primary lines. This pressure should be monitored closely and maintained at 3.5 ±0.5 psig.

5) When step 9 is completed, open vent valves 137 and 496 to establish a common atmosphere between the primary fill tank and the reactor.

If for any reason, line 156 cannot be used for filling the primary systems, the following alternate procedure may be followed:

a) Check to see that all preheat thermocouples on main primary and auxiliary primary systems and lines 112 and 113 are indicating between 300 and 500°F and that the reactor temperature is at 300°F.

b) Check to see that the freeze trap on line 176 is open for gas removal and that the freeze trap temperature is less than 150°F. This is accomplished by opening helium PC station 446-V305A & B. If helium flow is established, the trap is open. If flow cannot be established, energize heater switch SS-1 and melt the sodium seal. Trap temperature is read out on SS-5.

c) Check to see that the freeze trap on line 103 is open for gas removal and that the freeze trap temperature is less than 150°F. This is accomplished by opening helium PC station 436-V312 A & B. If helium flow is established, the trap is open. If flow cannot be established, energize heater switch SS-12 and melt the sodium seal. This temperature is read out on DH-2

d) Check that there is coolant flow on the main and auxiliary primary pump case and shaft freeze seals.

e) Open the following sodium valves: V-101, V-103, V-113, V-175, V-177.

f) Close reactor drain and fill valve V-112.

g) Check to see that there is coolant flow on the following valve freeze seals: V-101, V-103, V-175, and V-177.

h) Open vent valves V-137 and V-496.

i) Place the panel NN flowmeter switch in the "reactor fill" position and the control room flowmeter in the "fill" position.

j) Start the main and auxiliary primary pumps at minimum speed. (100 and 450 rpm respectively).

k) Start the reactor fill-and-drain pump in the fill direction and set the voltage at 50.

l) Open V-112 and increase fill-pump voltage to 150. Observe flowmeter to see that sodium is flowing. If sodium flow is not established, stop the fill pump and close V-112. Then close V-137 and V-496. Establish a delta P between the reactor and the primary fill tank by pressurizing the fill tank to 4.0 psig and venting the reactor to 0.5 psig. Start the drain pump and open V-112, V-137, and V-496.

m) When the level coil indicates the reactor level is 160 in., open the main and auxiliary primary-pump-case vent valves and allow sodium to enter the pumps. Then open vent valve V-199 for line 176 freeze trap. Continue to vent from line 176 freeze trap until flow is established in both main and auxiliary primary loops.

n) Continue to fill reactor until the level coil indicates reactor level is 126 in. Stop the drain-and-fill pump and close V-112.

o) Observe all freeze-seal temperatures and adjust coolant flows to maintain temperatures below 150°F. Establish helium backup pressure on both the pumps and valves. Pressure is to be maintained at 0.5 psig.

2. Main Secondary-Sodium System

The following procedure is to be used in filling the main secondary system:

E. DRAINING

Draining of the sodium systems is accomplished by energizing an EM pump and pressurizing the system to be drained. Sodium flows through a drain line to one of two sodium drain-and-fill tanks. As sodium fills the tank, inert gas is vented to the radioactive vent system or to atmosphere depending on whether the sodium is primary or secondary sodium.

The objective of this procedure is to permit the sodium systems to be drained to the fill tanks in a manner which permits easy maintenance on the systems by preventing holdup of sodium. The Shift Supervisor must authorize proceeding with the following steps:

1. Primary-Sodium Systems

Condition No. 1 - The reactor sodium level is to be reduced to the top of the moderator cans.

To drain:

a) The core loading must be reduced to dry critical (15 elements).
NOTE: A thimble must be provided for the portable level probe. This will be inserted in the reactor when fuel is being removed.

b) Observe the preheat thermocouples on the drain-and-fill lines 112 and 113 and the drain line strainer lines to see that the temperatures are all above 300°F.

c) Check that primary fill-tank temperature is being 350 and 400°F.

d) Place panel NN flowmeter selector switch in the reactor drain position. Place the control-room flowmeter selector switch in the drain position.

e) Check open-vent-valves V-137 and V-496.

f) Connect the portable level probe to the amplifier and energize the amplifier.

NOTE: The probe is sensitive to long time at temperature and must not remain in the thimble when not in use.

g) Reduce main primary and auxiliary primary pumps to minimum speed (100 and 450 rpm respectively).

f) Place the panel NN flowmeter selector switch in the "reactor drain" position. Place the control room flowmeter switch in the drain position.

g) Check that the reactor temperature is within 60°F of the primary fill-tank temperature.

h) Start the reactor fill-and-drain pump in the drain direction and set the voltage at 50.

i) Open sodium drain-and-fill valve V-112 and increase the drain pump voltage to 150. Check the sodium flowmeter on panel NN to see that flow is established. Continue to drain sodium from the reactor until the drain pump loses suction; then reduce the pump voltage to zero and stop the pump. Close V-112.

   1) When the main primary and the auxiliary primary pumps lose suction, stop the pumps.

   2) If sodium flow in the drain line is not established, stop the drain pump and close valve 112. Vent primary fill-tank pressure to 0.5 psig by cracking open V-497. (Close 497 completely when 0.5 psig is attained.) Close valves 496 and 137 and pressurize the reactor to 4.0 psig with helium solenoid valve SV-400. This should provide enough moving force to start drain-line sodium flow. This step of the procedure should also be followed when the reactor is to be completely drained.

j) To completely drain all auxiliary primary piping,

   1) Close V-177.

   2) Admit helium to the system through auxiliary-pump discharge-sodium-line-176 freeze trap by opening helium PC-446, V-305A and B. Energize heater switch SS-1 to melt the sodium in the freeze trap. Purge until the helium pressure will immediately fall off when V-305A is closed.

A. PURGING

Purging of the sodium service system consists of displacing the atmosphere in the lines and vessels to the vent system with helium. The following procedure will be initiated upon approval of the Shift Supervisor.

1. Cold Traps

In order effectively to purge the cold trap and its inlet-outlet lines, gas is admitted at the cold-trap freeze trap and the main primary system at the reactor. This gas is vented through the flush and drain tanks to the primary fill tank and from there to the vent system. This involves isolating the cover gas atmosphere of the primary fill tank and reactor.

To purge the cold trap:

a) Isolate the primary fill tank and reactor by closing V-496 and V-137.

b) Check that the following sodium valves are open: V-101, V-103, V-616, V-609, V-610, V-618, and V-619.

c) Check that the following sodium valves are closed: V-634, V-635, V-617, and V-620.

d) Open bypass valve V-295C, valve V-295D, and valve V-296. Check that valve V-561 to the vent system is closed. This introduces helium to the cold trap.

e) Introduce helium to the primary system by energizing SV-400.

f) Vent system to decay tanks by opening V-497C.

g) Check oxygen content with Beckman Oxygen Analyzer at V-554B to verify that O_2 content is below 0.25%.

2. Hot Trap

In purging the hot traps it is possible to admit helium through the freeze trap on hot trap A and vent through the freeze trap on hot trap B, or vice versa.

a) Check that the following sodium valves are closed, V-636, V-637, V-616, and V-609.

V. HELIUM SYSTEM

Purpose of the helium system is to establish and maintain an inert gas atmosphere for all piping, vessels, and equipment containing sodium. Table V-I is a list of pressure control stations and the components they serve.

TABLE V-I

Station	Pressure Indicator	Normal Pressure (psig)	Service
PC-400	PI-401	3	Fuel-element-cask service, at cask service area; moderator-cask service.
PC-402	PI-403	3	Cleaning cells; new fuel storage cells; service connection in fuel-storage-cell area
PC-410	PI-410	0-1	Main primary block valves
PC-413	PI-413	5-15	Main primary double-wall pipes
PC-417	PI-414 BPIM 465	3	Reactor atmosphere
PC-418	BPIM 465	0.5	Reactor atmosphere
PC-419	PI-420	8-12	Reactor shim rods
PC-421	PI-422	8-12	Reactor safety rods
PC-423	PI-424	3	Service Connection at reactor
PC-425	PI-426	0-1	Auxiliary primary block valves
PC-427	PI-428	5-15	Auxiliary primary double-wall pipe
PC-432	PI-429 BPIM 468	3	Primary fill-tank atmosphere
PC-433	BPIM 468	0.15	Primary fill-tank atmosphere
PC-434	PI-435	10	Primary cold trap; flush-and-drain-tank drain line; sodium line from primary cold trap; sodium flush line to primary cold trap
PC-436	PI-437	10	Main primary line at main intermediate heat exchanger
PC-442	PI-441	0-1	Main primary-sodium pump casing
PC-445	PI-444	0-1	Auxiliary primary-sodium pump casing
PC-446	PI-447	10	Auxiliary primary line at auxiliary intermediate heat exchanger

TABLE V-I (Continued)

Station	Pressure Indicator	Normal Pressure (psig)	Service
PC-448	PI-449	3	Sodium drums; sodium transfer line at melt station
PC-452	PI-453	3	Secondary-sodium fill-tank atmosphere
PC-454	PI-455	0-1	Main secondary-sodium pump casing
PC-457	PI-458	3	Main secondary expansion-tank atmosphere; main secondary cold trap
PC-459	PI-460	0-1	Auxiliary secondary-sodium pump casing
PC-462	PI-463	3	Auxiliary secondary expansion-tank atmosphere; auxiliary secondary plugging-meter piping
PC-469	PI-470	10	Reactor drain line
PC-473	PI-474	0-1	Core-tank cavity atmosphere
PC-475	PI-488	30	Sodium-service transfer tank
PC-489	PI-495		Secondary block valves (at Edison plant)
PC-490	PI-496	10	Steam generator

A. PURGING

Purging of the helium system may be accomplished at the same time the heat-transfer circuits and the sodium-service system are being purged. If purging of the sodium systems is accomplished in more than one step, the helium headers should be purged separately. Purging is accomplished in general by admitting helium at high points in the system and venting from the low points.

To purge entire helium system,

1) Open all stations in one of the two main branch headers consecutively from the start of the header.

2) Open the last station in the header not being purged.

3) Adjust flow to approximately 1.5 scfm using the PC upstream block valve.

4) If the individual station has more than one service line downstream of the PC, opening of the various spurs should be alternated.

3) PC station bypass valves are to be maintained closed at all times. Exceptions are during the purging operation, a period of maintenance on the PC valve or assembly, or when checking relief valve operability.

4) Changes in the following stations are required under the following operating conditions:

a) PC-417 and PC-418. (Reactor Atmosphere) PC-418 is set for 1/2 psig, and is to be used normally. PC-417 is set for 3 psig and is to be used only when the reactor atmosphere pressure is to be increased above 0.5 psig.

Should the pressure alarm sound (5 psig), an immediate investigation shall be initiated to ascertain the trouble. Check the helium-supply system for operability and the vent relief system. Close the helium-supply blocking valves (V-292A) (V-292AI) (V-268A) (V-268AI).

PC-418 should be always kept in service to guard against a negative pressure being established in the reactor.

b) PC-432 and PC-433. (Primary Fill Tank Atmosphere) PC-433 is set for 1/2 psig and is to be used normally. PC-432 is set for 3 psig and is to be used only when the primary fill-tank atmosphere is to be increased above 0.5 psig.

Should the pressure alarm sound (5 psig), an immediate investigation shall be initiated to determine the trouble. Check the helium supply system and the vent relief system for operability. Close the helium supply blocking valves V-268A, V-268AI, V-292A, V-292AI.

PC-433 shall always be kept in service to guard against a negative pressure being established in the primary fill tank.

c) PC-452. (Secondary Fill Tank) When filling the secondary fill tank with sodium the helium supply shall be cut off by closing the block valve downstream from PC-452 (V-298B) and the normally closed block valve in the vent line (V-299) is manually

VII. VENT SYSTEM

Operation of the vent system consists of routing all radioactive gases to the decay tanks by means of compressors. When sampling indicates that the activity of specific vent gases is sufficiently low, the gases may be routed directly to the stack by actuating solenoid valves in the several vent-system headers.

Gas samples taken from the decay tanks determine whether the gas can be vented to the stack and at what flowrate. Upon authorization of the Shift Supervisor, radioactive gases are then vented to the stack at the rate specified by Health Physics. The objective of the following procedures is to specify vent-system valve positions and to designate steps required to sample and dispose of radioactive gases.

A. NORMAL OPERATION

1. Radioactive Services

a. Reactor and Primary Fill-Tank Atmospheres

1) Bypass valves V-497 and V-551 shall be closed normally except during purging of the reactor and fill tank atmospheres. These valves may be opened to relieve excessive reactor and primary fill-tank pressures only with permission of the Shift Supervisor.

2) Freeze-trap vent valve V-523 will remain closed except during primary cold trap venting operation.

3) The sample valve V-497-B immediately upstream of the relief valve assembly is normally closed. Samples can be taken here to determine completeness of helium purges on the core and primary fill tank.

b. Helium Vent from Freeze Traps for Piping Adjacent to the Intermediate Heat Exchanger

1) These valves (V-140 and V-200) are normally closed.

2) They are only to be opened when lines are being filled with sodium, in accordance with Section II-B.

c. Cleaning Cells

The three fuel cleaning cell block valves (V-676-1, V-678-2, and V-676-3) shall be closed unless fuel washing operations are in progress.

d. Fuel-Handling-Machine Service Connections

1) Vent system block valves (V-504-G1 and V-504-G2) are to remain closed unless these service connections are in use.

2) Valve-to-atmosphere (V-518) from the coffin-service area shall be closed normally. Use of this valve requires approval of the Shift Supervisor.

e. Hot-Cell-Vent Service Connections

Vent-service header-block valve V-520 shall be closed unless header is in use. Use of this header requires prior approval of the Shift Supervisor.

2. Normally Nonradioactive Services

Gases from these services will be routed to the suction tank except under special conditions where a gas sample indicates levels low enough to be vented directly to the stack.

a. Operation of Diversion Station

1) Filter station bypass valve V-507B is normally closed. This valve may be opened in conjunction with valve V-507A for specific purging operations or during filter maintenance, with the approval of the Shift Supervisor.

2) Proper operation of radiation indicators will be checked monthly by Health Physics. Results of this check will be forwarded to the SRE Group Leader.

b. Main and Auxiliary Primary Pump Vents

1) The pressure controller bypass valves (V-521C-Main and V-522C-Auxiliary) shall be normally closed except during purging.

2) The pressure controllers shall be set to vent the pump casings at 15 psig.

3) Manual venting is accomplished after closing the helium inlet valve (V-319 main or V-327 auxiliary) by opening the pressure controller bypass valves (V-521-C and V-522-C) and observing the indicated pressure on the pump being vented.

c. Insulation Cavity Vent

1) Blocking valve (V-491A) shall be normally closed.

2) Sampling valve immediately upstream of blocking valve shall be normally closed.

d. Core Tank Cavity Vent

Blocking valve (V-459A) shall be normally closed.

e. Main and Auxiliary Primary Blocking Valves

1) Vent-system blocking valves (V-471A, V-473A, V-453A, V-559A, V-462A, and V-456) are normally closed. Always be sure that these valves are closed after they have been opened for a purging operation, to keep sodium out of the vent system.

2) Sample valves immediately upstream of the blocking valves shall be normally closed.

f. Main and Auxiliary Double-Wall Pipes Vent

1) Blocking valve (V-452A, V-469A, and V-467A) shall be closed normally.

2) Sample valves immediately upstream of blocking valves shall be closed normally.

B. SAMPLING AND DISPOSAL OF GAS

1. Sampling

Vent system gas samples are to be taken, using an evacuated sample chamber with adapter to connect to the various sample connections. Samples will be taken, once a week, from the following locations:

a) Core tank cavity (V-459D)

b) Insulation cavity (V-491D)

c) Auxiliary double-wall pipes (V-452D)

d) Main double-wall pipes (V-467 and V-469D)

e) Decay tanks

1) Decay tank samples will be taken as required to determine the gas decay rates. This information will be used to give a projected activity level and a date and rate for release.

2) A second sample will be taken just prior to release to obtain the exact release rate and activity level.

All gas samples will be taken by the Health Physics Unit, on request of the Reactor Operations Unit. See Section V-D for sampling procedure.

2. Disposal

As each decay tank is filled to 90 psig, an alarm will be sounded in the control room. The inlet valve to an alternate decay tank will then be opened, and the inlet to the decay tank with the high pressure will be closed.

Health Physics will then take a gas sample from the tank with the high pressure and will notify the Shift Supervisor of the recommended rate and date of release. The rate of release is controlled by V-537B, the decay-tank outlet valve. Gas flow is measured on a flowmeter located on the decay tank vault.

A solenoid valve (SV-604) is set to trip and stop the venting, if the stack monitor indicates a level higher than 5×10^{-7} μc/cc.

The display panel in the control room, showing which tank is being filled and which is being vented, shall always be kept up-to-date.

b) Vent the core tank cavity to the vent system by opening vent valve (V-459) and reduce the pressure to 0.2 psig. Close vent valve (V-459).

c) At 0.2 psig there should be no helium flow indicated at PC-473. If flow exists reset PC-473 for no flow at 0.2 psig.

d) By opening PC-473 bypass valve (V-270C) increase pressure to 4.5 psig. RV-402 should be set so that flow is just starting at 4.5 psig.

32) Primary-Fill-Tank Helium Pressure: >4 psig <0.5 psig

Corrective Action to Reduce the High Pressure

a) Notify Shift Supervisor.

b) Close the solenoid control valves for pressure control stations PC-432 and PC-417 by pressing the buttons on panel HH.

c) If the primary-fill-tank-pressure continues to rise, the primary fill tank relief valve will open at 5 psig. At 5.25 psig, the primary-fill-tank rupture disc will allow the pressure to be relieved to the vent system. As the pressure drops below 5 psig, the reactor relief valve will close to permit the rupture disc to be replaced.

d) Open the solenoid valve for helium-pressure control station PC-432 and vent the primary fill tank to 3 psig by opening the relief-valve bypass valve (V-497C).

e) With the primary fill tank pressure at 3 psig, observe the flow indicator on PC-432. If flow is indicated, the set point is above 3 psig and requires resetting.

f) If no flow is indicated, repeat steps c), and d) for PC-417.

g) If both pressure control stations and pressure gages are operating properly and the primary fill tank pressure again rises to 4 psig, a survey of the entire helium supply for the main-heat-transfer system is required. Check all branch lines for abnormal flow. Correct, if necessary.

Corrective Action to Increase the Low Pressure

a) Notify Shift Supervisor.

b) Check PC-417 and PC-432. Flow should be indicated for primary fill tank pressure lower than 0.5 psig. Check to be sure that the pressure gage is operating properly.

c) If pressure continues to drop, bring the pressure back up to 0.5 psig by opening PC-432 bypass valve V-292C.

d) If no flow is indicated at PC-417 and PC-432 for a pressure less than .05 psig, the controllers must be reset.

e) If PC-432 and PC-417 are operating properly, a leak is indicated. A survey of vent and helium corrections must be made to determine leakage point.

33) Reactor Helium Pressure: >4 psig <0.5 psig

Corrective Action to Reduce the High Pressure

a) Notify Shift Supervisor.

b) Close the solenoid control valves for pressure control stations PC-432 and PC-417 by pressing the buttons on panel HH.

c) If the reactor pressure continues to rise, the reactor relief valve will open at 5 psig, and the reactor rupture disc will allow the pressure to be relieved to the vent system. As the pressure drops below 5 psig, the reactor relief valve will close to permit the rupture disc to be replaced.

d) Open the solenoid valve for helium pressure control station PC-432 and vent the reactor core tank to 3 psig by opening the relief valve bypass valve V-551.

e) If no flow is indicated, repeat steps c) and d) for PC-432.

f) If both pressure control stations and gages are operating properly and the reactor pressure again rises to 4 psig, a survey of the entire helium supply for the main heat-transfer system is required. Check all branch lines for abnormal flow. Correct, if necessary.

Description of SRE Cover Gas System

Helium System

The purpose of the helium system is to establish and maintain an inert gas atmosphere for all piping, vessels, and equipment containing sodium.

The system consists of two supply manifolds of forty helium bottles each. Normal operating procedures shall have one manifold in standby or in the process of being reloaded with fresh bottles at all times. An alarm will sound in the control room when the pressure in the on-line manifold drops below 125 psig. Manual valving will then be required to put the other manifold on the line.

The gas passes from the manifold and is reduced from 2300 psig to 50 psig at pressure reducing station. A purification system consisting of NaK bubblers is then provided to remove oxygen from the gas before it is routed to the above services. Each service connection has its individual pressure control station which is made up of an adjustable pressure control valve, blocking valves, bypass valve, and pressure relief valves.

A pressure relief valve is located on the low pressure side of the pressure reducing valve assembly. The relief valve has sufficient capacity to remove the amount of gas that can be passed by the pressure reducer if it fails wide open, and at the same time not permit a pressure build-up in the line of more than 110% of the maximum working pressure of the vessel it is supplying. The exhaust from the relief valve on systems which may be contaminated by radioactive vapors is connected to the vent system. Relief valves in the secondary heat exchange systems and others where the gas will not normally become contaminated with radioactive vapor exhaust to the atmosphere.

Each control station has a pressure indicator and a flow meter. Where the relief valve exhausts to the vent system, the flow meter is located between the pressure control valve and the relief valve. In this manner the relief valve can be checked to see if it is open by noting the flow rate. Relief valves exhausting to the atmosphere are accessible and can be checked without the use of a flow meter.

Nitrogen System

The purpose of the nitrogen system is to establish and maintain an atmosphere of nitrogen gas in the following areas:

Main primary system gallery
Auxiliary primary system gallery
Primary fill tank vault
Disposable cold trap vault
Insulation cavity

The system consists of two supply manifolds of ten nitrogen bottles each. Normal operating procedure shall have one manifold in standby or in the

process of being reloaded with fresh bottles at all times. An alarm will sound in the control room when the pressure of the on-line manifold is low and manual valving in of the standby manifold is required. A connection is provided for a nitrogen trailer if large quantities of the gas are required for purging operations.

The gas passes from the manifold and is reduced from 2300 psig to 50 psig at a pressure reduction station. It is then fed to the above mentioned points from a low pressure surge tank. Individual pressure control stations further reduce the nitrogen pressure prior to entrance into the enclosure served. Each station has an adjustable pressure reducing valve, blocking valve, a bypass valve and a relief valve on the low pressure side.

A breathing section of the system ties the galleries and vaults to a single pressure control valve. In normal operation the spaces are in communication with each other. The insulation cavity is not included in the breathing section as it is connected to the vent system. Each space can be isolated from the breathing section by means of a plug valve in the branch connection to the space. There are two radioactive dust filters. One filter serves the galleries and the other filter serves the vaults. The filters have blocking valves and a bypass valve. A gas sample connection is located in the exhaust connection between the space or tank and the plug valve.

Normal flow through the system is small and is a result of leakage and breathing. The gas released during breathing is monitored by a radiation indicator and routed to gas storage or the atmosphere, as determined by its activation level.

A liquid nitrogen storage tank has been installed to replace the nitrogen cylinder manifold mentioned above. This installation has not been operationally accepted as of this date. It will not change the operation of the nitrogen system; it is intended to give a more adequate nitrogen supply.

Vent System

The vent system, by a system of headers, collects all process gases containing or possibly containing radioactivity. These gas sources are:

A. Normally Containing Radioactivity

1. Reactor and primary fill tank helium blanket atmospheres
2. Primary circuits and disposable cold trap vents
3. Cleaning cell vents
4. Fuel handling coffin service connections
5. Radioactive liquid waste accumulation and hold-up tanks
6. Hot cell radioactive gas service connections
7. Hot trap vents

B. Normally Non-radioactive

1. Primary circuit pumps (2)
2. Reactor drain line freeze trap vent
3. Helium supply system relief valves to the following services:

a. Main and auxiliary primary block valves (2)
b. Main and auxiliary primary double wall pipes (2)
c. Control rods
d. Safety rods
e. Disposable cold trap system
f. Main and auxiliary primary piping and equipment (2)
g. Core tank cavity

4. Nitrogen supply system relief valves to the following services:

   a. Main and auxiliary primary systems galleries (2)
   b. Primary fill tank vault
   c. Disposable cold trap vault
   d. Insulation cavity

5. Nitrogen system gallery atmosphere discharge
6. Insulation cavity vent (normally closed)
7. Core tank cavity vent (normally closed)
8. Main and auxiliary primary block valves (4) (normally closed)
9. Main and auxiliary double wall pipes (2) (normally closed)
10. New fuel storage cells

The gases from the services listed in section A above go directly to the compressor suction tank located north of the reactor building in a shielded pit. Gases from services B above are monitored for radioactivity in the main headers and, according to the activity, are passes to the compressor suction tank or bypassed through a filter to the atmosphere.

The compressor suction tank is maintained slightly below atmospheric pressure at all times, thus minimizing the intraflow of gas between service headers by assuring flow from higher pressure service outlets directly to the compressors. Two compressors in parallel draw from the suction tank; compress the gases to 114 psia and exhaust to one of four 350 cubic foot decay tanks. Here the gases may be sampled and exhausted upon sufficient decay. Exhaust rate is controlled by a pressure reducing valve and an associated hand operated globe valve. Exhaust gases pass through a CWS filter and are diluted by the hot cell ventilation exhaust stream. Should additional dilution be necessary, duct work and associated damper systems are installed to allow the use of a 25,000 cfm dilution fan.

heaters are installed on the outer periphery of the tank. These heaters are series-parallel connected and are controlled by a Wheelco temperature controller located in the sodium service building. Additional heat is available through the use of three immersion heaters located in thimbles in the tank. To provide for heat transfer, the thimbles are sealed at the top and filled with NaK. The immersion heaters are controlled manually from a breaker in the sodium service building. Thermocouples are welded to the tank in strategic locations and are read out on the Wheelco controller. Aluminum jacketed, super-X, block insulation is installed over the heaters and thermocouples. Sodium level and alarm coils are installed in thimbles in the tank to provide a means of measuring the tank sodium level.

Biological shielding in the form of a dense concrete vault is supplied for the radioactive primary sodium. The vault is situated north of the high bay and extends above ground about 6-ft. Elevation of the fill and drain tank is such that the reactor can be gravity - filled to the top of the moderator cans, but requires a pump for further filling or for draining. Thus, inadvertently opening the reactor drain valve cannot drain the sodium from the active core.

Fill and drain line 112 starts in the reactor at the bottom of the core tank and proceeds up through the reflector region on the north side of the core out through the core ear into the main gallery. The line then continues into the reactor drain pump pit (where V112 is located), and terminates at the electromagnetic (EM) drain pump. In the drain pump pit, a flowmeter is attached to line 112 on the reactor side of the EM pump. Line 113 starts at the EM pump and continues into the primary fill tank vault where valve 113 is located. Here the line is connected to the fill tank and extends to about 2-in. from the bottom. The EM pump can be stopped, started, and controlled either from panel K in the control room or panel N in the sodium service building. Line 112-113 and associated valves and pumps are normally frozen off after draining the reactor. It is for this reason that an additional fill line (155 and 156) is provided. This line starts on the south end of the fill tank 3/4-in. from the bottom and proceeds to an EM pump where line 155 starts and continues into the main gallery where it terminates at line 103. Valve 156 is located in the primary fill tank vault and the flowmeter is attached to line 155 in the drain pump pit. An EM

pump installed in line 155 facilitates draining the main system piping through these lines. Line 155 is connected to line 112 in the drain pump vault by line and valve 157. For normal conditions, valve 155 remains closed, and the valve actuator is covered and locked to prevent an accidental opening of the valve. Valve 156, however, is kept in an open position to provide for expansion of the sodium remaining in the line.

Through the use of the gas ballast line, 137, which connects the reactor cover gas region in the primary fill tank, the effective expansion volume of the reactor is greatly increased. This provides an improved safety factor in the event of an abnormal pressure excursion. In addition, conservation of the helium cover gas is assured during filling and draining operations as the gas is displaced by the sodium through line 137 and into the other vessel. So that a differential pressure may be established between the two vessels for special tests or for priming the drain pump, valve 137 is installed in the line. This valve is located in the reactor drain pump pit. Line 137 is equipped with heaters, thermocouples, and insulation in the event sodium vapor is carried over. Connected to line 137 through a sodium vapor trap is line 496, a carbon steel vent line whose primary purpose is to connect either vessel to the vent system. Valve 496 is installed in the line and serves the same purpose as valve 137. This valve is normally closed and individual vent valves and rupture discs are provided for each vessel; however, should the need arise, valve 496 may be opened and both vessels can be vented from one or the other vent stations. Also connected to line 137 is the condensate tank provided to collect condensed sodium vapor from the reactor pool and primary pumps. The tank is insulated and is provided with Wheelco controlled heaters, a sodium level probe (readout on panel GG), and a drain line. Drain line 133 connects to the reactor fill and drain line 113 at the EM pump, and is used to transfer sodium from the condensate tank to the primary fill tank as the need arises.

The primary sodium purification system is located in the sodium service vault and is connected to the main heat transfer loop by lines 616, 617 and 633 (Figures 3-21 and 3-25). Line 616 originates at the

header in which the leak occurs, but in checking the sections no big leak is found and when the main header is put back in service the leak rate is down due to pressure stabilization.

From the above, it can be seen that one of the first steps in checking the system for leaks is to be certain that the header pressure is not changing and that it is the same as the reference pressure.

B. NITROGEN SYSTEM

Areas such as the pipe galleries, insulation cavity, and vaults serve as secondary containment vessels for the primary sodium system. Inasmuch as these areas are usually closed to traffic, sealed, and have extremely high radiation fields, a sodium spill cannot be handled in the normal manner. To prevent a fire resulting from a sodium spill, the oxygen content of the atmosphere in the containment areas is maintained at less than one percent through the use of an inert gas. Because helium is more than 30 times as costly as nitrogen and the gas does not come into contact with the sodium in the circulating loops, nitrogen gas is used to establish and maintain the inert atmosphere in these areas.

1. System Description

Nitrogen can be supplied to the system from three sources: cylinder, trailer, or in the liquid state (Figure 5-3). Normally, liquid nitrogen is used because of the ease of handling and the cost.

Starting from the liquid nitrogen tank, the flow of liquid into the system is controlled by a General Controls pressure controller and an Annin valve. As the system pressure falls to the lower limit, the controller sends a signal to an air piston that opens the valve. Liquid then flows through the valve and into an evaporator section. This section consists of two finned U-tubes series connected. Here, the liquid absorbs heat from the surrounding air and evaporates into gas. The gas then passes through another pressure regulator and on to the gas heater. Between the evaporator and the gas pressure regulator is a connection from the top of the liquid tank. Any gas generated within the tank is relieved through this line. Also connected to the vent line is an atmospheric vent valve that is used

when filling the tank. Electrical heaters in the gas heater are used to evaporate any liquid that is carried over from the evaporator section. This can happen if the outside air temperature is low or if an excessive system use-rate occurs. From the gas heater, the nitrogen passes through the liquid nitrogen system stop valve, V-389, and on to the low pressure storage tank.

A standby source of supply consists of a high pressure rack and header to which 20 gas cylinders are attached, and a hose connection for a nitrogen trailer. Leaving the cylinders, the gas enters a high pressure header and passes on to one of two pressure regulators. Valves are provided so that either regulator can be used for each bank of ten cylinders. From the regulator, the gas passes through the "cylinder station" stop valve, V-351-B1 or V-351-B2, and on through the low pressure storage tank inlet valve, V-351, and into the tank. Between the regulator and the stop valve is the connection for the trailer. A connection for a purge supply to the wash cell area is also located here.

Because of its volume (35 standard cu. ft.), the low pressure storage tank tends to damp out any pressure surge caused by varying system use-rate. Between the LP storage tank and the main system stop valve, V-353, is the connection for the kerosene system surge tank cover gas. A service connection at the surge tank controls gas flow to maintain a constant three psig pressure within the tank.

In addition to the kerosene tank, six other service connections make up the nitrogen system: insulation cavity, main gallery, sodium service vault, primary fill tank vault, auxiliary gallery, and an inert atmosphere supply to the oxygen analyzer on the HB panel for zero checking of the instrument. Each service connection consists of an inlet valve, flow meter, pressure controller, outlet valve, bypass valve, relief valve, and pressure gauge. All valve handles and meters are painted green for system color coding. For ease of maintenance, pipe unions are installed around the pressure controller, and the bypass valve is arranged such that it can be throttled to provide service while the controller is removed.

A pressure of 20 psig is normally maintained within the low pressure header, and the individual pressure controllers are set

to provide sufficient nitrogen to maintain a slight positive pressure in the vaults and galleries and a pressure of three psig in the insulation cavity.

C. VAULT DEHUMIDIFICATION SYSTEM

As the vaults are not completely metal lined, moisture can enter these spaces through the concrete. A convection loop is then established with the water being evaporated by the heat, the water vapor rising and then being condensed by the vault cooling coils to start over again. Not only is the resulting rust problem not desirable, but should a sodium spill occur, the ensuing sodium-water reaction could cause equipment damage. For this reason the dehumidification system was installed. Basically, this system circulates the vault's humid atmosphere across external cooling coils where the moisture is removed. Dry nitrogen is then returned to the vault.

1. System Description

Presently, the sodium service vault, the drain pump vault, and the primary fill tank vaults are serviced by this system with each area having individual suction and return lines connected to common headers.

In each of the 4 lines (two suction and two return) is a Keystone throttling-butterfly valve. These valves have pneumatic positioners and can be positioned to any degree of opening desired. This is required to maintain the two areas under a balanced positive pressure. Vault pressure is recorded by vacuum-pressure recorders. Position control pushbuttons for the valves and the pressure recorders are installed in the JJ and GG panels in the control room.

A radiation detector is installed between the two suction lines in the secondary area to indicate radiation levels of the gas. If the radiation level in either of the two lines exceeds the setpoint, the alarm will sound in the control room and locally at the detector. No other automatic action takes place. The HP should be notified and a survey made to determine the magnitude of the radiation to aid the shift supervisor in subsequent action.

Nitrogen is drawn through the individual suction lines into a common suction header by the D-E fan (Figure 5-2). From the suction

located in the valve pit south of the compressor vault. The slope of the header is such that any liquid collected in the suction tank, drained from the compressor interstage coolers, or carried over from the wash cells, will drain by gravity into the radioactive liquid waste sump tank.

Two horizontal, reciprocating, vent compressors are provided, each with sufficient capacity to independently handle the normal system load. This allows one unit to be shut down for maintenance when required without impairing the system operating efficiency. Vent gases from the suction tank are pulled into the compressor and can be set so that either compressor, or both, are connected to the suction line. Vent gases within the compressor are compressed to 110 psig and discharged through lines 511 and 512 for compressors A and B respectively. Discharge valves V511 and V512 are provided to complete the isolation of either compressor when required. Line 511 is joined by line 512 and proceeds to the decay tank group selector valve (513K). This valve is a three way valve and is used to select either tanks A and B or tanks C and D, and through the construction of the valve when one group is selected, the other is isolated. Leaving valve 513K, the compressed gas follows the selected path to another three way valve (V498K for tanks A and B and V514K for tanks C and D). This valve is used to select the individual tank to be filled, and again through the construction of the valve when one tank is selected, the other is isolated. The compressor suction and discharge valves and the decay tank three way valves are all plug type and require a periodic application of grease to remain leak tight and sealed. Transmitting pressure switches on each decay tank send signals to a pressure readout on control room panel H. Pressure switches on each tank are set to sound an alarm when the pressure within the tank exceeds 90 psig. At this time another tank is selected for filling and the first tank is ready to be sampled for radioactivity and vented to the atmosphere if conditions permit. As a safety backup, a high pressure switch is connected to each compressor discharge line and will trip out the compressor if the pressure exceeds 110 psig thus protecting the system.

Radioactivity of the vent gases is determined by drawing a sample from the individual decay tank and counting it. The results are

reduce the absolute pressure to 25 in. Hg (5 in. vacuum). At this time a bypass solenoid valve between the compressor suction and discharge lines is opened, and the compressor will cycle for one minute. If the suction tank pressure has not increased at the end of the one minute cycling period, the compressor is stopped and the solenoid valve closed; however, if the tank pressure has increased, the solenoid valve closes and the compressor again starts to pump on the system. Operation of the helper compressor is only required when the system load is such that the leading compressor cannot keep up. In this case, when the suction tank absolute pressure rises to 29 in. Hg (1 in. vacuum), the helper compressor starts and continues to pump on the system until the absolute pressure in the tank is reduced to 25 in. Hg (5 in. vacuum). At this time both compressors cycle for one minute; the helper compressor then stops and the lead compressor either stops or continues to operate depending upon the system load.

For abnormal conditions, either compressor can be isolated by closing its suction and discharge valves and setting the control switch to the "off" position. Additionally, either or both compressors can be operated manually by manipulating the "off-on-auto" control switch.

Vent system leaks are detected by the rate of decay tank filling and by vent compressor operating time. When a system leak occurs, a larger volume of gas must be pumped by the compressor; therefore, the compressor operating time is increased, and the 2700 standard cubic feet decay tanks are filled more rapidly. As the vent system is quite complex and lengthy, with much of it underground and inaccessible, a system for locating leaks is required. The first step for vent system leak detection is to determine the logical sources of leakage; the most prevalent sources are: (1) inert gas pressure control station pressure set higher than the relief valve setting causing the valve to lift, (2) relief valve seats leaking, (3) wash cells or sump pump tank seals allowing air to enter the system, and (4) air leakage through fittings or ruptured piping. As header 492 services the pressure control stations, closing valve 492C and observing the decay tank filling rate will determine if the leak is class (1) or (2). If the filling rate remains unchanged, header 492 can be considered leak tight and valve 492C is opened and the search continued. Header 506 is isolated by

In the case of low level, a visual check will either verify or reject this cause. If the strainer basket should become plugged, change over to the standby strainer and remove and clean the plugged basket. After the basket is cleaned and reinstalled. be sure to check it for tightness by momentarily placing it in service, and then visually checking for leaks. This will insure against a leaking standby strainer when you next need one.

J. RADIOACTIVE LIQUID WASTE SYSTEM

Providing the same function for liquids as does the radioactive vent system for gases, the liquid waste system collects all radioactive (potential or active) liquids and stores them in a tank for monitoring and disposal. This portion of the manual will be devoted to a description and operation of the system.

1. System Description

Most of the radioactive liquid waste at the SRE is generated in the core element wash cells (approximately 15 gallons per element washed). Other contributors are the high bay floor vent stations, and the innerstage coolers on the vent compressors. (Figure 5-25) Three tanks (T-1, T-2, and T-3) are located west of the SRE building, two tanks (T-1 and T-2) are at the bottom of the liquid waste sump pit and the other tank (T-3) is located at ground level in the cubicle near the sump pit. Tanks T-1 (150 gallon capacity) and T-2 (350 gallon capacity) receive all the liquid waste generated. The other tank T-3 (3250 gallon capacity) provides an intermediate storage prior to disposition by RMDC. Drain valves V-901, V-902, and V-903 (for wash cells C, B, and A, respectively) join header 901 which connects to both sump tanks. The slope of the line is towards the sump tanks; therefore, the liquid from the wash cells flows by gravity to the tank. Connecting the sump tanks to the wash cell vent system is header 943-960. This header receives moist vent gases from the wash cell vacuum pump, excess liquid from the compression water tank, and flush water from the wash cell baffle box (Figure 4-11 and 5-11). Radioactive vent header 506 joins header 943 west of the SRE building. Vent gases from the wash cell are transferred to header 506, and liquid from the high bay floor vent stations and the vent compressor innerstage coolers is transferred to header 943.

16. Power and sodium flow will be increased in steps of 2%. Remove from automatic control. Power will be increased first. Flow will then be increased by the same percentage to compensate for the increased power generation rate.
17. When an equilibrium temperature and power condition exists, the next step increase can be made.
18. Reset the automatic flux level controller to NF (full power).
19. It will be necessary to increase the flow ratio control as power increases to hold the reactor cold leg temperature at 500° F.
20. As full power is approached, insert one control rod to increase the period to ∞ at full power.
21. The flow ratio control should be set on (1) one when the reactor is at full power.
22. Select either control Rod #3 or #4 to act as the regulating rod.

Note: Both regulating rods must be at their upper limit prior to switching to automatic control.

23. Move the "Indicator Range" selector for the selected rod to the "Vernier" position.
24. Move the "Rod Control Selector" to the rod selected "Auto" position.
25. Drive one control rod in or out to move the regulating rod to the center of its control range.
26. The reactor is now at full power on automatic control.
27. [illegible] trimming of flow and control rods will be required [illegible]ly until equilibrium is established.
28. [illegible]eration of the "Fission Chamber Control" switches [illegible] chambers to the out position. These chambers are required only during start-up and will be damaged by excessive flux if left adjacent to the core while in full power.
29. It may be necessary to adjust the auxiliary flow and auxiliary heat exchanger louvers to control the auxiliary primary cold leg temperature at 500° F.

(4) With the reactor pressure at 3 psig observe the flow indicator on PC-432. If flow is indicated, the set point is above 3 psig and requires resetting.

(5) If no flow is indicated repeat steps 3 and 4 for PC-417.

(6) If both pressure control stations are operating properly and the reactor pressure again rises to 4.5 psig a survey of the entire helium supply for the main heat transfer system is required. Check all branch lines for abnormal flow. Correct if necessary.

b. Corrective action to increase the low pressure:

(1) Check PC-417 and PC-432. Flow should be indicated for reactor pressure lower than 3 psig.

(2) If pressure continues to drop bring the pressure back up to 3 psig by opening PC-417 bypass valve (V-268C).

(3) If no flow is indicated at PC-417 and PC-432 for a pressure less than 3 psig the controllers must be reset.

(4) If PC-432 and PC-417 are operating properly a leak is indicated. Check vent relief PC-600 for set point. If normal a survey of vent and helium corrections must be made to determine leakage point.

20. Dummy fuel channel temperatures: Greater than 1050° F.

a. Corrective action to reduce the temperature:

(1) Reduce reactor power.

(2) If other reactor temperatures are normal this alarm may mean a plug in the sodium flow orifice at the bottom of the element. This will require removal of the element, after shut down, for cleaning and hot cell inspection. SRE-411.5.

21. Reactor sodium level: Less than 4' 3" above reactor outlet line.

a. Corrective action to increase the level:

(1) This alarm will indicate a leak in the primary sodium system.

(2) The reactor will be maintained at the power level present at the time of the alarm until the cause is investigated.

(3) If the situation is considered by supervision to warrant such, shut down using standard shut down procedure.

34. Helium pressure in core tank cavity greater than 5 psig or less than 2.5 psig.

a. Corrective action to reduce the high pressure:

(1) This alarm indicates failure of the helium pressure relief RV-402 (relieves at 4.5 psig). If pressure continues to rise it will be relieved at 6 psig by the vent system relief RV-600.

(2) Vent the core tank cavity to the vent system by opening valve (V-459) and reduce the pressure to 3 psig. Close valve (V-459).

(3) At 3 psig there should be no helium flow indicated at PC-473. If flow exists reset PC-473 for no flow at 3 psig.

(4) By opening PC-473 bypass valve (V-270C) increase pressure to 4.5 psig. RV-402 should be set so that flow is just starting at 4.5 psig.

b. Corrective action to increase the pressure:

(1) This alarm will indicate either a leak in the core tank cavity or a malfunction of the pressure control station and/or vent system relief valve.

(2) If pressure is decreasing check for flow at PC-473. If flow is indicated the helium system relief valve RV-402 or vent system relief valve RV-600 may be sticking in the open position. These valves must be checked and reset if necessary.

(a) If no flow is indicated at PC-473 this will indicate malfunction of PC-473 and will necessitate repair or resetting. Helium pressure can be controlled manually during this period by operation of PC-473 bypass valve (V-270C).

(a) If no flow is indicated at PC-505 this will indicate malfunction of PC-505 and will necessitate repair or resetting.

(3) Nitrogen pressure can be controlled manually during this period by operation of PC-503 bypass valve (V-358C).

42. Compressor suction tank pressure: Greater than 14 psia.

a. Corrective action to reduce the pressure:

(1) Check to see that both compressors are operating.

(2) See that suction tank valve (V-519A) is closed.

(3) Check the operation of the high pressure switches (HPS-612) (HPS-614).

(4) Operate one compressor at a time to isolate the defective unit (See SRE-407.2 Compressor Operation).

43. Mercury pressure in the boiler: Greater than 210 psig or less than 165 psig.

a. Corrective action for high pressure:

(1) A high pressure alarm will indicate a leak from the steam side of the boiler into the Hg.

(2) The reactor should be shut down using standard shut down procedure.

(3) The after glow heat load transferred to the auxiliary system.

(4) Valve off the steam generator from the main sodium system by closing valves (V-164) and (V-165).

(5) Close the main primary block valves (V-101) and (V-103).

(6) Drain the sodium from the steam generator by performing the following operations:

(a) Energize heaters on drain lines (166-2-A), (167-2-A), (132-2-A) and the secondary fill tank.

(b) Apply helium pressure to the steam generator freeze trap by opening valve (V-353).

(c) Energize freeze trap heater. A helium flow will indicate that the trap is free of sodium.

(d) Open the steam generator drain and vent valves (V-169 and V-170).

405.2 Normal Operation of Helium System

A. The helium manifold system alarm shall be set to trip at 125 psig. Upon receipt of this alarm in the control room the system shall be manually valved to the standby manifold. The exhausted bottles shall be replaced as soon as possible and this manifold purged and placed in standby service. The change in helium bottle inventory shall be entered on the Inert Gas Inventory Log, Form 609.

B. Normal operating pressure adjustment for the various stations shall be as shown under Helium System, SRE-403.

C. PC station bypass valves are to be maintained closed at all times. Exceptions are during the purging operation, a period of maintenance on the PC valve or assembly, or when checking relief valve operability.

D. Changes in the following stations are required under certain specific operating conditions.

1. PC-413 and PC-427 Main and Auxiliary Double Wall Pipes

These pressure control valves supply helium to these spaces at 12 psig to act as a heat barrier between the two pipes. Each double wall pipe enclosure vent valve to the vent system is normally closed (V-452) (V-467) (V-469). When the sodium pool in the reactor is to be drained, the supply block valves (V-265A) (V-285A) shall be closed and the vent valves opened.

2. PC-417 and PC-418 Reactor Atmosphere

PC-418 is set for 1/2 psig to be used during reactor start-up. PC-417 is set for 3 psig and is to be used after reactor atmosphere operating temperature has been attained. These PC stations are alternated by use of a two way solenoid valve located on panel HH. Upon reactor shutdown open PC-418. Should the pressure alarm (5 psig) associated with the reactor atmosphere sound, an immediate investigation shall be initiated to ascertain the trouble. Check the helium supply system for operability and the vent relief system as described in SRE-407.1, item A, 1 b. Close the helium supply blocking valves (V-292A) (V-292A1) (V-268A) (V-268A1) if the pressure reaches 7 psig.

3. PC-432 and PC-433 Primary Fill Tank

Identical to system 2, above, and should be operated in conjunction with reactor atmosphere supply as both systems are directly tied together.

4. PC-452 Secondary Fill Tank

When filling the secondary fill tank with sodium the helium supply shall be cut off by closing the block valve downstream from PC-452 (V-298B) and the normally closed block valve in the vent line opened (V-299). Upon initiating heating, the vent block valve shall be closed.

5. PC-457 Main Secondary Expansion Tank and Cold Trap

During normal operation the supply is set for 3 psig, the relief valve bypass (V-309C) is closed, the block valve ahead of the relief valve (V-309F) is open and the gas supply to the freeze trap (V-310) is closed.

a. Upon secondary system filling close the helium supply valve (V-307A), slightly crack open the PC bypass valve (V-307C) and open relief valve bypass (V-309C).

b. Upon secondary system draining the block valve to the relief valve (V-309F) is closed and the gas pressure raised to force the sodium into the fill tank.

6. PC-462 Auxiliary Secondary Expansion Tank

Similar to above.

7. PC-475 Sodium Service Transfer Tank

This tank is used only during the transfer of sodium from the 55 gal. drums to the primary or secondary fill tanks for operation of the helium system. See SRE-403.1 and 403.2.

8. PC-490 Steam Generator and Secondary Block Valves

During normal operation this pressure control station is supplying back-up helium pressure for the freeze traps on the main secondary system and the steam generator. Helium is supplied through this station for purging the steam generator and associated piping prior to filling with sodium.

E. During normal reactor operation each station flow indication and pressure shall be checked once each day and recorded on Form 610.

407 Vent System

The vent system, by a system of headers, collects all process gases containing or possibly containing radioactivity, Fig. 13. See Fig. 14 for P.C. station location. These gas sources are:

A. Normally Containing Radioactivity

1. Reactor and primary fill tank helium blanket atmospheres.
2. Primary circuits and disposable cold trap vents.
3. Cleaning cell vents.
4. Fuel handling coffin service connections.
5. Radioactive liquid waste accumulation and hold-up tanks.
6. Hot cell radioactive gas service connections.
7. Hot trap vents.

B. Normally Non-radioactive

1. Primary circuit pumps (2).
2. Reactor drain line freeze trap vent.
3. Helium supply system relief valves to the following services:
   a. Main and auxiliary primary block valves (2).
   b. Main and auxiliary primary double wall pipes (2).
   c. Control rods.
   d. Safety rods.
   e. Disposable cold trap system.
   f. Main and auxiliary primary piping and equipment (2).
   g. Core tank cavity.
4. Nitrogen supply system relief valves to the following services:
   a. Main and auxiliary primary systems galleries (2).
   b. Primary fill tank vault.
   c. Disposable cold trap vault.
   d. Insulation cavity.
5. [illegible] system gallery atmosphere discharge.
6. Insulation cavity vent (normally closed).
7. Core tank cavity vent (normally closed).
8. Main and auxiliary primary block valves (4) (normally closed).
9. Main and auxiliary double wall pipes (2) (normally closed).
10. New fuel storage cells.

The gases from the services listed in section A go directly to the compressor suction tank located north of the reactor building in a shielded pit. Gases from services B are monitored for radioactivity in the main headers, and according to the activity are passed to the compressor suction tank or bypassed through a filter to the atmosphere.

The compressor suction tank is maintained slightly below atmospheric pressure at all times, thus minimizing the intraflow of gas between service headers, by assuring flow from higher pressure service outlets directly to the compressors. Two compressors in parallel draw from the suction tank; compress the gases to 114 psia and exhaust to one of four 350 cu.ft. decay tanks. Here the gases may be sampled and exhausted upon sufficient decay, see SRE-410.3. Exhaust rate is controlled by a pressure reducing valve and an associated hand operated globe valve. Exhaust gases pass through a CWS filter and are diluted by the hot cell ventilation exhaust stream. Should additional dilution be necessary duct work and associated damper systems are installed to allow the use of a 25,000 cfm dilution fan.

407.1 Normal and Emergency Operation

The operating procedures for the various services of the vent systems are as follows, Fig. 13:

4. Radioactive Services Operation

1. Reactor and primary fill tank atmospheres.

These two spaces have a common helium atmosphere, which is supplied as per operating procedure SRE-405.2, item 4 b, operation of the helium system. The helium pressure control station servicing this atmosphere does not have a pressure relief valve. Relieving of excessive pressures, which could result from misoperation or malfunction of the helium supply valves, is accomplished by relief valves PC-600, RV-603, and RV-602 in the vent system. PC-600 is diaphragm operated and is set to relieve at 4.5 psig. The valve will close if the disphragm fails. RV-603 relieves at 6.5 psig upon failure of RV-600. RV-602, upon the failure of the other relief valves, will relieve the atmosphere of the vessels to the main gallery.

a. Relief valve bypass valve (V-497) will normally be closed. It will be opened only during purging of the reactor and fill tank atmospheres.

b. Bypass valve (V-497) may be opened to relieve the reactor and fill tank atmosphere if the pressure reaches 8 psig caused by malfunction of the regular relief system. This method of relief shall be by supervisor's permission only. (See SRE-408.2, item 4 b.)

c. Freeze trap vent valve (V-623) shall be closed except during disposable cold trap vent operation (see SRE-404.3).

d. A sampling valve, which is normally closed, is provided immediately upstream of relief valve assembly. Sample to provide information on completeness of helium purges or of gaseous fission product activity. (See SRE-410.4, sampling procedure.)

2. Helium vents from freeze traps for main and auxiliary primary cold traps and piping adjacent to intermediate heat exchangers. The lines are to be used in conjunction with primary system filling procedure (SRE-405.2).

3. Cleaning cells and new fuel storage cells vents.

Each of the three cleaning cells has two vent connections, one passes through a cold trap, a vacuum pump, and an oil trap before exhausting. The other bypasses the vacuum pump system.

The three new fuel storage cells are provided with a helium system supply and a vent system on the suction side of the above mentioned vacuum pump. Operation of the cleaning cell and the fuel storage cell vents are discussed under fuel handling procedure (SRE-414) normal procedures are:

a. The three block valves (V-670-1) (V-670-2) (V-670-3) and header valve (V-679) from the new fuel storage cells are closed unless purging of cells is in progress.

b. The six block valves (V-676-1) (V-676-2) (V-676-3) (V-501) (V-502) (V-503) from the fuel cleaning cell are closed unless washing operations are in progress.

c. The vacuum pump plug valve (V-683) is maintained closed unless the pump is being operated.

d. Cold trap drain valve (V-684) is normally closed. It shall be opened following use of the trap to allow the condensate to drain to the liquid waste sump tank.

e. Sample connection valves are provided at each fuel cleaning cell. These valves are normally closed. (See SRE-310.4, sampling procedure.)

4. Fuel handling coffin service connections.

Two connections to the vent system are provided for the fuel handling coffin. One is located immediately to the west of the hot dry fuel storage area and the other is in the coffin service area.

a. Block valves (V-504G1) (V-504G2) at each service connection are to remain closed unless service is being employed. Always make connection prior to opening block valves and close block valves, prior to making disconnect.

b. Valve to atmosphere (V-518) from the coffin service area connection is normally closed. Use of this valve requires approval of supervision.

5. Radioactive liquid waste and hold-up tanks.

The vent system ties in directly and drains to the liquid waste sump tank. A vent system header also serves the eight waste hold-up tanks. This line relief and associated bypass valves operation is discussed under hold-up and storage tanks (SRE-411.2).

6. Hot cell vent service connection

This connection is for the disposal of radioactive gases resulting from hot cell operations.

a. Vent service header block valve (V-520) shall be closed unless header is in use.

b. Use of header requires prior approval of supervision. See hot cell procedures (SRE-415).

. Normally Non-radioactive Services Operation

Gases from the below listed services are monitored by in-line radiation instruments. If the activity level is below 10^{-2} μc/cc the gases are directed to the atmosphere via indicator activated solenoid valves. If higher activity levels are present the gases are compressed for storage in the decay tanks.

1. Operation of the above diversion station is as follows:

a. Filter station bypass valve (V-508) is normally closed. This valve may be opened in conjunction with valve (V-507) for specific purging operations or during filter station maintenance with supervision approval.

b. Proper operation of radiation indicators and associated solenoid valves shall be checked montly by the health physics group.

2. Primary pumps vent.

This line connects to the main primary and auxiliary primary pump casing. A pressure controller, with associated block and bypass valves, is incorporated in each pump branch.

a. The pressure controller bypass valve is normally closed except during the purging operation.

b. The controllers are set to vent the pump casings at 15 psig.

c. Helium is supplied to the pump casings at a pressure determined by the operating characteristics of the pump.

d. Manual venting is accomplished, after closing the helium inlet valve (V-318 or V-326), by opening the pressure controller bypass valves (V-521C) (V-522C) and observing the casing pressure indicated on the pump being vented.

3. Reactor drain line vent.

This line connects to the vent system via a freeze trap and a block valve (V-528). By means of compressor suction tank bypass line 530 and associated valve (V-530) the gas compressors can be used to prime the reactor drain pump. For use of this pump see SRE-405.7.

Normal valve positions are:

a. Valve (V-528) closed excepting during primary sodium system purging operations.

b. Valve (V-530) closed except during reactor drain pump priming operations.

4. Helium supply system relief valves.

Operation of relief valves from pressure control stations PC-410, PC-413, PC-419, PC-421, PC-425, PC-427, PC-473, PC-434, PC-436, PC-438, PC-446, is described and discussed under helium system, SRE-408.

5. Nitrogen supply system relief valves.

Operation of relief valves from pressure control stations PC-507, PC-511, PC-509, PC-503 and PC-505 is described and discussed under nitrogen system, SRE-409.

6. Insulation cavity vent.

The nitrogen system maintains a minimum pressure of 3 psig in this space. Relief valve PC-505 on the supply line will exhaust into the vent system (see item 5, above) should the pressure exceed 4.5 psig. A relief valve is also incorporated in the [illegible] system and will relieve at 6 psig should the other relief valve fail.

a. Blocking valve (V-491) is normally closed. It is opened only during purging operations concerning the insulation cavity (see SRE-409.1).

b. A sampling valve, which is normally closed is provided immediately upstream of the blocking valve. Sample to provide information completeness of purging operations.

7. Core tank cavity vent.

The helium system maintains a minimum pressure of 3 psig in this space and relief valve PC-473 on the supply line will exhaust into the vent system (see item 4 above) should the pressure exceed 4.5 psig. A relief valve is also incorporated in the vent system and will relieve at 6 psig should the other relief valve fail.

a. Blocking valve (V-459 is normally closed. It is opened only during purging operations concerned with the core tank cavity or the helium system. (See SRE-408.1.)

b. A sampling valve, which is normally closed, is provided immediately upstream of the blocking valve. Sample to provide information on completeness of purging operation or integrity of core tank.

8. Main and auxiliary primary block valves.

Each of the four valves is directly vented. The vent is employed only when the valve stems are purged with helium. Purging should normally be required only following valve maintenance or loss of valve seal during operation.

a. Blocking valves (V-471) (V-473) (V-453) (V-456) are normally closed. They are opened only during controlled purging operation. Should these valves inadvertently be left open and should the valve freeze seal fail, sodium could flow into the vent system.

b. Sample valves are provided immediately upstream of each blocking valve. Sample to provide information on completeness of purging operation.

9. Main and auxiliary double wall pipes vent.

The helium system maintains a minimum pressure of 12 psig in each of these spaces. Relief valves on the supply system will relieve to the vent system at 110% of this pressure (see item 4, above).

a. Blocking valves (V-452) (V-469) are normally closed. They are opened only during purging operations.

b. Sample valves are provided immediately upstream of the blocking valves. Sample valves are closed excepting when sampling is in progress (SRE-410.4). Sample to provide information on completeness of purging and integrity of double wall pipes.

C. Vent System Purging

1. System Piping and Services

Individual services are to be purged according to the respective gas system purging procedures. Initial purges, in which radioactivity is not a problem, are to be diverted from the compressor suction tank by opening of block valves (V-507) and (V-508). Individual services purges following maintenance, or for other reasons, after the vent system is radioactive, are to be routed normally by the vent system, as described in SRE-410.

2. Compressors and Suction Tank

The suction tank and each of the compressors may be purged independently or all together, as required for accessibility purposes in the compressor vault or for unit maintenance reasons. Purging gas is supplied by connection of a helium bottle(s) to the pressure reducing station for this purpose adjacent to the compressor vault. This station will reduce the helium pressure from 2300 psig to 5 psig.

Valve manipulation sequence or procedures for any of the purging operations are to be authorized by supervision when the procedures are required. Extended purge success or termination point may be judged by a reduction in radiation intensities at the particular equipment piece.

The vent system valves that are normally closed are indicated in solid color on the P & I drawing.

408.2 Holdup and Storage Tank Operations

General

The ten waste holdup tanks are divided into two sections of five tanks each by a center shielding wall. Should line or vessel servicing be required, the waste in the section in question may be transferred to the other, thus allowing better accessibility. Transfer is accomplished by pressuring the vessel to be emptied, and provisions are installed for flushing of the vessel and lines.

The holdup tanks are connected to the radioactive vent system via 30 psig relief valves and an associated bypass valve for each of the two sections. The bypass valves are normally open; the relief valves used only when pressurizing or if isolation is required. Each holdup tank is connected to the vent header directly by a float type vent valve, which will allow gases to pass but closes against liquids, thus preventing flooding of the vent system by overfilling of a vessel.

Each of the holdup tank vaults is provided with a sump connected via a line and valve to the holdup tank drain header. The drain line valves are normally closed to contain any sump accumulation, which could indicate leakage of lines or vessels in the respective vault.

Two direct bury 5000 gallon storage tanks are tied into both the sump tank pump out line and the drain header from the holdup tanks. These vessels have radiation indicators and liquid level indicators which are read at the waste disposal station. They are vented directly to the atmosphere. A pump out line has been provided for each vessel should future removal of the liquid waste be required.

1. The holdup tanks and the waste storage tanks shall be inventoried once daily and the data recorded on Form 612.
2. The vault sumps shall be inspected once daily for liquid accumulation. Any liquid present may be sampled to determine the source or drained directly to the storage tanks at the discretion of the supervisor.

coffin is moved because of an interlock which interrupts all power to the coffin traverse and bridge motors when the index leaves its "up" position. The coffin must be properly indexed (all four lights on) before the rest of the sealing operations can be completed. A light on the control panel indicates the "index head up" position.

b. The gas seal may now be lowered by operating the "lower gas seal" switch on the control panel. Three solenoid operated two inch pneumatic cylinders raise and lower the gas interlock sleeve assembly which seals to the index ring previously placed into the fuel plug casing. Lights on the panel indicate the "up" and "down" positions.

c. When all four index lights are on an interlock is also completed which permits the pickup device to be lowered.

d. A switch lowers the pickup device from its travel position to its "down" position. A micro switch stops its downward motion and turns on a panel light (grapple down), and an interlock is completed that allows the vacuum buggy to be operated. This vacuum system will not operate unless the grapple plug is in the "down" position to hold down the fuel plug assembly during evacuation. In case the down switch fails, an override switch (emergency vacuum buggy) allows power to the vacuum buggy.

e. The shielded vacuum buggy is now used to evacuate the gas interlock. The vacuum buggy consists of a Kinney vacuum pump and shielded discharge tank. All valves on the buggy are solenoid type, operated from a separate pendant.

f. Prior to using the vacuum buggy for evacuating the interlock the following procedure will be carried out: The flex hose is connected to the vent system in the coffin service area. The tank is evacuated to the vent system. Solenoid valves are then arranged so as to permit evacuation of the interlock through the flex hose into the shielded tank.

Each of the above listed areas is served by a pressure control station comprised of a pressure control valve with a pair of blocking valves and a bypass valve and line. In the adjacent piping downstream from each control station there is a locally read flow indicator and pressure indicator, and a pressure relief valve connected to the reactor vent system. BPI-506 transmits the pressure as controlled by PC-505 to panel HH in the control room - BPHA-514 BLPA-514 actuate an alarm on panel HH in the control room if the pressure downstream from PC-505 goes above 5 psi or below 2.5 psi.

2.2.1 Testing

Using Dwg. 9693-978104 as a check list, all units to receive a nitrogen atmosphere will be sealed. Each pressure controller and relief valve will be set at the value indicated on the P&I diagram. The pressure gauges will be checked against the pressure controllers, and differences corrected by detail calibration. The pressure will be raised in each system to actuate the relief valve and verify its setting. The pressures in the insulation cavity will be varied to check the settings on BLPA-514 and BHPA-514.

2.3 Venting

Gases from the areas containing a nitrogen atmosphere, with the exception of the reactor insulation cavity, intercommunicate with each other through a parallel vent header system. Two tributaries, one collecting from the main primary and main auxiliary galleries, and one from the primary fill tank vault and disposable cold trap vault vent through a common vent line. Gases from each tributary pass through a radioactive dust filter station. They may be released to the vent system through a common pressure control station (PC-522) if the pressure exceeds 1.5 psig. Additional details of the venting process are given in section 3, "Vent System."

2.3.1 Filters (Equip. No. 781118-A & 781118-B)

Each radioactive dust filter station has two types of filtering media, a coarse section to remove

the major part of the duct, and a CWS section to remove fine particles. Each filter is installed with a pair of blocking valves and a bypass valve.

2.4 Purging

Purging of the nitrogen system will be performed after the galleries, vaults, and insulation cavity have been sealed for operational testing of the sodium systems.

With the constant pressure system vented to atmosphere through SV500 and the pressure controllers and valves throughout the entire system as prescribed on Dwg. 9693-978104, the system will be purged by a nitrogen supply from a 40,000 scf trailer. The flow rates to each unit will be set as prescribed on the above drawing by the globe valve on the low pressure side of the respective pressure control valves. Under these conditions one complete volume change should be effected in six hours. Three complete volume changes are possible from the trailer supply. The three changes should reduce the oxygen concentration in the purged volumes to 5%.

3. Vent System (Dwg. No. 9693-978102)

The vent system collects all gases that could be radioactive from the reactor, its auxiliaries, and service systems. The gases consist of varying amounts of helium, nitrogen, air, hydrogen and water vapor. Solenoid valves in the collection lines are operated from radiation monitors to route the gases to a retention system if they are radioactive and to the atmosphere if they are not.

3.1 Non-Radioactive Gas Disposal

Gases with activity below a certain level are routed by the solenoid valves mentioned above, to the reactor building ventilation exhaust stack. The collected gases are mixed with the hot cell exhaust and blown to atmosphere. At the valve assembly for diverting the gas to the ventilation stack, there is a manual valve connection direct to atmosphere, for use when the vent system serves as an exhaust for purging operations.

3.2 Radioactive Gas Disposal

Radioactive gases are routed to a compressor suction tank by the radiation monitor station. The gases are pulled from the suction tank by two compressors and moved to any of four 350 cu. ft. decay tanks as selected. The four tanks have a combined capacity of 10,000 scf of gas at 113.7 psi. Selection of a decay tank for storage is made by three-way valves which are always open in one of two possible flow paths. One valve directs flow to a manifold for tank A & B or C & D alternately. Another valve directs flow to tanks A or B alternately and the third to C or D alternately. Release of gas from a selected storage tank to the exhaust stack is done by a pair of three-way valves, one serving tanks A or B and one serving C or D. These valves have a position in which neither tank is vented to the stack. A third valve in the discharge manifold common to all tanks permits selection of use tanks A & B or tanks C & D.

A line from the discharge line common for all four tanks to the compressor suction tank permits transfer of gas from one tank to another. The flow of gas released from a decay tank to the vent stack is controlled by a pressure control station PC-611.

A radiation recorder-controller in the ventilation exhaust stack actuates a safety valve to stop the flow of gas from the decay tank if the radiation in the stack exceeds a set level.

A helium supply station comprised of two 225 scf helium bottles is connected to the system at the compressor suction tank, and at the suction and discharge sides of the compressor to permit purging of the system.

3.2.1 Suction tank (Equip. No. 78122)

The 100 cu. ft. suction tank is located on the hill side 170 ft. north of the reactor building, and is buried under earth for shielding purposes. There are a pair of dehydrators connected in parallel on the line between the suction tanks and the compressor. The humidity of gas passing through the dehydrators will be reduced to $\sim$ 16% @ 80° F.

3.2 1.1 Instrumentation

The pressure of gas in the suction tank is indicated locally on PI 601. If the pressure exceeds the prescribed setting, an alarm is actuated in the reactor control room as BHPA-602. Two high pressure switches and two low pressure switches (one pair for each compressor) control the compressor operation. These switches and the resultant compressor operation are covered in section 3.2.2.

3.2.1.2 Testing

The dehydrators will be checked for a proper chemical charge. All valves on lines entering the tank will be checked for operability and positive seating.

3.2.2 Compressors (Equip. No. 78121A & 78121B)

The two radioactive vent gas compressors are located in a concrete vault on the hillside 170 ft. north of the reactor building. Each compressor is rated at 20 scfm with a suction pressure of 10.7 psia and a discharge pressure of 113.7 psia.

3.2.2.1 Instrumentation and control

Each compressor is controlled from a high and low pressure switch on the suction tank. The pressure switches for compressor A will be set to start it when the pressure in the suction tank is 13.2 psia or higher and to stop it when the pressure drops to 10.7 psia. The switches for compressor B will be set to start it when the pressure in the suction tank reaches 13.7 psia with compressor A running and to stop it when the pressure drops to 11.2 psia with compressor A running.

When the low pressure switch drops out, the power to the compressor is maintained by a holding coil retained by a timer. Simultaneously a solenoid operated by the pressure switch opens

a bypass line between the suction and discharge of the compressor permitting circulation through the compressor. If the low pressure persists more than a pre-set time the timer cycle is completed, the holding circuit is broken, stopping the compressor. The above control system is designed to minimize cycling of the compressors. It is estimated that the operating pressure cycle will be between 10.7 and 13.2 psia and will be handled by compressor A with compressor B not functioning. It is recommended that the pressure control switch settings be altered periodically to permit the compressors to alternate carrying the load. The control switch for the compressors is located on the control panel adjacent to the compressors. An overload protection in the motor starter of each compressor serves as protection against damage to the compressor if the plug valve in the discharge line is inadvertently closed.

3.2.2.1.1 Testing

The control point settings of the pressure switches controlling the compressors and the operation of the compressor bypass solenoid valves will be verified. The function of the timers will be checked. The compressor motor starter overload protection will be tested on each compressor by purposely closing the blocking valve in the discharge line.

3.2.2.2 Testing

The compressors will be operated with no load for at least one hour prior to operation under load. They will be operated on air to check for motion restrictions resulting in overheating. The compressor control features will be rechecked with the compressor operating to insure proper sequencing.

3.2.2.3 Purging

The suction tank and compressors will be purged from the two 125 scf helium supply tanks described in section 3.2 above. Helium admitted to the suction tank and to the piping at either side of the compressors will be vented to atmosphere through the building ventilation stack. The 2" - 67 blocking valves normally closed in lines connecting the compressors to ventline 494-2"-H must be opened to permit this passage. Solenoid valves 605 and 606 in the by-pass lines around the compressors also must be open.

Pressure control stations PC-609 and 610 reduce the pressure of helium from the supply tanks from 2300 to 100 psig and from 100 to 5 psi gauge respectively, before it is used for purging.

3.2.3 Decay tanks (Equip. No. 78123A, 78123B, 78123C, 78123D)

The four 350 cu. ft. decay tanks are located on the hillside about 120 ft. north of the reactor building. The tanks are stainless steel, and have a combined capacity of 10,800 scf of gas at 113.7 psia. They are buried in a concrete vault for shielding purposes.

The purpose of these tanks is to hold radioactive gases until the gases decay to an activity level considered as safe for discharge to the atmosphere through the reactor building ventilation exhaust stack. The valving for these tanks is described in section 3.2 above.

3.2.3.1 Instrumentation

Each decay tank is equipped with s pressure pickup which indicates locally, transmits to a multipoint indicator on panel HH in the reactor control room, and actuates a high

pressure alarm on panel HH in the control room. The instrument numbers for each tank are tabulated below.

Tank No.	A	B	C	D
Local Press. Ind. (PI)	603	608	616	619
Press. Ind. in Control Room BPIM	603	608	616	619
Press. Alarm in Control Room BHPA 115 psi	603	608	616	619

3.2.3.2 Testing

All valves used in tank operations will be checked for positive shut off under full tank pressure. The pressure indication and high pressure alarm systems will be checked for operation under simulated operating conditions.

3.2.4 Gas vent control and filter station

There are two points from which gas may be exhausted from the vent system to the reactor building ventilation exhaust stack. A radiation monitoring station in the trunk line from the collection system to the compressors exhausts non-radioactive gases to the stack directly, and routes radioactive gas to the compressor. Flow of gas exhausted from the decay tanks to the stack is controlled by a pressure control valve. A second monitoring station on the exhaust line from the decay tank to the stack permits only gases below a prescribed activity level to be exhausted to the stack. Gases exhausted to the stack by either monitoring station are filtered to remove the particulate activity.

3.2.4.1 Exhaust direct from collection system

A Jordan ion chamber in the trunk line from the collection system monitors the passing gases. The output from the ion chamber is indicated on a simplitrol milivolt meter on panel GG in the reactor control room. Contacts on the simplitrol operate solenoid valves in the line to the compressor and in a shunt line to the

vent stack. Radioactive gases are routed to the compressor. Gases with an activity lower than a certain value are vented directly to the stack. A check valve downstream from the solenoid valve in the exhaust line prevents back flow into the collection system when venting the decay tanks to the stack.

3.2.4.2 Exhaust from decay tanks

The flow of gas released from the decay tanks to the reactor building ventilation stack is controlled by a pressure control station, and is filtered by a CWS type filter to prevent the discharge of radioactive particles to the atmosphere. A flame arrestor prevents back explosions into the control system piping.

3.2.4.2.1 Instrumentation and control

The pressure of gas released is controlled by pressure control No. PC-611 and indicated locally on pressure indicator PI-622. The flow rate of exhausted gases is indicated locally on flow indicator FI-600. The control station may be blocked out with blocking valves and bypassed by a normally closed valve in a bypass line.

The activity of gases passing up the vent stack is detected by a GM tube and recorded on radiation recorder No. RR-602. Limit contacts on the recorder actuate a solenoid valve in the exhaust line stopping the escape of gases from the decay tanks to the stack.

3.2.4.3 Testing

The pressure controller will be checked for ability to maintain the set pressure points. The safety shut off solenoid and radiation monitor will be tested for function by exposure of the GM tube to an artificial radiation source.

3.3 Collection System

The gas collected in the vent system falls into two basic categories; those normally expected to be radioactive, and those not normally radioactive, but which could be active under certain conditions.

3.3.1 Gases normally radioactive

The gases that are normally radioactive are piped directly to the suction tank for the compressors.

3.3.1.1 Reactor and primary fill tank atmosphere

A helium atmosphere is maintained in these two units as described in section 1, "Helium Systems," above. The atmosphere in these units is common. When sodium is transferred from one of these tanks to the other, the gas displaced from the receiving tank replaces the sodium in the discharging tank. Gas vented from the line connecting these two areas passes through a vapor trap (Equip. No. 73214A).

3.3.1.1.1 Control

The helium supply pressure control stations serving this atmosphere do not have pressure relief valves. All relieving is done by relief valves PC-600, RV-603, and RV-602 in the vent system. PC-600 will be set to relieve to the vent system if the pressure exceeds 4.5 psi. RV-603 is set to bypass PC-600 if the pressure exceeds 6.5 psi. Failure of RV-603 to open is backed up by spring loaded RV-602 which vents to the main sodium gallery when the pressure reaches 8.5 psi. RV-602 is installed to relieve to the gallery atmosphere on the premise that PC-600 and RV-603 could be prevented from relieving by failure of the vent gas compressor system.

3.3.1.1.2 Testing

The operation of RV-602 will be checked by setting PC-600 and RV-603 to relieve at higher than 8.5 psi and pressurizing the

atmosphere of the reactor and primary fill tank to 3.5 as indicated on EPI-468 and checking the relief. PV-603 will be checked by setting at 6.5 psi and observing that the relief through it is stopped when the atmosphere is reduced to 6.5 psi. PC-600 will be checked by setting at 4.5 psi and observing that the relief is stopped when the atmosphere reduces to 4.5 psi.

3.3.1.2 Helium vents from freeze traps

Vents from strategic locations in the sodium system where helium is fed for purging purposes are connected directly to the vent system. These vents are used only during the filling of the sodium system and are sealed by a gate valve and by frozen sodium seals under normal operating conditions. The operation of these traps as itemized below are covered under the description of the sodium systems.

a) Disposable Cold Trap
b) Main Primary Cold Trap
c) Main Primary Piping at the Main Intermediate Heat Exchanger
d) Auxiliary Primary Cold Trap
e) Auxiliary Primary Piping at the Auxiliary Primary Heat Exchanger

Gases vented from these freeze traps pass through the same control station as the reactor and primary fill tanks, as described in section 3.3.1.1.1 above.

3.3.1.2.1 Testing

The testing of these freeze seals is covered under section C.2, "Sodium Heat Transfer Systems."

3.3.1.3 Vents from cleaning cells

Each of the three fuel cleaning cells has two vent lines. One vent is through a condensate trap, and a vacuum pump. The other one

3.3.1.6 Liquid waste holdup tanks & liquid waste sump tanks

The holdup tanks vent through pressure relief valves. The four tanks in group A vent through RV-701 and the four in group B vent through RV-702.

The vent from the radioactive liquid waste sump tank is connected to the vent system at all times, unless isolated by a blocking valve at the sump tank.

3.3.1.6.1 Testing

Relief valves RV-701 and RV-702 will be checked for setting and operation at 10 psi.

3.3.1.7 Metallurgical hot cell

The decanning and washing compartment of the metallurgical hot cell is open to the vent system at all times.

3.3.2 Gases normally not radioactive

The gases normally not radioactive may be routed to the gas compressors or may be exhausted to the vent stack as described in section 3.1. These gases are collected from the sources enumerated below.

3.3.2.1 Galleries and vaults

These units normally containing a non-radioactive nitrogen atmosphere are as follows:

a) Main Primary Gallery
b) Auxiliary Primary Gallery
c) Primary Fill Tank Vault
d) Disposable Cold Trap Vault

The collection of the gases from these areas to a common header is described in section 2.3.

Gases passed from control station PC-522 pass through a radiation monitor station (BHRS-505). If the gases are radioactive, they are routed to the vent system compressor suction

tank. If they are not radioactive, they are routed directly to the building ventilation stack. The routing of these gases is effected by solenoid valves SV-501 in the compressor suction tank line and SV-500 in the stack line, as controlled from the monitor station BHRS-505.

3.3.2.1.1 Monitor station

The radioactivity of passing gases is detected by a Jordan ion chamber, and the output amplified to actuate relays to control the action of solenoid valves SV-500 and SV-501. The activity level is indicated on ERI-500 on panel GG in the control room.

3.3.2.1.1.1 Testing

The proper functioning of this monitor station will be checked by bringing a radioactive cource in proximity with the detector and observing the solenoid valve operation.

3.3.2.2 Reactor drain line to primary fill tank

The vent from this line is through a freeze trap in the primary fill tank vault. A bypass line around the compressor suction tank permits use of this line and the compressors to prime the reactor drain pump as described in section C. 1. "Sodium Service Systems."

The checkout of the freeze trap and its heaters is covered in the same section.

3.3.2.3 Helium pressure control stations

Gas exhausted from relief valves at various helium pressure control stations is collected into this system. The operation of the control stations as enumerated below are described in section I. 1,2 "Helium Systems."

PC-410, 413, 419, 421, 425, 427, 434, 436, 438, 446, 473

The cavity between the core tank and [illegible] tank is maintained at a minimum helium pressure of 3 psi and is relieved at 4.5 psi by pressure control station 473. A blocking valve in the vent line is normally closed. This valve is bypassed by a relief valve if the pressure exceeds 6 psi. The relief valve in the vent system (RV-600) functions only upon failure of the relief valve at the helium control station. There is a gas sampling valve in the vent line adjacent to the blocking valve.

3.3.2.3.1 Testing

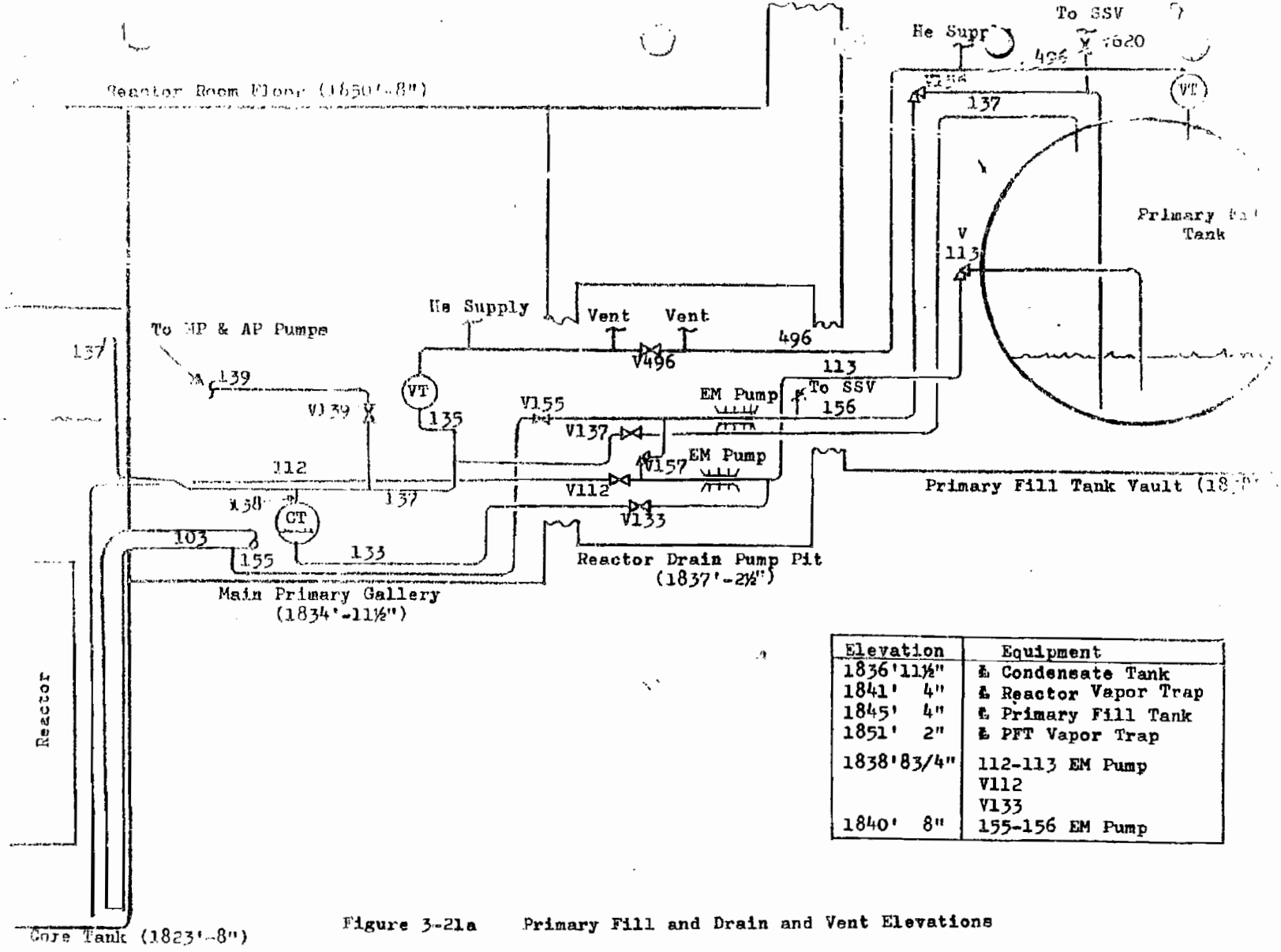
Relief valve RV-600 will be tested for setting and operation at 6 psi.

3.3.2.4 Nitrogen pressure control stations

Gas exhausted from relief valves at the nitrogen pressure control stations is collected into this system. The operation of these control stations as enumerated below is described in section 2.2 "Nitrogen System."

PC-503 Main Primary System Gallery
504 Primary Fill Tank Vault
505 Insulation Cavity
507 Auxiliary Primary System Gallery
511 Disposable Cold Trap Vault

Pressure control station PC-505 maintains a minimum nitrogen pressure of 3 psi in the insulation cavity. A relief valve at this station prevents the pressure from exceeding 4.5 psi. A blocking valve in the vent line is open only during specific operations such as purging. With the blocking valve normally closed, relief valve (RV-601) in the vent system is set to relieve at 6 psig, and will operate only as a safety backup for the relief valve at pressure station PC-505. A gas sampling valve permits sampling at the blocking valve. Pressure alarm



Elevation	Equipment
1836'11½"	& Condensate Tank
1841' 4"	& Reactor Vapor Trap
1845' 4"	& Primary Fill Tank
1851' 2"	& PFT Vapor Trap
1838'83/4"	112-113 EM Pump
	V112
	V133
1840' 8"	155-156 EM Pump

Figure 3-21a Primary Fill and Drain and Vent Elevations

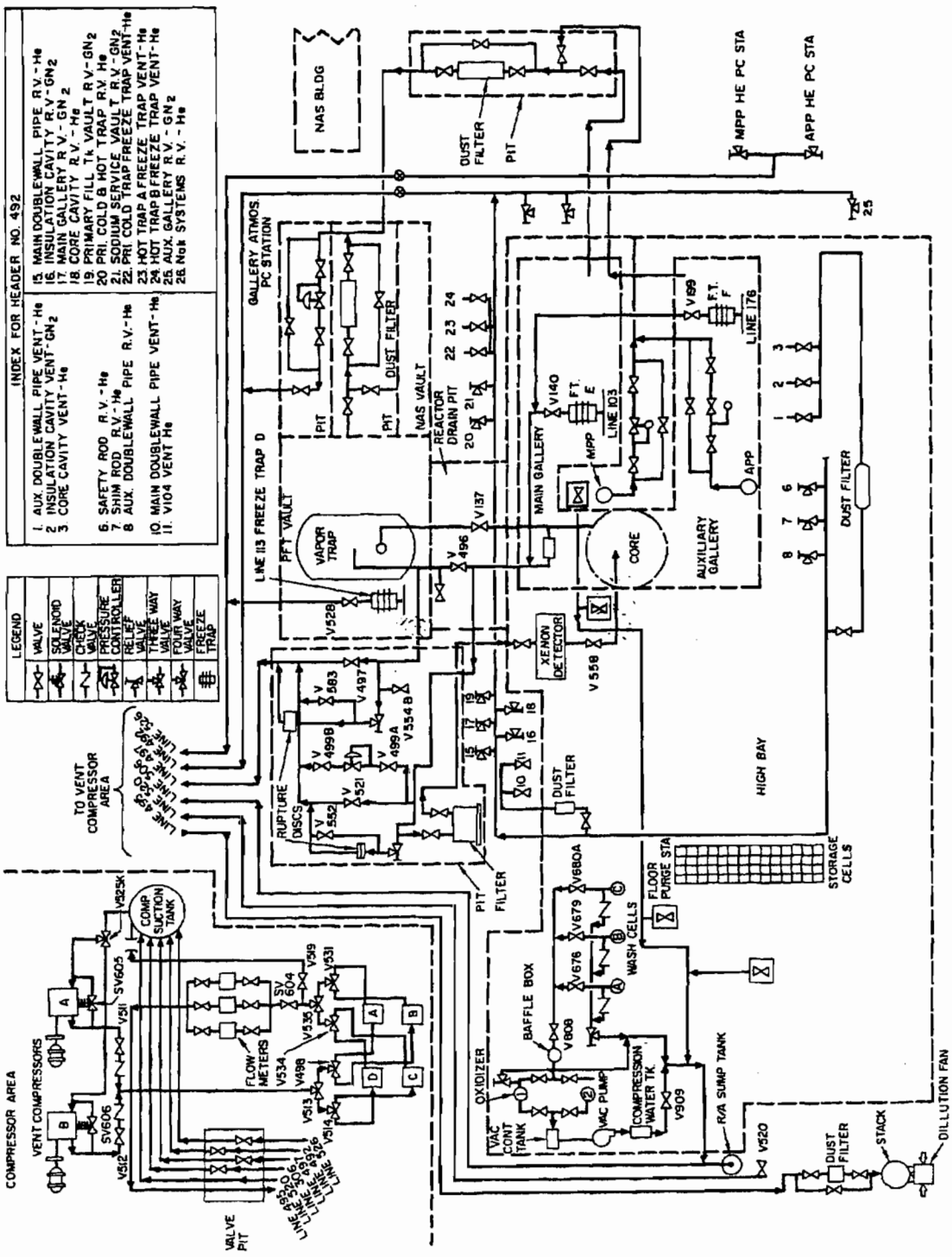


Figure 39. Radioactive Vent System

Appendix C

Holdup Tank Activities and Releases

W. L. Fisher W.L.F. 779 Santa Susana

G. Berg 779

74 Santa Susana 2 June 1959

Quarterly report of activity released to atmosphere.

The following report is a tabulation of radioactive gases released to the atmosphere from SRE hold-up tanks. This report covers the period of January 1 to April 1, 1959. The isotopes contained in the gas released generally consisted of Xe^{134}, Kr^{85}, Xe^{133} and mixed fission products.

Sample Date	Activity (μc/cc)	Total Activity (μc)	
1/1/59	2.0 x 10^{-7}	15 x 10^{0}	
1/2/59	2.5 x 10^{-5}	19 x 10^{2}	1.0×10^{3}
1/6/59	1.0 x 10^{-3}	76 x 10^{3}	7.6×10^{4} 7.8×10^{4}
1/6/59	2.4 x 10^{-5}	18 x 10^{2}	1.8×10^{3}
1/8/59	1.1 x 10^{-4}	8 x 10^{3}	
1/9/59	1.5 x 10^{-5}	11 x 10^{2}	1.1×10^{3}
1/14/59	3.7 x 10^{-3}	28 x 10^{3}	2.8×10^{4} } 3.4×10^{4}
1/14/59	4.0 x 10^{-5}	30 x 10^{2}	$3. \times 10^{3}$
1/15/59	3.0 x 10^{-5}	23 x 10^{2}	2.3×10^{3} } 6.1×10^{3}
1/15/59	5.0 x 10^{-5}	38 x 10^{2}	3.8×10^{3}
1/19/59	3.5 x 10^{-5}	27 x 10^{2}	2.7×10^{3} } 3.0×10^{3}
1/19/59	1.0 x 10^{-5}	86 x 10^{1}	2.6×10^{2}
1/20/59	3.0 x 10^{-5}	23 x 10^{2}	2.3×10^{3}
1/27/59	5.5 x 10^{-5}	42 x 10^{2}	4.2×10^{3} } 1.2×10^{4}
1/27/59	1.0 x 10^{-4}	76 x 10^{2}	7.6×10^{3}
2/4/59	7.8 x 10^{-5}	59 x 10^{2}	5.9×10^{3} } 1.7×10^{4}
2/4/59	5.0 x 10^{-5}	38 x 10^{2}	3.8×10^{3}
2/4/59	9.5 x 10^{-5}	72 x 10^{2}	7.2×10^{3}
2/6/59	4.0 x 10^{-4}	30 x 10^{3}	
2/7/59	2.4 x 10^{-4}	18 x 10^{3}	
2/9/59	8.0 x 10^{-5}	61 x 10^{2}	
2/10/59	3.5 x 10^{-5}	27 x 10^{2}	
2/12/59	1.4 x 10^{-4}	11 x 10^{3}	
2/13/59	5.0 x 10^{-5}	38 x 10^{2}	
2/15/59	3.3 x 10^{-5}	25 x 10^{2}	
2/17/59	3.5 x 10^{-5}	27 x 10^{2}	3.0×10^{3}
2/17/59	4.0 x 10^{-6}	30 x 10^{1}	
2/21/59	2.5 x 10^{-4}	19 x 10^{3}	
2/22/59	3.5 x 10^{-4}	27 x 10^{3}	
2/24/59	4.0 x 10^{-5}	30 x 10^{2}	
2/27/59	1.0 x 10^{-4}	76 x 10^{2}	

W. L. Fisher

From: G. Borg
2 June 1959

Date	Activity (μc/cc)	Total Activity (μc)	
3/4/59	2.5 x 10^{-5}	19 x 10^{2}	1.9×10^{3} } 1.6×10^{3}
3/5/59	1.8 x 10^{-4}	14 x 10^{3}	1.4×10^{3}
3/6/59	5.0 x 10^{-4}	38 x 10^{3}	3.8×10^{3}
3/8/59	5.0 x 10^{-4}	38 x 10^{3}	3.8×10^{3}
3/10/59	1.0 x 10^{-3}	76 x 10^{3}	7.6×10^{3}
3/11/59	6.0 x 10^{-3}	46 x 10^{3}	4.6×10^{3}
3/12/59	3.2 x 10^{-3}	24 x 10^{3}	2.4×10^{3} } 7.6×10^{3}
3/12/59	6.8 x 10^{-4}	52 x 10^{2}	5.2×10^{3}
3/13/59	6.5 x 10^{-4}	49 x 10^{2}	4.9×10^{3}
3/14/59	1.2 x 10^{-4}	12 x 10^{3}	1.2×10^{3}
3/17/59			
3/18/59	7.2 x 10^{-5}	55 x 10^{2}	5.5×10^{3}
3/22/59	1.8 x 10^{-4}	14 x 10^{2}	1.4×10^{3}
3/22/59	2.2 x 10^{-4}	17 x 10^{2}	17×10^{2} 1.7×10^{3} 3.1×10^{3}
3-23-59 3/23/59	1.8 x 10^{-4}	14 x 10^{2}	1.4×10^{3}
3/26/59	1.8 x 10^{-4}	14 x 10^{2}	1.4×10^{3}
3/27/59	2.2 x 10^{-4}	17 x 10^{2}	1.7×10^{3}
3/28/59	2.0 x 10^{-4}	15 x 10^{2}	1.5×10^{3} } 3.1×10^{3}
3/28/59	2.0 x 10^{-4}	15 x 10^{2}	1.5×10^{3}
3/31/59	Background	Background	

In each instance of venting the total volume released was approximately 7.6 x 10^{7} cc.

G. Borg
Health Physics Unit

GB:eb
cc: J. G. Long
Lois Bruhn
F.G. Bold
R. Wade

I Bruhn

ATOMICS INTERNATIONAL

A DIVISION OF NORTH AMERICAN AVIATION, INC.

INTER-OFFICE LETTERS ONLY

TO: W. L. Fisher NRD ADDRESS: 779

FROM: G. Borg ADDRESS: 779 SS

PHONE: X 74 DATE: 4 August 1959

SUBJECT: Quarterly report of activity released to atmosphere.

The following is a tabulation of radioactive gases released to the atmosphere from SRE hold-up tanks. This report encompases the period of April 1 to July 1, 1959. The isotopes contained in the gas released generally consisted of Na^{24}, Kr^{85}, Xe^{133} and mixed fission products.

Date 1959	Activity μc/cc	Total Activity μc		Release Rate CFM
4-2	4.0×10^{-7}	3.0×10^{1}		145
4-2	No detectable activity			145
4-4	" " "			145
4-5	" " "			145
4-11	3.0×10^{-3}	2.3×10^{5}	3.8×10^{5}	5
4-11	2.0×10^{-3}	1.5×10^{5}		100
4-22	3.5×10^{-5}	2.7×10^{3}	4.1×10^{3}	145
4-22	1.8×10^{-5}	1.4×10^{3}		145
4-24	5.0×10^{-5}	3.8×10^{3}	8.4×10^{3}	145
4-24	6.0×10^{-5}	4.6×10^{3}		145
4-27	3.0×10^{-4}	2.3×10^{4}		145
4-29	8.0×10^{-7}	6.1×10^{1}		145
5-1	3.5×10^{-6}	2.7×10^{2}		145
5-3	2.0×10^{-6}	1.5×10^{2}		145
5-4	2.0×10^{-6}	1.5×10^{2}		145
5-5	2.5×10^{-6}	1.9×10^{2}		145
5-11	4.0×10^{-6}	3.0×10^{2}		145
5-13	3.0×10^{-6}	2.3×10^{2}		145
5-14	4.0×10^{-6}	3.0×10^{2}		145
5-16	2.6×10^{-5}	2.0×10^{3}		145
5-18	0.5×10^{-4}	3.8×10^{3}	3.6×10^{4}	145
5-18	4.2×10^{-4}	3.2×10^{4}		145
5-20	7.5×10^{-5}	5.7×10^{3}		145
5-25	1.0×10^{-5}	7.6×10^{2}		145
5-26	3.5×10^{-5}	2.7×10^{3}		145
5-28	1.7×10^{-5}	1.3×10^{3}		145
5-29	3.2×10^{-6}	2.4×10^{2}		145
6-2	3.0×10^{-7}	2.3×10^{1}	3.8×10^{1}	145
6-2	2.0×10^{-7}	1.5×10^{1}		145
6-3	3.0×10^{-7}	2.3×10^{1}		145
6-4	2.5×10^{-6}	1.9×10^{2}	2.6×10^{4}	145
6-4	3.4×10^{-4}	2.6×10^{4}		60

N 6-R-5

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INTER-OFFICE LETTERS ONLY

TO: W. L. Fisher hxf ADDRESS: 779 88

FROM: G. Borg ADDRESS: 779 88

PHONE: X 74 DATE: November 20, 1959

SUBJECT: Quarterly report of activity released to atmosphere.

The following report is a tabulation of radioactive gases released to the atmosphere from SRE decay hold-up tanks. This report covers the period from July 1, 1959 to October 1, 1959.

Sample date	Activity μc/cc	Release Rate (CF/Hr)	Total Activity μc **
7/1/59	5.0×10^{-6}	145	3.8×10^{2}
7/1/59	7.0×10^{-6}	145	5.3×10^{2}
7/3/59	2.5×10^{-6}	145	1.9×10^{2}
7/9/59	2.0×10^{-6}	145	1.5×10^{2}
7/10/59	1.8×10^{-6}	145	1.4×10^{2}
7/11/59	2.2×10^{-6}	145	1.7×10^{2}
7/20/59	5.0×10^{-2}	4	3.8×10^{6}
7/25/59	1.4×10^{-2}	27	1.1×10^{6}
*7/25/59	8.0×10^{-2}	6	6.1×10^{6}
8/22/59	7.0×10^{-2}	4	5.3×10^{6}
9/16/59	6.5×10^{-2}	4	4.9×10^{6}
9/17/59	7.0×10^{-2}	12	5.3×10^{6}
9/18/59	2.0×10^{-3}	145	1.5×10^{5}
9/20/59	5.0×10^{-3}	145	3.8×10^{5}
9/21/59	1.3×10^{-3}	65	1.0×10^{5}
9/21/59	1.3×10^{-3}	145	1.0×10^{5}
9/21/59	1.1×10^{-3}	145	8.4×10^{4}
9/22/59	2.0×10^{-3}	145	1.5×10^{5}
9/27/59	1.8×10^{-3}	145	1.4×10^{5}
9/27/59	1.8×10^{-3}	145	1.4×10^{5}
9/28/59	1.8×10^{-3}	145	1.4×10^{5}
9/30/59	7.8×10^{-6}	15	5.8×10^{2}
9/30/59	6.0×10^{-6}	15	4.6×10^{2}

*Reason for gap between 7/25/59 to 8/22/59 to 9/16/59 is due to slow release rate.

**In each instance of venting the total volume released was approximately 7.6×10^{7} cc.

G. Borg

G. Borg
Health Physics Department - 88

GB:eb
cc: Lang, J. C.

FORM N 6-R-7 (R. 1-59)

ATOMICS INTERNATIONAL

A DIVISION OF NORTH AMERICAN AVIATION, INC.

INTER-OFFICE LETTERS ONLY

TO: W. L. Fisher ADDRESS 779 88

FROM: G. Borg ADDRESS 779 88

PHONE: 74 DATE: February 12, 1960

SUBJECT: Quarterly report of activity released to atmosphere.

The following report is a tabulation of radioactive gases released to the atmosphere from HRE hold-up tanks. This report covers the period from October 1, 1959 to December 31, 1959. The isotopes contained in the gas released generally consisted of [redacted] and mixed fission products.

Sample Date	Release(1) Rate in cfh	Activity uc/cc	Total Activity uc
10/1/59	145	1.5 x 10^{-3}	1.1 x 10^{3}
10/1/59	"	2.5 x 10^{-3}	1.9 x 10^{3}
10/2/59	900(2)	2.2 x 10^{-3}	1.7 x 10^{3}
10/2/59	"	2.4 x 10^{-3}	1.8 x 10^{3}
10/4/59	"	5.0 x 10^{-3}	3.8 x 10^{3}
10/5/59	"	4.6 x 10^{-3}	3.5 x 10^{3}
10/8/59	"	1.0 x 10^{-3}	7.6 x 10^{2}
10/11/59	"	2.3 x 10^{-3}	1.7 x 10^{3}
10/14/59	"	4.2 x 10^{-3}	3.2 x 10^{3}
10/20/59	"	2.3 x 10^{-3}	1.8 x 10^{3}
10/26/59	"	6.0 x 10^{-3}	4.6 x 10^{3}
10/26/59	"	3.0 x 10^{-3}	2.5 x 10^{3}
10/29/59	"	5.0 x 10^{-3}	3.8 x 10^{3}
10/29/59	"	2.5 x 10^{-3}	1.9 x 10^{3}
10/29/59	"	6.5 x 10^{-3}	4.9 x 10^{3}
10/29/59	"	6.5 x 10^{-3}	4.9 x 10^{3}
10/30/59	"	7.2 x 10^{-3}	5.5 x 10^{3}
10/30/59	"	4.0 x 10^{-3}	3.0 x 10^{3}
10/30/59	"	2.5 x 10^{-3}	1.9 x 10^{3}
11/2/59	"	2.0 x 10^{-3}	1.5 x 10^{3}
11/2/59	"	1.2 x 10^{-3}	9.0 x 10^{2}
11/5/59	"	6.0 x 10^{-3}	4.6 x 10^{3}
11/5/59	"	4.5 x 10^{-3}	3.4 x 10^{3}
11/8/59	"	7.0 x 10^{-3}	5.3 x 10^{3}
11/8/59	"	6.8 x 10^{-3}	5.2 x 10^{3}
11/13/59	"	6.0 x 10^{-3}	4.6 x 10^{3}
11/13/59	"	6.0 x 10^{-3}	4.6 x 10^{3}
11/19/59	"	2.1 x 10^{-3}	1.6 x 10^{3}
11/19/59	"	5.0 x 10^{-3}	3.8 x 10^{3}
12/7/59	"	2.0 x 10^{-3}	1.5 x 10^{3}
12/18/59	"	1.1 x 10^{-3}	8.0 x 10^{2}
12/21/59	"	3.8 x 10^{-3}	2.9 x 10^{3}
12/23/59	"	6.5 x 10^{-3}	4.9 x 10^{3}
12/23/59	"	7.2 x 10^{-3}	5.5 x 10^{3}

(1) Total volume approximately 7.6 x 10^{7} cc/release

(2) New release rate valve installed.

L. Bruhn

N 6-R-7

G. Borg

G. Borg
Health Physics Department - 88

Appendix D

Reactor Startup Checklist

Appendix E

Description of Damaged Fuel Assemblies

Core Channel	Description of Damage
R-10	At 2200, July 27, 1959, it was found that the fuel cluster was broken in two with approximately two-thirds of the fuel cluster remaining in the fuel Channel . Shield plug and broken section of feel cluster were stored In Storage Cell 69.
R-12	At 1743, August 2, 1959, It was found that the fuel cluster was broken in two with approximately two-thirds of the fuel cluster remaining in the fuel channel. Shield plug and broken section of fuel cluster are contained within the fuel transfer cask awaiting transfer to a storage cell.
R-21	At 2245, October 10, 1959, an unsuccessful attempt was made to remove the fuel cluster from R-21 . The hoist cable power was tripped because of overload~ 800 lbs. The fuel cluster was finally withdrawn from the core at 2330 . Observations made in the SRE hot cell indicated that the lower third of the fuel cluster had broken off and had probably remained in core channel R-21 . The portion of cluster removed has been canned and is being stored.
R-23	At 1413, October 11, 1939, the fuel element from R-23 was removed from the core . It was noted during removal operations that the element stuck momentarily after 4 ft of upward travel . The element was broken, with the lower third believed to be still In R-23 . The portion of cluster removed has been canned and is being stored.
R-24	Attempts to remove the fuel cluster from R-24 had failed . This cluster, with its moderator can, was raised and blocked-up approximately 1 in. in an attempt to free the fuel cluster from the moderator can . No change was noted
R-25	At 1915, July 16, 1959, the element was being viewed by using the portable television camera when it was noted that the cladding appeared to be split open on one of the fuel rods . The element was lowered back in the core, rotated 180°, and viewed again showing an additional ruptured fuel rod. Element was stored in storage cell 56.
R-31	At 0200, July 27, 1939, it was found that the fuel cluster was broken in two with approximately one -half of the fuel remaining in the fuel channel . Shield plug and broken section of fuel cluster were stored in storage cell 35 .
R-35	At 1700, July 27, 1959, it was found that the fuel cluster was broken in two with approximately one-half of the fuel remaining in the fuel channel. Shield plug and broken section of fuel cluster were stored in storage cell 68.
R-43	At 1903, October 15, 1959, the fuel element from R-43 was removed from the core. The fuel cluster was broken in half. The portion of cluster removed has been canned end is being stored.
R-55	At 1900, July 20, 1959, the experimental fuel element was found to be broken in two with approximately nee-half of the cluster remaining in the fuel channel . Broken section of fuel cluster was examined and photographed In the hot cell, then the remains of the cluster were removed from the shield plug section end placed in a container and left in the hot cell. The shield plug was placed in storage Call 73 .
R-68	At 2300, August l, 1959, it was found that the fuel cluster was broken in two with approximately two-thirds of the fuel cluster remaining in the fuel channel . Shield plug and broken section of fuel cluster were stored in storage cell 72.
R-69	At 2230, October 14, 1959, the feel element from R-69 were withdrawn from the reactor . The cluster was broken in half. The portion of cluster removed has been canned and is being stored .
R-76	Attempts to remove the fuel cluster from R-76 have failed. In-core observations of to cluster, during removal attempts, Indicate that the moderator can containing the cluster lifts as the element is raised. This cluster, with its moderator can, was raised and blocked-up approximately 1 in. in an attempt to free the fuel cluster from the moderator can. No change was noted.

Appendix F

Environmental Air Sample Data

Santa Susana & Van Owen Sample Points

PREPARED BY	ATOMICS INTERNATIONAL A Division of North American Aviation, Inc.	PAGE OF
CHECKED BY		REPORT NO.
DATE Jan-Feb 1959	SANTA SUSANA	MODEL NO.

CALIBRATED 30 Jan 59

Source: Be 210

Sch: 1.83

R E6 X 3.26 X 10^{-14} = mr/cc X 10^{-14}

DATE	BKG (hrs)	1ST. REGISTER	24TH REGISTER	55TH REGISTER	COMPLETE REGISTERS	mr/cc X 10^{-14}			
26 Jan 59	45	23	12	11	10	55.4			
27 Jan	45	60	28	25	22	71.7			
28 Jan	45	107	60	50	40	130.4			
29 Jan	46	179	103	100	97	316.2			
30 Jan	45	143	86	79	72	234.7			
31 Jan	46	70	42	36	30	97.8			
1 Feb	46	104	51	44	37	121.0			
2 Feb	46	90	42	37	32	104.5			
3 Feb	46	114	73	59	45	146.9			
4 Feb	46	154	62	53	44	143.5			
5 Feb	46	172	80	67	54	176.1			
6 Feb	46	113	76	68	60	195.8			
7 Feb	46	95	50	42	34	110.8			
8 Feb	46	76	40	34	28	91.2			
9 Feb	46	62	34	28	22	71.2			
10 Feb	46	28	19	19	19	61.9			
11 Feb	44	31	24	22	20	65.2			
12 Feb	44	63	29	29	29	94.5			
13 Feb	45	80	38	37	36	117.2			
14 Feb	45	120	69	61	53	173.0			
15 Feb	44	118	102	100	98	320.0			
16 Feb	44	23	10	9	8	26.1			
17 Feb	44	169	33	25	17	55.5			
18 Feb	45	73	40	29	18	58.6			
19 Feb	46	92	34	20	6	19.6			
20 Feb	44	89	40	36	32	104.4			
21 Feb	44	107	50	47	44	143.4			

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BNA02409993

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DATE		MODEL NO.

Rate X 3.26 = ~~c/cc~~ X 10^{-14}

DATE	Bg	1st	24th	55%	LL.	~~c/cc~~ X 10^{-14}			
22 Feb	43	73	40	33	26	85.0			
23 Feb	43	96	52	40	28	91.4			
24 Feb	43	94	64	50	36	117.5			
25 Feb	44	142	107	103	99	322.2			
26 Feb	45	177	100	98	96	313.0			
27 Feb									
28 Feb						—			
1 MAR									
2 Mar	47	96	69	58	47	153.0			
3 Mar	47	93	72	64	56	183.0			
4 Mar									
5 Mar									
6 Mar									
7 Mar									
8 Mar									
9 Mar									
10 Mar	48	54	24	16	8	25.1			
11 Mar	48	125	98	95	92	300.0			
12 Mar	48	10	—	—					
13 Mar	47	131	82	67	52	170.0			
14 Mar	44	144	112	98	84	274.0			
15 Mar	44	120	92	87	82	267.0			
16 Mar	45	135	78	70	62	202.0			
17 Mar	45	169	92	89	86	280.0			
18 Mar	45	155	81	69	57	186.0			
19 Mar	46	157	78	73	68	222.0			
20 Mar	46	167	100	87	74	241.0			
21 Mar	45	182	110	98	86	280.0			
22 Mar	44	80	62	57	52	219.0			4.21
23 Mar	48	33	18	9	0	0			

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DATE March – April 1959	AIR MONITOR Santa Susana	MODEL NO.

Date	Bg	1st Rdg	24 Rdg	55 Rdg	L.L. Rdg	uc/cc X10^{-14}		Conversion
24 MAR	46	108	46	33	20	84.0		Rdg X 4.2 X10^{-14} curies/cc
25 MAR		INSTRUMENT TROUBLE						
26 MAR	44	111	65	46	27	113.5		
27 MAR		NO DATA DUE TO INSTRUMENT MALFUNCTION						
28 MAR								
29 MAR	44	248	160	137	114 57	240.0 ✓		
30 MAR		NO DATA DUE TO MONITOR MALFUNCTION						
31 MAR	45	129	101	84	67	282.0		
1 APR 59	45	98	83	63	43	181.0		
2 APR	45	120	82	75	68	286.0		
3 APR		NO DATA DUE TO MONITOR MALFUNCTION						
4 APR	45	205	141	127	113 68	236.0		
5 APR		NO DATA						
6 APRIL	45	63	34	25	17	71.6		
7 APRIL	45	117	49	38	27	113.6		
8 APR	47	109	60	49	38	160.0		
9 APR	49	185	76	61	46	194.0		
10 APR		NO DATA						
11 APR	46	204	125	117	109 54	227.0	$\frac{109 + 44}{2}$ =	1 day Concentration 10 + 11 APRIL
12 APR	46	90	60	52	44	185.0		
13 APR	47	86	61	56	51	214.2		
14 APR	46	114	83	67	51	214.2		
15 APR	45	145	78	70	62	261.0		
16 APR	45	115	86	79	72	303.0		
17 APR	46	85	68	60	52	219.0		
18 APR	46	117	65	56	47	198.0		
19 APR	45	80	70	58	46	193.6		
20 APR	45	128	68	57	46	193.6		
21 APR	45	102	61	50	39	164.1		
22 APR	45	120	65	57	49	206.0		

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DATE	AIR MONITOR Santa Susana	MODEL NO.

Date	Bg	1st Req	24 hr Req	55 hr Req	LL. Req	$\mu c/cc \times 10^{-14}$			CONSTANT
23 APR	45	90	70	54	38	160.0			4.21
24 APR	45	87	44	40	36	151.6			
25 APR	44	34	26	20	14	58.9			
26 APR	44	47	24	22	20	84.3			
27 APR	44	92	48	47	46	193.6			
28 APR									
29 APR									
30 APR	44	203	167	152	137/3=46 Lab.	193.6		72 hour	Sample
1 MAY	45	48	36	28	20	84.3			
2 MAY	45	59	39	29	19	80.0			
3 MAY	45	73	38	33	28	118.0			
4 MAY	44	67	41	33	25	105.2			
5 MAY	44	95	51	42	33	139.0			
6 MAY	45	91	53	46	39	164.1			
7 MAY	46	83	56	52	48	202.0			
8 MAY	45	117	48	42	36	151.6			
9 MAY	44	69	45	38	31	130.5			
10 MAY	44	102	59	46	33	139.0			
11 MAY	44	109	59	46	33	139.0			
12 MAY	44	75	59	46	33	139.0			
13 MAY	44	81	65	61	57	240.0			
14 MAY	44	85	55	40	25	105.0			
15 MAY	43	135	75	70	65	274.0			
16 MAY	43	122	79	67	55	231.0			
17 MAY	No DATA	Due to	Instrument		malfunction				
18 MAY		"		"		"			
19 MAY	44	194	159	132	105	246.0			
20 MAY	46	105	65	56	47	330.0			
21 MAY		No DATA	Due to	Instrument		Mal function			
22 MAY		"		"		"			

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CHECKED BY		REPORT NO.
DATE YEAR: 1959	AIR MONITOR DATA SANTA SUSANA	MODEL NO.

Date	Bq	$\frac{137}{Bq}$	$\frac{24\ hr}{Bq}$	$\frac{55\ hr}{Bq}$	L.L. $\frac{}{Bq}$	$\mu c/cc \times 10^{-14}$			Corr. Factor
23 May 59	44	144	128	124	120	842.)	281.0 $\times 10^{-14}$	$\mu c/cc$	7.02
24 May	44	76	48	41	34	239.0			
25 May	44	88	42	31	20	140.5			
26 May	45	53	36	30	24	168.5			
27 May	43	67	30	21	12	84.4			
28 May	43	57	39	31	23	161.5			
29 May	45	63	25	25	25	175.6			
30 May	43	56	36	28	20	140.5			
31 May	44	88	34	30	26	182.5			
1 June	45	120	48	37	26	182.5			
2 June	46	86	56	41	26	182.5			
3 June	46	96	50	40	30	211.0			
4 June	45	95	52	39	26	182.5			
5 June	NO Data due to Instrument Mal-function								
6 June	45	107	62	45	28	196.5			
7 June	43	55	29	24	19	133.2			
8 June	43	79	39	30	21	147.5			
9 June	46	79	26	22	18	126.2			
10 June	45	84	34	27	20	140.5			
11 June	45	86	46	33	20	140.5			
12 June	46	141	50	30	10	70.25			
13 June	47	130	45	30	15	105.2			
14 June	NO DATA DUE TO INSTRUMENT MAL-FUNCTION								
15 June	47	103	51	43	35/18	126.4			
16 June	46	89	33	23	13	91.4			
17 June	43	95	35	30	25	176.0			
18 June	43	95	32	31	30	211.0			
19 June	43	123	44	28	12	84.4			
20 June	42	54	35	24	13	91.4			
21 June	42	129	49	27	5	35.1			

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DATE	Air Monitor S.S.	MODEL NO.

Date	B9	1st R89	24th R89	55th R89	L.L. R89	μc/cc x10⁻¹⁴			Conv. Factor
22 June									7.02
23 June									
24 June									
25 June	44	101	51	32	13	91.2			
26 June	43	52	26	23	20	140.5			
27 June	43	89	14	10	6	42.1			
28 June	44	57	17	6	—	NO L.L. ACTIVITY DETECTED			
29 June	45	96	29	20	11	77.1			
30 June	45	72	27	17	7	49.1			
1 July	44	90	32	20	8	56.1			
2 July	44	38	23	9	—	NO L.L. ACTIVITY DETECTED			
3 July	44	59	20	9	—	NO. L.L. ACTIVITY DETECTED			
4 July		NO Data							
5 July	40	72	40	17	—				
6 July	40	66	28	18	8	56.2			
7 July		NO Data							
8 July		NO Data							
9 July		NO Data							
10 July	41	61	17	11	5	35.1			
11 July	40	110	29	20	11	77.2			
12 July									
13 July	41	64	30	21	12	84.4			
14 July	42	62	32	20	8	56.2			
15 July	47	91	29	16	3	21.1			
16 July	44	98	41	20	—				
17 July	40	76	42	23	4	28.1			
18 July									
19 July									
20 July									
21 July	45	82	24	19	14	98.4			

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CHECKED BY		REPORT NO.
DATE 27 July 59	Air Monitor S.S.	MODEL NO.

Date	Bg	1st Reg	24th Reg	55th Reg	L.L. Reg	$\mu c/cc \times 10^{-14}$			Conv. factor
22 July 59	45	117	41	23	5	35.1			7.02
23 July 59	45	118	34	21	8	56.1			
24 July 59									
25 July 59	44	87	31	19	7	49.1			
26 July	43	80	37	24	11	77.4			
27 July	45	64	31	19	7	49.1			
28 July	No	Data							
29 July	44	139	34	18	2	14.0			
30 July	No	Data							
31 July	No	Data							
1 Aug	48	150	31	20	9/2	31.6			
2 Aug	No	Data							
3 Aug	No	Data							
4 Aug	46	218	50	38	26/2	13 91.4			
5 Aug	43	105	32	19	6	42.1			
6 Aug	45	100	21	16	11	77.2			
7 Aug	45	93	18	11	4	28.1			
8 Aug	45	185	24	16	8	56.2			
9 Aug	45	147	43	22	1	7.02			
10 Aug	45	287	2	0	0	0			
11 Aug	45	78	15	11	7	49.1			
12 Aug	41	45	21	13	5	35.1			
13 Aug	49	205	34	14	0	0			
14 Aug	46	182	0	0	0	0			
15 Aug	46	68	16	11	6	42.1			
16 Aug	No	Data							
17 Aug	44	33	14	10	6	42.1			
18 Aug	No	Data due to		instrument		maintenance			
19 Aug	"			"		"			
20 Aug	"			"		"			

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BNA02409999

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John Moore	A Division of North American Aviation, Inc.	OF
CHECKED BY		REPORT NO.
DATE		MODEL NO.
27 August 1959	Air Monitor SS.	

Date	BKG	1ST Reg	3&4 Reg	5S Reg	L.L. Reg	uuc/cc-4 x10-4			CONO. factor
21 Aug	No	Data	Due	To	Instrument	Maintenance			7.02
22 Aug	43	39	18	10	2	14.05			
23 Aug	43	63	22	15	8	56.1			
24 Aug	46	53	10	6	2	14.05			
25 Aug	46	61	22	13	4	28.00			
26 Aug	43	71	20	11	2	14.05			
27 Aug	43	98	22	13	4	28.0			
28 Aug	44	126	30	16	2	14.05			
29 Aug	44	108	29	16	3	21.10			
30 Aug	46	76	22	11	0	—			
31 Aug	45	62	19	10	1	7.02			
1 Sept	44	85	21	13	5	35.1			
2 Sept	44	88	21	12	3	21.1			
3 Sept	43	88	28	16	4	28.0			
4 Sept	43	86	26	14	2	14.05			
5 Sept	41	79	24	13	2	14.05			
6 Sept	42	104	29	17	5	35.1			
7 Sept		No Data							
8 Sept		No Data							
9 Sept	41	95	33	20	7	49.1			
10 Sept	42	96	29	17	5	35.1			
11 Sept	44	81	22	12	2	14.05			
12 Sept	43	92	26	14	2	14.05			
13 Sept	45	109	31	16	1	7.02			
14 Sept	44	54	20	10	0	N.A.D			
15 Sept	43	57	15	10	5	35.1			
16 Sept	43	47	14	7	0	N.A.D			
17 Sept	42	32	13	7	1	7.02			
18 Sept	42	15	10	6	2	14.05			
19 Sept	43	15	7	3	0	N.A.D			

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J. Moore	A Division of North American Aviation, Inc.	OF
CHECKED BY		REPORT NO.
DATE		MODEL NO.
29 SEPT 1959	AIR MONITOR SS	

* = INTERPOLATOR ON 8

Date	BKG	1ST REG	2496 REG	5594 REG	L.L. REG	mc/cc $\times 10^{-14}$			CONV. FACTOR
20 SEPT	44	67	16	6	0	N.A.D.			7.02
21 SEPT	44	51	18	12	6	42.12			
22 SEPT	46	72	18	7	0	N.A.D.			
23 SEPT	47	56	16	6	0	N.A.D.			
24 SEPT	48	104	29	15	1	7.02			
25 SEPT	43	122	34	18	2	14.04			
26 SEPT	46	129	35	19	3	21.06			
27 SEPT	46	70	17	10	3	21.06			
28 SEPT	46	80	24	13	2	14.04			
29 SEPT	48	94	23	13	3	21.06			
30 SEPT	91	100	26	14	2	7.02	*	3.51	
1 OCT	88	164	39	22	5	17.55	*		
2 OCT	90	206	38	24	10	35.1	*		
3 OCT	90	189	26	17	8	28.08	*		
4 OCT	88	250	54	32	10	35.1	*		
5 OCT	97	224	36	19	2	7.02	*		
6 OCT	90	190	55	30	5	17.55	*		
7 OCT	90	240	52	37	22	77.2	*		
8 OCT	88	278	76	41	6	21.06	*		
9 OCT	88	248	63	37	11	38.6	*		
10 OCT	89	293	78	43	8	28.1	*		
11 OCT	93	449	88	51	14	49.1	*		
12 OCT	92	387	103	59	15	52.6	*		
13 OCT	91	67	25	18	11	38.6	*		
14 OCT	93	185	28	NO DATA DUE TO GTZ SETTER-AMET FAILURE			*		
15 OCT	95		41	21	1	3.51	*		
16 OCT	95	296	60	40	20	70.2	*		
17 OCT	94	300	71	41	11	38.6	*		
18 OCT	93	320	82	48	14	49.1	*		
19 OCT	97	195	58	30	2	7.02	*		

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BNA02410001

PREPARED BY	ATOMICS INTERNATIONAL	PAGE
John Moore	A Division of North American Aviation, Inc.	OF
CHECKED BY		REPORT NO.
DATE 26 October	[redacted] Santa Susana 24 Hour Monitor	MODEL NO.

* = 3.51

DATE	Bq	1st REG	24th REG	55th REG	L.L. REG	mc/cc X10⁻¹⁴			CONV. FACTOR
20 OCT	98	312	61	31	1	3.51	*		3:1
21 OCT	94	165	50	32	14	48.2	*		
22 OCT	93	158	30	22	14	48.2	*		
23 OCT	93	125	27	17	7	24.6	*		
24 OCT	91	263	79	45	11	38.6	*		
25 OCT	88	220	72	45	18	63.2	*		
26 OCT	89	162	52	27	2	7.02	*		
27 OCT	91	294	64	33	2	7.02	*		
28 OCT	95	286	64	38	12	42.1	*		
29 OCT	92	114	34	20	6	21.1	*		
30 OCT	96	155	35	19	3	10.53	*		
31 OCT	98	130	40	21	2	7.02	*		
1 NOV	98	166	39	23	7	24.6	*		
2 NOV	97	190	39	24	9	31.6	*		
3 NOV	95	167	38	20	2	7.02	*		
4 NOV	93	280	58	33	8	28.1	*		
5 NOV	95	51	11	8	5	17.55	*		
6 NOV	96	59	15	10	5	17.55	*		
7 NOV	94	80	16	10	4	14.05	*		
8 NOV	95	107	20	6	<LL	N.A.D.	*		
9 NOV	97	166	29	10	<LL	N.A.D.	*		
10 NOV	94	223	45	25	5	17.55	*		
11 NOV	90	259	58	30	2	7.02	*		
12 NOV	90	220	54	28	2	7.02	*		
13 NOV	95	246	64	36	8	28.1	*		
14 NOV	90	271	64	35	6	21.1	*		
15 NOV	94	197	47	24	1	3.51	*		
16 NOV	48	134	29	15	1	7.02		10TH TO 16 CTS/MIN	POLAROID RESET
17 NOV	44	141	34	18	2	14.04			
18 NOV	44	92	30	18	6	42.10			

738-V-28 NOV NO DATA

PREPARED BY	ATOMICS INTERNATIONAL	PAGE OF
CHECKED BY	A Division of North American Aviation, Inc.	REPORT NO.
DATE	Santa Susana Air Monitor	MODEL NO.

DATE	Bg	1st Reg	249k Reg	555k Reg	L.L. Reg	μc/cc E-10			
19 Nov	46	144	31	16	1	7.02			8.2
20 Nov	45	83	13	8	3	21.10			
21 Nov	44	85	19	10	1	7.02			
22 Nov	43	28	9	7	5	41.00			
23 Nov	42	37	14	9	4	32.8			
24 Nov	44	94	20	13	6	49.2			
25 Nov	45	72	21	11	1	8.2			
26 Nov	44	29	9	5	1	8.2			
27 Nov	41	29	13	8	3	24.6			
28 Nov	41	41	11	8	5	41.0			
29 Nov	47	40	4	3	2	16.4			
30 Nov	46	58	10	7	4	32.8			
1 DEC	47	56	9	6	3	24.6			
2 DEC	48	87	14	8	2	16.4			
3 DEC	46	115	24	13	2	16.4			
4 DEC	46	56	16	9	2	16.4			
5 DEC	48	15	3	2	1	8.2			
6 DEC	47	34	5	3	1	8.2			
7 DEC	47	84	13	7	1	8.2			
8 DEC	No	DATA	To	12-11-59	due	to Vacuum	pump	repairs	
9 DEC									
10 DEC									
11 DEC	45	107	18	11	4	32.8			
12 DEC	45	65	15	8	1	8.2			
13 DEC	45	46	8	5	2	16.4			
14 DEC	44	34	11	6	1	8.2			
15 DEC	46	41	7	6	5	16.4			
16 DEC	45	106	25	13	1	8.2			
17 DEC	44	101	24	15	6	49.2			
18 DEC	45	82	23	13	3	24.6			

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BNA02410003

PREPARED BY	ATOMICS INTERNATIONAL	PAGE
CHECKED BY	A Division of North American Aviation, Inc.	OF / REPORT NO.
DATE: 22 DEC 59	Santa Susana Automatic Air Monitor	MODEL NO.

DATE	By	1st Beg	24hr Beg	55hr Beg	L.L. Beg	mc/cc x10⁻¹⁴	Remarks
19 DEC	45	69	16	11	6	49.2	
	No DATA	to	31 DEC	Due	to	Streeter - Amet failure	
31 DEC	47	48	7	4	1	8.2	
1 Jan 60	45	60	10	9	8	65.5	
2 Jan	47	64	9	6	3	24.6	
	No	DATA	TO	22	Jan 60	Due To	Streeter - Amet failure 8.2x10⁻¹⁴
22 Jan	46	84	17	9	1	8.2	
23 Jan	47	63	14	8	2	16.4	
24 Jan	46	66	16	9	2	16.4	
25 Jan							
26 Jan	46	20	2	2	2	16.4	
27 Jan	44	56	11	6	1	8.2	
28 Jan	45	51	15	8	1	8.2	
29 Jan	47	56	13	7	1	8.2	
30 Jan	47	68	14	7	1	8.2	
	Streeter - Amet printer failure resulted						
	in a Loss of data to					21 March 1960	
21 March 60	45	342	12	2	—	0	
22 Mar	37						
23 Mar	37	42	10	4	—	0	
24 Mar	37	47	14	—		0	
25 Mar	36	88	20	12	4	32.8	
26 Mar	37	103	23	13	3	24.6	
27 Mar	34	62	17	8	—	N.A.D.	
28 Mar	35	16	5	3	1	8.20	
29 Mar	35	57	14	8	2	16.40	
30 Mar	34	96	20	11	2	16.40	
31 Mar	35	29	11	7	3	24.60	
1 APRIL	36	18	4	3	2	16.40	
2 APR	35	19	6	5	4	32.8	

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PREPARED BY	ATOMICS INTERNATIONAL A Division of North American Aviation, Inc.	PAGE OF
CHECKED BY		REPORT NO.
DATE 1959		MODEL NO. V0

$Rx\ 4.55 \times 10^{-14} = mc/cc \times 10^{-14}$

$ESS = 2.87 - .348\%$

Date	KS	1ST	24th	55Lcs	4Lws	mc/cc x10^{-14}			
24 Dec									
[illegible]									
[illegible]									
[illegible]									
[illegible]	35	—	69	60	51	234.0	2.34	x10^{-12}	
[illegible]	35	118	78	76	74	339.6	3.39	x10^{-12}	
30 Dec	33	147	97	90	83	381.0	3.81	x10^{-12}	
31 Dec	34	141	67	55	76	349.0	3.49	x10^{-12}	
1 Jan 1959	36	1907	450	259	68	311.1	3.11	x10^{-12}	
2 Jan	37	1576	357	205	53	243.2	2.43	x10^{-12}	
3 Jan	36	1976	474	287	100	459.0	4.59	x10^{-12}	
4 Jan	35	2375	520	316	112	514.0	5.14	x10^{-12}	
5 Jan	33	375	258	168	78	358.0	3.58	x10^{-12}	
6 Jan	34	561	77	53	29	133.0	1.33	x10^{-12}	
7 Jan	35	743	101	59	17	78.0	7.80	x10^{-13}	
8 Jan	34	613	135	93	51	234.0	2.34	x10^{-12}	
9 Jan	35	231	113	80	47	215.7	2.15	x10^{-12}	
10 Jan	35	862	130	75	20	91.8	9.18	x10^{-13}	
11 Jan	35	1105	212	122	32	146.8	1.46	x10^{-12}	
12 Jan	35	694	183	123	63	289.1	2.89	x10^{-12}	
13 Jan	35	909	158	101	44	201.9	2.01	x10^{-12}	
14 Jan	35	884	178	119	60	275.4	2.75	x10^{-12}	
15 Jan	35	910							
16 Jan	35	1132	181	120	59	451.3	4.51	x10^{-12}	14h 20L
17 Jan	35	1467	336	222	108	495.7	4.95	x10^{-12}	
18 Jan	35	1622	345	214	83	380.9	3.80	x10^{-12}	
19 Jan	35	399	140	99	58	266.2	2.66	x10^{-12}	
20 Jan	35	501	194	177	160	734.4	7.34	x10^{-12}	
21 Jan	35	862	269	195	121	555.3	5.55	x10^{-12}	

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CHECKED BY		REPORT NO.
DATE	VO #2	MODEL NO.

Reg x 4.59 x 10^{-14} = m/cc x 10^{-14}

DATE	BP	1st Reg	2nd Reg	5th Reg	L.L. Reg.	m/cc x 10^{-14}			
22 Jan 59	35	1089	315	229	143	657.3			
23 Jan	35	1379	357	234	111	509.4			
24 Jan									
25 Jan	35	1438	328	201	74	339.6			
26 Jan									
27 Jan									
28 Jan	35	~~252~~	148	105	62	284.5			
29 Jan	35	858	241	171	101	463.5			
30 Jan	37	369	202	149	96	440.6			
31 Jan	35	137	78	60	42	192.7			
1 Feb	35	615	142	101	60	275.4			
2 Feb	34	182	73	64	45	252.4			
3 Feb	35	908	184	121	58	266.2			
4 Feb	35	1191	294	176	58	266.2			
5 Feb	35	563	2	3	4	18.36			
6 Feb	37	737	267	168	69	~~318.0~~	318.0		
7 Feb	36	439	144	96	48	220.3			
8 Feb	35	490	92	60	28	128.5			
9 Feb	36	739	141	86	31	142.2			
10 Feb	37	67	46	33	20	91.8			
11 Feb	33	25	14	9	4	~~29.0~~	2.90	0020 TO 1530	15 hour Sample
11 Feb	33	169	17	17	17	~~249~~	2.19	1530 TO 2400	
12 Feb	34	307	53	42	31	142.2			
13 Feb	35	115	0	0	0	0			
14 Feb	35	528	79	52	25	114.7			
	VanNowen Air	Monitor	under	repair			14 –	Feb 59	
17 Feb	32	400	49	41	33	292.0	Substitute Monitor		
18 Feb	36	323	66	59	52	247.5			

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PREPARED BY	ATOMICS INTERNATIONAL A Division of North American Aviation, Inc.	PAGE OF
CHECKED BY		REPORT NO.
DATE		MODEL NO. V0

4.76 x 10⁻¹⁴

Date	BS	1st	24th	55th	LL. 155	mlu x10⁻¹⁴			Monitor
19Feb59	35	667	89	66	43	205.			subs.
20Feb	35	652	104	74	44	209.			subs
21Feb	35	374	87	78	69	328			subs
22Feb	35	292	59	47	35	166.6			subs.
23Feb	34	240	93	71	49	233.2			
24Feb	34	198	121	104	87	414.1			
25Feb	34	507	206	186	166	790.1			
26Feb	32	843	241	199	157	747.3			
27Feb	33	943	291	201	111	528.3			
28Feb	34	597	240	177	114	542.6			
1 MAR	33	775	209	142	75	357.0			
2 Mar	35	929	242	163	84	399.8			
3 MAR	35	842	243	160	77	366.5			
4 MAR	34	605	195	164	133	633.0			
5 MAR	30	703	237	178	119	566.0			
6 MAR	34	1006	257	180	103	490.0			
7 MAR	34	907	269	180	91	434.0			
8 MAR	32	988	279	186	93	442.6			
9 Mar	32	592	242	169	96	456.9			
10 Mar	30	315	105	84	63	300.0			
11 Mar	33	576	224	181	138	656.0			
12 Mar	35	1022	271	201	131	624.0			
13 Mar	33	658	197	129	61	290.0			
14 Mar	32	339	215	175	135	642.0			
15 Mar	33	416	181	156	131	623.5			
16 Mar	31	526	168	137	106	504.0			
17 Mar	33	977	250	183	116	552.0			
18 Mar	33	911	249	167	85	405.			
19 Mar	32	879	247	164	81	386.			
20 Mar	32	1175	276	199	122	580.7			

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CHECKED BY		REPORT NO.
DATE		MODEL NO. XO

4.76 X 10^{-7}

Date	BP	1ST	24	55	11	mc/cc x 10^{-14}			
21 Mar 59	32	1211	302	223	144	685.0			
22 Mar	33	265	153	115	77	366.0			
23 MAR	33	199	80	57	34	162.0			
24 MAR	33	93	50	39	28	133.0			
25 MAR	34	[illegible]	144	88	32	152.0			
26 Mar	34	331	130	60	—	NO ACTIVITY DETECTED			
27 Mar	34	598	107	63	19	90.4			
28 Mar	34	634	208	154	100	476.			
29 MAR	34	963	207	132	57	271.0			
30 MAR	30	298	129	90	51	242.0			
31 MAR	30	561	176	142	108	514.0			
1 APRIL	30	1170	264	185	106.	504.0			
2 APRIL	30	445	197	167	137	650.0			
3 APRIL	35	1026	232	162	92	437.0			
4 APRIL	35	362	173	124	75	357.0			
5 APRIL	30	359	111	89	67	319.0			
6 APRIL	30	407	118	88	58	276.0			
7 APRIL	31	241	91	72	53	252.0			
8 APR	32	204	105	87	69	328.0			
9 APR	33	502	141	109	77	366.0			
10 APR	33	343	149	119	89	424.0			
11 APR	33	692	194	133	72	342.0			
12 APR	35	1025	200	130	60	285.0			
13 APRIL	33	717	155	110	65	309.0			
14 APRIL	33	334	207	178	149	710.0			
15 APRIL	32	533	160	142	124	591.0			
16 APR	30	656	186	155	124	591.0			
17 APR	33	292	139	118	97	461.7			
18 APR	32	461	171	136	101	481.0			

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BNA02410152

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CHECKED BY		REPORT NO.
DATE	VO #2	MODEL NO.

1959

Date	Bq	1-7 Req	24 Req	55 Req	42. Req	mc/cc X10^{-14}			Correction Factor
19 APRIL	31	423	144	115	86	409.36			4.76X10^{-14}
20 APRIL	33	284	143	112	81	386.0			
21 APRIL	33	387	118	93	68	323.9			
22 APRIL	33	831	142	105	68	323.9			
23 APRIL	33	229	133	100	67	318.5			
24 APRIL	33	646	123	93	63	299.9			
25 APRIL	32	141	61	45	29	138.0			
26 APRIL	32	215	40	0	0	0			
27 APRIL	32	310	43	28	13	183.3	8	Hour Sample Time	
28 APRIL	30	861	154	107	60	285.6			
29 APRIL	30	300	153	119	85	405.0			
30 APRIL	30	145	91	75	59	281.0			
1 MAY	32	140	51	47	43	204.9			
2 MAY	30	580	95	69	43	204.9			
3 MAY	30	210	84	60	36	171.3			
4 MAY	31	613	105	74	43	204.9			
5 May	32	450	124	86	48	228.1			
6 MAY	NO DATA DUE TO EQUIPMENT FAILURE								
7 MAY	31	112	51	46	41	195.0			
8 MAY									
9 MAY	NO DATA TO 11 MAY DUE TO EQUIPMENT FAILURE								
10 MAY									
11 MAY	31	320	81	63	45	339.	← 17 hour Sample		
12 MAY	31	122	81	74	67	319.0			
13 May	32	185	113	106	99	471.2			
14 May	32	519	115	77	39	186.0			
15 May	33	680	172	127	82	390.0			
16 May	33	419	151	114	77	366.0			
17 May	32	294	126	95	64	305.0			
18 May	33	611	148	107	66	314.0			

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BNA02410153

PREPARED BY	ATOMICS INTERNATIONAL	PAGE
J. Woods	A Division of North American Aviation, Inc.	OF
CHECKED BY		REPORT NO.
DATE	VO #2 24 hour Monitor	MODEL NO.
21 May 59		

DATE	Bq	1st Reg	24 hr Reg	55 hr Reg	Long Life Reg.	$\mu c/cc \times 10^{-14}$			Remarks
19 MAY 59	33	445	139	102	65	40% Recalibrated	with		Cond. Sect. 6.17×10^{-14}
20 May 59	31	860	202	142	82	50% Bi^{210}	Source		
21 May 59	30	195	118	95	72	445.0			Eff: 3.8
22 May 59	30	252	69	59	49	302.			
23 May	32	245	101	83	65	401.			
24 May	33	206	92	79	66	407.			
25 MAY	33	395	97	70	43	266.			
26 MAY	35	53	24	15	6	37.0			
27 MAY	31	370	67	54	41	639.			9 h. 20 m Sample
28 MAY	34	481	115	66	17	105.0			
29 MAY	33	370	111	68	25	154.5			
30 MAY	32	672	142	88	34	210.0			
31 MAY	32	308	115	77	39	241.0			
1 June	33	431	128	86	44	272.0			
2 June	32	470	157	109	61	377.0			
3 June	31	600	168	108	48	296.0			
4 June		NO Data Due to FILTER TAPE Breakage							
5 June	30	116	39	34	29	179.0	This data not Reliable Due to Instrument failure		15 h 40 m Sample
6 June	32	60	8	3 >	66	407.0			
7 June	32	346	18	45	72	445.0			
8 June		NO Data to 6 July 59 due to Repair & re-calibration of Air Monitor							
9 June									
10 June									
11 June		Air Monitor Returned to Service 6 July 59							
									5.15
6 July	32	188	39	34	29	4.07×10^{-12}			
7 July	38	509	109	61	13	67.0			5.15
8 July	37	650	135	77	19	98.0			
9 July	36	355	112	61	10	51.5			

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PREPARED BY	ATOMICS INTERNATIONAL	PAGE
J. Moore	A Division of North American Aviation, Inc.	OF
CHECKED BY		REPORT NO.
DATE	Vanowen 24 hour Air Monitor	MODEL NO.

Date	Bq	1st Reg	24th Reg	554 Reg	L.L. Reg	uuf/cc $\times 10^{-14}$			Remarks
10 July	35	282	83	48	13	67.10	interpolated		
11 July	35	703	120	No data to 568 follows	16	82.5	interpolated		
12 July	35	408	100	46 65	30	154.5	interpolated	NEW G-M Tube this date	
13 July	33	189	82	51	20	96.1			4.8
14 July	40	154	58	42	26	125.0			
15 July	40	65	53	30	7	33.6			
16 July	40	201	78	41	4	19.2			
23 July	42	126	29	16	3	28.5	12 hour Sample		
24 July	42	503	100	53	6	28.8			
25 July	42	333	83	45	7	33.6			
26 July	43	170	76	37	—	No L.L. activity			
27 July	40	198	60	32	4	19.2			
28 July	42	184	28	16	4	19.2			
29 July	42	150	45	25	5	24.0			
30 July	43	353	106	52	—	No LL activity			
31 July	43	489	112	58	4	19.2			
1 August	43	258	90	48	6	28.8			
11 Sept 59	48	210	47	33	19	35.8		500 minute Sample	
12 Sept	50	299	89	42	0	no L.L.	activity	detected	
13 Sept	48	346	56	31	6	28.8			
14 Sept	47	572	112	58	4	19.2			
15 Sept	48	786	123	68	13	62.5			
16 Sept	50	364	114	59	4	19.2			
17 Sept	102	1066	150	92	34	65.4			*
18 Sept	96	150	78	44	10	19.2			*
19 Sept	100	153	29	14	0	N.A.D.			*
20 Sept	97	1400	157	79	1	1.92			*

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* Interpolation set @ 8 cts/reg

PREPARED BY	ATOMICS INTERNATIONAL	PAGE
CHECKED BY	A Division of North American Aviation, Inc.	OF
		REPORT NO.
DATE	V0 24 Hour Monitor	MODEL NO.

+ 8 cts/req

x + 4 cts/req

Date	Bg	1st Reg	24 Reg	55 Reg	L.L. Reg	mc/cc x10^{-14}			Remarks
21 SEPT	995	505	135	74	13	25.0	x		
22 SEPT	96	534	166	84	2	3.84	+		
23 SEPT	100	582	160	86	12	23.04	+		
24 SEPT	96	345	127	76	25	48.00	+		
25 SEPT	55	378	42	55	68	130.0	+		
26 SEPT	95	805	121	69	17	32.6	+		
27 SEPT	96	592	86	54	22	42.2	+		
28 SEPT	90	720	180	92	4	7.68	+		
29 SEPT	93	752	178	92	6	11.52	+		
30 SEPT	94	266	94	54	14	26.88	+		
1 OCT	100	1318	169	91	13	24.96	+		
2 OCT	100	447	1	2	1	1.92	+		
3 OCT	98	1991	309	174	39	74.88	+		
4 OCT	98	1561	251	128	5	9.60	+		
5 OCT	107	1941	274	138	2	3.84	+		
6 OCT	93	2049	346	184	22	42.2	+		
7 OCT									
8 OCT	98	2037	356	188	20.	38.4	+		
9 OCT	98	1912	272	145	18.	34.6	+		
10 OCT	97	2096	381	202	23.0	44.1	+		conv. fact for 8 cts/s
11 OCT	No Data To 14 OCT 59 due Servicing								= 2.49 x 10^{14}
12 OCT	and Calibration of Monitor								
13 OCT									
14 OCT	89	1715	241	121	1	2.49	+		
15 OCT	86	2500	489	292	95	236.0	+		
16 OCT	83	2492	597	332	67	167.0	+		
17 OCT	94	2222	442	222	2	4.98	+		
18 OCT									
19 OCT	86	1537	11	6	1	~~2.49~~	2.49 +		
20 OCT	90	1436	233	125	17	42.3	+		

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PREPARED BY	ATOMICS INTERNATIONAL	PAGE
J. Moore	A Division of North American Aviation, Inc.	OF
CHECKED BY		REPORT NO.
DATE: 23 Oct 1959	V0 Air Monitor Data	MODEL NO.

* = 8 CTS/REG

Date	By	1st Reg	24 Reg	55 Reg	L.L. Reg	μc/cc ×10⁻¹¹			Remarks
21 Oct	89	1631	404	213	22	54.8			2.49
22 Oct	85	1468	289	175	61	152.0			
23 Oct	85	2411	306	174	42	104.5			
24 Oct	No Data to 27 Oct 59 due to Streeter-Amet failure								
25 Oct									
26 Oct									
27 Oct	84	136	39	23	7	17.4			
28 Oct									
29 Oct									
30 Oct									
31 Oct									
1 Nov								5.00	
2 Nov	50	463	52	31	10	50.0		Register reset to 16 c/log	
3 Nov	60	336	81	33	—	—			
4 Nov	69	1077	135	71	7	35.0			
5 Nov	60	210	30	17	4	20.0			
6 Nov	57	230	21	12	3	15.0			
7 Nov	52	572	100	52	4	20.0			
8 Nov	56	1142	163	84	5	25.0			
9 Nov	43	1539	222	122	22	110.0			
10 Nov	40	1653	276	145	14	70.0			
11 Nov	46	930	23	12	1	5.00			
12 Nov	No Data to 16 Nov 1959 Due to Streeter-Amet Relay failure								
13 Nov									
14 Nov									
15 Nov									
16 Nov	48	1105	126	65	4	20.0			
17 Nov	50	1868	259	131	3	15.0			
18 Nov	No Data to 25 Nov 59 due to Streeter-Amet Malfunction								
19 Nov									

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BNA02410157

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CHECKED BY		REPORT NO.
DATE 30 Dec 59	VO 24 hr. Air Monitor	MODEL NO.

DATE	BX	1ST REG	24 REG	55 REG	LL. REG	μc/cc X10⁻¹⁰			REMARKS
20 Nov									
21 Nov									
22 Nov									
23 Nov									
24 Nov									
25 Nov	46	1744	207	111	15	82.6			5.5
26 Nov	46	251	71	38	5	27.5			
27 Nov	48	776	106	54	2	11.0			
28 Nov	47	507	89	47	5	27.5			
29 Nov									
30 Nov									
1 DEC	45	502	81	44	7	38.5			
2 DEC	46	1645	193	103	13	71.5			
	NO DATA TO 22 Jan 1960 DUE TO MALfunction of the STREETER-AMET PRINT OUT mechanism. Repeated doubling STAMPING precluded the proper DATA recording cycle and thereby obviated the normal SAMPLING cycle AT the 24 hour SAMPLE change over. New STREETER-AMET "AMETRON" units have been ordered.								
22 Jan 60	45	—	122	62	2	11.00			5.5
23 Jan	45	500	108	55	2	11.00			
24 Jan	46	526	124	63	2	11.00			
25 Jan	46	168	20	11	2	11.00			
26 Jan	46	320	34	18	2	11.00			
27 Jan	43	583	74	38	2	11.00			
28 Jan	45	134	54	36	18	99.00			

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Appendix G

Description of RADTRAN Mathematical Model

1.Introduction and Background

The computer code RADTRAN was developed to provide a powerful and flexible tool for fission product transport analysis. The data libraries contain 367 nuclides in 100 decay chains, with a maximum chain length of six members. RADTRAN solves a set of coupled, linear first-order differential equations using a matrix exponential method to provide an exact solution for the time-dependent buildup and transport of radioactive fission products.

The primary use of RADTRAN is to provide time-dependent source terms that may be used to evaluate radiological conditions for plant operations. The code is very flexible, and offers many variations of calculations, depending on the definitions of coefficients described in the rate equation shown below. The most common use is to provide a time-dependent release from the plant via the ventilation systems, by defining nodal source terms inside the plant.

1.1. General Rate Equation

The computer code RADTRAN addresses time-dependent solution of radioactive material transport. The general rate equation is:

$$\frac{dN_f}{dt} = \alpha N_f + \lambda_{i-1} N_{i-1} - \lambda_i N_f - \beta N_f - \omega N_f$$

where:

N_f = population of radionuclide N [Atoms] in a decay chain of up to 6 members

α = production coefficient of radionuclide N

λ_{i-1} = decay constant of precursor of radionuclide in decay chain

N_{i-1} = population of precursor of radionuclide in decay chain

and

αN_f = production term for radionuclide N

$\lambda_{i-1} N_{i-1}$ = production term for radionuclide N by radioactive decay of precursor

$\lambda_i N_f$ = removal term for radionuclide N by radioactive decay

βN_f = removal term for radionuclide N with accumulation in separate node

ωN_f = removal term for radionuclide N with release rate output as Ci/Sec

The flexibility of the code is demonstrated in the use and definition of the coefficients $\alpha, \beta,$ and $\omega,$ which are described in the following sections. The βN_f and ωN_f terms allow modeling chemical and physical removal such as ionic exchange or filtration.

1.2. Production in Calculation Node by Physical Transport

In this mode of transport, the fission products are identified as a concentration in a fluid or gas as μCi/cc and the time-dependent release into the calculation node is specified by the α coefficient. The "leak rate" is specified as one of three choices:

- Lb_m /hr
- cc/sec
- Cu. Ft/min

Internal conversions in the code handle the conversions such that the "atoms/sec" production term is consistent with the rate equation for each radionuclide.

1.3. Removal Coefficients

1.3.1. Partial Removal Coefficient

The partial removal coefficient is used to allow filtration of the decay chain family with filtration efficiencies set as time-dependent values. The removal coefficient is defined by:

$$\beta = \frac{m_0}{M} F_i$$

where :

m_0 = Removal rate in consistent units, e.g., cfm

M = Denominator to obtain 1/sec when taken with m_0

F_i = Filtration efficiency for isotope i

Efficiencies must be specified in fractions instead of percent, and may vary with time. The unique feature of this removal term is that daughter products of an isotope that have efficiency values of 0 will be released from the "filter" and reintroduced into the main node. This models the behavior of a recirculation filter, for example. Any removal system that functions in a recirculation loop may be modeled. There is also an option called the "strongly interacting model" whereby gaseous daughters may be retained in the "filter" node. This feature allows accumulation for a time period, with subsequent leakage or release according to the value of the filtration efficiency and removal rate.

1.3.2. Total Removal Coefficient

The total removal coefficient is used to completely remove a decay chain or members thereof from the node. It may be thought of as a "purge" type node. The total removal coefficient is defined by:

$$\varpi = \frac{m_p F_i}{M}$$

where:

m_p = removal rate, in consistent units with M, e.g. cfm,

M = Denominator to obtain 1/sec when used with m_p

F_i = Filtration efficiency for total removal node.

The release from the node is calculated as atoms/sec removal from the node. Filtration efficiencies may be specified to vary with time.

JOHN A. DANIEL, SR.

University of Florida: BSNES, Nuclear Engineering Sciences, 1972

Mr. Daniel is currently providing engineering services to the power industry as President and Engineering Department Manager for Daniel and Associates, Inc. His primary area of expertise is in the area of nuclear safety; specifically in radioactive fission product behavior and radiation shielding in nuclear power plants. John is knowledgeable in both PWR and BWR reactor design and operation, as well as some foreign reactor designs.

Mr. Daniel is experienced in the calculation of dose rates to personnel from both high- and low-level radioactive sources. He derived equations and wrote a computer code to predict the radiation exposure from Bremmstrahlung radiation to military personnel using radiation calibration equipment.

He was the Principal Investigator for the reconstruction of the TMI-2 accident using data recorded by in-plant instruments and subsequently calculated the quantities of fission products released to the environment, using computer codes developed in-house. He was the Principal Investigator for an NRC project that developed a correlation between radioactive iodine resuspension and ambient temperature, published as NUREG/CR-4953.

He was associated with Three Mile Island from March, 1979 until May, 1984, as technical advisor to the Vice President of GPU following the accident, and during the accident recovery phase as Manager of the TMI-2 Radiological Analyses Group. In that capacity, he managed a group of engineers who investigated the transport of fission products from the reactor coolant system into the auxiliary building, and also provided radiation shielding calculations and review of shielding designs performed by others. The group also provided dose calculations and dose estimates for personnel who had entered radiation areas early in the accident without dosimetry. The group investigated unusual radiation exposures and contaminations including the airborne uptake of radioactive particulate material containing actinides. He also provided manrem estimates for decommissioning various systems and dose projections for work scheduled in high radiation areas.

Mr. Daniel devised techniques for the first remote access to the Reactor Building and sampling of Reactor Building sump water while still maintaining the integrity of the Reactor Building as a pressure vessel. He directed engineering projects for sampling high-level radioactive sources so that decontamination operations could proceed safely.

Mr. Daniel was TMI site manager for Science Applications, Inc., and was responsible for the laboratory's contribution to recovery operations, under contract to Department of Energy. He developed a systematic approach for documenting the fission product release from the fuel and

transport through the primary and auxiliary systems, utilizing a combination of specialized sampling, computer modeling and in-situ gamma spectrometer measurements. Mr. Daniel made several entries into the reactor building to perform scientific measurements related to the accident sequence and to measure the effectiveness of decontamination efforts.

As a Project Manager at United Engineers & Constructors, Inc., Mr. Daniel performed engineering and economic assessments of foreign and domestic reactor systems under contract to Department of Energy. His duties involved coordination and management of technical design reviews of the Canadian Heavy Water Reactor design and the French Phoenix and Super-Phoenix sodium-cooled breeder reactors. He also analyzed the design of heat exchangers, condensers, and engineered safeguard systems for all nuclear units contracted to UE&C, and developed mathematical models to determine their performance in normal and emergency operations.

Mr. Daniel was the Chief Licensing Engineer for the Sears Island Nuclear Unit, prior to its cancellation by the utility. His duties involved system-level design review from a U.S. licensing standpoint, and acting as liaison between the various state and federal regulatory agencies.

He has provided extensive mathematical modeling in all areas of nuclear safety analyses, particularly in fission product release and transport, pressure-temperature transient analyses, two-phase flow, and heat transfer through multi-material structures and components.

Mr. Daniel is the author of several major computer codes, including RADTRAN©. The code was used to calculate the allowable release rate of radioactive gases from the TMI-2 reactor building during the venting of those gases during 1980. He is also the author of Mesodif-D, an enhanced version of the NRC's segmented plume model, and was co-author of Contran, a thermodynamic code for predicting pressure-temperature response following PWR transients. He has also revised a version of the ORIGEN2 radioactive isotopic generation and depletion code. He has written numerous small codes related to off-site dose calculations, gamma-ray shielding, and two phase flow in large piping.

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