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REPORT ON RETIREMENT

OF

HALLAM NUCLEAR POWER FACILITY

AEC Research and Development Report

Atomics International
North American Rockwell

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Canoga Park, California 91304

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REACTOR TECHN

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Atomics International
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P.O. Box 309
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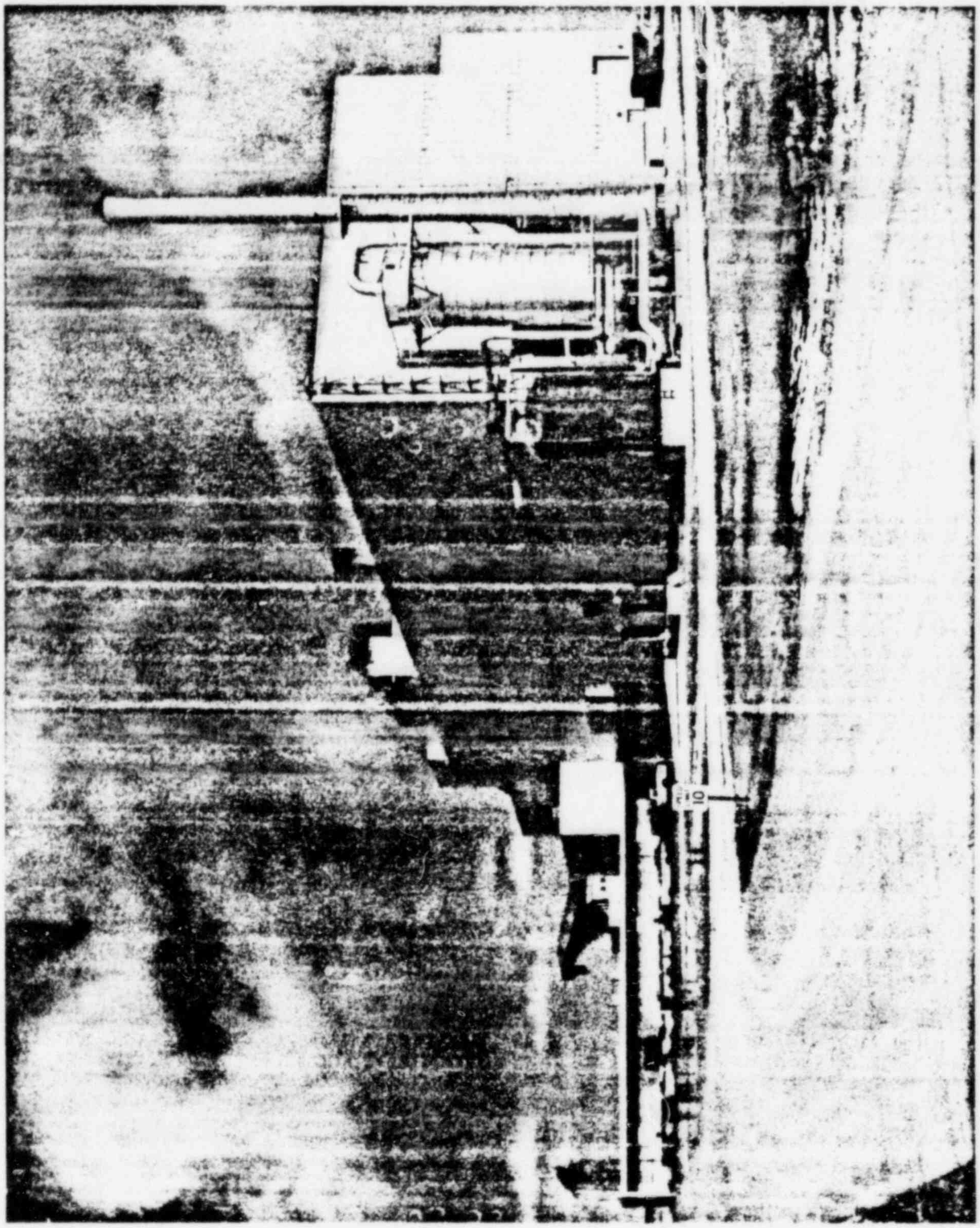

Frontispiece. Hallam Nuclear Power Facility of Consumers Public Power District

ABSTRACT

The Hallam Nuclear Power Facility (HNPF) was authorized as part of the Atomic Energy Commission's Power Demonstration Reactor Program. It was a sodium-cooled graphite-moderated power demonstration reactor owned by AEC and operated by Consumers Public Power District (CPPD)* at its Sheldon Station near Hallam, Nebraska. The HNPF was shut down in September 1964 as a result of failures in the cladding of certain moderator elements. Since the HNPF had significantly served its basic purposes under the Commission's Power Demonstration Plant Program, AEC determined that in lieu of restarting the facility it was to be decommissioned.

The Commission requested CPPD and Atomics International, the designer of the facility, to prepare a plan for decommissioning which provided for salvaging as much of the plant as economically feasible. The below-ground portion of the plant was to be adequately sealed to safely contain and restrict access to the contaminated and irradiated materials sealed therein. Major elements of the decommissioning plan included: (1) shipping all reactor fuel off-site; (2) shipping all bulk sodium off-site, and removing or reacting all residual sodium to eliminate chemical hazard; (3) disposing of all other contaminated or irradiated materials by shipment off-site or placement in on-site below-ground storage volumes; and (4) sealing such underground volumes to become a structure which isolates the contamination and radioactivity therein from release or access. The plan was directed at putting the plant in a radiation-safe condition such that after decommissioning no continuing AEC licensing of the premises would be required, and access to the exposed surfaces of the premises would not be restricted from a health and safety standpoint.

The decommissioning of HNPF in accordance with the above plan was substantially completed in October 1969. The AEC after review and approval of the program documentation will issue a termination of the operating authority.

*Recently renamed Nebraska Public Power District.

1.0 INTRODUCTION

The Hallam Nuclear Power Facility (HNPF) was a 254-Mwt sodium-cooled graphite-moderated nuclear power reactor located on land leased to the AEC by Consumers Public Power District (CPPD) at their Sheldon Station near Hallam, Nebraska. The AEC furnished the reactor plant under the Commission's Power Demonstration Reactor Program (second round). It was operated by CPPD under AEC operating authorization to provide steam to the existing conventional generating facility at the Sheldon Station.

The installation became operational early in 1963. Power generation continued through September 1964, at which time the plant was shut down because of defects developing in the cladding of certain moderator elements. Inasmuch as the facility had substantially served its purpose under AEC's Demonstration Program, by enhancing technology associated with a liquid-metal reactor plant and by demonstrating the safe operability of such a plant by a utility, the Commission determined not to repair the moderator elements and restore the reactor to power production, but rather to terminate the project and dismantle the reactor plant in accordance with its contract with CPPD.

To serve as a basis for development of a retirement plan for the HNPF facility the Commission established the general scope and objectives of the decontamination and dismantlement of the facility. The scope and objectives (see Section 4.0 of this report) were directed at making safe from a radiation standpoint the premises on which the reactor plant was located, with no continuing AEC license being required for use or access to the exposed surfaces of the premises after removal of the plant superstructure. Consistent with that direction the Retirement Plan for HNPF(1)* was developed by Atomics International, which had designed the facility, in conjunction with CPPD; the Plan was approved by AEC.

The overall responsibility for dismantling the nuclear facility in accordance with the Retirement Plan rested with CPPD. As holder of the Operating Authorization from AEC for the HNPF, CPPD was directly responsible for radiological, as well as industrial safety throughout the plant retirement. The work involved included removal of irradiated and unirradiated fuel, shipment of bulk

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

primary and secondary sodium, reacting of residual sodium, salvage equipment and components for reuse or storage, removal of plant systems, decontamination, disposal of irradiated materials, demolition of above-grade structures, and sealing of underground structures. All this was done by the utility either with its own manpower resources or by subcontractors under its supervision. CPPD also subcontracted with Atomics International for engineering assistance in connection with the plant retirement.

AEC exercised surveillance over the facility retirement through: approval of the Retirement Plan, specifications governing the retirement activities, and detailed procedures for carrying out those activities; an AEC technical representative at the site; and contractual arrangement with AI to provide management and engineering assistance, including personnel at the site. Through its regulatory processes the Commission imposed requirements that the plant decommissioning and decontamination be done by means that maintained the safety of the public, as well as those directly engaged, from radiological hazards.

At the time the Commission determined to retire the HNPF it was the first AEC-owned utility-operated reactor plant to be decommissioned. In the planning and carrying out of the plant retirement it was an intent of AEC to demonstrate the feasibility of a nuclear plant being dismantled by an operating utility utilizing existent technology. It is the purpose of this report to document the means by which the HNPF dismantlement was accomplished by CPPD. It should be recognized, however, that the extent of decontamination and removal of HNPF was dictated by the obligations of AEC under its contract and lease with CPPD and may not represent the extent of dismantlement necessary in connection with other nuclear power plants taken out of service. As indicated in Appendix VI, Page 25, other alternatives are available to utilities in retiring their nuclear plants. Unless the utility decides to remove all radioactive material from the site, further use of the site is subject to applicable regulatory requirements.

2.0 FACILITY DESCRIPTION

The HNPF, rated at 75-Mw electrical output, was of the sodium-cooled graphite-moderated reactor concept often referred to as SGR. It was an outgrowth of the Sodium Reactor Experiment (SRE) which also made use of an intermediate neutron-energy spectrum obtainable with a graphite moderator, and requiring only slightly enriched nuclear fuel.

The HNPF core consisted of fuel assemblies fitted between the corners of the hexagonal stainless-clad graphite moderator elements. Most of the fuel assemblies contained enriched U-Mo, with a few containing UC for experimental purposes. Neutron-absorber control elements occupied other fuel channel positions. The configuration of core, shielding, and sodium pipe connections is shown in Figure 2-1; the moderator element arrangement is shown in Figure 2-2; and a typical core loading is shown in Figure 2-3. Reactor-design data are given in Tables 1 through 7 of Appendix VII. Heat was transferred from the core to the intermediate heat exchangers by use of three sodium pumps, one in each of three primary sodium circuits designed to operate at 1200°F. However, actual operating temperature never exceeded 950°F. Each of the circuits (Figure 2-4) was independent and contained block, check, and throttle valves; instrumentation; temperature and flow sensors; and main primary centrifugal pumps. Storage for the primary sodium was provided by five horizontal, cylindrical tanks. Filling and draining operations were accomplished with linear induction electric pumps. A helium inert gas system protected the sodium from exposure to air at all times, and filled the space between the sodium and the top shield in the core tank during nuclear operation, thus allowing thermal expansion changes in the sodium.

From the intermediate heat exchangers, which provided the boundary between the radioactive primary sodium and the nonradioactive secondary system, heat was transferred to the three steam generators by the secondary sodium circuits. These were similar to the primary system, but each required a thermal expansion tank for the sodium volume change with temperature. Superheated steam at 825°F to the conventional turbine generators in the CPPD portion of the facility completed the energy path from the nuclear reactor to electric power.

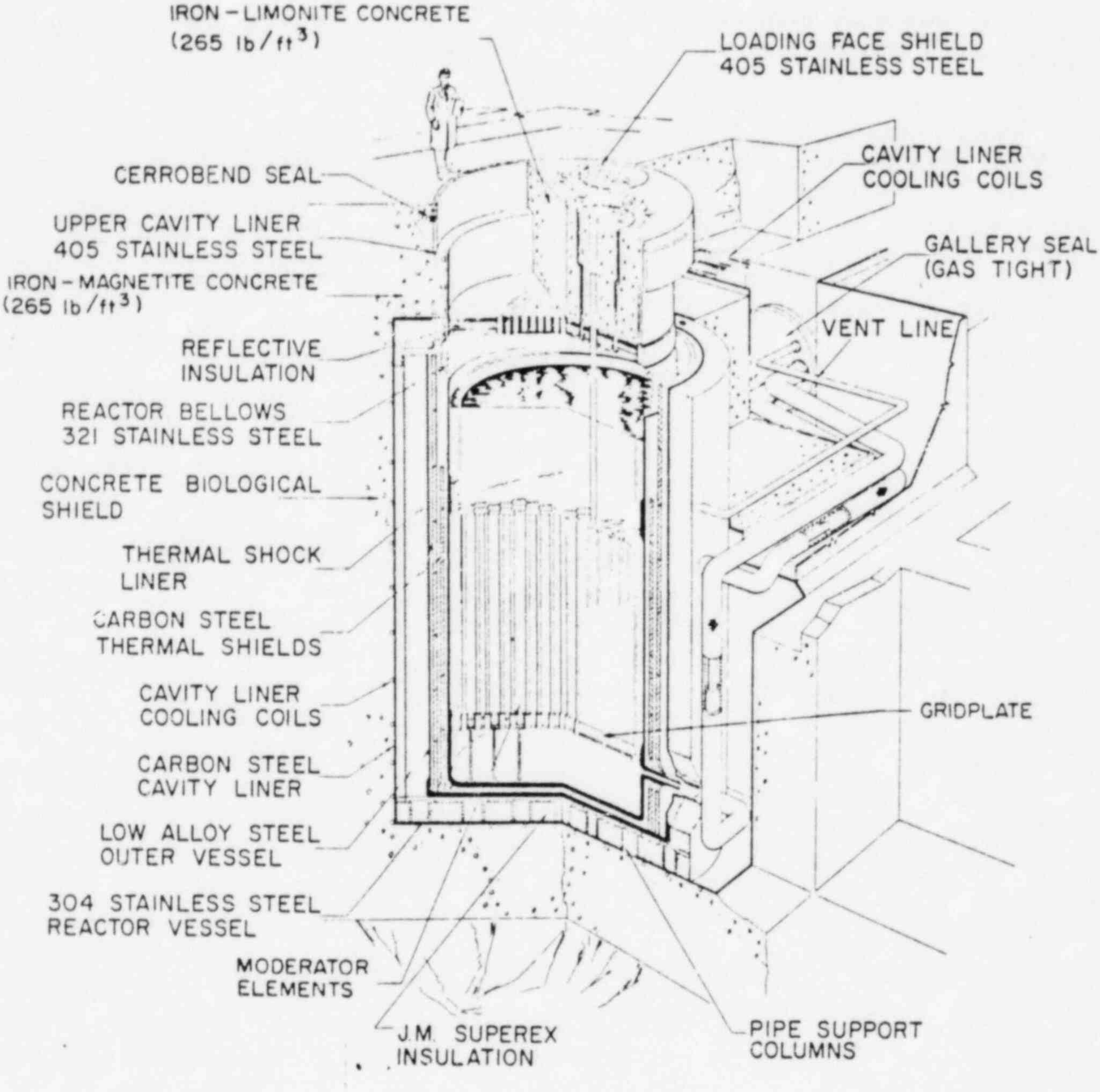

Figure 2-1. Core Configuration

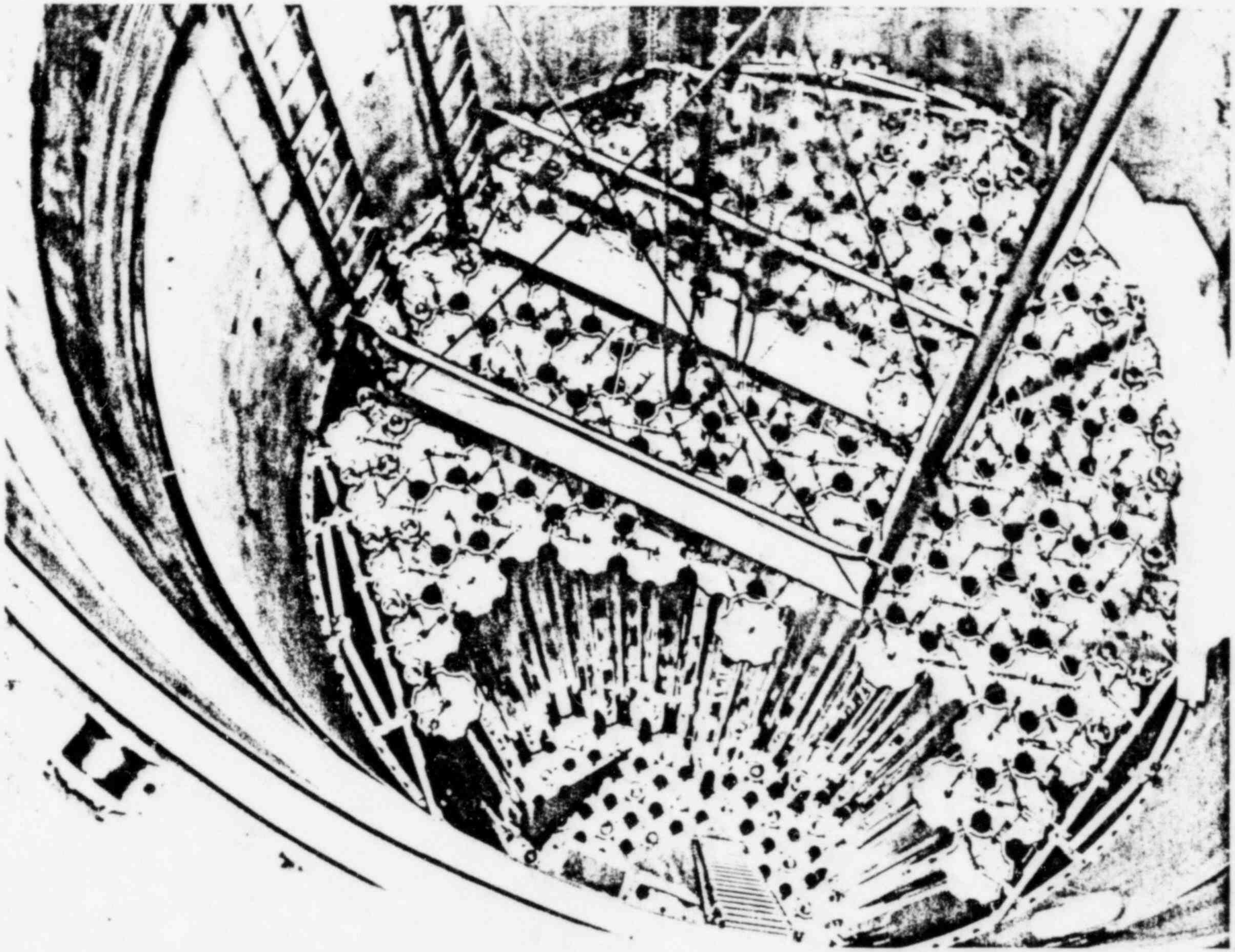

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Figure 2-2. Partial Assembly of Moderator Elements in the Core Tank

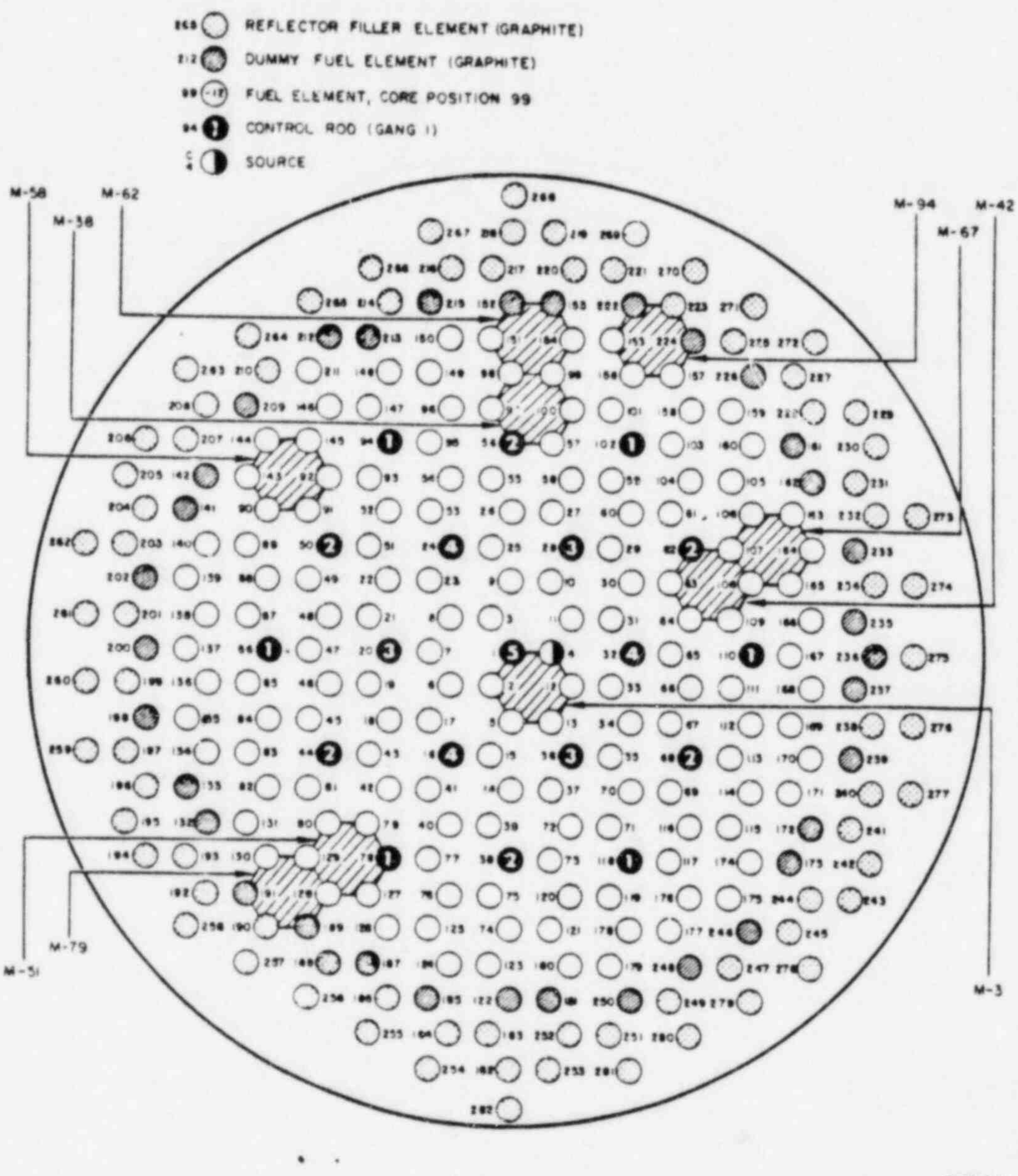

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Figure 2-3. Typical Core Loading (positions of moderator elements that cracked their cladding are shown shaded)

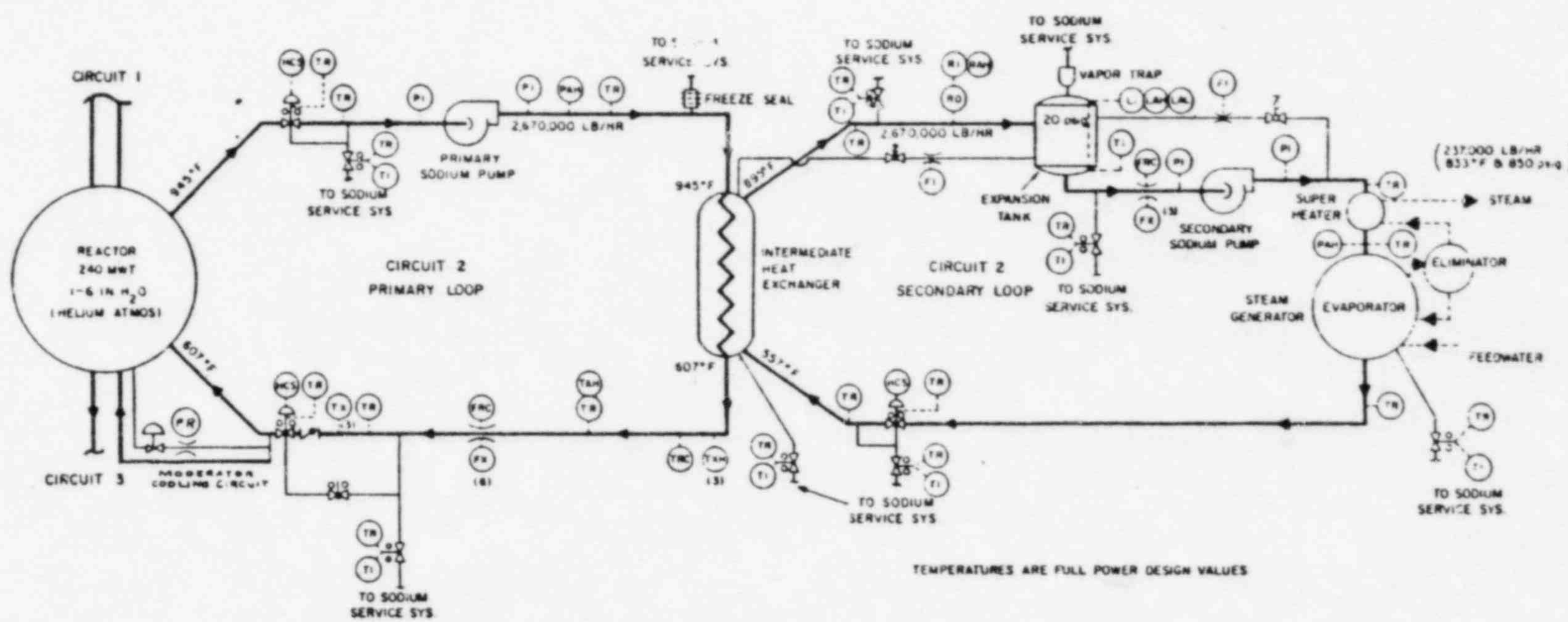

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Figure 2-4. Sodium Heat Transfer Circuit

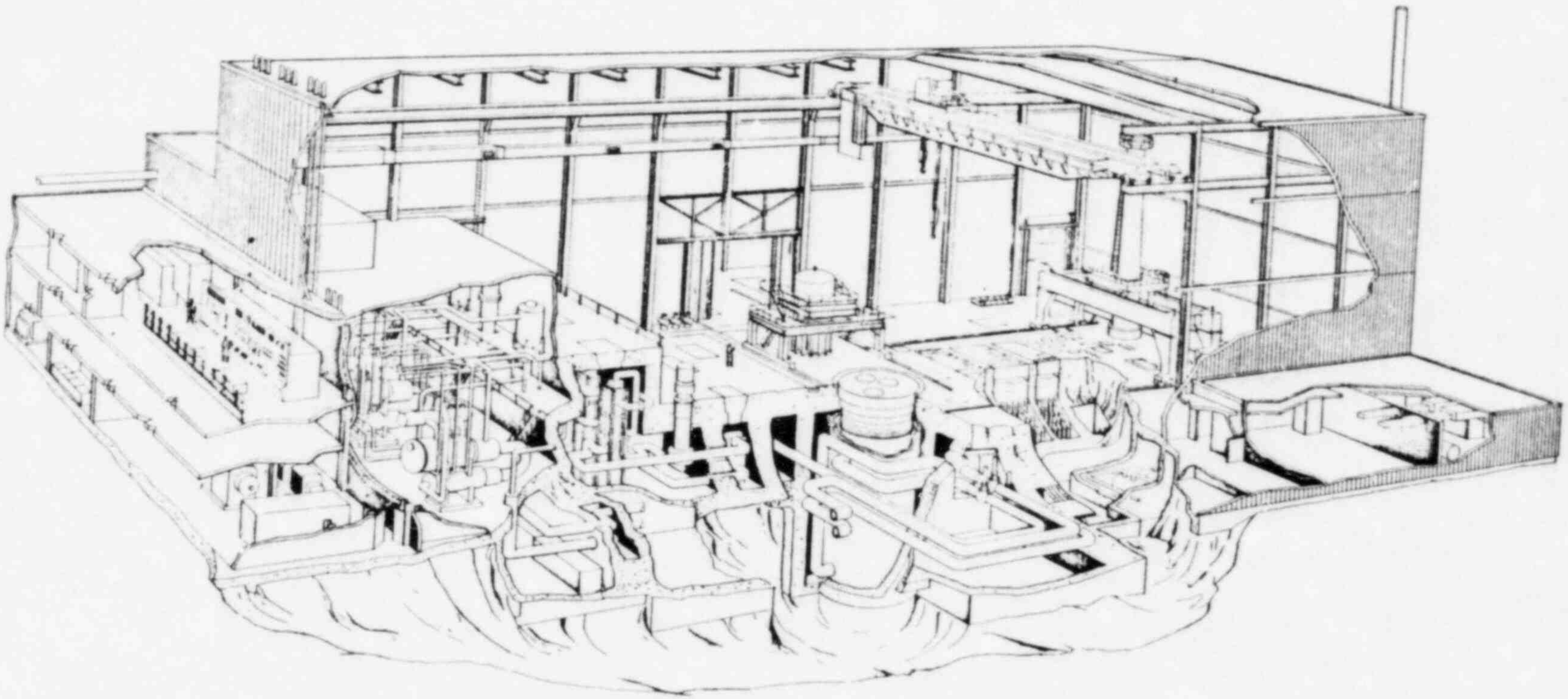

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Figure 2-5. HNPF Structure and General Arrangement

Figure 2-5 is a cutaway drawing of the nuclear building structure. The cylindrical reactor-vessel structure is in the center of the drawing. The sodium coolant piping extends from the reactor and contains thermal expansion loops. This piping carries the sodium to the intermediate heat exchangers (IHX) visible in the above-floor-level concrete structure. Located immediately to the left of the heat exchangers are the steam generators. Figure 2-6 shows the equipment locations in a sectional plan just below the operating floor of the reactor and Figure 2-7 is an elevation section through the center of the structure. These two figures show the plant major components and indicate their exact size and position relationship.

Important auxiliary facilities and equipment were: the fuel-handling machine; control and safety-element parking station; wash cells for fuel; control and safety elements and pumps; storage cells for fuel and moderator elements; the maintenance cell equipped with shielded windows and remote-handling manipulators; radioactive alarm system; radioactive-waste (gas and liquid) building; gas filter and discharge system including stack, laundry and change rooms; building service cranes; and emergency electrical system. The function and use of these items is largely self-explanatory, but further discussion will be found in the section on operations and in the appropriate retirement-activity specification, Appendix I.

The general construction of the plant is shown in Figure 2-7. Below-grade structures were steel-reinforced concrete. Figures 2-8 and 2-9 are photographs made during construction which show the below-grade work; the first shows the partially completed concrete structure with the core cavity liner in place, the second shows the same region at a later date ready for the concrete pour to the operating floor level.

Above grade the structure was of conventional design making use of steel columns, beams, and purlins with built-up insulated steel panels and built-up roof over steel decking. The structure included overhead cranes. Rails were provided in the floor for the fuel-handling machine. Figure 2-10 shows the interior views looking south from Column 17. The fuel-handling machine is prominent in the center background. Many structural details are apparent on the walls and ceiling. Fuel-storage cell covers are visible in the center foreground. The general floor-plan arrangement is shown in Figure 2-6.

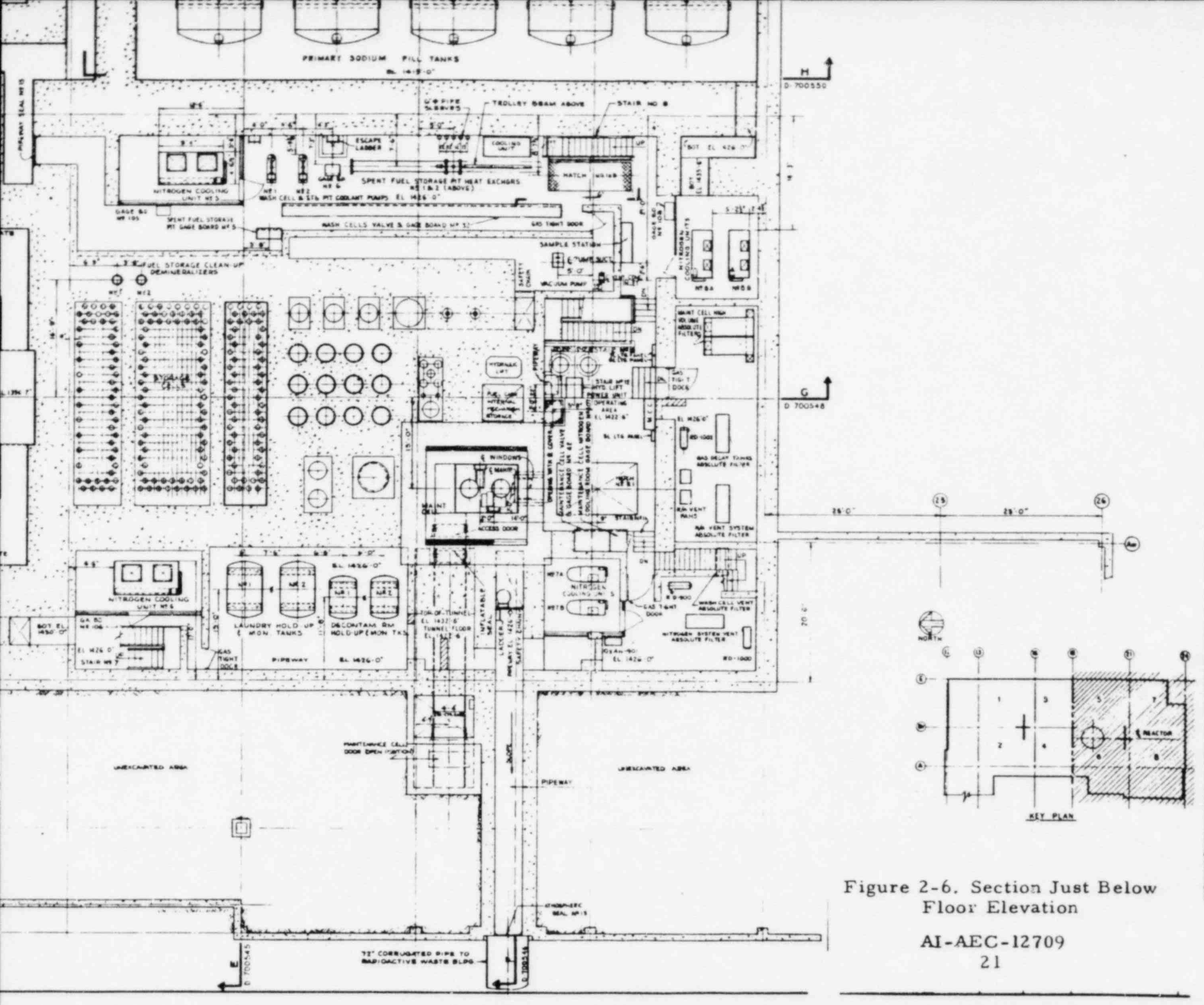
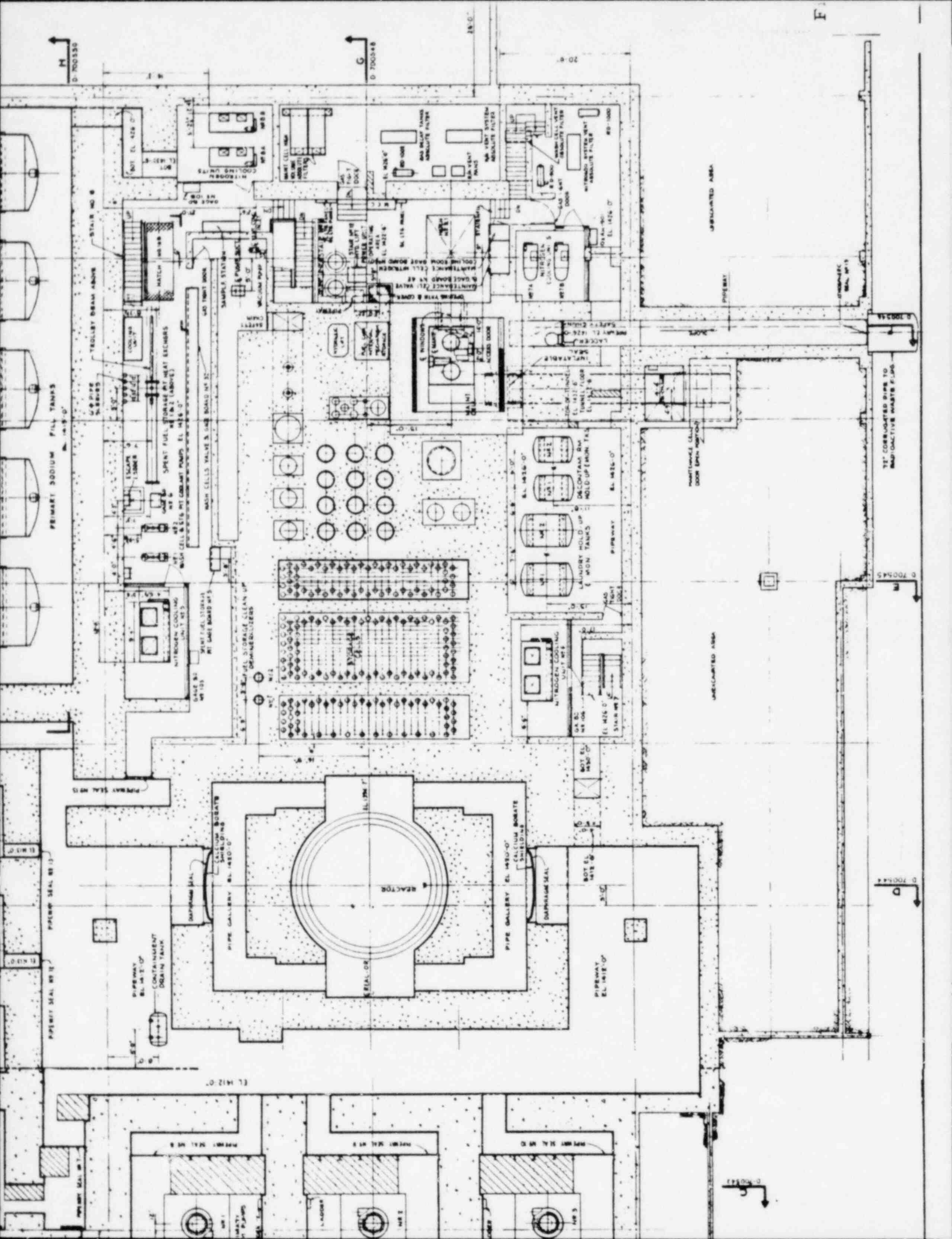
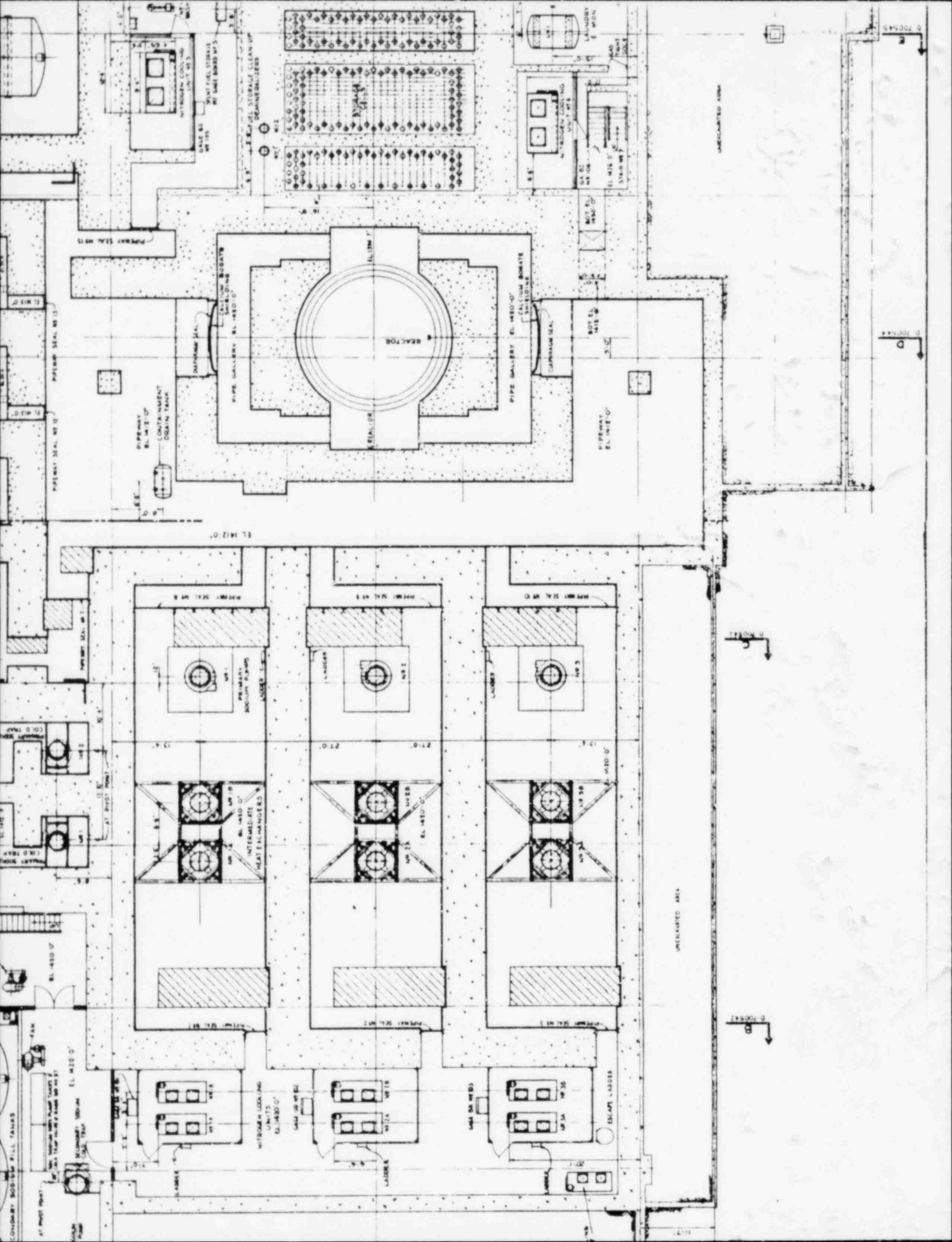
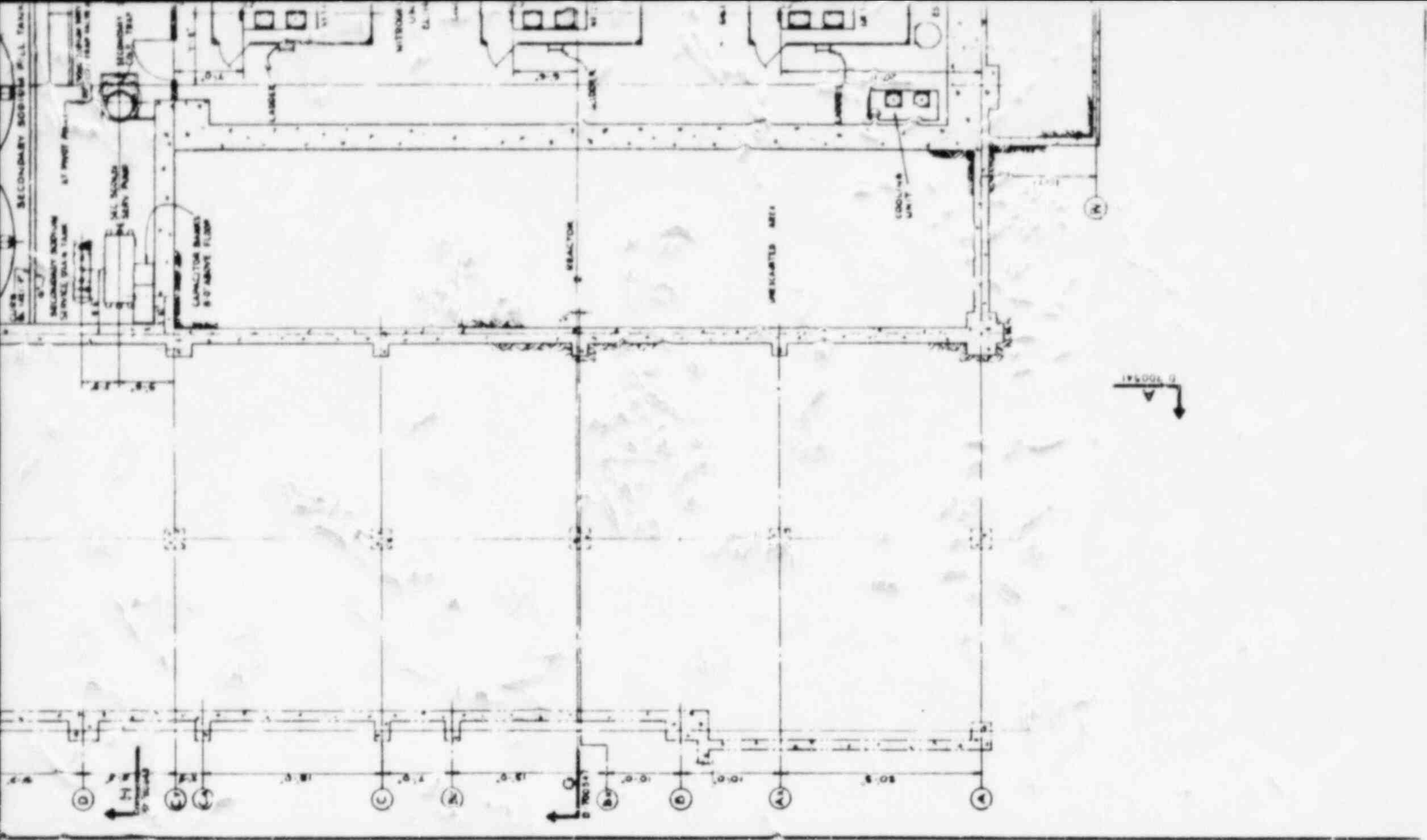


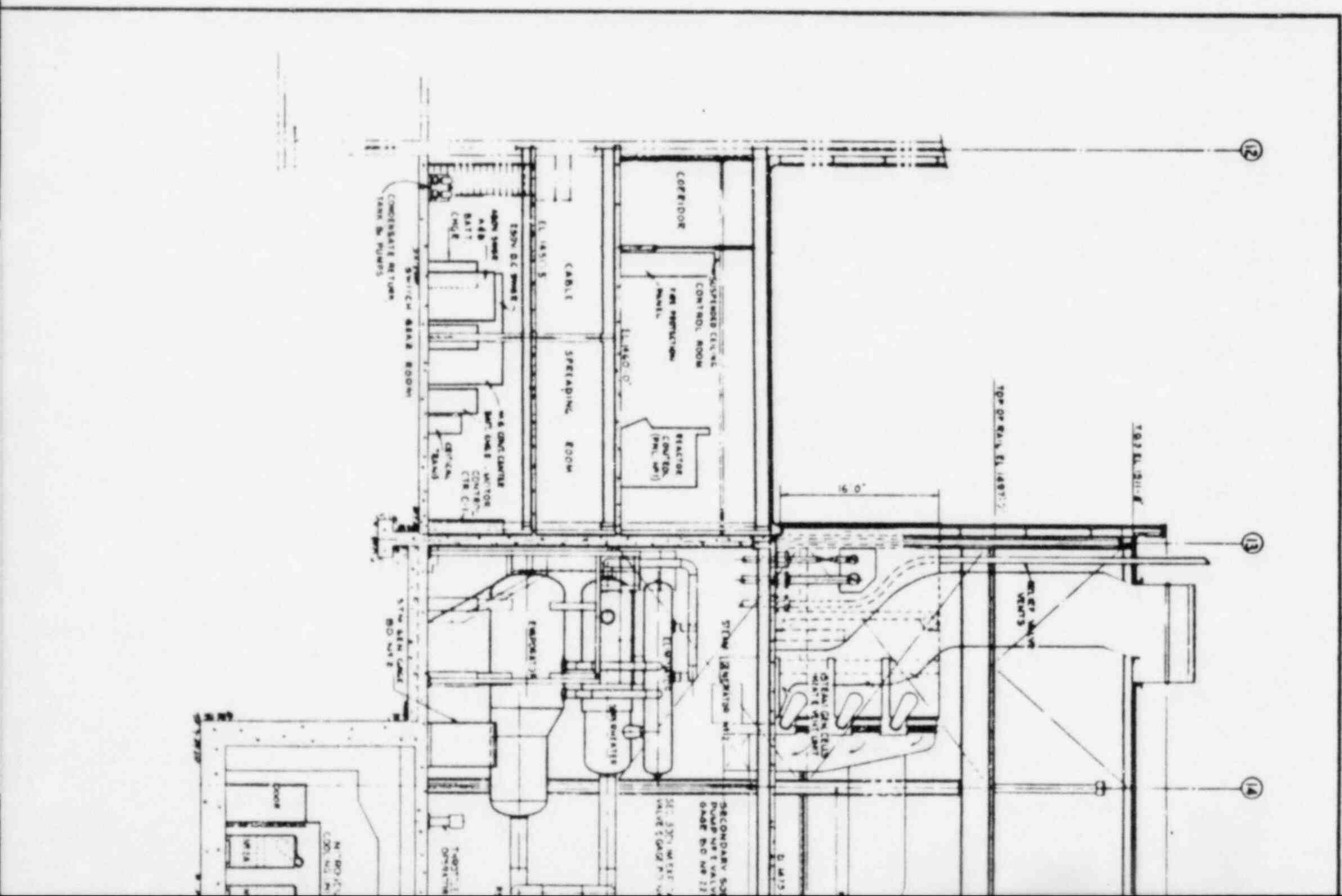
Figure 2-6. Section Just Below Floor Elevation



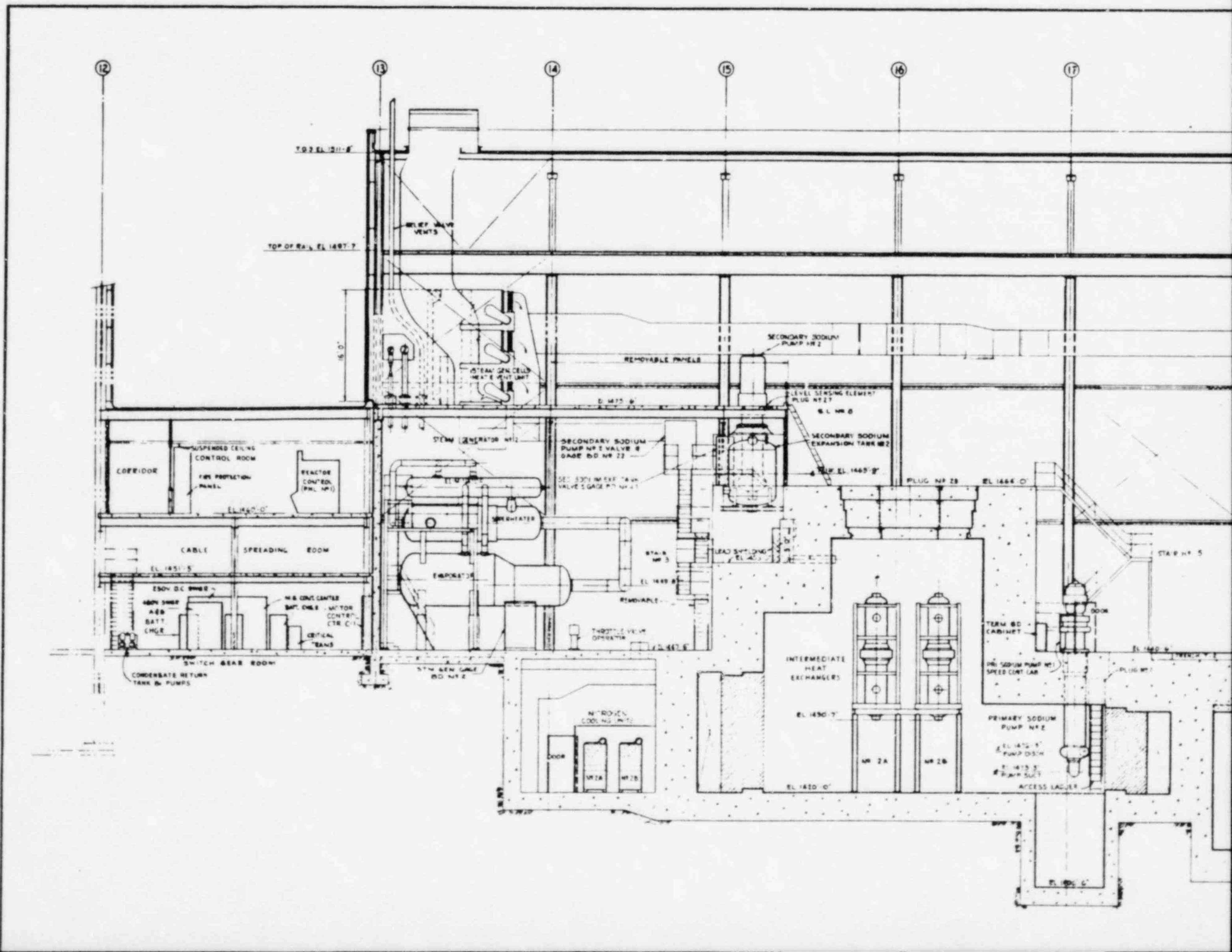
REACTOR
PIPE GALLERY
PRIMARY SODIUM FILL TANKS
CONTAINMENT DRAIN TANK
UNEXCAVATED AREA
PIPEWAY


REACTOR
PIPE GALLERY
CALCIUM BORATE SHIELDING
DIAPHRAGM SEAL
PIPEWAY
CONTAINMENT DRAIN TANK
EL 1412'-0"
UNEXCAVATED AREA
PRIMARY SODIUM PUMPS
INTERMEDIATE HEAT EXCHANGERS
NITROGEN COOLING UNITS
ESCAPE LADDER
LADDER
SECONDARY SODIUM FILL TANKS

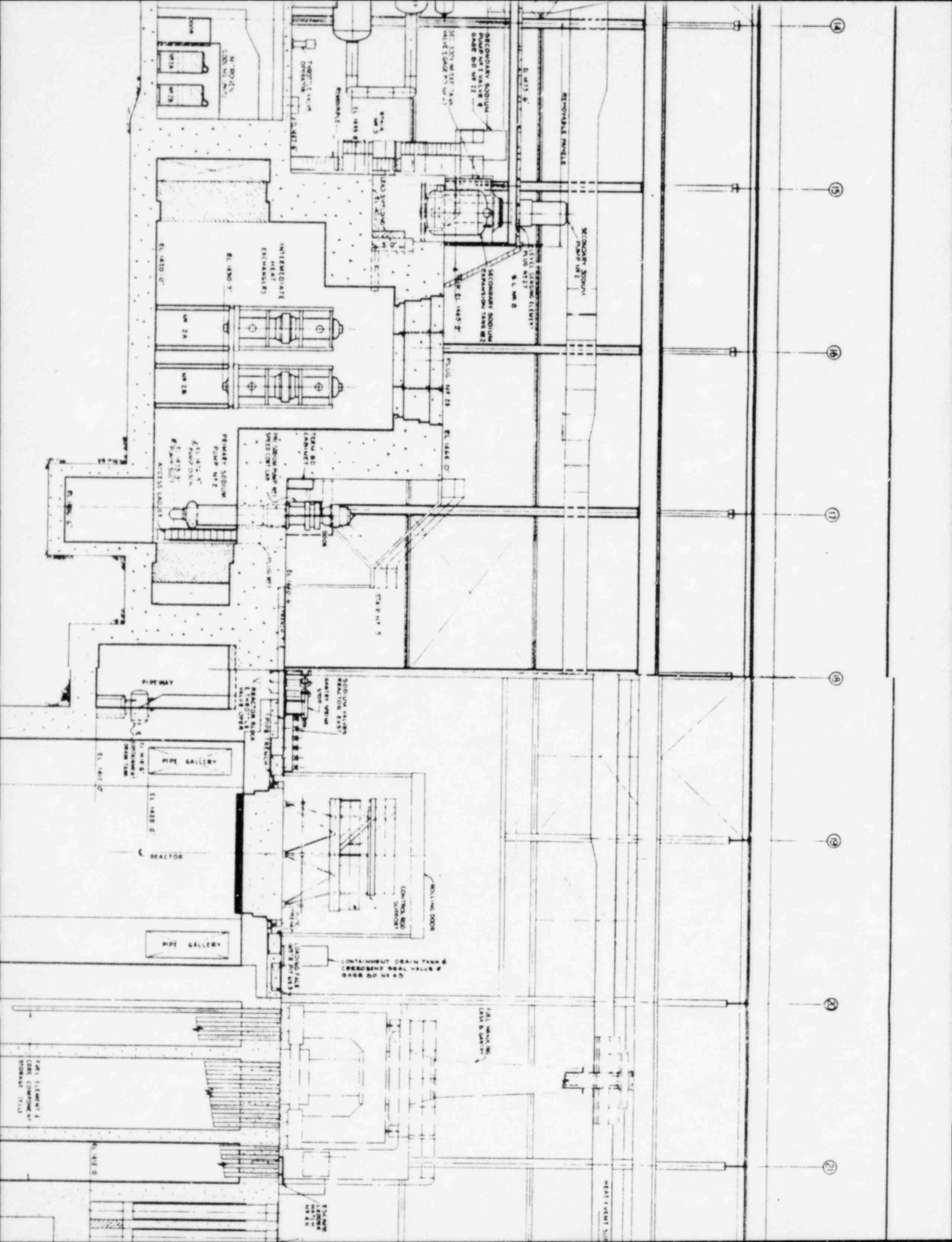




CORRIDOR
CONTROL ROOM
CABLE
SPREADING
ROOM
SWITCH GEAR ROOM
CONDENSATE RETURN TANK & PUMPS
REACTOR CONTROL (PNL NO.1)
FIRE PROTECTION PANEL
SUSPENDED CEILING
SECONDARY SYSTEM
RELIEF VALVE VENTS
16'-0"



12
13
14
15
16
17
TOP OF RAIL EL 1487'-7
RELIEF VALVE VENTS
16'-0"
REMOVABLE PANELS
SECONDARY SODIUM PUMP Nº 2
LEVEL SENSING ELEMENT PLUG Nº 27
SECONDARY SODIUM EXPANSION TANK Nº 2
STEAM GENERATOR Nº 2
SECONDARY SODIUM PUMP Nº 2 VALVE & GAGE BD Nº 22
SUSPENDED CEILING
CONTROL ROOM
CORRIDOR
FIRE PROTECTION PANEL
REACTOR CONTROL
EL 1460'-0"
CABLE SPREADING ROOM
EL 1451'-5"
250V D.C SWGR
SWITCH GEAR ROOM
CONDENSATE RETURN TANK & PUMPS
CRITICAL TRANS
SUPERHEATER
EVAPORATOR
STAIR Nº 3
REMOVABLE
THROTTLE VALVE OPERATOR
STM GEN GAGE BD Nº 2
NITROGEN COOLING UNITS
DOOR
PLUG Nº 28
EL 1464'-0"
INTERMEDIATE HEAT EXCHANGERS
EL 1430'-7"
Nº 2A
Nº 2B
EL 1420'-0"
TERM BD CABINET
PRI SODIUM PUMP Nº 2 SPEED CONT CAB
PRIMARY SODIUM PUMP Nº 2
ACCESS LADDER
STAIR Nº 5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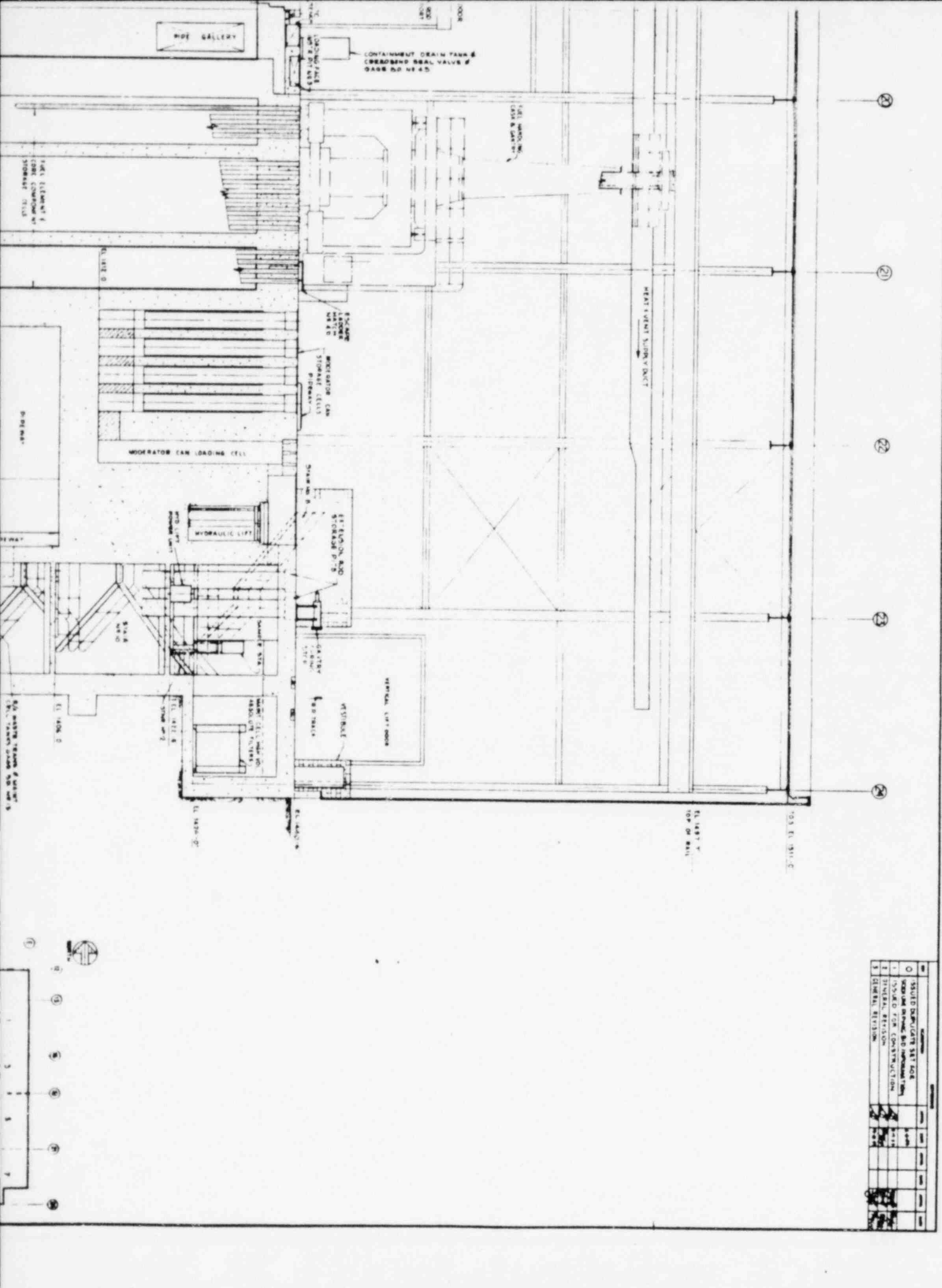
PIPEWAY
PIPE GALLERY
PIPE GALLERY
REACTOR



PIPE GALLERY
CONTAINMENT DRAIN TANK &
CROSSBEND SEAL VALVE &
GAGE BD NO 4.5
FUEL ELEMENT &
CORE COMPONENT
STORAGE CELLS
HEAT & VENT SUPPLY DUCT
MODERATOR CAN LOADING CELL
MODERATOR CAN
STORAGE CELLS
PIPEWAY
HYDRAULIC LIFT
EXTENSION ROD
STORAGE PITS
VESTIBULE
VERTICAL LIFT DOOR
ABSOLUTE FILTERS
TOP OF RAIL
21
22
23
24
ISSUED DUPLICATE SET FOR
ISSUED FOR CONSTRUCTION
GENERAL REVISION
GENERAL REVISION

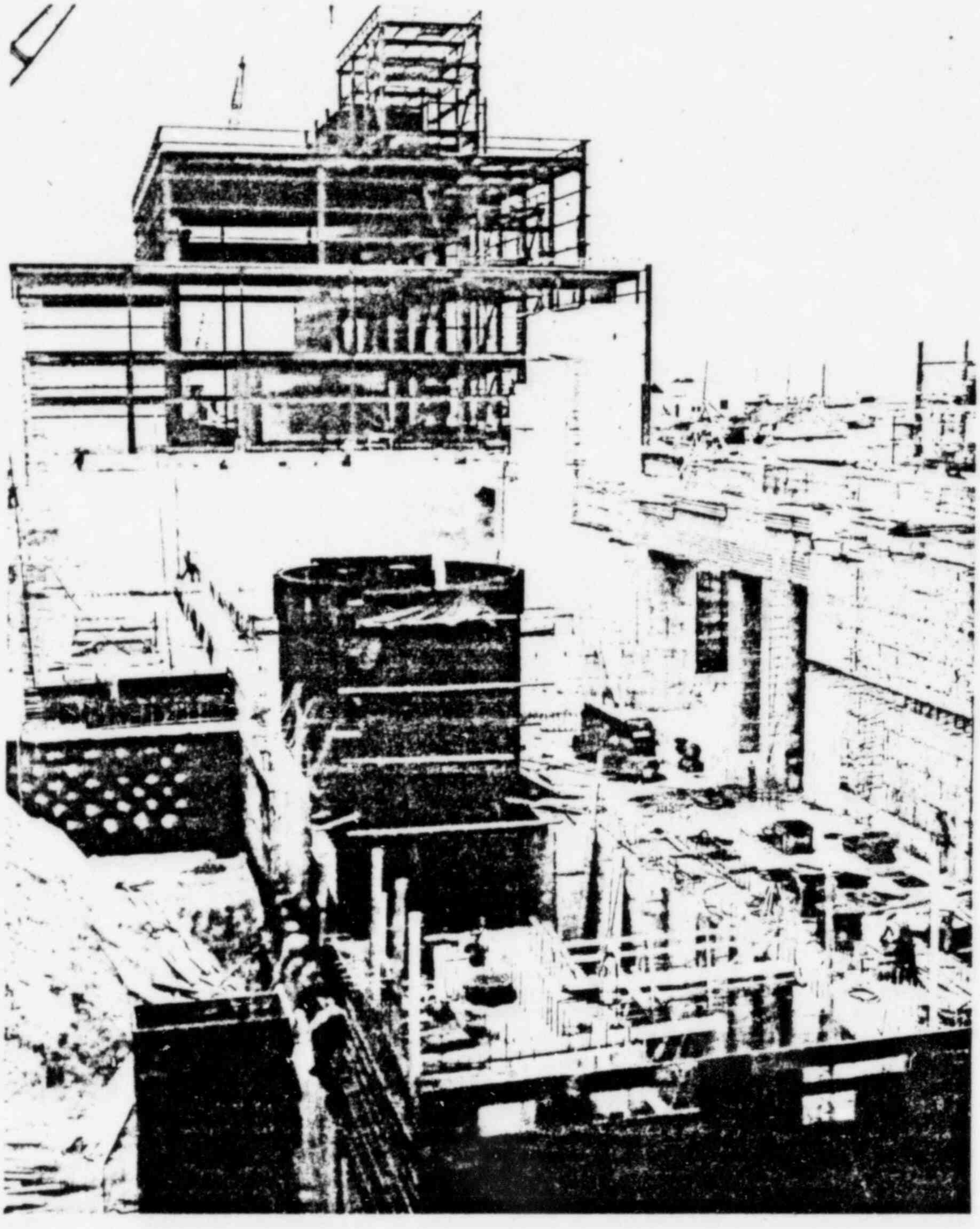


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Figure 2-8. Partially Completed Concrete, Core Liner in Place, and Structural Steel for Conventional Plant in Background

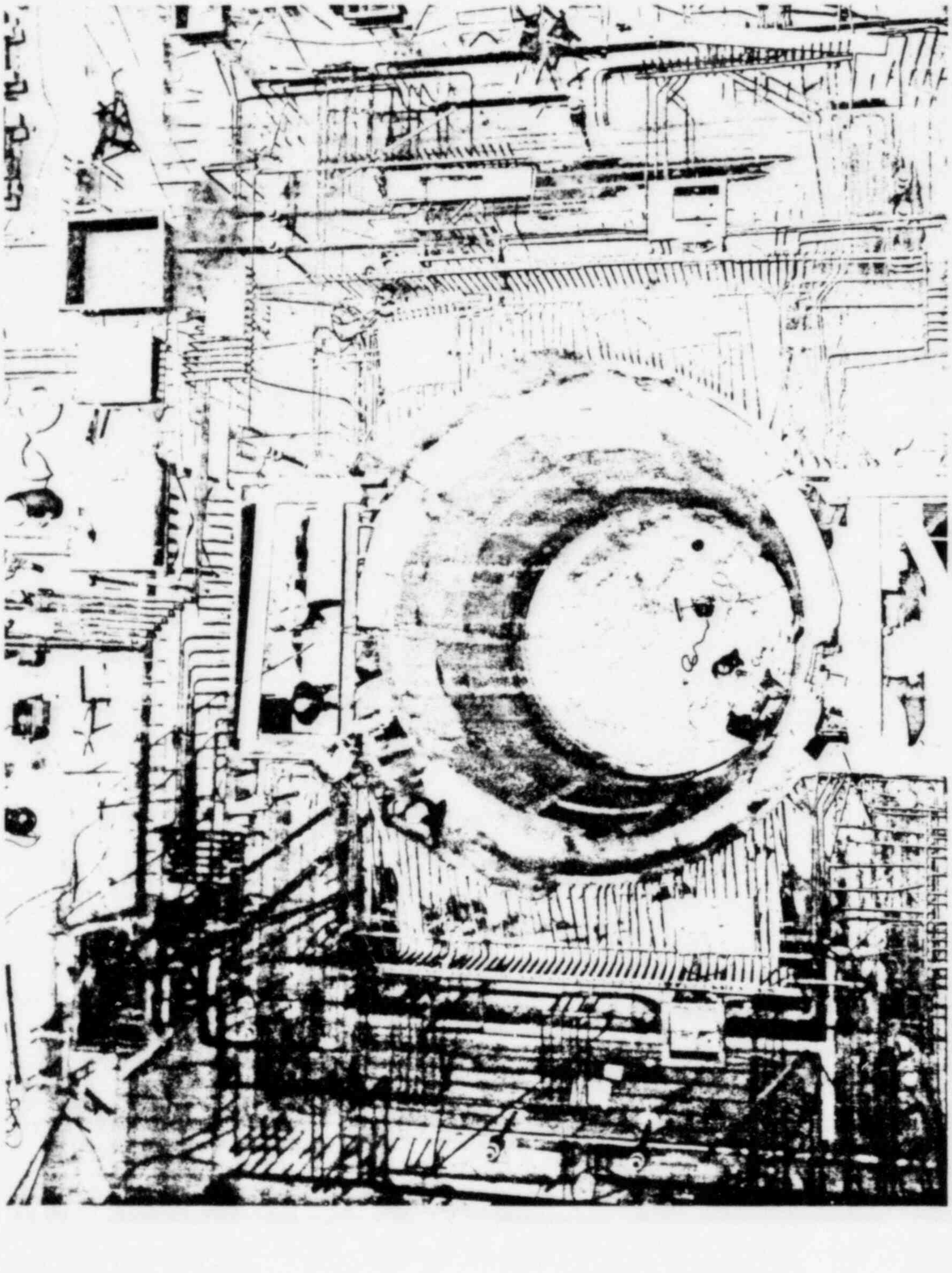

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Figure 2-9. Cavity-Liner Area

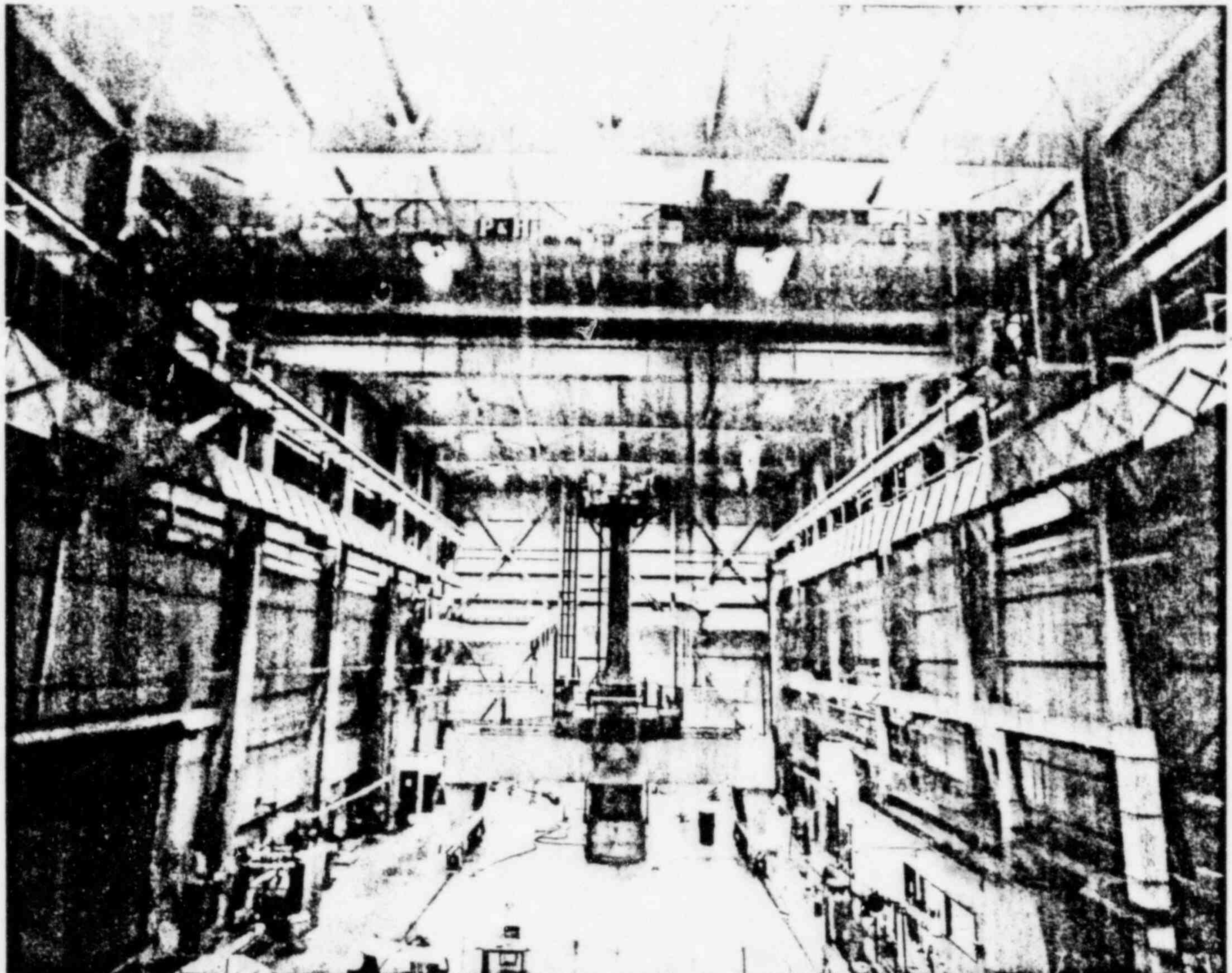

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Figure 2-10. Interior of HNPF Showing Fuel-Handling Machine and Structural Details (View towards south)

3.0 HISTORY OF OPERATIONS

Upon completion of the plant construction and preoperational testing of all systems, "wet criticality"* was achieved on August 25, 1962. The full 140-element loading of the core was then completed on October 10, 1962. Reactor operations were done by CPPD personnel with AI guidance. System checkout was then begun, low-power post-critical testing was completed, and plant official startup occurred on December 31, 1962. Figure 3-1 is a summary of HNPF reactor operating performance showing cumulative thermal power generated vs time, and reasons for various outages.

Failure of the stainless-steel cladding of the graphite moderator elements necessitated shutting down the plant for core repairs. By the end of September 1964 the cladding of seven moderator elements had failed resulting in saturation of these elements with sodium. These failures reduced core reactivity, caused perturbations of the flux, and locally depressed power; by these indications the failures were detected and the suspect failed elements identified.

The irradiated graphite expanded as much as 5% when saturated with sodium, therefore further cladding damage beyond the original cracking was experienced, as well as jamming of elements and control rod thimbles. Extensive failure analysis[2, 3] attributed the original cracking to high local-bending stress and low ductility of the cladding material.

Moderator element in manufacturing techniques had resulted in slightly irregular support to the cladding in the region of the upper head. The operating-pressure differential then caused the cladding to bend inward at the last point of support, resulting in surface bending stresses on the order of 40,000 psi. (Since Type 304 stainless steel above ~700°F has a ductility corresponding only to ~3% elongation at tensile failure, cracking for low $\sim 10^{-6}$ sec^{-1} strain rates occurred before the cladding could gain uniform support from the underlying structure through plastic deformation.) The fracture surfaces were intercrystalline, showing no deformation (therefore typical of "creep" or "time-to-rupture at constant stress and temperature" tests), with the crack progressing from the outside surface inward. The material was in no way defective, and neither corrosion nor weld-affected regions were contributing factors.

*Criticality after sodium addition to core

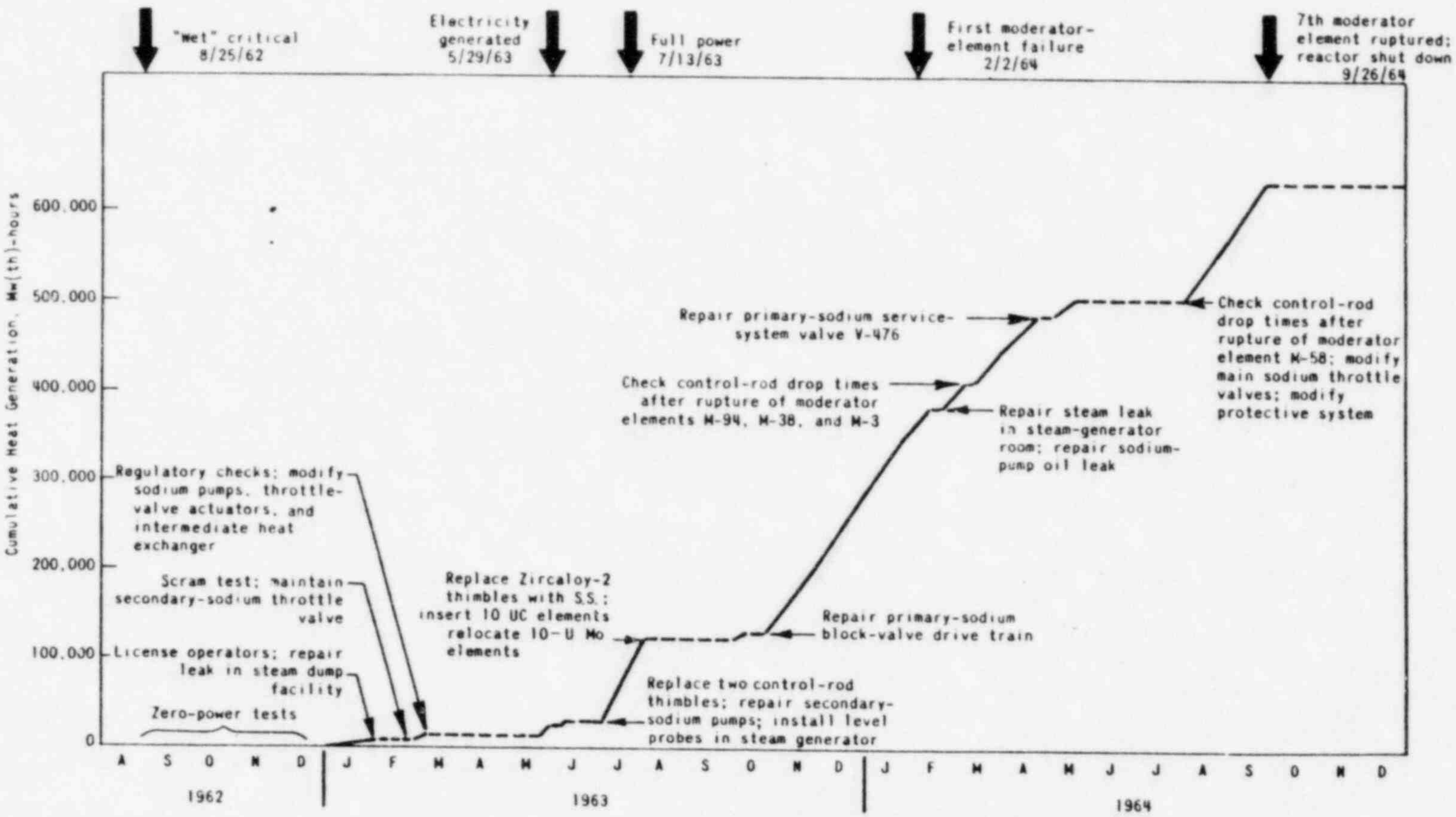


Figure 3-1. Summary of Hallam Reactor Operating Performance. (Before the last shutdown, moderator-element failures during 1964 seriously affected operations; a 1668-hr outage followed failure of moderator element M-58. Two other 1964 outages lasted 230 and 253 hr; 12 others lasted less than 100 hr each.)

The moderator elements were designed to accommodate 1% volume expansion of the graphite due to the possibility of a sodium leak. Actual expansion measured on leaky elements was as high as 5%. Further testing revealed the difference was due to a greater susceptibility of neutron-irradiated graphite to swelling from sodium.

Metallographic examination of the moderator cladding and some of the sodium piping revealed no evidence of corrosion or other damage by the sodium coolant after approximately 7000 hr of power operation, although some pitting on the order of 1/2-mil deep might have been expected, based on experimental studies of Na/stainless-steel interactions.

4.0 RETIREMENT PLAN DEVELOPMENT

The general scope for decontamination and dismantlement of the reactor plant was established by the AEC in a letter from the AEC to CPPD (see Appendix VI-2) with an enclosure which provided for:

1) Removal of all fuel from the site.
2) Removal from the site of all sodium that can be drained from the system.
3) Chemical reaction of residual sodium then remaining in the system, including all process lines, tanks, and the reactor vessel.
4) Disposal of radioactive residues from past reactor operations and from the decontamination activity (including those remaining after final processing of waste through the radioactive-waste disposal facility at the site), by burial within the reactor cavity or other adequately contained subsurface areas of the plant, controlled release to the environment, or removal from the site, depending upon the volumes and concentrations involved.
5) Decontamination of all components of the plant above the operating floor of the reactor building (the top of the IHX being considered as a continuation of the reactor building operating floor) such as the fuel-handling machine, the ventilation system, etc.; however any parts which are not to be further used or cannot be readily decontaminated will be buried in the reactor cavity or in other subsurface adequately contained areas of the reactor plant.
6) Prevention of release of contamination from and physical access to any subsurface portion of the plant, by permanently sealing all operating-floor-level penetrations of the entire facility (including all stairs and other entrances to the below-operating-floor-level of the reactor building); and removal of loose contamination on the surface of such areas as determined necessary on the basis of radiation surveys before permanently sealing access to these below-operating-floor areas.

7) Isolation of the reactor vessel by capping and sealing all pipes and pipeways thereto and by providing a permanent barrier against access from the operating-floor level to the reactor vessel and to the fuel-storage and IHX cells, by such means as welding existing plugs in place or covering openings with steel plate or approximately 6-in. layer of concrete; and decontamination and sealing of surface access (i.e. burial in place) to the underground tanks of the separate radioactive-waste disposal building or (if required for safety or economically preferable) removal of these tanks for burial in an appropriate subsurface area of the reactor vaults.

It was recognized that the foregoing AEC scope provided only broad definitions or descriptions of the measures to be taken to place the HNPF premises in a condition that was sufficiently safe from a radiation standpoint that no continuing AEC license would be required. To detail the requirement for accomplishing the Commission's objectives it was necessary to prepare an HNPF Retirement Plan to provide more specific definitions and descriptions of the activities to be performed and the related criteria and standards, as follows:

1) "Adequately contained subsurface areas" required interpretation on the basis of the materials to be contained and an evaluation of the adequacies of that containment.

2) It was recognized that "prevention of release of contamination" cannot be assured for the indefinite future; however, the Plan needed to set out the criteria to be met on a long-term basis for the safe release of minimal quantities of contamination to the surface or subsurface portions surrounding the plant. The extent to which "loose contamination on the surface of areas (below the operating floor) will be removed as determined necessary on the basis of radiation surveys" also required a determination of necessity based on criteria to be established and provided in the activity specification stated in the plan.

3) A definition of "permanent barriers" had to be provided in the plan that was consistent with the long-term disposition of this facility, and criteria were established as to "appropriate subsurface areas of the reactor vaults" for the on-site disposal of equipment and materials.

In preparing this HNPF Retirement Plan, attention was also given to the most effective means for accomplishing the following ends as requested by the AEC: (1) making the premises safe from a radiation standpoint including, as necessary, a removal of plant elements in whole or in part from the site, or their decontamination, disconnection, sealing, and containment as required for long-term disposition onsite; and (2) realizing the recoverable value of the remaining plant elements, by removing those determined economically advantageous for reuse elsewhere. To this end, six categories of material and equipment listed below were defined by the AEC, and these were to become the basic guide for disposition.

1) Category A, property authorized by AEC for shipment off-site except as scrap; such property to be removed from its present location and shipped off-site to AEC designated recipients.

2) Category B, property to be removed from its present location and shipped off-site as scrap.

3) Category C, property to remain in its present location at the HNPF.

4) Category D, property to be removed from its present location but whose ownership will be transferred to CPPD.

5) Category E, property to remain in its present location but whose ownership will be transferred to CPPD.

6) Category F, property which must be removed to accomplish the retirement of the HNPF, and which AEC authorizes to be removed from its present location pending determination of its ultimate disposition; such property to be provided with suitable protection against damage and temporarily stored until its final status as Category A, B, or D material is decided.

Development of the HNPF Retirement Plan involved numerous trade studies and feasibility investigations, with radiation, chemical, and physical safety of the operating personnel and long-range safety of the site and environs being the first consideration. As it turned out, a significant dollar value was identified for material and equipment to be reused after various degrees and processes of decontamination. For example the primary sodium would be reusable after allowing sufficient time for Na^{22} decay, the steam generators and certain other items such as the overhead cranes needed only to be reassembled to be ready for reuse.

The HNPF Retirement Plan, as developed by AI and approved by CPPD and AEC, identifies twelve activities as necessary and sufficient to meet all of the AEC criteria for retirement. These were written as individual activity specifications with the following titles:

1) Primary Sodium Disposal
2) Secondary Sodium Disposal
3) Irradiated Fuel Disposal
4) Unirradiated Fuel Disposal
5) Reaction of Residual Primary Sodium and Retirement of the Primary Sodium System
6) Reaction of Residual Secondary Sodium and Retirement of the Secondary Sodium System
7) Disposition of Contaminated and Irradiated Material
8) Reactor Isolation
9) Retirement of Auxiliary Systems
10) Securing of Isolation Structure
11) Retirement of the Radioactive Waste Facility
12) Final Closeout of the Facility

These are reproduced in entirety in Appendix I. This is also the order in which the retirement operations are reported herein; several activities were often carried out simultaneously. Some were interrupted to allow completion of a different activity as required by scheduling, effective use of manpower, and safety.

From each of these activity specifications, one or more detailed procedures were developed to show, on a step-by-step basis: how the work was to be done;

identification of all tools, materials, equipment; and processes to be used. These appear, as actually used, in Appendix II. Each of these required approvals from CPPD management, and a separate review and approval by the CPPD Safety Review Committee and AI and AEC.

The development of these specifications and procedures required very close cooperation between AI and CPPD; the AEC had final approval. The resulting organizational interface was basically as follows.

4.1 AI RESPONSIBILITIES

1) Prepare and recommend an overall plan which identifies the tasks to be undertaken for: (a) decontaminating HNPF in accordance with "AEC Plan for Decontamination of HNPF Premises," attached; and (b) disposing of plant components and equipment for use elsewhere, and decommissioning of the facility as AEC shall direct. This overall plan shall include a detailed sequence schedule which recognizes the inter-relationship between the various decontamination and decommissioning tasks and the needs to minimize the generation and distribution of radioactive wastes. The overall plan shall also recommend the engineering-and-safety criteria, and standards to be applied in carrying out the tasks identified therein.

2) Prepare outline procedures for accomplishing each task identified in 1) above, taking into account the radiation dose rates expected to be encountered. The procedures shall be adequate for use as a basis for CPPD preparation of the detailed working procedures.

3) Consult with CPPD in the preparation of Items 1 and 2 above in order to obtain their views and comments currently as those items are being developed.

4) Upon completion submit elements of the overall plan and outline procedures as they are developed to CPPD for review and comment with copies for information to AEC. Following resolution of CPPD comments AI shall submit the overall plan and outline-procedures to the AEC for approval by the contracting officer.

5) Act as a liaison point for the Commission with AEC contractors and others designated by the Commission to: receive components, equipment, and material from HNPF; work out removal, preparation for shipment, transportation, receiving, etc. requirements; and schedule.

6) Prepare jointly with CPPD an outline of a proposed technical report dealing with the decontamination and decommissioning of HNPF, and later participate in the preparation of such a report as AEC shall authorize.

4.2 CPPD RESPONSIBILITIES

1) Determine and advise AI and AEC of the existing radiation doserates in those locations in the plant for which such information is needed in AI's development of the overall plan and outline procedures for decontaminating and decommissioning in HNPF.

2) Provide views and comments of CPPD to AI while AI is developing the overall plan, and outline procedures referred to in Items 1 and 2 under AI Responsibilities.

3) Review and submit comments to AI with copies to AEC on the completed overall plan and outline of procedures as developed by AI.

4) Upon receipt from the contracting officer of AEC approval of elements of the AI-developed overall plan and outline procedures, establish detailed working procedures required to accomplish each task (using to the maximum extent practicable present CPPD operating procedures), and submit to AI for concurrence; the AEC approval to be obtained for these detailed working procedures will be indicated in the AEC action authorizing CPPD to carry out the approved AI outline procedures.

5) Execute the AEC authorized tasks (by CPPD personnel or subcontractors) in accordance with the approved CPPD detailed working procedures. CPPD to retain full responsibility for safety accomplishing all such authorized tasks.

6) Prepare jointly with AI an outline of a proposed technical report dealing with the decontamination and decommissioning of the HNPF and participate in the preparation of such a report as AEC shall authorize.

As indicated earlier, all aspects of safety during retirement operations as well as the long-range safety of the site were of paramount importance. Each of the activity specifications and each of the detailed procedures therefore took into account the requirements needed to minimize and safeguard against any radioactive, chemical, or physical hazard to people or property. The safety aspects were evaluated in "Final Summary Safeguards Report for the Hallam Facility, Supplement 5."[4] Some of the material contained therein is in answer to specific questions from the AEC on the control of hazards. For example, upon request, information was furnished in detail on the radiological safety condition expected when decommissioning was completed. Supplement 5 was the basis for AEC issuance to CPPD of the Dismantling Order set forth in Appendix VI-3.

Section 7.0 in this report, Health and Safety, deals with the actual final status of the facility with respect to safeguards at completion of the retirement program. A report titled "Final Status Report and Safety Analysis of the Hallam Nuclear Power Facility Site and Remaining Structures"[5] was prepared at the completion of the physical retirement activities. This report summarizes the retirement activities and indicates that accomplishment was in accordance with the safety requirements of the plan and the dismantling order.

5.0 PROGRAM ADMINISTRATION

As described earlier, the principals in the Hallam retirement were CPPD, AI, and the AEC. The basic responsibility for completing the work rested with the CPPD under contractual arrangements with the AEC, with AI providing engineering and management assistance. Since CPPD did not have all of the required labor skills available and little of the heavy equipment needed, a subcontract was made between CPPD and Stearns-Roger to provide the special capabilities needed. This resulted in the retirement organization shown in Figure 5-1. Stearns-Roger work was delineated by the requirements of the twelve activity specifications and the corresponding detailed procedures prepared by AI and approved by the AEC and CPPD. In addition Stearns-Roger provided structural designs for the structural strengthening of the floor slab north of the IHX and for the final weatherproofing cover. They also furnished

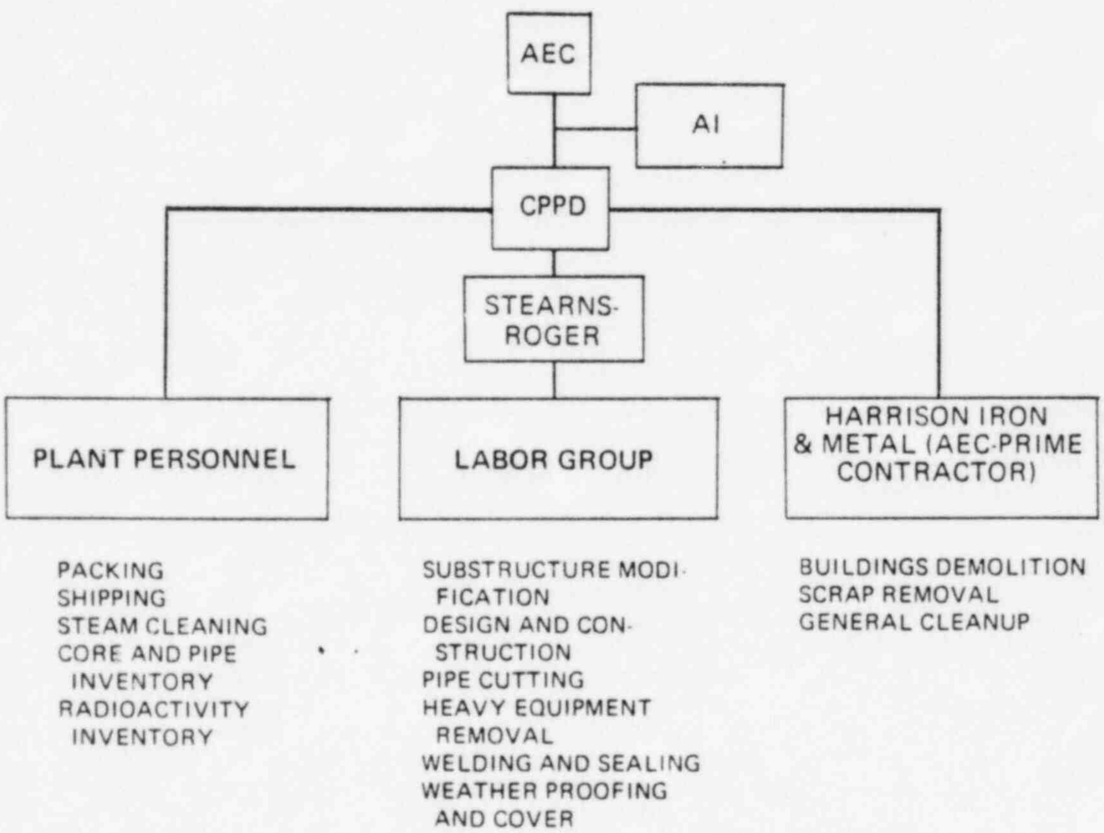


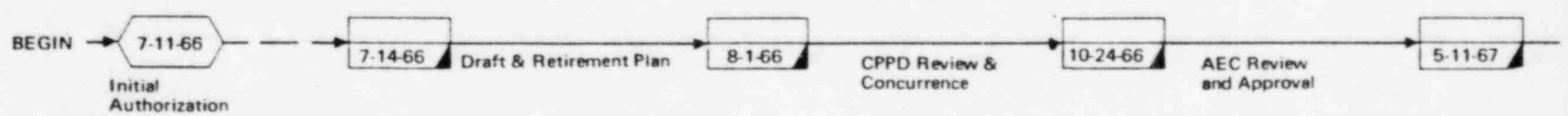
Figure 5.1. Retirement Organization

all of the skilled labor,* materials, and equipment for their contract. Activities such as the chemical reacting of residual sodium and disposition of irradiated and contaminated materials and of reactor fuel were done exclusively by CPPD personnel familiar with the hazards involved and skilled in implementing the safety procedures required. CPPD management of the program included: utilization of the HNPF Safety Review Committee, issuance of work permits and monitoring of hazardous locations or tasks, planning and coordination of work, and a preparation of schedules. The demolition work by Harrison Iron & Metal Co. was administered by CPPD even though the Harrison contract was made directly with AEC. Once the below-grade vaults were adequately sealed the reactor building, warehouse, calibration building, and moderator-assembly building were released to the Harrison Co. who then began dismantling the buildings.

During retirement AI maintained an on-site staff of three persons including the project manager. Several engineering specialists contributed as required to the retirement activities. These contributions supported the process experiments (for reacting sodium), preparation of process specifications, structural evaluation, radiation-inventory analysis, and other engineering studies. The number of persons at the peak of AI effort was approximately 12, the average was about 5. CPPD had a peak personnel load of 23 with an average of about 20.

Weekly meetings of AEC, AI, CPPD, and subcontractors were held to review the previous week's progress, map out and coordinate the next week's work, resolve problems, and check schedules. The overall schedule for the retirement program is shown in Figure 5-2.

*Five pipe fitters, seven iron workers, one millwright, seven carpenters, one sheet metal worker, one cement finisher, and ten laborers. A typical shift also included a field superintendent, a field accountant, an estimator, and a clerk typist.


BEGIN
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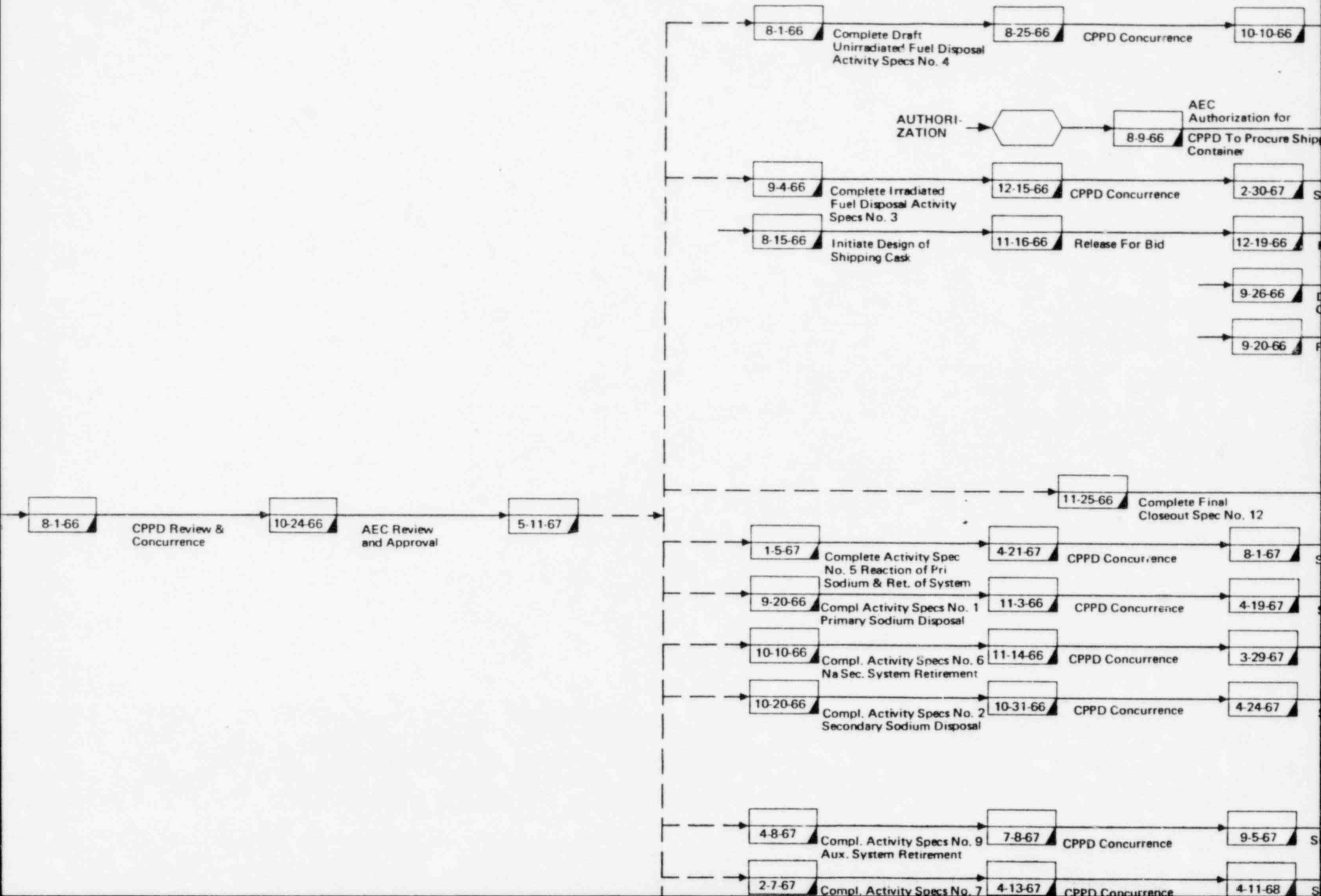
Initial
Authorization
7-14-66
Draft & Retirement Plan
8-1-66
CPPD Review &
Concurrence
10-24-66
AEC Review
and Approval
5-11-67



8-1-66
CPPD Review & Concurrence
10-24-66
AEC Review and Approval
5-11-67
8-1-66
Complete Draft Unirradiated Fuel Disposal Activity Specs No. 4
8-25-66
CPPD Concurrence
10-10-66
AUTHORI-ZATION
8-9-66
AEC Authorization for CPPD To Procure Shipping Container
9-4-66
Complete Irradiated Fuel Disposal Activity Specs No. 3
12-15-66
CPPD Concurrence
2-30-67
8-15-66
Initiate Design of Shipping Cask
11-16-66
Release For Bid
12-19-66
9-26-66
9-20-66
11-25-66
Complete Final Closeout Spec No. 12
1-5-67
Complete Activity Spec No. 5 Reaction of Pri Sodium & Ret. of System
4-21-67
CPPD Concurrence
8-1-67
9-20-66
Compl Activity Specs No. 1 Primary Sodium Disposal
11-3-66
CPPD Concurrence
4-19-67
10-10-66
Compl. Activity Specs No. 6 Na Sec. System Retirement
11-14-66
CPPD Concurrence
3-29-67
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Compl. Activity Specs No. 2 Secondary Sodium Disposal
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Compl. Activity Specs No. 9 Aux. System Retirement
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CPPD Concurrence
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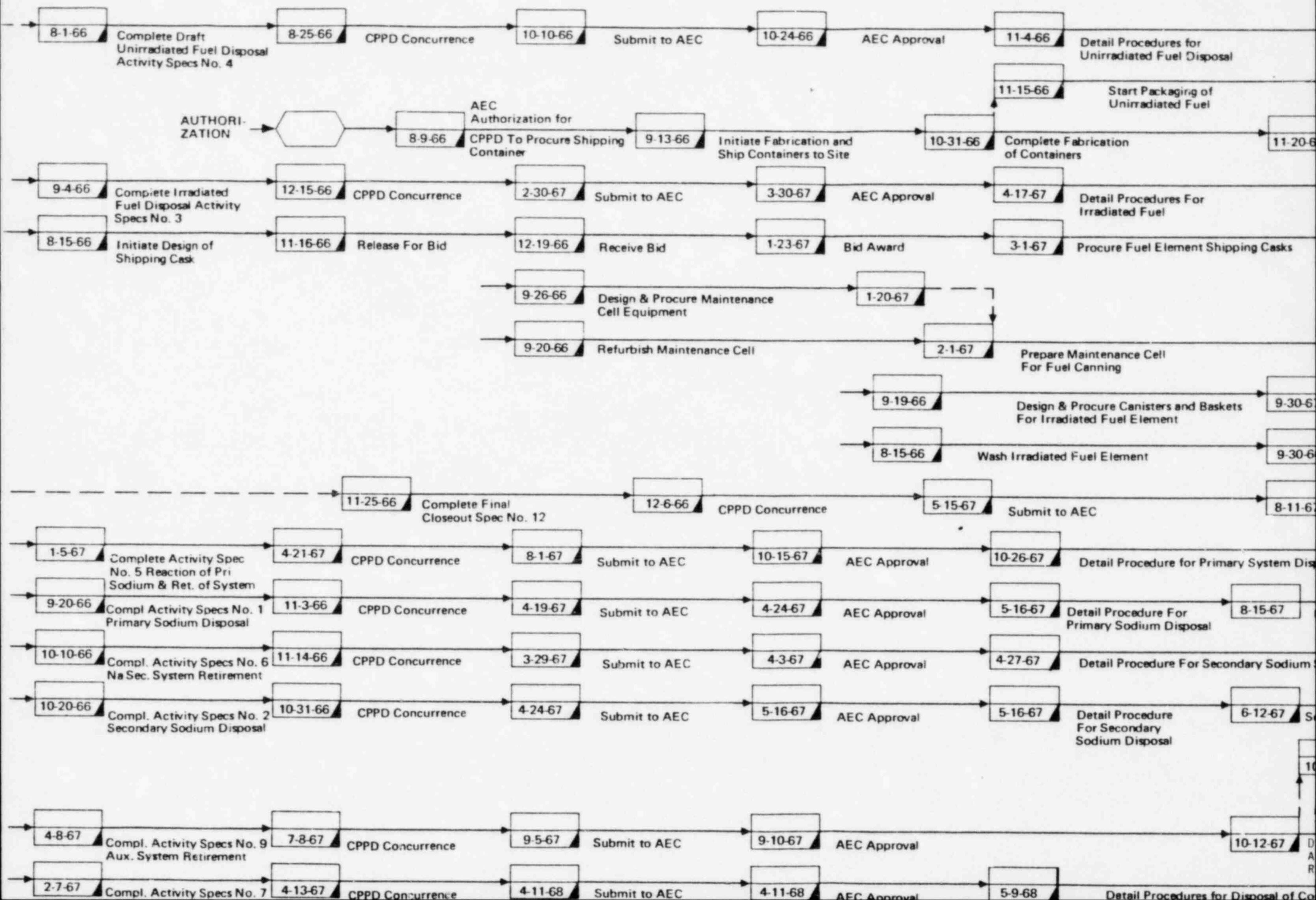
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Complete Draft Unirradiated Fuel Disposal Activity Specs No. 4
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CPPD Concurrence
10-10-66
Submit to AEC
10-24-66
AEC Approval
11-4-66
Detail Procedures for Unirradiated Fuel Disposal
11-15-66
Start Packaging of Unirradiated Fuel
AUTHORI-ZATION
8-9-66
AEC Authorization for CPPD To Procure Shipping Container
9-13-66
Initiate Fabrication and Ship Containers to Site
10-31-66
Complete Fabrication of Containers
9-4-66
Complete Irradiated Fuel Disposal Activity Specs No. 3
12-15-66
CPPD Concurrence
2-30-67
Submit to AEC
3-30-67
AEC Approval
4-17-67
Detail Procedures For Irradiated Fuel
8-15-66
Initiate Design of Shipping Cask
11-16-66
Release For Bid
12-19-66
Receive Bid
1-23-67
Bid Award
3-1-67
Procure Fuel Element Shipping Casks
9-26-66
Design & Procure Maintenance Cell Equipment
1-20-67
9-20-66
Refurbish Maintenance Cell
2-1-67
Prepare Maintenance Cell For Fuel Canning
9-19-66
Design & Procure Canisters and Baskets For Irradiated Fuel Element
8-15-66
Wash Irradiated Fuel Element
11-25-66
Complete Final Closeout Spec No. 12
12-6-66
CPPD Concurrence
5-15-67
Submit to AEC
1-5-67
Complete Activity Spec No. 5 Reaction of Pri Sodium & Ret. of System
4-21-67
CPPD Concurrence
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Submit to AEC
10-15-67
AEC Approval
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Detail Procedure for Primary System Dis
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Detail Procedure For Secondary Sodium
10-20-66
Compl. Activity Specs No. 2 Secondary Sodium Disposal
10-31-66
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Detail Procedure For Secondary Sodium Disposal
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Submit to AEC
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AEC Approval
10-12-67
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Compl. Activity Specs No. 7
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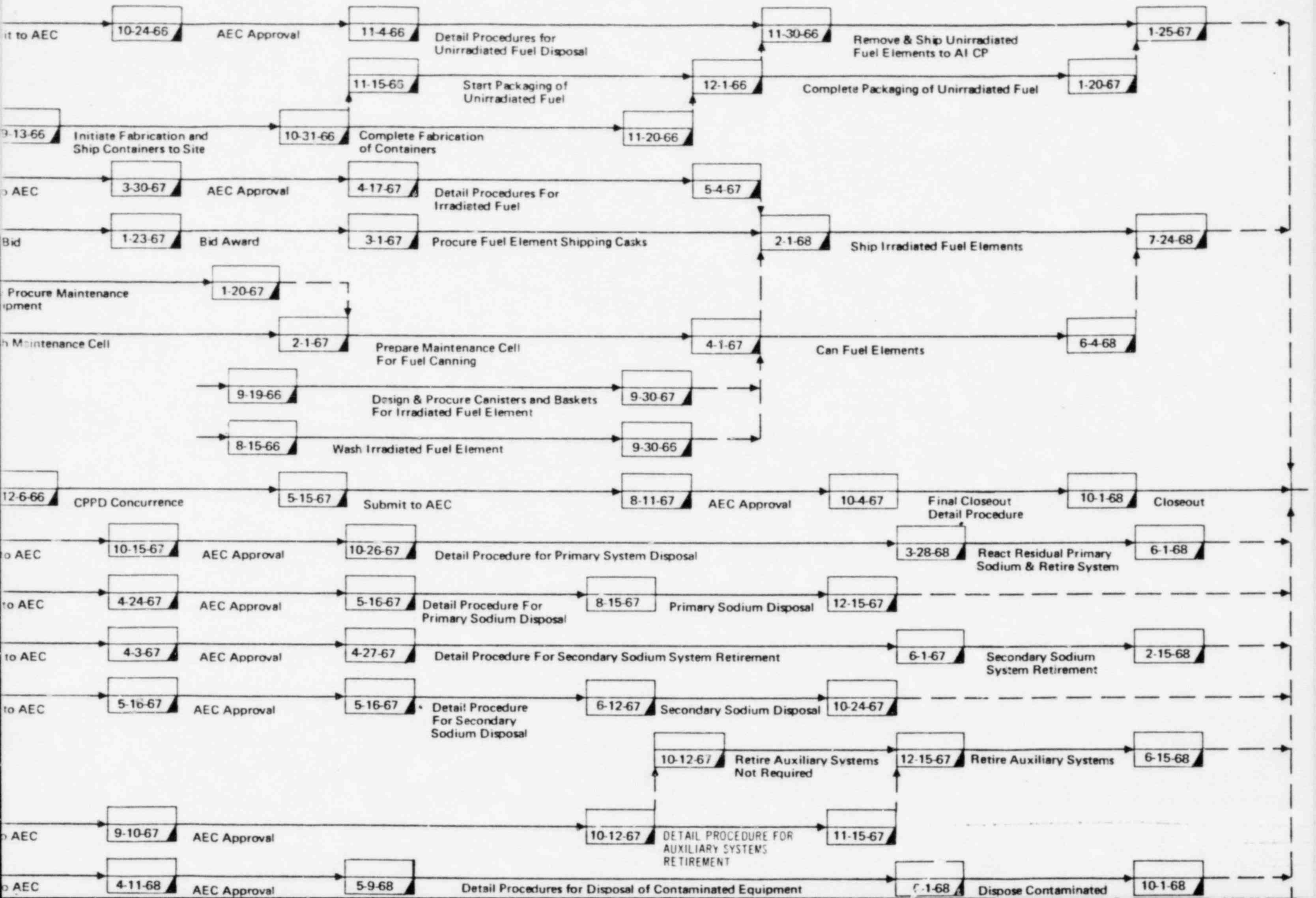

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Detail Procedures for Unirradiated Fuel Disposal
11-30-66
Remove & Ship Unirradiated Fuel Elements to AI CP
1-25-67
11-15-66
Start Packaging of Unirradiated Fuel
12-1-66
Complete Packaging of Unirradiated Fuel
1-20-67
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Initiate Fabrication and Ship Containers to Site
10-31-66
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Detail Procedures For Irradieted Fuel
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Final Closeout Detail Procedure
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Detail Procedure for Primary System Disposal
3-28-68
React Residual Primary Sodium & Retire System
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5-16-67
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AEC Approval
4-27-67
Detail Procedure For Secondary Sodium System Retirement
6-1-67
Secondary Sodium System Retirement
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Detail Procedure For Secondary Sodium Disposal
6-12-67
Secondary Sodium Disposal
10-24-67
10-12-67
Retire Auxiliary Systems Not Required
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Retire Auxiliary Systems
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AEC Approval
10-12-67
DETAIL PROCEDURE FOR AUXILIARY SYSTEMS RETIREMENT
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AEC Approval
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Detail Procedures for Disposal of Contaminated Equipment
Dispose Contaminated
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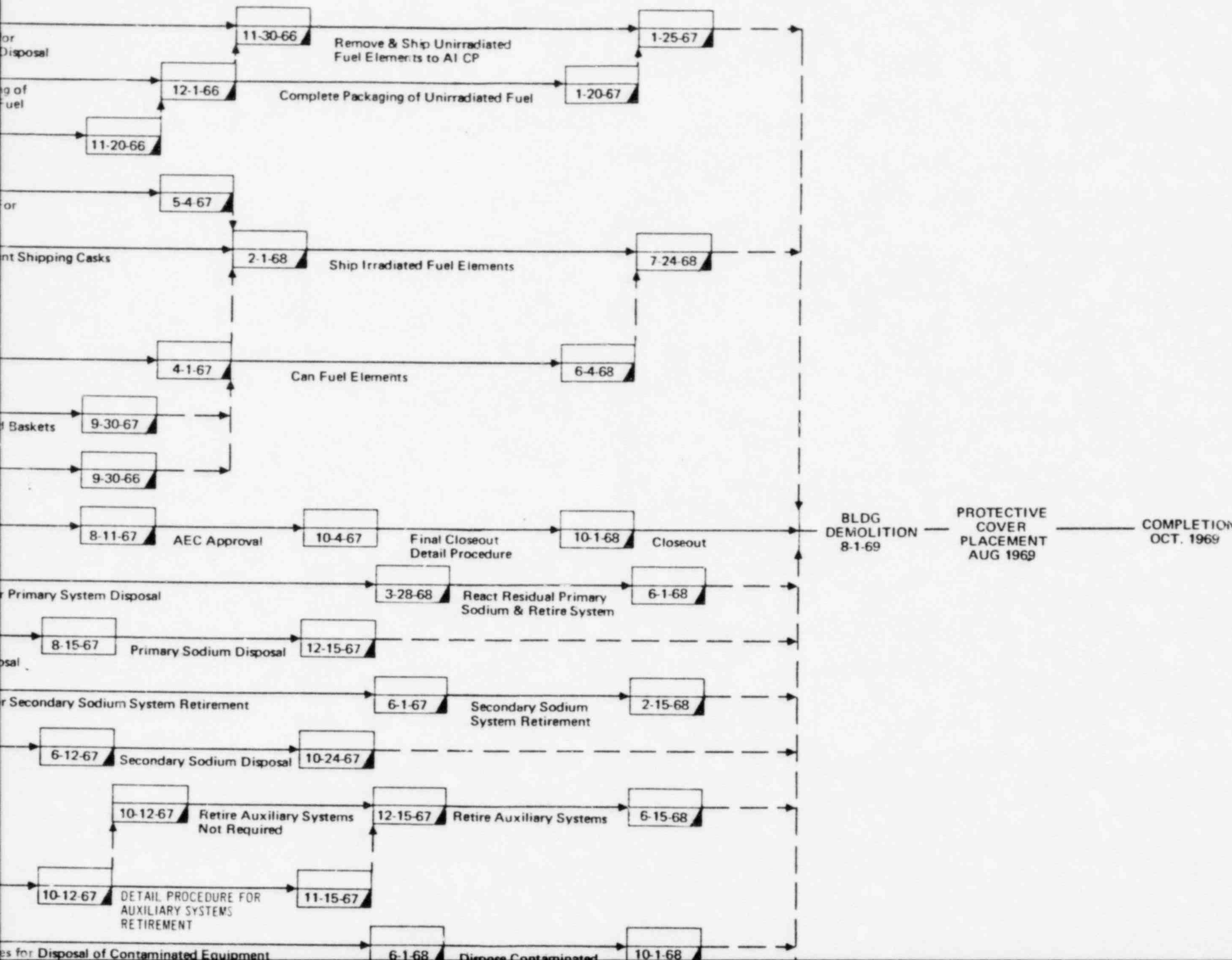

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6-12-67
Secondary Sodium Disposal
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10-12-67
Retire Auxiliary Systems
Not Required
12-15-67
Retire Auxiliary Systems
6-15-68
10-12-67
DETAIL PROCEDURE FOR
AUXILIARY SYSTEMS
RETIREMENT
11-15-67
es for Disposal of Contaminated Equipment
6-1-68
10-1-68

6.0 ACTIVITY IMPLEMENTATION

6.1 PRIMARY AND SECONDARY-SODIUM DISPOSAL

In preparation for retirement of the HNPF reactor the sodium from the primary and secondary systems was drained into eight existing underground fill-and-drain tanks; five were for primary sodium and three for secondary sodium. Each set of primary or secondary tanks was in a concrete vault with a hatchway leading to the main floor. The primary tanks were each 21-ft 3-in. long, 12-ft OD, and 2150 ft^3 capacity. The secondary tanks were each 14-ft 5-in. long, 12-ft OD, and 1375 ft^3 capacity. Each was equipped with electric heaters, thermocouples, and thermal insulation to permit maintaining stored sodium above its melting temperature (208°F).

After draining sodium from the primary and secondary heat-transfer systems into the tanks, the average loading was, for the primary tanks 110,000 lb (1932 ft^3) each, and for the secondary 76,600 lb (1330 ft^3) each. There was no significant radioactivity in the secondary sodium; the specific activity of the primary sodium was approximately 0.025 μci/gm (October 1967). The level of activity was low enough to permit safe direct personal contact with the tanks. The source of this activity was Na^{22} which has a 2.58 year half life. A chemical analysis of the primary sodium for carbon content indicated an impurity of 21 ppm (average of 2 samples). The analysis of the secondary sodium showed an impurity of 13 ppm (average of 3 samples).

Disposal of the sodium from its underground locations at the HNPF site required investigation of alternate means of disposal and consideration of their feasibility, time requirement, and cost. The alternatives considered were:

1) Chemical reaction of the metallic sodium to a passive form (Na_2CO_3 or NaOH) at the site or elsewhere, subsequent disposal of the reaction products in the underground vaults or packaging, and shipping the reaction products off-site for disposal,

2) Draining the metallic sodium into drums and shipment off-site for disposal, and

3) Shipment of the metallic sodium off-site in the existing tanks by

a) Removing the tanks, with their sodium content, from the underground vaults and loading them onto railroad flatcars, or

b) Draining the tanks into the plant sodium system, one at a time, then loading the empty tanks onto railroad flatcars and refilling them.

Evaluation of the alternatives resulted in the selection of 3(a) above. It was ascertained that both the tanks and the sodium were potentially reuseable elsewhere in the AEC programs; consequently the most economically feasible disposal consisted of shipping the sodium solidified in the tanks to storage pending further reuse.

In preparation for such shipment the tank heaters were shut off and the sodium allowed to cool and solidify. The five primary tanks were shipped to the AEC site at Richland, Washington, and the three secondary tanks to the AEC Liquid Metal Engineering Center (LMEC) in Santa Susana, California. All were shipped with their electric heaters and thermal insulation substantially intact.

Removal of the tanks and shipment by rail required reinforcing of the tank saddle supports, the addition of lugs and lifting eyes to permit handling of the loaded tanks with cranes, and anchoring to railroad flatcars. A special shipping permit was required because the tanks did not meet the requirements of Section 73.206 of the ICC Regulations for Transportation of Explosives and Other Dangerous Articles. Plans were also made for the type of railroad flatcars to be used, routing, and speed restrictions because the tanks constituted an oversized load with a high center of gravity. A detailed description of the removal and shipping plans is given in Reference 9.

After the plans for removal and shipment of the sodium were completed and reviewed, activity specifications were written by AI describing the methods to be used and the requirements to be met during operations. Activity Specification No. 1 covers primary-sodium disposal, and Activity Specification No. 2 covers secondary sodium disposal. A copy of the specifications is in Appendix I.

The initial plans recommended removal of the sodium tanks from the vault via the east side of the building by use of the building crane (25-ton rating) and one mobile crane. These recommendations were based on the limited availability of mobile cranes in the Hallam area at that time, and the marginal load-carrying capacity of the floor over the secondary-sodium tanks. Subsequent investigations before writing of the activity specifications indicated that mobile cranes of adequate capacity could be made available at Hallam, and that it was feasible and more economical to shore up the floor over the secondary tanks to support the combined crane-tank load.

Following AEC approval of the activity specifications, Retirement Detailed Procedures (RDP's) were prepared by CPPD. These procedures described in detail each step of the operations required to prepare the sodium tanks for removal and to transfer them from the vaults to railroad flatcars. Two detailed procedures were written for each of the two activity specifications. These were as follows:

Activity Specification No.	RDP No.	RDP Title
1	1-1	Isolation of the Primary-Fill Tanks
	1-2	Removal of Primary Sodium Fill Tanks and Transfer to Railroad Cars
2	2-1	Isolation of the Secondary-Fill Tanks
	2-2	Removal of Secondary-Sodium Fi.. Tanks and Transfer to Railroad Cars

The RDP's included sequence of events, identification of valves to be opened or closed, identification of lines to be cut and cut locations, precautions to be observed, and spaces for sign-off verifying that specific operations had been accomplished according to the procedure.

Removal of the sodium tanks proceeded as planned. After the tanks were isolated and the saddle supports were reinforced, the tanks were lifted out of the underground vaults and loaded onto special hydrocushioned suspension railroad cars. Anchoring of the tanks to the railroad cars (see Figure 6-1) was accomplished in accordance with tie-down plans furnished by AI. The tie-down arrangement was designed to withstand a 1.0-g acceleration in the fore and aft directions.

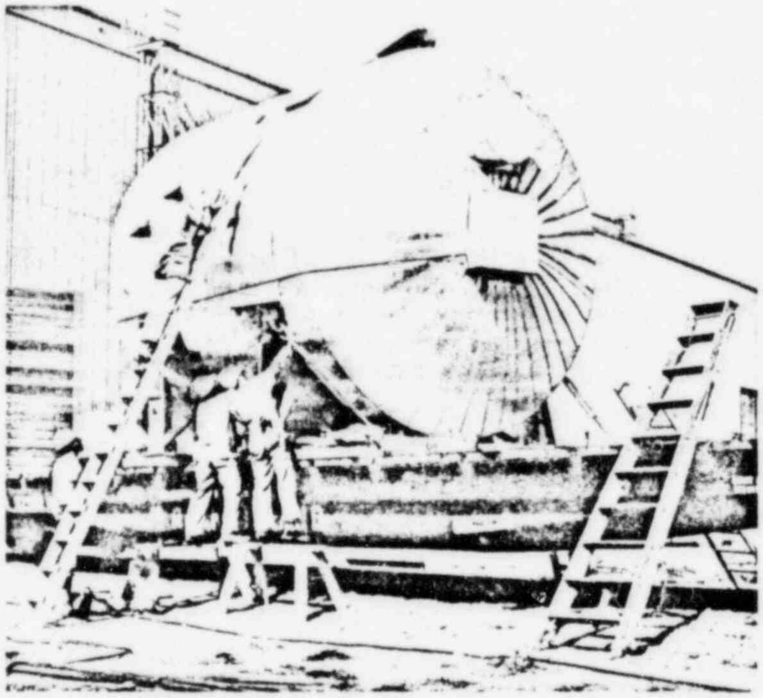

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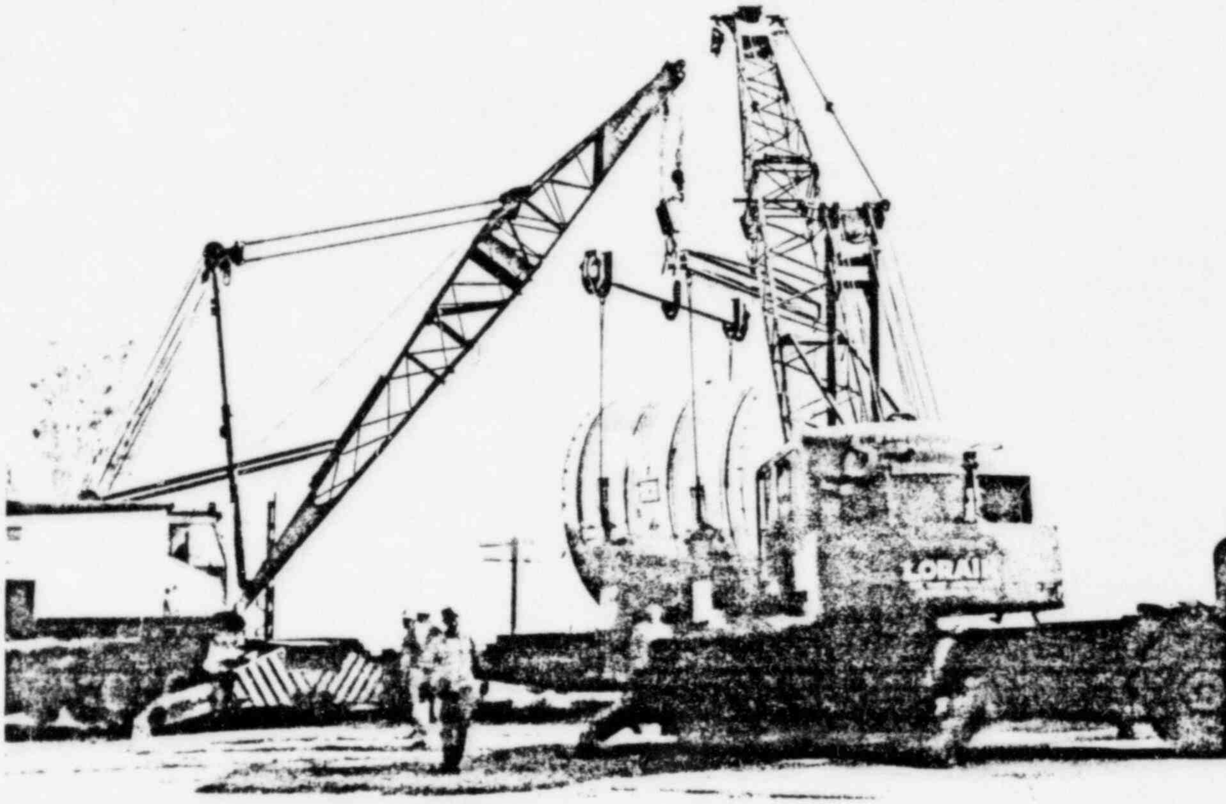
Figure 6-1. Secondary-Sodium Storage Tank Being Anchored to Railroad Flatcar

The three secondary-sodium tanks were removed first and shipped to the AEC-LMEC in California, Figure 6-2. The three flatcars and tie-down equipment were then returned to Hallam. For shipment of the five primary tanks two additional cars were obtained. The cars for shipment of the primary tanks were loaded with ballast to compensate for the high center of gravity for the car-tank combination. Shipment of the primary tanks to Richland, Washington was via special train with a two-man escort from AI. Duties of the escort were to check the tie-downs during stops, monitor train speed, and provide special instructions for handling of the sodium in the event of an accident. The escort rode in a special caboose, separate from the train crew caboose, and stayed with the sodium-laden cars throughout the trip. Train speeds were monitored by clocking the time required to pass 1/4-mile markers along the trackside. Reference 9 gives a detailed description of the moving and shipping operations.



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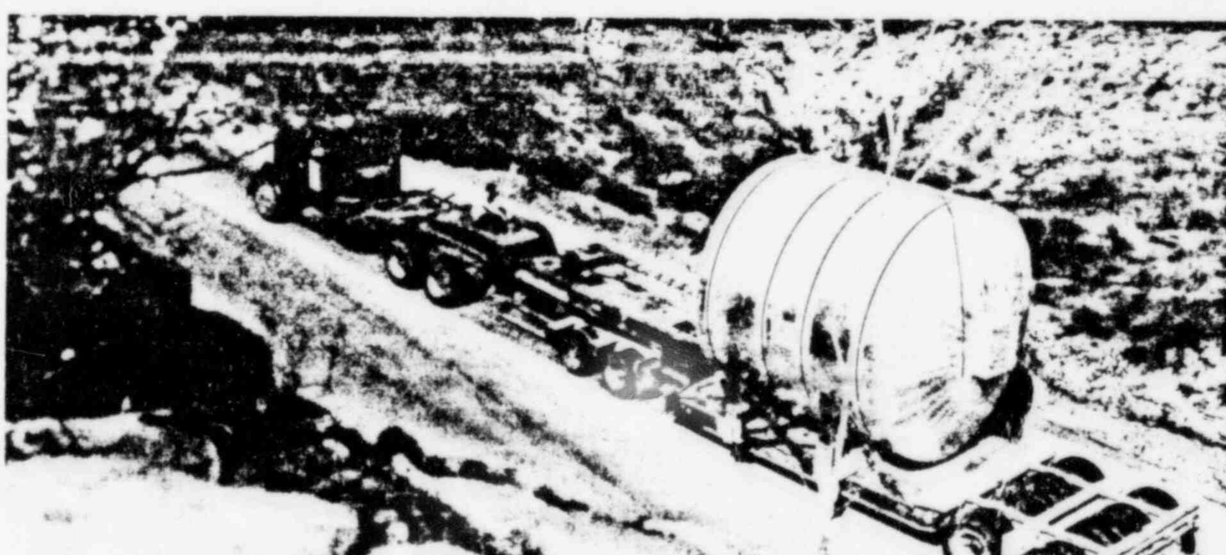

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Figure 6-2. Unloading Secondary-Sodium Tank, and Truck Transport to Santa Susana

6.2 DISPOSAL OF IRRADIATED FUEL

Removal of irradiated fuel was an important activity during retirement but was not unique to retirement, as fuel removal for reprocessing is a normal operating activity for any nuclear power plant. When all fuel materials had been shipped the major source of radiation was removed from the site and the potential criticality hazards were eliminated. Evaluation studies to identify and evaluate equipment, removal procedure time, and costs of packaging and shipping the irradiated fuel to an AEC designated reprocessing site began in May 1966.

Factors considered were the requirements of AEC Regulations (10CFR, Part 71) for "Packaging of Radioactive Materials for Transportation," and the handling requirements of the Atomic Energy Commission's Savannah River Plant, Aiken, South Carolina. Several handling methods were considered. These included:

1) Shipping the fuel elements intact, after removal from the fuel hanger rod and process tube, to the Atomics International Hot Laboratory where the fuel slugs would be removed from the stainless-steel cladding and sodium bonding, then repackaged into smaller containers for shipment to Savannah River for reprocessing. The same process was also considered for use in the maintenance cell at the HNPF
2) Removal of the sodium reservoir from each fuel bundle by use of the HNPF maintenance cell, then encapsulating the remainder of the fuel bundle, which included the fuel slugs, stainless-steel cladding, and sodium bond, in a double canister.
3) Removal of the fuel element from the hanger rod and process tube, then encapsulation of the complete fuel bundle inside two concentric canisters, both canisters' final closures to be welded remotely in the HNPF maintenance cell.

This last method was adopted.

The irradiated-fuel shipping equipment available to the AEC was evaluated for possible use, by modification, in shipping the HNPF irradiated fuel elements. No equipment was suitable without extensive modification. AI then recommended that a six-element shipping cask be constructed.

The AEC selected AI to design the fuel canisters and fuel shipping cask. After the design was approved by the AEC, AI was selected to fabricate the fuel canisters, and Allied Engineering Company of Alameda, California received the contract to build two six-element fuel-shipping casks and thirteen fuel-canister carrier baskets. The fuel-shipping cask, see Section 8.0, was designed to be mounted on a railroad flatcar for rail shipment. The extra canister baskets were to facilitate the encapsulation of fuel elements.

CPPD contracted with AI to: provide supervision; refurbish the HNPF maintenance cell for the special double-canistering of the irradiated fuel elements; prepare the activity specification and detailed procedures for such canistering; instruct certain CPPD personnel in the special maintenance cell operations involved in the double-canistering; and provide assistance in the design, procurement, and installation of the additional canistering equipment required.

The maintenance cell had three operating levels, Figure 6-3. The first operating level was used for positioning the fuel element, Figure 6-4, into the shipping canister. The second operating level was the work station equipped to do the welding operations necessary to encapsulate the fuel elements in the double shipping canister. The third operating level was primarily used for tool storage. The regular fuel-handling equipment installed in the maintenance cell at construction was supplemented by special equipment designed to permit double-encapsulation of the irradiated fuel elements (Figure 6-5).

The remote-welding of the fuel canisters required the designing of special welding equipment and controls to meet the limited handling features of the HNPF maintenance cell and the weld-integrity requirements of the AEC Savannah River Plant. A digital controller was designed to control the amperage, voltage, and torch speed of the weld head. The weld head was designed to weld two different diameters, both inner and outer canisters. The remote welder was designed and fabricated by Rytek, Inc. of Santa Fe Springs, California to AI specifications. The welding equipment was installed in the HNPF maintenance cell and test welds were made for conformance to the requirements of the E. I. duPont Company (operator of the AEC Savannah River Plant).

After examination of the welds, welding equipment, and welding procedures and a visit to the HNPF site by representatives of the AEC, SRO, and the duPont

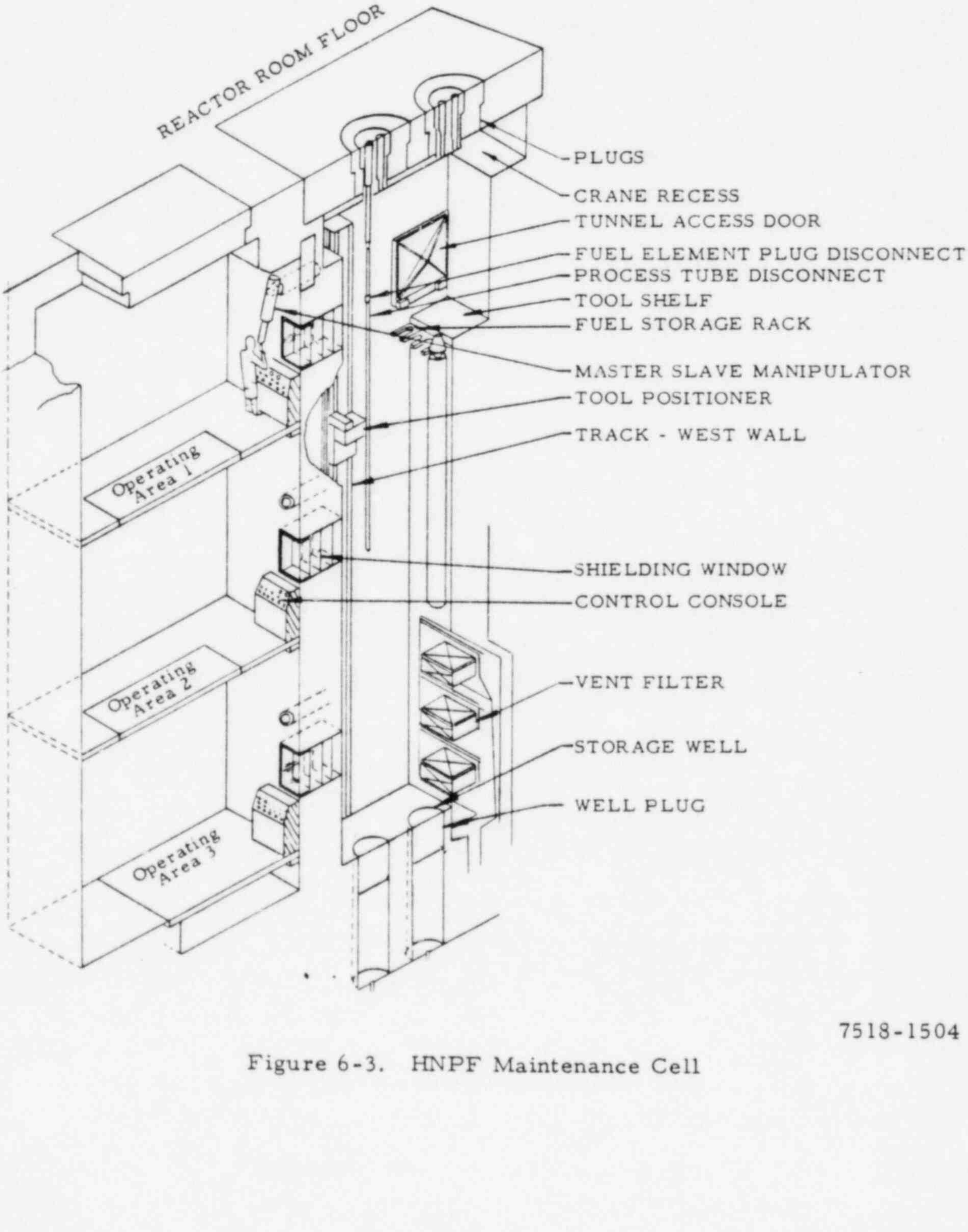

Figure 6-3. HNPF Maintenance Cell

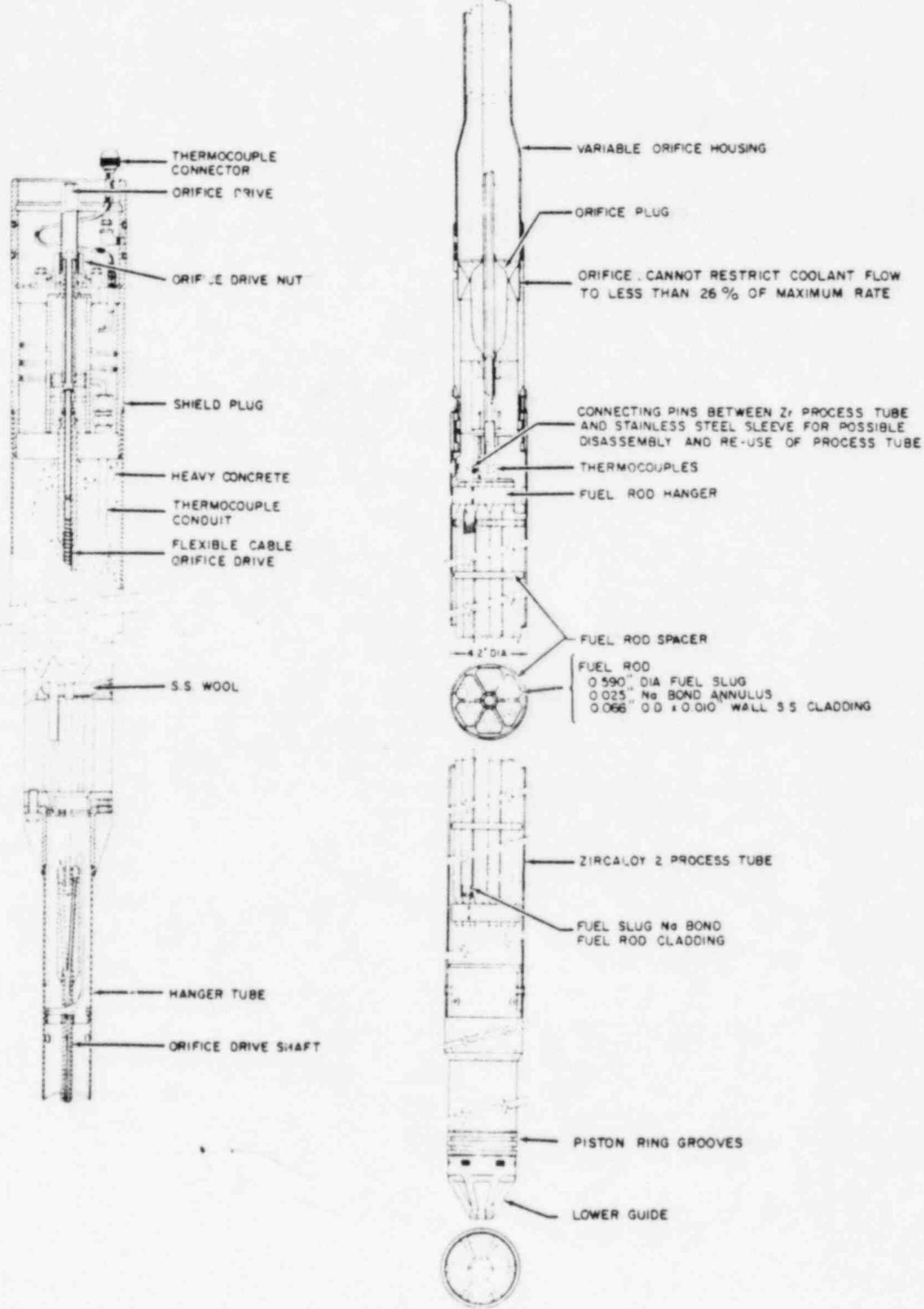

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Figure 6-4. HNPF Fuel-Element Assembly

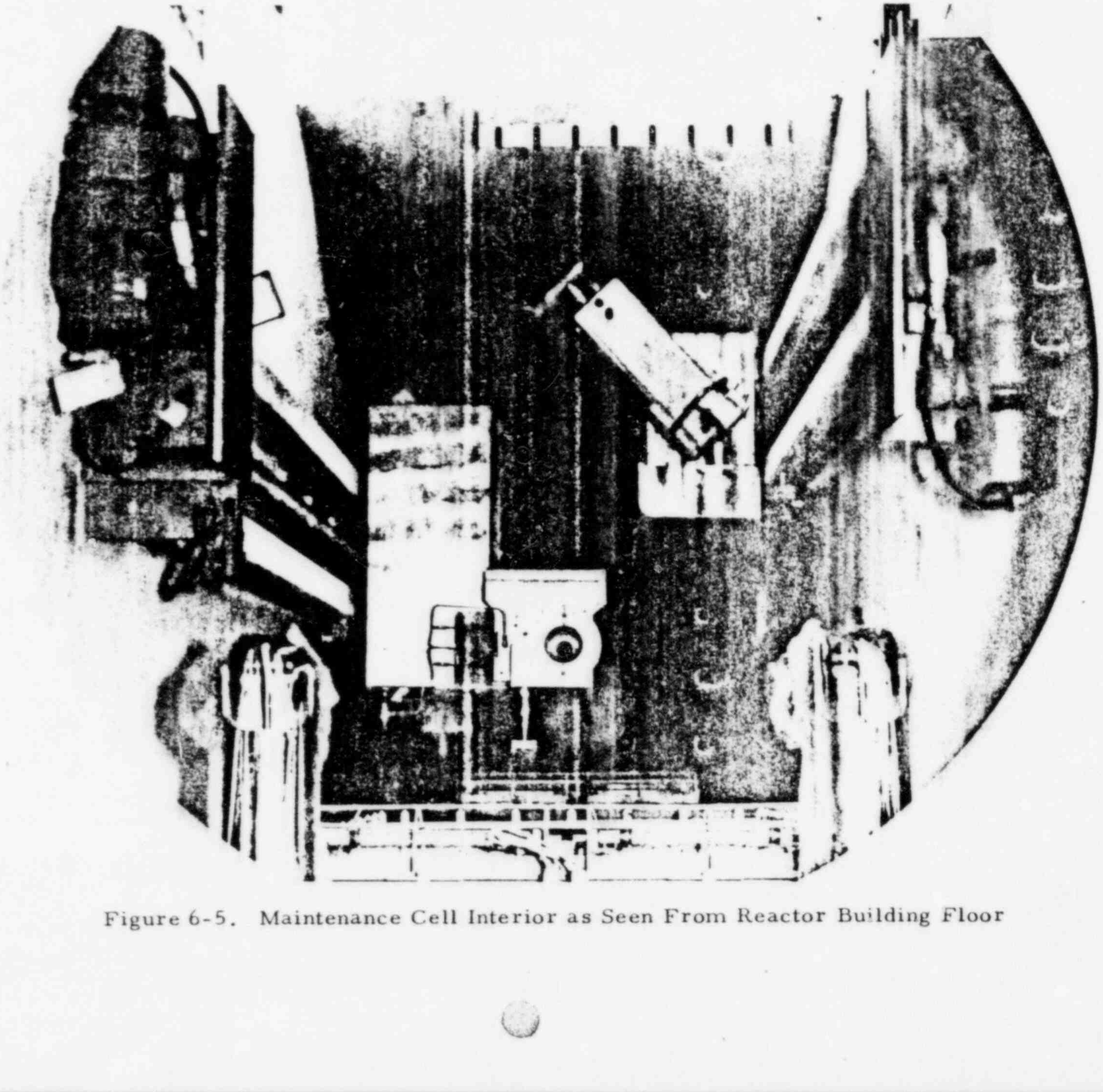


Figure 6-5. Maintenance Cell Interior as Seen From Reactor Building Floor

Company, the process was formally approved. The concern of the AEC and duPont Company was that the HNPF irradiated fuel was to be stored under water at the SRP and that the HNPF fuel, containing a sodium bond, was a potential hazard. The double-containment by use of two canisters was further protection. Inspection of the fuel-canister closure welds were visual; however, a chart record was made of the voltage and amperage used for each weld. If a deviation of voltage from the established normal was experienced, another weld pass was made. A typical volt-current chart is shown in Figure 6-6. After each group of six fuel elements was encapsulated a test canister without fuel was welded and removed from the maintenance cell for detailed inspection. This included a mass-spectrometer-type leak check using helium and a calibrated leak of 1×10^{-7} atm cc/sec, and micro-inspection (100X) of the weld, heat-affected zone, and parent metal. Sections of the weld were then sent to Savannah River for inspection and acceptance.

Before beginning fuel encapsulation a criticality study was made of the entire procedure to show that the technique was safe. This verified again that the maintenance cell could not hold enough elements to produce criticality assuming a complete flooding of the cell with water, which was the "worst case" postulated.

Intact irradiated fuel-element assemblies (except for the U-C) were steam-cleaned in the HNPF wash cells before encapsulation for shipment. The U-C elements were not steam-cleaned because of the remote possibility of occurrence of an uncontrolled water-carbide reaction. The U-C elements were canistered and shipped with the U-Mo elements to the Savannah River Plant. The encapsulation of U-10 Mo fuel was started July 11, 1967 and continued through August 30 by which time 54 irradiated fuel elements had been encapsulated. The further encapsulation of irradiated fuel elements was then temporarily discontinued because all storage facilities were filled and the fuel-shipping casks had not been delivered.

The fuel-shipping casks arrived at the HNPF January 29, 1968; after an operational checkout they were loaded with fuel and the first shipment departed February 1. The HNPF maintenance cell was returned to the operation of encapsulating irradiated fuel on February 6 and the operation continued through June 4, at which time the 150th irradiated fuel element was encapsulated. The

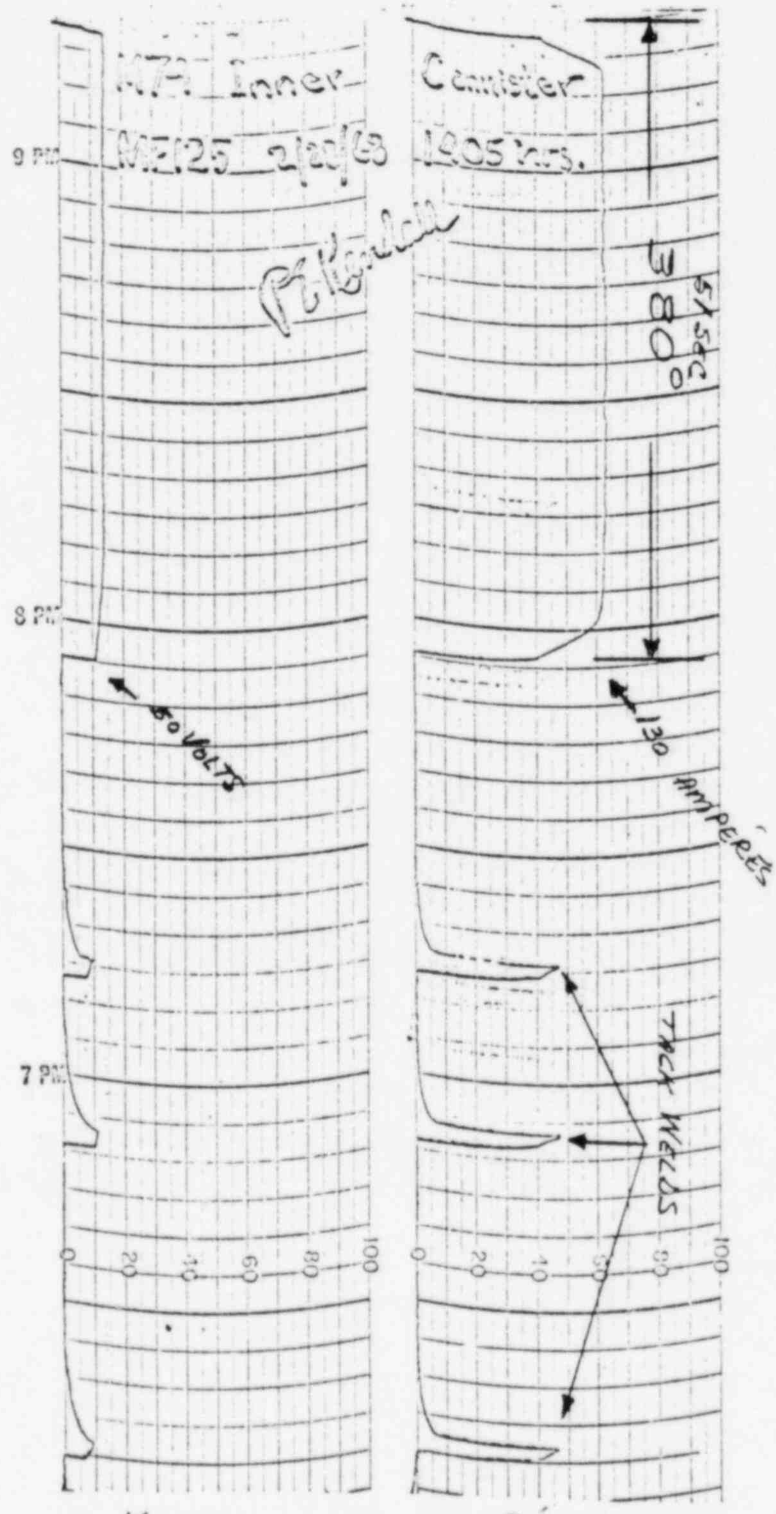


Figure 6-6. Typical Volt-Amperage Record of Canister-Welding Operation

transferring of irradiated fuel from the HNPF to Savannah River was completed July 24, 1968. The entire fuel removal and shipment was accomplished without any incident.

6.3 DISPOSAL OF UNIRRADIATED FUEL

The unirradiated fuel elements stored at the HNPF were intended to be used for the second-core loading, but because of the deactivation of the facility the AEC directed CPPD to disassemble the fuel elements and ship the fuel rods to AI for storage.

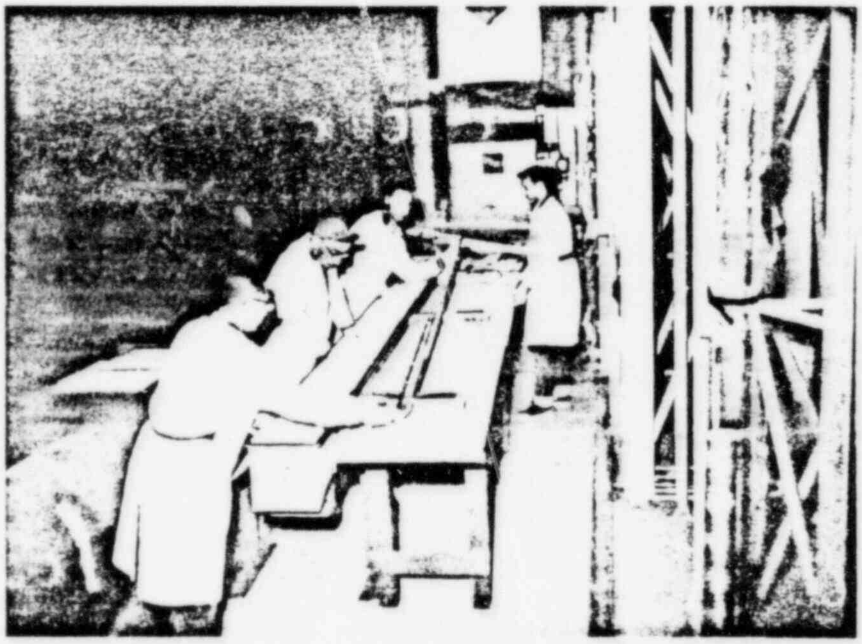
The activity specification for the disposal of the HNPF unirradiated fuel was written and accepted by the CPPD Safety Review Committee and the AEC site representative in October of 1966. The detailed Operating Procedure was accepted by the CPPD Safety Review Committee in November 1966.

The inventory of unirradiated fuel to be shipped to AI included 57 uranium carbide fuel elements with eight rods per element and 13 U-10 Mo fuel elements with 18 rods per element. There were also two elements of depleted U with eight rods in each element to be shipped with the unirradiated fuel. The shipping containers accepted for transporting the unirradiated fuel from the HNPF to AI, Figure 6-7, were similar to those designed by AI for the original shipment of fuel rods from AI to the HNPF. Forty-one shipping containers were fabricated by AI under a contract with CPPD. The capacity of each container was either 16 uranium carbide fuel rods or 36 U-10 Mo fuel rods.

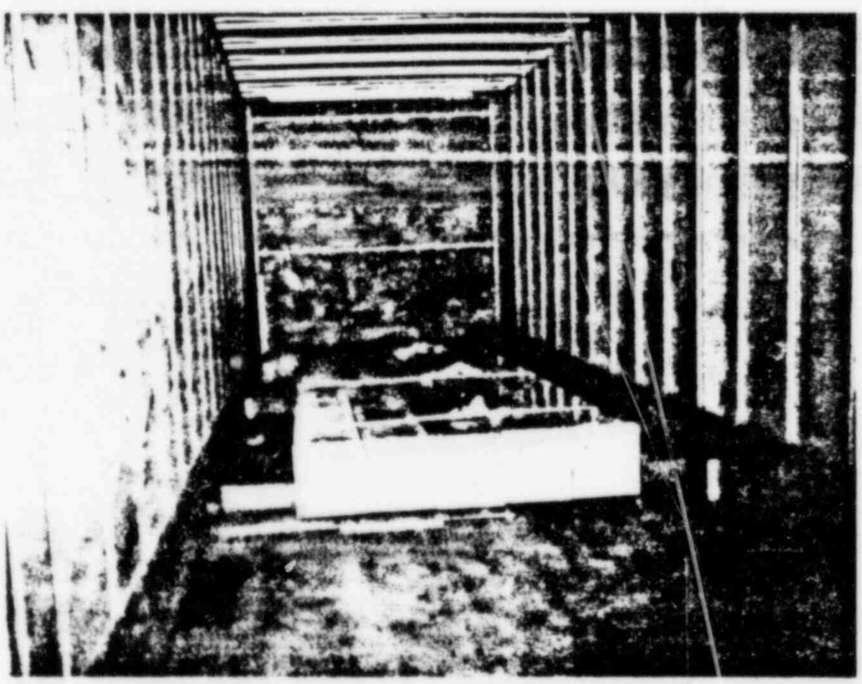
A criticality analysis by AI, based on flooding of the containers, indicated that no more than five containers of either UC or U-10 Mo fuel rods could be shipped in one truck trailer and that the fuel shipment could not contain a mixed load.

The disassembly of unirradiated fuel elements began December 6, 1966 with the disassembly of UC fuel elements.

The unirradiated fuel elements were stored in the fuel-storage section of the reactor high bay. For disassembly the unirradiated fuel elements were removed from storage in the reactor high bay and transported to the Moderator-Fuel Assembly building where they were disassembled and the individual fuel rods were packaged in the shipping containers.



a. Fuel-Rod Loading



b. Maximum Truck Load

Figure 6-7. Fuel Shipping

The first shipment of unirradiated fuel rods was made December 9 and the final shipment on January 5, 1967 to complete the disposal of unirradiated fuel at the HNPF. Final disposition of this fuel occurred when on February 18, 1969 the unirradiated fuel shipment from AI storage to National Lead for reprocessing was completed.

6.4 REACTION OF RESIDUAL PRIMARY SODIUM AND RETIREMENT OF PRIMARY-SODIUM SYSTEM

The AEC plan for decontamination of the HNPF premises required that all residual sodium remaining in the system, including all process lines, tanks, and the reactor vessel, be chemically reacted. The purpose of chemically reacting the residual sodium was to avoid any later potential hazard or damage to the containment after these vessels were sealed in place in accordance with the overall retirement plan.

Systems containing primary sodium were the primary heat-transfer system and the primary sodium-service system. The first consisted of the reactor vessel and three heat-transfer loops arranged in parallel and with the reactor vessel common to all three loops. Major equipment items included in each loop were a primary pump and two IHX's which coupled the primary- and secondary-sodium systems. Sodium piping between equipment was nominal 14- and 16-in. pipe. The second system included cold-traps, plugging meters, carbon trap, EM pumps, various drain tanks, and the primary fill and storage tanks. Piping between equipment items in the primary service system was nominal 2- and 3-in. pipe.

All primary-sodium equipment and piping was located in concrete vaults. All vaults were below the main floor except those for the IHX. Pipeways connecting the various equipment vaults contained seals through which the pipes were run.

Before starting activities for reaction of residual sodium the primary-sodium heat-transfer system had been drained to the primary fill-and-drain tanks. All primary heat-transfer equipment and piping was designed for gravity draining to the reactor vessel. The reactor vessel was drained by use of the EM pumps in the sodium-service system to the primary sodium fill-and-drain tanks. By use of the existing drain system all but about a 2-in. depth of sodium could be removed from the reactor vessel.

The problem of reacting the residual sodium was divided into two parts by physical configuration of the primary systems. The primary-sodium service system consisted of long lengths of 2- and 3-in. pipe with many valves and fittings. This pipe was cut readily into convenient size sections and was removed from the building for steam-cleaning in a special facility as described in Appendix III. Once the piping was removed, other primary-sodium service equipment was removed and disposed of by transporting it off-site for burial, or by cleaning it and saving it for salvage or scrap.

The reactor and heat-transfer loops presented a different problem. The large loop piping, 14- and 16-in. diam, could not be cut and removed easily from the underground pipeways. A 2-in.-thick layer of sodium remained at the bottom of the reactor vessel and a significant amount of residual sodium remained in other parts of the reactor such as on the gridplates and thermal-shield plates and at the bottom of horizontal pipe runs.

Several methods for reacting the residual sodium were investigated including reactions with oxygen, water, carbon dioxide, or alcohol. The experiments and analysis are reported in References 8, 9, 10, and 11. The first step was to drain the reactor heel sodium from its then 2-in. level to the minimum depth possible. Accordingly the sodium was siphoned from the reactor to the sodium-service system by using a combination of pressure in the reactor vessel and a vaccum on the service system. The reactor vessel was too deep (40 ft) for vacuum-siphoning only. Figure 6-8 illustrates the method used in draining the sodium "heel" from the reactor to a 3/16-in. level. (Approximately 314 pounds of sodium).

6.4.1 Reactor Vessel Sodium-Reaction Process Operation

Following removal from the reactor vessel of the sodium heel the valves in the primary-sodium loops were removed and shipped off-site and the piping stubbed and capped. Sodium pipes leading to and from the IHX's were cut and capped near the IHX locations. A 4-in. IHX bypass line was installed. The reactor and the parts of the primary-sodium loops which were left in the isolation structure were thus isolated from each other and from the balance of the primary system, and the residual sodium was reacted by means of a

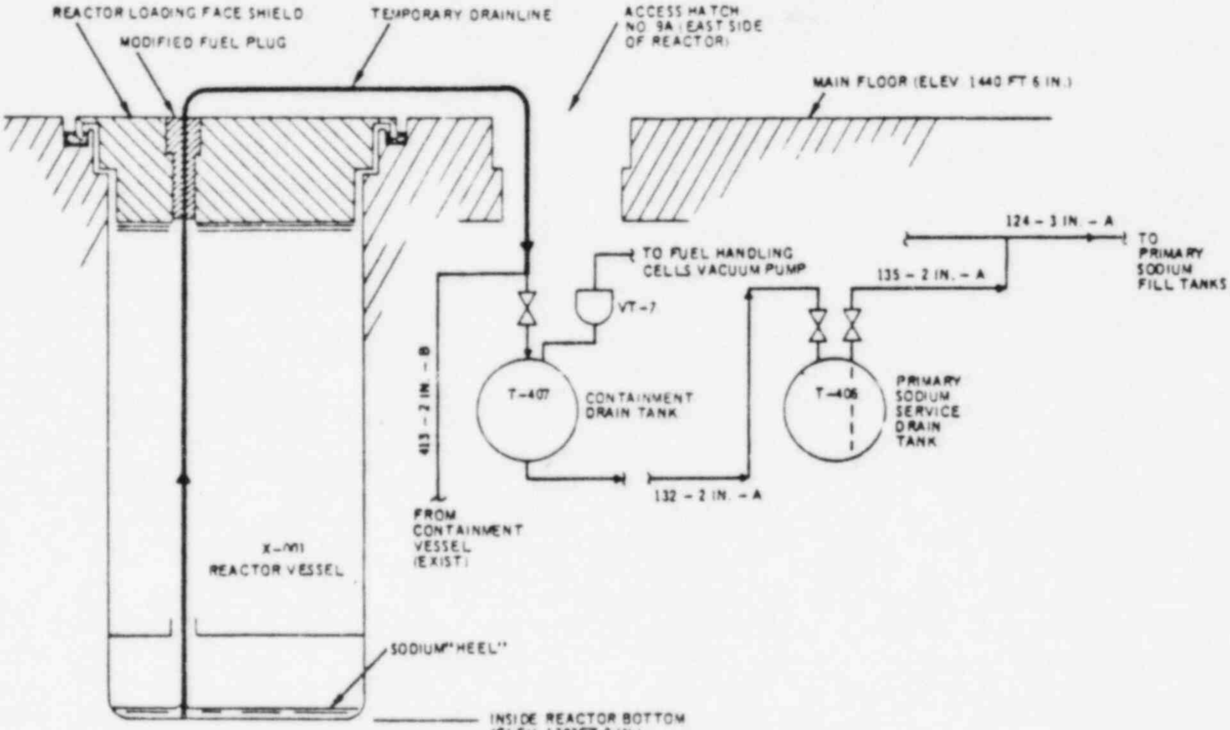


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Figure 6-8. Routing of Flow for Draining Sodium "Heel" From Bottom of Reactor

nitrogen-steam mixture. The steam of the system was made by introducing into the reactor through the reactor drain line and all other connected sodium piping a mixture of 80-vol % nitrogen and 20-vol % steam, which was subsequently adjusted to a mixture of 70 nitrogen and 30 steam. When stable conditions were attained in the reactor the nitrogen-steam mixture was changed to a mixture containing equal volumes of each. The reactor was steamed for 6 hr with this mixture, after which the mixture was changed to one of 80% steam and 20% nitrogen. The reactor was steamed with the 80% mixture until the hydrogen content of the reactor atmosphere was reduced to zero and then steamed for an additional 6 hr to assure that the reaction of the sodium was complete. The entire operation involved a total of 82 hr of steaming. Following steaming operations the reactor system was purged with nitrogen for a period of 1 hr to assure the removal from the atmosphere of all hydrogen. During steaming and purging operations noncondensible gases were vented to the reactor building

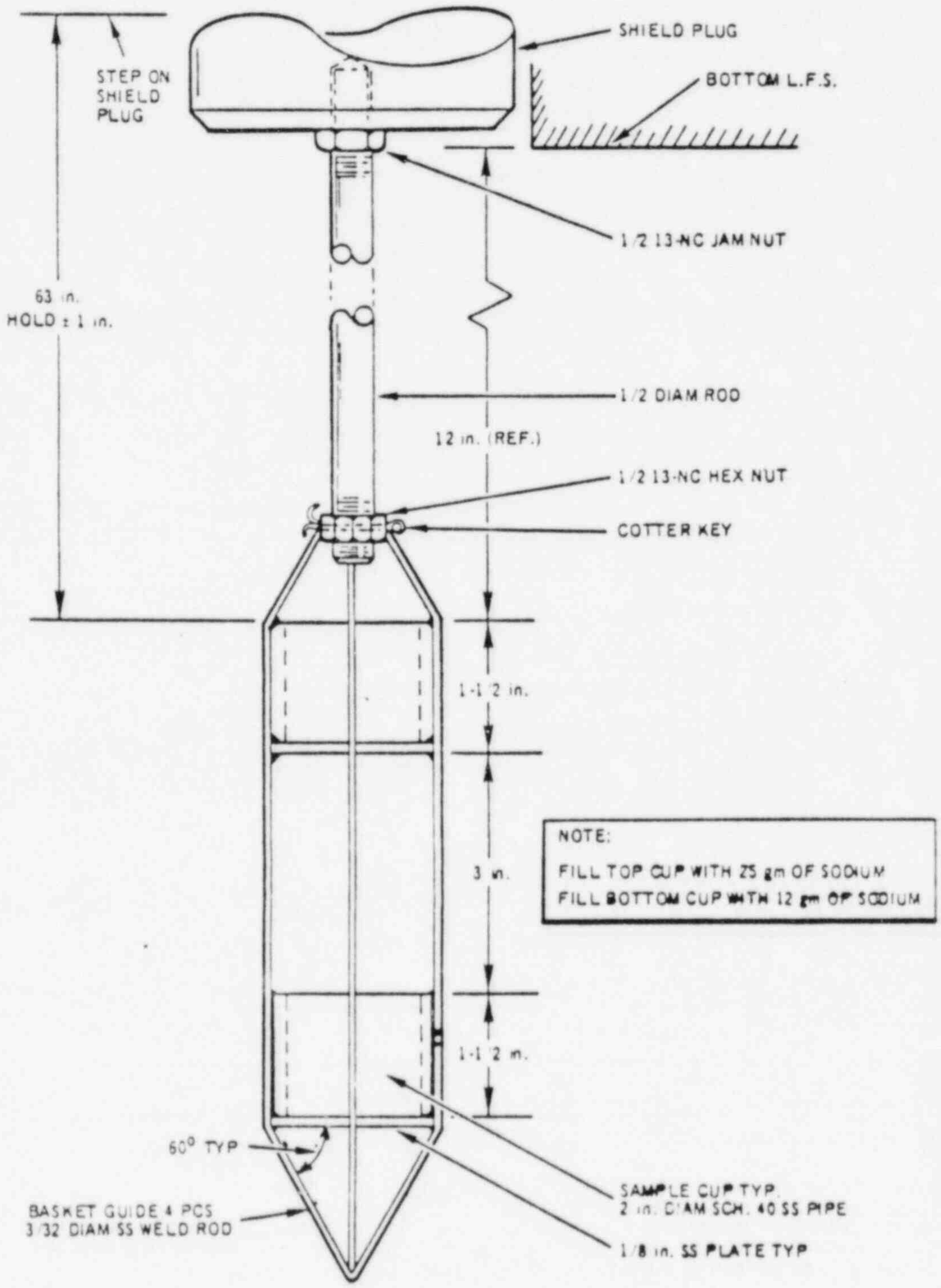


Figure 6-9. Sodium Sample Suspended From Shield Plug

stack. Finally the vessel atmosphere was dried by increased heating of the reactor vessel while circulating hot dry nitrogen through it for a period of 38 hr.

To provide a direct means of assessing the completion of the sodium-reaction process in the reactor vessel, sodium samples were placed in special containers and suspended inside the vessel before steaming operations. Each sample consisted of two stainless-steel cups, one suspended 3 in. above the other. The lower cup contained 12 gm of sodium and the upper 25. The sample cup assemblies were suspended 1 ft below the loading face shield as shown in Figure 6-9. Six such sample assemblies were installed in the reactor by attaching them to the bottom of reactor shield plugs and inserting them into selected locations in the reactor using the short gas-lock.

The first two sample assemblies were removed after the vessel was steamed for an additional 6 hr beyond the point at which the hydrogen content in the reactor atmosphere was reduced to zero. Examination of these samples revealed a reddish-brown liquid which, when cooled, set up into a bluish gelatin. A single small piece of solid material found in the liquid proved to be nonreactive when placed in water. Based on the absence of unreacted sodium in the samples and the complete absence of hydrogen in the reactor atmosphere it was concluded that all residual sodium in the reactor vessel had been reacted chemically. The reactor vessel preheaters were then turned up to 450°F and the 24-hr purge with hot (350 to 400°F) nitrogen was begun. At the end of the 24-hr period a sample of the effluent nitrogen was collected, cooled from 450 to 74°F, and tested for water vapor. The relative humidity of this sample was less than 2% at 74°F. The four remaining sample assemblies were then removed and inspected. All samples were found to be solidified and were unreactive when placed in water.

In developing the technique for the reaction of the sodium in the reactor vessel, consideration was given to flushing the reactor with water after all the sodium had reacted and formed sodium hydroxide. Flushing with water would have removed the sodium hydroxide but the large amount of water-required would have caused an extremely slow process because the liquid radioactive waste system was not designed to handle such volumes readily. In addition, published information on the properties of sodium-hydroxide solutions indicated that if the residue is dried sufficiently at high temperature (approximately 400°F), as was done, it would solidify and the solid residue will never react in any way with the stainless-steel containment vessels. The effects of sodium hydroxide on stainless steel are reported in more detail in Reference 12.

6.4.2 Heat-Transfer Loops

After completion of the reactor vessel steaming operation the steaming system was modified for steaming the primary-sodium heat-transfer loops. Each of the three heat-transfer loops was steamed with 75% steam, 25% nitrogen until the hydrogen concentration in the effluent had indicated zero for a period of 3 hr. The loops were steamed one at a time with the steam-nitrogen mixture entering each loop through the cover-plate at the top of the pump case and the effluent gases leaving through the lines leading to the pump suction, and from the IHX, which had been isolated and bypassed.

After steaming of the three loops was completed the loops were dried by purging each with hot nitrogen until the relative humidity of the effluent nitrogen was less than 1%. After the loops were dried, samples of the residue were taken from each of the six main pipe ends, two from each loop. Access to the samples was by means of 4-in.-diam inspection holes which were cut into the pipe walls near the ends of the loop piping. These pipe ends were low points in the sodium piping system. Inspection of the piping interiors revealed that the residual sodium hydroxide had solidified into a dry crystalline mass. Samples of the residue were taken where available. Two of the lines did not contain enough residue for a sample. All samples were solid and found to be completely unreactive when mixed with water. The sample holes in the loop piping were subsequently closed by welding.

Investigation of the hazards involved with steaming the primary sodium system indicated that the worst hazard would be a hyrogen explosion in the reactor vessel. The other hazard would be a rapid uncontrolled reaction which could overpressure and possibly rupture the reactor vessel. In order to eliminate the explosion hazard a number of protective steps were provided. First the controlling parameter, hydrogen concentration, would be monitored and kept below 5% by control of the process. Next the oxygen concentrations would be monitored and kept below 1/2%; the combustible minimum is about 4%. If the oxygen concentration exceeded 1/2% an alarm would be activated on the annunciator panel. To prevent air from being drawn into the reactor by a vacuum, an automatic valve would close the reactor effluent line when the pressure dropped below 1/4 psig. At the same time a separate nitrogen line with a pressure regulator would open to maintain the reactor pressure above 1/2 psig. These precautions were judged adequate to prevent a hydrogen explosion. In addition, hydrogen-oxygen mixtures of any proportions are not flammable in combination with more than 90% nitrogen.

Although no mechanism could be stipulated through which all of the sodium would react at once, it was estimated that damage resulting from that event would, at worst, rupture the reactor bellows and thereby relieve the reactor vessel to the containment vessel with no danger to personnel. In considering steep ramp-type pressure rises it was concluded that safety relief valves would not be practical; instead, automatic valves would be installed on the steam and nitrogen inlet lines to close at 10 psig. An alarm would also be activated on the annunciator panel. The effluent system in addition was sized to pass the maximum predicted gas flow without damaging the reactor vessel by overpressure.

Activity Specification No. 5 (see Appendix I) was written to describe the methods and requirements for reaction of residual primary sodium. In general the specification called for:

1) Isolation of the reactor vessel from the rest of the primary-sodium system and steam-cleaning in place,
2) Isolation of the primary-sodium heat-transfer loops from the IHX's and steam-cleaning in place with the primary-pump cases still in the loops, and
3) Isolation and dismantling of the primary-sodium service piping for ultimate disposal as scrap.

The specification defined requirements for the steam-cleaning process in detail, starting from initial draining requirements through the steam process and the final drying. Normal operation-ranges and limiting operating conditions were defined. To insure safe operation, alarm setpoints were established and operator actions in the event of an alarm were described.

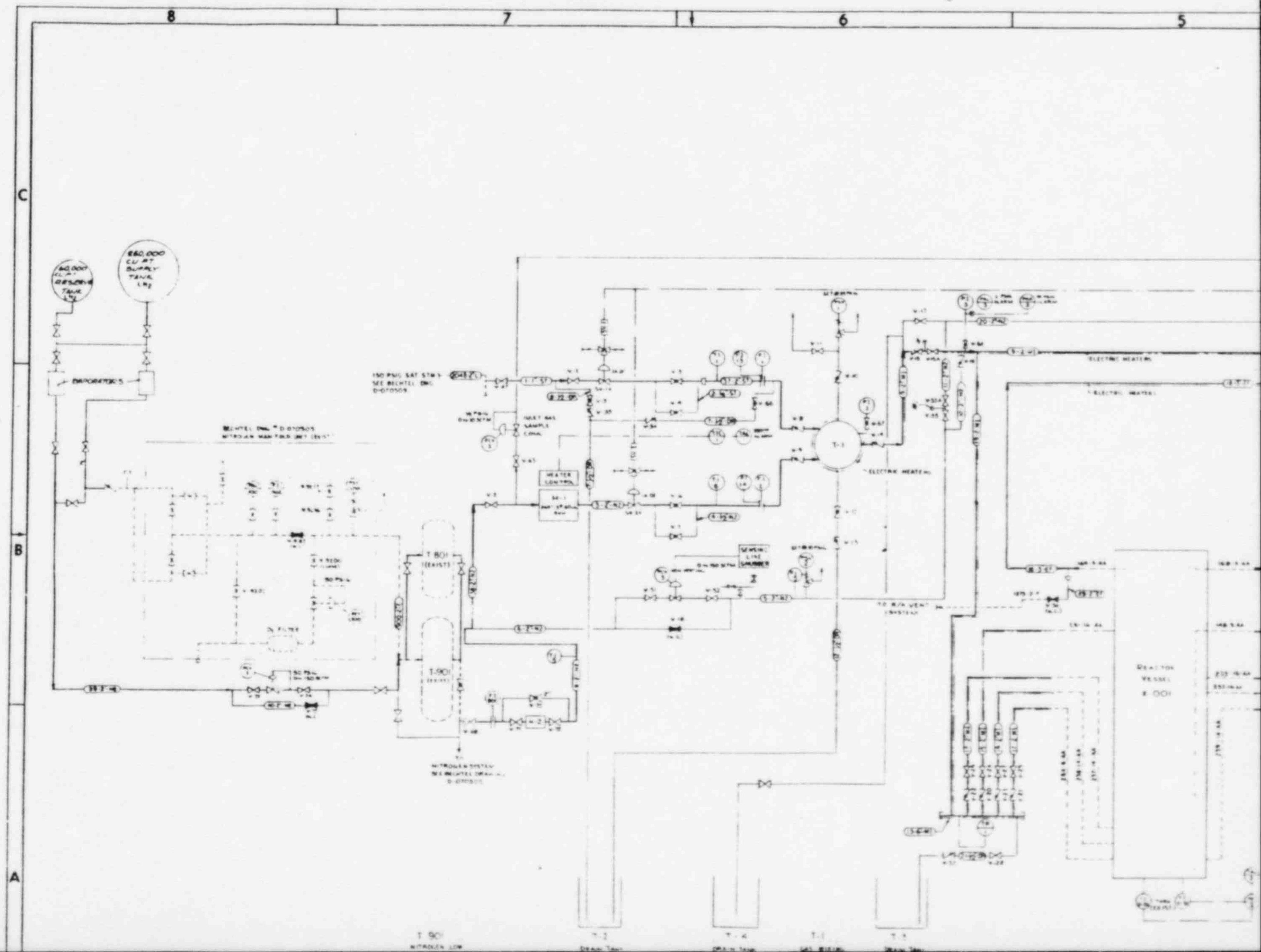
Writing of the activity specification proceeded concurrently with development of the P&I (Process and Instrumentation) diagram (see Figure 6-10). As much existing facility equipment as possible was used. Equipment selections were made on the basis of adequacy for the short operating time required, rather than on efficiency and long life; therefore, much of the detailed information contained in the activity specification was affected by the availability of equipment. Safety and assured results were prime considerations. No attempt was made to obtain experimental information from the steaming operation.

The reaction of residual sodium activity was divided into seven parts by CPPD. A Retirement Detailed Procedure (RDP) was written for each part. Information included in these documents covered step-by-step operations to be performed in isolation and removal of primary-sodium equipment and steaming of the reactor vessel and primary-sodium heat-transfer loops. Actual construction of the steaming system was done by a subcontractor to CPPD (Stearns-Roger). Installation of instruments and steaming operations were done by CPPD personnel. During the steaming-loop construction period close cooperation was maintained between AI and CPPD in selecting equipment and making detail piping and instrumentation plans. The installation is shown in Figure 6-11.

A complete description of the operations for reaction of residual primary sodium is given in Reference 9. Calculation of the results of steaming the reactor vessel indicated that 483 lb of sodium were reacted with steam. No calculation was made for the amount of sodium reacted in the primary heat-transfer loops; however, visual inspection of these loops indicated that 1/2 to 3/4 in. of sodium-hydroxide residue was left at the low point of some loops (pipe ends nearest to reactor). The primary-sodium service piping was dismantled and cleaned in a special cleaning facility built at the site. This is described in Appendix III.

Major equipment items were removed and shipped off-site in accordance with the activity specifications and detailed procedures.

The primary-pump internals were removed before the steaming operation, since the three pump casings were used to admit the steam and nitrogen mixture



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A
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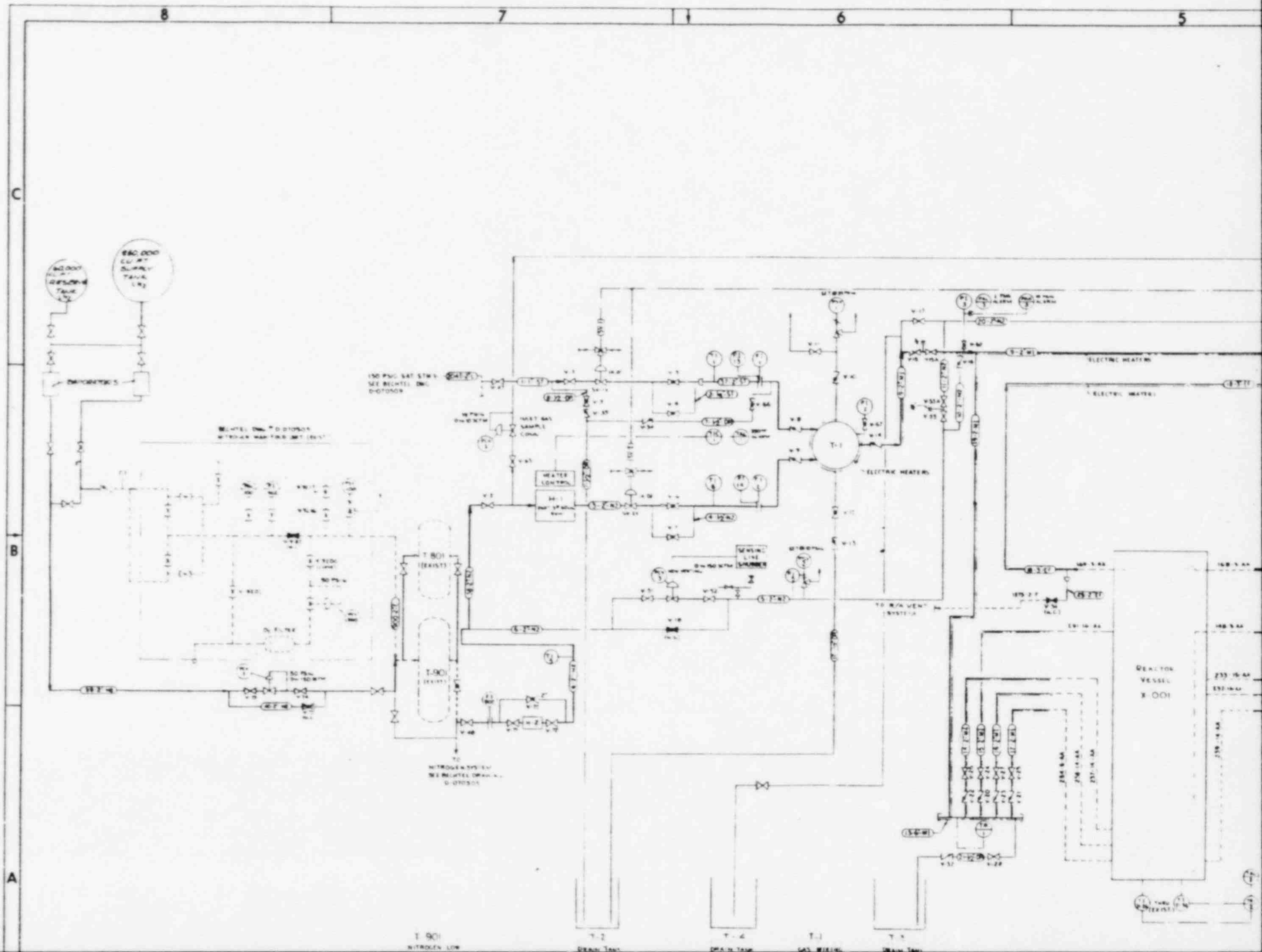
EVAPORATORS
HEATER CONTROL
SENSING LINE SNUBBER
T-801 (EXISTS)
T-901
T-1
ELECTRIC HEATERS



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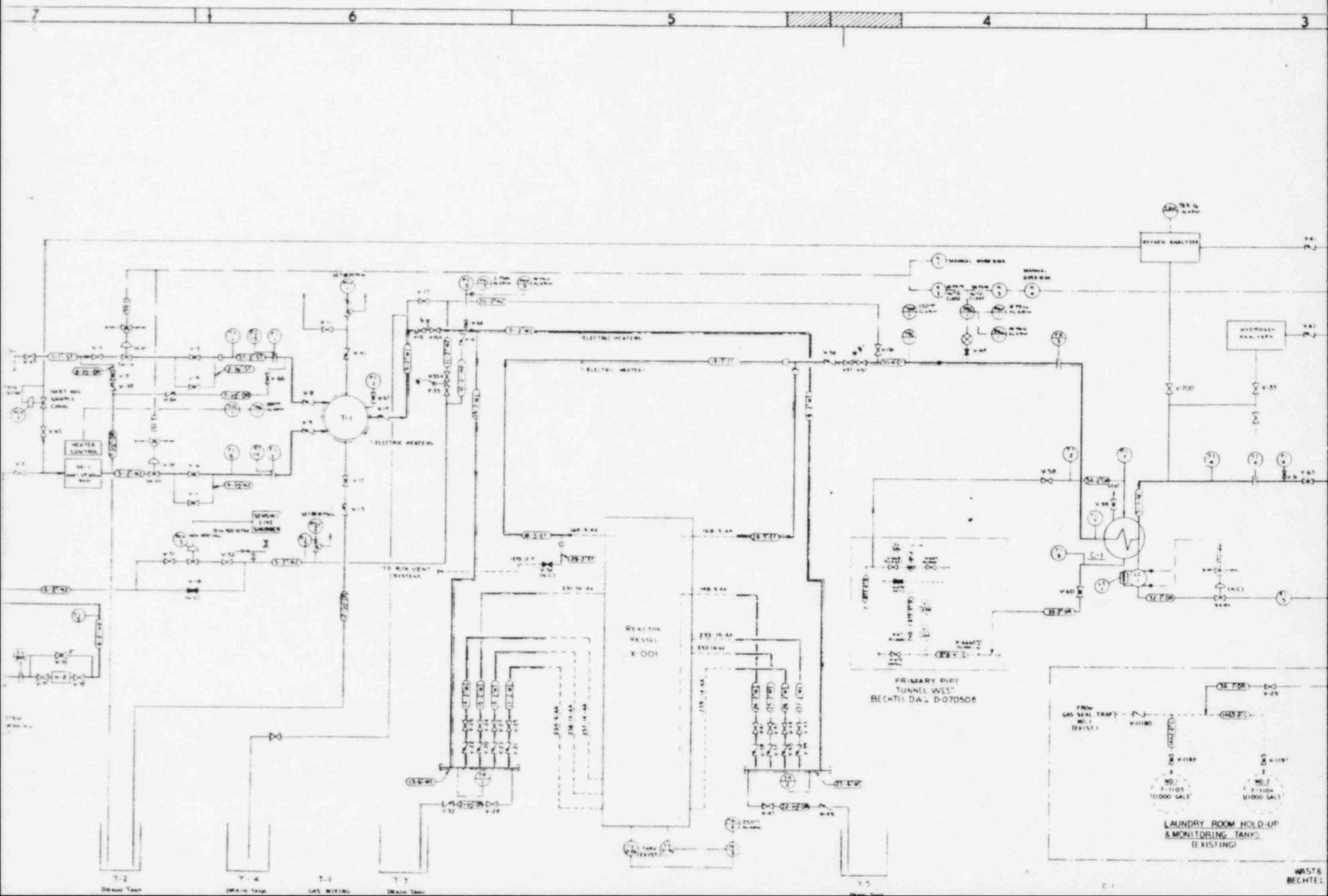
A
Reactor Vessel X-001
T-1
Electric Heaters
Heater Control
Sensing Line Snubber
To R/A Vent System
150 PSIG SAT STM S
See Bechtel Dwg. D-070508
Waste Gas Sample Conn.
Bechtel Dwg. D-070507
Nitrogen Manifold Set (Exist)
T-801 (Exist)
T-901 (Exist)
To Nitrogen System See Bechtel Dwg. D-070501
Vaporizers
850,000 Cu Ft Supply Tank
T-901 Nitrogen Low
T-2 Drain Tank
T-4 Drain Tank
T-1 Gas Mixing
T-3 Drain Tank



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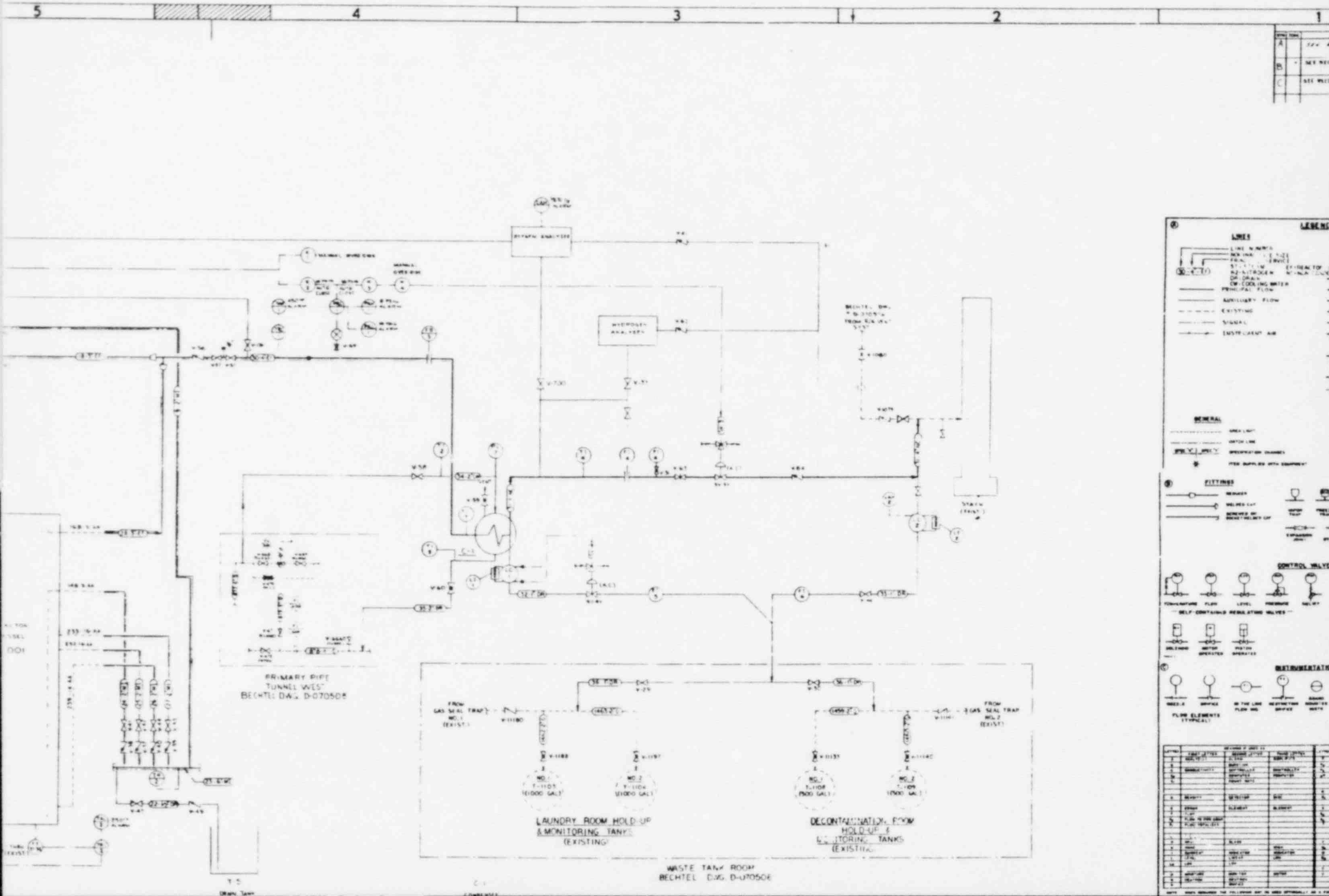
4
3
HEATER CONTROL
T-1
ELECTRIC HEATERS
TO R/A VENT SYSTEM
REACTOR VESSEL X-001
HYDROGEN ANALYZER
OXYGEN ANALYZER
MANUAL OVERRIDE
C-1
PRIMARY PIPE TUNNEL WEST BECHTEL DWG. D-070508
FROM GAS SEAL TRAP NO.1 (EXIST.)
NO.1 T-1103 (1000 GAL)
NO.2 T-1104 (1000 GAL)
LAUNDRY ROOM HOLD-UP & MONITORING TANKS (EXISTING)
T-2 DRAIN TANK
T-4 DRAIN TANK
T-1 GAS MIXING
T-3 DRAIN TANK
WASTE
BECHTEL



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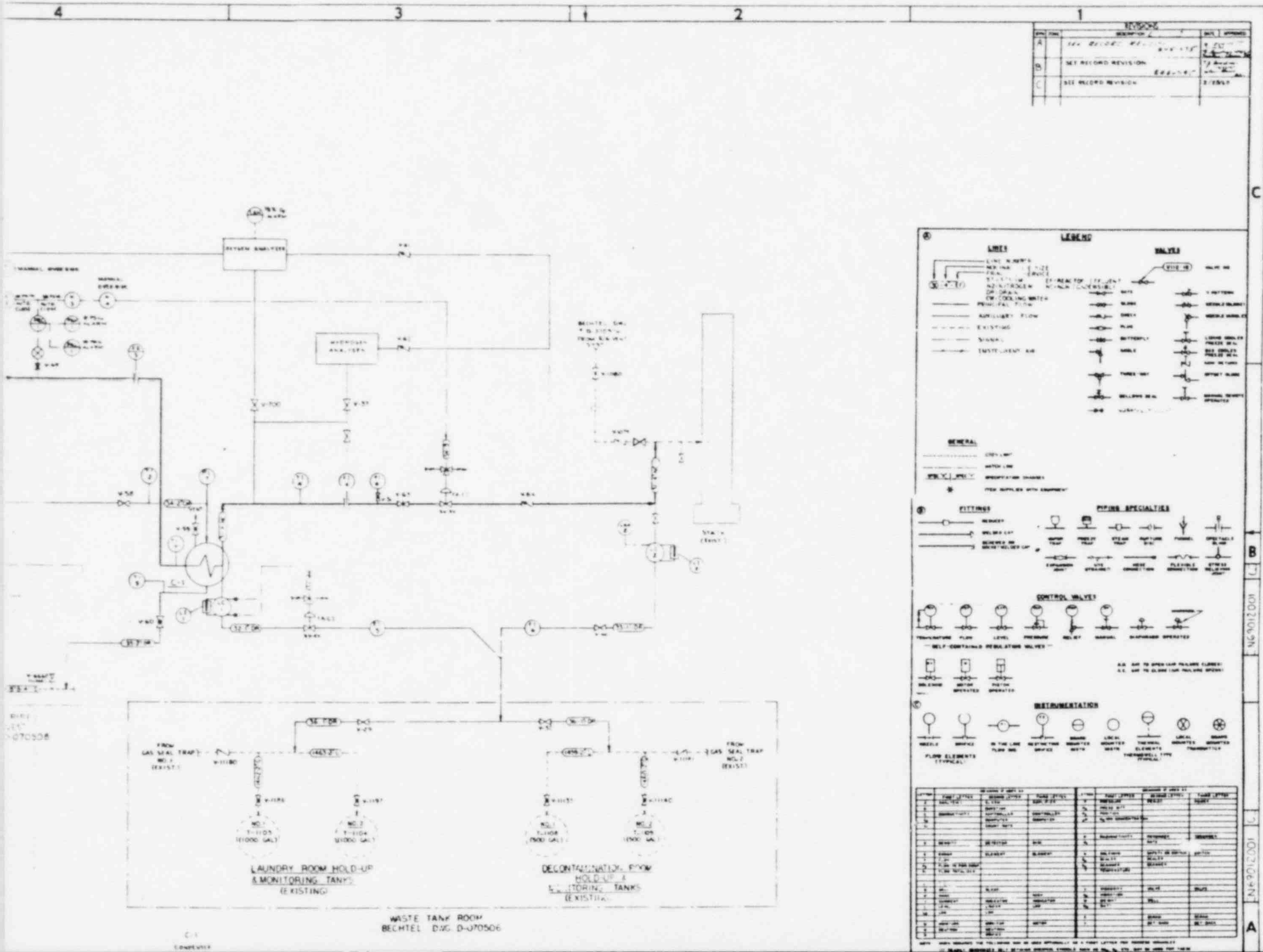
1
LEGEND
LINES
PRINCIPAL FLOW
AUXILIARY FLOW
EXISTING
SIGNAL
INSTRUMENT AIR
GENERAL
FITTINGS
CONTROL VALVES
SELF-CONTAINED REGULATING VALVES
INSTRUMENTATION
FLOW ELEMENTS (TYPICAL)
HYDROGEN ANALYZER
STACK (EXIST.)
PRIMARY PIPE TUNNEL WEST BECHTEL DWG. D-070508
FROM GAS SEAL TRAP NO. 1 (EXIST.)
FROM GAS SEAL TRAP NO. 2 (EXIST.)
NO. 1 T-1103 (1000 GAL)
NO. 2 T-1104 (1000 GAL)
NO. 1 T-1108 (1500 GAL)
NO. 2 T-1109 (1500 GAL)
LAUNDRY ROOM HOLD-UP & MONITORING TANKS (EXISTING)
DECONTAMINATION ROOM HOLD-UP & MONITORING TANKS (EXISTING)
WASTE TANK ROOM BECHTEL DWG. D-070506



REVISIONS
A SEE RECORD REVISION
B SEE RECORD REVISION
C SEE RECORD REVISION
LEGEND
LINES
VALVES
GENERAL
FITTINGS
PIPING SPECIALTIES
CONTROL VALVES
SELF CONTAINED REGULATION VALVES
INSTRUMENTATION
FLOW ELEMENTS (TYPICAL)
OXYGEN ANALYZER
HYDROGEN ANALYZER
STACK (EXIST)
FROM GAS SEAL TRAP NO. 1 (EXIST)
FROM GAS SEAL TRAP NO. 2 (EXIST)
NO. 1 T-1105 (1000 GAL)
NO. 2 T-1106 (1000 GAL)
NO. 1 T-1108 (500 GAL)
NO. 2 T-1109 (500 GAL)
LAUNDRY ROOM HOLD-UP & MONITORING TANKS (EXISTING)
DECONTAMINATION ROOM HOLD-UP & MONITORING TANKS (EXISTING)
WASTE TANK ROOM
BECHTEL DWG D-070506
N69012001 C

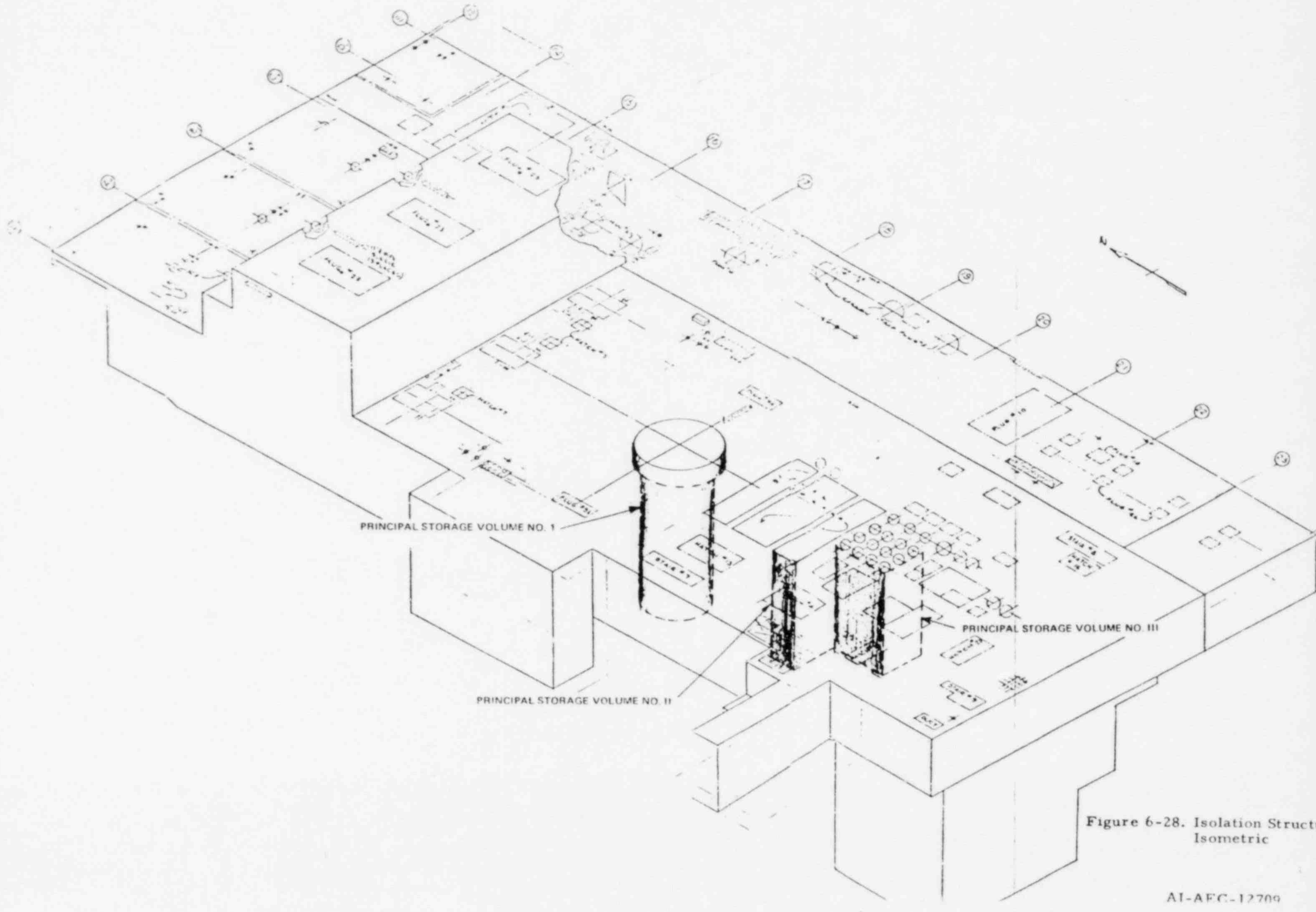
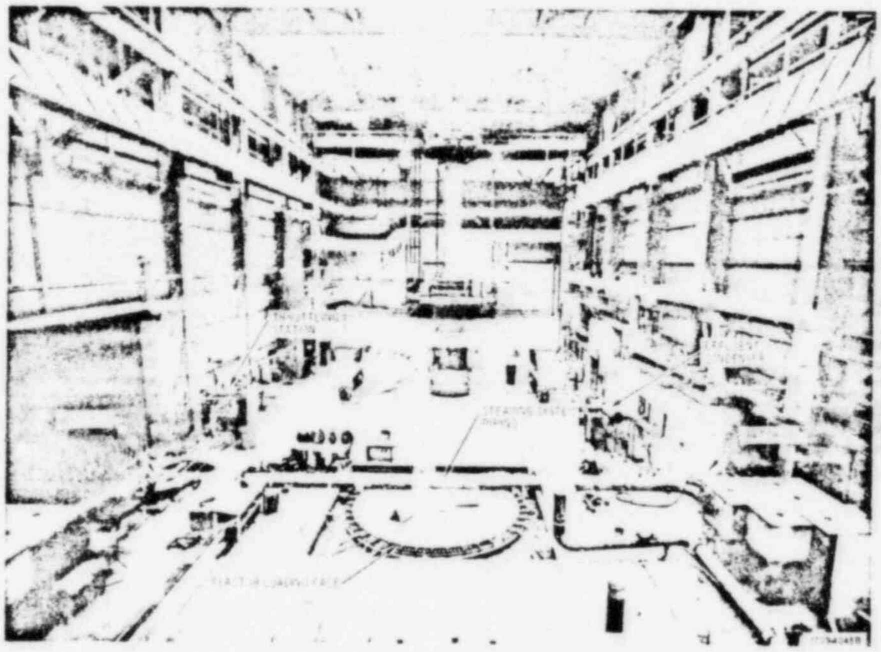
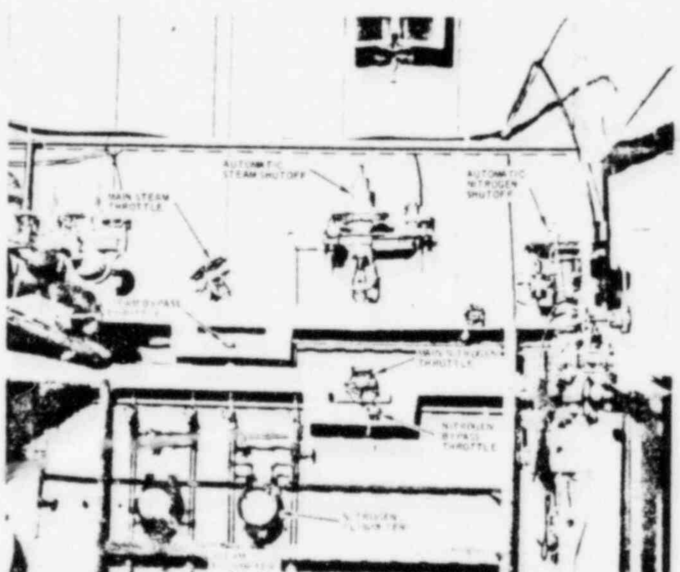


Figure 6-28. Isolation Structu
Isometric



a. Overall View of Reactor Building Main Floor
with Steaming System Installed



b. Manual Throttling Station

Figure 6-11. Steam Reaction Installation to part of the primary-pipe system. The steaming operation thereby reacted residual sodium in the pump case and volute. The pump pipe connections were then cut and capped (see Figure 6-12).

The pump internals were cleaned in the wash cell, then transferred to the operating floor where additional decontamination was done as the hydroxide exuded from the metal surfaces which had been exposed to sodium. The pump cases were removed, Figures 6-13 and -14, and the residual hydroxide was cleaned off; the discharge side was free of hydroxide. The pumps were then reassembled and, with all openings closed, were crated and shipped to the National Reactor Test Site (NRTS) for storage. The pumps were stored with an internal nitrogen cover-gas and with provisions (sampling valve) for testing and adding nitrogen.

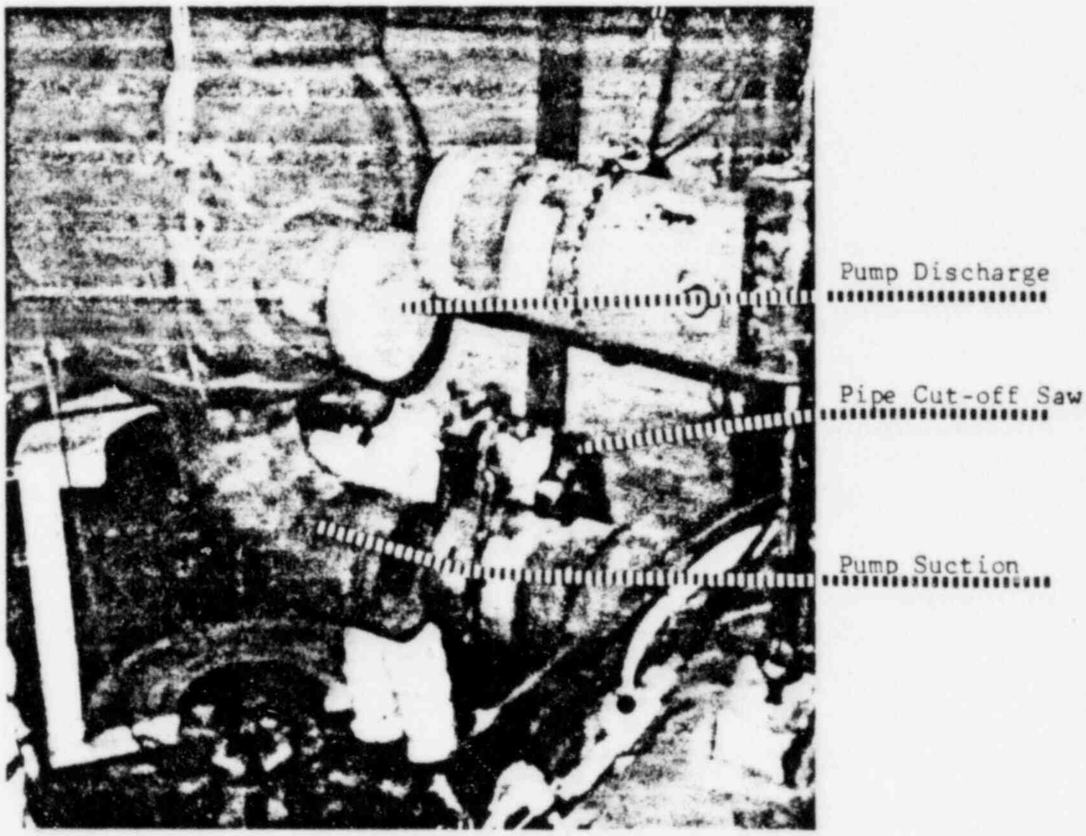


Figure 6-12. Preparation for Removal of Primary Pump

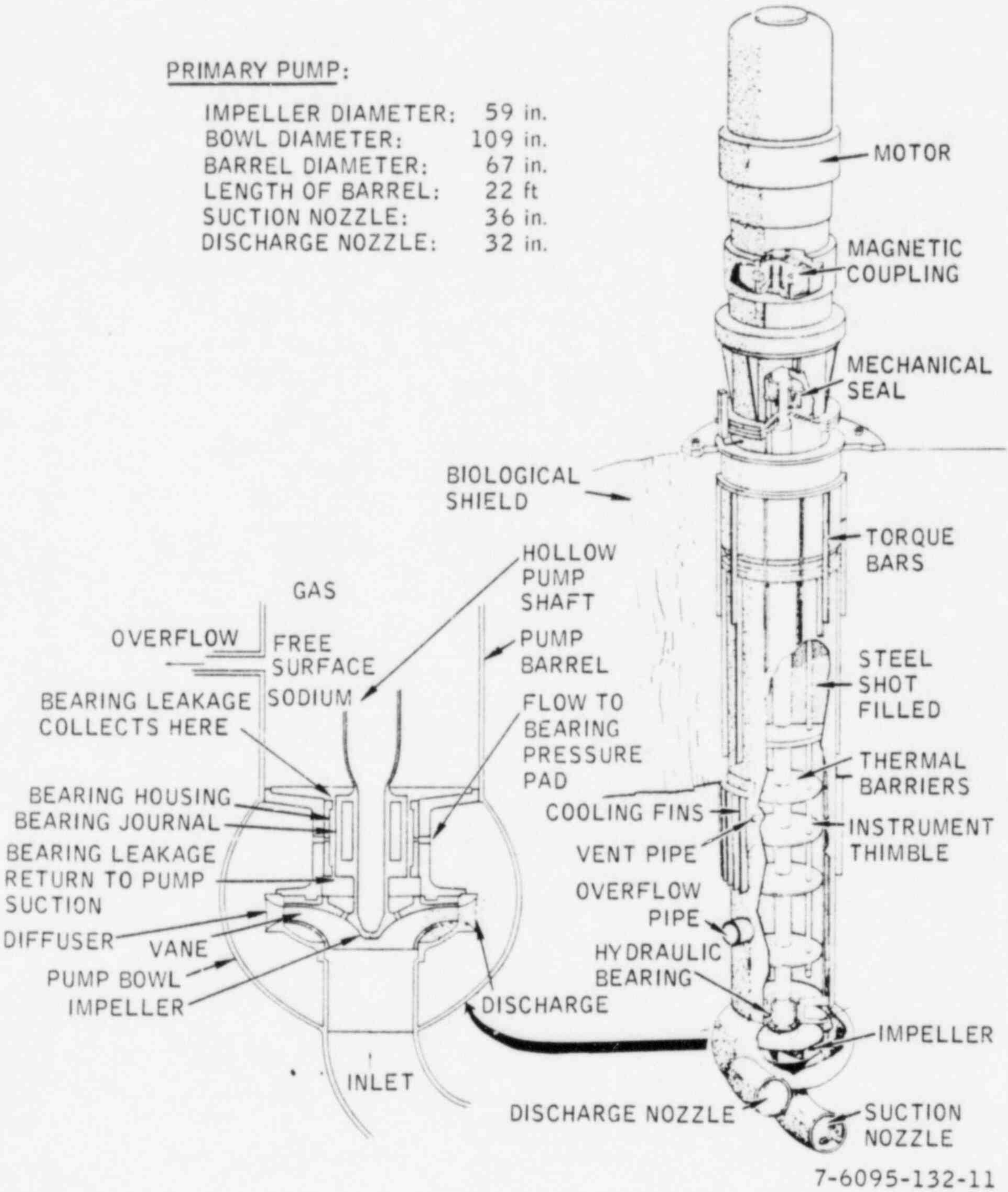


Figure 6-13. Free-Surface Sodium Pump

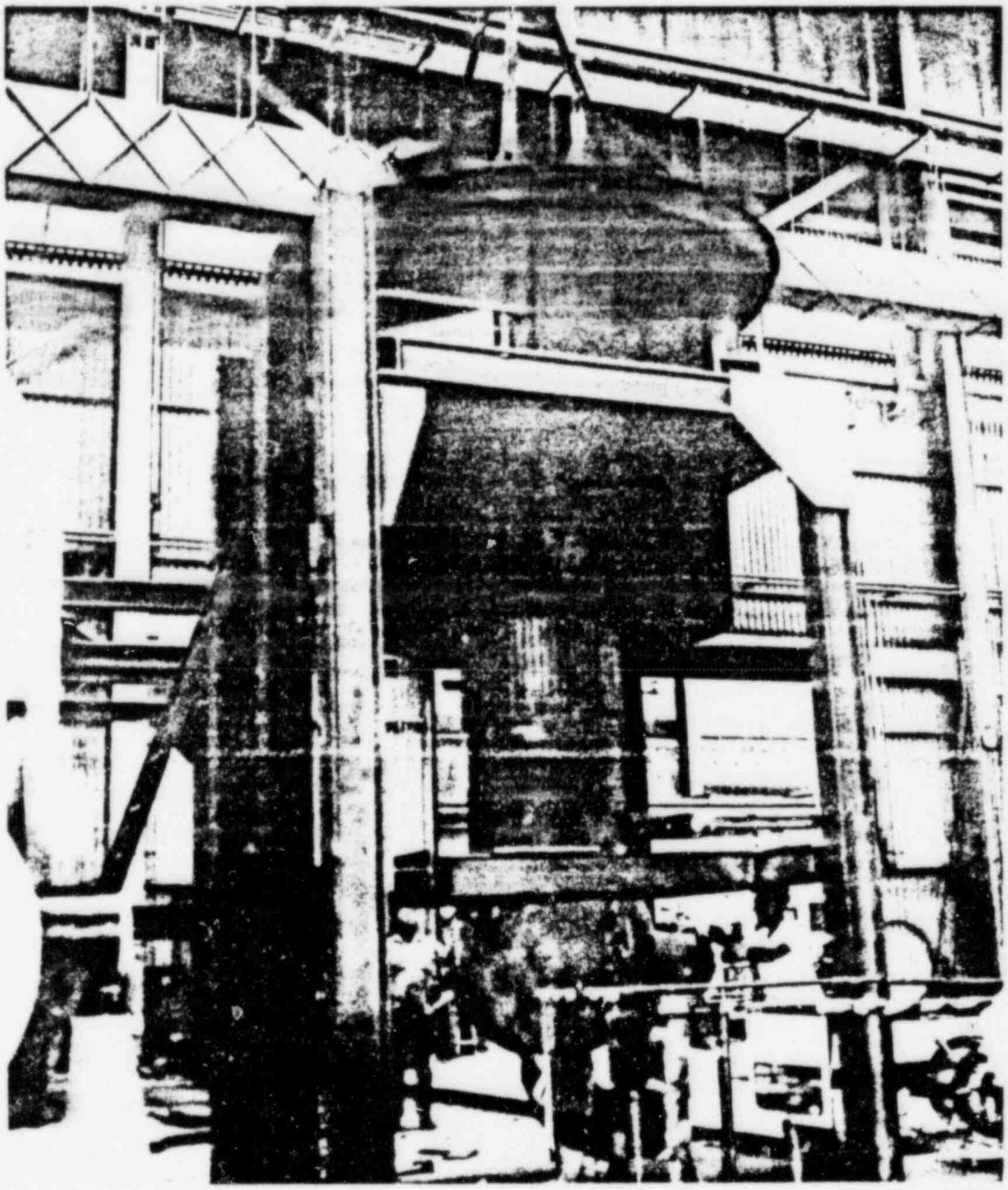


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Figure 6-14. Primary-Pump Assembly After Removal

The primary-sodium balancing legs fabricated of pipe had only scrap value and were not worth decontaminating. They were capped, removed, crated, and shipped off-site to Beatty, Nevada for disposal by burial. This was also the disposition of the primary system EM pump throats, plugging meters, vapor traps, small valves, and similar contaminated articles of little reuse value.

The three IHX's, each consisting of two modules (see Figures 6-15 and -16), having been drained were cut and capped on both primary and secondary sides while under inert gas. These were prepared for shipment by clamping the bellows and bolting wood skids to the mounting legs. With a surface activity of only 0.5 mr/hr no special precautions were required for shipment. These three units sealed and protected with a cover-gas were sent to NRTS for storage. It is not expected that internal cleaning will be needed to make them serviceable.

The primary blocking, throttle, and check valves, having been removed from the system before any steaming, contained some sodium with Na^{22} activity. This was adequately retained by use of a capping plate welded over each pipe stub. These items were shipped protected in this manner to the AEC-LMEC where they were decontaminated by steam-cleaning.

The primary oxygen cold-traps, carbon traps, and associated equipment such as economizers and valves having little reuse value and a high cost of decontamination were disposed of as radioactive scrap at offsite burial. The activity levels were not high enough to require shielding or unusual packaging for shipment to burial.

6.5 REACTION OF RESIDUAL SECONDARY SODIUM AND RETIREMENT OF SECONDARY-SODIUM SYSTEM

The secondary-sodium system contained no radioactive material; therefore, no in-place reaction of residual sodium was attempted. After draining the bulk of the secondary sodium from the system and shipping it off-site the dismantling of the secondary-sodium piping was begun.

Retirement of the secondary-sodium system was done under the requirements of Activity Specification No. 6 (Appendix I). In addition to describing material categories for disposal and safety requirements, the specification

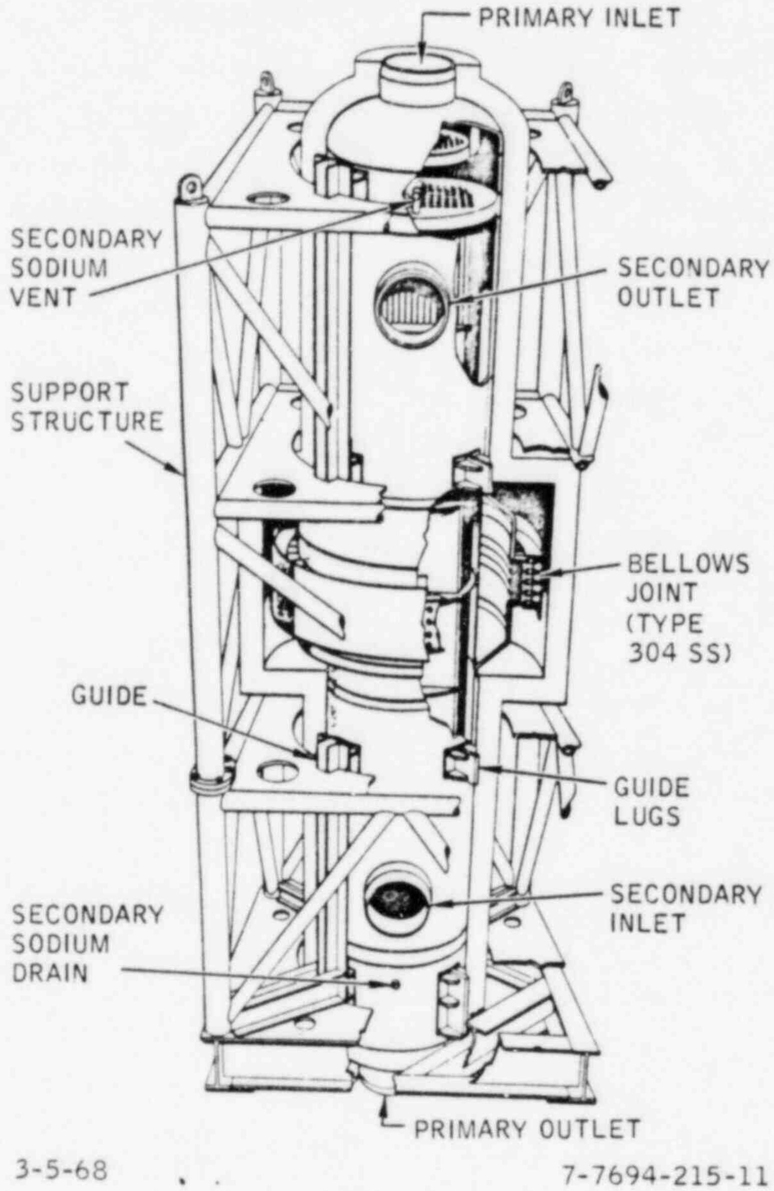

Figure 6-15. HNPF Intermediate Heat Exchanger

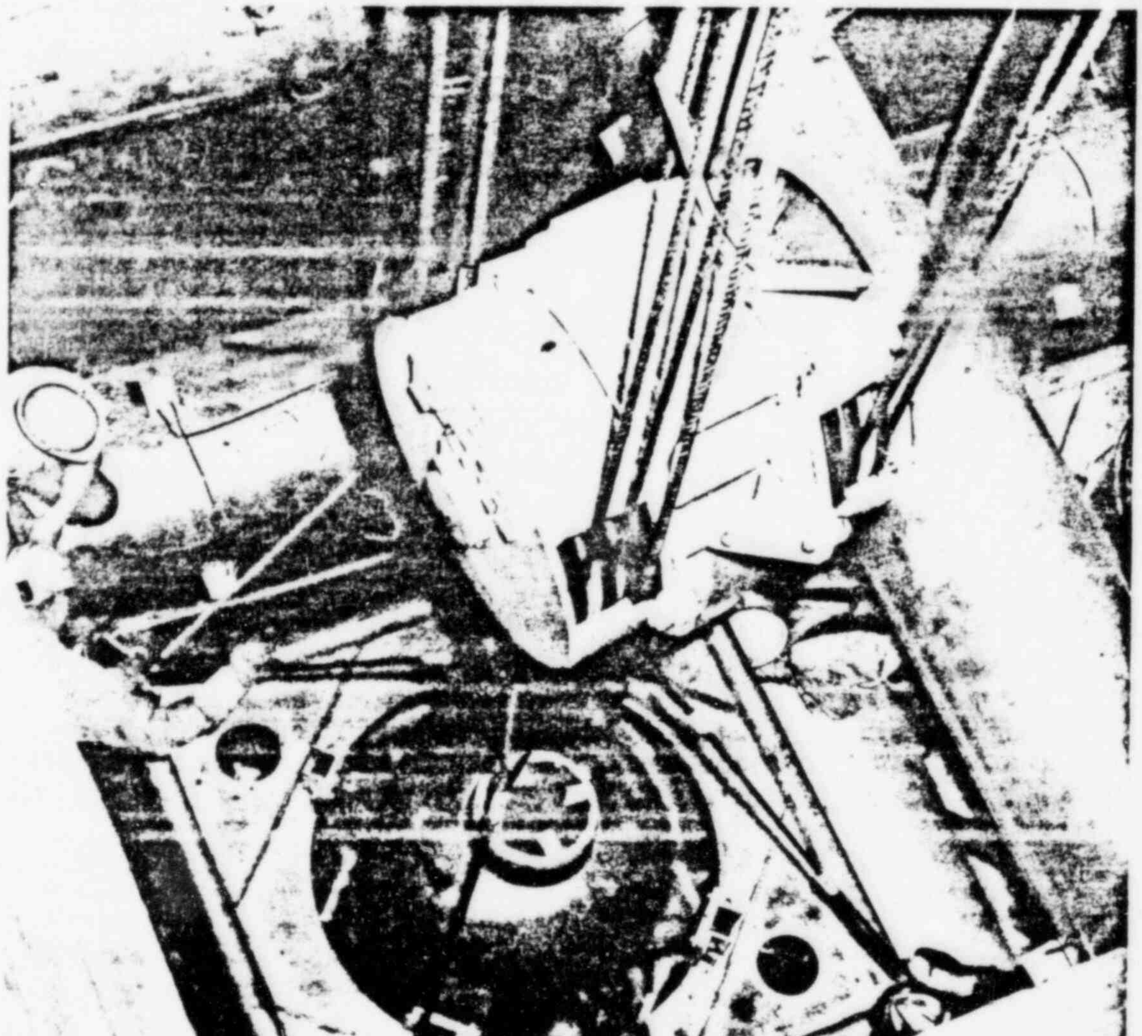

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Figure 6-16. Intermediate Heat Exchanger Module Being Removed From Vault included requirements for isolation of major equipment such as steam generators, IHX's, and centrifugal pumps. Under the specification, pipe sections containing residual sodium less than 1/8-in. thick could remain open to the atmosphere. Previous tests run at AI indicated that sodium layers under 1/8-in. thick would react completely with air within 30 days and would not be hazardous; reaction of sodium layers over 1/4-in. thick with air was considered hazardous.

In the tests at AI samples of sodium 1/2-in. thick were placed in 3-in. deep tins and allowed to react with the moisture in the air. The reaction progressed slowly during the first four weeks. A layer of wet sodium hydroxide formed over the unreacted sodium. In the fourth week the reaction terminated violently. The sodium hydroxide liquid being more dense than solid sodium caused the sodium to suddenly float to the top. The abundance of moisture set off a rapid sodium/water reaction. The hydrogen evolving mixed with the air and was ignited by the heat of the reaction. The explosion blew the sodium out of the pan and into the walls of the test containment. Subsequently sodium samples less than 1/4-in. thick tested in a similar manner completely and slowly reacted to form sodium hydroxide.

Under the activity specification, Retirement Detailed Procedures were prepared by CPPD. These procedures gave step-by-step instructions for each phase of the operation. The six RDP's covered:

1) Isolation and disassembly of the secondary-sodium service system,

2) Isolation of the IHX's from the secondary-sodium systems,

3) Isolation and removal of the secondary-pump cases,

4) Isolation of steam generators from the secondary system,

5) Removal of secondary heat-transfer piping, and

6) Disassembly and removal of steam generators.

Before dismantling the secondary-sodium system all heaters were turned off and the system was allowed to cool. Drive motors, EM couplings, and internal parts were removed from the three pumps in the heat-transfer system. The pump internals were cleaned in the existing pump-wash cells, then disassembled, cleaned in the wash cells again, and packaged for shipment off-site.

Dismantling of the pipe proceeded with first isolating major equipment items and then isolating the heat-transfer system from the service system. Next piping was cut into convenient lengths and placed in temporary storage before cleaning. Before removing equipment in pipe sections, thermal insulation was removed and dumped into an underground vault where it could be abandoned. All auxiliary systems including cover-gas, instrument, and electric-heating systems were then disconnected. Instrument elements, primarily thermocouples, and electric heaters were disposed of as scrap. At each step precautions were observed to avoid accidents due to cutting lines connected to a power (electric or pneumatic) source. Also, systems being disconnected were checked to ensure that no other operations would be disrupted.

Secondary heat-transfer piping was transferred to a concrete pad outside the plant where it was cleaned by use of a fire hose; the sodium cleaning facility had not been completed at that time. Some of the secondary-sodium service piping was cleaned on the concrete pad, but most service piping was cleaned in the cleaning facility. All secondary-sodium piping was removed and cleaned with steam or water. The activity-specification criteria would have permitted pipe sections with less than 1/8 in. of sodium to be reacted with the atmosphere. An inspection of the pipe internals and a measurement of the depth required that the pipe be cut to short lengths. The benefits of added safety, scrap value of the pipe, and fewer cuts in the long lengths removed made total removal of the secondary-sodium piping a more desirable procedure than inspecting pipe for 1/8-in. sodium depths. Salvageable items such as valves were separated from the pipe sections for possible future use. All pipe was designated as saleable for scrap.

No cleaning operations were performed on the steam generators or the secondary-pump casings. After removal of the secondary sodium heat-transfer piping, each steam generator was dismantled into three sections: evaporator, superheater, and eliminator. Cover-plates were welded over the pipe nozzles while using an inert (nitrogen) atmosphere inside, under a small positive pressure. Cradle extension assemblies were installed on the vessels to provide support during shipment. Subfloor shoring was needed to support the total load while the steam generator was positioned under the crane hook. Wood

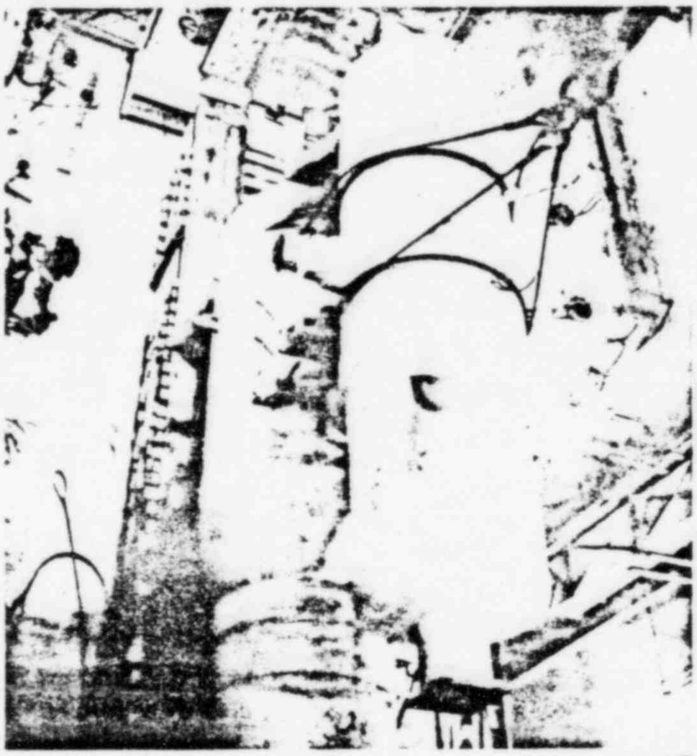

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Figure 6-17. Removal of Superheater from Vault

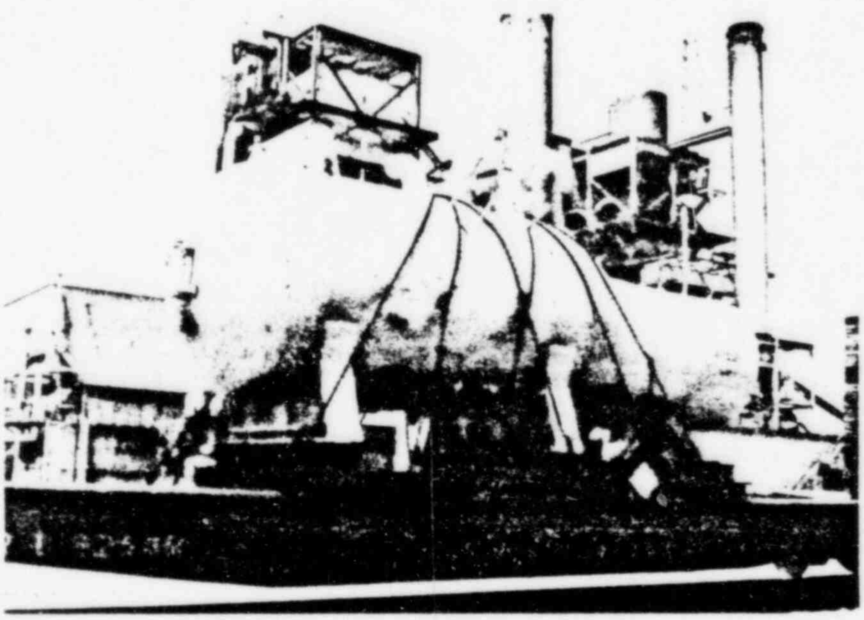

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Figure 6-18. Steam Generator Loaded for Shipment cribbing was used to distribute the load on the floor. Removal rigging and the flatcars load are shown in Figures 6-17 and -18. The steam generators were shipped to NRTS.

A gasketed steel plate was installed on each secondary-pump barrel. Seal-plates had been previously welded over the pipe nozzles during the isolation procedures. After the casings were removed they were crated for shipment along with the drive equipment and internal parts. Two pumps were shipped to the NRTS and one to the LMEC.

The two EM pumps, one sodium service, and one sodium transfer were removed during dismantling of the secondary-sodium service system and shipped to PNL in Richland, Washington.

6.6 DISPOSITION OF CONTAMINATED AND IRRADIATED MATERIAL

The HNPF Retirement Plan defined the requirements and stipulations for the disposition of material and equipment. Allowable radiation levels for unrestricted use are given in Appendix VIII.

By following these rules valuable reusable contaminated equipment was salvaged. For example the primary pumps having been drained of sodium were easily decontaminated by steam-cleaning and water washing; the IHX's were shipped to storage with a small residual amount of irradiated sodium on the primary side. Since there was no immediate user for these items they are being held as AEC property at NRTS. The radioactivity in the residual primary sodium in the IHX units will have decayed to a very low level in a few years. Decontamination at that time can be conveniently accomplished with a hot-sodium or NaK flush without physical damage to the IHX fabrication materials. Appendix V lists the disposition of all reusable material and equipment.

Nonreuseable contaminated and/or irradiated material and equipment (by previously established definition) was either permanently stored within the shielded sealed containment structure or removed from the site for burial at a licensed disposal site (Beatty, Nevada).

As described in Section 2.0 the reactor vessel, which is surrounded by massive shielding concrete and is specifically designed to provide proper containment for many c· ·.tions, was used to permanently store in place most of the radioactive scrap items. These consisted primarily of the moderator elements, the control and safety elements, the neutron-source elements, and miscellaneous fuel element and core hardware. The total calculated activity inventory was $\sim 3 \times 10^5$ curies (see Section 6.9 following for an Isolation-Structure Activity Inventory). Other spaces within the isolation structure were used to store items of lower-level activity.

Fuel-storage Pit No. 3 was used as a disposal for such miscellaneous irradiated hardware as hanger rods, M-3 snorkels, cut pieces of Zircaloy thimbles, and dummy elements. Three moderator elements previously removed from the core for examination were stored in the moderator cell area. Materials to be stored in these three significant volumes (reactor vessel, fuel-storage Area No. 3, moderator storage) were identified in the Activity Specification No. 7 requirements (see Appendix I). Specification No. 7 also gave an option in the cases of those items not specifically identified, for storage either in appropriate containment structure spaces or by burial off-site.

Most of the radioactive material designated for off-site burial was of such low level that a plastic film covering was sufficient for contamination containment. About one truckload of such debris was packaged in sealed plastic bags which were then packed in corrugated paper boxes for shipment by truck. Other items of low-level activity buried off-site were the primary-pump sodium-level balancing legs, plugging-meter components, primary-sodium service-system pipe, and components such as pumps, flowmeters, and valves. As noted previously the large primary-system piping was removed and cleaned with the intent of salvage; however, the radioactivity could not be economically reduced to the unrestricted use level for some of the pipe and as a result it was also shipped off-site for burial.

Some additional scrap having a radioactive level sufficiently high to require shielding was given special attention. A shielded shipping cask and several canisters had been designed and built for removal of used moderator elements. Four of the canisters were loaded with the high-level radioactivity scrap and were shipped by means of the cask. The canisters were buried intact at the Beatty facility.

Established procedures for monitoring and handling radioactive materials were used in accomplishing this operation, and no personnel overexposures or unusual events occurred.

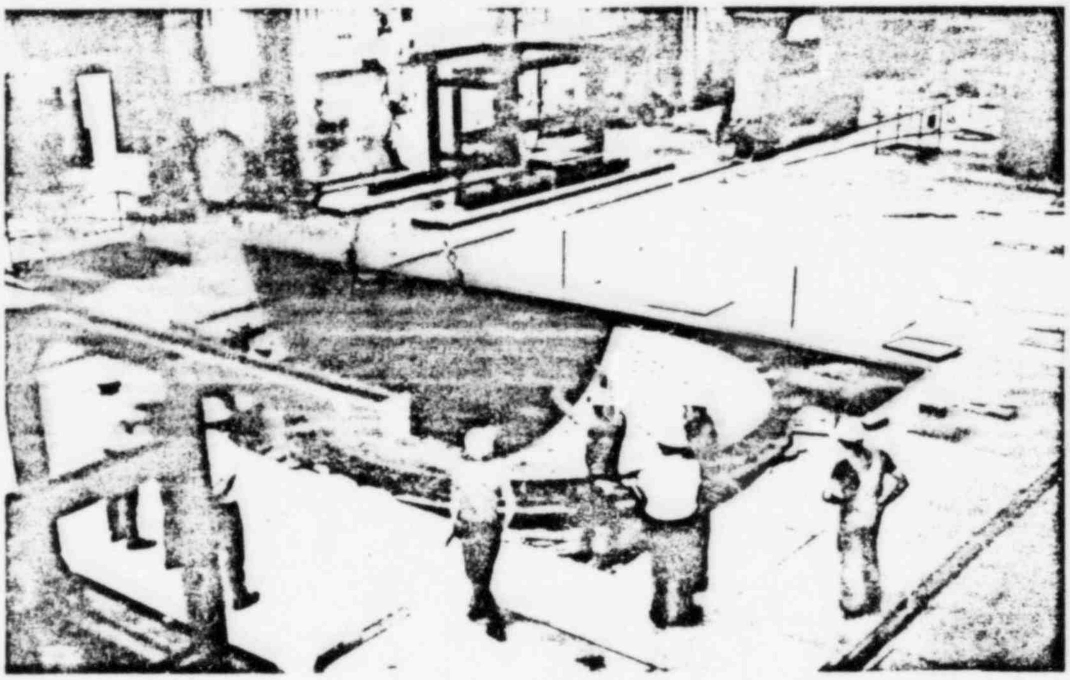
6.7 REACTOR ISOLATION

Because the reactor vessel had been installed surrounded with massive concrete shielding and had excellent containment afforded by the stainless-steel construction, it was used to store some 208 irradiated items. Studies of corrosion problems resulting from groundwater attack on the stainless-steel reactor vessel at the thinnest wall dimension indicated that penetration might not be expected before 500 yr based on the published corrosion rate of 0.1 mil/yr. The items stored in the reactor vessel consisted of the following:

- 21 control elements (18 stainless-steel thimbles and 3 Zircaloy-2 thimbles),
- 2 neutron sources,
- 40 dummy fuel elements,
- 2 sodium-level instruments,
- 3 sodium-temperature instruments,
- 133 subassemblies (shield-plug, hanger-rod, and process-tube),
- 1 shield plug and cut hanger rod, and
- 6 more subassemblies (shield-plug, cut hanger-rod, and process-tube).

Adequate containment of the reactor was provided below the reactor operating floor level by cutting all pipe and other connections which penetrated the reactor vessel, then seal-welding these openings with pipe caps or steel plates. Eighteen pipes 6- to 20-in. diam were cut and capped. Weld-quality inspection and leak-testing was maintained at the same standards as those for new sodium-system assembly.

After taking a final inventory of the items stored in place in the reactor core vessel, the loading face shield plugs were covered and seal-welded with a fitted 1/2-in. carbon steel plate. Thirty-one plug welds were made on a 16-in.



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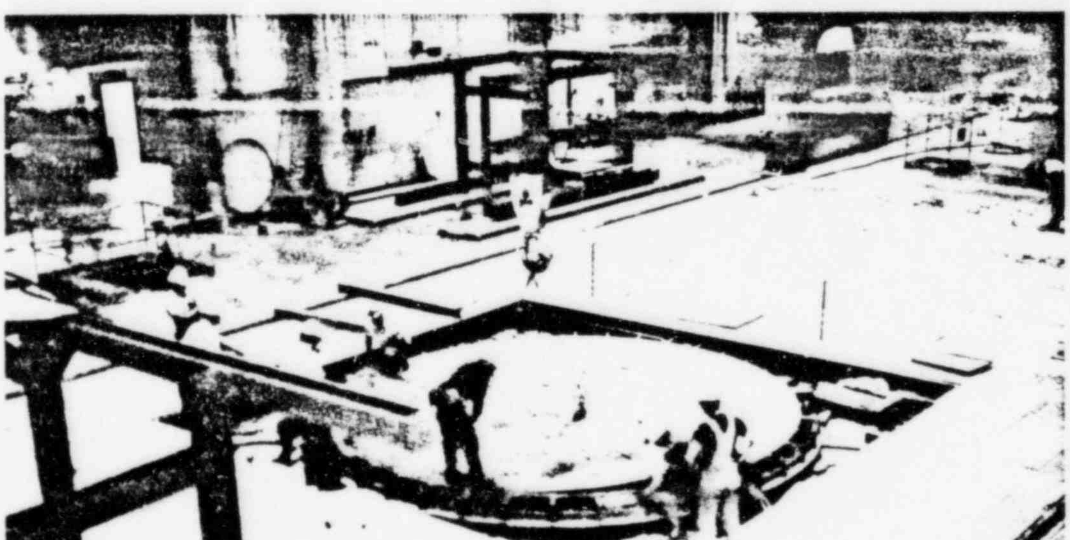

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Figure 6-19. Positioning Loading-Face Shield Cover-Plate triangular lattice, and the circumferential weld anchored the plate to the loading face shield (see Figures 6-19 and -20). The cerrobend seal area was then coated with a cold-setting bituminous sealant.

The reactor cavity liner and the primary pipe gallery were sealed in the following way. The three 2-in. pipes draining and venting the reactor cavity were cut and capped by welding. The pipe hanger penetrations into the gallery were weld-capped (see Figures 6-21 and -22) as were the electrical and instrument penetrations into the core vessel and cavity.

Sixteen nitrogen and ten thermocouples penetrations on the vertical periphery of the loading-face shield were sealed by welding as shown in Figure 6-23.

Leak-testing of the core vessel was accomplished with a helium leak-detector and by means of pressure decay. The reactor cavity and reactor vessel were purged with helium until a 25%-helium 75%-nitrogen atmosphere was obtained. Vessel pressure was then brought to 2.30 psig. Finding no leak in any of the closure welds, a 24-hr pressure-decay test was begun at the same pressure. Barometric pressure corresponded to 28.62-in. Hg at the beginning and at the end of the test. The test gages followed interim barometric pressure changes, and still indicated the original pressures at the end of the test. With the gages used a leakrate of 0.002 scfm could have been detected in the 24-hr period.

In a similar way the reactor cavity liner was leak-checked at 2 psig with the helium mass spectrometer. Several leaks around instrument penetrations and the pipe hanger caps were located and repaired. A 24-hr pressure-decay test from 2 psig indicated after barometric correction a leakrate from this region of 0.03 scfm. This was considerably less than the allowable 0.5 scfm for the core vessel; therefore, the liner and vessel were considered sealed according to the specification. The vessel pressure was then reduced, and the valve on Line-148 cap was removed. This 1/4-in. pipe connection was welded closed and a guard (pipe and cap) was welded over the connection.

The gage was removed from the loading-face shield cover-plate and the opening (1/4-in. pipe stub) was also seal-welded. The reactor vessel was then pressurized to 1.1 psig through the remaining 1/4-in. bar stock valve at the helium line cap. With this valve closed the 1/4-in. helium supply line was

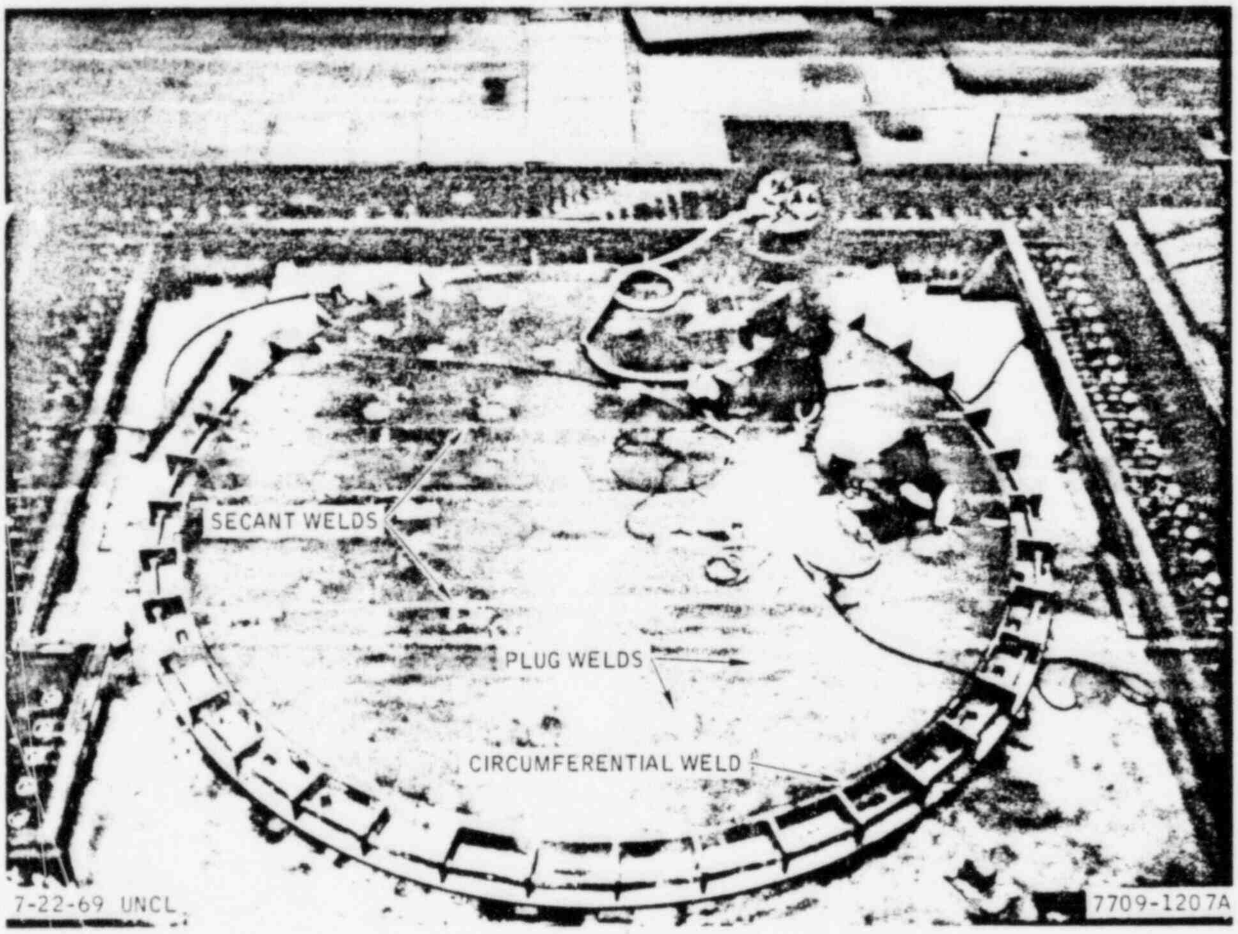

Figure 6-20. Welding of Cover-Plate

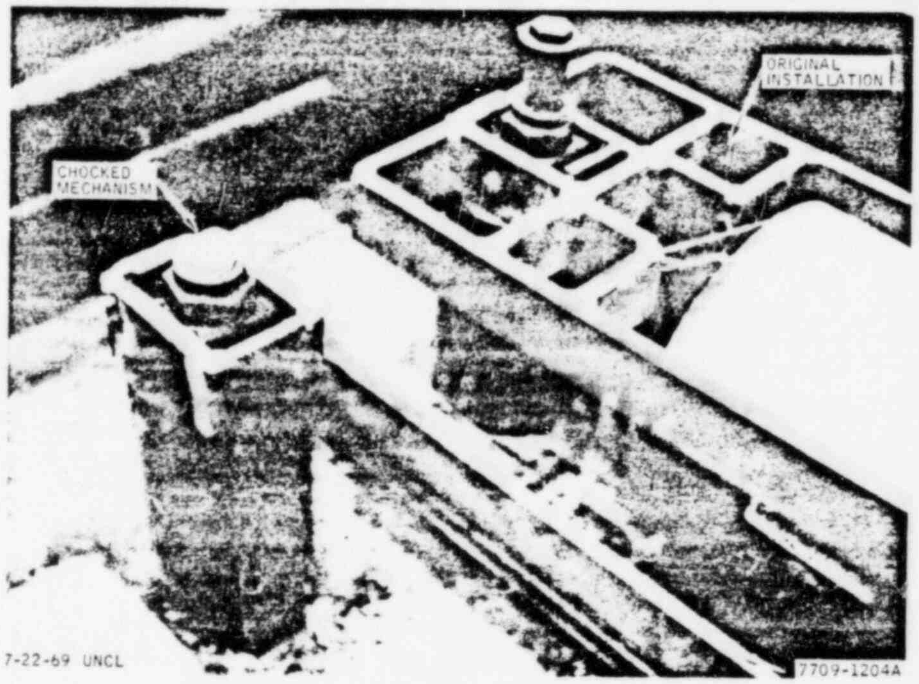

Figure 6-21. Cavity Pipe Hanger Chock Installation

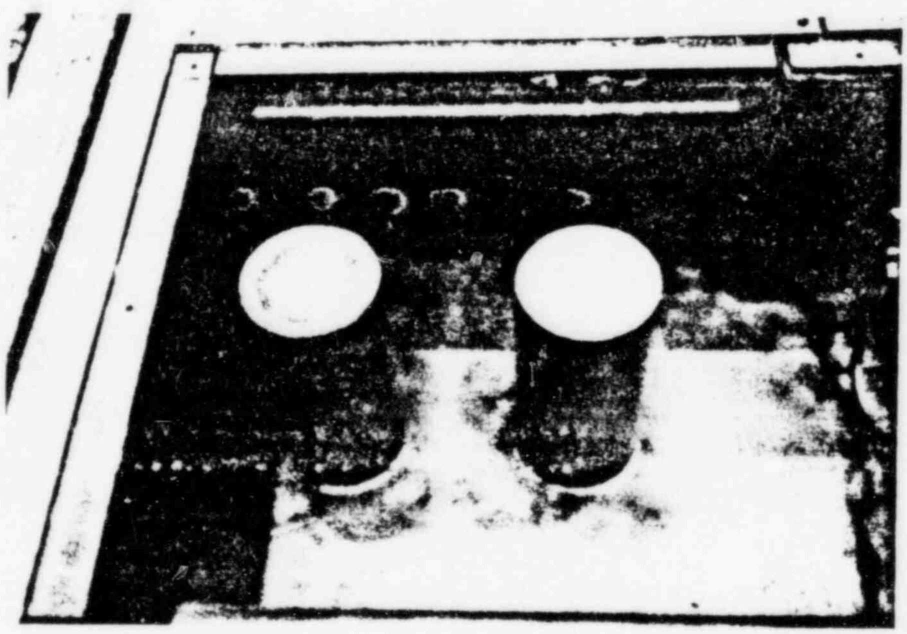

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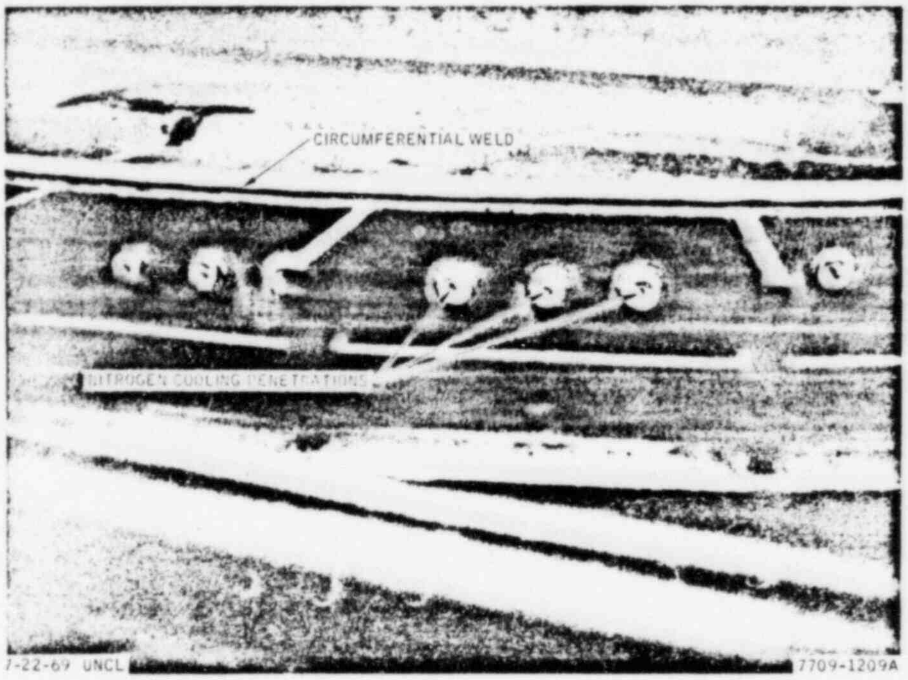
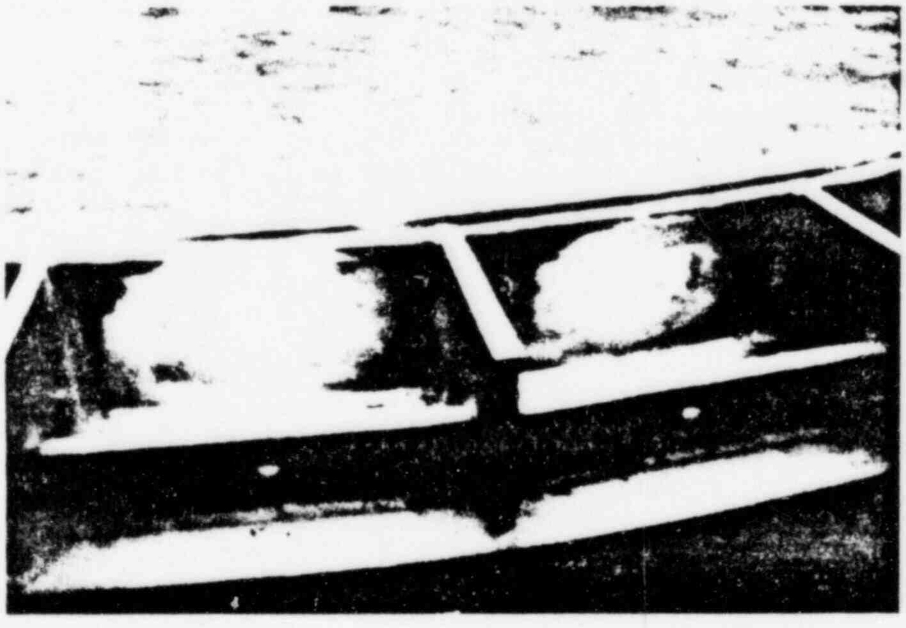
Figure 6-22. Completed Chock and Seal Installation



a. N_2 Cooling Penetrations



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b. Thermocouple Penetrations

Figure 6-23. Loading-Face Shield Seal Welding cut and welded, and the valve stem was seal-welded to the valve body. The valve and stub were then enclosed by welding a guard (pipe and cap) over them as shown in Figures 6-24 and -25. A guard (pipe and cap) was also welded over the reactor vent-line stub.

Following the reactor-cavity leaktest, the helium supply line was cut just outboard of the valve. This valve was also sealed and covered by welding a guard (pipe and cap) over it.

The annular space around the loading-face shield and the pipe trenches were then filled to floor level with concrete. Another rectangular steel plate 1/2-in. thick was then fitted and welded over this area as shown in Figure 6-26.

This effort completed the isolation of the reactor vessel and cavity liner. The long-term condition for the vessel therefore began with approximately 75% N_2 and 25% He_2 at 1.1 psig.

6.8 RETIREMENT OF AUXILIARY SYSTEMS

This activity included all HNPF systems and equipment not specifically covered by the other eleven activity specifications. The major items of concern were as follows:

1) Main steam and feedwater, and emergency feedwater system.
2) Chemical feed system.
3) Reactor control and protection system.
4) Radioactive liquid-waste system.
5) Radioactive-vent system.
6) Preheat system.
7) Helium system.
8) Nitrogen system.
9) Auxiliary steam system.
10) Loading-face shield nitrogen cooling system.
11) Emergency power (including diesel load bank) system.

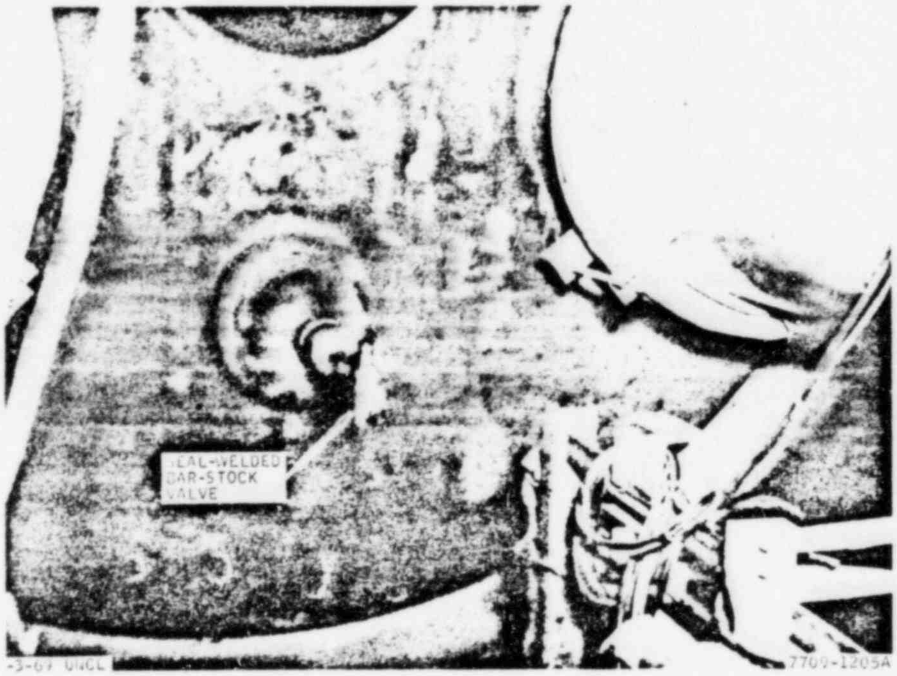


Figure 6-24. Reactor Vessel Helium Supply Line Isolation

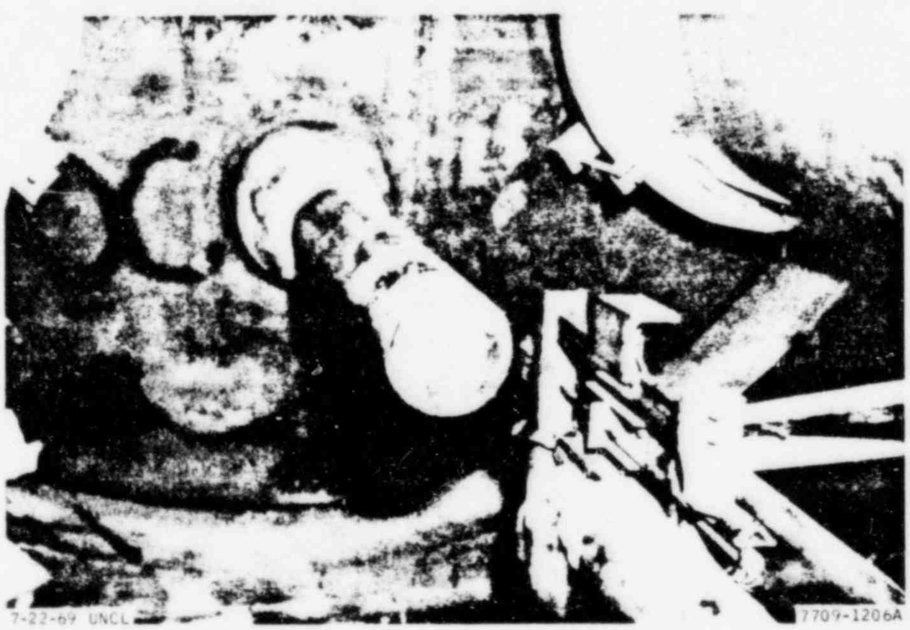


Figure 6-25. Reactor Vessel Helium Supply Final Closure

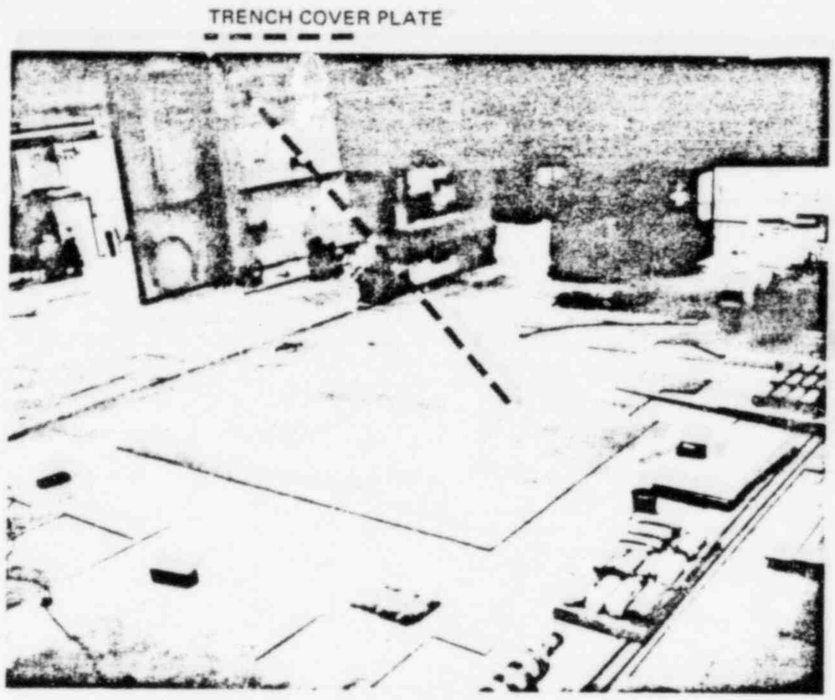


Figure 6-26. Completed Installation of Reactor Trench Cover-Plate

12) Radiation-detection and monitor system.

13) Heating and ventilating system.

14) Compressed-air system.

15) Fire-detection and protection systems.

16) Fire-fighting equipment.

17) Electrical systems.

18) Liquid-nitrogen system.

19) Communication and alarm system.

20) Building services (water, sewer, etc.).

21) Fire protection external to building.

22) Building cranes.

23) Fuel-handling machines.

24) Maintenance cell shielding window, Kollmorgen periscope, and master slave manipulator.

This activity turned up the most reusable items in number if not dollar value; the disposal criteria were in accordance with the category designations as described in the Retirement Plan.

Generally the removal and disposition of the auxiliary systems was a straightforward mechanical operation, using conventional safety precautions and radiation monitoring and personnel exposure control as required.

The retirement of building services (electricity, water, communications, sewer) was completed toward the end of the program so that they would be available to the retirement crew as long as practical. Sewer, steam, water, and electrical pipes and conduits penetrated the isolation structure; therefore, cutting these and sealing them at the building foundation constituted part of the structure final closure.

The retirement of a few auxiliary items of particular interest is described in the balance of this section.

The service cranes were dismantled and packed for shipment under the supervision of the manufacturer's representative. One went to the AEC-LMEC, the other was assigned to Pacific Northwest Laboratory.

The fuel-handling machine was designated as scrap. A portion of it, however, grapple and internal drive, was too radioactive to remove from the site as scrap. It was therefore placed in Moderator Element Storage Pit No. 19 for permanent storage.

The periscope, two manipulators, and two shielding windows were salvaged. These were easily decontaminated, then packaged for storage and future use at another site.

The radiation-detection system was not contaminated and was in good working condition when dismantled and packaged for future use at another site.

The emergency power system (diesel generator, switch gear, and loadbank) was sold to CPPD and left in place.

6.9 SECURING OF ISOLATION STRUCTURE

Primary considerations for treatment of the isolation structure were sealing and weatherproofing to prevent water intrusion and closure of all penetrations to prevent personnel access. In addition the structure was required to be one which would require no maintenance over the design life objective of 100 yr. An evaluation of methods for safely securing the isolation structure shows that removal of the reactor building superstructure was desirable economically. Pouring of a thick concrete slab inside the building was considered but the accomplishment of a proper seal around building columns and at the periphery of the building would be too costly.

Within the containment structure three regions were designated as the principal storage areas: (1) the core tank and environs, (2) fuel-storage cell Group 3, and (3) moderator-element storage cells. All radioactive storage space within the containment structure is identified in Figure 6-27.

The radioactive inventory for the three high-level radioactivity regions is listed in Tables 6-1, -2, and -3. All references in the tables and text of this section to radioactivity levels are based on levels as of September 1969. Some low-level activity exists in the remainder of the isolation structure in the locations designated and in the items stored in it; the latter are described as follows:

1) Fuel-handling machine and grapple storage pit (Pit 19).

   Items placed in Pit 19 before filling this volume with concrete were:

   a) Fuel-handling machine lower unit (gas-lock assembly) and grapples,

   b) Spent-reactor startup source in lead cask, and

   c) Building ventilation filters.

   These items contain a total of approximately 1.3×10^{-1} curies of radioactivity. Most of this activity is represented by the activated stainless-steel cladding on the spent-reactor startup source.

2) Primary-sodium fill-tank vault

   The primary-sodium fill-tank vault contains 72 filters from the building ventilation system and the cut-up remains of the low-level

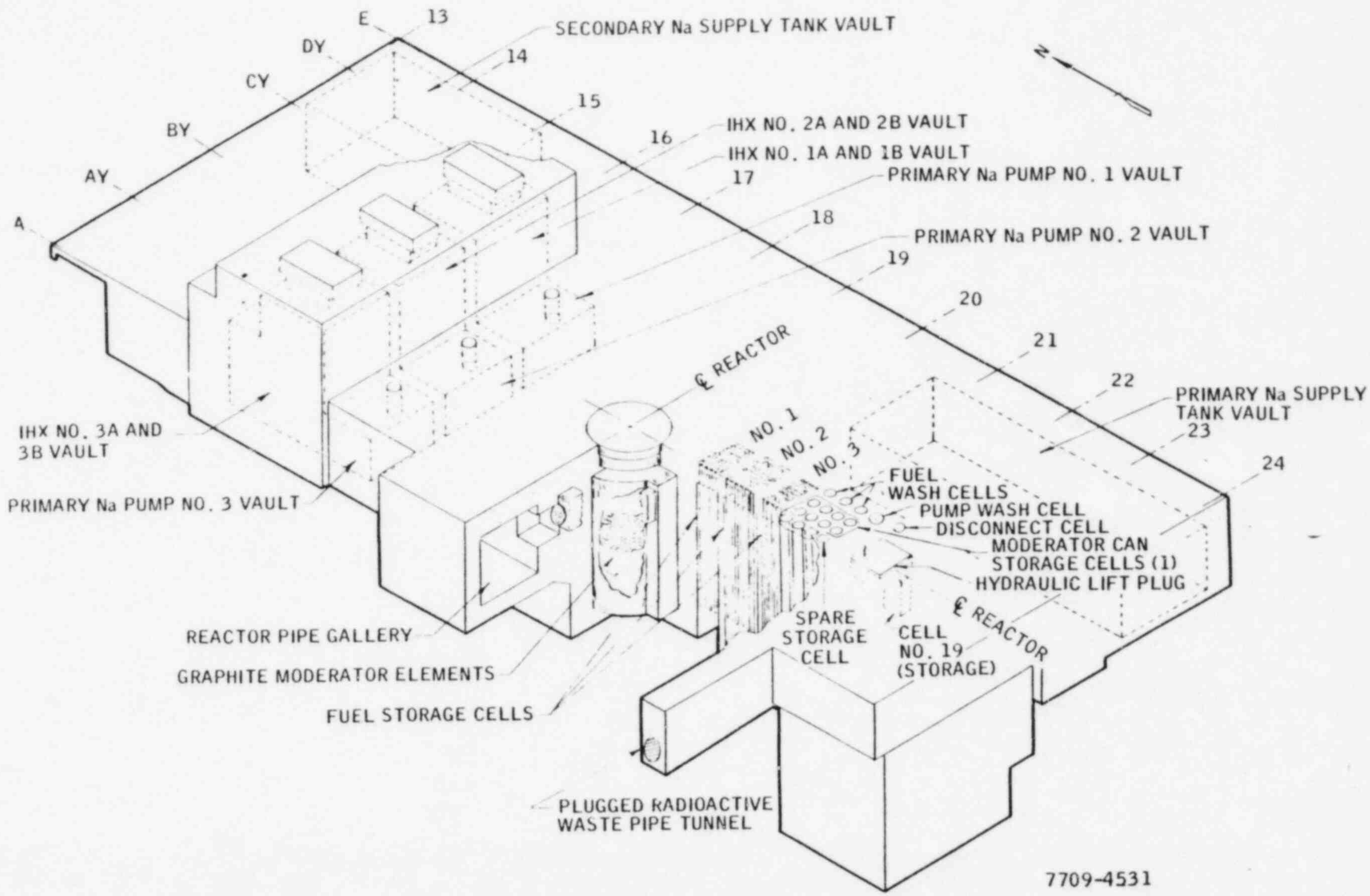


Figure 6-27. Radioactivity Storage Spaces

TABLE 6-1

RADIOACTIVITY IN AREA 1 FIVE YEARS AFTER REACTOR SHUTDOWN (September 1969)

Component	Activity (Ci)			
	Fe^{55}	Co^{60}	Ni^{63}	Sm^{151}
Skin of Moderator Cans (304 SS)	1.1×10^{5}	3.1×10^{4}	6.8×10^{3}	
Thermal-Shock Liners (304 SS)	3.8×10^{3}	1.1×10^{3}	2.4×10^{2}	
Gridplate (304 SS)	1.3×10^{4}	3.8×10^{3}	8.5×10^{2}	
Control Elements (Hastelloy-X skin and samarium internal)	2.2×10^{2}	4.4×10^{3}	2.1×10^{3}	6.3×10^{3}
Reactor Bellows (321 SS)	3.2×10^{-1}	1.4×10^{-2}	3.2×10^{-3}	
Upper Cavity Liner (405 SS)	2.5×10^{-1}	7.4×10^{-2}	1.6×10^{-2}	
Loading-Face Shield Bottom Face (304 SS)	7.0×10^{-1}	2.0×10^{-1}	4.5×10^{-2}	
Loading-Face Shield Cooling Coils (carbon steel)	3.1×10^{-2}	6.1×10^{-3}		
Thermal Shields (carbon steel)	1.6×10^{4}	3.2×10^{3}		
Cavity Liner (carbon steel)	1.2×10^{2}	2.3×10^{1}		
Reinforcing Bar outside Cavity Liner (carbon steel)	5.5×10^{2}	1.1×10^{2}		
Concrete Around Cavity Liner	*			
Reflector Elements (304 SS)	2.4×10^{4}	6.9×10^{3}	1.5×10^{3}	
Reactor Vessel (304 SS)	7.3×10^{3}	2.1×10^{3}	4.7×10^{2}	
Dummy Elements and Instrument Thimbles (304 SS)	1.5×10^{4}	3.2×10^{3}	9.0×10^{2}	
Process Tubes (Zircaloy-II)	2.0×10^{2}	2.3×10^{4}	2.6×10^{2}	
Support Pins (304 SS)	7.8×10^{3}	2.2×10^{3}	4.9×10^{2}	
Sodium Piping in Pipe Chase (304 SS)	9.0×10^{0}	2.7×10^{0}	5.9×10^{-1}	
Loading-Face Reflector Plates (304 SS)	3.0×10^{-1}	8.6×10^{-2}	1.9×10^{-2}	
Superex Insulation	2.7×10^{-1}	5.4×10^{-5}	3.1×10^{-6}	
Loading-Face Shield Reinforcing Bar (carbon steel)	1.2×10^{-1}	2.3×10^{-2}		
Outer Vessel (carbon steel)	4.3×10^{2}	8.7×10^{1}		
Cavity-Liner Cooling Coils (carbon steel)	3.5×10^{2}	7.0×10^{1}		
Reactor Support Piping (carbon steel)	3.5×10^{0}	3.3×10^{-1}		
Concrete in Loading-Face Shield	*	*	*	*
Total	2.0×10^{5}	7.6×10^{4}	1.2×10^{4}	6.3×10^{3}

*See Appendix II of AI-AEC-MEMO-12794

TABLE 6-2

RADIOACTIVITY IN AREA 2 FIVE YEARS AFTER REACTOR SHUTDOWN (September 1969)

Component	Activity (Ci)		
	Fe^{55}	Co^{60}	Ni^{63}
Process Tubes (Zircaloy-II)	1.2×10^{1}	1.1×10^{3}	1.6×10^{1}
Control Rod Thimbles (Zircaloy-II)	2.7×10^{1}	2.5×10^{3}	3.5×10^{1}
Control-Rod Thimbles (304 SS)	1.6×10^{3}	4.6×10^{2}	1.0×10^{2}
Dummy Elements (304 SS)	9.6×10^{2}	2.8×10^{2}	6.0×10^{1}
Reactor Source Element (304-SS Clad) SbO_4 and Be	3.2×10^{2}	9.2×10^{1}	2.0×10^{1}
Total Activity	2.9×10^{3}	4.4×10^{3}	2.3×10^{2}

NOTE: The radioactivity in Area No. 2 of the isolation structure is limited to the induced activity of the irradiated components which are stored in that area. No induced radioactivity and no significant radioactive contamination is inherent to this portion of the isolation structure.

TABLE 6-3

RADIOACTIVITY IN AREA 3 FIVE YEARS AFTER REACTOR SHUTDOWN (September 1969

Cell	Item	Activity (Ci)
MS-1	No irradiated or contaminated items.	0
MS-2	Radioactive liquid-waste transfer pumps, pipe and valves.	$\sim 6.9 \times 10^{-6}$
MS-3	Canistered lower portions of control-rod thimbles TZ-6 and TZ-12.	$\sim 6.9 \times 10^{-2}$
MS-4	Canistered moderator element SN-142.	Fe^{55} 7.8×10^{2}
MS-5	No irradiated or contaminated items.	0
MS-6	Radioactive liquid-waste pipe and valves.	$\sim 6.3 \times 10^{-6}$
MS-7	Solidified residue from the radioactive liquid-waste storage tanks.	$\sim 1.2 \times 10^{-2}$
MS-8	Canistered moderator element SN-58.	Fe^{55} 7.8×10^{2}
MS-10	Radioactive liquid-sample car and one nuclear instrument.	$\sim 2.4 \times 10^{-8}$
MS-11	Radioactive liquid-waste storage tank recirculation lines.	$\sim 7.8 \times 10^{-5}$
MS-12	Canistered moderator element SN-45.	Fe^{55} 7.8×10^{2}
		Co^{60} 2.2×10^{2}
		Ni^{63} 4.8×10^{1}
		Fe^{55} 2.3×10^{3}
		Co^{60} 6.6×10^{2}
		Ni^{63} 1.4×10^{2}

NOTE: The radioactivity contained in Area No. 3 of the isolation structure is limited to radioactive contamination and induced radioactivity in the reactor components and other items stored in the area. No induced radioactivity and no significant radioactive contamination is inherent to this portion of the isolation structure.

and intermediate-level-radioactive liquid-waste storage tanks. Early in 1970 there was a total activity of approximately 1.5×10^{-2} curies of radioactivity in these items.

3) Moderator Elements Storage Cells

The moderator storage cells contain two 55-gal. drums of control-rod actuator parts and 12 control-rod actuator housings. A total activity of approximately 1.5×10^{-3} Ci is in these items.

4) Primary Pipeway

The primary pipeway contains the primary-sodium heat-transfer system which was steam-cleaned in place (approximately 550 ft of 14-in.-diam pipe). A total activity of approximately 7.0×10^{-4} Ci is in this piping.

5) Intermediate Heat Exchanger Vaults

The No. 1 and 2 heat-exchanger vaults each contains approximately 140 ft of the 14-in. primary-sodium heat-transfer piping which was cleaned in place. The No. 3 heat-exchanger vault contains, in addition to the No. 3 primary-sodium heat-transfer system piping, the primary pipe systems which were removed to provide access for the removal of the heat-exchanger vessels. Vault No. 3 contains a total of 320 ft of primary-system heat-transfer pipe. A total activity of approximately 5.4×10^{-4} Ci is in the primary piping in the three vaults.

6) Maintenance Cell

The radioactivity in the maintenance cell is the result of contamination of the internal surface areas of the cell. The items disposed of in the cell were items used for in-cell work: the tool positioner, jib manipulator, vacuum cleaner, and the in-cell crane. A total activity of approximately 2.3×10^{-4} Ci is in the maintenance cell.

7) Laundry and Decontamination Holdup-Tank Vault — Figure 2-6

The laundry and decontamination room liquid-waste holdup tanks were emptied before isolating the system; they contain a total activity of approximately 9.7×10^{-7} Ci.

8) Fuel-Storage Pit No. 2

This pit contains the M-3 moderator-element snorkel which is the upper end of the instrumented moderator element. The M-3 snorkel, which is in storage S-104, contains a total activity of approximately 0.1 Ci.

9) Maintenance-Cell Holdup-Tank Vault

The maintenance-cell holdup-tank vault and the adjoining pipe chases contain the two maintenance-cell holdup tanks, the radioactive liquid-waste transfer tank, and associated liquid-waste system piping. These tanks, which were emptied before isolating the systems, contain a total activity of 4.8×10^{-3} Ci.

Other spaces in the containment structure were either left empty or contain nonradioactive materials.

6.9.1 Sealing of Penetrations

All penetrations into the isolation structure were sealed by welded closures and/or by pouring expanding concrete (ordinary concrete in which 20 wt % calcium sulfoaluminate was added to the cement) into recesses. In addition certain areas of the isolation structure were specifically isolated from the balance of the structure; all penetrations (i.e. sodium piping, etc.) into the reactor-cavity liner from other parts of the isolation structure were sealed. Figure 6-28 locates the penetrations into the isolation structure.

6.9.1.1 Fuel-Storage Cells

All fuel-storage cells in Fuel-Storage Pits No. 1, 2, and 3 were sealed by welding the shield plugs or closure plates to the storage thimbles, and the tops of the storage thimbles to the liners of the penetrations through the reactor room floor. Additional welds were made wherever necessary to assure that the tops of the cells form seals against penetration of the cells or the fuel-storage pits. Fuel-storage cells in Pits No. 1 and 2, except Cell S-104 in No. 1 which contains the M-3 snorkel, were further sealed by filling with expanding concrete the recess between the reactor room floor and the top of the fuel-cell shield plug. The voids above the components located in Pit No. 3 and in Cell S-104 were filled with expanding concrete to a point approximately 1 in. below floor level.

The steel dust covers were then reinstalled over the concrete in these cells and welded to the tops of the cell liners. The method by which the fuel-storage cells were sealed is described in Figure 6-29.

6.9.1.2 Moderator Storage Cells

The moderator storage cells were sealed by replacing all cell plugs, welding the plugs to the cell liners, and filling the voids above the plugs with expanding concrete (see Figures 6-30 and -31).

6.9.1.3 Access Penetrations

Equipment- and man-access penetrations through the surface of the isolation structure were sealed by welding the stepped-concrete access plugs to the liners of the hatch penetrations and by filling recesses with expanding concrete (see Figures 6-32 through -35).

6.9.1.4 Stairway Openings and Pipeway Hatches

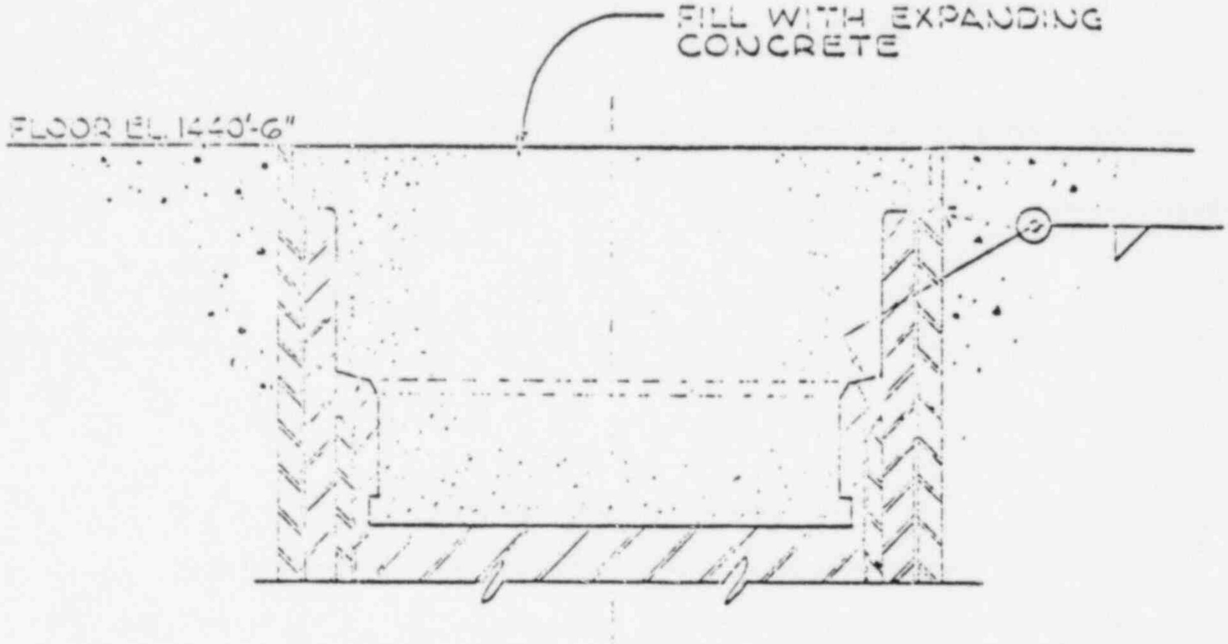
Stairway openings and pipeway hatches were sealed with expanding concrete as described by Figures 6-36, -37, and -38. Stairs, handrails, concrete curbing, etc., were removed from the openings, and enough of the existing concrete was removed to expose the reinforcing bars. Forming material was installed under the openings to support wet concrete, and mats of reinforcing bars. The openings were then filled with expanding concrete.

6.9.1.5 Secondary-Sodium Vent-Line Pipe Sleeves

The secondary-sodium vent-line pipe sleeves that penetrate the IHX cells were sealed as shown in Figure 6-39. Plugs were installed in the tapered end of the pipe sleeve and the pipe sleeves were filled with expanding grout. Form lumber was then placed around the oblong indentation and the penetration was filled with expanding concrete.

6.9.1.6 Pipe Sleeves in Steam-Generator Room Floor

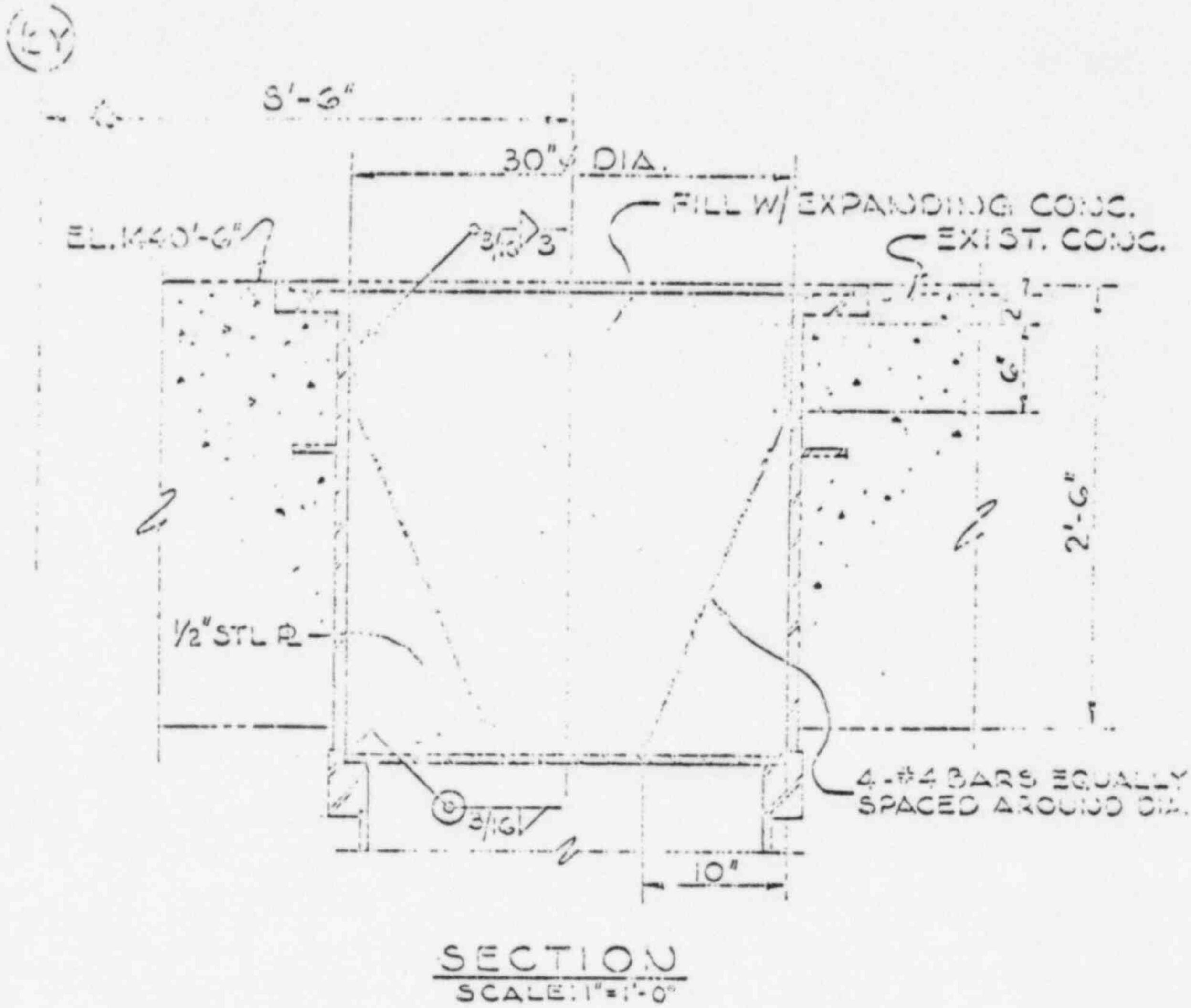
The vertical pipe sleeves in the steam-generator room floors were sealed as described in Figures 6-40 and -41. Forms and mats of reinforcing steel were placed in the openings, which were then filled with expanding concrete.



SECTION
NOT TO SCALE

NOTE: CELLS WITH STEEL CASED CONCRETE PLUGS SHALL BE WELDED AROUND PERIMETER TO LINING AND FILLED WITH EXPANDING CONCRETE.

Figure 6-29. Typical Fuel-Storage Cell Detail



NOTE: THIS DETAIL TYPICAL FOR CELLS WHERE PRECAST PLUG HAS BEEN REMOVED. WHERE PLUG IS STILL IN PLACE, WELD IT IN PLACE AND FILL TO EL.1440'-5½" WITH EXPANDING CONCRETE.

Figure 6-30. Sealing of Moderator Storage Cells

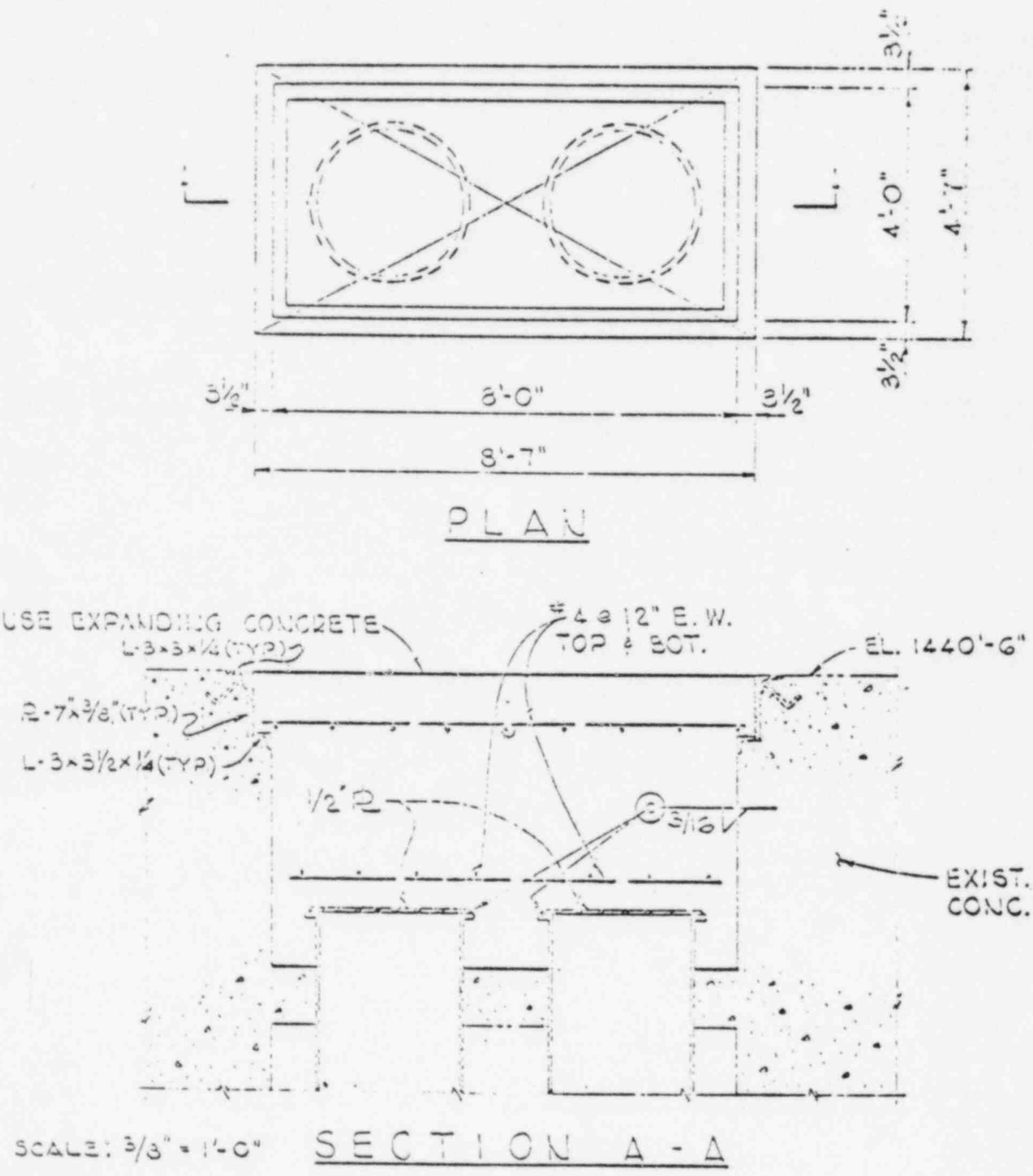
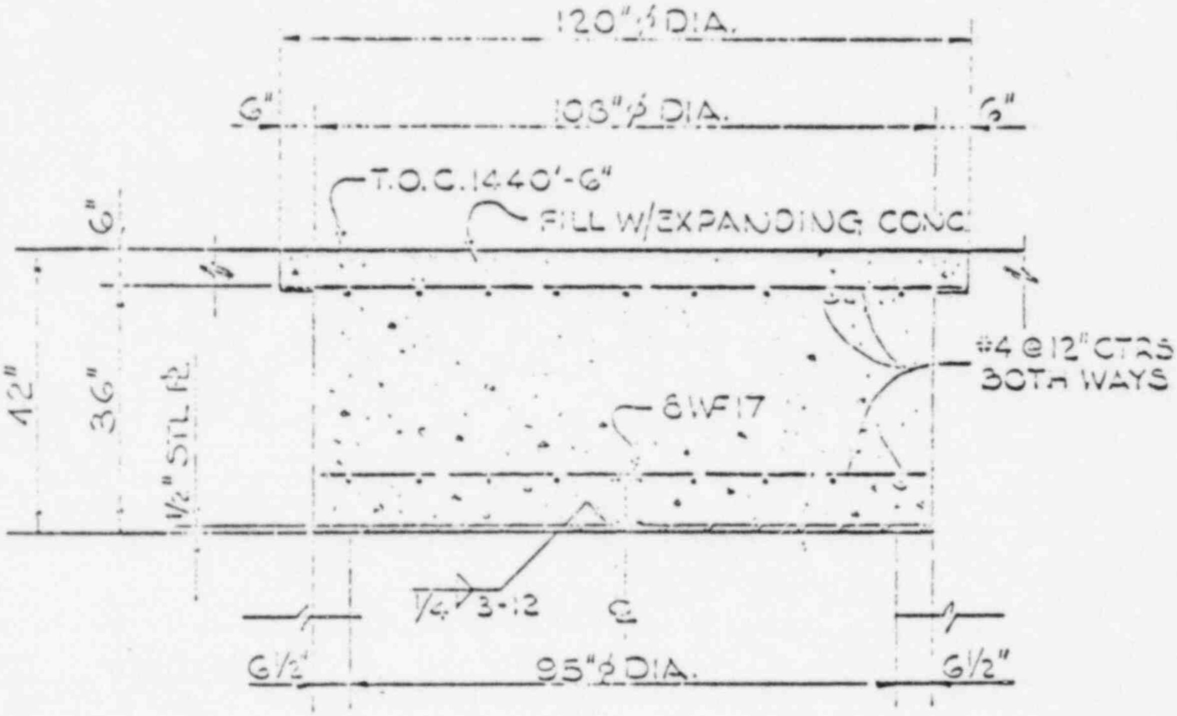


Figure 6-31. Sealing of Moderator-Can Grapple Storage Pit



(TYPICAL IN 3 PLACES)
SCALE: 3/8" = 1'-0"

NOTE:
BEFORE FILLING CUT OFF LEVELING
BOLTS @ EL. 1440'-6".

Figure 6-32. Sealing of Primary-Sodium Pump Opening

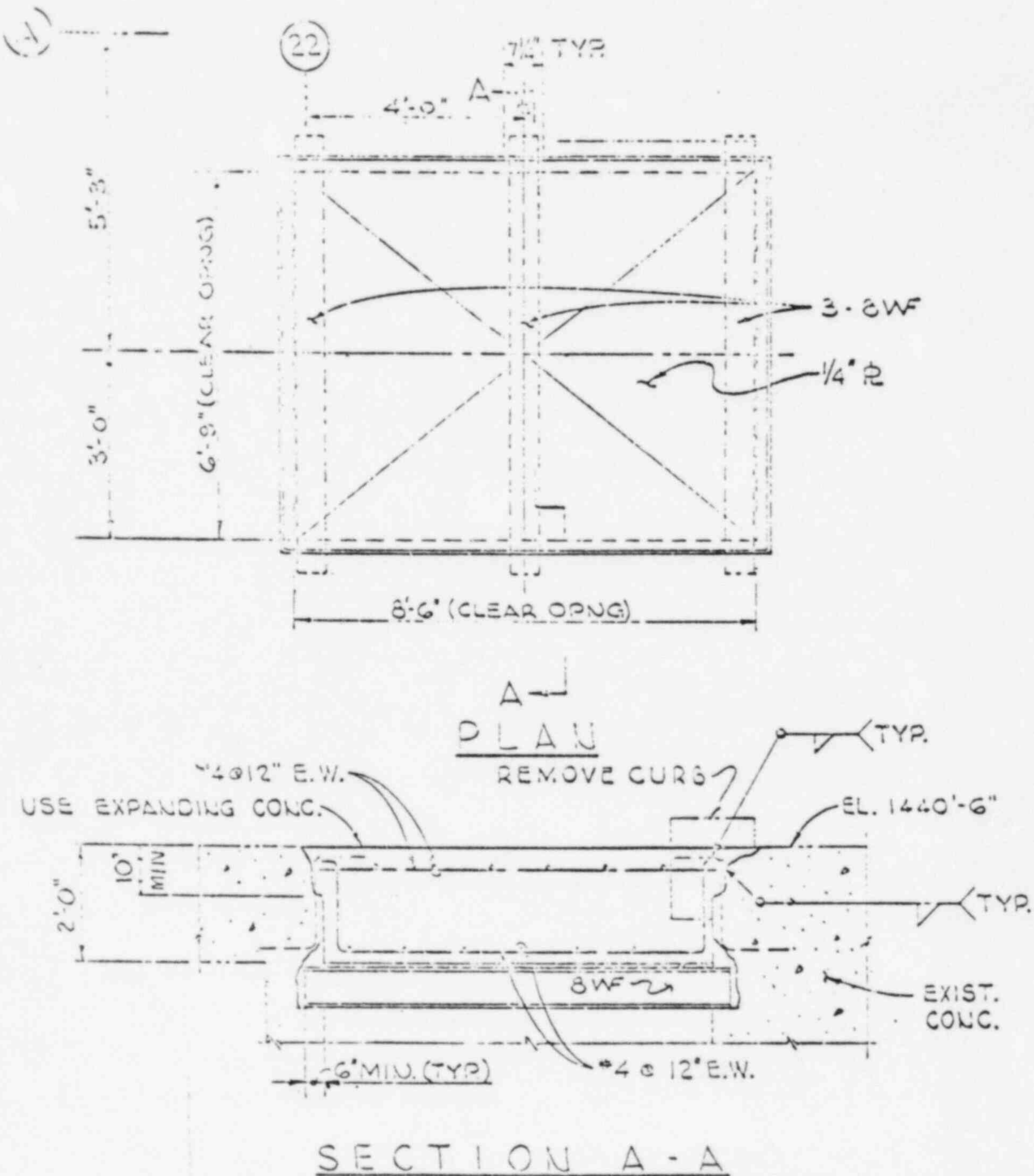
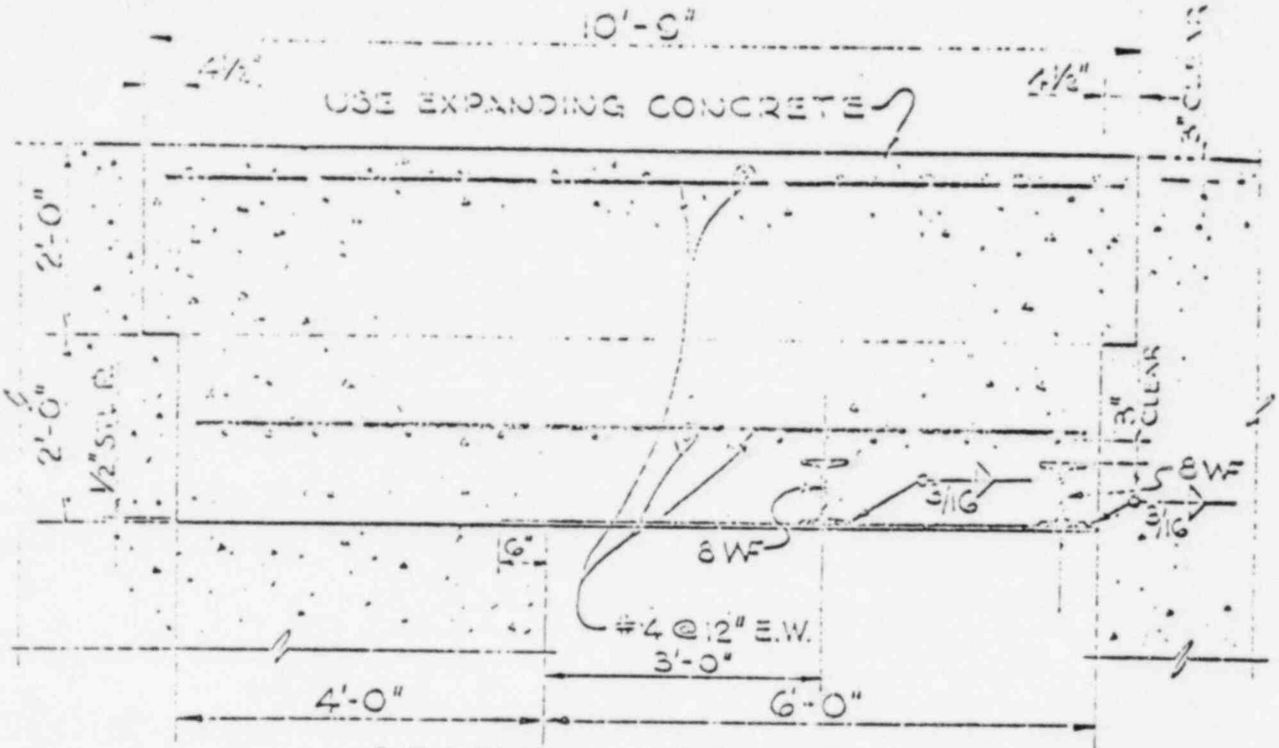
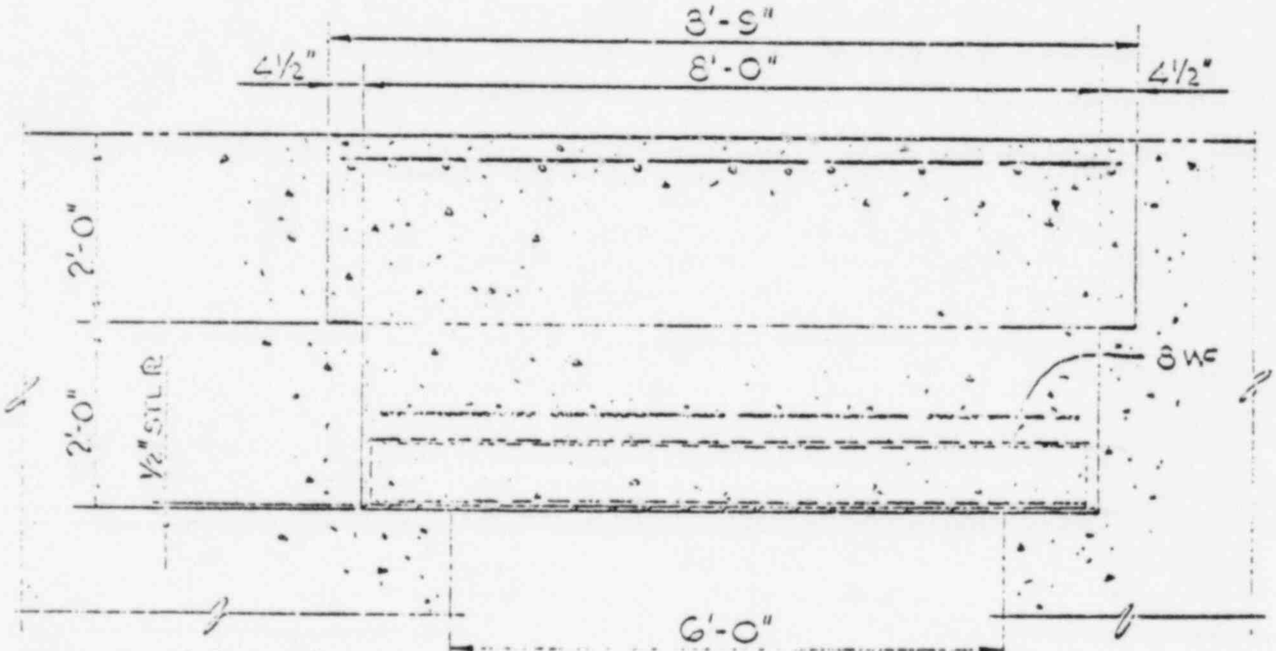


Figure 6-33. Sealing of Hatch 17



SECTION LOOKING EAST



SCALE: ½" = 1'-0"

SECTION LOOKING NORTH

Figure 6-34. Sealing of Fuel Shipping-Cask Loading and Storage-Pit Detail

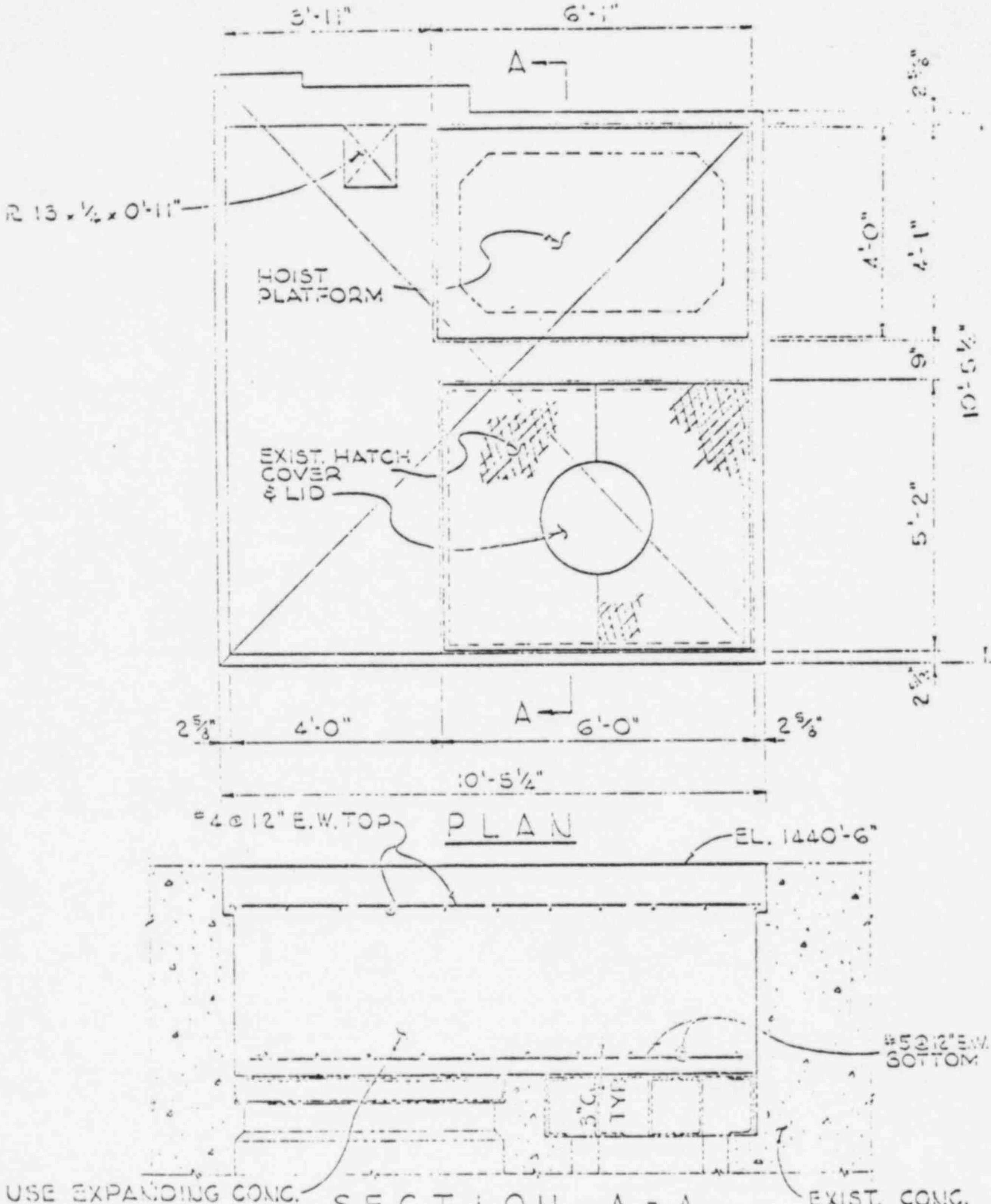
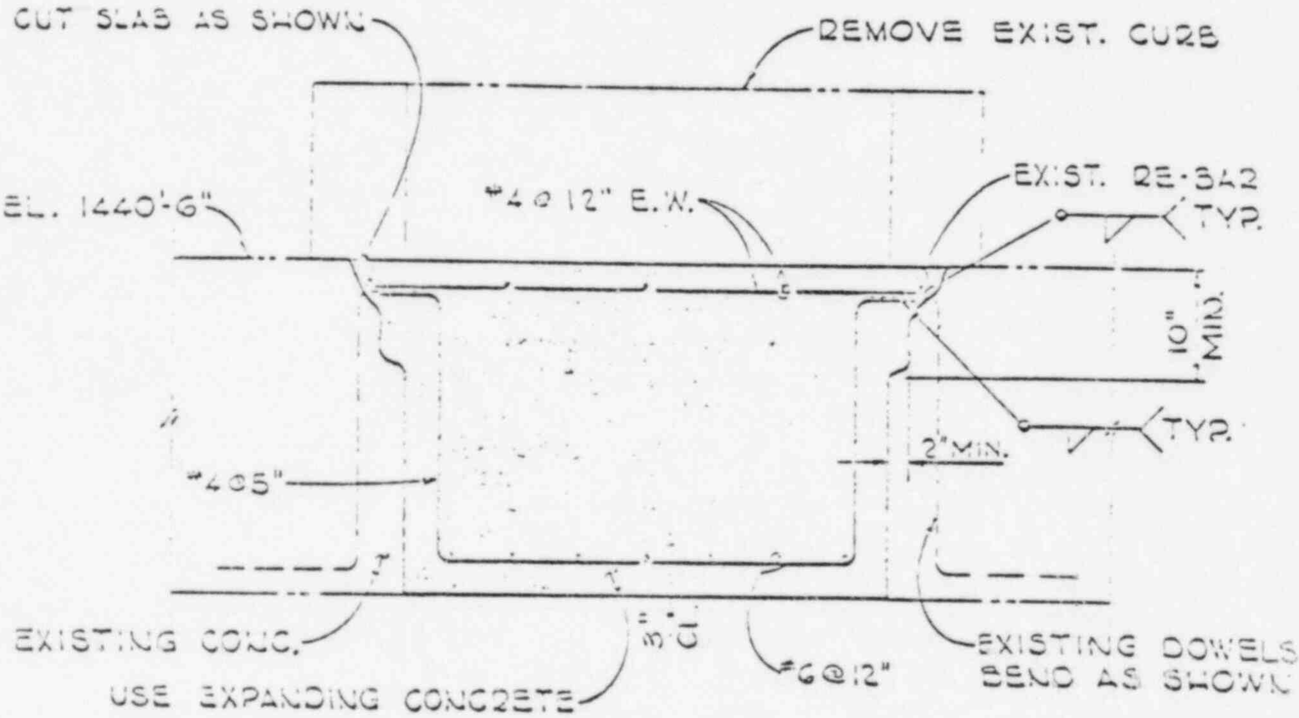


Figure 6-35. Sealing of Unloading Pit 19



(NO SCALE)

Figure 6-36. Sealing of Stairwells 1, 7, and 8

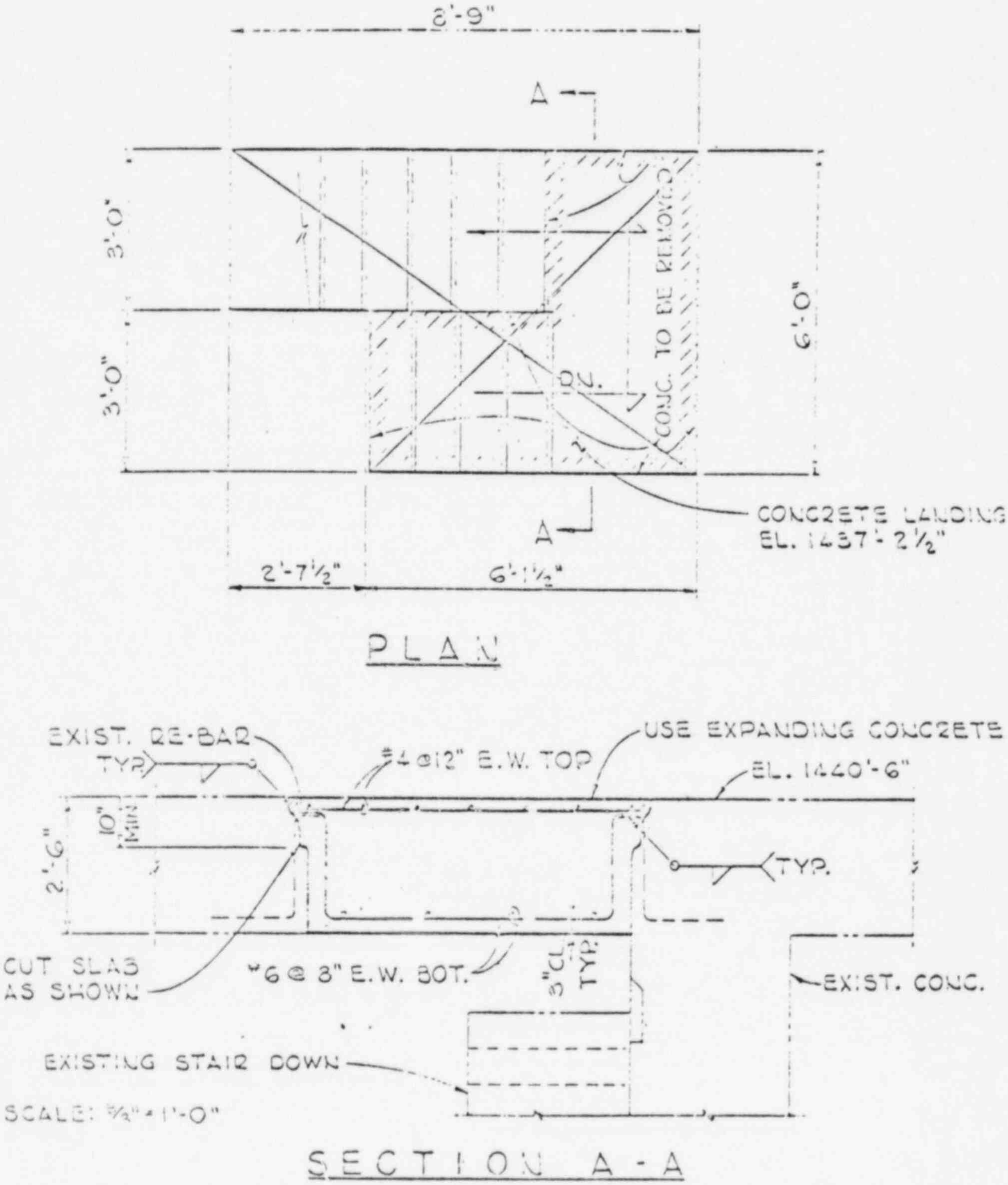


Figure 6-37. Sealing of Stairwell 9

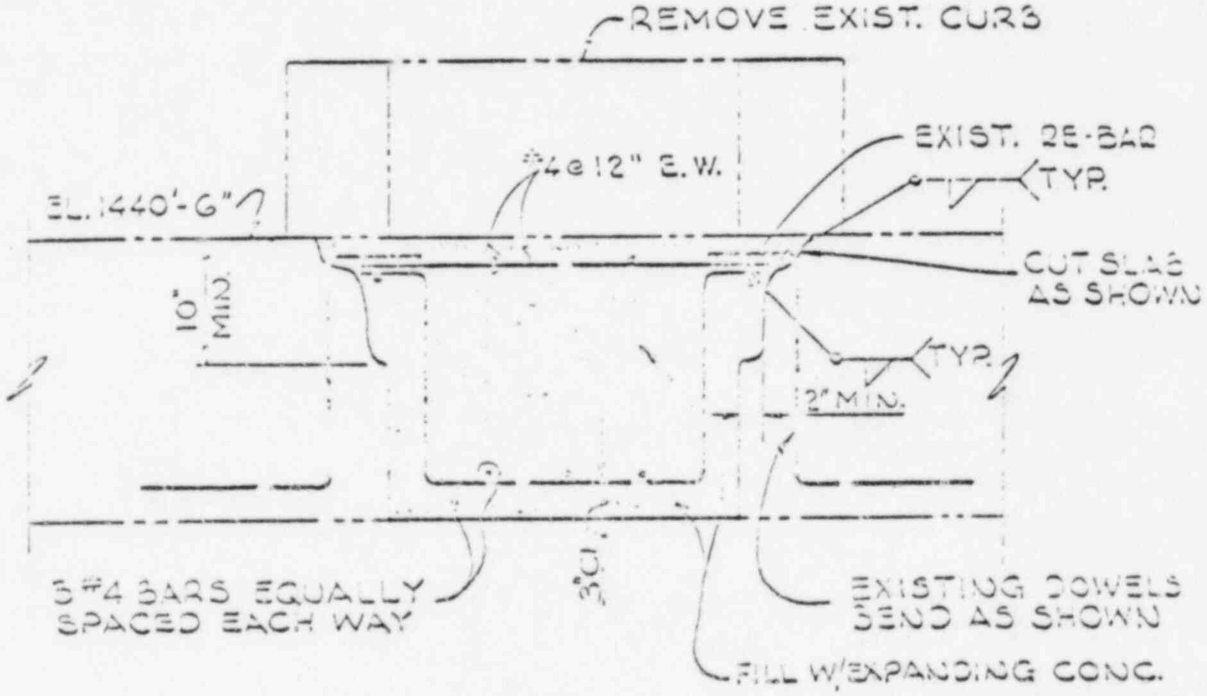


Figure 6-38. Sealing of Hatches and Pipeways to 2-ft 6-in. Square

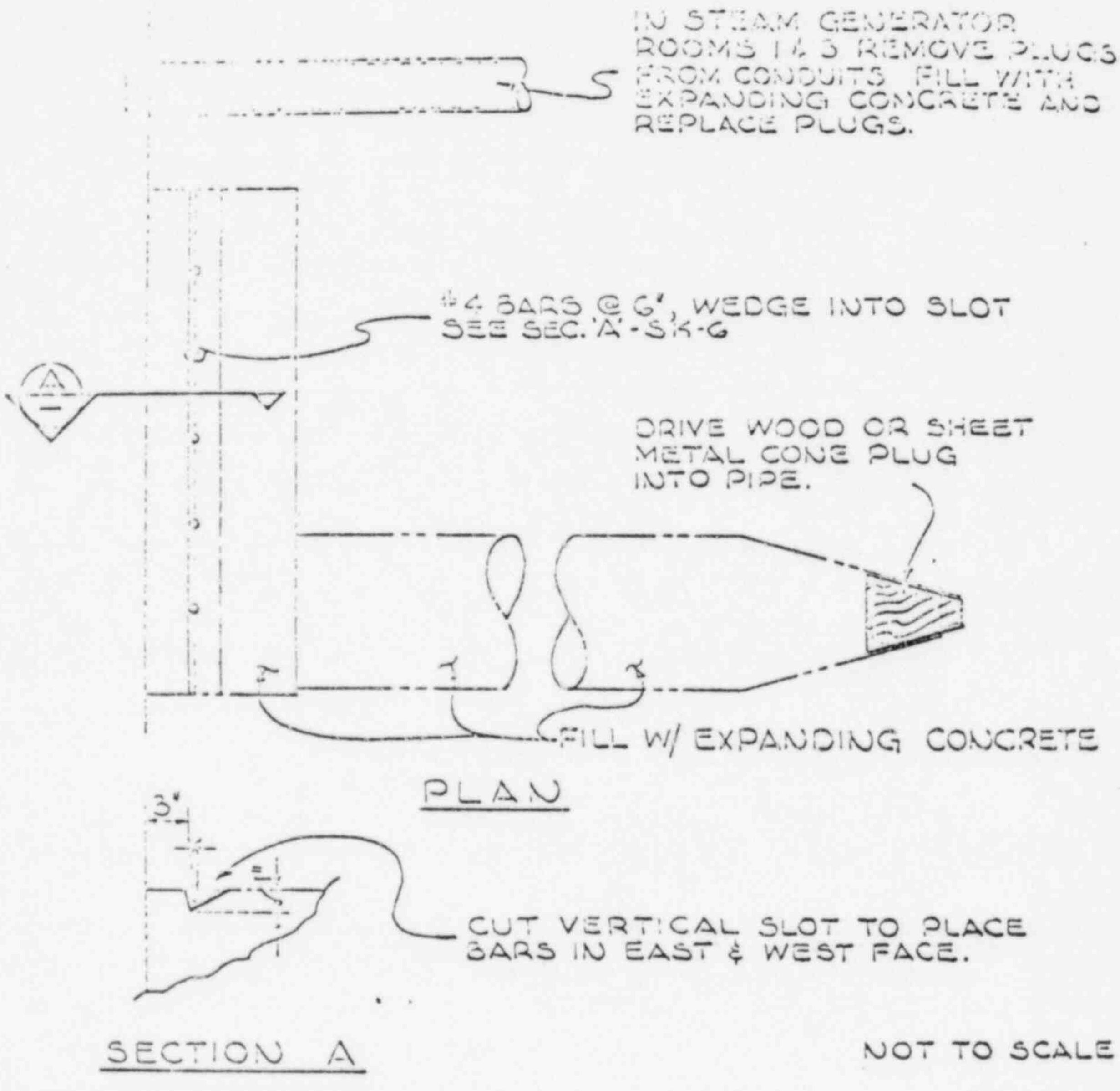
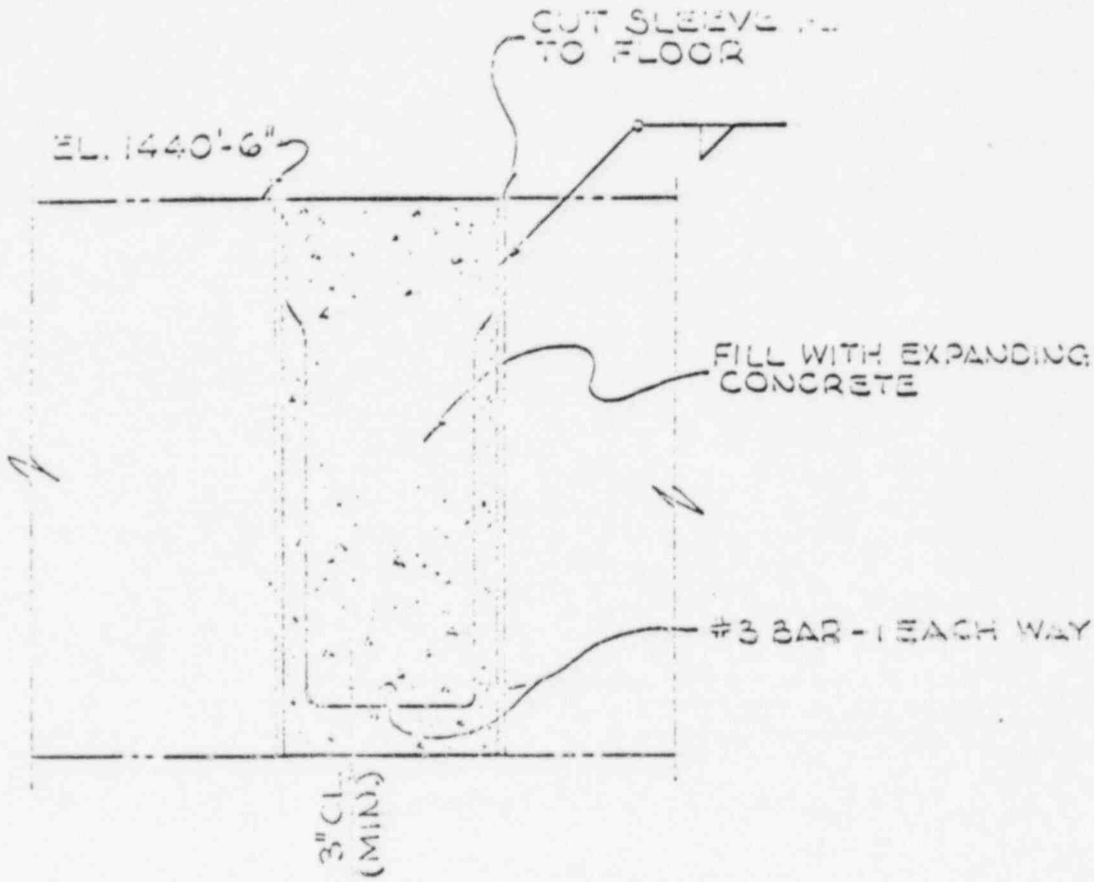


Figure 6-39. Sealing of Intermediate Heat Exchanger Cell Penetrations



NOT TO SCALE

<u>NOTE:</u> FOR 4"ϕ PIPE SLEEVES & CONDUIT
SURFACE AREA > 300 in^2
LOAD - H_2O = 12.6 x 51 psi = 640 lbs
$\therefore$ UNIT AREA SHEAR = $640/300$ = 2.14 psi
THUS FOR 4"ϕ OR SMALLER PIPES & CONDUITS THE WELDED STEEL REINFORCING BARS CAN BE ELIMINATED.

Figure 6-40. Sealing of Pipe Sleeves and Conduits Less Than 16 in. in Diameter

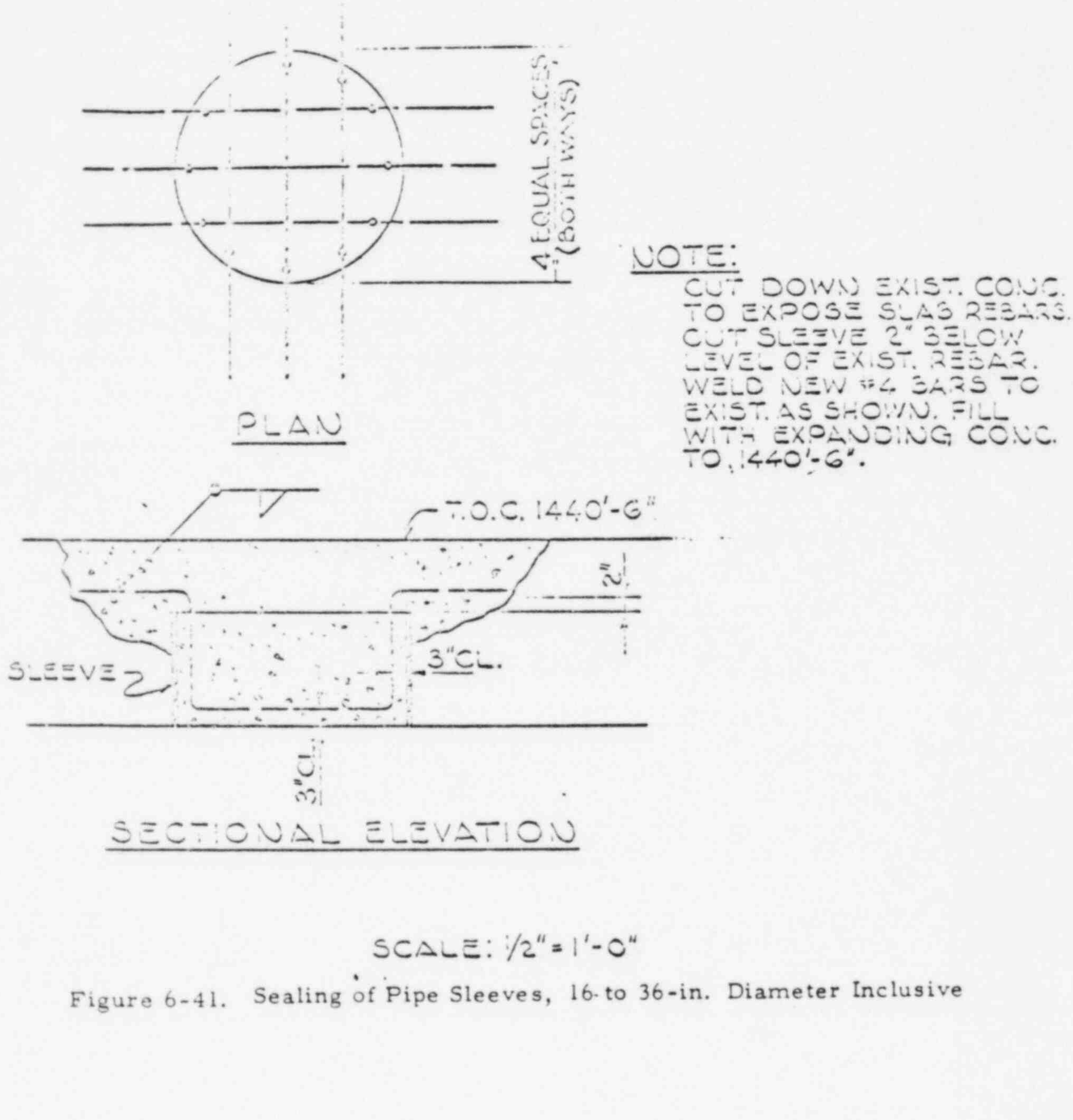


Figure 6-41. Sealing of Pipe Sleeves, 16- to 36-in. Diameter Inclusive

6.9.1.7 Primary-Sodium Pump-Case Openings

The primary-sodium pump-case openings were sealed as described in Figure 6-42. Plates fabricated from 1/2-in.-thick carbon steel were installed in the pump-case openings. The plates were welded to the penetration liners and sections of 8-in. I-beam were positioned on and welded to the plates. Mats of reinforcing steel were installed in the penetrations, which were then filled with expanding concrete.

6.9.1.8 Pipe Trenches and Instrumentation Pits

Pipe trenches in the high bay and auxiliary bay were filled with concrete to the operating floor level. All instrumentation pits were sealed by filling the pits with expanding concrete to the operating floor level.

6.9.1.9 Reactor-Building/Waste-Facility Tunnel

The tunnel between the reactor building and the radioactive-waste facility was sealed, as shown in Figure 6-43, at the point where it penetrates the west wall of the isolation structure. The piping was cut on both sides of the isolation-structure wall and removed. The bulkhead door leading from the reactor building to the tunnel was welded shut and plates were welded over all penetrations through the wall. Concrete was chipped from the wall around the tunnel entrance to provide an anchor for the closure, and mats of reinforcing steel were welded in place. Forms were placed around the penetration and filled with concrete.

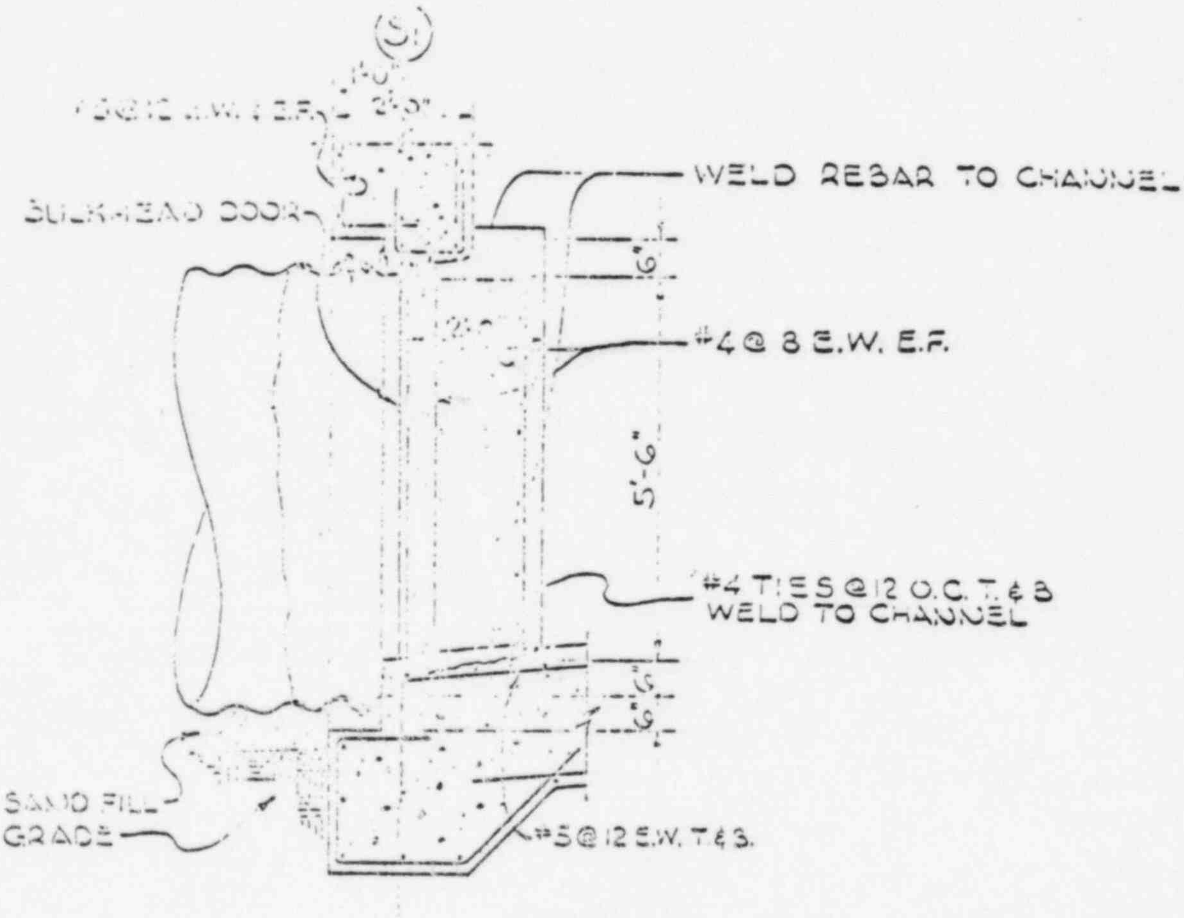
6.9.1.10 Miscellaneous Penetrations

All conduits, valve operators, tee-wrench penetrations, piping penetrations, and structural-liner vents were sealed by filling with expanding concrete. Wherever possible, plates were welded over the lower ends of these penetrations before filling with concrete (see Figures 6-44 and -45).

6.9.1.11 Preparation of Upper Surface of the Isolation Structure

Following completion of the sealing of penetrations into the isolation structure and the removal of the reactor building, a detailed inspection was made of welds followed by extensive preparation of the upper surface of the isolation structure. The entire surface was washed with water, and all exposed metal

Figure 6-42. Penetration Seal Near Primary-Sodium Pumps

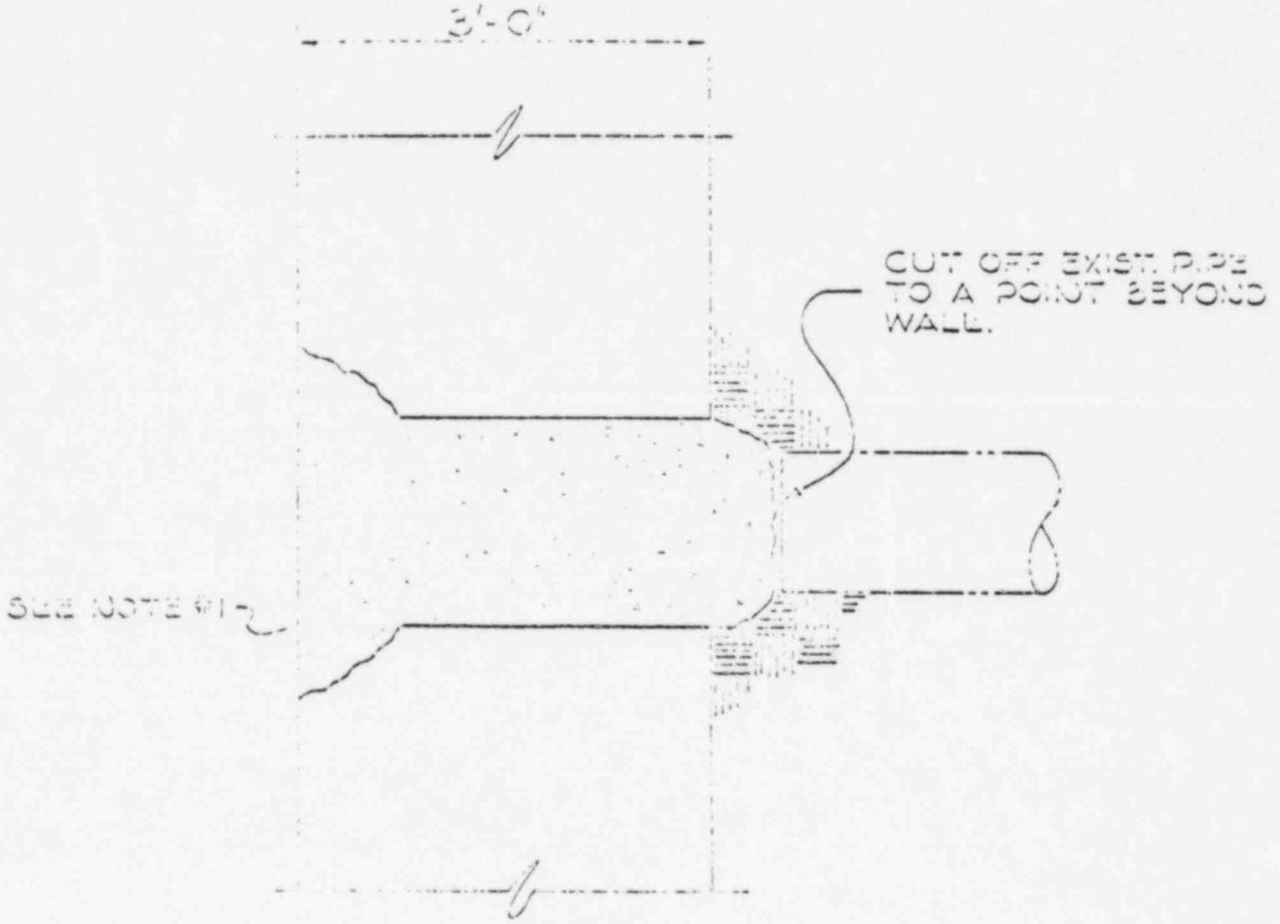


SCALE: 3/8" = 1'-0"

NOTES:

1. CUT ALL PIPES OUT OF BULKHEAD.
2. TACK WELD DOOR LOCKS.
3. WELD PLATES IN PLACE OVER ALL PENETRATIONS.
4. CHIP CONCRETE AS REQUIRED IN SKETCH.
5. PROCEED WITH FORMING & PLACING OF CONCRETE.
6. FORM STRIPPING CAN BE ACCOMPLISHED 48 HOURS AFTER CONCRETE PLACING IS COMPLETE.

Figure 6-43. Sealing of Bulkhead No. 19



SCALE: 3/4" = 1'-0"

NOTE 1

a) REMOVE PIPE AND GROUT FROM HOLE.
b) IF THIS EXPOSES EXISTING REBAR, WELD 3-#4 BARS IN PLACE EQUALLY SPACED.
c) IF THIS DOES NOT EXPOSE EXISTING REBAR - CHIP EXIST. CONCRETE BACK AS SHOWN UNTIL REBAR IS EXPOSED AND THEN WELD 3-#4 BARS IN PLACE EQUALLY SPACED.
d) PACK WITH EXPANDING CONCRETE GROUT.

Figure 6-44. Horizontal-Wall Penetration Seal

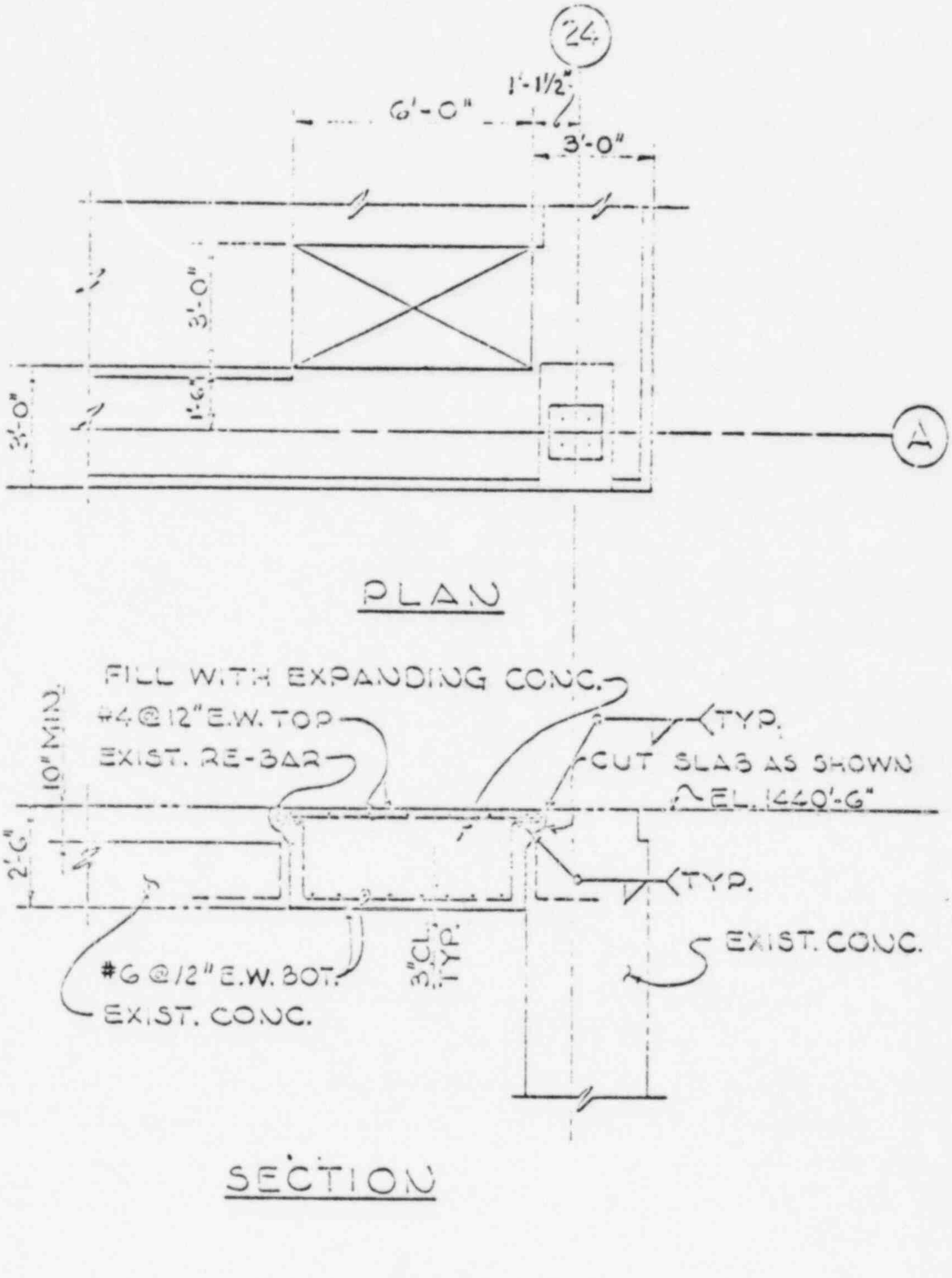


Figure 6-45. Sealing of Duct Opening

surfaces such as welds, cover-plates, hatch cover liners, etc., were sandblasted. Following sandblasting, the metal surfaces were covered with a coal-tar epoxy material known as "Tarset." Finally, a 1/8-in.-thick layer of asphalt was sprayed on to the entire upper surface of the structure (Figure 6-46).

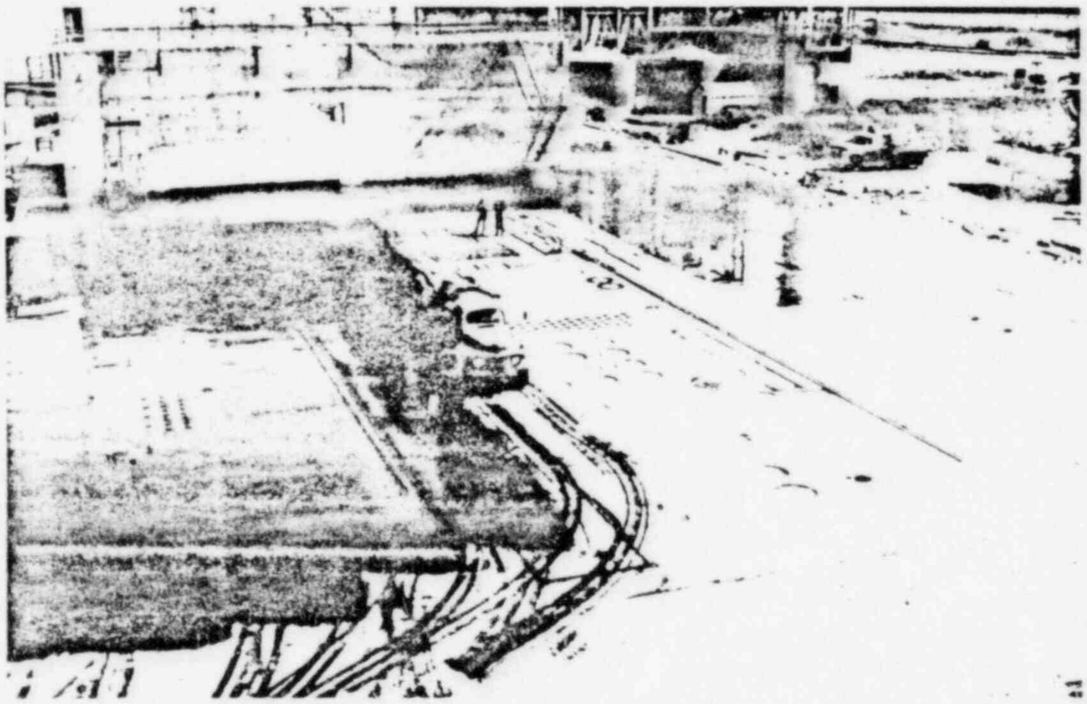


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Figure 6-46. Application of Bituminous Coating to Isolation Structure

6.9.2 Load-Bearing Capability

In compliance with the HNPF Retirement Plan and with Activity Specification No. 10, the reactor building operating level floors in the steam-generator room and over the secondary-sodium storage vaults were reinforced to increase their load-bearing capability to AASHO-H20-44 wheel loading. These floor areas are part of the top surface of the isolation structure and are the only existing floor areas which required reinforcing to comply with the H-20 rating.

The AASHO-H20-44 wheel loading refers to standards established by the American Association of State Highway Officials. Specifically the loading is that which is applied to a road by a 20-ton truck carrying 8 tons on each rear wheel and 2 tons on each front wheel. The truck front and rear axles are 14-ft apart and the wheels on the axles are 6-ft apart. For the HNPF application static loading is assumed. No impact factors need be applied because of the final covering of dirt over the structure.

Increasing the strength of the floor areas over the steam-generator room and secondary-sodium fill-tank vault area required the design and construction of special bearing walls. The plan view showing the location of these special concrete bearing walls is in Figure 6-47. The details of the wall construction under the steam-generator area are shown in Figure 6-48, those for the secondary-sodium fill-tank area are in Figures 6-49 and -50.

The access plug over the secondary-sodium fill-tank area did not have an H-20 load-carrying capability as originally constructed and it was necessary to design and construct a new plug having this strength. Following fabrication of the new plug, however, further study revealed that the floor slab surrounding the plug was overstressed. Therefore it was also necessary to pour a 22-in.-thick cover slab over this area. Details of the construction of the plug and cover slab are shown in Figure 6-51.

The concrete seals for other penetrations in the isolation structure were also designed and constructed to accommodate the H-20 floor-loading requirement.

6.9.3 Weatherproofing of the Isolation Structure

Following completion of the sealing and final inspection of the isolation structure, the surface above the structure was weatherproofed and secured by the addition of sand, earth, and a water-impermeable polyvinyl membrane. These materials were graded in such a manner that the resulting final surface sloped away from the center of the reactor toward a peripheral drainage system. Finally this surface was sodded with grasses suitable for prevention of erosion. One portion of the isolation structure, the IHX vaults, which protrudes above the graded surface was sealed by addition of a water-impermeable polyvinyl membrane capped by a 1-ft-thick layer of concrete.

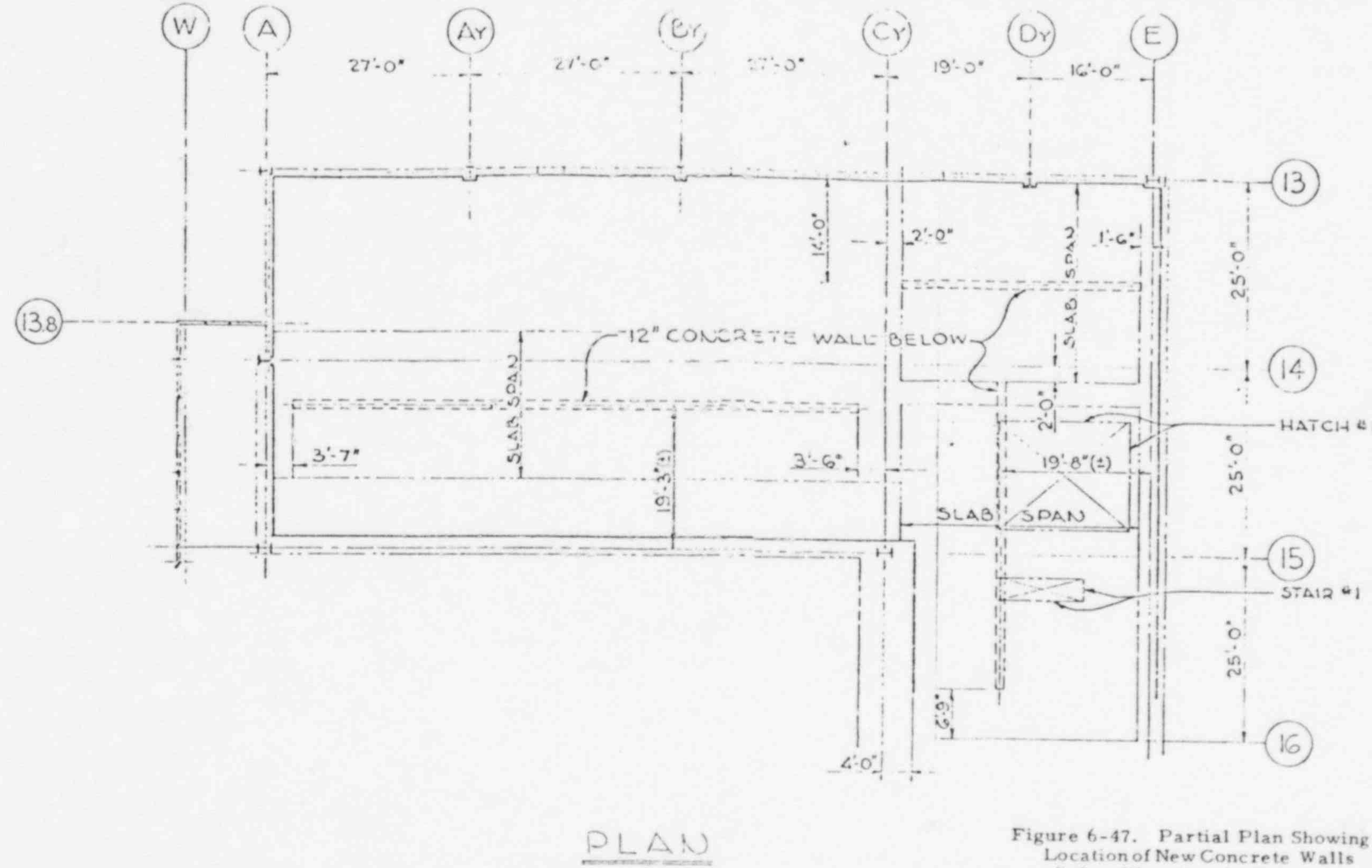


Figure 6-47. Partial Plan Showing Location of New Concrete Walls Below Elevation 1440 ft, 6 in.

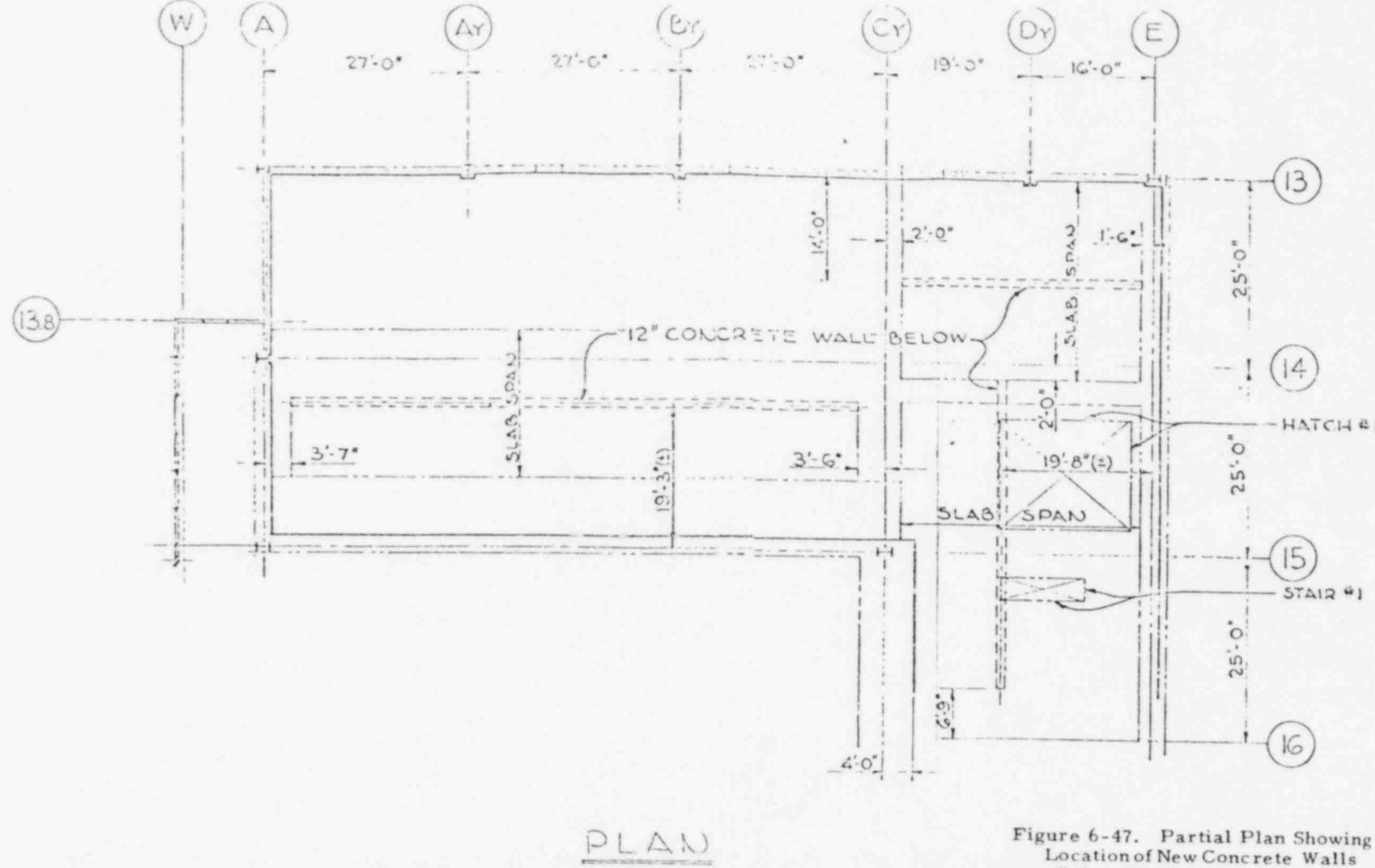
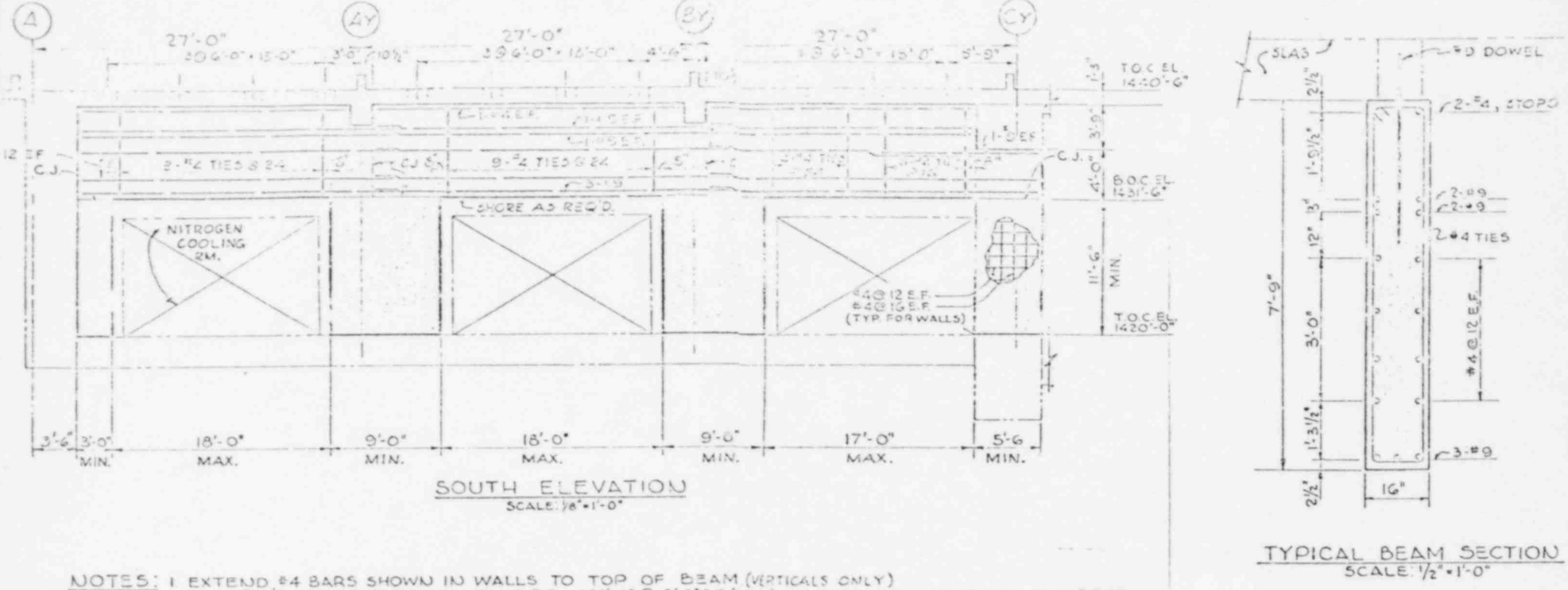


Figure 6-47. Partial Plan Showing Location of New Concrete Walls Below Elevation 1440 ft, 6 in.



NOTES:

1. EXTEND #4 BARS SHOWN IN WALLS TO TOP OF BEAM (VERTICALS ONLY)
2. CUT REQ'D HOLES THROUGH SLAB @ EL. 1440'-6" @ 6'-0" O.C. (ON ℄ OF THIS WALL TO PLACE CONCRETE, WHEN POURING BEAMS INSTALL 1 #9 x 4'-0" DOWEL IN EACH HOLE, TOP OF DOWELS TO BE 3" BELOW TOP OF FLOOR.
3. CUT 4" ϕ HOLES x 1'-0" DEEP IN SLAB @ EL. 1420'-0" @ 18" O.C. ON ℄ OF THIS WALL FOR GROUTING OF 1-#9 DOWEL EACH WHERE WALLS MEET FLOOR SLAB.
4. USE EXPANDING CONCRETE w/ A MINIMUM STRENGTH OF 3000 PSI FOR WALLS.
5. LAP ALL SPLICES 24 BAR DIA.
6. SEE SK-A-12 FOR PLAN LOCATION
7. DRY PACK ANY SPACE BETWEEN EXISTING SLAB & NEW WALL

Figure 6-48. Wall Under Steam-Generator Area

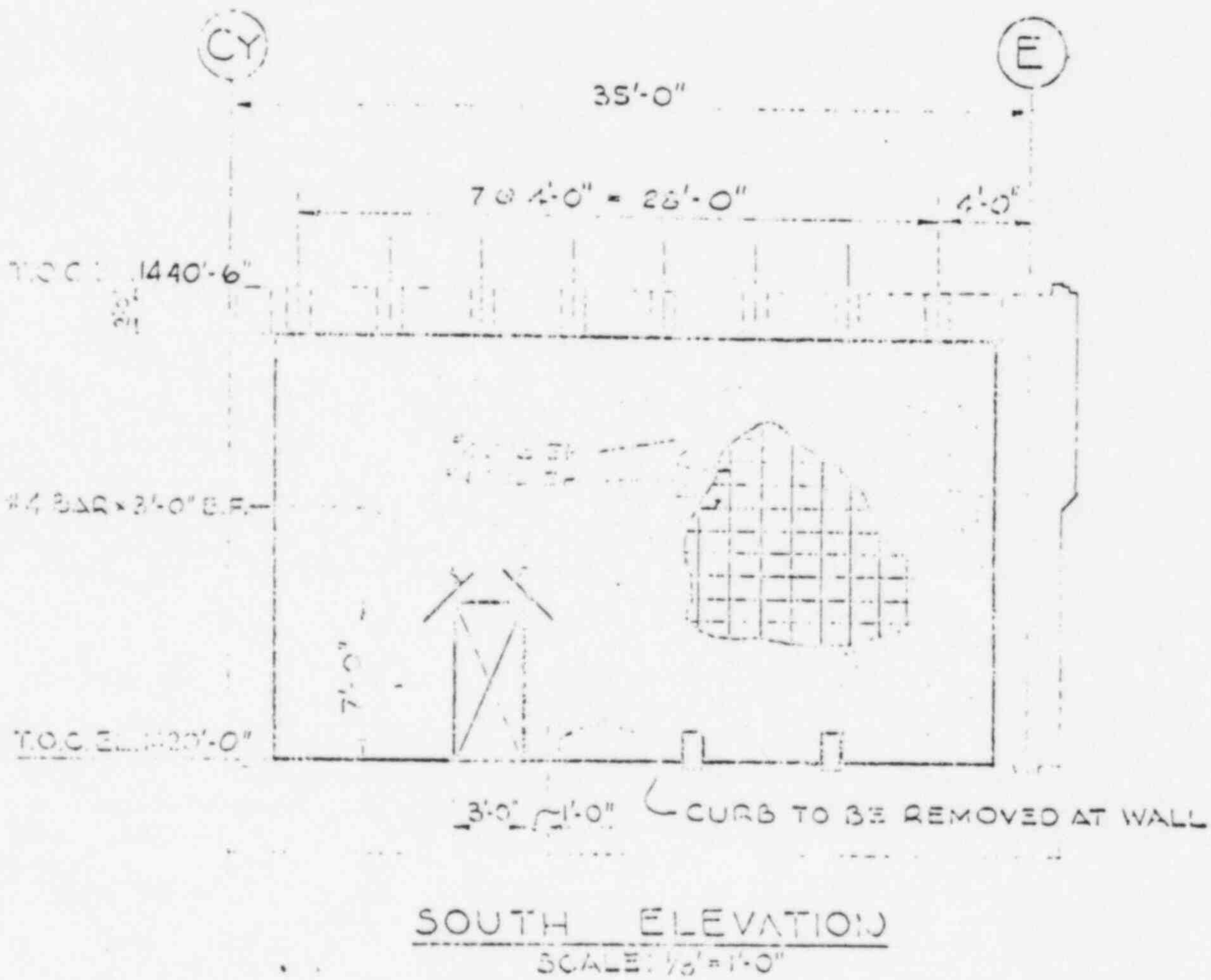


Figure 6-49. 12-in. Wall in Secondary-Sodium Tank Area

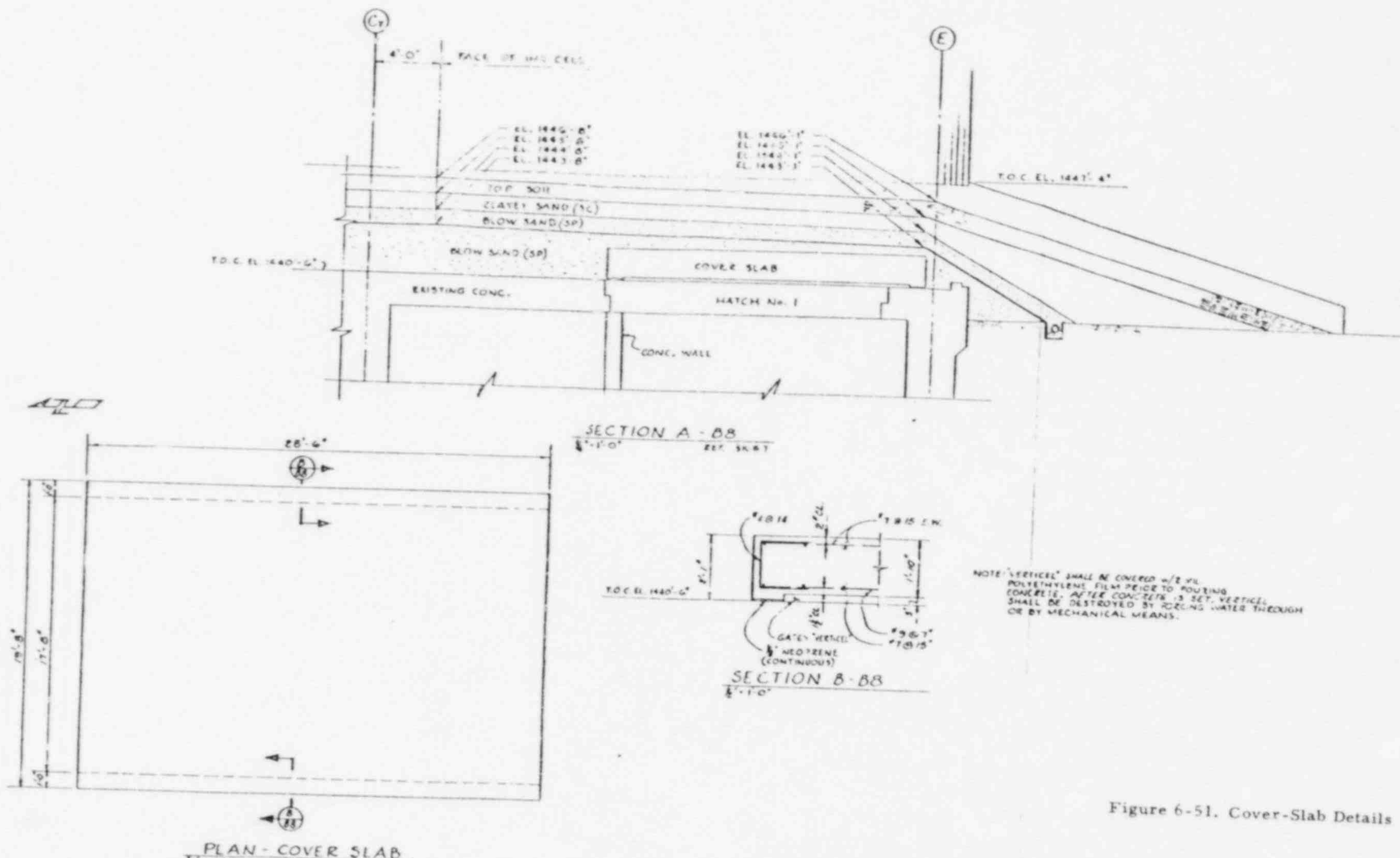


Figure 6-51. Cover-Slab Details

6.9.3.1 Slope of Graded Surface

The graded surface crowns at the center of the reactor-vessel loading face and slopes in two directions away from the reactor vessel at the rate of 1/4-in./ft. The elevation of the surface at the crown is 1446-ft 8-in. The grading plan for the structure is described in Figure 6-52.

6.9.3.2 Composition of the Graded Structure

The fill in the graded structure above the isolation structure is made up of five layers.

6.9.3.2.1 Core Layer

The first layer of fill (core layer) was placed directly on the upper surface of the isolation structure and is made up of fine-to-medium-grained poorly graded sand (blow sand) free of vegetation, stones, and foreign, or frozen objects. The sand was compacted to 70% of theoretical density and contoured to the slope and thickness described in Figure 6-53. The layer of sand is 2-ft 2-in. thick at the crown and tapers to about 6-in. thick at the edges of the isolation structure. The upper surface of this layer slopes at 1/4 in./ft and the lower surface is horizontal.

6.9.3.2.2 Impermeable Barrier

A water-impermeable barrier consisting of 40-mil-thick polyvinyl chloride sheets was placed immediately above the core layer of sand. This material is resistant to fresh and salt water, fungus and bacterial action, most mineral and organic acids, dilute alkaline chemicals, and highly corrosive salts. The sheets are joined together by means of an adhesive and activator using a 4-in. joint strip over butt-joints. Extreme care was taken to assure that no holes developed in the membrane during installation. All joints were thoroughly pressed together with rollers to assure their integrity. A spark test of all joints was made and repairs made of leaks detected. The membrane extends over the entire isolation structure and terminates at the edge of the graded area. In places where concrete foundations, retaining walls, or other objects penetrate or are present on the perimeter of the earthen structure the sheets are attached to the concrete or metal with flashings and adhesive. The impermeable barrier slopes in two directions away from the center of the reactor vessel at the rate of 1/4 in./ft. The layout of the membrane is described in Figure 6-54.

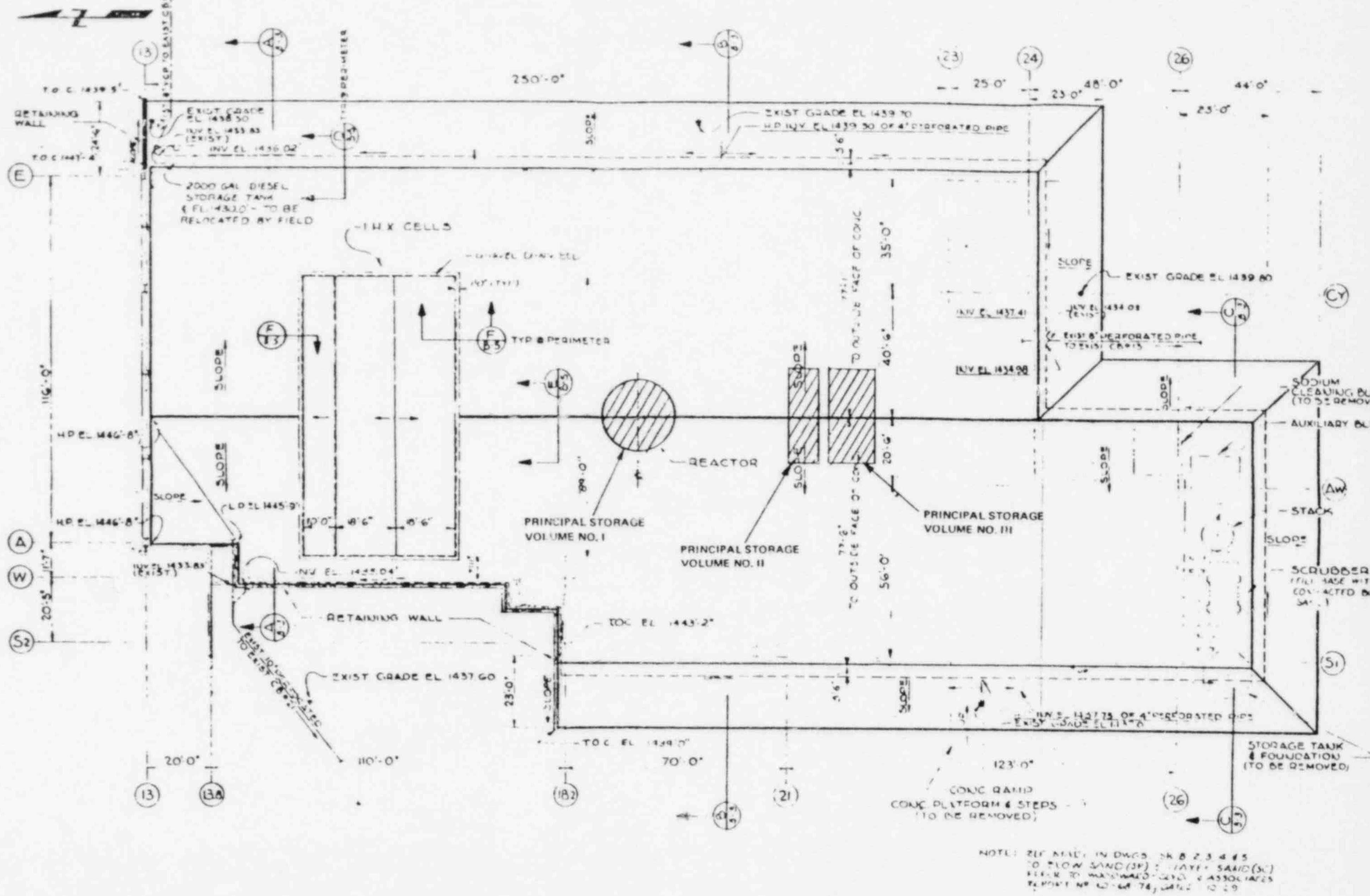


Figure 6-52. Grading Plan Over Isolation Structure

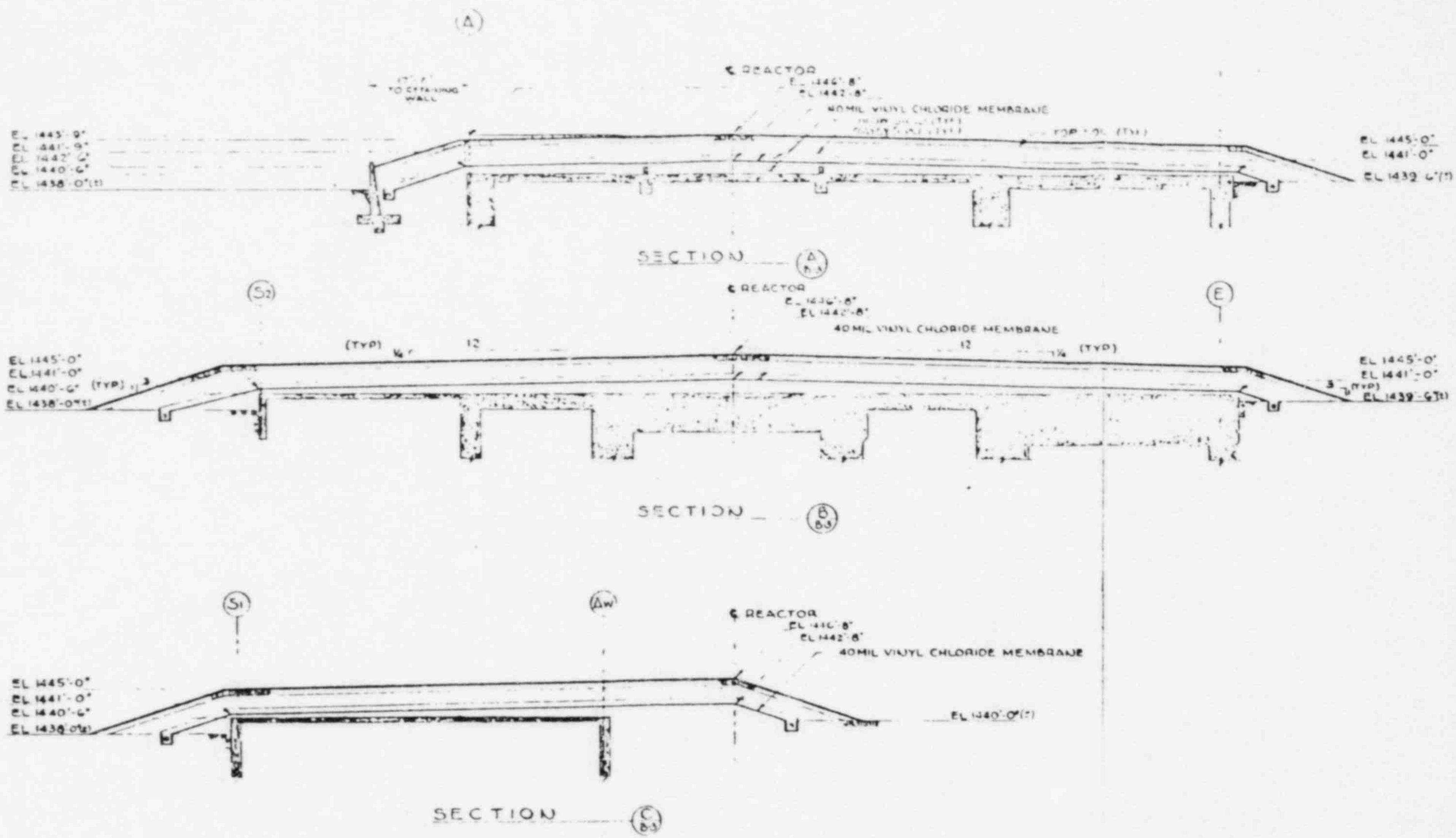


Figure 6-53. Grading Sections Over Isolation Structure

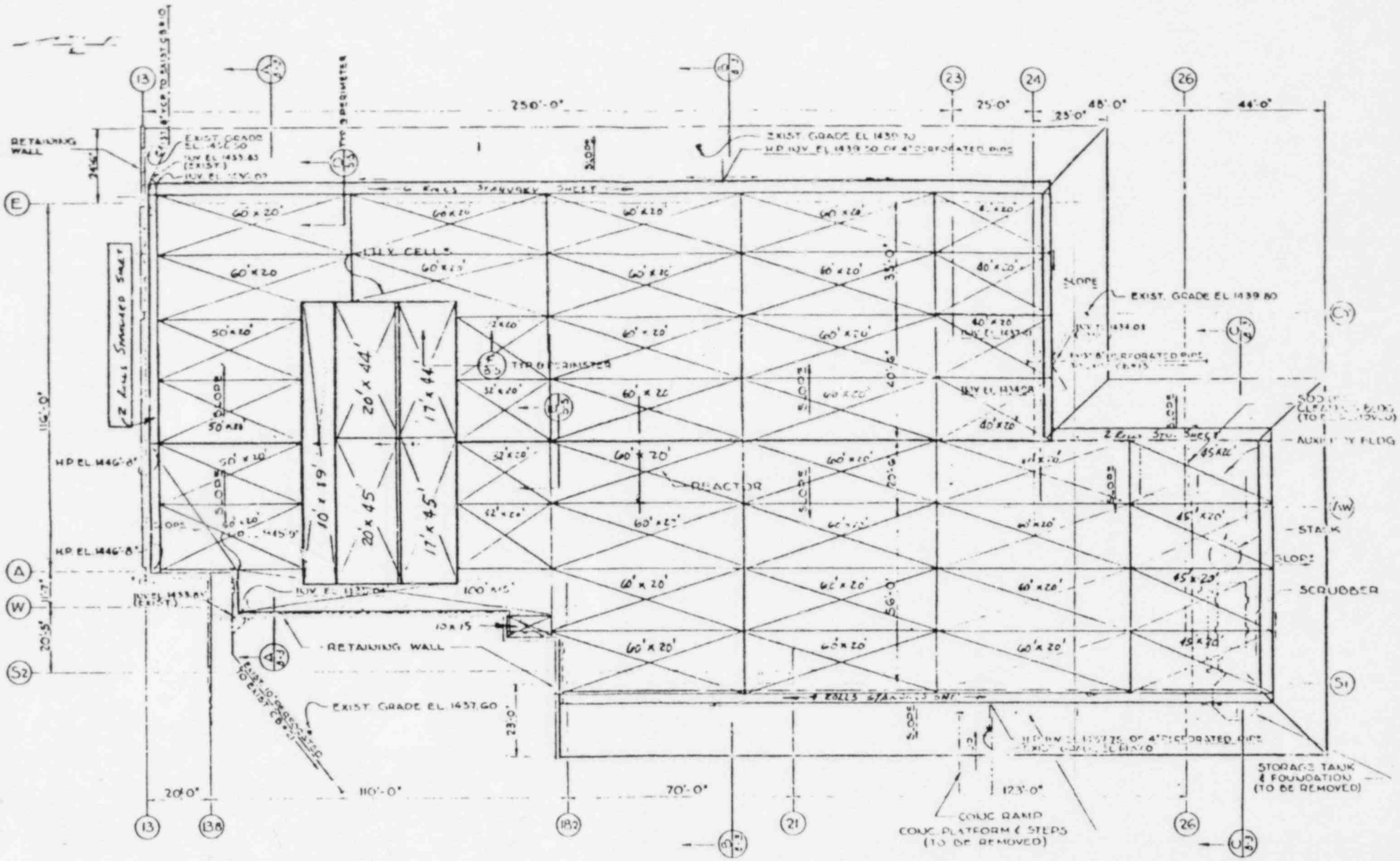


Figure 6-54. Membrane Layout Over Isolation Structure

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6.9.3.2.3 Drainage Belt

Directly above the impermeable barrier was placed a 1-ft-thick layer of fine-to-medium-grained, poorly graded sand (blow sand) free from vegetation, stones, and foreign or frozen materials, compacted to 70% of theoretical density, and contoured to the slope of the impermeable barrier. This layer of blow sand, termed the drainage belt, provides a drainage path to the system of drainage tile that surrounds the periphery of the isolation structure, for water which may collect above the impermeable barrier.

6.9.3.2.4 Drainage-Belt Cover

Immediately above the sand-drainage belt a 2-ft-thick layer of fine-grained clayey sand free from vegetation, stones, and foreign or frozen materials was placed. This layer was compacted to 90% of theoretical density and contoured to the level of the top of the sand-drainage belt. The layer provides a cover for the drainage belt.

6.9.3.2.5 Topsoil

The top surface of the graded structure is a 1-ft-thick layer of topsoil contoured to the level of the top of the sand-drainage belt cover layer and harrowed to a depth sufficient to maintain the sodding of grass. A layer of sod was placed on the topsoil and a temporary sprinkling system was installed to assure a good first-year growth. The surface of this layer slopes in two directions away from the center of the reactor vessel at the rate of 1/4 in./ft.

6.9.3.3 Perforated Tile Drainage System

A drainage system has been constructed around the periphery of the graded structure. This system consists of 4-in.-diam vitrified-clay perforated drainage tile laid in a 1- by 1-ft ditch surrounding the entire isolation structure. The tile is centered in the ditch by means of fill composed of 3/4-in. clean crushed hard limestock rock with no fines. The limestone is covered with 3 in. of Platte River sand with fines removed to prevent the overlying blow sand from filling the voids. Details of the drainage system are shown in Figure 6-55. The layer of blow sand constituting the drainage belt of the graded structure terminates on all sides at the drainage-system ditch, and the impermeable membrane extends into the bottom of the drainage-system ditch. The portion of the graded

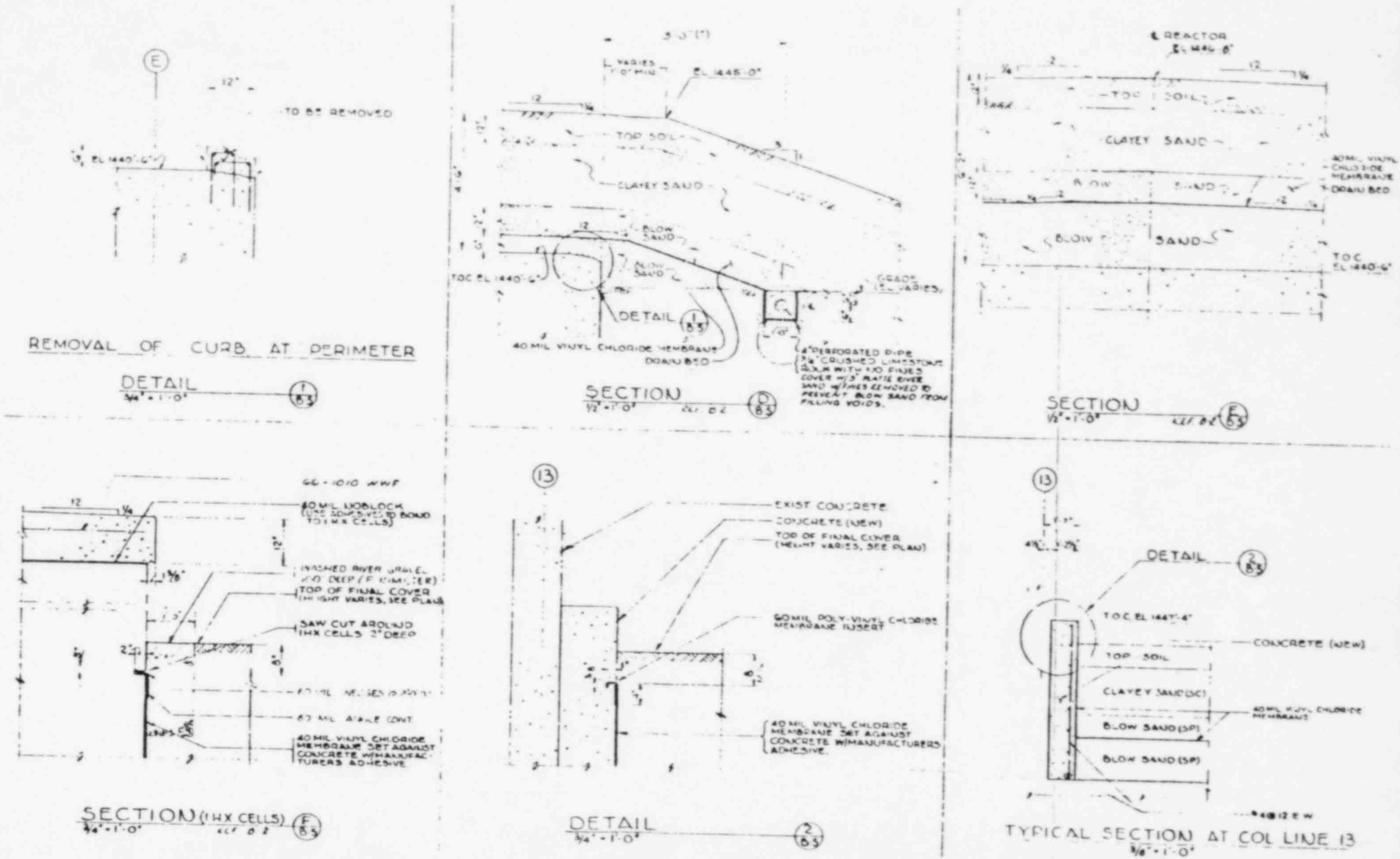

Figure 6-55. Isolation Closure Details

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structure that lies above the drainage belt extends over and covers the drainage system. The drainage system is constructed with a minimum of 1/8 in./ft fall throughout its course. The flow from the system exhausts into the Sheldon Power Station site surface-drainage system and subsequently is released to the natural surface drainage.

6.9.3.4 Retaining Walls

Throughout certain parts of the periphery of the isolation structure the slope of the graded structure is terminated by reinforced-concrete retaining walls. These areas, as indicated on Figure 6-52, are the areas along the site parking lot and along a section of the Sheldon Power Station facility wall. The retaining walls extend approximately 5 ft above the existing grade levels at these points.

6.9.3.5 Intermediate Heat Exchanger (IHX) Vaults

The IHX vaults extend 23.5 ft above the reactor operating floor level and thus extend about 17 ft above the surface of the graded structure. Weatherproofing of the IHX vaults was done by covering the top of the vaults with a water-impermeable barrier of 40-mil polyvinyl chloride "Nob Lock" sheets and capping the structure with a 1-ft-thick layer of concrete. Since the original structure of the IHX vaults included concrete wall and ceiling thicknesses of 7.5 ft, the addition of the impermeable barrier and additional concrete results in a minimum of 7.5 ft of concrete separating the inside of the IHX vaults from the environs on the sides of the vault structure and 8.5 ft of concrete separating the vaults from the environs on the top.

6.10 RETIREMENT OF THE RADIOACTIVE-WASTE FACILITY

Provision was made for the decontamination and disposition of components and materials associated with this facility (see Figure 6-56) which consisted of a 40- by 52-ft concrete and steel structure with a massively shielded (concrete) basement, and a 5-ft-diam tunnel connection from its basement to the reactor building basement approximately 170 ft away. The basement contained four radioactive-gas decay tanks, three liquid-waste storage tanks, and two pumps as major items of equipment. Releases of stored gases to the atmosphere were performed in accordance with established operating procedures. Liquid radioactive

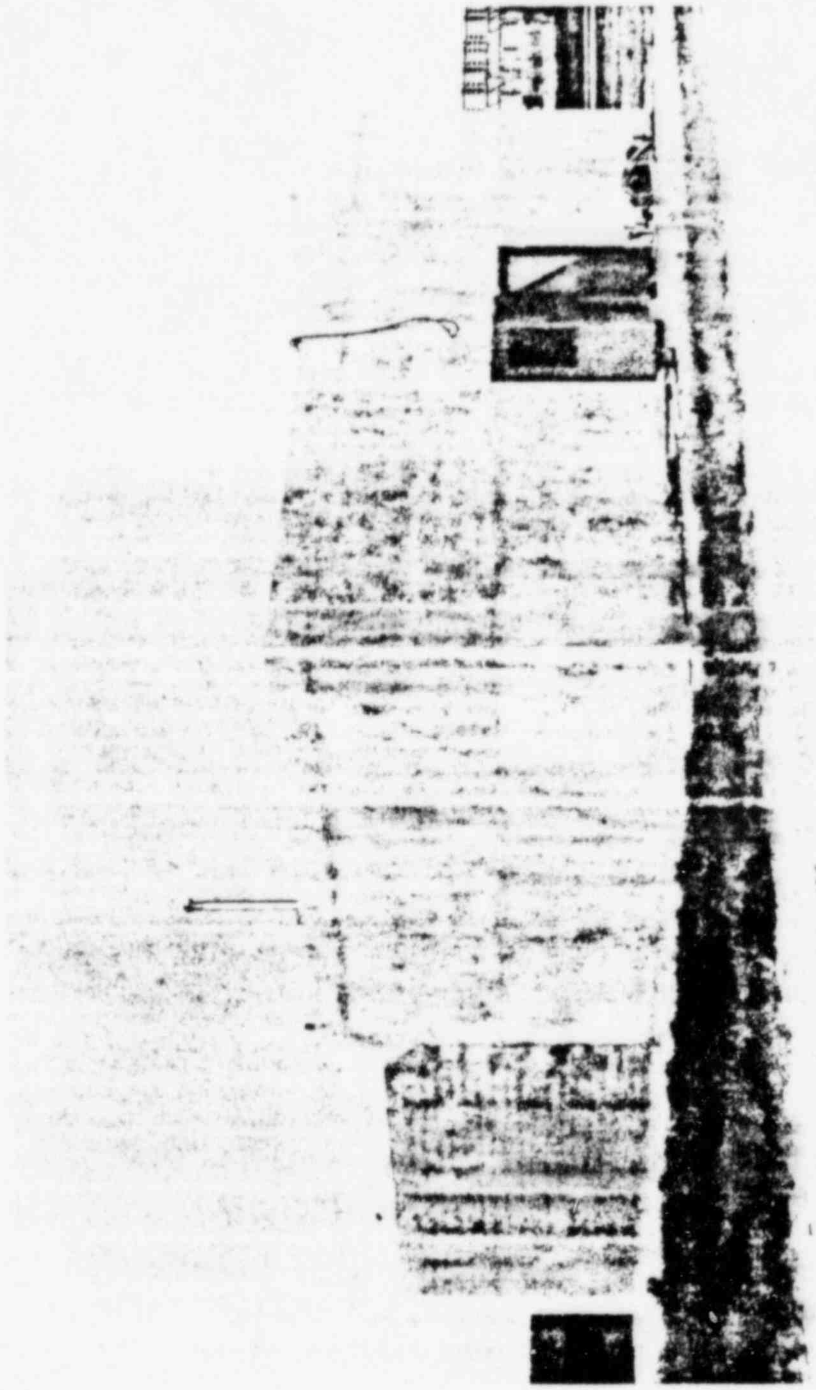

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Figure 6-56. Radioactive-Waste Disposal Facility waste, after suitable decay, was diluted and transferred by established techniques to the leach ponds; an additional water supply was required to facilitate this procedure. Openings into the radioactive-storage space were the tunnel which contained gas-vent and liquid-waste drain lines, and several shielded plugs and hatches in the floor.

Several categories of equipment and material were designated on the basis of value, possible future use, and ease of decontamination. As a result some items (pumps and compressors) were cleaned and shipped off-site to other users, some material (mostly pipe) was disposed of as scrap, and other items such as the high level liquid waste tank were left in place. The carbon steel liquid-waste tanks and pipes were cut up and placed in the primary-sodium vault because of a residual radioactivity level higher than that permissible for salvage scrap.

After an inventory and a radiological survey to assure an acceptably low level of activity and no storage of radioactive materials, the basement space was isolated. This was done by welding the floor plugs and hatches closed, grouting all cracks and openings, disconnecting the tunnel at both ends, and plugging (with concrete) both the building foundation walls and the tunnel ends, thereby isolating the tunnel approximately 10 ft below grade. In it are pipe lines having permissible levels of radioactivity. Lines with a higher activity were placed in the primary fill-tank vault for permanent disposal.

The above-grade structure, now clean and empty, has been signed over to CPPD; it is suitable for further use, probably as storage space.

6.11 FINAL CLOSE-OUT OF FACILITY

This activity included the disposition of buildings, a general cleanup of the area, and a final radiation survey. The buildings removed from the site were the reactor building, moderator-assembly building, calibration building, storage building, and the steam-cleaning facility buildings. The steam-dump facility and the radioactive-waste handling facility are the only remaining buildings. They are now being used by CPPD for storage.

Removal of buildings was a Harrison Metal and Iron Company responsibility under prime contract to the AEC. In addition to removing the buildings and the remaining contents they cleared the area of all debris and salvageable scrap materials.

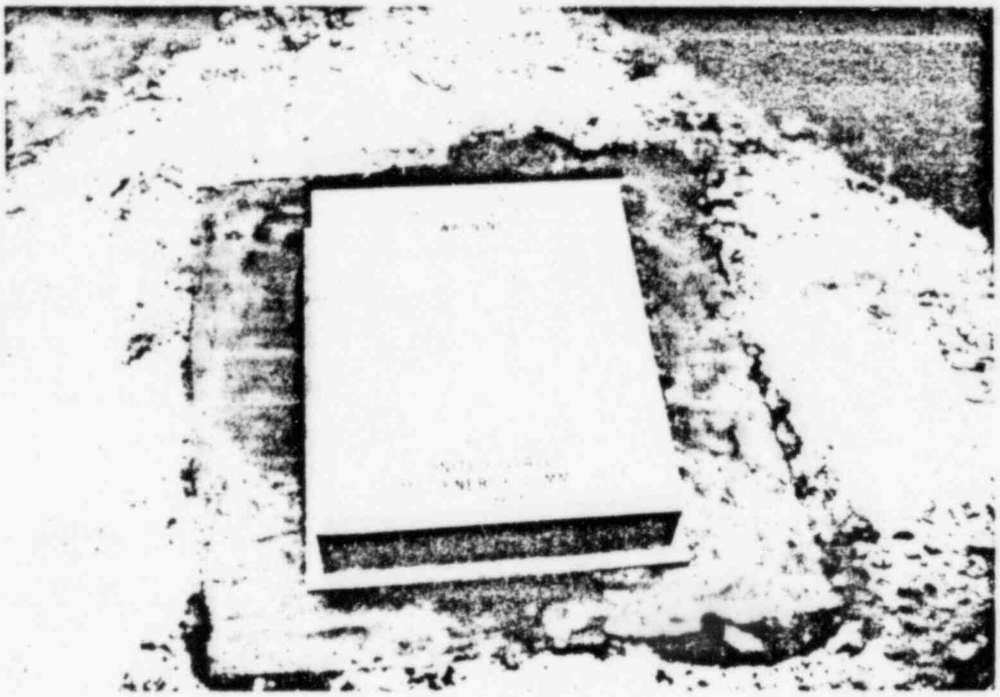

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Figure 6-57. Time Capsule on Reactor Core Cover-Plate

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Figure 6-58. Time Capsule in Wall of Intermediate Heat Exchanger Facility

Harrison Metal and Iron Co. began their work in November 1968, after the isolation-structure penetrations-sealing was completed thereby preventing access by Harrison personnel to areas containing radioactive materials.

A final site-radiation survey was taken. A radiation level slightly above acceptable limit was found in the inlet of the retention-pond sediment. The radioactive sediment was removed and shipped for off-site burial. Another survey of the pond area was then taken and found to be well below acceptable limits for unrestricted areas. The radiation survey of the total plant area showed levels below the acceptable limits.

Sealed in two stainless-steel boxes were descriptive material of the isolation structure construction and sealing, and the Final Status Report and Safety Analysis of the HNPF and remaining structures, which lists the location, component nomenclature, and radionuclides stored in the structure. To each box a plaque was affixed which briefly stated the potential danger and warned against entry without contacting the AEC. One box was located directly over the centerline of the reactor vessel and welded to the cover-plate, and is now 6 ft below the grass surface. The other box and plaque are mounted in the concrete south wall of the IHX structure about 5 ft above the grass sod. Figure 6-57 shows the buried box and 6-58 the location of the above-grade box.

7.0 HEALTH AND SAFETY

Since CPPD had the basic responsibility for implementing the retirement program and used their operating crews to a large extent, the burden of safety operations in the physical activities of retirement fell totally to them. Operating crews and supervisors, as a result of their reactor operations training and experience, were already well indoctrinated in hazards of radiation and of chemical reactions of metallic sodium; therefore, only a familiarization with the Retirement Plan and procedures was required for CPPD personnel. A large amount of dismantling and heavy-equipment removal was subcontracted by CPPD to Stearns Roger, a general contractor who essentially hired local labor for the job. The contractor was specifically directed by CPPD in following CPPD safety requirements. To cope with the general problem of health and safety CPPD management utilized the established Safety Review Committee to review plans and procedures. The CPPD health and safety group monitored all activities, provided control of personnel access to radiation areas, and supplied film badges and dosimeters to all personnel as necessary. The dose to about 90% of the personnel was less than 10 mrem per month. Occasionally a man did receive 100 mrem per month. The greatest radiation hazard resulted from removal, disassembly, and shipment of irradiated fuel. Because existing CPPD health-safety rules and established fuel-handling procedures were followed, no accident or overlimit exposure of personnel occurred as a result of the HNPF retirement work.

Before the start of the HNPF dismantling work AI prepared Supplement 5 to the HNPF Safeguards Analysis.[4] This document analyzed the potentially hazardous activities as delineated in the HNPF Retirement Plan and indicated the safety precautions taken to accomplish these activities safely. AEC-DRL reviewed the document and after clarification of some of the procedures issued a Dismantling Order (Appendix VI-3).

A "Final Status Report and Safety Analysis of the HNPF Site and Remaining Structures"[5] was prepared by AI. Besides describing the structures and their contents, the report concludes that the radioactivity in the isolation structure has been safely contained indefinitely; in conjunction with the final environmental survey it forms the basis for AEC-DRL cancellation of the operating authorization for HNPF.

8.0 CASK FOR IRRADIATED-FUEL SHIPMENT

The importance of the fuel shipping casks in the retirement program necessitated that a special section of this report cover the cask design, procurement, operation, and disposition. A complete treatment of the cask design and procurement is reported in Reference 13.

The HNPF retirement program required that the 150 irradiated fuel elements in the fuel-storage cells (140 U - 10% Mo and 10 UC) be shipped to the Savannah River Plant (SRP) at Aiken, South Carolina. Off-site shipment of the radioactive fuel was the first major task in the dismantlement of the plant.

Original operating equipment at HNPF included a one-element shipping cask designed for fuel-reprocessing requirements, but equipment suitable for shipping this volume of irradiated elements in a reasonable time was nonexistent. A study of the retirement-program costs and schedule showed that two six-element casks would be needed to meet the reqiurements. The AEC-contracted with AI on August 31, 1966 to design the shipping casks.

The design conditions and criteria for the shipping casks were based on HNPF and SRP requirements and on those in AEC Regulation 10CFR71, and ICC Regulations Title 49, Code of Federal Regulations, Part 73.

HNPF Requirements

The HNPF requirements for the cask were as follows:

1) The cask entered the building mounted on a railroad car on a track that was serviced by the overhead crane.
2) The overhead crane had a 60-ton capacity on one hook and a 15-ton capacity on an auxiliary hook. The hook height was 59 ft.
3) The cask mated vertically with the moderator storage cell containing baskets of fuel.

SRP Requirements

The SRP requirements were as follows:

The canned fuel assemblies had to comply with the following requirements of the Savannah River Plant:

1) Each fuel assembly was to be canned within two stainless-steel containers providing double enclosure. The containers were to be sealed by welded construction in such a way as to assure water tightness.

2) The canned fuel assemblies were to be shipped intact and in casks meeting the handling requirements of the Savannah River Plant.

3) The sodium in the fuel rods was to be maintained in a solid state during fuel shipment, i.e. at a temperature less than 208°F.

The SRP receiving facility is designated the receiving basin for off-site fuels (RBOF). It is equipped with water-filled basins for underwater unloading, transfer, and storage of fuel. The basin area contains a decontamination pit, two cask-unloading basins, a storage basin, and interconnecting canals for movement of elements between basins.

Approximately 10 ft of water was to be maintained over the elements when removed from the cask vertically in order to restrict the radiation at the water surface to less than 0.1 mr/hr. The two water-filled unloading basins have their floor level at a water depth of 28 ft; in addition each contains a pit 9-1/2 ft in diameter and a water depth of 44 ft.

The RBOF facility also has a railroad spur into the building. A bridge crane of 100-ton capacity operates over the cask vehicle unloading position, the decontamination pit, and the two unloading basins. The crane has two 50-ton hoists arranged for separate travel on the bridge. The bridge travel is 15 ft which posed a problem as to how best to upright the cask as the cask length exceeded 15 ft. This was solved by allowing the car to free-wheel during a portion of the cask translation from horizontal to vertical and vice-versa.

8.1 DESIGN AND ANALYTICAL EFFORT

The plan for shipping the spent-fuel elements and the design of the shipping cask and other accessories was engineered by AI. AI also worked with the Chicago, Rock Island, and Pacific Railroad (CRI & PRR) to provide suitable railroad flat cars for the fuel shipment.

The shipping operations were begun by canning the irradiated-fuel assemblies according to SRP requirements in the maintenance cell (as described in previous sections), and transferring them in a fuel-shipping basket to a moderator-storage cell, using the moderator cask for shielding during the transfer.

The fuel-shipping basket was removed from the storage cell by means of one of the two shipping casks. The loaded cask was transferred to the railroad car which was used for transportation to the SRP site. The route of the shipment was selected as the result of an economic study of the AEC Chicago Traffic Department. The unloading of the fuel assemblies took place in a water basin at the SRP.

Two complete fuel-shipping cask assemblies were manufactured. They were supplemented by 13 fuel-shipping baskets to facilitate the preparation of fuel for shipment and the fuel-transfer operations. The selected quantities of casks and baskets assured economical planning and execution of the shipment. The two railroad cars used for transportation were each equipped with a support cradle and a sunshade. Two cask-handling yokes were furnished, one each specially designed for HNPF and SRP.

The fuel-shipping cask assembly had to be compatible with the existing facilities which required a cask for bottom loading at the HNPF and top unloading at SRP under water. The cell arrangement below floor level at HNPF also required the cask to have the ability to automatically grapple the fuel basket in the storage cell and raise the basket into the cask. This handling feature of the cask was achieved by provisions similar to those of the fuel-transfer cask. The fuel-shipping basket was designed to be compatible with the handling features of both the fuel-shipping and fuel-transfer casks.

The inner shell of the fuel-shipping cask is made of ASTM A240 Type 304L stainless steel for corrosion resistance and for ease of decontamination. The outer shell was fabricated of carbon-manganese steel, ASTM A516, Grade 55.

The size of the fuel-transfer cask and the 3-ton lifting capacity of the maintenance-cell in-cell crane set the fuel load at 6 assemblies and fixed the basket diameter at 18 in.

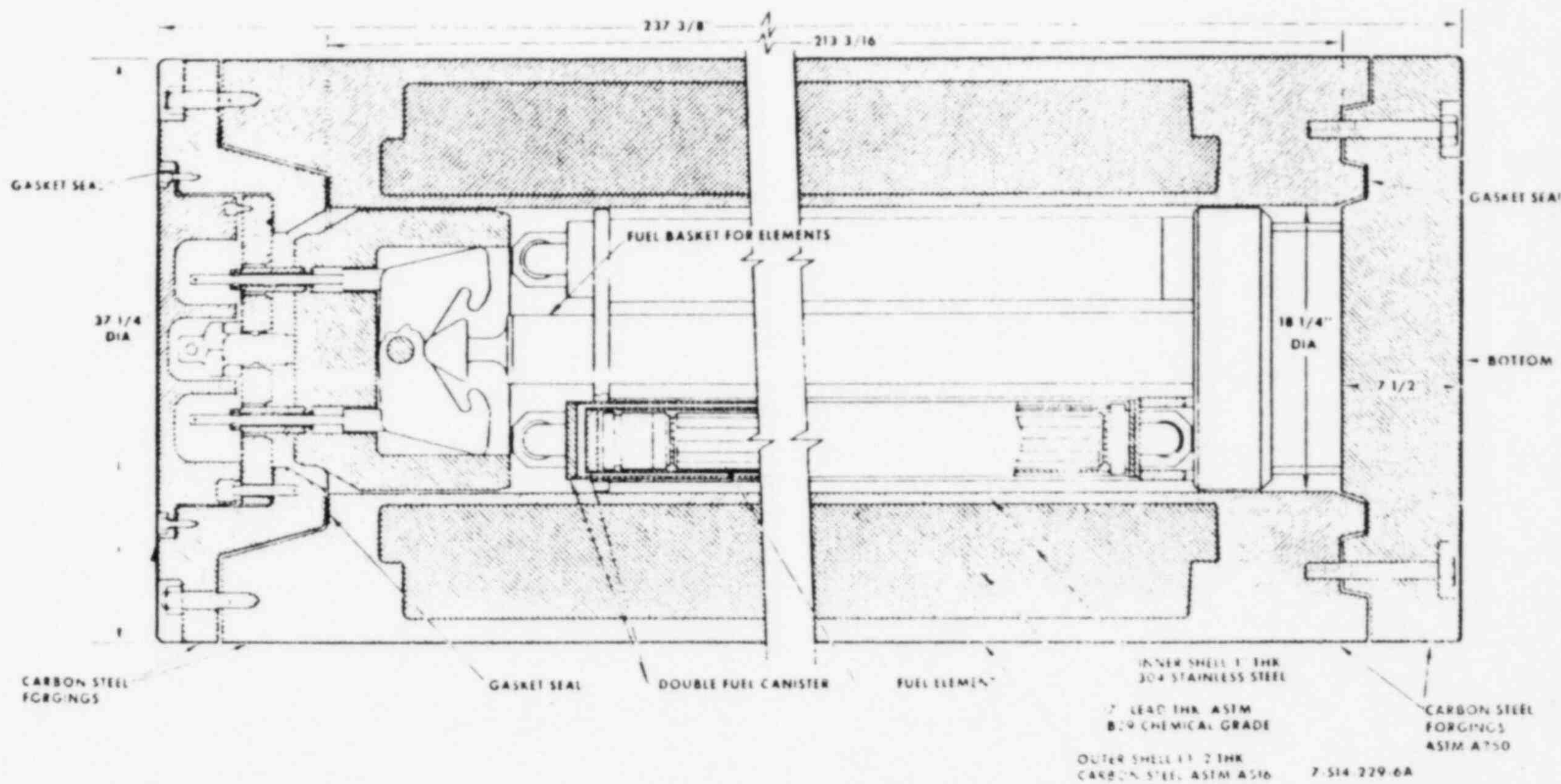

Figure 8-1. Six-Element Irradiated-Fuel Shipping-Cask Assembly

The size of the basket set the ID of the cask at 18-1/4 in. Analyses to meet the requirements of USAEC Manual Chapter 0529 resulted in a selection of a 1-1/2-in. carbon steel outer shell and a 1-in. stainless-steel inner shell; this, with the 7 in. of lead required to achieve the 8-1/2-in. total lead equivalent, set the OD of the cask at 37-1/4 in. The general arrangement of the cask in the shipping configuration is shown in Figure 8-1.

Two 54-ft flat cars of the CRI & PRR were modified by them in their Kansas City shops to accommodate the fuel-shipping operations. The modifications consisted of: installing hydrocushion couplings and new car wheels; reinforcing the deck and furnishing new wood decking; and applying steel plates to facilitate attachment of the support cradle by welding.

Due to the long decay time of the fuel (18 mo as of May 1966) the cask axial-shield requirements were determined primarily by the Co^{60} activity (induced activity of the Cobalt) in the stainless steel in the fuel element. The radial-shield requirements were determined by the Ce^{144} fission-product acitvity in the spent fuel. The actual radial-shielding requirement for shipping six peak HNPF fuel elements as of May 1966 was 7-1/2 in. of equivalent lead thickness. Based on a postulated accident, a minimum lead equivalent of 3.7 in. must be maintained for the radial shield.

The total decay heat due to the beta and gamma decay of all fission products for an average fuel element was calculated. The first core loading of U - 10 wt % Mo fuel elements had undergone a total burnup of 0.29 full-power years. The decay heat of the elements at various decay times in watts per average element is: May 1966, 54; May 1967, 30; December 1967, 24; and May 1968, 20. The decay power for a peak element is 1.45 (peak-to-average ratio) times these values.

To satisfy the thermal conditions of SRP relating to safe reprocessing, a fuel element must not exceed 208°F, the melting temperature of sodium. The maximum summer air temperature along the shipping route between the reactor and reprocessing facilities (Nebraska to South Carolina) was found by a check of the weather records to be 110°F. The treated-surface temperature was calculated to be 157°F. The resulting fuel temperatures are: for 30 watts power per fuel element, 208°F; for 45 watts, 234°F.

It was therefore necessary that a sunshade be used to shield the cask from solar exposure. With a sunshield, shown in Figure 8-2, the temperature drop from the cask surface to the air is 11°F for 45 watts and 7°F for 30 watts. Thus the fuel temperatures for 110°F ambient air are 198°F for 45 watts and 168°F for 30 watts.

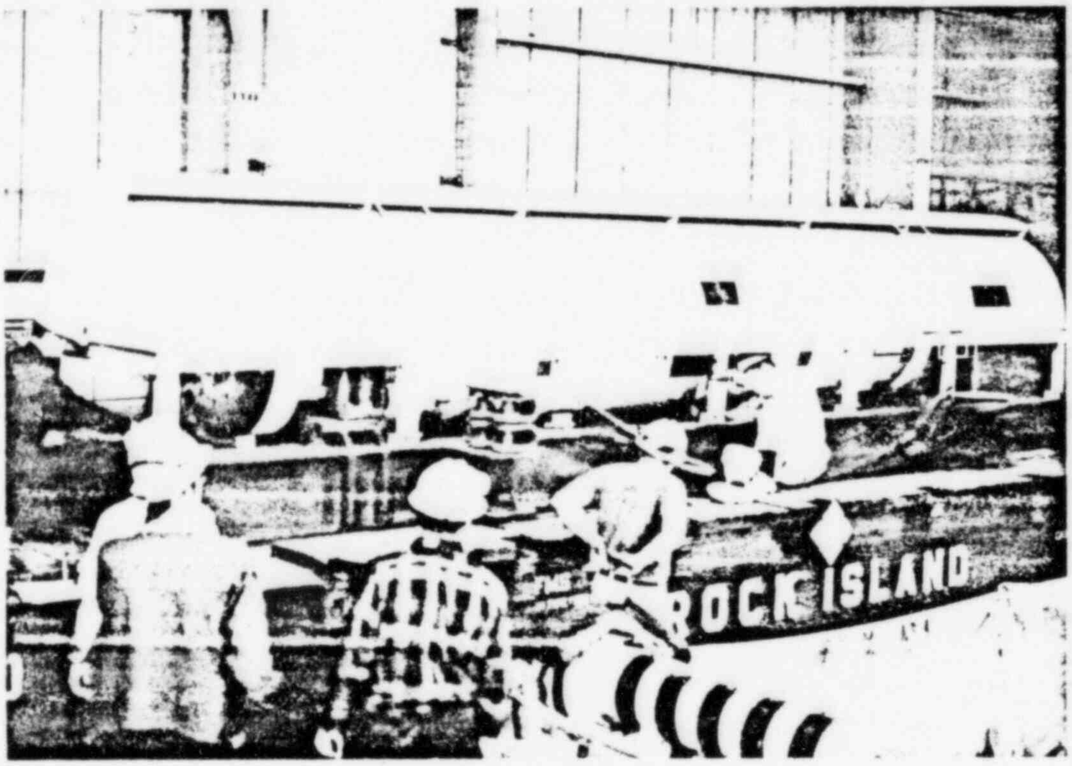

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Figure 8-2. Shipping Cask with Sunshield on Railroad Flatcar

For the purpose of the fuel shipping cask heat dissipation analysis, fuel elements were assumed to be unirradiated since reactivity decreases with fuel exposure. Water was assumed to be around and within the cask and the gas-tight fuel canisters as required. K_{eff} was calculated for four values of uniform water density within the cask and canisters, while maintaining full density exterior to the cask. A large range of water densities was selected in order to bracket the optimum fuel-to-moderator ratio which results in maximum reactivity.

TABLE 8-1

HNPF FUEL-ELEMENT MASS

Fuel	Enrichment (% U^{235})	Fuel Element Mass (kgm)	U^{235} per Element (kgm)
U-10 Mo	3.6	220	7.13
UC	3.7	165	4.81
UC	4.9	165	7.70

The requirement pertaining to the subcriticality of two identical casks in contact with one another is satisfied provided the lead thickness is greater than 5 in. It has been shown experimentally that 10 in. of lead are adequate for neutronic isolation.

8.2 MANUFACTURING

The HNPF casks were procured by the AEC Chicago Operations Office. After competitive proposals to fabricate two casks were evaluated, Allied Engineering and Production Corp. (AEPC) of Alameda, California was awarded the contract on March 27, 1967. Atomics International, the cask designer, was retained to provide design consultation, surveillance, and quality assurance services during fabrication at the plants of AEPC and its subcontractors.

8.3 UTILIZATION

The two casks were shipped from Allied Engineering & Production Corp. in Alameda, California on Wednesday, January 24, 1968 on their specially equipped railroad flat cars. They arrived at Hallam, Nebraska on Sunday, January 28.

At the HNPF the two casks were inspected visually and put through a pre-operational checkout. The hoisting equipment was assembled to each cask in turn and the casks were checked out by going through the following cycle: removal of cask from car; transfer of cask to position over storage cell; grapple of fuel basket loaded with dummy fuel; raising it into the cask; and transfer to the horizontal position on the railroad car and back to the storage cell to unload

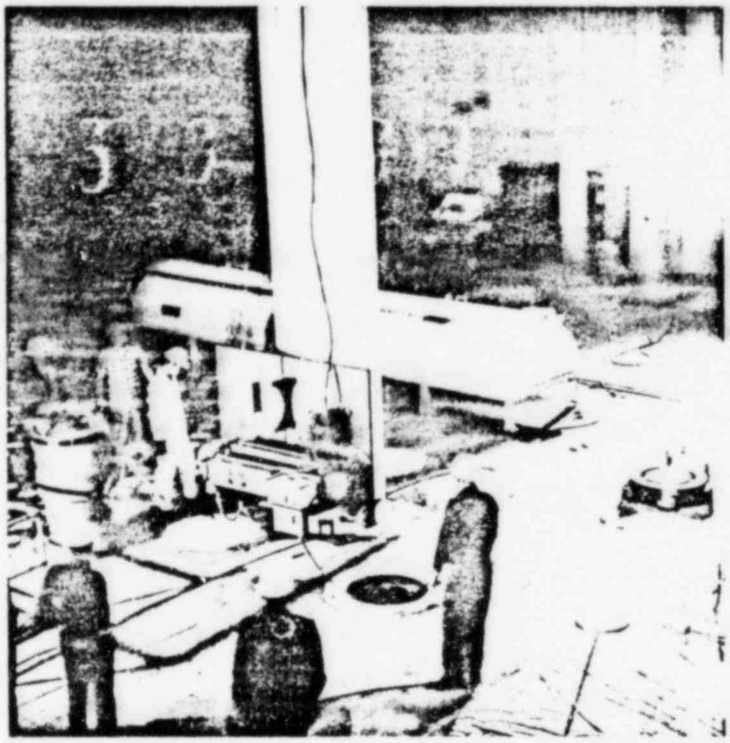

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Figure 8-3. Checkout of Cask at HNPF Moderator Cell

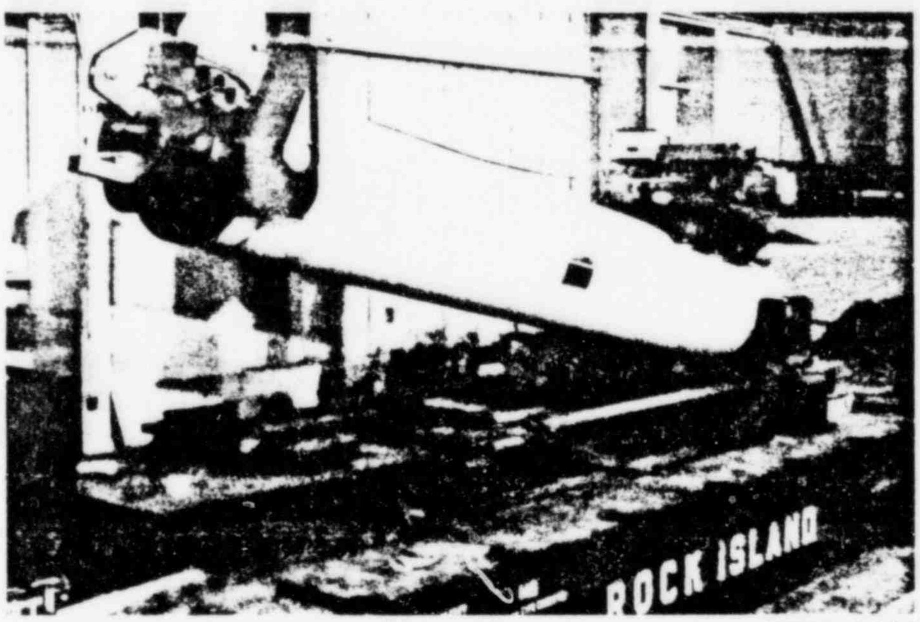

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Figure 8-4. Loading Configuration

the basket with dummy fuel. Limit switches and load-cell high- and low-load relay cutouts were adjusted. Figure 8-3 shows checkout operations at the storage cells.

In operation the cask had a configuration for loading at HNPF as shown in Figure 8-4. This included a demountable hoist, pulleys, wire rope, and a load cell for operating the fuel pickup grapple. With the previously made six-element fuel basket ready and placed in a HNPF moderator-element storage cell, the cask was removed from its cradle on the railway car and placed on end near the storage cell. The cask lower head was detached and left on the floor while the plug from the moderator cell was removed. With the cask placed over the open cell and fuel basket, the cask grapple was run down to engage the basket (with bottom shielding) which was then drawn into the cask. The cask was then placed on its lower head, which was refastened, then put into its shipping configuration (Figure 8-2) by removing the grapple-hoist equipment, placing the cask in its railroad car cradle, and adding the sunshade.

At the Savannah River Plant unloading and fuel storage were under water-shielding. This was done by removing the sunshade and moving the cask from its cradle to a vertical position submerged in the tank. The upper head was then taken off under water and the fuel removed as shown in Figure 8-5. Removal, drying, and reassembly in the railroad car completed the operational cycle.

The first cask was loaded with fuel and prepared for shipment on Wednesday, January 31, 1968, the second on Friday, February 2.

The first cask arrived at Savannah River Plant on Monday, February 5; Figure 8-5 shows a fuel bundle being removed from the submerged cask.

The SRP procedure for receiving the HNPF fuel included a test to prove the absence of sodium on the external surfaces of the fuel canisters or basket. This test consisted of passing moist nitrogen through the cask to react any sodium present with a small amount of moisture which would then produce hydrogen. A sample of the cask atmosphere was then taken and sent to the laboratory for a check on the presence of hydrogen. This test was done four times with samples taken periodically over a 6-hr period; none of the samples tested

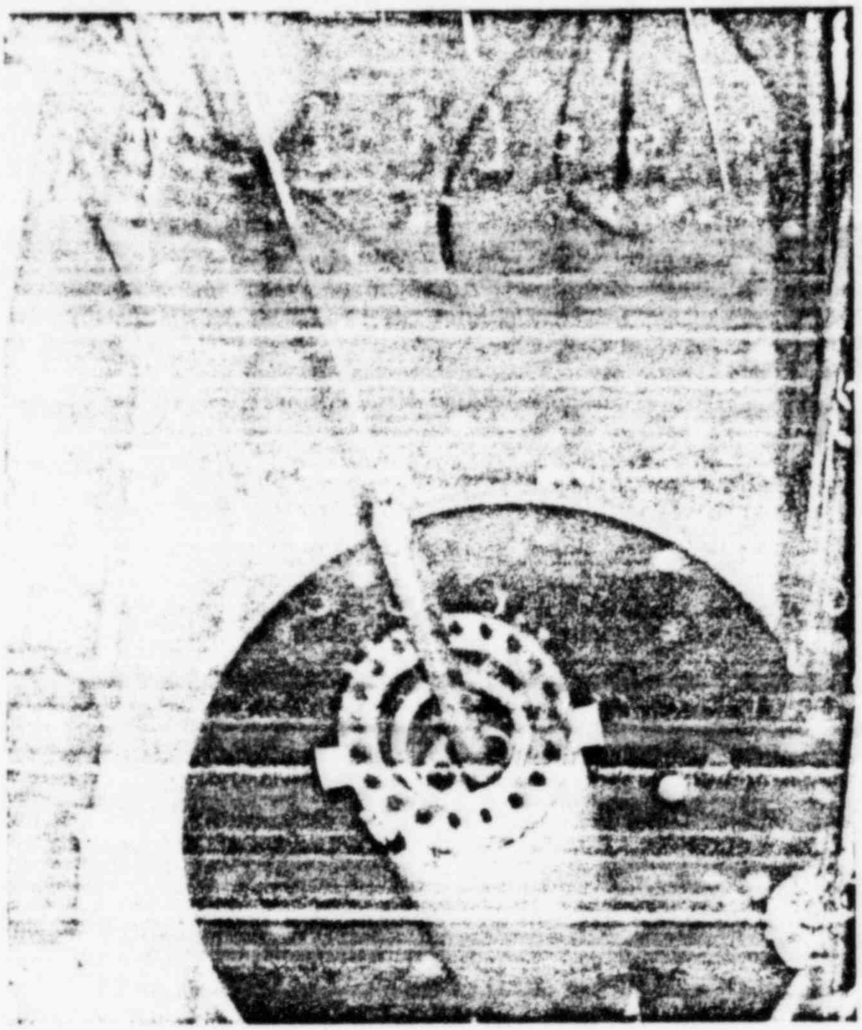

Figure 8-5. Unloading Fuel from the Cask Under Water at Savannah River Plant

showed the presence of hydrogen. During shipping, a change was effected to reduce the 6-hr testing time at SRP. Wet nitrogen was put into the cask at HNPF before shipping, thus providing sufficient time for reaction of any sodium that might be present. The first cask was unloaded and ready for its return trip to HNPF on Wednesday, February 7, 1968. The second cask arrived on February 8, and was unloaded and ready for return to HNPF the following day. Twenty-five shipments were made from HNPF to SRP.

The actual fuel shipment proceeded very smoothly. The turnaround time at each plant was consistently 1 to 2 days instead of the 3 to 4 predicted. Careful coordination with the five railroads which handled the shipments* and the excellent cooperation received from their management and operating personnel resulted in an average round-trip time of 2 wk rather than the 3 which had been estimated for the approximately 2600-mi round trip. The last shipment of fuel arrived at SRP on July 31, 1968. The fuel-shipping casks and cradles were then removed from the railroad cars and put into storage at SRP along with the cask-handling and ancillary equipment. The railroad cars were then released to the CRI & PRR.

*Chicago, Rock Island, & Pacific (originating line); Cheasapeake & Ohio; Carolina, Clinchfield, & Ohio; Columbia, Newberry, & Laurens; and the Seaboard Coast Line.

9.0 COST OF HNPF RETIREMENT

The cost of retiring the HNPF classified by the decontamination and decommissioning activities described in Section 6.0 of this report was as follows:

Activity	Cost (In Thousands)
(1) Primary-sodium disposal	$ 142
(2) Secondary-sodium disposal	149
(3) Irradiated-fuel disposal	883*
(4) Unirradiated-fuel disposal	176*
(5) Reaction of residual primary sodium and retirement of the primary-sodium system	718
(6) Reaction of residual secondary sodium and retirement of the secondary-sodium system	323
(7) Disposition of contaminated and irradiated material	191
(8) Reactor isolation	142
(9) Retirement of auxiliary systems	349
(10) Securing of isolation structure	810
(11) Retirement of the radioactive-waste facility	53
(12) Final close out of HNPF	271
Total	$4,207

At the time of its termination the Hallam project had not yet acquired casks for shipping its irradiated fuel. During the HNPF retirement two such casks were obtained as described in Section 8.0 of this report. The cost associated with them ($649,000) is not included in Activity 3 above since they were designed and fabricated as general-purpose casks for continuing use in the shipment of other fuels after their initial use in transporting the Hallam irradiated-fuel subassemblies to Savannah River for reprocessing and recovery of the contained uranium. It should also be pointed out that the retirement costs tabulated above have not been reduced by the estimated value of the uranium that will be

*Fuel-disposal costs are necessary nuclear plant operational costs and can be excluded from the retirement costs.

recovered from the Hallam fuel (both irradiated and unirradiated) less the estimated cost of reprocessing. Note also that the Activity 3 and 4 cost total of \$1.05 M can be regarded as normal plant-operating expenditure necessary regardless of plant disposition and therefore the retirement cost can be decreased to \$3.148 M.

In the course of decommissioning the HNPF, plant equipment, and components (other than the fuel) having an initial acquisition cost of \$6.5 M, which represents about 15% of the total construction cost of the HNPF, were salvaged and shipped for reuse elsewhere in other AEC programs, as shown in Appendix V. The approximately \$400,000 cost of removing and packaging these items was charged to the recipients, who also bore the shipping costs, and has therefore been excluded from the above tabulation.

43.3 Million
Construction Costs

7.3%

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2. AI-65-217, "HNPF Moderator Element Examination Program, January 1966

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5. AI-AEC-MEMO-12794, "Final Status Report and Safety Analysis of the Hallam Nuclear Power Facility Site and Remaining Structures," W. F. Heine, November 1968

6. NAA-SR-9799, Report No. 3, September 1, 1963 to February 29, 1964

7. "HNPF Monthly Retirement Report CPPD Sheldon Station," L. J. Cooper, Ed.

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APPENDICES TO

REPORT ON RETIREMENT

OF

HALLAM NUCLEAR POWER FACILITY

AI-AEC-12709

APPENDIX I

RETIREMENT ACTIVITY SPECIFICATIONS 1 THROUGH 12

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SPECIFICATION 1.
PRIMARY-SODIUM DISPOSAL

I. PURPOSE

The purpose of this activity specification is to state the method to be used to prepare and ship primary sodium from the HNPF site to the AEC designated recipient at Richland, Washington.*

II. ACTIVITY PLAN

The 9662 ft^3 (555K lb) of primary sodium is presently stored in the five primary sodium fill tanks located on the 1415-ft level of the reactor building. The specific activity of the sodium is approximately 0.03 $\mu c/gm$. The source of this activity is Sodium-22 which has a 2.58 yr half-life.

The tank heaters will be turned off and the sodium allowed to solidify. The fill tanks will be removed from the fill tank vault, placed on railroad cars, and shipped to the AEC designated recipient in accord with AEC and ICC regulations.

III. CRITERIA

A. ENGINEERING, SCHEDULING, AND REQUIREMENTS

Primary sodium disposal activities are to be coordinated with other HNPF retirement activities (e.g., irradiated fuel disposal, removal of residual sodium, and secondary sodium disposal) to optimize safety and the overall schedule. Special moving equipment will be provided under a heavy equipment moving subcontract.

The method of disposal of primary sodium (shipping in fill tanks) as outlined in this specification, was selected as the result of engineering feasibility and economic trade-off studies.† Alternate methods considered were shipping in drums or in tank cars. Removal and shipment of sodium in the fill tanks proved to be the most economical and feasible method of disposal. The full primary fill tanks, with relatively minor modifications to their existing cradles, are structurally adequate to withstand removal, handling, and shipping of loads. Modifications include the addition of lifting lugs and cross bracing to the existing fill-tank cradle and to the tank support saddles welding the tank support shoes. Each fill-tank cradle structure will then be lifted from the primary fill-tank vault by use of a mobile crane and will subsequently be transported across the auxiliary bay and to the railroad car by means of rollers and skids; lastly, it will be crane loaded onto the railroad car and tied down for shipping. To facilitate fill tank movement through the auxiliary bay, certain HNPF structures and hardware must be removed and/or modified. Such structures and hardware include the south helium bottle bank, the helium low pressure tank and gauge board, and portions of the sodium melt station structure.

Care shall be exercised to minimize damage to tank heaters, insulation, and instrumentation during removal and shipment. The structural integrity of the tanks and their support structures shall be preserved during removal and shipping operation.

Atomics International will have the prime responsibility for on-site inspection of the preparation and tiedown of the sodium fill tanks to the railroad cars, for compliance with ICC regulations, and shall make all necessary arrangements for the shipment.

*Letter, Milton Shaw to K. Dunbar, March 13, 1967, RDT:PE:A044, Disposition of Hallam Facility Primary Sodium

†NAA-SR-MEMO-12269, HNPF — Proposed Methods for Removal and Shipment of Bulk Sodium.

Major subcontractual work items, engineering studies, and equipment design and procurement to be conducted in support of this activity are:

1. Subcontractual work in connection with isolation of the fill tanks from the service system, and capping the nozzles of the tanks and end of lines,
2. Heavy-equipment moving, subcontracting for removal of fill tanks from the primary fill tanks vault, transporting the tanks to railroad cars, and placing the tanks on the cars,
3. Design of tank-to-railroad car tiedown devices and supplementary engineering studies in support of ICC Special Permit No. 4913 (ICC Special Permit No. 4913 is the authorization to ship sodium metal), and
4. Subcontractual work in connection with removal of HNPF hardware and auxiliary bay structures to facilitate tank movement through the auxiliary bay and the restoration of such structures and hardware consistent with safety and further retirement activities.

B. HEALTH AND SAFETY

This activity will be carried out under normal HNPF and CPPD safety regulations. Safety during sodium handling, sodium pipe cutting, sodium component handling; and radiological safety will be in accord with the HNPF Operations Manual, Volume I-a, Part III. Heavy equipment lifting and moving will be conducted in accord with the CPPD Safety Manual.

C. STANDARDS

Welding procedures, welder qualifications, and inspection for seal-capping cut piping on the removed fill tanks shall be in conformance to the ASME Boiler and Pressure Vessel Code, Section VIII with the exception that welding of structural support members may be performed under welding procedures and welder qualifications of the American Welding Society.

Closure, packaging, and shipment of the sodium tanks shall be in compliance with ICC regulations except as modified by ICC Special Permit No. 4913.

IV. METHOD OF ACCOMPLISHMENT AND SEQUENCE OF EVENTS

1. Fill tank heaters will be turned off and the sodium allowed to solidify.
2. An inert atmosphere will be preserved via the usual techniques used during sodium pipe cutting operations at the HNPF (e.g., use of plastic bags to preserve inert atmosphere during cutting).
3. Electrical instrumentation and wiring will be disconnected at breaker panels and tank junction boxes. In cases where electrical leads must be cut, as long a lead as is reasonably practical will be left attached to the tank. Level and temperature instrumentation associated with the tank will be salvaged, cleaned, packaged, and shipped. Electrical heaters and instrument leads will be salvaged, in so far as practical, together with the tank and shall be labeled to facilitate reassembly. Pertinent wiring, instrumentation diagrams, system descriptions, and drawings shall accompany the tank shipment.
4. Vent, fill, and drain lines will be cut and the pipe and nozzle stubs sealed by welding on caps. The length of pipe stubs to be left will be specified in the detailed procedures. Stub lengths shall be in accordance with shipping requirements and shall allow for future removal of tank contents. The nitrogen atmosphere of the tanks shall be preserved during such cutting and sealing operations.

5. Existing tank cradle support structures shall be modified so as to provide suitable pickup points (lugs) and cross bracing for lifting and shipping operations.

6. A representative sodium sample shall be obtained from each tank. Chemical and radiochemical analyses shall be performed by AI, and the information (properly identified as to the tank to which it pertains) shall be provided to the AEC.

7. Each tank and its modified cradle support structure will be lifted from the primary sodium vault area, transported to, and placed on a railroad car by heavy equipment obtained under a heavy-moving subcontract. Portions of the tank insulation, heaters, and instrumentation may be removed to facilitate lifting of the tanks through the primary fill tanks vault access plug. Reasonable care will be exercised to protect remaining insulation, heaters, and instrumentation during tank lifting and transport.

8. In order to facilitate transport of the fill tanks to the railroad car loading area, certain HNPF auxiliary bay structures and hardware may be removed (e.g., south helium bottle bank, helium low pressure tank and gauge board, and portions of the sodium melt station structure). Such structures and hardware will subsequently be modified and/or restored to a condition compatible with safety requirements and further retirement activities.

9. Each tank and its modified support structure will be tied down to the railroad car in accordance with ICC regulations and good engineering practice.

10. Prior to shipment, Atomics International will inspect the tied down tank and support structure for conformance with all pertinent regulations and good engineering practice. Final on-site sign-off for shipment will be by CPPD with AI concurrence.

SPECIFICATION 2.
SECONDARY-SODIUM DISPOSAL

I. PURPOSE

The purpose of this activity specification is to state the method to be used to prepare and ship secondary sodium from the HNPF site to the Santa Susana Facility of Atomics International.*

II. ACTIVITY PLAN

The 4000 ft^3 (220K lb) of secondary sodium is presently stored in the three secondary fill tanks located on the 1420-ft level of the reactor building. The tank heaters have been turned off and the sodium allowed to solidify. The fill tanks will be removed from the secondary sodium service area, placed on railroad cars, and shipped to Chatsworth, California. At Chatsworth, the tanks will be transferred from railroad car to truck and shipped to Atomics International at the Santa Susana Facility.

III. CRITERIA

A. ENGINEERING, SCHEDULING, AND REQUIREMENTS

Secondary sodium disposal activities are to be coordinated with other HNPF retirement activities (e.g., irradiated fuel disposal, removal of residual sodium, and primary sodium disposal) to optimize safety and the overall schedule. Special moving equipment will be provided under a heavy-equipment moving subcontract.

The secondary sodium fill tanks will be isolated, and the cut lines will be capped. The tanks will be lifted from the secondary sodium area by use of a crane located outside the reactor building and extending through the auxiliary bay wall.

The method of disposal of secondary sodium (shipping in the fill tanks) as outlined above, was selected as a consequence of engineering feasibility and economic tradeoff studies.† Alternate methods considered were shipping in drums or tank cars. Removal and shipment of the sodium while in the fill tanks proved to be the most economical and feasible method of disposal. The full secondary fill tanks, with relatively minor modifications to their existing cradles, are structurally adequate to withstand removal, handling, and shipping loads. Modifications include the addition of lifting lugs and cross bracing to the existing fill tank cradle, and welding the tank support shoes to the tank support saddle. The load carrying capacity of the floor slab in the vicinity of the secondary fill tanks access plug is questionable; hence, removal through the side of the building was selected.

Care shall be exercised to minimize damage to tank heaters, insulation, and instrumentation during removal and shipment. The structural integrity of the tanks and their support structure shall be preserved during removal and shipping operations.

Atomics International will have the prime responsibility for on-site inspection of the preparation and tiedown of the sodium fill tanks to the railroad cars for compliance with ICC regulations, for arrangements for the shipment, and for details and supervision of each secondary sodium tank shipment after the tank has left the HNPF site.

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*Letter, Milton Shaw to K. Dunbar, September 27, 1966, Disposition of Hallam Secondary Sodium
†NAA-SR-MEMO-12269, HNPF – Proposed Methods for Removal and Shipment of Bulk Sodium

Major subcontractual work items, engineering studies, and equipment design and procurement to be conducted in support of this activity are:

1. Subcontractual work in connection with isolation of the fill tanks from the service system and capping the nozzles of the tanks and end of lines,
2. Heavy-equipment moving, subcontracting for removal of fill tanks from the secondary sodium service area, transporting each tank to a railroad car, and placing the tank on the car,
3. Subcontractual work in connection with restoration of the auxiliary bay outer structure and HNPF auxiliary systems after removal of the tanks, and
4. Design of tank-to-railroad car tiedown devices and supplementary engineering studies in support of ICC Special Permit No. 4913. ICC Special Permit No. 4913 is the authorization to ship sodium metal.

B. HEALTH AND SAFETY

This activity will be carried out under normal HNPF and CPPD safety regulations. Safety during sodium handling, sodium pipe cutting, and component handling will be in accordance with the HNPF Operations Manual, Volume I-a, Part III. Heavy equipment lifting and moving will be conducted in accordance with the CPPD Safety Manual.

C. STANDARDS

Welding procedures, welder qualifications, and inspection for seal-capping cut piping on the removed fill tanks shall conform to the ASME Boiler and Pressure Vessel Code, Section VIII with exception that welding of structural support members may be performed under welding procedures and welder qualifications of the American Welding Society.

Closure, packaging, and shipment of the sodium tanks shall comply with ICC Regulations except as modified by ICC Special Permit No. 4913.

IV. METHOD OF ACCOMPLISHMENT AND SEQUENCE OF EVENTS

1. Fill tank heaters will be turned off and the sodium allowed to solidify.
2. An inert atmosphere will be preserved via the usual techniques used during sodium pipe cutting operations at the HNPF (e.g., use of plastic bags to preserve inert atmosphere during cutting).
3. Electrical instrumentation and wiring will be disconnected at breaker panels and tank junction boxes. In cases where electrical leads must be cut, as long a lead as is reasonably practical will be left attached to the tank. Level and temperature instrumentation associated with the tank will be salvaged, cleaned, packaged, and shipped. Electrical heaters and instrument leads will be salvaged, in so far as is practical, together with the tank and shall be labeled to facilitate reassembly. Pertinent wiring and instrumentation diagrams, system descriptions, and drawings shall accompany the tank shipment.
4. Vent, fill, and drain lines will be cut, and the pipe and nozzle stubs sealed by welding on caps. The length of pipe stubs to be left will be specified in the detailed procedures. Stub lengths shall be in accordance with shipping requirements and shall allow for future removal of tank contents. The nitrogen atmosphere of the tanks shall be preserved during such cutting and sealing operations.

5. Existing tank cradle support structures shall be modified to provide suitable pickup points (lugs) and cross bracing for lifting and shipping operations.

6. A representative sodium sample shall be obtained from each tank. Chemical analyses shall be performed by AI, and the information (properly identified as to the tank to which it pertains) shall be provided to the AEC.

7. Each tank and its modified cradle support structure will be lifted from the secondary sodium service area, transported to, and placed on a railroad car by heavy equipment obtained under a heavy-moving subcontract. Portions of insulation on tank ends may be removed to facilitate lifting of the tanks through the auxiliary bay access plug. Reasonable care will be exercised to protect remaining insulation, heaters, and instrumentation during tank lifting and transport.

8. Each tank and its modified support structure will be tied down to the railroad car in accord with ICC regulations and good engineering practice.

9. Prior to shipment, Atomics International will inspect the tied down tank and support structure for conformance with all pertinent regulations and good engineering practice. Final on-site sign-off for shipment will be by CPPD with concurrence by AI.

10. The tanks will be shipped by rail to Chatsworth, California. Atomics International will assume full responsibility for all aspects of the shipment once the tanks have left the HNPF site.

11. Shipment of the secondary sodium in fill-tanks will be made under ICC Special Permit No. 4913.

12. Reactor building (auxiliary bay) outer structure removed to facilitate removal of the fill tanks, and HNPF auxiliary systems equipment and lines affected by removal operations will be restored or retired in a condition consistent with safety and further retirement activities.

SPECIFICATION 3.
DISPOSAL OF IRRADIATED FUEL

I. PURPOSE

The purpose for this activity is to remove all irradiated fuel from the HNPF site to the Savannah River Reprocessing Plant, at Aiken, South Carolina, as directed by the Atomic Energy Commission. The remaining irradiated process tubes will be reassembled on their hanger rods and returned to the reactor. Irradiated miscellaneous scrap will be packaged and disposed of as radioactive waste.

II. ACTIVITY PLAN

The present inventory of irradiated fuel elements at the HNPF site is as follows:

1. 140 U - 10 Mo elements, and
2. 10 UC elements.

The irradiated U - 10 Mo and UC fuel elements are stored in the HNPF fuel storage cells. The condition of these elements is as follows: 136 U - 10 Mo fuel elements are complete with shield plug and hanger rods, and have been washed; sixteen of these elements were separated from their shield plug but have been reconnected with a special coupling and washed. Three U - 10 Mo fuel elements have no process tubes, and have not been washed. One U - 10 Mo fuel element has no hanger rod or process tube, and is contained in a temporary canister in storage. The 10 UC fuel elements have not been washed but are intact with hanger rod and shield plug. One of these UC elements has been separated from its process tube.

Disassembly and encapsulation of irradiated fuel element assemblies will be performed in the HNPF maintenance cell. The U - 10 Mo fuel element now stored in a temporary canister will be removed from this canister by use of the method by which it was placed in the canister. This element will then be double encapsulated by use of the same procedures as are used for the other element assemblies.

The fuel elements will be removed from their process tubes and hanger rods, and double encapsulated in stainless steel canisters which will be capped and remotely welded in the cell. The encapsulated fuel elements will be placed in a six element shipping basket, and the basket will be placed in an AEC approved HNPF irradiated fuel shipping cask designed and fabricated to meet the requirements of the AEC Manual Chapter 0529, 10CFR, Part 71. Irradiated fuel shipments will be made in accordance with AEC and ICC regulations.

The remaining process tubes will be reattached to the hanger rods and returned to the reactor. Retaining pins and other irradiated scrap will be deposited in small metal containers and removed as radioactive waste.

The U - 10 Mo elements will be encapsulated and shipped first. Since there are 140 U - 10 Mo fuel elements, the last shipment of U - 10 Mo fuel will contain only 2 U - 10 Mo elements; however, 4 UC elements will be added to make a normal six element shipment. This shipment will be identified so that the Savannah River Plant will be sure to recognize the two types of fuel, in the event that their reprocessing procedures require different handling for the two types of fuel. There are no safety problems involved in "mixing" the two types of fuel since the criticality hazard is a function of enrichment, and the cask has been analyzed for a full load of the highest enrichment fuel, which is the 4.9% enriched UC, although there are only two of these elements in the plant, see addendum.

III. CRITERIA

A. ENGINEERING, SCHEDULING, AND REQUIREMENTS

This activity will be conducted in coordination with other plant activities during plant retirement activities. The fuel handling machine must be available for transfer of the fuel from the storage cells to the maintenance cell, the removal of the process tube, the removal of the fuel element from the hanger rod, and the return of the process tube to the reactor.

The regular maintenance cell operating equipment will be repaired and serviced to assure operational reliability. The maintenance cell air inleakage will be at a rate of about 10 SCFM; a rate low enough that the cell can be purged with nitrogen when necessary and held at an oxygen content level of 2 to 4% which is below the combustion limit for sodium. " .e maintenance cell will be purged with nitrogen, and the oxygen concentration of the cell atmosphere will be maintained at between 3 and 4% during encapsulation of UC and unwashed U - 10 Mo fuel elements. This oxygen level is safe for these operations and conserves nitrogen.

The maintenance cell will be equipped with special equipment for handling and encapsulating the irradiated fuel elements. The special equipment, in addition to the regular maintenance cell equipment is:

1. A digital controlled remotely operated program welder,
2. A fuel element and canister holding fixture,
3. Shipping canister handling fixtures,
4. Wall mounted tables and fixtures, for mounting the remote welding fixture,
5. Tool trays,
6. Process tube adapters to reattach the process tube to hanger rod,
7. Process tube retaining pins, and
8. An assortment of miscellaneous small tools modified for use with master slave manipulators.

One UC fuel element will be visually inspected for damage incurred during plant operation; the element will be photographed for record purposes and any abnormalities will be reported.

B. HEALTH AND SAFETY

Normal existing HNPF safety procedures will be employed. Radiological safety procedures and records will be in accordance with HNPF Operations Manual, Volume I-a, Part III.

Nuclear fuel accountability records will be in accordance with normal HNPF procedures and the HNPF Operations Manual, Volume I, Part IV.*

The fuel shipping cask will be inspected for loose foreign objects before being placed upright over the maintenance cell. A foreign object in the cell could cause damage to the cell equipment and/or cause unnecessary contamination. Extra precaution will be taken in the inspection of the cask hoisting equipment. A failure in the hoist wire rope or grapple could drop a loaded fuel shipping basket from the shipping cask into the cell. The maximum distance the fuel shipping basket could fall is fifty feet.

*For cask unloading procedures, see Section III-B of NAA-SR-TDR 12274, revision issued April 28, 1967, "HNPF Six-Element Irradiated Fuel Shipping Cask."

An analysis was made by Atomics International to ascertain the possible damage to the six encapsulated fuel elements and their canisters in the event of such a drop. The study revealed that there might be stress cracks in the fuel canisters but that no crack would be large enough to release fuel or solid sodium. The false floor in the maintenance cell would absorb a great portion of the shock; this floor is also strong enough to absorb the shock without collapsing. If such an incident should occur, recovery procedures would have to be written after the damage had been evaluated.

An analysis was also made for the supposition of dropping a bare fuel element into the maintenance cell without an inert atmosphere. There is only one operation wherein a fuel element is not contained, and that is after the fuel element is removed from its process tube and inserted into the canister. If a fuel element is dropped at this point, it could rupture; however, there would be no criticality hazard involved in any rearrangement of the fuel which could take place as a result of its fall, see addendum. The sodium in each fuel rod would be in the solid state, the heat of the element being so low (less than 30 watts per element) that the element temperature would be essentially ambient; well below the melting point of sodium. There being no water in the cell, the sodium would slowly oxidize; no burning would occur. An inert atmosphere would then be established in the cell. Calcium carbonate will be placed in several containers on the false floor of the cell. The master slaves and jib manipulator will be able to cover any sodium that might be exposed with calcium carbonate.

C. STANDARDS

Criticality control will be in accordance with normal HNPF irradiated fuel handling procedures. These procedures are supported by several analyses made by Atomics International. Only one loaded canister shipping basket will be in the cell at any one time.

Welding procedures, welder qualifications, and inspection for capping element canisters shall conform to that part of the ASME Boiler and Pressure Vessel Code, Section VIII, that complies with the requirements of the AEC designated recipient of the encapsulated fuel. Welds will be visually inspected by using the Kollmorgan periscope for deformities that would affect the integrity of the weld. A strip recorder coupled with the heliarc power source will record the welding amperage and voltage during each weld; this record of each weld will be identified with the canister by identity numbers.

Loading and blocking of the fuel transport cask on a flatcar will be inspected by the CPPD Plant Superintendent or his designated representative.*

Test canisters will be welded in-cell by use of the standard production procedures before the program for the encapsulation of fuel elements commences. A test canister will be welded after every six fuel elements encapsulated.

The test canisters consist of tube-to-tube welds made from 2-ft sections of regular canister tubing.[1] Each closure weld shall be visually inspected by use of the Kollmorgan Periscope while in-cell. The test canister welded assemblies will be removed from the cell and given the following bench inspections:

1. Liquid penetrant dye, visual,

2. Mass spectrometry by use of helium and a calibrated leak of 1×10^{-7} atmosphere cc/sec,

*For cask unloading procedures, see Section III-B of NAA-SR-TDR-12274, revision issued April 28, 1967, "HNPF Six-Element Irradiated Fuel Shipping Cask."

3. Hydrostatic burst proof test, and

4. Macro inspection (100X) of the weld heat-affected zone and parent material.

IV. METHOD OF ACCOMPLISHMENT AND SEQUENCE OF EVENTS

Irradiated fuel elements will be handled and shipped as indicated below.

1. Wash each element in the fuel wash cell per HNPF SOP-5301.

   NOTE: UC fuel elements and U - 10 Mo fuel elements without process tubes or separated from their shield plug or otherwise suspected of failure will not be washed.

2. Transfer each element to the maintenance cell with the fuel handling machine per HNPF SOP-5201.

3. Remove process tube and fuel element from shield plug and hanger rod. Replace process tube on hanger rod and shield plug by using as the adaptor, an upper casting from a disassembled unirradiated fuel element or one of the upper castings from Core II that is in stock.

4. Return shield plug, hanger rod, and process tube to reactor or storage cell with the fuel handling machine.

5. Double encapsulate fuel elements in previously inspected stainless steel canisters. Place elements in a six-element shipping basket that has been previously placed in the cell per the cask operating procedures.

   NOTE: The fuel element is placed in double nested canisters. A cap is placed on the inner canister, remotely welded, and inspected. A cap is then placed on the outer canister, and is remotely welded and inspected. Welds will be visually inspected by using the Kollmorgan periscope for deformities that would affect the integrity of the weld. A strip recorder coupled with the heliarc power source will record the welding amperage and voltage during each weld; this record of each weld will be identified with the canister by identity numbers.

6. After six fuel elements have been double encapsulated and loaded into the shipping basket, six previously inspected double nested canisters will be loaded into the maintenance cell by removing the 30-in. cell plug with the reactor high bay crane.

7. Attach to the 15 ton high bay crane hook, a sling and auxiliary hook long enough to reach the fuel element and canister holding fixture stored at the first operating level of the cell.

8. The fuel-element-and-canister holding fixture is attached to the auxiliary hook on the 15-ton high bay hoist and raised to the level of the reactor floor. A double nested canister is placed in each of the six holding fixtures. The holding fixtures loaded with the double canisters are placed in the cell storage rack after removal from the high bay crane.

9. End caps for the canisters, installed in the cell, are lowered into the cell with the high bay 15-ton hook and basket.

10. Steps 2 through 5 are repeated for the six canisters just installed in the cell. The above six canisters, after loading with fuel and welding, will be left in the holding fixture and placed in the cell storage rack until the previously loaded shipping basket is loaded into a fuel shipping cask and an empty shipping basket is lowered into the cell.
11. The fuel shipping cask is placed over a port in the top of the maintenance cell. The cask shall have been previously checked for foreign material such as water. The fuel shipping basket with six encapsulated fuel elements is raised into the cask with the cask grapple and hoist. The cask will be removed from the cell and set on the bottom head for closure.
12. The shipping cask will be placed on its shipping cradle on a railroad flatcar. It will be sealed, and inspected for proper loading,* and shipped to the Savannah River Reprocessing Plant, Aiken, South Carolina.
13. A second fuel shipping cask is placed over a port in the top of the maintenance cell. The shipping basket is lowered into the cell and loaded with the six canistered fuel elements from the cell storage rack. The loaded fuel shipping basket is then raised into the cask with the cask grapple and hoist.
14. Repeat Step 12 for the second cask.
15. Repeat the preceding steps until all irradiated fuel has been shipped from the HNPF site.
16. Shipping of all irradiated fuel shall be in compliance with the AEC Manual, Chapter 0529 regulations and with all ICC regulations.

V.† PACKAGING AND TRANSPORT OF IRRADIATED HALLAM FUEL ELEMENTS – CRITICALITY SAFEGUARDS EVALUATION

A criticality study has been made to evaluate the nuclear safety in the packaging and transport of the irradiated 3.6% U^{235} enriched Hallam U - 10 Mo elements (18-rods/element), the 3.7% U^{235} enriched UC elements, and the 4.9% U^{235} enriched UC elements (8 rods/element). Since the unirradiated elements are more reactive than the corresponding irradiated ones, the nuclear safety evaluation was based on unirradiated fuel elements. Based on this study it was determined that[2] two canister baskets spaced on 28-in. centers, are safe when each one is loaded to its capacity of six elements. Two casks so loaded are safe, provided the minimum lead thickness of each cask is at least 6 in. The safety established is independent of the degree of water moderation and reflection.

In an earlier study[2] it was determined that the minimum critical number of the U - 10 Mo elements was 9+: when there is optimum spacing between elements, a water moderator, and water reflection.

Although the minimum critical number of the 3.7% U^{235} enriched UC elements under the above conditions was not directly calculated, it may be estimated from a comparison of the reactivity of seven U - 10 Mo elements in a given shipping cask with that of the 3.7% U^{235} enriched UC elements in the same cask. In an earlier criticality study[3] in which the safety of a 7-element shipping cask was evaluated, it was determined that the k_{eff} of the package would be 0.653 when loaded with the U - 10 Mo fuel, compared with a k_{eff} of 0.641 when loaded with the 3.7% U^{235} enriched UC fuel. Under these conditions, the cask and surroundings were flooded with water, but there was no water

*For cask unloading procedures, see Section III-B of NAA-SR-TDR-12274, revision issued April 28, 1967, "HNPF Six-Element Irradiated Fuel Shipping Cask."

†This section was originally designated Appendix I of Specification 3.

inside the gas tight fuel canisters. With water inside the fuel canisters as well as inside the casks and surroundings, the corresponding k_{eff}'s were found to be 0.884 and 0.868 respectively; therefore it is safe to load a cask with six elements (less than two-thirds the minimum critical number under optimum conditions of water moderation and reflection) of either the U - 10 Mo or the 3.7% U^{235} enriched fuel.

Determination of the nuclear safety of the shipping cask, evaluated in the referenced report,[3] when loaded with the 4.9% U^{235} enriched UC elements, included the neutron poisoning effect of the canister surrounding each element; while the analysis for the 3.7% U^{235} enriched UC as well as the U - 10 Mo loading did not. The reactivity of the cask loaded with the 4.9% U^{235} enriched elements was 0.92 when no allowance was made for the poisoning effect of the canister[4] (water flooding within fuel elements as well as between them). It is planned to place each element in two concentric steel canisters (double containment). There is about 45% more steel/element in these canisters than in the reference design.[3] Since the reactivity of the reference cask, when loaded with the 4.9% U^{235} enriched fuel (including steel poisoning effect) was less than 80% of that when loaded with the fuels of lower enrichment (excluding steel poisoning effect), it can be concluded that the minimum critical number of the 4.9% U^{235} fuel elements in their canisters is greater than nine; therefore, six elements, so packaged, are safe independent of container configuration.

It is desirable to have two canister baskets, each loaded with six elements, inside the maintenance cell at a center-to-center separation between baskets of 28 in. If it were possible to flood the cell (not likely), there would be at least 11 in. of water between fuel in adjacent baskets. This is more than enough between baskets to isolate them from each other.[5] If there is no water or other type of moderator material in the cell, it is not possible to make an infinite number of elements (having the above enrichments) critical. The minimum critical enrichment[6] in an unmoderated system is greater than 5.0% U^{235}.

The safety of two baskets loaded with six elements each and spaced on 28-in. centers was also evaluated under the conditions of there being water inside the element but none between elements. If this were possible, the assembly would be more reactive than under either of the above conditions. The allowable solid angle[6] is greater than 2.4 steradians (based on a k_{eff} 0.66[3] for a water moderated and reflected assembly of 4.9% U^{235} enriched UC elements). This is conservative since the allowable solid angle should be based on the moderated but unreflected system. The actual solid angle is less than 1.7 steradians; therefore, the array is also safe under these conditions.

Two casks, each loaded with six elements, placed side by side are also safe, provided the lead thickness in each cask is greater than 6 in. (at least 12 in. of lead between fuel in adjacent casks). Experimental data indicate that about a 10-in. thickness of lead is adequate for neutronic isolation.[5]

REFERENCES

1. AI Drawing N69081014, Parts 9 to 13 and 3 to 6.

2. N. Ketzlach, "Criticality Study for Hallam Fuel Fabrication and Shipment," TDR No. 5975, December 16, 1960.

3. S. Berger, R. A. Hewson, et al., "HNPF Spent Fuel Shipping Cask," NAA-SR-TDR-10549, October 9, 1964.

4. H. Rood, personal communication, September 8, 1966

5. T. G. McCreless, R. R. Smith, et al., "Neutronic Isolation Characteristics of Concrete, Lead, Wood, Polyethylene and Beryllium," Trans. Am. Nucl. Soc. Vol 8, No. 2, November 1965.

6. "Nuclear Safety Guide," TID-7016 Rev. 1, 1961.

7. NAA-SR-TDR-5975, Criticality Study for Hallam Fuel Fabrication, by N. Ketzlach, December 16, 1960.

8. NAA-SR-TDR-6279, Criticality Study for UC Fuel Fabrication for Hallam Core I, N. Ketzlach, April 12, 1961.

9. NAA-SR-TDR-9168, Storage of Hallam Core II Fuel Rod Cluster, by N. Ketzlach, October 29, 1963.

10. NAA-SR-TDR-12274, HNPF 6 Element Irradiated Fuel Shipping Cask, December 15, 1966, page 199, paragraph b, by B. Hewson.

SPECIFICATION 4.
DISPOSAL OF UNIRRADIATED FUEL

I. PURPOSE

The purpose of this activity is to prepare and ship all unirradiated fuel rods from the HNPF site to Atomics International, Canoga Park, California.

II. ACTIVITY PLAN

The present inventory of unirradiated (new) fuel at the HNPF site is as follows:

57 UC elements,

12 UC fuel rods,

13 U - 10 Mo elements,

12 U - 10 Mo fuel rods (assembled into a partial element), and

2 depleted U elements.

There are 8 UC fuel rods per element and 18 U - 10 Mo fuel rods per element; hence, a total of 246 U - 10 Mo and 468 UC rods are to be shipped. Fuel elements will be disassembled and shipped in accordance with AEC and ICC regulations.

Seven assembled UC elements are in the Fuel-Moderator Assembly Building. The remaining assembled UC and U - 10 Mo elements, along with a partial assembly of 12 U - 10 Mo rods, are stored in HNPF fuel storage Pits No. 1 and 2. The 12 UC rods are stored in the Fuel-Moderator Assembly Building. Two depleted U elements are stored in the Fuel-Moderator Assembly Building.

The UC fuel rods will be prepared and shipped first.

III. CRITERIA

A. ENGINEERING, SCHEDULING, AND REQUIREMENTS

This activity will be conducted in coordination with other plant activities during initial plant retirement activities. The high bay crane and fuel transport vehicle must be available prior to removal of fuel elements from storage cells. The fuel handling fixtures used in the original assembly of the fuel elements will be used in disassembly.

The unirradiated fuel rods will be shipped in containers identical with the containers fabricated under B of E Permit No. 1215 for shipment of unirradiated fuel rods to Hallam, Nebraska from Canoga Park, California. The unirradiated fuel rod shipping containers shall be numbered by stamping with a one half inch metal stamp in the following manner: Mark the containers on each end above the humidity indicator with the number H-1 through H-40, there being 40 shipping containers. There shall be painted on the top of the containers in red enamel the same number that has been stamped on the end plates. This number shall be noted on all records and shipping forms.

Inspection of packaging will be made by the supervisor of the crew performing the disassembly operation. Release for off-site shipment will be authorized by the Deactivation Supervisor or his designated representative. CPPD shall be responsible for shipment of the fuel containers.

B. HEALTH AND SAFETY

Normal existing HNPF and CPPD safety procedures will be employed in this activity. Radiological safety and records will be in accordance with HNPF Operations Manual, Volume Ia, Part III. Fuel inventory and accountability will be in accordance with normal HNPF procedures and the HNPF Operations Manual, Volume I, Part IV. Criticality control will be maintained in accordance with the restrictions outlined in the following HNPF criticality studies:

NAA-SR-MEMO-5975,

HNPF Operations Manual, Volume I, Part IV,

NAA-SR-MEMO-6279,

NAA-SR-MEMO-6479,

NAA-SR-MEMO-8151,

AI-MEMO-8810, Rev 1, and

NAA-SR-MEMO-9168

Handling, storage, and shipment of U - 10 Mo and UC rods shall be performed in such a manner that mixing of U - 10 Mo and UC fuel rods will not occur. At no time shall U - 10 Mo and UC rods both be present in the Fuel-Moderator Assembly Building, nor be shipped together. The depleted U fuel rods will be packaged in the same manner as the U - 10 Mo and UC fuel rods. The depleted U fuel rods may be shipped with a regular shipment of either U - 10 Mo or UC fuel rods. They will be shipped with the first shipment of UC fuel containers. Each shipping container has a capacity of 16 UC fuel rods or 36 U - 10 Mo fuel rods. All containers will have an argon atmosphere.

The disassembly area will be roped off, no smoking will be permitted, and the appropriate radiation signs will be posted. Dry-powder fire extinguishers will be located within easy reach of work stations where acetone is utilized in wiping the fuel rods and components. There will also be available a supply of calcium carbonate to be used on sodium in the event a rod is ruptured.

Minimal quantities of acetone will be stored and used in the Moderator Assembly Building and will be contained in metal safety cans. Used acetone moistened pads shall be deposited in covered metal containers and disposed of by appropriate means.

C. STANDARDS

The fuel rods shall be cleaned with acetone before being placed in the shipping container. The fuel rods shall be handled with lint-free gloves at all times. Visual inspections shall be performed as disassembly proceeds, to detect any damage present which affects the integrity of the cladding. In the event of damage to the cladding of a fuel rod, the Deactivation Supervisor (CPPD) and the AEC Site Representative shall be notified. The rod will be examined under the cognizance of these responsible persons and the following general disposal procedures shall be followed: Where the integrity of the cladding is not breached, the fuel rod shall be shipped in the regular manner and the damage shall be noted on the records. Should the integrity of the cladding be breached, the fuel rod shall be inserted into an aluminum pipe that is enclosed on both ends with threaded caps and purged with argon. The fuel rod enclosure shall be so constructed that no movement of the fuel rod inside the pipe can occur. The number of the fuel rod shall be painted on the outside of the aluminum pipe in red enamel. The encased fuel rod will be shipped in a regular shipping container with the normal number of fuel rods. The styromethane packing groove will be enlarged to accept the encased fuel rod. The condition of the fuel rod and its number shall be noted in the shipping data.

Shipment shall be in compliance with AEC Manual Chapter 0529 and Bureau of Explosives Permit No. 1215. The fuel rod packaging shall be inspected for compliance with this permit.

IV. METHOD OF ACCOMPLISHMENT AND SEQUENCE OF EVENTS

Unirradiated fuel rods shall be handled and shipped as listed below.

1) Remove unirradiated fuel assemblies from fuel storage in reactor building. Smear for contamination and clean (if necessary).
2) Move complete fuel assembly to fuel transport fixture, enclose in plastic sleeve, and place assembly in fixture.
3) Disconnect shield plug and hanger rod from fuel element. Support fuel element as required.
4) Move the shield plug and hanger rod away from fixture and place in a storage cell.
5) Repeat Steps (1) through (4) and place a second fuel element on the transport fixture.
6) Transport the two fuel elements to the Fuel-Moderator Assembly Building.
7) Place the fuel element in a storage fixture.
8) Remove retaining pins and raise fuel cluster out of process tube.

NOTE

Process tube is still contained within the plastic sleeve and supported in the storage fixture. Smear survey fuel element as it is raised, and clean if necessary.

9) Transfer the fuel element to a disassembly fixture.
10) Smear survey the internal surface of process tube and clean if necessary. Remove the plastic sleeve and transfer the process tube to the HNPF storage building.
11) Remove the fuel rod retaining clips.
12) Wipe each fuel rod with acetone on a lint-free cloth as it is removed from the fuel element disassembly fixture, and inspect for damage of cladding.
13) Dry the fuel rod with a lint-free cloth and place it in the shipping container. Dispose of acetone soaked clothes in a metal container.
14) Repeat Steps (7) through (13) for the second element.
15) Place a cover on the container and leak check same in accordance with specifications designated on Atomics International Drawing No. AX 20003.
16) Purge the shipping container with argon.
17) Ship in accordance with AEC and ICC regulation as specified in Part III-C of this activity specification.
18) Return the fuel element transport fixture to the high bay.
19) Repeat these operations until all unirradiated fuel elements have been shipped.

RETIREMENT OF PRIMARY-SODIUM SYSTEM

I. PURPOSE

The purpose of this activity specification is to state the method to be used for reaction or removal of residual sodium remaining in the primary sodium system and for disposal of primary sodium system hardware and components.

II. ACTIVITY PLAN

The primary sodium system includes the primary sodium main heat transfer system, the reactor vessel, and the primary sodium service system.

Sodium has been drained from the primary sodium system into primary fill tanks. The sodium-filled primary fill tanks will be disposed of per Activity Specification No. 1, "Primary Sodium Disposal." Residual sodium remains in certain portions of the system. Primary sodium system components and hardware will be classified in categories and disposed of accordingly. Component category classifications refer basically to the ultimate methods of disposal of the component; these are contained in Part II, C3, of the HNPF Retirement Plan and are summarized below.

Category A = Property authorized by CH for shipment off site except as scrap. Such property will be removed from its present location and shipped off site to AEC designated recipients.

Category B = Property which will be removed from its present location and shipped off site as scrap.

Category C = Property which will remain in its present location at the HNPF.

Category D = Property which will be removed from its present location and placed in adequately contained subsurface volumes.

Category E = Property which will remain in its present location but whose ownership will be transferred to CPPD.

Category F = Property which must be removed to accomplish the retirement of the HNPF and which CH authorizes to be removed from its present location pending determination of its ultimate disposition. Such property will be provided with suitable protection against damage and temporarily stored until its final status as Category A, B, or D material is decided.

The reactor vessel will be steam cleaned in place as described in Sections IV and V of this activity specification. Other primary system components and hardware will be disposed of as follows:

Category A and F items will be removed from the system, residual sodium present will be removed and/or reacted as necessary, and prior to ultimate disposal, the item will be sealed containing an inert atmosphere if necessary. Category B items will be removed from the system and residual sodium present removed or reacted as necessary. Residual sodium in Category C items will be reduced by reaction and/or physical removal to the specified limits for its safe containment. Where alternatives are stated, the steps employed shall be the most economical of those determined to be technically feasible.

III. ENGINEERING

A. ENGINEERING, SCHEDULING, AND REQUIREMENTS

Operations connected with this activity will be scheduled in coordination with other HNPF Retirement activities. Work contemplated under this activity, however, is largely independent of other activities.

The methods of reaction of residual primary sodium and retirement of the primary sodium system were selected on the basis of system engineering analysis of the problems involved.

Major subcontractual work items, engineering studies, and equipment design and procurement to be conducted in support of this activity are listed below.

1. Design, procurement, and installation of sodium pumping apparatus for removing the residual sodium heel remaining in the reactor vessel
2. Detailed design, equipment procurement, and system modifications to permit the in-place steam cleaning of the reactor vessel
3. Detailed design, equipment procurement, and system modifications to permit the in-place reaction and/or removal of residual sodium in primary system components, other than the reactor vessel, as necessary (e.g., primary main heat transfer system piping)
4. Subcontractual work in connection with design, equipment procural, and existing hardware and structures modifications to permit the removal and/or reaction of residual sodium in Category A, B, and F items (It is recommended, on the basis of engineering feasibility and economic considerations, that auxiliary sodium cleaning facilities be provided. The pump wash cell will be used for sodium cleaning operations which can be economically and reliably performed by remote controlled steam cleaning. Auxiliary facilities will be required for the majority of sodium cleaning operations; e.g., manual removal, swabbing, water flush, heated oil bath, and manual steam cleaning)
5. Heavy equipment removal subcontracts for such removal as is deemed necessary
6. Miscellaneous engineering studies, design, and equipment procural in support of this activity

B. HEALTH AND SAFETY

This activity will be carried out under normal HNPF and CPPD Safety regulations. Radiological safety will be in accordance with the HNPF Operations Manual Volume Ia, Part III, and 10CFR20 regulations. Hardware lifting and moving will be conducted in accordance with the CPPD Safety Manual.

C. STANDARDS

Welding procedures, qualifications, and inspection for all components utilized for reaction fluid containment during in-place steam cleaning operations shall conform to the ASME Boiler and Pressure Vessel Code, Section VIII.

Care will be exercised to minimize damage to Category A and F components during removal, handling, and cleaning operations. Category B and C components are scrap items and require no special handling to preserve the equipment.

Criteria for sodium removal and cleaning operations are contained in Sections IV and V of this activity specification.

IV. METHOD OF ACCOMPLISHMENT AND SEQUENCE OF EVENTS

Sodium pumping apparatus shall be procured and utilized to pump the residual sodium heel from the reactor vessel. The depth of the residual sodium heel will be measured before and after pumping.

Primary sodium fill tanks will be removed, and the primary sodium service system will be modified in accord with Activity Specification No. 1, "Primary Sodium Disposal."

Each component for which an approved CH purchase order has been received will be disposed of in accord with its respective removal requirement.

It should be noted that category designations as described in Section II of this activity specification define the ultimate disposal of the item.

A. MAIN HEAT TRANSFER SYSTEM

1. The residual sodium heel in the reactor vessel will be pumped out to reduce the quantity of residual sodium to a minimum. The average depth of the sodium heel after pumping shall be approximately 1/4-in. or less.
2. The reactor vessel will be isolated from the primary system and steaming lines, and hardware will be installed as described in Section V.
3. The primary balancing legs will be removed from the system and disposed of a Category C items.
4. The IHX units will be isolated and sealed by cutting and capping all primary sodium lines leading to the IHX units inside the IHX cells. Cutting and capping will be accomplished so that an inert atmosphere is maintained within the IHX. Secondary sodium lines will be cut and capped as per Activity Specification No. 6. Ultimate disposal of the IHX units will be as Category A items.
5. The primary main heat transfer system 14- and 16-in. piping will be modified to permit in-place steaming reaction of residual sodium. The modification consists essentially of providing steam inlet lines from the steam mixing tank to be utilized in reactor vessel steaming), exhaust lines to the condenser (to be utilized in reactor vessel steaming), and of isolating the main loops from the Na service system by cutting and capping and installing a bypass line to replace the flow path formerly provided by the IHX units. The primary pumps have already been removed from the system. The primary pump cases will be modified to permit in-place steam cleaning of the 14- and 16-in. main piping. Sodium piping of lesser diameters (6, 4, 3, and 2-in.) that is not utilized for steaming activities will be removed from the primary pipe galleries and ultimately disposed of as Category B items.
6. The primary pumps, pump cases, and associated instrumentation and hardware will ultimately be disposed of as Category A items.
7. After modifications are complete, the main primary piping (14- and 16-in.) will be steam cleaned in-place one loop at a time by utilizing essentially the same procedure as that to

be employed in steaming the reactor vessel (Section V). The primary pump cases will ultimately be removed from the system and the 14- and 16-in. piping will be sealed while under a dry nitrogen atmosphere and disposed of as Category C (left-in-place).

8. The reactor vessel will then be steamed as described in Section V.

B. PRIMARY SODIUM SERVICE SYSTEM

1. Primary sodium service system hardware, components, and piping for which approved CH purchase orders are not received will be classified as Category B items and will be removed from the system with the following exceptions.

   a) Pipe sections which extend through vault walls and which are difficult to remove will be classified as Category C and left in place. Residual sodium present will be removed and/or reacted in the most expeditious and economic manner.

   b) Piping which contains relatively insignificant amounts of sodium will be classified as Category C items and left in place or stored nearby in accord with operational convenience and economic considerations. For purposes of determining whether residual sodium is present in "significant" quantities, the following procedure will be followed.

      1) The piping will be cut into convenient lengths so that internal surfaces may be visually inspected.

      2) If sodium is not present in layers exceeding 1/8-in. thickness, the piping will be classified as Category C. Sodium layers of 1/8-in. or less will react with moisture from the atmosphere within approximately 30 days. This procedure will be followed provided that the radioactivity of the residual sodium present does not present a problem from the contamination control standpoint. If contamination control presents a problem, the piping will be removed and disposed of as Category B material (with residual sodium being removed and/or reacted as necessary).

2. Residual sodium in the below-grade Category B primary sodium system items will be removed and/or reacted as necessary by utilizing the pump wash cell and/or auxiliary cleaning facilities. If residual sodium removal in a Category B item is technically or economically unfeasible, the item will be suitably sealed and/or packaged and shipped off site for disposal elsewhere.

V.* REACTION OF RESIDUAL SODIUM IN THE REACTOR VESSEL

Residual sodium in the Hallam Nuclear Power Facility reactor vessel will be reacted with a nitrogen-steam mixture. Significant considerations are summarized below under two headings: (A) System Modifications and Preliminary Tasks and, (B) Process Control and Instrumentation.

A. SYSTEM MODIFICATIONS AND PRELIMINARY TASKS

1. The residual sodium heel in the reactor vessel will be pumped out to reduce the quantity of residual sodium in the reactor vessel to a minimum. The average depth of the sodium heel after pumping shall be approximately 1/4-in. or less in depth.

2. The reactor vessel will be isolated from the main heat transfer system by cutting and capping the following lines on the primary pipeway side of the gallery seals:

*This section was previously designated Appendix I of this specification.

Quantity	Size (in.)	Material Type	Remarks
3	14	304	Na inlet
3	16	304	Na outlet
1	6	304	Moderator coolant inlet
3	3	304	One fill and drain and two vents
3	18	Carbon steel	Guard pipe
3	20	Carbon steel	Guard pipe
1	10	Carbon steel	Guard pipe
1	6	Carbon steel	Guard pipe for fill and drain

3. A steam-nitrogen mixing station to introduce the reaction fluid into the reactor vessel and exhaust lines and a condensation tank to exhaust reaction products to the reactor building stack will be designed and installed.

4. A central control and instrumentation panel will be designed and installed for monitoring and control of the process.

B. PROCESS CONTROL AND INSTRUMENTATION

1. General Description of Process

a. The reactor vessel and associated piping will be preheated (250 to 300°F) by utilizing existing procedures and equipment, and heated nitrogen flow.

b. Nitrogen-steam mixtures at 250 to 300°F will be introduced into the system for reaction purposes. The initial steam/nitrogen ratio will be zero (i. e., pure heated nitrogen will be introduced into the system). The steam-to-nitrogen ratio will be slowly increased and the following parameters monitored.

1) Oxygen content

2) Hydrogen content

3) Pressure

4) Selected reactor vessel temperatures

5) Pressure

6) Vessel effluent temperature

7) Flow rates (both inlet and exit).

Instrumentation will be provided so that mass flows in and out of the reactor vessel may be estimated on the basis of recorded data.

c. Steam flow rates, steam-nitrogen percentages, and reaction times to be utilized will be specified in the Detailed Procedures. These operational criteria will be based on experimental studies of the process conducted by Atomics International.

2. Normal Operating Conditions

Tentative values for the normal conditions under which the steaming process will be operated are listed below. These values are based on the studies of the process conducted by Atomics

International. The Detailed Procedures will contain more definitive data for the process as necessary.

a) Reactor vessel pressure, 1 to 6 psig

b) Reactor vessel atmosphere temperature, 250 to 450°F

c) Reactor vessel structure temperature (bottom of vessel) 250 to 550°F

d) Oxygen content of reactor vessel, trace quantities only

e) Nitrogen and steam mixture input to reactor vessel 250 to 300°F.

3. <u>Tentative Maximum and Minimum Values</u>

a) Maximum deliberate reactor vessel operating pressure to be utilized for steaming operations, 10 psig

b) Maximum reactor vessel pressure for flushing and venting operations (normal or emergency), 15 psig

c) Maximum reactor vessel oxygen content, 1/2%

d) Maximum vessel structure temperature (TC's on bottom of reactor vessel), 650°F

e) Minimum temperature of reactor vessel structure, 240°F

f) Maximum reactor vessel atmosphere temperature, 600°F

g) Minimum reactor vessel pressure, 1/2 psig.

The above maximums and minimums are to a greater or lesser extent arbitrary but are currently considered to be reasonable operational values on the basis of studies conducted to date. The above parameters will be more precisely defined as necessary in the Detailed Procedures. It is assumed that the operator will take corrective action when significant deviations from normal values are observed before the above values are reached. Routine operation somewhat outside the above limits is not necessarily hazardous and may be permitted provided there is an operational reason for such procedures and appropriate approval is granted.

In no event, however, will steaming operations be continued if 2% or greater oxygen concentrations are present, since such oxygen concentrations would very likely indicate a gross malfunction of the system. Pressures in excess of 25 psig are considered to be possibly damaging to the reactor bellows (rupture of the bellows does not present a hazard to personnel but would present other problems). Prolonged vessel atmosphere and structural temperatures in excess of 700°F are considered to represent an excessively fast rate of Na-steam reaction due to the use of an unreasonably high steam flow rate and/or steam/nitrogen ratio for the amount of sodium present. Reactor temperature should be maintained at 240°F or above to avoid any hazards due to possible steam condensation within the reactor vessel. Reactor pressure should be maintained at greater than atmospheric to preclude the possibility of air inleakage to the system.

4. <u>Alarms</u>

Tentative alarms and their setpoints are listed below. As discussed in Item 3 above, the setpoints are to a greater or lesser extent arbitrary, and final values and/or additional alarms will be contained in the Detailed Procedures.

a. High oxygen content in reactor vessel, 1/2%

b. High reactor vessel pressure, 2 psig above desired operating pressure

c. High reactor vessel atmosphere temperature, 550°F

d. High reactor vessel structural temperature, 650°F

e. Low reactor vessel structural temperature, 240°F

f. Low reactor vessel pressure, 1/2 psig

g. Automatic action high pressure alarm (4 psig above desired operating pressure)

h. High radiation effluent alarm.

All alarms shall give both visible and audible alarm signals at the central control and instrumentation panel.

5. Operator Action in Case of Alarm

Detailed operator actions, in case of alarm, will be contained in the Detailed Procedures. The general nature of operator action in case of alarm is briefly summarized below.

On high vessel oxygen content alarm, the operator shall cut off steam flow and commence nitrogen flushing operations until the oxygen content returns to normal values.

On high reactor vessel pressure alarm, high reactor vessel atmosphere temperature alarm, and high reactor vessel structural temperature alarm, the operator shall reduce steam and total flow rate as necessary to return conditions to normal. The automatic-action high-pressure alarm will automatically shut off both the steam and nitrogen supplies to the reactor vessel.

On low reactor vessel temperature alarm, the operator shall energize the reactor vessel heaters as necessary to return conditions to normal.

On low reactor vessel pressure alarm, the operator shall increase nitrogen flow rate as necessary to return conditions to normal.

On high radiation effluent alarm, the operator shall begin to reduce steam and nitrogen flow rate preparatory to shutting down steaming operations and an investigation will be immediately initiated as to the cause of the alarm. Release of quantities of radioactive material sufficient to cause a high radiation effluent alarm is considered to be an extremely unlikely occurrence, and instrument malfunction and/or operations elsewhere in the plant are considered to be more likely causes. If investigation indicates that the alarm is valid, the steaming process shall be shut down.

The occurrence of a valid high-oxygen alarm, high-radiation effluent alarm, or of a significant sodium reaction pressure-transient requires that, after immediate operator action is taken, the steaming operation be shut down and that an investigation of the causes of such occurrences be conducted. Steaming operations may be recommenced after the cause(s) of the observed phenomena have been determined, damage (if any) has been assessed, adequate provision made for prevention of future re-occurrence, and appropriate approval (to be specified in the Detailed Procedures) is obtained.

6. Reaction Completion

The reaction steaming process shall be performed in a manner such that there is a high degree of assurance, on the basis of studies conducted by Atomics International, that the reaction of residual sodium will be essentially complete. That is, reaction times utilized for specified steam percentages and flow rates shall be conservative in regard to the actual reaction times experimentally recorded in order to ensure reaction of the estimated thicknesses of residual sodium present. These minimum reaction times and other pertinent data will be specified in the Detailed Procedures. After the minimum reaction run times have been completed, the additional requirement that the hydrogen evolution must be shown to be essentially negligible over a prolonged period (e.g., 6 hr) prior to the cessation of steaming operations shall be made. Finally, sodium samples of various thicknesses shall be placed in the reactor vessel prior to the steaming operations and removed after the above requirements are met; thus, a direct experimental check on the efficacy of the steaming operations will be available.

After steaming operations have been completed, the steam supply will be cut off, the reactor vessel heaters will be energized as necessary, and heated nitrogen will be passed through the vessel to dry the NaOH which was formed during steaming. This will insure that the NaOH remaining in the vessel will be in solid form when the reactor is cooled to ambient temperature. Finally, the reactor vessel will be sealed and contain a dry nitrogen atmosphere such that a small positive gage pressure will be present under ambient conditions. The drying and sealing in of a nitrogen atmosphere reduces dry corrosion rates to negligible proportions.

SPECIFICATION 6.
REACTION OF RESIDUAL SECONDARY SODIUM AND
RETIREMENT OF SECONDARY SODIUM SYSTEM

I. PURPOSE

The purposes of this activity are to place the secondary sodium system in a condition so that residual sodium is in such a form that exposure to the anticipated environment will not reasonably result in a hazard to the HNPF Isolation Structure and to provide for ultimate disposal of secondary sodium system hardware and components. Where alternative disposal methods are stated in this specification, the steps employed shall be the most economical of those determined to be technically feasible.

II. ACTIVITY PLAN

For purposes of this activity specification, the secondary sodium system is assumed to include the secondary sodium main heat transfer system, the secondary sodium service system, and the sodium melt station including sodium Line 411 and Line 410 up to the removable spool piece.

All secondary sodium system components will be classified in categories and disposed of accordingly. Component category classifications refer basically to the ultimate method of disposal of the component. These category classifications are defined in Part II C-3 of the HNPF Retirement Plan and are summarized below.

Category A = Property authorized by CH for shipment off-site except as scrap. Such property will be removed from its present location and shipped off-site to AEC designated recipients.

Category B = Property which will be removed from its present location and shipped off-site as scrap.

Category C = Property which will remain in its present location at the HNPF.

Category D = Property which will be removed from its present location and placed in adequately contained subsurface volumes.

Category E = Property which will remain in its present location but whose ownership will be transferred to CPPD.

Category F = Property which must be removed to accomplish the retirement of the HNPF, and which CH authorizes to be removed from its present location pending determination of its ultimate disposition. Such property will be provided with suitable protection against damage and temporarily stored until its final status as Category A, B, or D material is decided.

Category A and F items will be removed from the system, and residual sodium present will be removed and/or reacted as necessary; the items will be sealed and will contain an inert atmosphere, if necessary, prior to ultimate disposal. Category B items will be removed from the system, and residual sodium present will be removed or reacted as necessary. Residual sodium in Category C items will be reduced by reaction and/or physical removal to the specified limits established for its safe containment.

III. CRITERIA

A. ENGINEERING, SCHEDULING, AND REQUIREMENTS

Operations connected with this activity will be scheduled in coordination with other HNPF Retirement activities; however, reaction of residual secondary sodium and retirement of the secondary system can be commenced immediately, since this activity is largely independent of other activities.

Major subcontractual work items, engineering studies, and equipment design and procurement to be conducted in support of this activity are listed below.

1. Detailed design, equipment procurement, and system modifications to permit the in-place steam cleaning of secondary system components as necessary
2. Subcontractual work in connection with design, equipment procurement, and existing hardware and structures modifications to permit the removal and/or reaction of residual sodium in Category A, B, and F items (It is recommended, on the basis of engineering feasibility and economic considerations, that auxiliary sodium cleaning facilities be provided. The pump wash cell will be used for sodium cleaning operations which can be economically and reliably performed by remote controlled steam cleaning. Auxiliary facilities will be required for the majority of sodium cleaning operations; e.g., manual removal, swabbing, water flush, heated oil bath, and manual steam cleaning.)
3. Heavy equipment removal subcontracts for such removal as is deemed necessary
4. Miscellaneous engineering studies, design, and equipment procurement in support of this activity.

B. HEALTH AND SAFETY

This activity will be carried out under normal HNPF and CPPD safety regulations. Sodium safety, sodium pipe cutting, sodium component handling practices, and radiological safety will be in accord with the HNPF Operations Manual, Volume I-a, Part III. Heavy equipment lifting and moving will be conducted in accord with the CPPD Safety Manual.

C. STANDARDS

Welding procedures, weldor qualifications, and inspection for seal-capping the cut piping of Category A and F items shall be in conformance with the ASME Boiler and Pressure Vessel Code, Section VIII.

Special care will be exercised to minimize damage to all Category A and F items, and their associated hardware and instrumentation during cutting, handling, cleaning, and disposal operations. Category B, C, and D components are scrap items and require no special handling criteria.

IV. METHOD OF ACCOMPLISHMENT AND SEQUENCE OF EVENTS

Secondary sodium fill tanks will be removed, and the secondary sodium service system will be modified in accord with Activity No. 2, Secondary Sodium Disposal.

Each component for which an approved CH purchase order has been received will be disposed of in accord with respective removal requirement. Ultimate disposal of Secondary Sodium System components shall be in accordance with (a) AEC Memo, RDT:D:A:174, March 20, 1967, Milton Shaw to K. A. Dunbar, and (b) AEC Memo, RDT:PE:A060, April 10, 1967, Milton Shaw to K. Dunbar. In the absence of an approved CH purchase order, disposal of secondary system components will proceed as listed below. It should be noted that category designations as described in Section II of this activity specification define the ultimate disposal of the item.

1. The IHX units will be isolated and sealed by cutting and capping all secondary sodium lines leading to the IHX units inside the IHX cells. Cutting and capping will be accomplished

such that an inert atmosphere is maintained within the IHX. Secondary sodium lines will also be cut outside the IHX cells to free the secondary sodium main heat transfer loop lines for disposal. Ultimate disposal of the IHX units will be as for Category A items.

2. The steam generators and associated hardware will be classified as Category A. Initially, each steam generator unit will be isolated and sealed, by cutting and capping its sodium inlet and outlet line. An inert gas atmosphere will be preserved during the cutting and capping operations. Further disposal operations will be in accordance with the steam generator removal requirements.

3. Below grade secondary sodium system hardware, components, and piping (with exception of the IHX units) will be classified as Category B items and will be removed from the system with the following exceptions.

   a) Pipe sections which extend through vault walls and which are difficult to remove will be classified as Category C and left in place. Residual sodium present will be removed and/or reacted in the most expeditious and economic manner.

   b) Piping which contains relatively insignificant amounts of sodium will be classified as Category C items and left in place or stored near by in accordance with operational convenience and economic considerations. For purposes of determining whether residual sodium is present in "significant" quantities, the following procedure will be followed.

      1) The piping will be cut into convenient lengths so that internal surfaces may be visually inspected.

      2) If sodium is not present in layers exceeding 1/8-in. thickness, the piping will be classified as Category C. Sodium layers of 1/8-in. or less will react with moisture from the atmosphere within 30 days.

4. Residual sodium in the below grade Category B secondary sodium system items will be removed and/or reacted by utilizing the pump wash cell and/or auxiliary cleaning facilities, with the following exceptions.

   a) Secondary sodium drain tank

   b) Secondary sodium cold trap.

   The above items may be cleaned on-site as per other Category B items or sealed and shipped off-site for disposal. The most economic of the two alternatives will be followed.

5. The secondary sodium system centrifugal pumps will be removed and steam cleaned. The secondary sodium system centrifugal pumps, pump cases, and associated hardware are presently classified as Category A.

6. Other above grade secondary sodium hardware components and piping (with exception of the steam generators, centrifugal sodium pumps, and cases previously discussed) will be disposed of as Category B or C items depending upon which alternative is economically advantageous.

SPECIFICATION 7.
DISPOSITION OF CONTAMINATED AND IRRADIATED MATERIAL

I. PURPOSE

The purpose of this activity is to provide for ultimate disposal of HNPF contaminated and irradiated material.

II. ACTIVITY PLAN

Radiological contamination is present in certain plant systems and components as a result of normal plant operations and the Moderator Removal Program. In addition, there are a number of HNPF irradiated components currently stored on-site which must be disposed of.

A. CONTAMINATED MATERIAL

Systems and equipment of which certain portions may be radiologically contaminated and which may require special decontamination effort are as follows.

1. Radioactive Liquid Waste System
2. Radioactive Vent System
3. Maintenance Cell
4. Equipment Decontamination Room Components
5. Fuel Handling Machine
6. Maintenance Cell Ventilation System
7. Portions of Heating and Ventilation System
8. Portions of Control Rod Drive Mechanisms
9. Surfaces of Vaults and Cells Which Are to be Sealed (Excluding Irradiated Component Disposal Areas)
10. Miscellaneous Components.

Decontamination and disposal of the above systems and equipment will be accomplished as outlined in Section IV of this activity specification.

B. IRRADIATED MATERIAL

Irradiated material currently on-site which must be disposed of is identified in the following three lists.

List 1

Control Elements (21 rod assemblies in 18 steel and 3 Zircalloy-2 thimbles)	21
Sources	2
Dummy Fuel Elements	40
Sodium Level Elements	2
Sodium Temperature Elements	3
Process Tubes	133
Shield Plug and Cut Hanger Rod	1
Shield Plug, Cut Hanger Rod, and Process Tube	6
Total	208

List 2

Shield Plug, Cut Hanger Rod, and Process Tube	10
M-3 Snorkle	1
Filler Plug, Process Tube, and Spacer Now in Storage Cells S275, S281, S289, and S290	4
Cut Dummy Elements	2
Lower Portion of TZ 6 and TZ 12 (Zircalloy Thimbles)	2
Zircalloy Thimbles	20
Filler Plug, Process Tube and Sleeve in S373	1
Total	40

List 3

Canistered Moderator Elements	3
Total	3

The above irradiated components (List 1, 2, and 3) will be permanently disposed of on-site in adequately contained subsurface volumes of the HNPF Isolation Structure as follows. List 1 material will be placed in the reactor vessel. The reactor vessel will be sealed as outlined in Activity 5, Reactor Isolation and Retirement. A brief analysis of possible neutron hazards resulting from the burial of the two HNPF Sb-Be neutron sources in the reactor vessel is given in Section VII of this activity specification.

List 2 materials will be placed in stainless steel thimbles in Fuel Storage Pit No. 3, and the thimbles will be sealed as described in Section IV of this activity specification.

List 3 materials are currently in Moderator Element Storage Pits. The pits will be sealed as described in Section IV of this activity specification.

III. CRITERIA

A. ENGINEERING SCHEDULING AND REQUIREMENTS

Operations connected with this activity will be scheduled in coordination with other HNPF Retirement activities. Disposal of List 3 materials (irradiated moderator cans) could be commenced immediately. Fuel bundles will be removed from process tubes under Activity No. 3, Disposal of Irradiated Fuel; such removal must of course, be accomplished before the associated Lists 1 and 2 process tubes can be disposed of. Decontamination operations depend on the retirement date for the particular component or system; and hence, will be scheduled in accord with the retirement of auxiliary systems under Activity No. 9, Retirement of Auxiliary Systems.

Major subcontractual work items, engineering studies, and equipment design and procurement to be conducted in support of this activity are as follows.

1. Detailed engineering studies and work in support of assessing the type, quantity, and associated hazards of radioactive material to be permanently disposed of on-site
2. Miscellaneous engineering studies and subcontractual work in support of this activity.

B. HEALTH AND SAFETY

On-site work connected with this activity will be carried out under normal HNPF and CPPD safety regulations. Radiological safety will be in accord with the HNPF Operations Manual, Volume Ia, Part III. Heavy equipment lifting and moving will be conducted in accordance with the requirements in the CPPD Safety Manual.

Radioactive contamination limits for components, hardware, and structures which are to be left in place outside the Isolation Structure or shipped off-site as scrap for unrestricted use are contained in Section V of this activity specification. Criteria for disposal of radioactively contaminated items within the HNPF Isolation Structure are contained in Section VI.

C. STANDARDS

Welding procedures, weldor qualifications, and inspection for the sealing of irradiated component storage areas shall be in conformance to the ASME Boiler and Pressure Vessel Code, Section VIII.

No material will be disposed of within the Isolation Structure which could create, by combustion or chemical reaction, a hazard to the containment integrity of the Isolation Structure.

IV. METHOD OF ACCOMPLISHMENT AND SEQUENCE OF EVENTS

A. CONTAMINATED MATERIAL DISPOSAL

Items for which an approved CH purchase order is received will be removed, decontaminated, and shipped in accordance with their respective removal requirements. If a CH purchase order is not received, HNPF hardware, components, and structures will be disposed of as follows.

1. The Radioactive Waste Facility, its associated hardware and components, and the tunnel connecting the Reactor Building and the Radioactive Waste Facility will be disposed of as per Activity 11, Retirement of the Radioactive Waste Facility.
2. The Reactor Building below grade structure, components, and hardware will remain in place and be disposed of in the HNPF Isolation Structure. The general procedure for disposal of such radioactive material is as follows.
   a) The type, quantity, and location of material to be disposed of will be recorded.
   b) The disposal of such material will be shown to be nonhazardous in accordance with the Part II criteria of the HNPF Retirement Plan and Section VI of this activity specification.
3. Contaminated above grade HNPF property in the Reactor Building or elsewhere (other than the R/A Waste Facility) will be surveyed to determine whether or not it is within the unrestricted use disposal limits contained in Section V of this activity specification. If an item is not contaminated in excess of the unrestricted use disposal limits, it will be disposed of as scrap for unrestricted use. If contamination is present in excess of the unrestricted use disposal limits, the item will be disposed of by one of the following methods.
   a) Decontamination to unrestricted use disposal limits in the most expeditious manner (provided that such decontamination may be performed using existing HNPF facilities) and disposed of as scrap

b) Buried within the HNPF Isolation Structure in accordance with Section VI of this activity specification

c) Shipped off-site for disposal elsewhere

4. Radioactive waste residue material produced by this activity will be (a) disposed of by dilution and discharge, and/or by packaging and shipment off-site in accordance with existing procedures of the HNPF Operations Manual, Volume Ia, Part III, or (b) disposed of in adequately contained subsurface volumes of the HNPF Isolation Structure in accord with Section VI criteria.

B. IRRADIATED MATERIAL DISPOSAL

1. List 1 irradiated components will be placed in the reactor vessel by utilizing HNPF Moderator Removal Program and normal operating procedures.
2. The reactor vessel and cavity will be sealed as described in Activity No. 8, Reactor Isolation and Retirement.
3. List 2 irradiated components will be placed in HNPF Storage Pit 3 thimbles by utilizing HNPF Moderator Removal Program and normal operating procedures, and the thimbles will be sealed.* The sealing method to be employed is as follows: the shield plug will be sealed to the thimble with an asphalt layer; next, the dust cover will be welded to storage cell casing; and, the Fuel Storage Pit 3 and the entire high bay floor will then be covered with an access barrier and weatherproof covering layer as described in Activity No. 10, Retirement of the Reactor Building.
4. List 3 components (3 moderator cans) are currently in three HNPF Moderator Can Storage Cells. The cells will be sealed† as follows: the shield plug will be seal-welded to the cell liner, and the dust cover will be seal-welded to the top flange of the cell liner; and the three Moderator Storage Cells and the entire High Bay floor will then be covered with an access barrier and weatherproof covering layer as described in Activity No. 10, Retirement of the Reactor Building.

V.§ RADIOACTIVE CONTAMINATION LIMITS TO PERMIT RELEASE OF AN ITEM FOR UNRESTRICTED USE

HNPF hardware, structures, and scrap which is to be disposed of so as to permit unrestricted use will be decontaminated to specified limits as outlined below. The specified contamination limits shall be in accordance with proposed Appendix A 10CFR20, Proposed Amendments to Establish Limits for Contamination on Premises, Equipment or Scrap to be Released by Licensees.

For reference purposes, Tables I and II of the above referenced document are attached.

*NAA-SR-MEMO-12270, "HNPF — Proposed Methods for Closing, Sealing, and Disposition of Structure," Figure 14, p 43
†loc. cit., Figure 15, p45
§This section was previously identified as Appendix I of this specification

RADIOACTIVE SURFACE CONTAMINATION LIMITS TO PERMIT RELEASE OF AN ITEM FOR UNRESTRICTED USE

Isotope[a]	Table I		Table II	
	Total[c]	Removable[b, c]	Total[c, e]	Removable[b, c]
U-Nat, U-235, U-238, Th-Nat, Th-232, and associated decay products	10,000 dpm $\alpha/100\ cm^2$	1,000 dpm $\alpha/100\ cm^2$	Average = 5,000 dpm $\alpha/100\ cm^2$ Maximum = 25,000 dpm $\alpha/100\ cm^2$	1,000 dpm $\alpha/100$ cm
Other isotopes which decay by alpha emission or by spontaneous fission	1,000 dpm $\alpha/100\ cm^2$	100 dpm $\alpha/100\ cm^2$	Average = 500 dpm $\alpha/100\ cm^2$ Maximum = 2,500 dpm $\alpha/100\ cm^2$	100 dpm $\alpha/100$ cm
Beta-gamma emitters (Isotopes with decay modes other than alpha emission or spontaneous fission)	0.4 mrad/hr at 1 cm[d]	1,000 dpm $\beta\gamma/100\ cm^2$	Average = 0.2 mrad/hr at 1 cm[d] Maximum = 1.0 mrad/hr at 1 cm[d]	1,000 dpm $\beta\gamma/100$ cm

a. Where surface contamination by both alpha and beta-gamma emitting isotopes exists, the limits established for alpha- and beta-gamma emitting isotopes shall apply independently.

b. The amount of removable radioactive material per 100 cm^2 of surface area shall be determined by wiping that area with dry filter- or soft absorbent paper and with the application of moderator pressure; and assessing the amount of radioactive material on the wipe with an appropriate instrument of known efficiency. In determining removable contamination on objects of lesser surface area, the pertinent levels shall be reduced proportionally, and the entire surface shall be wiped.

c. As used in this table, dpm (disintegrations per minute) means the rate of emission by radioactive material as determined by correcting the counts per minute observed by an appropriate detector and count rate meter, for background, efficiency, and geometry factors associated with the instrumentation.

d. Measured through not more than 7 mg/cm^2 of total absorber.

e. Measurements of total contaminant shall not be averaged over more than 10 m^2. For objects of lesser surface area, the average shall be derived for each such object.

Note: Either Table I or Table II may be used; for example, if all beta-gamma readings were less than 0.4 mrad/hr, material could be released under Table II providing the average was less than 0.2 mrad/hr.

VI.* DISPOSAL OF RADIOACTIVE MATERIALS BY PLACEMENT IN ADEQUATELY CONTAINED SUBSURFACE VOLUMES WITHIN THE HNPF ISOLATION STRUCTURE

A. GENERAL CRITERIA

The term "disposal" as used herein refers to the permanent disposition of a radioactive item by placement within an adequately contained subsurface volume of the HNPF Isolation Structure. General Disposal Criteria are contained in Section II, Criteria of the HNPF Retirement Plan and are summarized below.

The general criteria for adequacy of a subsurface volume for disposal of radioactive material at the HNPF are as follows.

1. Containment afforded by an adequately contained subsurface volume must be such that no failure in this containment which could occur by reasonable means would result in the release of a quantity of contained material sufficient to create a hazard to personnel in the immediate vicinity or to the general public over an indefinitely long period of time.
2. A disposal volume is considered adequate if it meets the general criteria in (1) above as substantiated by an engineering study. An engineering study will take into account
   a) The particular containment features of the volume,
   b) The type and quantity and relative hazard of radioactive material to be contained within the volume, and
   c) The mechanism by which the containment could be violated and the contained material released to the environment.

In addition to the above requirements, external radiation levels produced at the outer surfaces of the HNPF Isolation Structure, by materials contained therein must be in accordance with 10CFR20 regulations for an unrestricted area.

B. HNPF RADIOACTIVE MATERIAL DISPOSAL PROCEDURE

General Procedure

HNPF radioactive material to be left within the HNPF Isolation Structure will be treated as follows.

1. The type, quantity, and general location of such radioactive material will be recorded.
2. The general hazard imposed by the disposal of such material must be assessed and shown to be in accord with Part A, General Criteria of this section.

VII. DISPOSAL OF HNPF ANTIMONY-BERYLLIUM START-UP SOURCES IN THE HNPF REACTOR VESSEL

The start-up sources each originally contained approximately 436 gm of Sb_2O_4 encapsulated in stainless steel. The calculated saturated activity, at the maximum flux of 2×10^{13} n/cm^2-sec, of each source would be about 1400 curies of antimony-124. Since, at the present time, the reactor

*This section was previously identified as Appendix II of this specification.

has been shut down for 26 months, the maximum activity of each antimony source (60-day half life) would be 0.139 curies. The neutron yield for antimony-beryllium is 1.9×10^5 n/sec-curie; therefore, each source would produce a flux of approximately 2 n/cm^2-sec at 1 ft from the source if the source were placed in its beryllium sleeve at the present time. Neutrons are produced by a γ-n reaction with beryllium; the threshold energy of the reaction is 1.7 Mev. Although there is a significant quantity of gamma activity in the reactor vessel, the γ energies emitted are less than the threshold energy required for the γ-n reaction with beryllium. The gamma radiation level (including activity due to impurity activation in the source capsule) produced by the more active of the two sources is approximately 5 r/hr at 1 ft at the present time.

From the above data, it is concluded that permanent storage of the start-up sources in the reactor will not result in a hazardous situation.

SPECIFICATION 8.
REACTOR ISOLATION

I. PURPOSE

The purpose of this activity is to permanently seal and isolate the reactor vessel and cavity.

II. ACTIVITY PLAN

The sealed reactor vessel will be utilized for permanent containment of certain HNPF irradiated components. These components include 141 moderator elements and 75 reflector filler elements which are currently in the reactor. In addition, certain irradiated components which are currently on-site will be placed in the reactor vessel. These irradiated components are enumerated in List 1.

List 1

Control Elements (21 rod assemblies in 18 steel and 3 zircalloy-2 thimbles)	21
Sources	2
Dummy Fuel Elements	40
Sodium Level Instruments	2
Sodium Temperature Instruments	3
Shield Plug, Hanger Rod, and Process Tube	133
Shield Plug and Cut Hanger Rod	1
Shield Plug, Cut Hanger Rod, and Process Tube	6
Total	208

Reactor isolation and retirement will be accomplished as follows. Residual sodium will be pumped out and/or reacted as described in Activity Specification No. 5, Reaction of Residual Primary Sodium and Retirement of Primary Sodium Systems. Piping penetrating the reactor cavity will be cut, capped, and seal welded on the primary pipeway side of the gallery seals. Pupe hanger penetrations to the reactor cavity will be sealed. After List 1 irradiated components have been placed in the reactor, the Loading Face Shield (LFS) penetrations will be sealed by welding a carbon steel plate to the top of the existing LFS.

III. CRITERIA

A. ENGINEERING, SCHEDULING, AND REQUIREMENTS

Operations connected with this activity will be scheduled in coordination with other HNPF Retirement activities. Pipe hanger penetrations sealing and below-grade pipe capping activities are not primarily related to other retirement work and could be commenced immediately if desired. Final sealing of the LFS must necessarily be scheduled after residual sodium in the reactor vessel has been reacted and the List 1 components have been placed in the reactor. Prior to LFS sealing, therefore, the fuel bundles must be removed from all process tubes which are to be placed in the reactor (removal of fuel bundles will be accomplished under Activity 3, Disposal of Irradiated Fuel), and residual sodium in the reactor vessel must be removed and/or reacted (this will be accomplished under Activity 5, Reaction of Residual Primary Sodium and Retirement of Primary Sodium System).

The method of reactor isolation as outlined in Section IV of this specification was chosen as a result of an engineering analysis of the problems involved.*

*NAA-SR-MEMO 12263, "Study of Methods for Permanent Sealing of HNPF Reactor and Cavity"

Major subcontractual work items, engineering studies, and equipment design and procurement to be conducted in support of this activity are as follows:

1. Subcontractual work in connection with design, fabrication, handling, and installation of the LFS carbon steel cover plate,
2. Subcontractual work in connection with design, fabrication, handling, and installation of pipe hanger seal structures, and
3. Subcontractual work in connection with pipe cutting, sealwelding, and other miscellaneous work associated with this activity.

Detailed procedures shall include the following items: checkoff lists for all penetrations, the welding sequence for the loading face shield cover plate to minimize distortion, and detailed requirements for the pressure drop testing of the reactor vessel and reactor cavity.

B. HEALTH AND SAFETY

This activity will be carried out under normal HNPF and CPPD safety regulations. Sodium safety, sodium pipe cutting, sodium component handling practices, and radiological safety will be in accordance with the requirements of the HNPF Operations Manual, Volume I-a, Part III. Heavy equipment lifting and moving will be conducted in accordance with the CPPD Safety Manual.

C. STANDARDS

Welding procedures, weldor qualifications, and inspections for the plate covering the loading face shield, for seal capping cut piping, and for all electrical or instrumentation penetrations shall be in conformance to the ASME Boiler and Pressure Vessel Code, Sections VIII and IX. All weld passes shall be dye-penetrant or magnetic-flux checked to Section VIII criteria.

The sealed reactor vessel shall be gas tight and shall be capable of sustaining all normal atmospheric pressure swings without inducing stresses in any portion of its structure in excess of ASME Boiler and Pressure Vessel Code, Section VIII allowables; in addition, it shall be capable of sustaining the maximum pressure drop associated with the eye of a tornado without rupturing. Helium leak checks will be made on seal welds. The sealed LFS shall be capable of supporting an AASHO H-20 wheel loading in addition to its own dead weight.

The structural capabilities of reactor cavity penetration seals provided under this activity specification to withstand external and internal pressure shall be at least equivalent to that of the presently existing reactor cavity structure.

IV. METHOD OF ACCOMPLISHMENT AND SEQUENCE OF EVENTS

1. Residual sodium in the reactor vessel will be removed and/or reacted in accordance with Activity Specification No. 5.
2. The following pipe lines will be cut and capped on the primary pipeway side of the gallery seals:

Quantity	Size (in.)	Material Type	Remarks
3	14	304	Na inlet
3	16	304	Na inlet
1	6	304	Moderator coolant inlet
3	3	304	One filled and drain and two vents
3	18	Carbon steel	Guard pipe
3	20	Carbon steel	Guard pipe
1	10	Carbon steel	Guard pipe
1	6	Carbon steel	Guard pipe for fill and drain
1	2	Carbon steel	Containment vessel drain
1	2	Carbon steel	Cavity vent and drain
1	2	Carbon steel	Cavity vent

3. The 17 pipe hanger penetrations into the reactor cavity will be sealed by permanently supporting the pipe hanger rod and by replacing the bellows with a welded pipe cap.*

4. Electrical and instrumentation penetrations to the reactor vessel and cavity will be closed by seal welding.

5. Plan List 1 irradiated components will be placed in the reactor utilizing existing HNPF procedures. This will be accomplished under Activity 7 - Disposition of Contaminated and Irradiated Material.

6. A 1/2-in. thick carbon steel plate will be placed on the top of the LFS and attached to the LFS in accordance with Figure 2 of the Op. Cit. in Section III A. The plate and the cerrobend seal structure will be coated with a bituminous sealing material applied at a temperature below the melting point of the cerrobend.

7. The sealed reactor-vessel reactor-cavity complex will be pressure drop tested at 2 psig. Allowable leak rates for the reactor vessel and cavity are 1/2 scfm and 1 scfm respectively. The reactor vessel and cavity will be permanently sealed and will contain a dry nitrogen atmosphere.

8. The below operating floor level area around the LFS and the reactor services pipe trenches will be filled with concrete to the operating floor level of 1440 ft 6 in.

9. As a final step, the LFS and the entire High Bay floor will be covered with an access seal and weatherproof covering layer. This access seal and weatherproof covering layer will be specified in Activity 10, Securing of Isolation Structure.

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*See p 5 of Op. Cit. in Section III A.

SPECIFICATION 9.
RETIREMENT OF AUXILIARY SYSTEMS

I. PURPOSE

The purpose of this activity specification is to provide for the deactivation of the HNPF auxiliary systems and disposal of auxiliary systems hardware and components.

II. ACTIVITY PLAN

Auxiliary systems are considered to be all HNPF systems exclusive of the Sodium Heat Transfer System and the Sodium Service Systems.

A listing of the auxiliary systems follows.

1) Main Steam and Feedwater Systems and Emergency Feedwater System
2) Chemical Feed Systems
3) Reactor Control and Protective Systems
4) Radioactive Liquid Waste System
5) Radioactive Vent System
6) Preheat System
7) Cooling Water System
8) Helium System
9) Nitrogen System
10) Auxiliary Steam Systems
11) LFS Nitrogen Cooling System
12) Emergency Power System (including Diesel Load Bank)
13) Radiation Detection and Monitoring System
14) Heating and Ventilating Systems
15) Compressed Air Systems
16) Fire Detection and Protection System
17) Fire Fighting Apparatus
18) Electrical System
19) Liquid Nitrogen Tank and System
20) Building Communication and Alarm Systems
21) Miscellaneous Building Service Systems (e.g., water, sewerage, drainage)
22) Fire Protection Water System External to Reactor Building

Auxiliary system hardware and components will be classified in categories and disposed of as per the category definitions in Part II-C3 of the HNPF Retirement Plan. The Category definitions are summarized below.

Category A = Property authorized by CH for shipment off-site except as scrap. Such property will be removed from its present location and shipped off-site to AEC designated recipients.

Category B = Property which will be removed from its present location and shipped off-site as scrap.

Category C = Property which will remain in its present location at the HNPF.

Category D = Property which will be removed from its present location and placed in adequately contained subsurface volumes.

Category E = Property which will remain in its present location but whose ownership will be transferred to CPPD.

Category F = Property which must be removed to accomplish the retirement of the HNPF, and which CH authorizes to be removed from its present location pending determination of its ultimate disposition. Such property will be provided with suitable protection against damage and temporarily stored until its final status as Category A, B, or D material is decided.

Retirement of auxiliary systems will consist of disconnecting electrical power, draining of any fluids present, radiological decontamination of hardware and components (where necessary), and disposal of hardware and components.

III. CRITERIA

A. ENGINEERING, SCHEDULING, AND REQUIREMENTS

Disposition of the auxiliary systems will follow a sequence compatible with the requirements of the entire Retirement Program; auxiliary systems shall remain operable as necessary to ensure safety in carrying out subsequent operations.

Portions of auxiliary systems and/or specific hardware and components may be deactivated and disposed of whenever their particular services are no longer required for Retirement Program operations, provided that such deactivation and/or disposal will not be inimical to radiological or industrial safety.

The following general considerations apply in regard to deactivation sequences.

1) The main steam and feedwater systems, the emergency feedwater systems, the chemical feed systems, and the reactor control and protective systems may be retired at any convenient time commencing immediately (portions of these systems have already been retired).

2) The helium system will be modified for a nitrogen supply; hence, retirement of existing helium system hardware will coincide with the final retirement of the nitrogen system. After retirement of the nitrogen system, the Liquid Nitrogen System tank and associated apparatus will be returned to the vendor. Retirement of the nitrogen system shall be in accordance with the stipulations in Item 5 below.

3) The LFS nitrogen cooling system may be retired whenever it is no longer required (appropriate modifications to this system have already commenced).

4) The sodium components Preheat System may be retired after completion of sodium cleaning and removal operations in Activities 5 and 6.

5) Other systems listed in Section II of this specification shall remain operable until all operations involving radiological hazards and sodium cleaning and handling hazards are

completed. For practical purposes, this stipulation requires that Activities 1 through 8 and necessary decontamination operations of the R/A Vent and Liquid Systems under Activity 11 be completed. As previously stated, portions of auxiliary systems may be retired at any convenient earlier date, provided that it is clearly established that such retirement will not be inimical to radiological and industrial safety.

6) Retirement of portions of systems contained in the Radioactive Waste (R/A) Facility is described in Activity 11. Retirement of the R/A Waste Facility systems shall be scheduled in accordance with the stipulations of Item 5 above.

7) The retirement schedule for systems listed below shall conform to the stipulations of Item 5 above and; these systems shall remain in operation until Isolation Structure penetration closure and protective covering placement operations under Activity 10 necessitate their retirement.

   a) Electrical lighting and appliance plug connections (Note: The Emergency Power System may be retired at an earlier date in accord with the safety stipulations of Item 5 above.)

   b) Building Communication and Alarm System, including evacuation sirens

   c) Portions of Heating and Ventilation (H&V) System as necessary to secure a reasonable working environment for retirement operations (This also requires that appropriate portions of subsidiary systems serving the H&V system; e.g., cooling water, auxiliary steam and compressed air, remain operable. This requirement is strictly included for "creature comfort" purposes and should not be construed in terms of any former criteria in regard to radiological or industrial hazards.)

   d) Fire Detection System, excluding any former automatic protective actions

   e) Fire fighting apparatus

   f) The fire protection water system external to the Reactor Building

   g) Miscellaneous Building Service Systems; e.g., water, sewerage, drainage

Systems enumerated in a) through g) above constitute the last auxiliary systems to be deactivated and disposed of under the HNPF Retirement Program.

Subcontractual work items, engineering studies, and equipment design and procurement to be conducted in support of this activity are listed below.

Miscellaneous minor studies and/or investigations will be conducted in support of this activity. This activity is almost entirely concerned with shift operations and maintenance activities which have been more or less routinely performed at the HNPF; hence, no special preliminary engineering studies or design efforts are deemed necessary.

B. HEALTH AND SAFETY

On-site work connected with this activity will be carried out under normal HNPF and CPPD safety regulations. Radiological safety will be in accord with the HNPF Operations Manual, Volume Ia, Part III. Any necessary heavy equipment lifting and moving will be conducted in accordance with the CPPD Safety Manual.

Radiological decontamination operations and radioactive material disposals shall be carried out in accordance with Activity Specification No. 7 for Reactor Building systems and Activity No. 11 for Radioactive Waste Facility Systems.

C. STANDARDS

No special nonradiological standards are deemed necessary for this activity. Work connected with this activity shall be conducted in accordance with good engineering practice and existing procedures.

IV. METHOD OF ACCOMPLISHMENT AND SEQUENCE OF EVENTS

Retirement of auxiliary systems consists of disconnecting electrical power supplies; draining of any fluids present; radiological decontamination of hardware and components, where required; and final disposition of hardware and components.

Disconnecting of electrical power supplies and draining of fluids present are routine operations at the HNPF and will be performed in accordance with existing operational procedures (HNPF Operations Manual, Volumes I through IV).

Retirement of auxiliary systems shall follow a sequence compatible with health, safety, and the requirements of the Retirement Program. Scheduled retirement dates for individual auxiliary systems, or portions thereof, shall be in accordance with the criteria and sequences given in Part III-A of this specification.

Radiological decontamination and radioactive material disposal operations shall be conducted in accordance with Part III-B of this specification.

All HNPF property for which approved CH purchase orders are received will be designated as Category A and will be disposed of in accordance with its respective removal requirements. In the absence of CH approved purchase orders, auxiliary systems hardware and components shall be disposed of as follows.

a) Disposal of Radioactive Waste Facility hardware and components shall be in accordance with Activity Specification No. 11.

b) A relatively small amount of HNPF property is located north of Column Line 13 (e.g., emergency turbine and pump, HNPF boiler chemical feed tanks, closed loop cooling water head tank, electrical equipment, and switchgear). In addition, certain auxiliary system lines (e.g., compressed air, main and auxiliary steam, HNPF boiler feedwater, and cooling water) connect the nuclear and conventional plants. The disposition of these items will to a greater or lesser extent affect CPPD operation of their conventional plant. The disposition of HNPF property residing north of Column Line 13 (if CH approved purchase orders are not received for such property) and the modification (north of Column Line 13) of lines crossing the Column Line 13 interface shall be considered the subject for separate negotiations between the Commission and CPPD, and are therefore not included in this Activity Specification. In this regard, it should be noted that a significant number of the above items have already been disposed of via such separate negotiation.

c) Auxiliary systems equipment and hardware south of Column Line 13 which are currently located in volumes of the proposed Isolation Structure will be classified as Category C and will remain in place in their present location.

d) Auxiliary systems equipment and hardware south of Column Line 13 which are currently located outside the proposed Isolation Structure (i.e., above operating floor levels) shall be considered as part of the Reactor Building structure. These will be disposed of in accordance with the results of the engineering feasibility and economic advantage studies pertaining to the Reactor Building structures to be conducted under Activity 10. Specifically, such hardware will be classified as Category C and remain in place if removal of the building structure to which it is attached is not economically advantageous to the Commission and/or such removal is not necessary to render the premises radiation safe. If building structure removal is found to be economically advantageous to the Commission, or necessary for rendering the site radiation safe, such attached auxiliary systems hardware and components shall be classified as Category B (scrap) and will be disposed of during structures demolition and/or modification.

SPECIFICATION 10.
SECURING OF ISOLATION STRUCTURE

I. PURPOSE

The purpose of the Activity Specification is to provide for final closure and sealing of all penetrations to the HNPF Isolation Structure, and criteria for the weather and access proof covering to be placed over exposed surfaces of the Isolation Structure.

II. ACTIVITY PLAN

The HNPF Isolation Structure is defined as the closed concrete structure formed by the IHX cells structure and the Reactor Building vault structures after all accesses and penetrations have been closed and the exposed surfaces weatherproofed. The HNPF Isolation Structure includes all below grade cells in the Reactor Building.

HNPF hardware structures and components (other than structures which form part of the Isolation Structure) will be classified in categories and disposed of accordingly. Category classifications are contained in Part II, C-3 of the HNPF Retirement Plan and are summarized below.

Category A = Property authorized by CH for shipment off-site except as scrap. Such property will be removed from its present location and shipped off-site to AEC designated recipients.

Category B = Property which will be removed from its present location and shipped off-site as scrap.

Category C = Property which will remain in its present location at the HNPF.

Category D = Property which will be removed from its present location and placed in adequately contained subsurface volumes.

Category E = Property which will remain in its present location but whose ownership will be transferred to CPPD.

Category F = Property which must be removed to accomplish the retirement of the HNPF, and which CH authorizes to be removed from its present location pending determination of its ultimate disposition. Such property will be provided with suitable protection against damage and temporarily stored until its final status as Category A, B, or D material is decided.

Work items outlined in Activity Specifications 1 through 9 which are pertinent to a particular subsurface volume of the Isolation Structure shall be performed, and any Category D items to be disposed of shall be placed in the volume prior to the final sealing of the volume. All hardware, components, and structures which would interfere with the selected method of sealing of the Isolation Structure shall be removed and disposed of. Penetrations to the Isolation Structure will be sealed, and a protective covering against access and the elements shall be placed over all exposed operating floor surfaces of the HNPF Isolation Structure.

III. CRITERIA

A. ENGINEERING, SCHEDULING, AND REQUIREMENTS

Prior to the sealing of penetrations to a given volume within the HNPF Isolation Structure, the following shall be accomplished.

1) Work items of Activities 1 through 9 pertinent to that volume shall be completed.

2) Any Category D items to be disposed of in that volume shall be placed therein.

3) Any necessary radiological decontamination of components on surfaces in that volume shall be performed in accord with Activity Specification No. 7 criteria.

4) The final radiological status of the volume shall be recorded.

After the above have been accomplished, penetrations to the Isolation Structure will be sealed; an access and weatherproof protective covering will be placed over exposed horizontal surfaces of the Reactor Building floor structure and the IHX cells structure, and the covering will be suitably integrated to these structures to meet the access and weatherproof requirements specified in Part C of this section. Details concerning the proposed nature of the weather and access proof protective covering, and the proposed nature of sealing penetrations, have been presented elsewhere.[a] General criteria for the protective covering and penetration seals are listed in Part C of this specification. Further detailed design of the protective covering and penetrations seals will be carried out as outlined in subsequent paragraphs of this section of this specification. Prior to the placement of the protective covering, a determination shall be made as to what portion of the existing Reactor Building structure must be demolished, if any, in order to render the premises radiation safe in the most feasible and economic fashion. For this determination, the following ground rules shall apply.

1) The protective covering and penetration closures selected must meet the radiological and engineering criteria of Parts B and C of this section.

2) Existing above grade Reactor Building structures shall be removed only if such removal is economically advantageous to the Atomic Energy Commission and/or necessary for rendering the premises radiation safe.

3) The removal of the building shall be deemed economically advantageous if any of the following three evaluations are found to result in a net financial gain to the Commission:

   a) The cost for removing the structure vs any gain from resulting scrap sales,

   b) The difference in cost of the penetration seals and protective covering required to meet the criteria of Parts B and C which would be incurred if the structure remains in place rather than being removed, and

   c) Any other economic advantages and/or monetary reimbursements which the Commission may accrue if the Reactor Building structures are removed and/or modified in some specified manner.

4) For purposes of determination of the value of the Reactor Building and its associated contents as scrap to a prospective demolition or modification subcontractor, the following definition of the Reactor Building structure shall apply.

The Reactor Building structure is considered to consist of all above grade structures and hardware south of Column Line 13 with exception of the following: the HNPF Isolation Structure itself as defined in Section II of this specification, any items on AEC excess property lists for which CH approved purchase orders are received, and any other property whose disposal has been specified in Activity Specifications 1 through 9.

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[a]NAA-SR-MEMO-12263, "Study of Methods for Permanent Sealing of Reactor and Cavity," and NAA-SR-MEMO-12270, "HNPF-Proposed Methods for Closing, Sealing, and Disposition of Structures"

Major subcontractual work items, engineering studies, equipment design, and procurement to be conducted in support of this activity are:

1) Economic and engineering feasibility determinations in regard to the advantages or disadvantages of Reactor Building structures removal and/or restoration as outlined above, including the collection of bids from subcontractors on the cost for desired demolition and/or modification activities,

2) Detailed design of Isolation Structure penetration closures,

3) Detailed design of the HNPF Isolation Structure protective covering, and

4) Heavy equipment moving, modification and/or demolition subcontracts as necessary in accord with the results of the economic and engineering determinations in Item (1) above.

B. HEALTH AND SAFETY

Operations connected with this activity will be carried out under normal HNPF and CPPD safety regulations. Radiological safety will be in accordance with the HNPF Operations Manual Volume I-a, Part III and 10CFR20 regulations. Necessary hardware lifting and moving will be conducted in accordance with the CPPD Safety Manual.

Radiological decontamination and radioactivity disposal operations shall be in accordance with Activity Specification No. 7.

The weatherproof protective covering of the HNPF Isolation Structure shall have a design life objective of preventing access to the contained disposal volumes for a period of time necessary to reasonably assure that in the event of failure of the Isolation Structure at some time beyond its design life no radiological hazard will result which will be inimical to the health and safety of the public.

C. STANDARDS

Sealing of the reactor vessel and cavity shall be in accordance with Activity Specification No. 8.

The integrity of containment provided by penetration seals shall be at least equivalent to that provided by the adjacent Isolation Structure.

The Isolation Structure shall be capable of carrying all structural loads which can be imposed by reasonable means.

The weatherproof protective covering of the Isolation Structure shall have a design life objective of 100 years. Closure of the penetrations and accesses shall be designed to prevent personnel access except as might be gained through extensive use of equipment such as jack hammers, pneumatic drills, cutting torches, and explosives. Adequate drainage of exposed surfaces of the Isolation Structure shall be provided.

IV. METHOD OF ACCOMPLISHMENT AND SEQUENCE OF EVENTS

1) Prior to the sealing of penetrations to a given volume within the HNPF Isolation Structure the following shall be accomplished.

a) Work items of Activities 1 through 9 pertinent to that volume shall be completed.

b) Any Category D items to be disposed of in that volume shall be placed therein.

c) Appropriate radiological surveys will be conducted, and any necessary radiological decontamination of components or surfaces in that volume shall be performed in accordance with Activity Specification No. 7 criteria.

d) The final radiological status of the volume shall be recorded.

2) Penetrations to the Isolation Structure will be sealed in accordance with the criteria in Part III-C of this specification.

3) Reactor Building structures removal and/or modification necessary to permit installation of the access and weatherproof protective covering will be initiated and the protective covering will be installed and integrated to the designated concrete structures to produce the completed Isolation Structure which will meet the criteria of Parts III-B and III-C of this specification.

4) Further removal and/or modification of Reactor Building structures will be performed in accord with the results of engineering feasibility and economic studies performed as outlined in Part III-A of this specification. In order for such removal and/or modifications to be authorized, the above studies must show that the proposed operations are an engineering necessity to render the HNPF site radiation safe and/or such operations will be economically advantageous to the Atomic Energy Commission.

SPECIFICATION 11.
RETIREMENT OF THE RADIOACTIVE-WASTE FACILITY

I. PURPOSE

The purpose of this activity is to provide for the decontamination and disposition of components and materials associated with the HNPF Radioactive Waste Facility.

II. ACTIVITY PLAN

A tunnel containing radioactive vent- and liquid-system piping extends below grade from the Reactor Building to the Radioactive Waste Facility Building vaults. The Radioactive Waste Facility contains portions of the Radioactive Vent- and Liquid-Systems hardware, piping, and components both above grade in the building, and below grade in the vaults. Decontamination operations will be carried out as described in Section IV of this activity specification to achieve the limits specified in Section V. Disposal operations within the HNPF Isolation Structure will be carried out in accordance with the criteria in Section VI of Activity Specification No. 7. Radioactive Waste Facility components and materials will be classified in categories and disposed of in accordance with the HNPF Retirement Plan category classification system which is summarized below.

Category A = Property authorized by CH for shipment off-site except as scrap. Such property will be removed from its present location and shipped off-site to AEC designated recipients.

Category B = Property which will be removed from its present location and shipped off-site as scrap.

Category C = Property which will remain in its present location at the HNPF.

Category D = Property which will be removed from its present location and placed in adequately contained subsurface volumes.

Category E = Property which will remain in its present location but whose ownership will be transferred to CPPD.

Category F = Property which must be removed to accomplish the retirement of the HNPF, and which CH authorizes to be removed from its present location pending determination of its ultimate disposition. Such property will be provided with suitable protection against damage and temporarily stored until its final status as Category A, B, or D material is decided.

III. CRITERIA

A. ENGINEERING, SCHEDULING, AND REQUIREMENTS

Operations connected with this activity will be scheduled in coordination with HNPF Retirement activities.

The Radioactive Liquid Waste and Vent Systems will be retired under Activity No. 9, Retirement of Auxiliary Systems, in accordance with the updated HNPF Retirement schedule. Generally speaking, other major decontamination operations and radioactive materials disposition must be completed prior to the decontamination and dismantling of components of the Radioactive Liquid Waste and Vent Systems contained within the Radioactive Waste Facility.

The most economic alternative, where technically feasible alternatives exist, will be used for disposal of a given item.

B. HEALTH AND SAFETY

Operations connected with this activity will be carried out under normal HNPF and CPPD safety regulations. Heavy equipment lifting and moving will be conducted in accordance with the CPPD Safety Manual. Radiological safety will be in accordance with the HNPF Operations Manual, Volume Ia, Part III. Those major components, for which it is uneconomical to meet the limits of Section V, will be evaluated by AI on a case basis to determine their ultimate disposition, with AEC approval. Radioactive contamination limits for components which are to be left in place or shipped off-site as scrap for unrestricted use are contained in Section V of this activity specification. Criteria for disposal of radioactively contaminated items within the HNPF Isolation Structure are contained in Section VI of Activity Specification Number 7.

C. STANDARDS

Operations connected with this activity will be conducted in accordance with good engineering practice. No special nonradiological standards are deemed necessary.

IV. METHOD ACCOMPLISHMENT AND SEQUENCE OF EVENTS

Items for which approved CH purchase orders are received (Category A items) will be disposed of in accordance with their respective removal requirements. Remaining items will be disposed of as follows.

1) Piping in the pipe tunnel connecting the Reactor Building with the Radioactive Waste Facility will be:

   a) Externally and internally decontaminated to the unrestricted use disposal limits (Appendix I) and disposed of as Category C items, or

   b) Disposed of as Category D items in the HNPF Isolation Structure in accord with the criteria of Section VI of Activity Specification No. 7 if in-place decontamination is economically unfeasible.

2) Pipe tunnel surfaces will be decontaminated to the unrestricted use disposal limits, see Section V. The pipe tunnel will be disposed of as a Category C item.

3) The tunnel penetration to the Reactor Building (HNPF Isolation Structure) will be sealed and weatherproofed in accord with Activity Specification No. 10, Securing of the Isolation Structure.

4) Above grade Radioactive Waste Facility hardware and components will be:

   a) Decontaminated to the unrestricted use disposal limits, Section V, and disposed of as Category B items, or

   b) Disposed of as Category D items in the HNPF Isolation Structure in accord with the criteria of Section VI of Activity Specification No. 7 if decontamination is economically unfeasible.

5) Below grade Radioactive Waste Facility hardware and components will be:

a) Decontaminated to the unrestricted use disposal limits, Section V, and disposed of as Category C items, or

b) Disposed of as Category D items in the HNPF Isolation Structure in accord with Section VI criteria if decontamination is economically unfeasible, or

c) Decontaminated to the unrestricted use disposal limits, Section V, and disposed of as Category B items if such disposal is economically advantageous to the AEC.

6) Above grade Radioactive Waste Facility Structures will be decontaminated to the unrestricted use disposal limits, Section V, with the additional restriction that radiation levels at exposed surfaces shall in accordance with 10CFR20 requirements for an unrestricted are and will be:

a) Disposed of as Category C items, or

b) Disposed of as Category B items if such disposal is economically advantageous to the AEC.

7) Radioactive waste residue material produced by this activity will be (a) disposed of by dilution and discharge and/or packaged and shipped off-site in accord with existing procedures of the HNPF Operations Manual, Volume Ia, Part III, or (b) disposed of in adequately contained subsurface volumes of the HNPF Isolation Structure in accord with the criteria of Section VI of Activity Specification No. 7.

V. RADIOACTIVE CONTAMINATION LIMITS TO PERMIT RELEASE OF AN ITEM FOR UNRESTRICTED USE

HNPF hardware, structures, and scrap which is to be disposed of so as to permit unrestricted use will be decontaminated to specified limits as outlined below. The specified contamination limits shall be in accordance with proposed Appendix A, 10CFR20, Proposed Amendments to Establish Limits for Contamination on Premises, Equipment or Scrap to be Released by Licensees.

For reference purposes, Tables I and II of the above referenced document are attached.

RADIOACTIVE SURFACE CONTAMINATION LIMITS TO PERMIT RELEASE OF AN ITEM FOR UNRESTRICTED USE

Isotope[a]	Table I		Table II	
	Total[c]	Removable[b, c]	Total[c, e]	Removable[b, c]
U-Nat, U-235, U-238, Th-Nat, Th-232, and associated decay products	10,000 dpm α/100 cm^2	1,000 dpm α/100 cm^2	Average = 5,000 dpm α/100 cm^2 Maximum = 25,000 dpm α/100 cm^2	1,000 dpm α/100 cm
Other isotopes which decay by alpha emission or by spontaneous fission	1,000 dpm α/100 cm^2	100 dpm α/100 cm^2	Average - 500 dpm α/100 cm^2 Maximum - 2,500 dpm α/100 cm^2	100 dpm α/100 cm
Beta-gamma emitters (Isotopes with decay modes other than alpha emission or spontaneous fission)	0.4 mrad/hr at 1 cm[d]	1,000 dpm $\beta\gamma$/100 cm^2	Average = 0.2 mrad/hr at 1 cm[d], Maximum = 1.0 mrad/hr at 1 cm[d]	1,000 dpm $\beta\gamma$/100 cm

a. Where surface contamination by both alpha and beta-gamma emitting isotopes exists, the limits established for alpha- and beta-gamma emitting isotopes shall apply independently.

b. The amount of removable radioactive material per 100 cm^2 of surface area shall be determined by wiping that area with dry filter- or soft absorbent paper and with the application of moderator pressure; and assessing the amount of radioactive material on the wipe with an appropriate instrument of known efficiency. In determining removable contamination on objects of lesser surface area, the pertinent levels shall be reduced proportionally, and the entire surface shall be wiped.

c. As used in this table, dpm (disintegrations per minute) means the rate of emission by radioactive material as determined by correcting the counts per minute observed by an appropriate detector and count rate meter, for background, efficiency, and geometric factors associated with the instrumentation.

d. Measured through not more than 7 mg/cm^2 of total absorber.

e. Measurements of total contaminant shall not be averaged over more than 10 m^2. For objects of lesser surface area, the average shall be derived for each such object.

Note: Either Table I or Table II may be used; for example, if all beta-gamma readings were less than 0.4 mrad/hr, material could be released under Table II providing the average was less than 0.2 mrad/hr.

SPECIFICATION 12.
FINAL CLOSEOUT OF THE FACILITY

I. PURPOSE

The purpose of this activity is to provide for the disposition of HNPF structures and components located outside the Reactor Building and Radioactive Waste Facility, to provide for the disposal of loose debris still present on the HNPF site as a result of Retirement activities, and to conduct a final radiation survey of the site prior to termination of the Retirement Program.

II. ACTIVITY PLAN

In addition to the Radioactive Waste Facility, there are four HNPF structures which reside outside the Reactor Building:

1) Fuel and Moderator Assembly Building,
2) Storage Building south of Fuel and Moderator Assembly Building,
3) Steam Dump Facility, and
4) Calibration Building.

The above structures and their contents will be classified and disposed of in accordance with the HNPF Retirement Plan category classification system which is reproduced below.

Category A = Property for which approved AEC-CH purchase orders are received. Such property will be removed from its present location and shipped off-site to AEC designated recipients.

Category B = Property which will be removed from its present location and shipped off-site as scrap.

Category C = Property which will remain in its present location at the HNPF.

Category D = Property which will be removed from its present location and placed in sub-surface volumes of the HNPF Isolation Structure.

Category E = Property which will remain in its present location but whose ownership will be transferred to CPPD.

Category F = Property which must be removed to accomplish the retirement of the HNPF, and which CH authorizes to be removed from its present location pending determination of its ultimate disposition. Such property will be provided with suitable protection against damage and temporarily stored until its final status as Category A, B, or D material is decided.

The structures of Items 1 through 4 will be disposed of as Category B, C, or E items. Prior to final disposal, a radiological survey of the structures will be conducted and decontamination will be conducted and decontamination will be performed, as necessary, to achieve the following.

1) If a structure is classified as Category B, radiation levels shall be in accord with the unrestricted scrap disposal limits of Section V of Activity Specification No. 7.

2) If a structure is classified as Category C or E, the radiation levels shall be in accord with 10CFR20 requirements for an unrestricted area.

After the structures have been decontaminated, as necessary, to meet the above criteria, the method of disposal of the structures will no longer be of concern in regard to rendering the HNPF premises safe from a radiation standpoint; hence, in general, economic considerations will govern. Structures will be classified as Category C unless economic analyses show that disposal as Category B or E would be economically advantageous to the Atomic Energy Commission. For purposes of determination of economic advantage to the Commission, the structures may be considered individually or as part of any general scrap disposal contracts for Category B items or property transfers to CPPD as Category E items which may be conducted under Activity Specifications No. 9 and 10 or other Retirement activities.

Any loose debris still left on site as a result of HNPF Retirement activities will be disposed of.

A final radiological survey of the site will be conducted.

III. CRITERIA

A. ENGINEERING, SCHEDULING AND REQUIREMENTS

Operations connected with this activity will be scheduled in coordination with other HNPF Retirement activities.

Steam Dump Facility retirement can be commenced immediately if desired.

The Fuel and Moderator Assembly Building can be retired as soon as sodium component cleaning (carried out under Activities 5 and 6) has been completed. The Calibration Building and the Storage Building may be disposed of at any convenient time during the Retirement Program. Final debris removal must necessarily be conducted after completion of Activities 1 through 11.

Major subcontractual work items, engineering studies, and equipment design and procurement to be conducted in support of this activity are:

1) Economic studies as necessary to establish the economic advantages to the Commission for the various possibilities of disposal of structures, and

2) Subcontractual work in connection with miscellaneous debris disposal.

B. HEALTH AND SAFETY

This activity will be carried out under normal HNPF and CPPD safety regulations. Any necessary heavy equipment lifting and moving will be conducted in accordance with the CPPD Safety Manual. Radiological safety associated with any required decontamination operations will be conducted in accord with the HNPF Operations Manual, Volume I-a, Part III. Based on present estimates, it is considered that decontamination operations, if required, will be quite minor in nature.

C. STANDARDS

Any necessary demolition and disposal operations will be conducted in accordance with accepted good engineering practice. No special standards are deemed necessary.

IV. METHOD OF ACCOMPLISHMENT AND SEQUENCE OF EVENTS

1) The structures listed below will be radiologically surveyed and decontaminated, as necesary, in accord with the procedure outlined in Part II of this specification.

   a) Fuel and Moderator Assembly Building

   b) Storage Building south of Fuel and Moderator Assembly Building

   c) Steam Dump Facility

   d) Calibration Building

2) The above structures will be disposed of as Category B, C, or E items in accordance with the result of economic studies conducted as per Part II of this specification.

3) All loose debris still present on site as a result of HNPF Retirement activities will be disposed of.

4) A final radiation survey of the HNPF site will be conducted to verify that all exposed above grade surfaces of remaining structures and of the site in general with 10CFR20 requirements for an unrestricted area.

APPENDIX II

DETAILED PROCEDURES 1 THROUGH 14

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PROCEDURE 1.

CUTTING AND CAPPING REACTOR CONTAINMENT DRAIN LINE AND INSTALLATION OF REACTOR SODIUM HEEL DRAIN LINE

I. PURPOSE

This procedure establishes a step-by-step sequence for cutting and capping the reactor containment drain line and installation of a temporary drain line extending through the loading face shield. This permits removing most of the sodium heel (~2 in.) from the bottom of the reactor to the containment drain tank.

II. TASK DESCRIPTION

The activities to be accomplished by this procedure are as follows. Deenergize, tag, and remove electrical supplies to line Heaters 413-1, 413-2, and 413-3; remove insulation, reflective material, and heaters in area where cut is to be made; reduce core containment cavity pressure to atmospheric; cut and remove ~ 3 ft section of pipe from Line 413; seal the reactor side of the line by welding a 2-3/8 in. diameter x 1/4 in. carbon steel plate over the line opening; reestablish regulated inert atmosphere in reactor containment cavity; install a 2-3/8 in. daimeter x 1/4 in. carbon steel plate with a 90° elbow (fabricated as per attached drawing) on containment drain tank side of Line 413. Install the modified reactor plug in the loading face shield and insert the reactor portion of the drain line. Complete the assembly of the drain line, and leak check same. Install heater cable and thermocouples for temperature control. Insulate line, and establish control barrier around it.

III. CRITERIA

A. REQUIREMENTS AND PREREQUISITES

1. Coordinate activities with CPPD shift supervisor.
2. The following tools and equipment will be required.
    - a. Hand tools - wrenches, pliers, screwdrivers, hammer, wirecutters, spatulas, scrapers, and sodium augers
    - b. Bandsaw - portable (3 in. throat)
    - c. Portable grinder
    - d. Hand operated pipe cutter (2 in. throat) and spare cutting wheels
    - e. Chain falls, 1/2-ton capacity - 2 each.
    - f. 1/4 in. x 2-3/8 in. diameter carbon steel capping plates - 2 each (One of these plates shall be equipped with a capped 90° elbow per attached sketches)
    - g. Bread pans, steel - 4 each
    - h. Ohm meters - 2 each

i. Portable drop cord lights, with safety guards - 2 each

j. Plastic caps for 2 in. pipe

k. Cloth tape, sealed fiber, smooth finish, 2 in. wide - 2 rolls

l. Calcium carbonate (dry) - 1 bag.

m. Welding machine

n. Sheet polyethylene for apron under cutting locations to contain steel and sodium saw chips

o. Seamless steel tubing - 160 ft - 1 in. O.D. x 0.120 in. wall thickness for drain line

p. Electrical heater cable - 89 ft

q. Transformer for heaters - 1 each

r. Thermostats for heaters - 3 each

s. Insulation - John Manville - 1-1/2 in. x 1 in. thick thermobestos

3. Personnel assigned to the task shall study and sign the procedures.

4. Remove access block east of reactor; (CPPD personnel).

5. Close V-486 sodium inlet to containment drain tank.

6. Deenergize and tag out of service in accordance with normal HNPF tagging procedures line Heaters 413-1, 413-2, and 413-3. (CPPD personnel)

B. HEALTH AND SAFETY

1. Special Work Permits are required for all penetrations, cutting, handling, cleaning, and welding of sodium piping and for sodium handling operations.

2. Sodium protective clothing shall be worn at all times by all personnel while within the working areas of sodium pipe cutting, handling or cleaning, welding, and for handling or disposal of sodium scrap.

3. The buddy system is required for all work within below grade vaults and for all sodium handling, pipe cutting, and welding operations.

4. Deposit all sodium scrap in metal containers containing calcium carbonate and deliver them to the storage location designated by Health and Safety.

NOTE: No sodium scrap or sodium contaminated component may be left open or unattended except as specifically authorized by Health and Safety

5. Calcium carbonate and sodium fire fighting equipment must be kept standing by at all times during cutting, cleaning, and welding of sodium piping and druing handling of sodium scrap.

6. Disposal of sodium scrap and sodium contaminated equipment and materials shall be performed only as directed by Health and Safety.

7. Oxygen concentrations of working area atmospheres within enclosed volumes and below grade vaults shall be suitably monitored whenever inert gas releases may occur which could affect these enclosed working areas. No entry into atmospheres with less than 18% O_2 will be permitted except as specifically authorized by Health and Safety.

C. STANDARDS

Work performed in connection with this activity shall be carried out in accordance with normal HNPF Health and Safety regulations and the specific requirements of this procedure.

Inspection of all pipe cap and connection weldments shall include dye penetrant testing of each root pass and final pass.

IV. METHOD OF ACCOMPLISHMENT AND SEQUENCE OF EVENTS

1. Obtain Special Work Permit
2. Enter main pipe gallery and remove ~ 3 ft of pipe insulation from Line 413 - 9 ft down-steam of the elbow from the reactor on Line 413.
3. Check line Heater 413-2 for voltage, and disconnect heater lead wires at the junction box.
4. Bend line Heaters 413-2 out of the way to permit cutting and removing ~ 3 ft of pipe from Line 413.
5. Shut off the helium supply to the reactor containment cavity by closing V-8109 - reactor cavity helium supply inlet block; (Ref. Drawing D-070504, Sheet 2 of 2, Helium System).
6. Reduce reactor containment cavity pressure to atmospheric by closing valve V-10140, reactor cavity R/A vent FCV inlet; and V-10141 reactor cavity R/A vent FCV outlet; and by opening V-10142, reactor cavity R/A vent FCV bypass; (Ref. Drawing D-070507, Sheet 2 of 2).
7. Shut V-10142 - reactor cavity R/A vent FCV bypass when containment cavity pressure is reduced to atmosphere.
8. Tag the above valves (V-486, V-8109, V-10140, V-10141, and V-10142) out of service in accordance with normal HNPF tagging procedures; (CPPD personnel).
9. Remove ~ 3 ft of Line 413 and temporarily seal the cut ends of the pipes.
10. Weld a carbon steel capping plate over the open end of the reactor side of Line 413, and dye check weld.
11. Remove out of service tags mentioned in step No. 8 with normal HNPF back in service procedures, (CPPD personnel).
12. Reestablish the regulated inert atmosphere in the reactor containment cavity by opening V-8109 (reactor cavity helium supply pressure inlet), V-10140 (reactor cavity - R/A vent FCV inlet), and V-10141 (reactor cavity - R/A vent FCV outlet); (Ref. Drawing D-070504, Sheet 2 of 2, Helium System).
13. Weld a carbon steel capping plate (equipped with capped 1 in. 90° elbow) on the reactor containment drain tank side of Line 413, and dye check weld.

    NOTE: This elbow connection will remain sealed until the 1 in. drain line has been fabricated and is ready for installation (Step 17).

14. Field route the sodium drain line (by using 1 in. O. D. seamless carbon steel tubing with 0.120-in. wall) from Line 413, containment tank inlet, to the reactor loading face shield. Route the line about 4-in. above the high bay floor to provide clearance for heaters and insulation by use of angle iron support braces to maintain this elevation. Establish necessary barriers to prevent damage to drain pipe.

15. Fabricate a metal catch tray the full length of the horizontal run of the drain pipe. Dry calcium carbonate shall be added, as needed, to the catch tray.

16. Place a sodium drip pan containing dry calcium carbonate beneath the vertical run of the drain line which connects to Line No. 413.

17. Remove cap from 1-in. elbow on Line 413.

18. Weld the 1-in. reactor heel sodium drain line to the elbow on 413. Use dye penetrant to check the weld. Cap this portion of the drain line and perform a vacuum decay test.

19. Install the modified shield plug with the gas seal cap installed into the reactor loading face sheild by using the Fuel Handling Machine; (CPPD personnel).

20. Fabricate by use of 1-in. O. D. seamless tubing, the portion of the drain line to be inserted into the reactor. Use an empty storage cell to permit assembly of the full length (43-ft 8-in). Dye penetrant check each weld. Install a temporary plug in the upper end of this line to seal in the reactor atmosphere until assembly of the line is to be completed.

21. Transfer the assembled reactor portion of the drain line from the storage cell to the reactor by using the high bay crane. Remove the gas seal cap and immediately insert the reactor drain line. Lower the drain line; allow ~ 20 min for the line to warm before entering the sodium pool, until it rests on the bottom of the vessel. Four inches of drain line will be extending above the loading face shield.

22. Remove the temporary plug which was installed in Step 20, tape seal this connection until weldment is complete. Complete the assembly of the reactor drain line by welding the portion which was inserted into the reactor to the portion of line connected to Line 413. Dye penetrant check this weld.

23. When the drain line is completely assembled, increase the reactor pressure as per SOP-6005 to 5 psig. Leak check the drain line from the loading face shield to line No. 413 by using the soap bubble leak checking method. The drain line may also be leak checked by opening V-486 and establishing vacuum for a vacuum decay test.

24. If leaks are not indicated, reduce the reactor pressure to normal operating pressure and install drain line heater cables. Attach thermocouples for thermostat controls. Insulate the drain line from the loading face to Line 413 with 1-1/2-in. x 1-in. thick block insulation.

25. Complete the preheat circuitry and release the tagging order.

    All draining of reactor heel sodium operations will be done by CPPD personnel.

26. All tools, waste, etc., will be stored in a proper place and the area where work was done will be left in an orderly condition in accordance with HNPF policies.

PROCEDURE 2.
REMOVAL OF NaK FILLED INSTRUMENTATION

I. PURPOSE

This procedure establishes a step-by-step sequence for removing the NaK-filled pressure transmitting instrumentation from the HNPF sodium systems.

II. DESCRIPTION

The activities to be accomplished by this procedure are as follows. Deenergize and tag out the electrical power supplies to the line heaters in the general area of the pressure sensing element; remove the thermal insulation from around the housing; reduce the inert gas pressure on the particular sodium system in question to 1/4-in. w.g. pressure (use oil in the manometer with specific gravity of 1.0), and remove the assembly as a unit. In the primary sodium system, it will be necessary to cut the capillary transmitting tubes because they are encased in concrete.

III. CRITERIA

A. REQUIREMENTS AND PREREQUISITES

1. This activity will be coordinated with other plant activities during the HNPF retirement program.
2. All operations of plant equipment and deenergizing of electrical circuitry required in the performance of this procedure shall be performed only by CPPD personnel.
3. Notification shall be made to the CPPD shift supervisor prior to initiating this work and also immediately prior to making any penetration into piping or components which may affect or require special operation of any associated plant system or components.
4. Personnel assigned to this task will not be reassigned to other tasks and replaced with other personnel; in other words, once a man gains experience he will continue with this assignment until it is completed. This doesn't mean that he will not be given other work when this task is not in progress.
5. Tools and equipment required for the work detailed in this procedure are as follows.
   a. Hand tools - wrenches, pliers, hammers, lagging saws, tin snips, wire cutters, screw drivers, scrapers, etc
   b. Porter-Cable band saw - spare blades
   c. Welding machine - equipped for inert gas welding
   d. Oil manometer, specific gravity of 1.0, equipped with a connecting hose
   e. Portable drop cord lights with safety guards - 2 each
   f. Capping plates - 2-in. diameter by 1/4-in. - 42 each - 21 SS - 2 ft mild steel
   g. Ladders and scaffolding - as necessary
   h. Dry ice - procure as needed
   i. Plastic caps for 2-in. pipes, or cloth tape

j. Bread pans or buckets

k. Dry calcium carbonate

l. Hand pipe cutter - sized for 2-in. pipe

m. Side cutters - for crimping NaK lines

6. The following prerequisite items shall be completed and signed off prior to initiating performance of the work detailed in Section IV, "Method of Performance and Sequence of Events," of this procedure.

   a. Personnel assigned to the task shall study and sign this procedure.

   b. Potential hazards relating to the work to be performed shall be discussed with the personnel involved before the operation is initiated.

   c. Electrical power to the sodium lines where the sensing elements are located shall be tagged "out of service" in accordance with normal HNPF tagging procedures.

   d. Confirm that the particular heat transfer system where the sensing elements are being removed is at ambient temperature.

B. HEALTH AND SAFETY

1. Special work permits are required for all penetrations, cutting, handling, and welding of sodium piping and for sodium handling operations.

2. Sodium protective clothing shall be worn at all times by all personnel while within the working areas of sodium pipe cutting, handling, cleaning, welding, and for handling or disposal of sodium scrap.

3. The buddy system is required for all sodium handling, pipe cutting, and welding operations.

4. Deposit all sodium scrap in metal containers containing calcium carbonate and deliver them to the storage location designated by Health and Safety.

   NOTE: No sodium scrap or sodium contaminated component may be left open or unattended except as specifically authorized by Health and Safety.

5. Calcium carbonate and sodium fire fighting equipment must be kept standing by at all times during cutting, cleaning, welding of sodium piping, and handling of sodium scrap.

6. Disposal of sodium contaminated equipment and materials will be performed only as directed by Health and Safety.

7. The portable oxygen analyzer (with an audible alarm set at 19% oxygen) shall monitor continuously the work [illegible] atmosphere when work is being done in vaults or pipe galleries.

C. STANDARDS

Work performed in connection with this activity will be carried out under normal HNPF Health and Safety regulations (including sodium safety regulations) in accordance with Part III of the HNPF

Operations Manual, and the specific requirements of this procedure. Inspection of capping plate weldments will consist of dye penetrant testing of the final pass.

IV. METHOD OF ACCOMPLISHMENT AND SEQUENCE OF EVENTS

1. All preparatory work shall be completed before starting the work of this section of the procedure.
2. Confirm that the particular sodium line where a pressure sensing element is to be removed is at ambient temperature.
3. Confirm that the electrical supplies to the sodium line where the sensing element is located have either been removed or tagged "out of service."
4. Remove the thermal insulation from the pressure sensing element housing and the related sodium line as necessary to permit access for cutting.
5. Connect a manometer to the sodium pump case cover plate or other desirable location for the particular loop in question.
6. Reduce the inert gas pressure on the sodium loop to 1/4-in. w.g. pressure. Loop inert gas pressure shall be controlled by CPPD operating personnel.

CAUTION

The pressure transmitting instruments are filled with NaK (sodium-potassium compound) that burns when exposed to air. NaK remains in a liquid state until it is cooled to ~ 10°F. The bellows which is located inside the element housing is made of stainless steel with a wall thickness of ~ 0.003-in. The NaK is contained by the bellows; therefore, these instruments must be handled with care. Personnel will be <u>suited</u> as per CPPD Health and Safety instructions.

7. Confirm that the inert gas pressure on the loop in question is correct (1/4-in. w.g. pressure).
8. Saw or cut the sensing element pipe nozzle near the weld that attaches the assembly to the sodium line. This is 2-in. diameter pipe and can be cut rather quickly.

   NOTE: Do not permit the sensing element to fall.
9. Install temporary seals over the pipe stubs by using plastic caps secured with tape.
10. Disconnect the transmitter capillary tube from its support rail that leads to an instrument panel.
11. Disconnect the electrical connection from the terminal strip at the readout panel. This is the signal cable from the electronic transmitter to the terminal strip.
12. Remove the mounting bolts from the transmitting pot in the instrument panel and remove the assembly from the area.
13. Remove the seal (plastic cap) from the 2-in. diameter pipe stub on the sodium line and scrape out the residual sodium from interior surfaces. The gas pressure on the loop must be controlled.

14. Weld a 2-in. diameter x 1/4-in. mild steel capping plate on the line stub. Use dry ice if it is deemed necessary.

15. Dye penetrant inspect the capping plate weld.

16. Install a 2-in. diameter x 1/4-in. SS capping plate on the 2-in. diameter pipe spool that remains on the sensing element. This must be done carefully and welded slowly to prevent overheating the pressure sensing bellows.

NOTE: Nine of the 21 NaK-filled pressure sensing instruments are located in pipe vaults, and the pressure transmitting capillary tubes lead through conduits in the reactor high bay floor that are filled with concrete.

17. The nine pressure instruments with the pressure transmitting tubes sealed in the concrete floor will be removed as described above with the following exceptions.

    a. The 0.090-in. stainless steel tube will be crimped in four places to seal it.

    b. Cut the tube such that two crimps are on each side of the cut.

    c. The transmitting capillary tube shall be cut in the vault near its enterance into the concrete and at the high bay floor seal box.

    d. Assuming that ten feet of capillary tube is left in the vault floor and that the ID of the tube is 0.020-in., it will contain ~ 0.6 gm of NaK.

PROCEDURE 3.
ISOLATION OF THE PRIMARY SODIUM FILL TANKS

I. PURPOSE

This procedure establishes a step-by-step sequence for isolating the five primary sodium fill tanks from their related systems as a part of the retirement of the Primary Sodium Service System and to permit the removal of the fill tanks for off-site shipment.

II. TASK DESCRIPTION

The activities to be accomplished by this procedure are as follows: Disconnect and remove the electrical service to the primary sodium fill tanks and the associated sodium piping; disconnect and remove the temperature indicating instrumentation from the tanks and the associated sodium piping; disconnect and remove the sodium level instrumentation from the tanks; cut and remove the sodium piping and vent lines connecting to the fill tanks; install capping plates on the fill tank nozzles (the capping plates on the sodium inlet nozzles will be equipped with purge valves to permit pressurizing the tanks with inert gas).

III. CRITERIA

A. REQUIREMENTS AND PREREQUISITES

1. This activity will be coordinated with other plant activities during the HNPF retirement program.
2. All operations of plant equipment and deenergizing of electrical circuitry required in the performance of this procedure shall be performed only by or under the supervision of CPPD personnel.
3. Notification shall be made to the CPPD shift supervisor prior to initiating this work and <u>also</u> immediately prior to making any penetrations into piping or components which may affect or require special operation of any associated plant system or components.
4. Personnel assigned to specific noninterruptable tasks (such as pipe cutting, closure welding, and sodium removal and handling) shall not be given additional assignments until their particular phase of that task is completed.

   An inert atmosphere shall be maintained on the Primary Fill Tanks during the isolation process.
5. Tools and equipment required for the work detailed in this procedure are as follows:
   a. Hand tools - wrenches, pliers, hammers, lagging saws, tin snips, wire cutters, screw drivers, spatulas, sodium augers, and scrapers
   b. Bandsaw, Porter-Cable (3-in. throat)
   c. Electrically driven portable grinder
   d. Hand operated pipe cutter (4-in. throat) and spare cutting wheels
   e. Welding machine - equipped for inert gas welding

f. Dye Penetrant Weldment Inspection Kit

g. 1/4-in. 304 SS capping plate, 3-in, 10 each. Five 3-in. caps shall be equipped with 1/4-in. bar stock valves

h. 1/4-in. 304 SS capping plates, 2-in., five each

i. Bread pans, steel, four each

j. Ohm meters, 2 each (in good operating condition)

k. Portable drop cord lights, with safety guards, four each

l. Roll of 1/2-in. manila rope

m. Expandable rubber stoppers for 2- and 3-in. pipe

n. Plastic pipe caps for 2- and 3-in. pipe

o. Cloth tape, sealed fiber, smooth finish, 2-in. wide, 2 rolls

p. Supply of dry calcium carbonate

q. Dry ice (if required); procure as needed

r. Portable oxygen analyzer equipped with audible alarm

s. Cylinder of argon gas equipped with regulating valves and hoses

6. All sodium valves must be properly positioned (open or closed as required for performance of the procedure) before removal of instrumentation and electric power to heaters. No attempt should be made to operate sodium valves at temperatures below 300°F. The heaters will remain energized on the fill tank vent valves (V-419, V-422, V-425, V-435, and V-438) until the respective tank inlet and outlet lines have been capped; after which the vent valve will be closed, tagged, and the heaters secured.

7. The following prerequisite items shall be completed and signed off prior to initiating performance of the work detailed in Section IV, "Method of Performance and Sequence of Events," of this procedure.

   a. Attach to this procedure a copy of the appropriate P and I drawing marked to show the equipment and system isolation to be performed. NOTE: Drawing shall be marked by physical inspection of the system.

   b. Personnel assigned to the task shall study and sign the procedure.

   c. Potential hazards relating to the work to be performed shall be discussed with the personnel involved before the operation is initiated.

   d. The primary sodium service system shall be isolated from the primary main heat transfer system by closing the following block valves.

      V-108 - Balancing Leg, Primary Na Loop No. 1, drain

      V-109 - Balancing Leg, Primary Na Loop No. 1, drain

      V-208 - Balancing Leg, Primary Na Loop No. 2, drain

      V-209 - Balancing Leg, Primary Na Loop No. 2, drain

      V-308 - Balancing Leg, Primary Na Loop No. 3, drain

V-309 - Balancing Leg, Primary Na Loop No. 3, drain

V-439 - Primary Na fill tanks and reactor vent tie block

V-453 - Primary Sodium Loop No. 1 fill and drain block valve

V-454 - Primary Sodium Loop No. 2 fill and drain block valve

V-455 - Primary Sodium Loop No. 3 fill and drain block valve

V-456 - Drain from primary block valve No. 1

V-457 - Drain from primary block valve No. 2

V-458 - Drain from primary block valve No. 3

V-459 - Reactor drain block valve

V-460 - Primary Na Loop No. 1 reactor inlet fill and drain

V-461 - Primary Na Loop No. 2 reactor inlet fill and drain

V-462 - Primary Na Loop No. 3 reactor inlet fill and drain

V-463 - Primary Na Loop No. 1 throttle valve drain

V-464 - Primary Na Loop No. 2 throttle valve drain

V-465 - Primary Na Loop No. 3 throttle valve drain

e. Confirm that the following primary sodium service system valves are in the closed position.

V-416 - Sodium Melt Station outlet header valve

V-417 - No. 1 Primary Fill Tank inlet block valve

V-418 - No. 1 Primary Fill Tank drain and tie valve

V-420 - No. 2 Primary Fill Tank inlet block valve

V-421 - No. 2 Primary Fill Tank drain and tie valve

V-423 - No. 3 Primary Fill Tank inlet block valve

V-424 - No. 3 Primary Fill Tank drain and tie valve

V-430 - Primary Fill Tank drain header block valve

V-433 - No. 4 Primary Fill Tank inlet block valve

V-434 - No. 4 Primary Fill Tank drain and tie valve

V-436 - No. 5 Primary Fill Tan inlet block valve

V-437 - No. 5 Primary Fill Tank drain and tie valve

V-440 - Primary Fill Tank drain and inlet headers tie valve

V-480 - Primary Cold Traps outlet header valve

V-482 - Primary Sodium Service drain tank outlet valve

V-483 - Primary Sodium Service drain tank inlet valve

f. Electrical power to the primary sodium service system piping and component heaters with the exception of the fill tank vent valve heaters shall be deenergized and tagged "out of service" in accordance with normal HNPF tagging procedures.

g. Electrical power to the primary sodium fill tanks level instrumentation shall be deenergized and tagged "out of service" in accordance with normal HNPF tagging procedures.

h. Check to confirm that the entire primary sodium service system, except for the fill tank vent valves, is at ambient temperature prior to initiating the cutting of any of the sodium piping.

B. HEALTH AND SAFETY

1. Special work permits are required for all penetrations, cutting, handling, cleaning, and welding of sodium piping, and for handling or disposal of sodium scrap. The special work permit shall specify the use of the portable audible alarm equipped oxygen analyzer for continuous monitoring of the work area atmosphere.

2. Sodium protective clothing shall be worn at all times by all personnel while within the working areas of sodium pipe cutting, handling, or cleaning, welding, and for handling or disposal of sodium scrap.

3. The buddy system is required for all sodium handling, pipe cutting, and welding operations.

4. Deposit all sodium scrap in metal containers containing calcium carbonate and deliver them to the storage location designated by Health and Safety.

   NOTE: No sodium scrap or sodium contaminated component may be left open or unattended except as specifically authorized by Health and Safety.

5. Calcium carbonate and sodium fire fighting equipment must be kept standing by at all times during cutting, cleaning, and welding, of sodium piping and handling of sodium scrap.

6. Disposal of sodium contaminated equipment and materials will be performed only as directed by Health and Safety.

C. STANDARDS

Work performed in connection with this activity will be carried out under normal HNPF Health and Safety regulations (including sodium safety regulations) in accordance with Part III of the HNPF Operations Manual (Volume I-a), the CPPD Safety Manual, and specific requirements of this procedure. Inspection of the fill tank nozzle weldments will include dye penetrant testing of each root pass and final pass.

IV. METHOD OF ACCOMPLISHMENT AND SEQUENCE OF EVENTS

1. All preparatory work shall be completed before starting the work of this section of the procedure.

2. Deenergize, tag out, and disconnect the fill tank heater electrical supplies at the breakers and terminal boards, and then remove the leads from the fill tank heater connection.

3. Remove conduits and junction boxes associated with the fill tank heaters as required for clearance.

4. Deenergize, tag out, and disconnect the electrical supply to heaters on the sodium piping associated with the primary sodium fill tanks at the breakers and terminal boards, and then remove the leads from the heaters on sodium lines connecting to the fill tanks.

    NOTE: The extent of the pipe heater circuitry removal will be determined by physical inspection of the system.

5. Remove the conduits and junction boxes associated with these sodium piping heaters.

6. Disconnect and remove the temperature indicating instrumentation from the primary sodium fill tanks. These items shall include the thermocouple lead wires, conduits, and junction boxes as is necessary to permit subsequent removal of the tanks.

7. Remove thermal insulation from sodium lines No. 111, 125, and 170 where they connect to primary fill tank No. 1.

8. Remove the line heater reflector oven that covers the lines heaters from the uninsulated sections of sodium lines No. 111, 125, and 170.

9. Remove the line heater attaching devices from line heaters in the above described locations.

10. Remove or bend the line heaters away from the pipes to permit access for cutting in the above mentioned locales.

11. Remove thermal insulation from sodium lines No. 111, 125, and 170 upstream from the fill tank. The locations for removing the insulation will be determined by visual inspection of the pipes with the objective of moving the tanks being the determining factor.

12. Remove the line heater reflector oven from the sodium lines at the locations specified by Step 11.

13. Remove the line heater attaching devices and remove or bend the heaters away from the pipes to provide working access.

14. Prepare the corresponding lines connecting to primary sodium fill tanks No's. 2, 3, 4, and 5 for cutting by repeating Steps 7 through 13 for each tank.

15. Close valve V-889 (Reactor - Primary Fill Tanks Helium Supply Tie) and reduce the set point on PIC-805 to maintain a pressure of 2 in. of water in the primary fill tanks.

16. Open the primary sodium fill tank purge and vent valves.

Valve No.	Description
V-419	No. 1 Primary Fill Tank purge and vent valve
V-422	No. 2 Primary Fill Tank purge and vent valve
V-425	No. 3 Primary Fill Tank purge and vent valve
V-435	No. 4 Primary Fill Tank purge and vent valve
V-438	No. 5 Primary Fill Tank purge and vent valve

17. Check the inert gas pressure in the tanks as indicated on PIC-805. Should it be necessary to reduce the inert gas pressure in the tanks, operate the vapor trap pressure control valve bypass.

Valve No.	Description
V-1037	Bypass on CV-1005

18. Close the purge and vent valves on primary sodium fill tanks No's. 2, 3, 4, and 5.

Valve No.	Description
V-422	No. 2 Primary Fill Tank purge and vent valve
V-425	No. 3 Primary Fill Tank purge and vent valve
V-435	No. 4 Primary Fill Tank purge and vent valve
V-438	No. 5 Primary Fill Tank purge and vent valve

19. Properly support the selected spool of sodium line No. 111 to prevent it from falling or springing when it is cut. If pipe hangers are suitably located, they can be used as support mechanisms.

    CAUTION: The personnel assigned the task of cutting and installing capping plates on the fill tank lines shall be properly suited for working contaminated sodium as specified by the CPPD health physics representative. Notify CPPD shift supervisor prior to making each penetration into sodium lines.

20. Sodium line No. 111 will be completely full of sodium because of the physical location of the line connection on the tank. No inert gas purge is required.

21. Cut line No. 111 at the upstream locations, and tape seal the cut. If the line should spring apart, seal the open ends with plastic caps. Use the Porter-Cable bandsaw or the hand pipe cutter, whichever tool is the more applicable for the particular location.

22. Cut line No. 111 at the tank nozzle. The exact location of the cut will be determined by inspection of the tank nozzle with the objective of installing a capping plate that doesn't protrude beyond the tank insulation.

23. Seal the open stubs with plastic caps. Use cloth tape as a holding mechanism for the caps.

24. Lower the removed section of line No. 111 to the floor. This operation will be accomplished by attaching chain falls and rope lifting slings to the pipe spool, or it may be possible to place it on the floor by use of other means.

25. Remove the pipe spool from the work area to a predetermined storage area.

26. Sodium samples shall be obtained from the primary fill tanks. These samples will be taken from the following line stubs: No. 111, No. 112, No. 113, No. 114, and No. 115. The sodium is to be analyzed for carbon; therefore exposing the sample to air will not affect the analysis. Quantity of sodium required for each sample, 1/2 lb.

    The sodium samples will be placed in approved containers for shipment to the Liquid Metals Engineering Center, Atomics International.

27. Remove the plastic cap from line No. 111 tank nozzle, and extract the frozen sodium from the tank nozzle to a depth of 6 in. The sodium extraction is to be accomplished by use of spatulas, knives, sodium augers, or other suitable tools.

28. Position the capping plate inside of the tank nozzle stub, and make the weldment as per standard HNPF sodium pipe welding procedures. Dry ice is suggested as a heat sink to facilitate welding.

29. Perform dye penetrant inspection on the weldment root pass and the final pass.

30. Install a welded capping plate on the cut pipe stub on tank side of No. 1 fill tank drain line block valve V-418.

31. The steps in this procedure pertaining to the removal and capping of sodium line No. 111 shall be applicable to all the sodium lines connected to the primary fill tanks; reference attached P and I drawing.

32. Sodium line No. 125 should be empty; therefore, an inert gas will be admitted to the tank during the cutting and capping operation by purging inert gas through vapor trap No. 1 and Valve 419. A capping plate equipped with a bar stock valve shall be used to seal the tank penetration for line No. 125.

33. Purge and vent line No. 170 will be empty. The tank will be purged during the cutting and capping operation by admitting argon gas into the tank through the bar stock valve installed in the capping plate for sodium line No. 125.

34. Following the isolation of primary sodium fill tank No. 1, and each successive tank, close the purge and vent valve and deenergize the valve heaters before proceeding to the next tank.

35. Open the purge and vent valve on the next tank to be isolated and repeat Steps 19 through 33 for the corresponding lines.

36. Remove the level instrumentation and its related circuitry from the primary sodium fill tanks. These items shall be properly identified and placed in shipping crates.

37. Upon removal of a primary fill tank from the fill tank vault, the following work shall be accomplished: seal the terminals of the fill tank heaters with a moisture proof compound, install moisture proof caps over the tank instrument thimbles, and seal the thermal insulation to prevent rain water from saturating it during shipment.

38. A daily progress record (log book) of the task shall be kept, and delivered to CPPD at the completion of the Primary Sodium Fill Tanks isolation. This record shall include man hours, unusual problems encountered, and any unusual conditions that may have developed in the course of this work.

39. By signature, acknowledgment is made that this procedure "No. RDP 1-1 for Isolation of the Primary Sodium Fill Tanks" is studied and understood.

40. Procedure RDP 1-1. Isolation of the Primary Sodium Fill Tanks. Work completed in accordance with this procedure and the attached support drawings, weldment inspection reports, and other data shall be dated, listed, and signed.

PROCEDURE 4.

ISOLATION OF THE SECONDARY SODIUM FILL TANKS

I. PURPOSE

This procedure establishes a step-by-step sequence for isolating the three Secondary Sodium Fill tanks from their related systems as a part of the retirement of the Secondary Sodium Service System and to permit the removal of the tanks filled with sodium for off-site shipment.

II. TASK DESCRIPTION

The activities to be accomplished by this procedure are as follows. Disconnect and remove the electrical service to the secondary sodium fill tanks and the associated sodium piping, disconnect and remove the temperature indicating instrumentation from the tanks and the associated sodium piping, disconnect and remove the sodium level instrumentation from the tanks, cut and remove the sodium piping connecting to the fill tanks, install capping plates on the nozzles of the fill tanks, remove the helium system supply lines from the tank vapor traps, cut the fill tank vent lines, and install capping plates with purge valves to permit pressurizing the tanks with inert gas.

III. CRITERIA

A. REQUIREMENTS AND PREREQUISITES

1. This activity will be coordinated with other plant activities during the HNPF retirement program.
2. All operation of plant equipment and deenergizing of electrical circuitry required in the performance of this procedure shall be performed only by CPPD plant personnel.
3. Notification shall be made to the CPPD shift supervisor prior to initiating this work and also immediately prior to making any penetrations into piping or components which may affect, or require special operation of, any associated plant system or components.
4. Personnel assigned to specific noninterruptable tasks (such as pipe cutting, closure welding, and sodium removal and handling) shall not be given additional assignments until their particular phase of that task is completed.
5. Tools and equipment required for the work detailed in this procedure are as follows.
   a. Hand tools - wrenches, pliers, hammers, lagging saws, tin snips, wire cutters, screw drivers, spatulas, sodium augers, and scrapers
   b. Bandsaw, Porter-Cable (3-in. throat)
   c. Electrically driven portable grinder
   d. Hand operated pipe cutter (4-in. throat) and spare cutting wheels
   e. Chain falls, 1/2-ton capacity - 2 each
   f. Welding machine - equipped for inert gas welding
   g. X-ray machine or gamma-graphing equipment and supplies for weldment inspection

h. 1/4-in. 304 SS capping plates, 3-in. - 9 each - 3 capping plates shall be equipped with 1/4-in. bar stock valves.

i. Bread pans, steel - 4 each

j. Ohmmeters - 2 each (in good operating condition)

k. Portable drop cord lights, with safety guards - 4 each

l. Roll of 1/2-in. manila rope

m. Expandable rubber stoppers, for 3-in. pipe

n. Plastic pipe caps for 3-in. pipe

o. Cloth tape, sealed fiber, smooth finish, 2-in. wide - 2 rolls

p. Supply of dry calcium carbonate

q. Dry ice (if required) - procure as needed

r. Sodium sample shipping containers - 3 each

s. Argon bottle - equipped with regulator

t. Pressure gage - (0 - 10 psig)

u. Relief valve - set at 5 psig

6. All sodium valves must be in the proper position (open or closed as required for performance of the procedure) before removal of instrumentation and electric power to heaters. No attempt should be made to operate sodium valves at temperatures below 300°F.

7. The following prerequisite items shall be completed and signed off in the spaces provided prior to initiating performance of the work detailed in Section IV, "Method of Performance and Sequence of Events" of this procedure.

a. Attach to this procedure a copy of the appropriate P and I drawing marked to show the equipment and system isolation to be performed.

NOTE: Drawings shall be marked in accordance with physical inspection of the system.

b. Personnel assigned to the task shall study and sign the procedures.

c. Potential hazards relating to the work to be performed shall be discussed with the personnel involved before the operation is initiated.

d. The secondary sodium service system shall be isolated from the secondary main heat transfer system by closing the following block valves.

V-4104 - Secondary Sodium loop No. 1 fill and drain block valve (cold trap supply)

V-4107 - Secondary Sodium loop No. 1 fill and drain block valve (cold trap return)

V-4110 - Secondary Sodium loop No. 1 drain block valve

V-4105 - Secondary Sodium loop No. 2 fill and drain block valve (cold trap supply)

V-4108 - Secondary Sodium loop No. 2 fill and drain block valve (cold trap return)

V-4111 - Secondary Sodium loop No. 2 drain block valve

V-4106 - Secondary Sodium loop No. 3 fill and drain block valve (cold trap supply)

V-4109 - Secondary Sodium loop No. 3 fill and drain block valve (cold trap return)

V-4112 - Secondary Sodium loop No. 3 drain block valve

e. Electrical power to the secondary sodium service system piping and components heaters shall be deenergized and tagged "out of service" in accordance with normal HNPF tagging procedures.

f. Electrical power to the secondary sodium fill tanks level instrumentation shall be deenergized and tagged "out of service" in accordance with normal HNPF tagging procedures.

g. Check to confirm that the entire secondary sodium service system is at ambient temperature prior to initiating the cutting of any of the sodium piping.

h. The techniques described in Section IV (Method of Accomplishment and Sequence of Events) of this procedure for maintaining an inert gas atmosphere in secondary sodium fill tank No. 1 shall also be applicable to secondary sodium fill tanks No. 2 and 3 during their isolation. The helium system valves that apply to fill tanks No. 2 and 3 shall be determined from P and I drawing D-070504.

i. Sodium samples will be obtained from secondary fill tank lines nozzles No. 178, 179, and 180.

B. HEALTH AND SAFETY

1. Special Work Permits are required for all penetrations, cutting, handling, cleaning, welding of sodium piping, and for sodium handling operations.
2. Sodium protective clothing shall be worn at all times by all personnel while within the working areas of sodium pipe cutting, handling, cleaning, welding and for handling or disposal of sodium scrap.
3. The buddy system is required for all sodium handling, pipe cutting, and welding operations.
4. Deposit all sodium scrap in metal containers containing calcium carbonate and deliver them to the storage location designated by Health and Safety.

   NOTE: No sodium scrap or sodium contaminated component may be left open or unattended except as specifically authorized by Health and Safety.

5. Calcium carbonate and sodium fire fighting equipment must be kept standing by at all times during cutting, cleaning, and welding of sodium piping and during handling of sodium scrap.
6. Disposal of sodium contaminated equipment and materials shall be performed only as directed by Health and Safety.

C. STANDARDS

Work performed in connection with this activity shall be carried out under normal HNPF Health and Safety regulations (including sodium safety regulations) in accordance with Part III of the HNPF Operations Manual (Volume I-a), the CPPD Safety Manual, and specific requirements of this procedure.

Inspection of the fill tank nozzle cap weldments shall include dye penetrant testing of each root pass and final pass.

IV. METHOD OF ACCOMPLISHMENT AND SEQUENCE OF EVENTS

1. All preparatory work shall be completed before starting the work of this section of the procedure.
2. Tag out and disconnect the fill tank heaters electrical supplies at the breakers and terminal boards, and then remove the leads from the fill tank heater connections.
3. Remove conduits and junction boxes associated with the fill tank heaters.
4. Tag out and disconnect the sodium piping heater electrical supplies at the breakers and terminal boards, and then remove the leads from the heaters on sodium lines connected to the fill tanks.
5. Remove the conduits and junction boxes associated with these sodium piping heaters.

   NOTE: The extent of the pipe heater circuitry removal will be determined by the physical inspection of the system.

6. Remove the sodium level instrumentation and its related circuitry from the secondary sodium fill tanks. These items shall be properly identified and placed in shipping crates.
7. Disconnect and remove the temperature indicating instrumentation from the secondary sodium fill tanks. These items shall include the thermocouples lead wires, conduits, and junction boxes as necessary to permit subsequent removal of the tanks.
8. Confirm that the following secondary sodium service system valves are in a closed position.

Valve No.	Description
V-490	No. 1 Secondary Sodium Fill Tank Inlet Block Valve
V-491	No. 1 Secondary Sodium Fill Tank Outlet Block Valve
V-492	No. 2 Secondary Sodium Fill Tank Inlet Block Valve
V-493	No. 2 Secondary Sodium Fill Tank Outlet Block Valve
V-494	No. 3 Secondary Sodium Fill Tank Inlet Block Valve
V-495	No. 3 Secondary Sodium Fill Tank Outlet Block Valve
V-498	Secondary Sodium Fill and Drain Tie Valve
V-499	Secondary Sodium Service Drain Block Valve
V-4103	Secondary Sodium Service Fill Block Valve
V-488	Secondary Sodium Service Make-up Block Valve

9. Close the helium gas supply valves to vapor traps No. 4, No. 5, and No. 6.

Valve No.	Description
V-8194, V-8195	Helium supply valves to VT-4
V-8200, V-8201	Helium supply valves to VT-5
V-8206, V-8207	Helium supply valves to VT-6

10. Reduce the inert gas pressure in thetanks to 0.25 psig or less by operating the vapor trap vent valves.

Valve No.	Description
V-9198	Helium vent valve from VT-4
V-8204	Helium vent valve from VT-5
V-8210	Helium vent valve from VT-6

11. Remove the thermal insulation from sodium line No. 108 where it connects to Secondary Fill Tank No. 1. A small amount of tank insulation may have to be removed around the pipe nozzle to provide proper access.

12. Remove a section of the line heater reflector oven from over the line heaters.

13. Disconnect the line heater attaching devices to permit bending the heaters away from the pipe. Provide access for cutting the pipe.

14. Select a desirable location on sodium Line 108 upstream from the tank. (Location to be determined by on-site inspection of the pipe.) Remove the thermal insulation from the pipe. Three foot section.

15. Remove the line heater reflector oven and remove the line heater attaching devices.

16. Bend the heaters away or remove from the pipe to permit access for cutting.

17. Support the section of Line 108 that is to be removed by rope slings and chain falls.

NOTE: Sodium samples shall be obtained from secondary sodium fill tank nozzles No. 178, 179, and 180. The sodium shall be extracted in as large pieces as possible and placed in approved gas tight shipping containers for shipment to the Liquid Metals Engineering Laboratory. The shipping containers shall have an inert gas atmosphere. The sodium will be analyzed for carbon; therefore, limited exposure to air when obtaining the samples will not affect the analysis. The tool used to extract the sample must be clean and free of all foreign matter.

(CAUTION)

The personnel performing the sodium line cutting operation shall suit in the proper protective clothing as specified by Health and Safety and detailed in the special work permit issued for this work.

Notify CPPD Shift Supervisor prior to making each penetration into sodium lines.

NOTE: When cutting and removing a section of sodium Line 108 an inert atmosphere shall be maintained on the fill tank. This will be accomplished by purging inert gas into the fill tank through vapor trap No. 4. The following designated helium system valves shall be operated as described to supply the inert gas atmosphere.

Valve No.	Description	Valve Position
V-8194	Secondary Sodium Fill Tank No. 1 Supply C.V. in	Open
V-8196	Secondary Sodium Fill Tank No. 1 Supply C.V. out	Open
V-8197	Secondary Sodium Fill Tank No. 1 Supply	Opened and Closed as required

The above listed helium system valves will be closed upon completion of capping sodium Line 108.

Valve No.	Description
V-8194	Secondary Sodium Fill Tank No. 1 Supply C.V. in
V-8196	Secondary Sodium Fill Tank No. 1 Supply C.V. out
V-8197	Secondary Sodium Fill Tank No. 1 Supply

18. Cut sodium Line 108 at the nozzle where it connects to the fill tank. The cut shall be made to permit installation of a capping plate on the nozzle; the top of the pipe cap shall be flush with the tank insulation. The pipe cutting tool may be either the Porter-Cable saw or the hand operated cutter, whichever is more suitable for the particular work area.

19. Tape seal the cut; or, if the pipe should spring apart, install plastic caps over the open ends of the pipe.

20. By using the same technique, cut the pipe at the previously prepared location upstream from the tank.

(CAUTION)

When this cut is made, steady the pipe to prevent it from swinging and possibly causing someone to fall or be injured.

21. Install plastic caps over the open ends of the pipe.

22. Lower the removed section of Line 108 to the floor and remove it from the work area.

23. Remove the plastic pipe cap from the line stub nozzle on the tank.

24. By use of a spatula, knife, sodium auger or other suitable tool, remove the sodium from the tank nozzle to a depth of about 4 in.

(CAUTION)

The sodium that is extracted from the tank nozzle shall be placed in bread pans containing dry calcium carbonate and delivered to the storage location previously designated by Health and Safety.

25. Upon completion of the sodium extraction operation, install an expandable rubber stopper in the open end of the nozzle. The stopper will prevent metal cutting from contaminating the sodium during the next operation.

26. Prepare the tank nozzle for welding. Consult with the welder on the preparation which is desired.

(CAUTION)

When using an electrically driven grinding machine in awkward positions do not lock or tape the switch in the on position. Grinders have been known to cause injury due to the wheel catching and wrenching the tool from the users hand.

27. When the nozzle preparation is completed, remove all of the metal cuttings from the pipe and rubber stopper.

28. Remove the rubber stopper and immediately position a 3-in. bar stock valve equipped with a capping plate on the tank nozzle and seal weld it to the nozzle.

NOTE: Standard techniques for welding sodium pipe shall be used. This includes inert gas purge inside the pipe capping plate to prevent oxidation. Dry ice may be employed as a heat sink to prevent melting the frozen sodium in the tank nozzle.

29. If a purge hole has been drilled in the capping plate, weld it closed.

30. Connect a regulator equipped argon bottle to the 1/4-in. bar stock valve on line Stub 108. The line connecting the argon bottle to the capping plate will be equipped with a low range pressure gage (0 - 10 psig) and a relief valve set to relieve at 5 psig.

31. Cut line No. 227 as described in procedural steps 9 through 29. Maintain the inert atmosphere on fill tank No. 1 by purging argon gas into the tank through the 1/4-in. bar stock valve on line stub No. 108.

(CAUTION)

The work area shall be well ventilated when permitting inert gas to escape from the fill tank.

32. Sodium lines No. 109, 110, 178, 179, and 180 shall be removed as per procedural steps 9 through 29 for Line 108.

33. Sodium lines No's. 228 and 229 shall be removed and the tank nozzles capped in the same manner as that for line No. 227 (steps 9 through 29 above); except, the vapor traps must be removed and the inert atmosphere shall be maintained on the tanks.

34. The secondary fill tank vapor trap line stubs shall be capped by welding capping plates on them.

35. The helium supply lines to the vapor traps shall be removed as necessary to permit removal of the fill tanks.

36. A daily progress record (log book) of the task shall be kept and delivered to CPPD at completion of the Secondary Sodium Fill tanks isolation. This record shall include man hours, unusual problems encountered, and any unusual conditions that may have developed in the course of this work.

PROCEDURE 5.

REMOVAL OF SECONDARY SODIUM FILL TANKS AND TRANSFER TO RAILROAD CARS

I. PURPOSE

This procedure establishes a step by step sequence for removal of the three secondary sodium fill tanks from their present below grade location and for transferring them to railroad cars for transfer off-site.

II. TASK DESCRIPTION

The activities to be accomplished by this procedure are as follows.

Prepare the tanks for removal by welding lifting devices and support bracing to the tank cradles. Remove the access plugs from above No. 3 fill tank.

Install temporary shoring under the auxiliary bay floor above stairway No. 1 and elsewhere as necessary to support the weight of tank. The estimated weight of a filled tank is 50 tons. Rig the auxiliary bay crane with a 100,000 lb capacity dynometer to the north side of tank by using 1-1/4-in. diameter cable with 1-1/2-in. diameter shackles. Position a 60 to 75 ton mobile crane in the auxiliary bay, keeping the weight of the crane south of the floor above No. 1 stairway. Rig the mobile crane to the south side of the fill tank by using 1-1/4-in. cables and 1-1/2-in. shackles. Remove tank cradle hold down nuts and lift tank by both cranes. When the tank is above floor level, turn it by moving the overhead crane and set it down on the supported floor. Move the tank southward, out through the auxiliary bay, on rollers by using truck mounted mobile crane as the motive force. Lift the tank by using the mobile crane and position the railroad car underneath it. Secure the tank with tie downs and blocking for shipment. Fill tanks No. 1 and No. 2 are not located beneath the access plugs, and therefore will be moved to the position left by No. 3 tank before they can be lifted. This will be accomplished by positioning cribbing between the tank pedestals and by sliding them into position below the access hole.

III. CRITERIA

A. REQUIREMENTS AND PREREQUISITES

1. This activity will be coordinated with other plant activities during the HNPF retirement program. This includes removal of the loading face shield compressor enclosure and the enclosure above No. 1 stairway.
2. All operation of plant equipment required in the performance of this procedure will be performed or authorized by CPPD plant personnel.
3. Notification shall be made to the CPPD shift supervisor prior to interruption of any normal operating equipment.
4. Tools and equipment required for the work detailed in this procedure are as follows.
   a. 60- to 75-ton mobile crane
   b. Lumber for shoring - 8-in. x 8-in. fir

c. Two I-beams 8-in. x 8-in. wide flange 48 lb/ft - 16 ft long

d. Bolts 1-in. NC x 3-1/2-in. long, quantities - 16

e. Dynometer - 100,000-lb capacity - 1 required

f. Cable spreading device

g. Lifting devices - 12 each per AI drawing AX-22156

h. Jacking lugs

i. 1-1/2-in. diameter cable - 10 ft - 4 each

j. 1-1/2-in. diameter cable - 12 ft - 4 each

k. 1-1/2-in. diameter shackles - 8 each

l. Pipe rollers 24, 3-ft schedule-80, 4-in. diameter

m. 15-in. channel iron - 20 ft

n. Three-ton chain falls - 2 each

o. Welding machine

p. 4-in. x 4-in. Tee iron bracing - 280 ft.

q. Plywood 1/4-in. - 4 ft x 8 ft

r. Mild steel plate 1/4-in. - 4 ft x 8 ft

5. The following prerequisite items shall be completed and signed off and dated prior to initiating performance of the work detailed in Section IV, "Method of Performance and Sequence of Events" of this procedure.

   a. The secondary fill tanks have been isolated from the secondary service system per RDP 2-1.

   b. Verify that the auxiliary bay crane has been inspected by a factory representative and is in good condition.

   c. Potential hazards relating to the work to be performed shall be discussed with the personnel involved.

   d. Personnel assigned to this task shall have studied and signed the procedure.

   e. Attach one copy of each of the following drawings to this procedure.

AI - AX-22156	Cradle Modification to Sec. Fill Tanks
AI - AX-22157	Shipping Arrangements for Na Storage Tanks

B. HEALTH AND SAFETY

1. A fixed barrier must be assembled prior to removal of the access plug over the fill tanks.

2. Normal HNPF safety practices pertaining to working near open holes will be used as specified by Health and Safety.

IV. METHOD OF ACCOMPLISHMENT AND SEQUENCE OF EVENTS

NOTE: The following description is for removal of tank No. 3 which will be the first tank removed. The following sequence of removal will also pertain to tanks No. 1 and No. 2.

1. Weld the secondary fill tank lifting devices (four for each tank) and support braces to the tank cradles as per Atomics International drawing AX-22156.
2. Install temporary floor support shoring under the area of stairway No. 1 as shown on the enclosed drawing.
3. Assemble a rigged barrier around the fill tank access plugs.
4. Remove the secondary sodium fill tank access plugs and place them out of the way against the west wall of the auxiliary bay. Remove or relocate, as needed, the air duct and cable tray to provide tank clearance. Heater cable covers on the ends of the tanks can be removed to provide added clearance.
5. Rig the auxiliary bay bridge crane to the north side of the tank, as shown on the enclosed drawing, with the 100,000-lb capacity dynometer attached.
6. Locate the mobile crane south of stairway No. 1 such that the weight imposed by the crane will bear on the supported flooring between No. 1 stairway and column line No. 16. (Refer to Bechtel Corp. Dwg. No. D-700549, Rev. 3.)
7. Rig the mobile crane as shown in enclosed drawing.
8. Remove the cradle hold down nuts (16 per tank).
9. Lift the tank by simultaneously operating the two cranes such that each is lifting only one-half the total weight of the tank. Do not exceed the rated capacity of the auxiliary bay crane which is 50,000 lb (25 tons). One man shall be assigned to coordinate the actions of the two crane operators. The lifts shall be made by using slow speeds and by stopping periodically to assure that a simultaneous lift is being accomplished. In addition, a level will be placed on one of the spreader bars, and measurements or markings will be utilized to assure that load line speeds of the two cranes are the same.
10. When the tank is above floor level turn it 90° by moving the boom of the mobile crane and traveling south with the overhead crane.
11. Bolt two 16-ft I-beams between the tank cradles as skids for moving the tank out of the building. Lay 1/4-in. plywood with 1/4-in. steel plates on top of the floor covering.
12. Set the tank on rollers, which are positioned on the floor covering, above the temporary shoring. Rig the tank to the mobile crane.
13. Move the tank by using the mobile crane through the auxiliary bay door to the railroad loading area.
14. Lift the tank by using the mobile crane and position the railroad car underneath the tank.
15. Lower the tank onto the railroad car properly positioned for shipment.

PROCEDURE 6.

ISOLATION AND DISASSEMBLY OF THE PRIMARY SODIUM SERVICE SYSTEM

I. PURPOSE

This procedure establishes a step-by-step sequence for isolating the Primary Sodium Service System from the main heat transfer system and removing the Primary Sodium Service System piping and components from the facility.

II. DESCRIPTION

The electrical services to the Primary Sodium Service System will be deenergized, tagged, and disconnected. These electrical services include the electrical supply to the pipe and component preheat units, sodium leak detectors, local instrumentation, and service pumps.

Cooling air ducts which supply cooling air to the cold traps, freeze traps, and plugging meter assemblies will be disassembled and removed. Thermal insulation, sodium pipe heaters, associated conduit, and temperature indicating instruments will be removed to provide access for pipe cutting.

Sodium lines connecting the service system to the primary heat transfer system will be cut near the heat transfer pipes.

Seal plates will be welded onto the line stubs left on the main heat transfer system. When the service system is isolated from the heat transfer system, the service pipe and individual components will be removed for disposal.

III. CRITERIA

A. REQUIREMENTS AND PREREQUISITES

1. This activity will be coordinated with other plant activities during the HNPF retirement.
2. All operation of plant equipment and deenergizing of electrical circuitry required in the performance of this procedure shall be performed only by CPPD plant personnel.
3. Temporary support shall be provided as necessary for any item (piping or component) prior to removing its normal support devices.
4. Notification shall be made to the shift supervisor prior to initiating this work and also immediately prior to making any penetrations into piping or components which may affect, or require special operation of, any associated plant system or components.
5. The following prerequisite items shall be completed and signed off prior to initiating performance of the work detailed in Section IV, "Method of Performance and Sequence of Events" of this procedure.
   a. Close and tag out all helium valves to components in the primary sodium service system.
   b. Personnel assigned to this task shall study the procedure and sign same.

c. Potential hazards relating to the work to be performed shall be discussed with the personnel involved before the operation is initiated.

d. Electrical power to the piping, component heaters, leak detectors, motors, and EM pumps shall be deenergized and tagged out per HNPF tagging procedures.

6. Dry ice shall be employed as a heat sink as necessary when installing capping plates or pipe caps.

7. The inert gas pressure on the primary heat transfer loops shall be carefully controlled during the process of cutting and capping the sodium lines connecting the service system to the primary heat transfer loops.

8. The expandable rubber stopper technique shall be used to maintain an inert atmosphere on the primary heat transfer loops.

9. Plastic caps shall be used as temporary seals on exposed sodium piping. Metal seal plates will be welded on the reusable components and the primary heat transfer loop connections as specified.

10. Tools and equipment required for the work detailed:

a. Hand tools - wrenches, pliers, hammers, lagging saws, hammers, tin snips, wire cutters, spatulas, scrapers, sodium augers, etc.

b. Porter-Cable band saw - supply of spare blades

c. Welding machine - equipped for inert gas welding

d. Cloth tape - 4 rolls

e. Plastic pipe caps - sized for 2-in. and 3-in. pipe

f. 36 capping plates - 1-15/16-in. diameter - 1/4-in. thick type 304 SS

g. 24 capping plates - 2-15/16-in. diameter - 1/4-in. thick type 304 SS

h. Scaffolding - as required

i. Ladders - as necessary

j. Hand operated pipe cutters for 2-in. and 3-in. pipe

k. Portable oxygen analyzer equipped with audible alarm

l. Hoisting and rigging equipment as required for hoisting pipe and components from the vaults

B. HEALTH AND SAFETY

1. Special work permits are required for all vault entries, pipe penetrations, cutting, cleaning, capping, and sodium handling operations.

2. Sodium protective clothing shall be worn at all times by all personnel while within the confines of the working areas during sodium pipe cutting and capping operations.

3. Deposit all sodium scrap in metal containers containing calcium carbonate and deliver them to the storage location designated by Health and Safety. All pipe containing radioactive sodium shall be identified before removal from the working area.

4. Calcium carbonate and sodium fire fighting equipment must be kept standing by at all times during cutting, cleaning, capping, and handling of sodium scrap.
5. The buddy system is required for all vault entries, sodium handling, pipe cutting, and welding operations.
6. Disposal of sodium contaminated equipment and other radioactive contaminated materials will be performed only as directed by Health and Safety
7. No sodium scrap or sodium contaminated component may be left open or unattended except as specifically authorized by Health and Safety.
8. During work in vaults or cells which may release inert gas, a portable oxygen analyzer shall be used to continuously monitor the work area atmosphere. This analyzer shall have an audible alarm which will be actuated at 18% or less of oxygen.
9. Any source of water within the area shall be noted and sodium fire equipment shall be readily available.

C. STANDARDS

Work performed in connection with this activity will be carried out under normal HNPF Health and Safety regulations (including sodium safety regulations) in accordance with Part III of the HNPF Operations Manual (Volume I-a), the CPPD Safety Manual, and specific requirements of this procedure. Inspection of the capping plate weldments will consist of dye penetrant inspection of the final pass.

IV. METHOD OF ACCOMPLISHMENT AND SEQUENCE OF EVENTS

1. All preparatory work shall be completed prior to starting the work detailed by this section of the procedure.
2. Confirm that the preheat electrical supplies have been deenergized and tagged out per Part III, Section A, Step 5-d.
3. Disconnect the primary sodium service electrical supplies at the breakers by removing fuses and/or by lifting electrical leads feeding the service pumps, sodium line heaters, cold trap furnaces, carbon trap heater, leak detectors, electrical controls, and instruments. Electrical leads disconnected at the breakers must be taped with electrical tape.
4. Remove the leads at the line heater terminals, furnaces, and electrically supplied equipment.
5. Remove electrical conduits, junction boxes, and wire trays necessary to facilitate disassembly of the pipe system and its components.
6. Remove the temperature indicating instrumentation from the piping and components including its related conduit and junction boxes.
7. Remove the thermal insulation, heater reflector oven, and line heaters.
8. Reduce the inert gas pressure to 1/4-in. w.g. on primary heat transfer loops by adjusting the set points on PIC-803 reactor pressure control. A slight positive inert gas pressure (1/4-in. w.g.) shall be maintained on the primary heat transfer loops when cutting and capping the service system sodium lines that connect to the primary heat transfer loops.

9. The sodium service system lines connecting to the primary heat transfer loop No. 1 shall be cut and capped by use of the following technique.

10. Cut a section from sodium line No. 137 (3-in. diameter loop No. 1 fill and drain) where it connects to line No. 231 (16-in. diameter reactor outlet line); leave a 6-in. line stub to cap. Install temporary seals (plastic caps) over the exposed pipe openings (four seals required).

11. Remove the pipe spool from the work area to a specified storage area.

12. Remove the temporary seal from line stub No. 137 on sodium line No. 231 and insert an expandable stopper into the line stub to seal the pipe.

13. Remove the residual sodium from the interior of the pipe stub to a depth of 4 in.

14. Remove the expandable stopper and position a capping plate 1 in. inside the pipe stub, tack weld in position.

15. Complete the weld and dye penetrant inspect the final pass.

16. Cut a section from sodium line No. 136 (2-in. diameter drain from sodium block valve V-101) where it connects to valve V-101; leave 6 in. of line stub to cap. Install temporary seals (plastic caps) over the exposed pipe openings (four seals required).

17. Remove the pipe spool from the work area to a specified storage area.

18. Remove the temporary seal from line stub No. 136 on sodium valve V-101 and install an expandable stopper in the pipe stub.

19. Remove the residual sodium from the interior of the line stub to a depth of 4 in.

20. Remove the expandable stopper from the line stub and position a capping plate 1 in. inside the pipe; tack weld in position.

21. Complete the weld and dye penetrant inspect the final pass.

22. Cut a section from sodium line No. 138 (2-in. diameter throttle valve V-103 drain) where it connects to the valve body, leaving 6 in. of line stub to cap. Install temporary seals (plastic caps) over the exposed pipe openings (four seals required).

23. Remove the pipe spool from the area and store in a specified storage area.

24. Remove the temporary seal from line stub No. 138 on sodium valve V-103.

25. Insert an expandable stopper into the line stub and seal the pipe.

26. Remove the residual sodium from the interior of line stub No. 138 to a depth of 4-in.

27. Remove the expandable stopper and position a capping plate 1 in. inside the line stub; tack weld it in position.

28. Complete the capping plate weld and dye penetrant inspect the final pass.

29. Cut a section from sodium line No. 139 (3-in. diameter fill and drain line, loop No. 1) where it connects to sodium line No. 237 (14-in. diameter reactor inlet line); leave a 6-in. pipe stub. Install temporary seals (plastic caps) over the exposed pipe openings; four seals required.

30. Remove the pipe spool from the area and store it in a specified storage area.

31. Remove the temporary seal from line stub No. 139 on sodium line No. 237 and insert an expandable stopper to seal the pipe.

32. Remove the residual sodium from the interior of line stub No. 139 to a depth of 4 in.

33. Remove the expandable stopper and position a capping plate 1 in. inside the line; tack weld it in position.

34. Complete the capping plate weld and dye penetrant inspect the final pass.

35. Cut a section from sodium line No. 224 (2-in. diameter balancing leg drain, loop No. 1) where it connects to the pipe tee; leave a 6-in. pipe stub to cap (service system side, junction of sodium lines No. 224 and No. 318).

36. Install temporary seals (plastic caps) over the exposed pipe openings (four seals required). Remove the pipe spool from the area to a specified storage area.

37. Remove the temporary seal from line stub No. 224 (balancing leg side) and position an expandable stopper inside the pipe to seal the pipe. Remove the residual sodium from the pipe to a depth of 4-in.

38. Remove the expandable stopper and position a capping plate in line stub No. 224. Tack weld it in position.

39. Complete the weld and dye penetrant inspect the final pass.

40. The following sodium service system lines connecting to primary heat transfer loops No. 2 and No. 3 will be cut and capped using the same techniques specified in procedural steps No. 9 through No. 33.

PRIMARY HEAT TRANSFER LOOP NO. 2

Line No.	Line Diameter (in.)	Description
140	2	Drain from block valve V-201
141	3	Fill and drain loop No. 2 reactor outlet line
142	2	Drain from throttle valve V-203
143	3	Fill and drain from loop No. 2 reactor inlet line
225	2	Balancing leg drain

PRIMARY HEAT TRANSFER LOOP NO. 3

Line No.	Line Diameter (in.)	Description
144	2	Drain from block valve V-301
145	3	Fill and drain loop No. 3 reactor outlet line
146	2	Drain from throttle valve V-303
147	3	Fill and drain loop No. 3 reactor inlet line
226	2	Balancing leg drain

41. Cut a section from sodium line No. 148 (3-in. diameter reactor fill and drain) where it connects to the service system side of sodium valve V-459. Leave a 6-in. pipe stub to cap.

42. Install temporary seals (plastic caps) over the exposed pipe openings (four seals required). Remove the pipe spool from the area to a specified storage area.

43. Remove the residual sodium from line stub No. 148 at sodium valve V-459. No expandable stopper is required because of the closed valve.

44. Place a capping plate in line stub No. 148 and tack weld it in position.

45. Complete the capping plate weld and dye penetrant inspect the final pass.

46. Cut a section from sodium line No. 305 (2-in. diameter moderator coolant piping drain) where it connects to sodium line No. 284 (6-in. diameter moderator coolant inlet line) leaving a 6-in. line stub. Install temporary seals (plastic caps) over the exposed pipe openings; four seals required. Remove the pipe spool from the area to a specified storage area.

47. Remove the temporary seal from line stub No. 305 on sodium line No. 284 and install an expandable stopper in the pipe.

48. Remove the residual sodium from the interior of the pipe to a depth of 4 in.

49. Remove the expandable stopper and place a capping plate in the pipe stub; tack weld it in position.

50. Complete the capping plate weld and dye penetrant inspect the final pass.

51. Cut a section from line No. 159 (2-in. diameter interconnecting line from VT-14 to drain tank) where it connects to valve V-475 (service system side of valve); leave a 6-in. line stub. Vapor trap No. 14 is to remain in place to provide a normal source of helium to the reactor. Install temporary seals over the exposed pipe openings and remove the pipe spool from the area.

52. Remove the temporary seal from the line stub on valve V-475 and remove the residual sodium from the interior of the pipe. Position a capping plate inside the line stub and weld it in position. Dye penetrant inspect the final pass.

53. Cut and remove a section from line No. 169 (3-in. diameter vent line) at the junction (pipe tee) of lines No. 169 and No. 176. Vapor trap No. 3 is to remain in place to provide normal venting of reactor. The pipe section is to be removed from the service system side of the pipe tee.

54. Cap line stub No. 169 (reactor side) as per step No. 52.

> NOTE: The primary sodium system is now isolated from the main heat transfer system. The following portion of this procedure is descriptive of isolation and removal of primary sodium service components which include: (1) drain tanks (2), (2) service pumps (2), (3) carbon trap (1), (4) plugging meters (2), (5) freeze traps (7), and (6) vapor traps (2).

Due to the configuration of the service system piping within the confines of the Primary Sodium Service Vaults, the sodium lines will be cut and removed as necessary to permit removal of the items specified above.

55. Cut and remove a section from sodium line No. 132 (2-in. diameter containment tank T-407 drain line) near the point where it connects to the containment drain tank.

56. Install temporary seals (plastic caps) over the exposed pipe openings; four seals required. Remove the pipe spool from the area.

57. Cut a section from line No. 413 (2-in. diameter containment tank inlet line) where it connects to the containment drain tank.

58. Install temporary seals (plastic caps) over the exposed pipe openings.

59. Cut and remove a section from line No. 133 (3-in. diameter containment drain tank vent line) where it connects to the drain tank.

60. Install temporary seals (plastic caps) over the exposed pipe openings. Four seals required. Remove the pipe spool from the area.

61. Remove the containment drain tank from its mounting and place it aside in the pipe vault until procedure RDP 5-2 (Isolation of the Reactor) has been completed. The containment drain tank can easily be removed at that time. Completely remove line 413.

62. Remove the protective cage from the primary sodium service pumps (P-401 and P-402).

63. Disconnect the electrical leads from the pump coils and remove the coils from the vault. Store as per CPPD's instructions.

64. Cut a section from sodium line No. 117 (3-in. diameter primary service pump No. 1 header) where it connects to the pump throat. Install temporary seals (plastic caps) over the exposed pipe openings; four seals required.

65. Cut a section from sodium line No. 119 (3-in. diameter primary sodium service pump No. 1 header) where it connects to the pump throat.

66. Install temporary seals (plastic caps) over the exposed pipe openings (four seals required). Remove the pump from the vault.

67. Remove the residual sodium from line stubs No. 117 and No. 119 on the pump throat, and install capping plates over the line stubs. Dye penetrant inspect the final pass of the capping plate weldments.

68. Store the pump as per CPPD's instructions.

69. Remove sodium service pump No. 2 as per Steps 62 through 68. The sodium lines connecting to the pump throat are No. 118 and No. 120.

70. Cut a section from sodium line No. 160 (2-in. diameter primary plugging meter No. 1 inlet) where it connects to the plugging meter economizer. Install temporary seals (plastic caps) over the exposed pipe openings. Remove the pipe spool from the vault to a specified storage area.

71. Cut a section from sodium line No. 163 (2-in. diameter primary plugging meter No. 1 outlet) where it connects to the plugging meter economizer. Install temporary seals (plastic caps) over the exposed pipe openings and remove the pipe spool from the vault to a specified storage area.

72. Remove plugging meter No. 1 support devices (pipe hangers) and remove the plugging meter assembly from the vault.

73. Remove the temporary seals from line stubs No. 160 and No. 163 on the plugging meter assembly.

74. Remove the residual sodium from line stubs No. 160 and No. 163 on the plugging meter assembly and install capping plates over the line stubs. Dye penetrant inspect the final pass of the capping plate welds.

75. Store the plugging meter as per CPPD's instructions.

76. Remove primary plugging meter assembly No. 2 as per procedural Steps 70 through 75.

> NOTE: Due to the physical location of the primary sodium system drain tank, which is the low point in the system, it shall be removed when enough piping has been taken from the service vault to provide clearance for its removal.

77. Cut and remove a section from sodium line No. 106 (2-in. diameter carbon trap economizer vessel. Install temporary seals (plastic caps) over the exposed pipe openings; four seals required. Remove the pipe spool from the work area and store in a specified storage area.

78. Cut and remove sodium line No. 321 (2-in. diameter carbon trap economizer outlet to carbon trap) one cut is to be made where it connects to the economizer vessel the second cut at the carbon trap vessel. Install temporary seals (plastic caps) over the exposed pipe openings. Four seals required. Cut helium line No. 1172 (1/2-in.) and R/A vent line No. 2430 (3/4-in. diameter freeze trap No. 16 R/A vent), and tape seal the exposed pipe openings. Remove the loose pipe spool from the area including freeze trap No. 16.

79. Cut and remove a section from sodium line No. 107 (2-in. diameter carbon trap economizer outlet) where it connects to the economizer vessel. Install temporary seals (plastic caps) over the exposed pipe openings. Four seals required. Remove the pipe spool from the work area and store in a specified storage area.

80. Cut and remove sodium line No. 323 (2-in. diameter carbon trap heater to economizer line) one cut to be made where it connects to the economizer vessel and one cut to be made at the heater. Install temporary seals (plastic caps) over the exposed pipe openings. Four seals required. Cut helium line No. 1173 and R/A vent line No. 2431 to freeze trap No. 17. Remove the pipe spool including freeze trap No. 17 from the area and store in a specified storage area.

81. Remove the carbon trap economizer (X-407) support devices and remove the economizer from the vault.

82. Remove the temporary seal from line stub No. 106 on the economizer. Remove the residual sodium from the interior of the pipe stub and install a capping plate. Dye penetrant inspect the final pass of the capping plate weldment.

83. Capping plates shall be installed on line Stubs 107, 321, and 323 on the carbon trap economizer per Step 82. Store the economizer per CPPD's instructions.

84. Cut sodium line No. 322 (3-in. diameter carbon trap Na outlet) and install temporary seals (plastic caps) over the exposed pipe openings; two seals required.

85. Remove sodium heater (H-401) including the attached line stubs No. 322 and No. 323 from the area.

86. Cut the attached pipe sections from the heater vessel; remove residual sodium; and install capping plates. Dye penetrant inspect the final pass of the capping plate weld.

87. Remove the support devices from the carbon trap vessel and remove it from the vault.

88. Install capping plates on line stubs No. 321 and No. 322 on the carbon trap vessel as specified by step No. 82.

89. The remainder of the primary sodium service system consists of piping valves, freeze traps, and vapor traps. Cut the pipe in suitable lengths for removal and cleaning. All exposed pipe openings shall be covered with plastic caps secured with tape. Freeze traps and vapor traps will be removed from pipes and shall be sealed by welding capping plates onto exposed openings.

90. Clearly identify piping with "metallic sodium" and "radioactive" tags before removal from the work area.

91. Remove piping to storage area as designated by Health and Safety.

92. A daily progress report will be made and delivered to CPPD. This report will include manpower, unusual problems encountered, and any unusual conditions that may have developed in the course of the work.

PROCEDURE 7.

REMOVAL OF THE PRIMARY SODIUM BLOCK AND THROTTLE-VALVES, AND ISOLATION OF THE REACTOR FROM ITS RELATED SODIUM PORTS

I. PURPOSE

This procedure establishes a step-by-step sequence for removal of the primary system block and throttle valves, check valves, and moderator coolant line, including its valves; and for isolation of the reactor and the related primary heat transfer loops in preparation for steam cleaning the reactor and heat transfer loops.

II. TASK DESCRIPTION

The activities to be accomplished by this procedure are as follows.

Disconnect and remove the electrical service to the primary system block and throttle valves and adjacent sodium pipes; disconnect and remove the temperature indicating instrumentation from the primary block and throttle valves; remove the nitrogen cooling ducts from the vicinity; remove the valve operator gear boxes, shafting, and other miscellaneous items from the work area; remove the thermal insulation from the valve bodies and the connecting pipes as necessary to permit working access; cut and remove sections from the guard pipes; cut the sodium lines connecting to the block and throttle valves; remove the block and throttle valve assemblies; install ring type sealing flanges on the guard pipes; and install valve equipped pipe caps on the sodium lines.

III. CRITERIA

A. REQUIREMENTS AND PREREQUISITES

1. This activity shall be coordinated with other plant activities during the HNPF retirement program.
2. The root pass of all pipe cap weldments specified in Section IV of this procedure will be made by the tungsten inert gas welding process with an inert gas backup.
3. Dry ice shall be used as a heat sink when welding on sodium lines, provided the particular location requires it.
4. All operations of plant equipment and deenergizing of electrical circuitry required in the performance of this procedure shall be performed only by CPPD personnel.
5. Notification shall be made to the CPPD shift supervisor prior to initiating this work and <u>also</u> immediately prior to making any penetration into piping or components which may affect, or require special operation of, any associated plant system or components.
6. Personnel assigned to specific noninterruptable tasks (such as pipe cutting and handling) shall not be given additional assignments until their particular phase of that task is completed.
7. The inflatable bladder technique shall be used to maintain an inert atmosphere on the reactor vessel and the primary heat transfer loops during the performance of Section IV of this procedure.

8. The items that are being removed (valves and pipe spools) shall be properly supported prior to starting their removal. This requirement will also apply to piping being left in position.

9. The pipe caps specified for sodium lines No. 231, 232, 233, 237, 238, and 239 will each be fitted with a 4-in. diameter SS pipe nipple prior to their installation. These pipe nipples will be fitted with 2-in. reducers and 2-in. valves after the pipe caps have been installed and the inflatable seals removed. The 2-in. valves will be sealed with pipe plugs. Line No. 148 (3-in. diameter reactor drain line) and line No. 284 (6-in. diameter moderator coolant line) shall be fitted with 2-in. diameter SS pipe nipples. These pipe nipples will be sealed with 2-in. valves and pipe plugs.

10. Tools and equipment required for the work detailed in this procedure are as follows.

    a. Hand tools - wrenches, pliers, hammers, lagging saws, tin snips, wire cutters, screw drivers, scrapers, etc.

    b. Band saw - Porter-Cable (3-in. throat)

    c. Pipe saw - Fein or equivalent

    d. Welding machine - equipped for inert gas welding

    e. Dye penetrant weldment inspection kit

    f. Oil manometer - equipped with connecting hose

    g. Ohmmeters

    h. Four portable drop cord lights with safety guards

    i. Six - 14-in. diameter type 304 SS Schedule-10 pipe caps, butt weld

    j. Six - 16-in. diameter type 304 SS Schedule-10 pipe caps, butt weld

    k. One - 6-in. diameter type 304 SS Schedule-40 pipe cap, butt weld

    l. One - 3-in. diameter type 304 SS Schedule-40 pipe cap, butt weld

    m. Scaffolding as required

    n. Hoisting and rigging gear as required

    o. Portable oxygen analyzer - equipped with audible alarm

    p. Bladders - sized for sealing 6-in. to 16-in. pipe - 4 each

    q. Three ring flanges - 1/4-in. thick, type 304 SS, 17-3/8-in. OD - 14-1/8-in. ID

    r. Three ring flanges - 1/4-in. thick, type 304 SS, 19-3/8-in. OD - 16-1/8-in. ID

    s. One ring flange - 1/4-in. thick, 9-15/16-in. OD - 6-3/4-in. ID

    t. One ring flange - 1/4-in. thick, 5-15/16-in. OD - 3-5/8-in. ID

    u. Dry ice - procure as needed

    v. Ten hole saws - 4-in. diameter

    w. Two hole saws - 2-1/2-in. diameter

    x. Portable 1/2-in. electric drill motor

y. Supply of dry calcium carbonate

z. Power source required for pipe saw, 220 volt, 3 phase

za. Three expandable rubber stoppers for 3-in. diameter pipe

zb. Sixteen - 2-in. valves - steam service

11. Plastic caps or other temporary sealing devices shall be used on exposed sodium piping. Metal seal plates shall be welded in place before leaving the work area unattended provided time permits. However, in cases where temporary seals are used for extended periods suitable surveillance of the area shall be maintained.

12. The following prerequisite items shall be completed and signed off prior to initiating performance of the work detailed in Section IV, "Method of Performance and Sequence of Events," of this procedure.

    a. Attach to this procedure a copy of the appropriate P and I drawing marked to show the equipment and system isolation to be performed.

    NOTE: Drawings shall be marked in accordance with physical inspection of the system

    b. Personnel assigned to the task shall study the procedure and sign same.

    c. Potential hazards relating to the work to be performed shall be discussed with the personnel involved before the operation is initiated.

    d. Electrical power to the block valves, throttle valves, and primary piping heaters shall be deenergized and tagged "out of service" in accordance with normal HNPF tagging procedures.

    e. Confirm that the entire primary heat transfer system is at ambient temperature prior to initiating the cutting of any of the sodium piping.

    NOTE: The only exceptions to item (e) above are the portions of the primary system lines which connect to the reactor core vessel. The reactor core vessel shall remain preheated to ~300°F; therefore, the sodium lines within the confines of the reactor cavity shall remain above ambient temperature due to heat conduction.

    f. The primary sodium fill tanks shall have been isolated from their related sodium systems prior to initiation of Section IV, "Method of Performance and Sequence of Events" of this procedure.

    g. The primary sodium service system lines connecting to the primary heat transfer loops shall have been cut and capped prior to initiating Section IV of this procedure.

B. HEALTH AND SAFETY

1. Special work permits are required for all penetrations, cutting, handling, and welding of sodium piping and for sodium handling operations, and for all vault entries.

2. Sodium protective clothing shall be worn at all times by all personnel while within the working areas of sodium pipe cutting, handling, cleaning, welding and for handling or disposal of sodium scrap.

3. The buddy system is required for all sodium handling, pipe cutting, and welding operations and all vault entries.

4. Deposit all sodium scrap in metal containers containing calcium carbonate and deliver them to the storage location designated by Health and Safety. All primary R/A sodium must be labeled as such and segregated from non-R/A sodium.

   NOTE: No sodium scrap or sodium contaminated component may be left open or unattended except as specifically authorized by Health and Safety.

5. Calcium carbonate and sodium fire fighting equipment must be kept standing by at all times during cutting, cleaning, welding of sodium piping, or handling of sodium scrap.

6. Disposal of sodium contaminated equipment and materials shall be performed only as directed by Health and Safety.

7. The portable oxygen analyzer shall be used to continuously monitor the work area atmosphere; audible alarm shall be set at 19% oxygen.

C. STANDARDS

Work performed in connection with this activity shall be carried out under normal HNPF Health and Safety regulations (including sodium safety regulations) in accordance with Part II of the HNPF Operations Manual and specific requirements of this procedure. Pipe cap and ring flange welds shall be dye penetrant inspected.

IV. METHOD OF ACCOMPLISHMENT AND SEQUENCE OF EVENTS

1. All preparatory work shall be completed before starting the work of this section.

2. Confirm that the primary heat transfer system preheat electrical supplies have been deenergized and tagged out per Part III, Section A, Step 12d. Disconnect the electrical supplies to the following sodium valves by removing fuses and/or by lifting the leads at the breakers, V-101, V-103, V-104, V-201, V-203, V-204, V-301, V-303, V-304, V-001, V-002, and sodium drain line No. 148 where it connects to the reactor seal plate. All electrical leads must be taped with electrical tape.

3. Remove electrical conduits, wire tray, and junction boxes only as necessary to permit removal of the above specified sodium valves and to install capping plates on the related sodium lines, reference step No. 2.

4. Disconnect and remove the temperature indicating instrumentation from the above specified sodium valves, reference step No. 2.

5. Confirm that the sodium leak detection system has been deenergized and tagged out per tag-out procedure. Disconnect the electrical supplies to the sodium leak detectors on the sodium valves specified in procedural step No. 2.

6. Remove the conduits, wire trays, and junction boxes only as necessary to permit access for working.

7. Remove the thermal insulation from the items specified in step No. 2. The insulation will be removed only as necessary to permit removal of the valves and installation of capping plates on the related pipes.

8. Remove the heater reflector oven from the heaters on the valve bodies and sodium line No. 148 at the reactor seal plate.

9. Remove the heaters from the valve bodies and one set on line No. 148 ~6 in. from where it enters the reactor cavity liner.

10. The line heaters on each side of the specified sodium valves will not be removed; however, it will be necessary to remove the heater holding devices sufficiently to permit bending the heaters away from the pipes on both sides of the valve bodies to pe..nit access for cutting the lines.

11. Close and tag out the inert gas supply valves to the reactor liner cavity (helium system), V-8109.

12. Reduce the inert gas pressure on the reactor cavity to 0-gage pressure by operating R/A vent system valve V-10142.

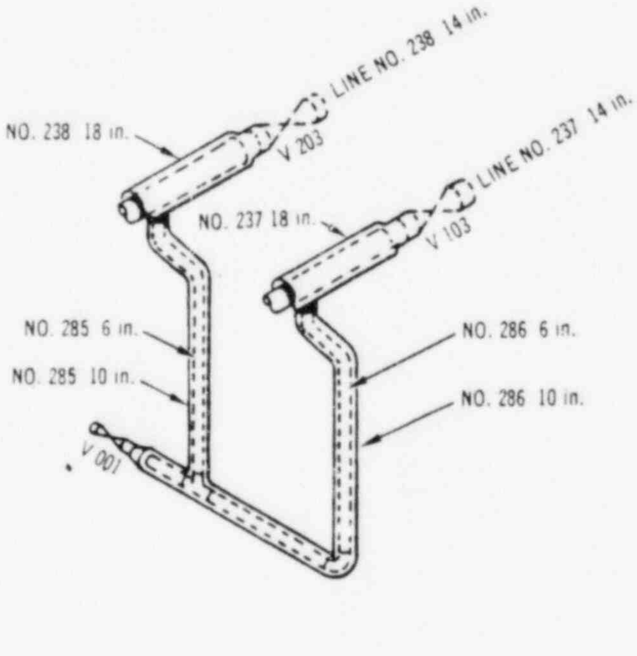
13. Cut line No. 237 (18-in. guard pipe) near the point where it connects to sodium valve No. 103 (14-in. sodium throttle valve, loop No. 1), see sketch attached.

14. Cut line No. 237 (18-in. guard pipe) on the reactor side of the junction (pipe tee) of line No. 237 and line No. 286 (10-in. guard pipe).

15. Cut line No. 286 (10-in. guard pipe) ~16-in. below the point where it connects to the pipe tee (junction of guard pipes No. 237 and No. 286).

16. Cut the loose section of guard pipe longitudinally to permit removing it.

17. Cut line No. 238 (18-in. guard pipe) near the point where it connects to sodium valve V-203 (14-in. throttle valve, loop No. 2).



18. Cut line No. 238 (18-in. guard pipe) on the reactor side of the junction (pipe tee) of guard pipes No. 238 and No. 285.

19. Cut line No. 285 (10-in. guard pipe) ~16 in. below the point where it connects to the pipe tee (junction of guard pipes No. 238 and No. 285).

20. Cut the loose section of guard pipe longitudinally to permit removing it.

21. Cut line No. 284 (10-in. guard pipe) near the point where it connects to sodium valve V-002 (6-in. moderator coolant inlet valve).

22. Cut line No. 284 (10-in. guard pipe) ~2 ft downstream of sodium valve V-002 toward the reactor; remove the guard pipe.

CAUTION

> The inert gas pressure in the reactor core and the primary heat transfer loops will be at ~1/4-in. w.g. pressure. Personnel will be suited in protective clothing and equipped as per instructions from CPPD Health and Safety to work contaminated sodium.

23. Penetrate sodium line No. 237 (14-in. reactor inlet line, loop No. 1) with a 4-in. diameter hole saw at the exposed pipe tee (junction of sodium lines No. 237 and No. 286).

24. Insert an inflatable bladder into the sodium line No. 237, and position it toward the reactor vessel just beyond the point where sodium line No. 286 connects to sodium line No. 237.

25. Inflate the bladder sufficiently to seal sodium line No. 237.

26. Tape seal the hole saw penetration in sodium line No. 237.

27. Penetrate sodium line No. 238 (14-in. reactor inlet line, loop No. 2) with a 4-in. diameter hole saw at the exposed pipe tee (junction of sodium lines No. 238 and No. 285).

28. Insert an inflatable bladder into sodium line No. 238 and position it toward the reactor vessel just beyond the point where sodium line No. 285 connects to sodium line No. 238. Inflate the bladder sufficiently to seal sodium line No. 238. Tape seal the hole saw penetration in sodium line No. 238.

29. Penetrate sodium line No. 284 (6-in. moderator coolant inlet line) with a 2-1/2-in. diameter hole saw on the reactor side of sodium valve V-002.

30. Insert an inflatable seal into sodium line No. 284 and position it toward the reactor. Inflate it sufficiently to seal the pipe. Tape seal the hole saw penetration.

31. Penetrate sodium line No. 237 near the point where sodium line No. 237 connects to check valve No. 104 (piping system side) with a 4-in. diameter hole saw.

32. Insert an inflatable seal into sodium line No. 237 and position it upstream of piping system side of the hole saw penetration.

33. Inflate the seal sufficiently to seal the pipe and tape seal the hole saw penetration.

34. With a 4-in. diameter hole saw penetrate sodium line No. 238 (14-in. diameter reactor inlet line, loop No. 2) near the point where it connects to check valve V-204 (piping system side of V-204). Insert an inflatable seal into sodium line No. 238 and position it upstream of piping side of the hole saw penetration.

35. Inflate the seal sufficiently to seal the pipe and then tape seal the hole saw penetration.

NOTE: All pipe cuts are to be made so that the hole saw penetrations will be removed with the pipe spool and will not have to be resealed.

36. Cut sodium line No. 286 (6-in. moderator coolant line) near where it connects to the pipe tee in sodium line No. 237; tape seal the cut.

37. Cut sodium line No. 285 (6-in. moderator coolant line) near where it connects to the pipe tee in sodium line No. 238; tape seal the cut.

38. Cut sodium line No. 285 (6-in. moderator coolant line) near where it connects to the downstream side of sodium valve V-001 (4-in. moderator coolant block valve).

39. Lower the loose piping assembly sufficiently to permit installation of temporary seals (plastic caps) over the exposed pipe openings. Six seals are required.

40. Complete removal of the piping assembly from the work area and lay it aside.

NOTE: The configuration of the piping assembly is such that it will not pass through the vault access opening; therefore, it will be cut into pieces.

41. Cut sodium line No. 237 on the piping system side of check valve No. 104. The inflatable seal is to remain in position on the piping side of the cut; tape seal cut.

42. Cut sodium line No. 237 on the reactor side of the pipe tee (junction of sodium lines No. 237 and No. 286). The inflatable seal is to remain in position on the reactor side of the cut.

43. Hoist the check-valve, throttle-valve assembly, including the pipe tee, upward to permit installation of temporary seals over the pipe openings; four seals (plastic caps) required. If necessary because of the confined working area, the throttle valve may be separated from the check valve and removed from the area separately.

44. Hoist the check-valve, throttle-valve assembly from the pipe vault and store per CPPD's instructions.

45. Cut sodium line No. 238 on the piping system side of check valve No. 204 just upstream of the hole saw penetration. The inflatable seal is to remain in place on the piping side of the cut.

46. Cut sodium line No. 238 on the reactor side of the pipe tee (junction of lines No. 238 and No. 285). The inflatable seal is to remain in position on the reactor side of the cut.

47. Hoist the check valve-throttle valve assembly, including the pipe tee, upward sufficiently to permit installation of temporary seals over the exposed pipe openings; four seals (plastic caps) required.

48. Remove the temporary seal from line stub No. 237 (reactor side).

49. Remove the residual sodium from the interior of the line to a depth of ~10 in.

50. Position a 17-3/8-in. OD by 14-1/8-in. ID SS ring flange in the annulus between the guard pipe and sodium line No. 237. The flange must be in position before welding the pipe cap on.

51. Prepare sodium line stub No. 237 for welding (chamfer end of pipe) and position a 14-in. SS pipe cap with 4-in. diameter nipple on the line stub.

52. Weld a pipe cap to the sodium line stub No. 237. Leak testing of the weld root pass is to be made by the tungsten inert gas method.

> NOTE: The inflatable seals must remain in position until all the pipe caps are seal welded in place; they shall then be removed through the 4-in. diameter nipples.

53. Position the ring flange ~2 in. into guard pipe No. 237; weld it to the interior surface of the guard pipe and the exterior surface of sodium line No. 237.

54. Install a 14-in. SS pipe cap on sodium line stub No. 238 (reactor side) as per procedural steps No. 48 through No. 53. This installation shall include the ring flange that seals the annulus between the guard pipe and the sodium line.

55. Cut sodium line No. 284 (6-in. moderator coolant inlet line) into random lengths to permit removing it from the pipe vault. Visual inspection of the line indicates that approximately three cuts are required. The final cut shall be made near the point where the line attaches to sodium valve V-002 (upstream side of valve). Install temporary seals over the exposed line stubs. This removal shall include freeze trap No. 14 and its related helium and vent line.

56. Cut sodium line No. 284 (6-in. moderator coolant line) on the reactor side of V-002. The expandable seal is to remain in position. Install temporary seals over the exposed line stubs and remove valve V-002 from the area.

57. Remove the temporary seal from line stub No. 284 and remove the residual sodium from the interior of the pipe up to the inflated seal or to a minimum depth of 8 in.

58. Prepare line stub No. 284 for welding (chamfer the pipe).

59. Position a 9-15/16-in. OD by 6-3/4-in. ID ring flange in the annulus between the guard pipe and line No. 284.

60. Position a 6-in. diameter SS pipe cap on sodium line stub No. 284 and weld it to the line stub.

> NOTE: The inflatable seal must be located in sodium line No. 284 such that welding the ring flange will not damage or burn it.

61. Position the ring flange ~2-in. into the guard pipe and weld it to the interior surface of the guard pipe and to the exterior surface of sodium line No. 284.

62. Install 14-in. diameter SS pipe caps on sodium line stubs No. 237 and No. 238 (piping system side) as in procedural steps No's. 48, 49, 51, and 52.

63. Cut line No. 231 (20-in. diameter guard pipe) near the point where it connects to the reactor side of primary block valve V-101 and near the bellows toward the reactor; this pipe spool is ~2-in. long.

64. Cut the loose pipe spool longitudinally to permit its removal.

65. With a 4-in. diameter hole saw, penetrate sodium line No. 231 (16-in. diameter reactor outlet line, loop No. 1) near the point where the pipe connects to the sodium valve V-101 (reactor side of valve).

66. Position an inflatable seal inside sodium line No. 231 upstream or toward the reactor and inflate it sufficiently to seal the pipe. Tape seal the hole saw penetration.

67. With a 4-in. diameter hole saw, penetrate sodium line No. 231 near the point where the pipe connects to the valve body, V-101 (piping system side).

68. Position an inflatable seal on the downstream (piping system side) of the hole saw penetration, and inflate it sufficiently to seal the pipe; tape seal the hole saw penetration.

69. Cut sodium line No. 231 on the reactor side of valve V-101 so that the hole saw penetration is removed with the valve; tape seal the cut.

70. Cut sodium line No. 231 on the piping system side of valve V-101 downstream of the pipe system side of the hole saw penetration.

71. Hoist sodium valve V-101 upward sufficiently to permit installation of temporary seals over the exposed pipe openings; four seals required.

72. Hoist the valve from the pipe vault and store it per CPPD's instructions.

73. Remove the temporary seal from sodium line stub No. 231 (reactor side), and remove the residual sodium from the interior of the line to a depth of ~10 in. The inflatable seal shall be retained in place.

74. Position a 19-3/8-in. OD by 16-1/8-in. ID ring flange in the annulus between the guard pipe and the sodium pipe.

75. Prepare sodium line stub No. 231 for welding (chamfer the end of the pipe).

76. Position a 16-in. diameter SS pipe cap and weld it to line stub No. 231.

77. Insert the ring flange ~2 in. into guard pipe No. 231; weld it to the interior surface of the guard pipe and to the exterior surface of sodium line No. 231.

78. Install a 16-in. diameter SS pipe cap on line stub No. 231 (piping system side) as per procedural steps No's. 73, 74, and 76.

79. Cut and remove a 3 ft section from line No. 148 (6 in guard pipe) ~6 ft from the point where it connects to the gallery seal plate.

80. Cut sodium line No. 148 (3-in. diameter reactor fill and drain line) ~3-in. from the end of the guard pipe (reactor side of exposed portion of sodium line No. 148); tape seal the cut.

81. Cut sodium line No. 148 ~3 in. from the guard pipe on the piping system side of the exposed portion of sodium line No. 148.

82. Remove the pipe spool and install temporary seals over the exposed pipe openings; four seals required. Remove the pipe spool from the area to storage.

83. Remove the temporary seal from sodium line stub No. 148 (reactor side) and insert an expandable stopper into the line to a depth of 6 in. Expand the stopper sufficiently to seal the pipe.

84. Remove the residual sodium from the interior of the pipe to a depth of ~4 in.

85. Prepare sodium line stub No. 148 for welding (chamfer end of pipe).

86. Position a 5-15/16-in. OD by 3-5/8-in. ID ring flange in the annulus between guard pipe No. 148 and sodium line No. 148.

87. Remove the expandable rubber stopper from the line stub, and position the pipe cap on the line. Tape seal the annulus between the pipe and pipe cap.

88. Remove the tape sufficiently to permit tack welding the pipe cap at approximately three places around its circumference. Replace the tape to prevent excessive inert gas leakage.

89. Proceed to make the weldment root pass; remove the tape as necessary to permit welding.

90. Complete the weldment. The inert gas pressure in the reactor shall be carefully controlled to prevent blowing a hole in the weld.

91. Position the ring flange ~1 in. into the guard pipe No. 148; weld it to the interior of the guard pipe and to the exterior surface of sodium line No. 148.

92. Cut the remaining section of sodium line No. 148 into random lengths to permit removing it from the area. This removal will include sodium valve V-459. Temporary seals (plastic caps) shall be installed on all exposed pipe openings.

> NOTE: The preceding procedural steps that pertain to the removal of sodium valves V-101, V-103, V-104, V-203, V-204 and to capping of their related sodium lines shall be applicable to removal of sodium valves V-201, V-301, V-303, and V-304 and to capping of their related sodium lines.

93. Dye penetrant inspect all pipe cap weldments. Repair welds if necessary and inspect repairs.

94. Remove the inflatable seals from the sodium lines indicated below. The seals shall be deflated and pulled through the 4-in. diameter pipe nipple on the pipe caps. When a seal is removed, a 4-in. to 2-in reducer and a 2-in. valve shall be installed (welded) onto the 4-in. diameter pipe nipple. To insure a positive seal, install a pipe plug in each valve.

Sodium Line Number

L237 - Reactor Side
L237 - System Side
L238 - Reactor Side
L238 - System Side
L231 - Reactor Side
L231 - System Side
L232 - Reactor Side
L232 - System Side
L233 - Reactor Side
L233 - System Side
L239 - Reactor Side
L239 - System Side
L284 - Reactor Side

95. Dye penetrant inspect the ring flange weldments and report the results to CPPD in writing.

96. A daily progress report shall be made and delivered to CPPD. This report shall include manpower, unusual problems encountered, and any unusual conditions that may have developed in the course of this work.

97. By signature acknowledge that this procedure has been studied and understood, RDP 5-2, Removal of the Primary Block and Throttle Valves and Isolation of the Reactor from its Related Sodium Systems.

98. Procedure RDP 5-2. In reference to removal of the Primary Sodium Block and Throttle Valves, and Isolation of the Reactor from its Related Sodium Systems, the following statement shall be signed. Work was completed this date in accordance with this procedure and that attached support drawings, weldment inspection reports, and other data as listed below. Indicate the data; attach drawings; and list pertinent data.

PROCEDURE 8.

ISOLATION OF INTERMEDIATE HEAT EXCHANGER MODULES FROM THEIR RELATED PRIMARY SYSTEM

I. PURPOSE

This procedure establishes a step-by-step sequence for isolating the three intermediate heat exchangers (six modules) from their related primary systems, as a part of the retirement of the primary heat transfer system and to permit the removal of the heat exchangers for off-site shipment.

II. TASK DESCRIPTION

The activities to be accomplished by this procedure are as follows.

Disconnect and remove the electrical service to the primary system piping in the immediate area of the intermediate heat exchangers; disconnect and remove the temperature indicating instrumentation from the heat exchangers primary system lines; disconnect and remove the sodium leak detectors from the intermediate heat exchanger vessels; remove conduits, wire tray, and junction boxes from the vault as necessary to facilitate removal of the heat exchanger modules; remove thermal insulation and line heaters as required; cut the primary sodium lines connected to the heater exchanger modules; install capping plates on the heat exchanger vessel line stubs; pressurize the vessels with inert gas for subsequent off-site shipment and storage; and install pipe caps and an interconnecting line on the line primary system line stubs in preparation for steam cleaning the piping systems.

III. CRITERIA

A. REQUIREMENTS

1. This activity shall be coordinated with other plant activities during the HNPF retirement program.
2. All operation of plant equipment and deenergizing of electrical circuitry required in the performance of this procedure shall be performed only by CPPD plant personnel.
3. Prior to cutting sodium lines or removing pipe support devices (pipe hangers), temporary support shall be provided. The temporary support may be by use of chain falls and lifting slings or by other safe devices.
4. Notification shall be made to the CPPD shift supervisor prior to initiating this work and also immediately prior to making any penetrations into piping or components which may affect, or require special operation of, any associated plant system or components.
5. Exposed electrical leads shall be checked for voltage prior to making physical contact with the wire.
6. Personnel assigned to specific noninterruptable tasks (such as pipe cutting, closure welding, sodium removal, and handling) shall not be given additional assignments until their particular phase of that task is completed.

1) Work items of Activities 1 through 9 pertinent to that volume shall be completed.

2) Any Category D items to be disposed of in that volume shall be placed therein.

3) Any necessary radiological decontamination of components on surfaces in that volume shall be performed in accord with Activity Specification No. 7 criteria.

4) The final radiological status of the volume shall be recorded.

After the above have been accomplished, penetrations to the Isolation Structure will be sealed; an access and weatherproof protective covering will be placed over exposed horizontal surfaces of the Reactor Building floor structure and the IHX cells structure, and the covering will be suitably integrated to these structures to meet the access and weatherproof requirements specified in Part C of this section. Details concerning the proposed nature of the weather and access proof protective covering, and the proposed nature of sealing penetrations, have been presented elsewhere.* General criteria for the protective covering and penetration seals are listed in Part C of this specification. Further detailed design of the protective covering and penetrations seals will be carried out as outlined in subsequent paragraphs of this section of this specification. Prior to the placement of the protective covering, a determination shall be made as to what portion of the existing Reactor Building structure must be demolished, if any, in order to render the premises radiation safe in the most feasible and economic fashion. For this determination, the following ground rules shall apply.

1) The protective covering and penetration closures selected must meet the radiological and engineering criteria of Parts B and C of this section.

2) Existing above grade Reactor Building structures shall be removed only if such removal is economically advantageous to the Atomic Energy Commission and/or necessary for rendering the premises radiation safe.

3) The removal of the building shall be deemed economically advantageous if any of the following three evaluations are found to result in a net financial gain to the Commission:

   a) The cost for removing the structure vs any gain from resulting scrap sales,

   b) The difference in cost of the penetration seals and protective covering required to meet the criteria of Parts B and C which would be incurred if the structure remains in place rather than being removed, and

   c) Any other economic advantages and/or monetary reimbursements which the Commission may accrue if the Reactor Building structures are removed and/or modified in some specified manner.

4) For purposes of determination of the value of the Reactor Building and its associated contents as scrap to a prospective demolition or modification subcontractor, the following definition of the Reactor Building structure shall apply.

The Reactor Building structure is considered to consist of all above grade structures and hardware south of Column Line 13 with exception of the following: the HNPF Isolation Structure itself as defined in Section II of this specification, any items on AEC excess property lists for which CH approved purchase orders are received, and any other property whose disposal has been specified in Activity Specifications 1 through 9.

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*NAA-SR-MEMO-12263, "Study of Methods for Permanent Sealing of Reactor and Cavity," and NAA-SR-MEMO-12270, "HNPF-Proposed Methods for Closing, Sealing, and Disposition of Structures"

10. The inflatable seal technique shall be used to prevent air from entering the heat exchanger modules and the heat transfer loops.

11. The following prerequisite items shall be completed and signed off prior to initiating performance of the work detailed in Section IV, "Method of Performance and Sequence of Events," of this procedure.

    a. Attached to this procedure there shall be a copy of the appropriate P and I drawing marked to show the equipment and system isolation to be performed.

    b. Responsible personnel assigned to the task shall study and sign the procedure.

    c. Potential hazards relating to the work to be performed shall be discussed with the personnel involved before the operation is initiated.

    d. Electrical power to the IHX and primary heat transfer system piping heaters in the immediate area of the IHX shall be deenergized and tagged "out of service" in accordance with normal HNPF tagging procedure.

    e. Electrical power to the IHX sodium leak detectors shall be deenergized and tagged "out of service" in accordance with HNPF tagging procedure.

    f. Check to confirm that the primary heat transfer system piping is at ambient temperature prior to initiating the cutting of any of the sodium piping.

B. HEALTH AND SAFETY

1. Special Work Permits are required for all penetrations, cutting, handling, cleaning, welding, and handling or disposal of sodium scrap.

2. Approved protective clothing shall be worn at all times by those personnel directly involved in making penetrations, cutting, handling, cleaning, and welding of sodium piping.

3. The buddy system is required for all vault entries, sodium handling, pipe cutting, and welding operations.

4. Deposit all sodium scrap in metal containers containing calcium carbonate and deliver them to the storage location designated by Health and Safety.

NOTE

No sodium scrap or sodium contaminated component may be left open or unattended except as specifically authorized by Health and Safety.

5. Calcium carbonate and sodium fire fighting equipment must be immediately available at all times during cutting, cleaning, and welding of sodium piping, and during handling of sodium scrap. No water shall be permitted in the area.

6. The portable oxygen analyzer shall be used to continuously monitor the work area atmosphere, with its associated audible alarm set at 19% oxygen.

7. Disposal of sodium contaminated equipment and materials shall be performed only as directed by Health and Safety.

c) Appropriate radiological surveys will be conducted, and any necessary radiological decontamination of components or surfaces in that volume shall be performed in accordance with Activity Specification No. 7 criteria.

d) The final radiological status of the volume shall be recorded.

2) Penetrations to the Isolation Structure will be sealed in accordance with the criteria in Part III-C of this specification.

3) Reactor Building structures removal and/or modification necessary to permit installation of the access and weatherproof protective covering will be initiated and the protective covering will be installed and integrated to the designated concrete structures to produce the completed Isolation Structure which will meet the criteria of Parts III-B and III-C of this specification.

4) Further removal and/or modification of Reactor Building structures will be performed in accord with the results of engineering feasibility and economic studies performed as outlined in Part III-A of this specification. In order for such removal and/or modifications to be authorized, the above studies must show that the proposed operations are an engineering necessity to render the HNPF site radiation safe and/or such operations will be economically advantageous to the Atomic Energy Commission.

10. Pressure shall be maintained at 1/4 in. w. g. by operating the following valves.

Loop No. 1	V-847	Modules 1A and 1B Freeze Trap No. 11 Helium Supply Throttle Valve
	V-844	Modules 1A and 1B Freeze Trap No. 11 Helium Supply Block Valve
Loop No. 2	V-848	Modules 2A and 2B Freeze Trap No. 12 Helium Supply Throttle Valve
	V-845	Modules 2A and 2B Freeze Trap No. 12 Helium Supply Block Valve
Loop No. 3	V-849	Modules 3A and 3B Freeze Trap No. 13 Helium Supply Throttle Valve
	V-846	Modules 3A and 3B Freeze Trap No. 13 Helium Supply Block Valve

CAUTION

The personnel performing the sodium line cutting operation shall suit in the proper protective clothing as specified by Health and Safety and detailed in the Special Work Permit issued for this work. Notify CPPD Shift Supervisor prior to making each penetration into sodium lines.

NOTE

An inert gas atmosphere will be maintained on the heat exchangers during the isolation operation.

11. Penetrate sodium line No. 242 (14 in. diameter – Module 1A outlet) with a 4-in. diameter hole saw near the point where it connects to Module 1A outlet pipe elbow. Pipe cuts are to be made on the horizontal section of piping.
12. Remove the hole saw blank and immediately install inflatable seals inside the line. Insert one into Module 1A outlet pipe elbow and one downstream of hole saw penetration. Inflate the seals sufficiently to seal the pipe.
13. Penetrate sodium line No. 243 (14-in. diameter – Module 1B outlet) with a 4-in. diameter hole saw near the point where it connects to Module 1B outlet pipe elbow.
14. Remove the hole saw blank and immediately install inflatable seals inside the line. Insert one into Module 1B outlet elbow and one downstream of the hole saw penetration. Inflate the seals sufficiently to seal the pipe.

The most economic alternative, where technically feasible alternatives exist, will be used for disposal of a given item.

B. HEALTH AND SAFETY

Operations connected with this activity will be carried out under normal HNPF and CPPD safety regulations. Heavy equipment lifting and moving will be conducted in accordance with the CPPD Safety Manual. Radiological safety will be in accordance with the HNPF Operations Manual, Volume Ia, Part III. Those major components, for which it is uneconomical to meet the limits of Section V, will be evaluated by AI on a case basis to determine their ultimate disposition, with AEC approval. Radioactive contamination limits for components which are to be left in place or shipped off-site as scrap for unrestricted use are contained in Section V of this activity specification. Criteria for disposal of radioactively contaminated items within the HNPF Isolation Structure are contained in Section VI of Activity Specification Number 7.

C. STANDARDS

Operations connected with this activity will be conducted in accordance with good engineering practice. No special nonradiological standards are deemed necessary.

IV. METHOD ACCOMPLISHMENT AND SEQUENCE OF EVENTS

Items for which approved CH purchase orders are received (Category A items) will be disposed of in accordance with their respective removal requirements. Remaining items will be disposed of as follows.

1) Piping in the pipe tunnel connecting the Reactor Building with the Radioactive Waste Facility will be:

   a) Externally and internally decontaminated to the unrestricted use disposal limits (Appendix I) and disposed of as Category C items, or

   b) Disposed of as Category D items in the HNPF Isolation Structure in accord with the criteria of Section VI of Activity Specification No. 7 if in-place decontamination is economically unfeasible.

2) Pipe tunnel surfaces will be decontaminated to the unrestricted use disposal limits, see Section V. The pipe tunnel will be disposed of as a Category C item.

3) The tunnel penetration to the Reactor Building (HNPF Isolation Structure) will be sealed and weatherproofed in accord with Activity Specification No. 10, Securing of the Isolation Structure.

4) Above grade Radioactive Waste Facility hardware and components will be:

   a) Decontaminated to the unrestricted use disposal limits, Section V, and disposed of as Category B items, or

   b) Disposed of as Category D items in the HNPF Isolation Structure in accord with the criteria of Section VI of Activity Specification No. 7 if decontamination is economically unfeasible.

28. Install a 13-3/8-in. diameter SS capping plate (fitted with 1/4-in. SS bar stock valve) on line stub No. 243 per procedural steps No. 22 through No. 27.

29. Remove the temporary seal from line stub No. 237 and remove the residual sodium from the interior of the pipe to a depth of ~10-in. The inflatable seal shall be retained in position.

30. Prepare the line stub for welding (chamfer end of pipe).

31. Position a 14-in. diameter schedule-10 pipe cap with 4-in. nipple on line stub No. 237 and weld it to the pipe; an inert gas backup is required.

32. Deflate and remove the inflatable seal through the 4-in. pipe nipple.

33. Seal the 4-in. diameter pipe nipple with a 4-in. threaded pipe cap until interconnecting line from inlet header to outlet header is assembled and ready for installation.

34. Connect a regulator equipped nitrogen gas bottle with a flow indicator to each bar stock valve on heat exchanger line stubs No. 242 and No. 243. Connect a manometer to each module between the nitrogen supply and the bar stock valve.

NOTE

Maintain an inert atmosphere by using a bottled supply.

35. Penetrate sodium line No. 240 (14-in. diameter – Module 1A inlet) with a 4-in. diameter hole saw near where the line connects to the elbow of the inlet nozzle. The cut shall be made on the horizontal section of piping.

36. Remove the hole saw blank, and immediately install inflatable seals on each side of the hole saw penetration. Inflate the seals sufficiently to seal the pipe. Tape seal the hole saw opening.

37. Penetrate sodium line No. 241 (14-in. diameter Module 1B inlet) with a 4-in. diameter hole saw where the line connects to the elbow of the inlet nozzle.

38. Remove the hole saw blank and immediately install inflatable seals on each side of the hole saw penetration. Inflate the seals sufficiently to seal the pipe. Tape seal the hole saw opening.

39. Penetrate sodium line No. 234 (14-in. diameter primary sodium pump discharge line) with a 4-in. diameter hole saw approximately 1 ft upstream of the pipe tee (junction of sodium lines No. 234, 240 and 241).

40. Remove the hole saw blank and immediately install inflatable seals on each side of the hole saw penetration. Inflate the seals sufficiently to seal the pipes. Tape seal the hole saw opening.

CAUTION

Insure that the piping to be removed is adequately supported.

41. Cut sodium line No. 240 between the seals which were inserted in step 36. Tape seal the cut.

42. Cut sodium line No. 241 between the seals which were inserted in step 38. Tape seal the cut.

43. Cut sodium line No. 234 between seals which were inserted in step 40 and hoist the pipe assembly upward sufficiently to permit installation on temporary seals (plastic caps) on all exposed pipe openings.

44. Complete the removal of disassembled piping from the vault and store it as per CPPD's instructions.

45. Remove the temporary seal from line stub No. 240 on heat exchanger Module 1A and remove the residual sodium from the interior of the pipe to a depth of ~10 in. The inflated seal shall be retained in position.

46. Determine that the 13-3/8-in. by 1/4-in. thick capping plate will fit inside the line stub.

CAUTION

Assure that the inert gas pressure on the heat exchanger Module 1A is 1/4 in. w.g.

47. Deflate the inflatable seal sufficiently to permit sliding it out of line stub No. 240.

48. Immediately position the capping plate inside the line stub 1/2 in. and tack weld it into position.

49. Tape seal the annulus between the capping plate and the interior of the pipe.

50. Apply the capping plate weldment root pass, removing the tape as necessary to permit welding.

51. Complete the weldment and perform a dye penetrant inspection of the final filler pass.

52. Install a capping plate on line No. 241 (heat exchanger Module 1B) as per procedural steps No. 45 through No. 51.

53. Close the bar stock valves on Modules 1A and 1B; replace manometer with a 0 to 10 psig pressure gage.

54. Pressurize heat exchangers Module 1A and Module 1B to 5 psig with nitrogen gas. Close the bar stock valves, and remove the nitrogen supply. Seal the bar stock valves with pipe plugs.

55. Remove the temporary seal from the sodium line stub No. 234. Remove the residual sodium from the interior of the pipe to a depth of ~10 in. The inflatable seal shall be retained in position.

56. Prepare the line stub No. 234 for welding (chamfer end of pipe).

57. Position a 14-in. diameter schedule-10 SS pipe cap, fitted with a 4-in. diameter SS pipe nipple, on the line stub. This pipe nipple must be threaded for temporary sealing with a pipe cap.

58. Weld the pipe cap to the line stub. Inert gas backup is required. Inspect the pipe cap weldment.

59. Deflate and remove the inflatable seal through the 4-in. pipe nipple.

60. Seal the 4-in. pipe nipple on line No. 234 pipe cap with a 4-in. to 2-in. reducer and a 2-in. valve. Seal the valve with a pipe cap.

61. Remove insulation as required from the heater ovens and from the bellows region of heat exchanger Modules 1A and 1B.

62. Install the special bellow clamps on the modules to prevent the bellows from flexing when the heat exchangers are being handled.

63. Attach lifting slings to the heat exchanger mounting frames.

NOTE

Estimated weight of modules – 28,000 lb.

64. Detach the frame mounting bolts and hoist the modules from the vault.

65. Transport the heat exchanger modules to the reactor high bay floor area for packaging and shipment off-site.

66. Heat exchangers Modules 2A, 2B, 3A and 3B shall be isolated and removed as per Section IV of this procedure. The line numbers will be changed on the field copy of Procedure RDP 5-3 to correspond with heat exchangers No. 2 and No. 3.

67. A daily progress report shall be made and delivered to CPPD. This report shall state manpower, unusual problems encountered, and any unusual conditions that may have developed in the course of the work.

PROCEDURE 9.

ISOLATION AND REMOVAL OF PRIMARY PUMP BALANCING LEGS

I. PURPOSE

This procedure establishes a step-by-step sequence for isolating the three primary sodium pump balancing legs for removal from the facility.

II. DESCRIPTION

The activities to be accomplished by this procedure are as follows. Confirm that the electrical service to the balancing legs and their related piping has been deenergized and tagged out; disconnect and remove electrical power supplies to the balancing legs, including the sodium leak detectors; disconnect and remove the temperature indicating instrumentation from the balancing legs, including their associated piping; remove thermal insulation, heater reflector oven, and line heaters as required to provide access for cutting the lines; and cut and cap the sodium and vent lines as specified by Section IV of this procedure. Remove the balancing legs from the pipe vault for disposal.

III. CRITERIA

A. REQUIREMENTS AND PREREQUISITES

1. This activity shall be coordinated with other plant activities during the HNPF retirement.
2. All operation of plant equipment and deenergizing of electrical circuitry required in the performance of this procedure shall be performed only by CPPD plant personnel.
3. An inert gas atmosphere shall be maintained on the balancing legs and the heat transfer loops during the performance of Section IV of this procedure.
4. Notification shall be made to the shift supervisor prior to initiating this work and also immediately prior to making any penetrations into piping or components which affect, or require special operation of, any associated plant system or components.
5. Personnel assigned to specific noninterruptable tasks, such as pipe cutting, closure welding, removal, and handling, shall not be given additional assignments until their particular phase of that task is completed.
6. Confirm that the following primary sodium pump case vent valves are in the closed position.

Valve No.	Description
V-107	Pump Case No. 1 (P101) Vent
V-207	Pump Case No. 2 (P201) Vent
V-307	Pump Case No. 3 (P301) Vent

7. Dry ice shall be employed as a heat sink when installing capping plates or pipe caps, provided the particular location requires it.

8. Tools and equipment required for the work detailed are listed below.
   a. Hand tools — wrenches, pliers, hammers, lagging saws, tin snips, wire cutters, screw drivers, spatulas, sodium augers, and scrapers.
   b. Hand-operated pipe cutters for 2-in. and 3-in. pipe
   c. Porter-Cable band saw — spare blades
   d. Welding machine — equipped for inert gas welding
   e. Cloth tape — 2 rolls
   f. Two hole saws — 2-1/2-in. diam
   g. Capping plates — 1/4-in. thick, 304 SS, 6-3/4-in. diam
   h. Plastic pipe caps — 2 in.
   i. Plastic pipe caps — 6 in.
   j. Three capping plates — 1/4 in. thick, Type 304 SS, 3-5/8 in. diam
   k. Three capping plates — 1/4 in. thick, Type 304 SS, equipped with 1/4 in. bar stock valve
   l. Six pipe caps — 2 in. diam, 304 SS, butt weld type, Sched-40
   m. Six pipe caps — 6 in. diam, 304 SS, butt weld type, Sched-40
   n. Scaffolding as required
   o. Four expandable rubber stoppers for 2-in. pipe
   p. 1/2-in. portable drill motor
   q. Hoisting and rigging gear for handling pipe spools
   r. Dry ice — procure as needed
   s. Ohmmeters
   t. Pans and supply of dry calcium
   u. Regulator equipped nitrogen bottle with connecting hose
9. The expandable rubber stopper and inflatable seal technique shall be used to maintain an inert atmosphere on the primary heat transfer loops.
10. The inert gas pressure on the primary heat transfer loops shall be carefully controlled during the process of cutting and capping the sodium lines connecting the balancing legs to the pumps.
11. The following prerequisite items shall be completed and signed off prior to initiating performance of the work detailed in Section IV, "Method of Performance and Sequence of Events," of this procedure.
   a. Attach to this procedure a copy of the appropriate P and I drawing marked to show the system isolation to be performed.

NOTE

Drawings shall be marked in accordance with the physical inspection of the system.

b. Personnel assigned to the task shall study and sign the procedure.

c. Potential hazards relating to the work to be performed shall be discussed with the personnel involved before the operation is initiated.

d. The electrical power supply to the balancing legs and the associated piping will be deenergized and tagged out per CPPD tag-out procedure.

e. Confirm that the primary heat transfer loops No. 1, 2, and 3 are at ambient temperature.

B. HEALTH AND SAFETY

1. Special work permits are required for all penetrations, cutting, cleaning, welding of sodium piping, and for sodium handling operations.

2. Sodium protective clothing shall be worn at all times by all personnel while within the confines of the working areas during pipe cutting and pipe capping operations.

3. Deposit all sodium scrap in metal containers containing calcium carbonate and deliver them to the storage location designated by Health and Safety. Primary R/A sodium piping shall be labeled as such before removal from the working area.

4. Calcium carbonate and sodium fire fighting equipment must be kept standing by at all times during cutting, cleaning, welding of sodium piping, and handling of sodium scrap.

5. The buddy system is required for all sodium handling, pipe cutting, and welding operations.

6. Disposal of sodium contaminated equipment and materials shall be performed only as directed by Health and Safety.

7. A portable oxygen analyzer equipped with an audible alarm shall be used to continuously monitor the work area atmosphere; alarm set at 18% oxygen.

C. STANDARDS

Work performed in connection with this activity shall be carried out under normal HNPF Health and Safety regulations (including sodium safety regulations) in accordance with Part III of the HNPF Operations Manual and specific requirements of this procedure. The root pass of all pipe cap and capping plate welds shall be made by the tungsten inert gas method. The pipe cap welds shall be x-ray inspected. The capping plate welds shall be dye penetrant inspected.

IV. METHOD OF ACCOMPLISHMENT AND SEQUENCE OF EVENTS

1. All preparatory work shall be completed before starting the work of this section of the procedure.

2. Confirm that the primary sodium pump balancing leg electrical supplies have been deenergized and tagged "out of service" per Part III, Section A, Step 10-d of this procedure.

3. Disconnect the electrical supplies to the primary sodium pump balancing legs and the following specified sodium lines at the breakers by removing fuses and/or lifting the heater supply leads; then remove the leads at the heaters. Electrical leads disconnected at the breakers must be taped with electrical tape. Balancing leg No. 1-177, 196, 207, 224, and 318; balancing leg No. 2-186, 201, 208, 225, and 319; balancing leg No. 3-188, 202, 212, 226, and 320.

4. Disconnect the temperature indicating instrumentation from the items specified in Step No. 2.

5. Remove conduits, wire tray, and junction boxes as necessary to provide working access to the items specified in Step No. 2.

6. Remove the thermal insulation from the balancing legs piping as required to permit access for cutting and capping the lines.

7. Remove the line heater reflector oven and line heaters from the items specified in Step No. 2 as necessary to permit cutting and capping the pipes.

8. Cut vent line No. 177 (2-in. diameter vent line) in three places as follows: near the point where it connects to the pump case, valve body No. 107, and at the top of the balancing leg. Tape seal the cuts until temporary seals (plastic caps) can be installed over the exposed pipe openings; six seals required. Remove the loose pipe spool from the area.

9. Remove the temporary seal from line stub No. 177 on the pump case and remove any sodium frost that may be present in the pipe stub.

10. Position a 2-in. diameter pipe cap on the line stub and tack weld in position. Tape seal the annulus between the pipe cap and the line stub.

11. Complete the pipe cap weldment, removing the tape as necessary to permit welding.

12. Remove the temporary seal from the line stub No. 177 on the top of the balancing leg. Remove any sodium frost that may be present in the line stub.

13. Position a capping plate (equipped with 1/4-in. bar stock valve) in the line stub and tack weld in position.

14. Complete the capping plate weld, and dye penetrant inspect the final pass.

15. Connect the regulator equipped nitrogen bottle to the 1/4-in. bar stock valve on the balancing leg (line stub No. 177) and purge inert gas into the balancing leg as necessary to maintain an inert atmosphere during the remainder of the isolation operation.

16. Penetrate sodium line No. 196 (6-in. diameter sodium pump overflow) with a 2-1/2-in. diameter hole saw near the point where it connects to the pump case and also near the balancing leg nozzle. Two penetrations are required.

17. Install inflatable seals (bladders) in line No. 196, position one in the pump case nozzle and the other in the balancing leg nozzle. Inflate the seals sufficiently to seal the pipes. Tape seal the hole saw penetrations.

18. Cut sodium line No. 196 near the point where it connects to the balancing leg nozzle. A cut shall also be made between the hole saw penetration and the inflatable seal; tape seal the cut.

19. Cut sodium line No. 196 near the point where it connects to the pump case nozzle. The cut shall be made between the hole saw penetration and the inflatable seal. Install temporary seals (plastic caps) over the exposed pipe openings; four seals required.

20. Remove the loose pipe spool from the area and store it in a specified storage area or lay it aside within the vault.

21. Remove the temporary seal from line stub No. 196 on the pump case nozzle and remove the residual sodium from the interior of pipe to a depth of approximately 6-in. The inflatable seal shall be retained in position.

22. Prepare the end of line stub No. 196 on the pump case for welding (chamfer end of pipe).

23. Check the fit of the 6-in. diameter pipe cap to pump case line stub No. 196.

24. Deflate the inflatable seal sufficiently to permit sliding it out of the pipe opening. Position the pipe cap as quickly as possible and secure it in position with a "C" clamp. Tape seal the annulus between the pipe cap and the line stub to prevent excessive inert gas leakage.

25. Tack weld the pipe cap in position, removing the tape only as necessary to permit tack welding.

26. Remove the "C" clamp and make the weldment root pass, removing the tape as required.

27. Complete the pipe cap weld.

28. Remove the temporary seal from line stub No. 196 on the balancing leg and remove the residual sodium from the interior of the pipe to a depth of approximately 6 in. The inflatable shield shall be retained in position.

29. Determine that the capping plate will fit the pipe stub.

30. Deflate the seal and position the capping plate. Tack weld the capping plate in position. Tape seal the annulus between the capping plate and the interior of the pipe to prevent excessive inert gas leakage.

31. Makethe capping plate weldment root pass, removing the tape as required to permit welding.

32. Cut, remove, and cap sodium line No. 207 (6 in. diameter balancing line pump suction line) as per Steps No. 8 through 24. The line shall be cut near the point where it connects to the balancing leg nozzle and at the connection on sodium line No. 231 (16 in. diameter pump suction line).

33. Cut sodium line No. 318 (2-in. diameter sodium service drain) near the point where it connects to sodium line No. 234 (14-in. diameter primary sodium pump No. 1 discharge); tape seal the cut. This sodium line is not part of the balancing leg.

34. Cut sodium line No. 224 (2-in. diameter balancing leg drain) near the point where it connects to the balancing leg.

35. Move the loose pipe assembly, which includes sodium valves No. 108 and 109, sufficiently to permit installing temporary seals (plastic caps) over the exposed pipe openings; four seals required.

36. Lower the loose pipe assembly to the floor and cut it as required to facilitate removing it from the vault.

37. Remove the temporary seal from line seal No. 318 and install an expandable stopper. Remove the residual sodium from the interior of line stub No. 318 to a depth of approximately 4 in.

38. Prepare line stub No. 318 for welding (chamfer end of pipe).

39. Remove the expandable stopper from line No. 318 and position a 2-in. diameter pipe cap on the line stub. Tack weld it in position. Tape seal the annulus between the cap and the line stub to prevent excessive inert gas leakage.

40. Complete the weld, removing the tape as necessary to permit welding.

41. The following sodium and vent lines connecting balancing legs No. 2 and 3 to their respective sodium pumps and heat transfer loops shall be isolated as per procedural Steps No. 8 through 39.

Balancing Leg No. 2 — Loop No. 2

Line No.	Line Diameter (in.)	Line Description
186	2	Pump case vent
201	6	Pump case overflow
208	6	Balancing leg suction
319	2	Loop No. 2 sodium drain line

Balancing Leg No. 3 — Loop No. 3

Line No.	Line Diameter (in.)	Line Description
188	2	Pump case vent
202	6	Pump case overflow
212	6	Balancing leg suction
320	2	Loop No. 3 sodium drain line

42. Upon completion of the isolation of the balancing legs, each balancing leg shall be charged with inert gas to a pressure of ~5 psig. The inert gas shall be purged into the balancing legs through the 1/4-in. bar stock valve located on the top of the assembly.
43. Upon completion of the work detailed by Section IV of this procedure, the work areas shall be cleaned and the balancing legs shall be retained in their present position until the primary pump cases have been removed. The balancing legs shall be hoisted from the pipe vaults through the openings provided by removing the pump cases.
44. Attach P and I drawing No. D-070501 that shall be red lined as the piping and balancing legs are removed.
45. A daily progress report shall be kept and delivered to CPPD.
46. By signature, acknowledgment shall be made that this procedure, RDP-5-4, Isolation and Removal of Primary Pump Balancing Legs has been studied and understood.
47. Procedure RDP-5-4, Isolation and Removal of Primary Pump Balancing Legs. Work completed, specify date, in accordance with this procedure and the attached support drawings, weldment inspection reports, and other data shall be listed and signed.

PROCEDURE 10

ISOLATION AND DISASSEMBLY OF THE SECONDARY SODIUM SERVICE SYSTEM

I. PURPOSE

This procedure establishes a step-by-step sequence for isolating the Secondary Sodium Service System from the Secondary Heat Transfer Systems and removing the Secondary Sodium Service System piping and components from the facility.

II. TASK DESCRIPTION

The activities to be accomplished by this procedure are as follows.

Confirm that the electrical service to the Secondary Sodium Service System is deenergized and tagged out as was partially accomplished during the performance of Procedure RDP 2-1; disconnect and remove the temperature indicating instrumentation from the Secondary Sodium Service System piping and components; disconnect and remove the air cooling ducts from the service system freeze traps, plugging meter assembly, and cold trap furnace; remove thermal insulation from the service system piping and components to provide access for cutting the piping; cut and cap the sodium lines connecting the service system to the Secondary Heat Transfer Systems; and remove the service system piping and components from the facility for disposal.

III. CRITERIA

A. REQUIREMENTS AND PREREQUISITES

1. This activity will be coordinated with other plant activities during the HNPF retirement program.
2. All operation of plant equipment and deenergizing of electric circuitry required in the performance of this procedure shall be performed only by CPPD plant personnel.
3. Notification shall be made to the shift supervisor prior to initiating this work and also immediately prior to making any penetrations into piping or components which may affect, or require special operation of, any associated plant system or components.
4. Personnel assigned to the specific noninterruptable tasks (such as pipe cutting, closure welding, removal, and handling) shall not be given additional assignments until their particular phase of that task is completed.
5. Tools and equipment required for the work detailed.
   a. Hand tools - wrenches, pliers, hammers, lagging saws, tin snips, wire cutters, screw drivers, spatulas, sodium augers, and scrapers.
   b. Hand operated pipe cutter for 3-in. pipe
   c. Porter-Cable band saw - supply of spare blades
   d. Welding machine - equipped for inert gas welding
   e. Cloth tape - 4 rolls

f. Plastic pipe caps - 3-in. diameter

g. Plastic pipe caps - 2-in. diameter

h. Capping plates 1/4-in. thick SS - 12 each - 2-in. diameter

i. Capping plates 1/4-in. thick SS - 12 each - 3-in. diameter

j. Scaffolding - as required

k. Stepladders - as required

l. Hand operated pipe cutter for 2-in. diameter pipe

m. Lifting and rigging equipment as required for handling pipe spools and components

n. Ohmmeters - 2 each

o. Dry ice (procure as needed)

p. Expandable rubber stoppers for 2 and 3-in. pipe - 2 each

q. Oil manometer - specific gravity of 1

6. The expandable rubber stopper technique shall be used to maintain inert atmosphere on the Secondary Heat Transfer Loops.

7. Plastic seal caps shall be used as temporary seals on exposed sodium piping. Metal seal plates will be welded on the reusable components only.

8. The inert gas pressure on the Secondary Heat Transfer Loops shall be carefully controlled during the process of cutting and capping the sodium lines connecting the service system to the Secondary Heat Transfer Loops. Air shall not be permitted to enter the Secondary Heat Transfer Loops.

9. Dry ice shall be employed as a heat sink when installing capping plates, providing the particular location requires it.

10. The following prerequisite items shall be completed and signed off prior to initiating performance of the work detailed in Section IV, "Method of Performance and Sequence of Events," of this procedure.

    a. Attach to this procedure a copy of the appropriate P and I drawing marked to show the system isolation to be performed.

       NOTE: Drawing shall be marked in accordance with physical inspection of the system.

    b. Personnel assigned to the task shall study and sign the procedure.

    c. Potential hazards relating to the work to be performed shall be discussed with the personnel involved before the operation is initiated.

    d. The electrical power supply to the Secondary Sodium Service System Preheat was partially deenergized and tagged out during the performance of Procedure RDP 2-1 (Isolation of the Secondary Fill Tanks). The remainder of the Sodium Service System electrical power supply shall be deenergized and tagged out per CPPD tag-out procedure.

e. Confirm that the Secondary Heat Transfer Loops No. 1, No. 2, and No. 3 are at ambient temperature.

B. HEALTH AND SAFETY

1. Special Work Permits are required for all penetrations, cutting, cleaning, welding of sodium piping, and for sodium handling operation.
2. Sodium protective clothing shall be worn at all times by all personnel while within the confines of the working areas during pipe cutting and pipe capping operations.
3. Deposit all sodium scrap in metal containers containing calcium carbonate and deliver them to the storage location designated by Health and Safety.
4. Calcium carbonate and sodium fire fighting equipment must be kept standing by at all times during cutting, cleaning, welding of sodium piping, and handling of sodium scrap.
5. The buddy system is required for all sodium handling, pipe cutting, and welding operations.
6. Disposal of sodium contaminated equipment and materials shall be performed only as directed by Health and Safety.

C. STANDARDS

Work performed in connection with this activity shall be carried out under normal HNPF Health and Safety regulations (including sodium safety regulations) in accordance with Part III of the HNPF Operations Manual, and specific requirements of this procedure. Inspection of capping plate weldments shall consist of dye penetrant testing of the final pass.

IV. METHOD OF ACCOMPLISHMENT AND SEQUENCE OF EVENTS

1. All preparatory work shall be completed prior to starting the work of this section of the procedure.
2. Confirm that all secondary sodium preheat has been deenergized and tagged out per Part III, Section A, Step 10d of this procedure.
3. Disconnect the piping preheat electrical supplies at the breakers by removing fuses and/or lifting heater supply leads and them remove the leads at the heaters. Electrical leads disconnected at the breaker must be taped with electrical tape.
4. Remove conduits and junction boxes associated with the Secondary Service System piping and component heaters.
5. Disconnect and remove the temperature indicating instrumentation from the Secondary Sodium Service System piping and components. These items shall include the thermocouple lead wires, conduits, and junction boxes as is necessary to permit subsequent removal of the service system piping and components.
6. Remove insulation from the Secondary Sodium Service System piping and components as required to permit access for cutting the piping into suitable lengths to facilitate removing the piping and components from the area.
7. Remove the service system piping and component heaters as required to permit cutting the piping at the desired locations.

8. Confirm that the following helium system valves are in the closed position.

Valve No.	Description
V-8218	He Supply Block Valve to Freeze Trap No. 6 and No. 7
V-8219	He Supply By Pass Valve to Freeze Trap No. 6 and No. 7
V-8228	He Supply Block Valve to Freeze Trap No. 10
V-8229	He Supply by Pass Valve to Freeze Trap No. 10

8a. Connect an oil manometer (with specific gravity of 1) to the No. 1 Secondary Heat Transfer Loop.

NOTE: The manometer shall be used on Secondary Heat Transfer Loops No. 2 and No. 3 when they are isolated.

9. Reduce the inert gas pressure on the Secondary Heat Transfer Loops No. 1, No. 2, and No. 3 to ~2-in. of oil pressure by operating the following helium vent valves.

Valve No.	Description
V-8179 (Loop 1)	He Vent from Vapor Trap No. 11
V-8185 (Loop 2)	He Vent from Vapor Trap No. 12
V-8191 (Loop 3)	He Vent from No. 3 Loop

NOTE: A slight positive inert gas pressure (1/4-in. oil pressure or less) shall be maintained on the Secondary Heat Transfer Loops when cutting and capping the Service System Sodium lines that connect to the Secondary Heat Transfer Loops.

10. Cut a section from sodium line No. 191 (2-in. diameter Secondary Sodium throttle valve drain) where it connects to throttle valve V-102.

11. Seal the open line stubs with plastic caps. This will include the removed pipe spool and the open line stubs remaining on the system.

12. Remove the pipe spool from the work area to a specified storage area.

13. Remove the plastic cap from line stub No. 191 on throttle valve No. 102. Position an expandable rubber stopper inside the line stub to a depth of ~6 in.

14. Prepare line stub No. 191 for welding on a capping plate. Remove any residual sodium from inside the line stub up to the expandable stopper.

15. Remove the expandable stopper from line stub No. 191, immediately position the capping plate, and tack weld in position.

16. Complete the capping plate weld, and dye penetrant inspect the final pass.

17. Cut a section from sodium line No. 192 (3-in. diameter - fill and drain from Secondary Loop No. 1) where it connects to sodium line No. 273 (14-in. diameter - steam generator outlet line).

18. Seal the open ends of the pipes with plastic caps, including the removed pipe spool.

19. Remove the pipe spool from the work area to a specified storage area.

20. Remove the plastic cap from line stub No. 192 on sodium line No. 273. Position an expandable rubber stopper inside the line stub to a depth of about 6 in.

21. Prepare line stub No. 192 for installing a 3-in. diameter capping plate. Remove any residual sodium from the inside walls of the line stub up to the expandable rubber stopper.

22. Remove the expandable stopper from the line stub No. 192 and immediately position the capping plate. Tack weld the capping plate in position.

23. Complete the capping plate weld and dye penetrant inspect the final pass.

24. Cut and remove a section from sodium line No. 209 (3-in. diameter - fill, drain and cold trap return line from Secondary Loop No. 1) where it connects to sodium line No. 267 (14-in. diameter secondary pump suction line).

25. Seal the open end of the pipes with plastic caps, including the removed pipe spool.

26. Remove the pipe spool from the work area to a specified storage area.

27. Remove the plastic cap from line stub No. 209 where it connects to line No. 267 and immediately position an expandable rubber stopper inside the line stub to a depth of about 6 in.

28. Prepare line stub No. 209 for installing a 3-in. diameter capping plate by removing the residual sodium from inside the line stub up to the expandable rubber stopper.

29. Remove the expandable rubber stopper from line stub No. 209 and immediately position a 3-in. diameter capping plate. Tack weld the capping plate in position.

30. Complete the capping plate weld and dye penetrant inspect the final pass of the weld.

31. Cut and remove a section from sodium line No. 194 (2-in. diameter - drain from Loop No. 1).

32. Seal the open ends of sodium line No. 194 with plastic caps including the removed pipe spool.

33. Remove the pipe spool from the work area to a specified storage area.

34. Remove the plastic cap from line stub No. 194 where it connects to sodium line No. 264 and immediately position an expandable rubber stopper inside the pipe to a depth of about 6 in.

35. Prepare line stub No. 194 for installing a capping plate by removing any residual sodium from the inside walls of the pipe up to the expandable stopper.

36. Remove the expandable stopper and immediately position the capping plate. Tack weld the capping plate in position.

37. Complete the capping plate weld and dye penetrant inspect the final pass of the weld.

38. The following Secondary Sodium Service System lines connecting to Secondary Heat Transfer Loops No. 2 and No. 3 shall be cut and capped by using the same techniques specified by procedural steps No. 9 through No. 37.

Secondary Heat Transfer Loop No. 2

Line No.	Line Diameter (in.)	Line Description
197	2	Drain from Sec. Throttle Valve No. 2
198	3	Fill and Drain from Sec. Loop No. 2
200	2	Drain from Sec. Loop No. 2
210	3	Fill and Drain from Sec. Loop No. 2
		Secondary Heat Transfer Loop No. 3
203	2	Fill and Drain from Sec. Loop No. 3
204	3	Drain from Sec. Throttle Valve No. 3
206	2	Drain from Loop No. 3
211	3	Fill and Drain from Sec. Loop No. 3

39. Confirm that the electrical power to the Secondary Service Pump has been deenergized and tagged out.

40. Remove the protective cage from the Secondary Service Pump (B-403).

41. Remove the electrical coils from the pump throat and remove them from the work area.

42. Cut sodium line No. 181 where it connects to the secondary sodium service pump throat. Seal pipe spool from line No. 181 and the pump throat line stub with plastic caps. Remove pipe spool from line No. 181 from the work area.

43. Cut sodium line No. 182 where it connects to the Secondary Sodium Service pump throat.

44. Remove the pump throat from its mounting and install 1/4-in. SS capping plates over the pump throat line stubs. Remove the pump throat from work area.

45. Remove air cooling equipment from the plugging meter assembly.

46. Cut sodium line No. 218 at the upstream side of sodium valve No. 445 and at the plugging meter economizer. Seal the open pipes with plastic caps. Remove the pipe spool from the work area.

47. Cut sodium line No. 221 at the upstream side of sodium valve No. 446 and where it connects to the plugging meter economizer. Seal the open pipes with plastic caps and remove the pipe spool from the work area.

48. Remove the plugging meter assembly from its mounting and lower it to the floor.

49. Install 1/4-in. thick SS capping plates over economizer line stubs No. 218 and No. 221. Move the plugging meter to an approved storage area.

> NOTE: The remainder of the Secondary Sodium Service System consists of sodium piping, sodium valves, two sodium filters, and two freeze traps. The following techniques will be followed for removing the remainder of the service system.

50. Cut the piping into suitable lengths to facilitate removing it from the area.

51. Each pipe spool will be sealed with plastic caps as it is removed.

52. The sodium inlet and outlet lines connecting to the sodium filters will be cut so the valves remain on the filter vessels, thereby preventing exposing the sodium filter cartridges to air. Install plastic caps on the pipe stubs of the valve.

53. The attached P and I drawing No. D-070502 will be red lined as the piping and components are removed.

54. A daily progress record (log book) of the task will be kept and delivered to CPPD at the completion of the removal of the Secondary Sodium Service System.

PROCEDURE 11.

ISOLATION OF THE INTERMEDIATE HEAT EXCHANGERS FROM THEIR RELATED SECONDARY SYSTEMS

I. PURPOSE

This procedure establishes a step-by-step sequence for isolating the three Intermediate Heat Exchangers (six modules) from their related secondary systems as a part of the retirement of the Secondary Heat Transfer Systems and to permit the removal of the heat exchangers for off-site shipment.

II. TASK DESCRIPTION

The activities to be accomplished by this procedure are as follows. Disconnect and remove the electrical service to the heat exchangers and the associated secondary piping; disconnect and remove the temperature indicating instrumentation from the heat exchangers and the associated piping; disconnect and remove the sodium leak detector circuitry from heat exchangers; cut and cap the secondary sodium lines connecting to the heat exchangers; and provide a means of pressuring the secondary or shell side of the heat exchanger modules with inert gas.

III. CRITERIA

A. REQUIREMENTS AND PREREQUISITES

1. This activity will be coordinated with other plant activities during the HNPF retirement program.
2. All operation of plant equipment and deenergizing of electrical circuitry required in the performance of this procedure shall be performed only by CPPD plant personnel.
3. Notification shall be made to the CPPD shift supervisor prior to initiating this work and also immediately prior to making any penetrations into piping or components which may affect, or require special operation of, any associated plant system or components.
4. Personnel assigned to specific noninterruptable tasks (such as pipe cutting, closure welding, and sodium removal and handling) shall not be given additional assignments until their particular phase of that task is completed.
5. Tools and equipment required for the work detailed are listed below.
   a. Hand tools - wrenches, pliers, hammers, lagging saws, tin snips, wire cutters, screw drivers, spatulas, sodium augers, and scrapers
   b. Skill saw - equipped with carborundum blade
   c. Hand-operated pipe cutter - 4-in. throat
   d. Fein pipe saw or equivalent
   e. Welding machine - equipped for inert gas welding
   f. Cloth tape - 4 rolls

g. Electric drill motor - 1/2-in.

h. Hole saws - 4 in. diameter - 8 each

i. Porter-Cable band saw - 3-in. throat

j. Capping plates, 6 each - 14-in. diameter, 1/4-in. thick - type 304 SS - equipped with 1/4-in. bar stock valves

k. Inflatable bladders - 4 each - sized for sealing 14-in. pipe

l. Capping plates - 6 each - 14-in. diameter - 1/4-in. thick - type 304 SS
Capping plates - 12 each - 14-in. diameter - 1/4-in. thick - mild steel

m. Capping plates - 6 each - 2-in. diameter - 1/4-in. thick - type 304 SS
Capping plates - 6 each - 2-in. diameter - 1/4-in. thick - mild steel

n. Plastic pipe caps - sized for pipes - 14-in. and 2-in. diameter

o. Toledo pipe cutter - electrically driven or equivalent

p. Weldment dye penetrant inspection kit

q. Scaffolding - as required

r. Step ladders

s. Nitrogen bottles - equipped with regulators, fitting and hose

t. Pressure gauge (range 0 to 5 psig)

u. Relief valve - gas service - set at 5 psig

v. Chain falls - 2 each - 1-ton capacity

w. Lifting slings - sized for task

x. Portable O_2 analyzer - equipped with audible alarm

y. Manometer - oil with specific gravity of 1.

6. Secondary sodium loops shall have been drained and isolated from the service system per HNPF SOP 3101 (Rev. 1) before removal of instrumentation and electric power to heaters. No attempt shall be made to operate sodium valves at temperatures below 300°F.

7. The inflatable bladder technique shall be used to maintain an inert atmosphere on the Intermediate Heat Exchanger Modules.

8. Plastic seal caps shall be used as temporary seals on exposed sodium piping. Metal seal plates must be welded in place before leaving work area unattended.

9. The following prerequisite items shall be completed and signed off prior to initiating performance of the work detailed in Section IV, "Method of Performance and Sequence of Events," of this procedure.

   a. Attach to this procedure a copy of the appropriate P and I drawing marked to show the equipment and system isolation to be performed.

      NOTE: Drawings shall be marked in accordance with physical inspection of the system.

b. Personnel assigned to the task shall study and sign the procedures.

c. Potential hazards relating to the work to be performed shall be discussed with the personnel involved before the operation is initiated; (example, inert gas)

d. Electrical power to the Intermediate Heat Exchangers shall be deenergized and tagged "out of service" in accordance with normal HNPF tagging procedures.

e. Electrical power to the Intermediate Heat Exchanger Sodium Leak Detectors shall be deenergized and tagged "out of service" in accordance with normal HNPF tagging procedures.

f. Confirm that the entire Main Secondary Heat Transfer System is at ambient temperature prior to initiating the cutting of any of the sodium piping.

B. HEALTH AND SAFETY

1. Special Work Permits are required for all penetrations, cutting, handling, cleaning, and welding of sodium piping, and for sodium handling operations.
2. Sodium protective clothing shall be worn at all times by all personnel while within the working areas.
3. The buddy system is required for all sodium handling, pipe cutting, and welding operations.
4. Deposit all sodium scrap in metal containers containing calcium carbonate and deliver them to the storage location designated by Health and Safety.

   NOTE: No sodium scrap or sodium contaminated component may be left open or unattended except as specifically authorized by Health and Safety.

5. Calcium carbonate and sodium fire equipment must be kept standing by at all times during cutting, cleaning, and welding of sodium piping and handling of sodium scrap.
6. Disposal of sodium contaminated equipment and materials shall be performed only as directed by Health and Safety.
7. A portable oxygen analyzer (equipped with audible alarm) shall be in service monitoring the work area atmosphere prior to making any penetrations into the sodium systems; alarm shall be set at 19% oxygen.

C. STANDARDS

Work performed in connection with the activity shall be carried out under normal HNPF Health and Safety regulations (including sodium safety regulations) in accordance with Part III of the HNPF Operations Manual, and specific requirements of this procedure. Inspection of the capping plate weldments shall include dye penetrant testing of each root pass and final pass.

IV. METHOD OF ACCOMPLISHMENT AND SEQUENCE OF EVENTS

1. All preparatory work shall be completed before starting the work of this section of the procedure.

2. Confirm that the preheat electrical supply has been deenergized and tagged out per Part III, Section A, Step 9d. Disconnect the Intermediate Heat Exchanger preheat electrical supplies at the breakers by removing fuses and/or by lifting the electrical leads feeding the heaters. All electrical leads must be taped with electrical tape.

3. Entry into the Intermediate Heat Exchanger Cells shall be made per SOP's 5605, 5606, and 5607

4. Remove electrical leads at the heaters and associated conduits and junction boxes as necessary to facilitate cutting and capping of sodium lines. The extent of the pipe heater circuitry removal will be determined by the physical inspection of the system.

5. Confirm that the electrical supply to the leak detectors has been deenergized and tagged out per Part III, Section A, Step 9e. Disconnect and remove the sodium leak detector power leads, conduits, and junction boxes from the heat exchangers.

6. Disconnect and remove the temperature indicating instrumentation from the Intermediate Heat Exchangers. These items shall include the thermocouple lead wires, conduits, and junction boxes as is necessary to permit subsequent removal of the heat exchangers.

7. Remove thermal insulation from secondary sodium lines No's. 252 (outlet), 254 (inlet), and 296 (vent), where they connect to Heat Exchanger No. 1 - Module 1A. See P and I drawing for line numbers pertaining to Modules 1B, 2A, 2B, 3A, and 3B. A small amount of insulation may have to be removed from the heat exchanger around the line nozzle connections to provide proper access.

8. Remove a section of line heater reflector oven from the uninsulated areas on these lines.

9. Disconnect the line heater attaching devices to permit removing or bending the line heaters to provide access for cutting the pipes.

CAUTION

The personnel performing the sodium line cutting operation shall suit in the proper protective clothing as specified by Health and Safety and detailed in the Special Work Permit issued for this work. Notify CPPD shift supervisor prior to making each penetration into sodium lines.

NOTE: An inert gas atmosphere shall be maintained on the heat exchangers during the isolation operation.

10. Reduce to 1/2 to 1-in. w.g. (use oil with a specific gravity of 1) the inert gas pressure to Secondary System Main Heat Transfer Loop which is pertinent for the Intermediate Heat Exchanger being isolated by opening the following valves.

Loop No. 1	V-8179	Secondary Sodium Expansion Tank No. 1 Vent
Loop No. 2	V-8185	Secondary Sodium Expansion Tank No. 2 Vent
Loop No. 3	V-8191	Secondary Sodium Expansion Tank No. 3 Vent

11. The following helium system valves shall be operated to maintain 1/2 to 1-in. w.g. pressure of nitrogen gas in the sodium line piping and in the secondary sodium side of the Intermediate Heat Exchanger during the cutting and capping operation.

Loop No. 1	V-8175	Secondary Sodium Expansion Tank No. 1 Helium Supply Block Valve
	V-8177	Secondary Sodium Expansion Tank No. 1 Helium Supply Throttle Valve
Loop No. 2	V-8181	Secondary Sodium Expansion Tank No. 2 Helium Supply Block Valve
	V-8183	Secondary Sodium Expansion Tank No. 2 Helium Supply Throttle Valve
Loop No. 3	V-8187	Secondary Sodium Expansion Tank No. 3 Helium Supply Block Valve
	V-8189	Secondary Sodium Expansion Tank No. 3 Helium Supply Throttle Valve

12. Make a 4-in. hole saw penetration into secondary sodium line No. 254 (Intermediate Heat Exchange Inlet) approximately 1 ft upstream from heat exchanger - Module 1A nozzle.

13. Remove the hole saw blank and immediately install inflatable bladders inside the line. Insert one into the heat exchanger nozzle and one upstream from the hole saw penetration. Inflate the bladders sufficiently to seal the lines.

14. Tape seal the hole saw penetration.

15. Cut sodium line No. 254 near the heat exchanger. The cut shall be made between the bladders and on the heat exchanger side of the hole saw penetration; tape seal the cut.

16. Cut sodium line No. 254 upstream from the hole saw penetration but not beyond the inflated bladder so that about 18 in. of line No. 254 can be removed. Both inflated bladders must remain in position during the cutting operation to seal the exposed sodium lines. The 18-in. section of line No. 254 shall be supported to prevent it from falling.

17. Remove any residual sodium from the line stub No. 254 on the Heat Exchanger Module.

NOTE: The inflated bladders shall remain in position during the sodium removal. Clean the line stubs to a depth of 8 in. or more.

18. Install a plastic cap in the heat transfer line stub as a temporary seal.

19. Determine that the capping plate will fit inside the Intermediate Heat Exchange line stub. It is suggested that a handle be tack welded to the capping plate to permit positioning the capping plate quickly.

NOTE: Use stainless steel capping plates to cap the Intermediate Heat Exchanger.

20. Deflate the bladder sufficiently to permit sliding it out of the line stub.

21. Immediately insert the SS capping plate (equipped with 1/4-in. bar stock valve) into the line stub to a depth of 3/8-in.; tack weld in position.

22. Tape seal the annulus between the capping plate and the inside wall of the line stub to prevent the inert atmosphere from escaping from the heat exchanger.

23. Proceed to make the weldment root pass. Remove the tape as necessary when making the weldment.

24. Dye penetrant inspect the weldment root pass. Determine that the weldment is satisfactory.

25. Proceed to apply the filler passes to the weldment root pass.

26. Dye penetrant inspect the weldment final filler pass.

27. Remove the temporary plastic cap on the heat transfer line and seal weld a mild steel capping plate as per Steps 17 through 26.

28. Proceed to cut and cap sodium line No. 252, Intermediate Heat Exchanger No. 1A Secondary Outlet, as specified by procedural Steps 12 through 27. Use a stainless steel capping plate without a bar stock valve to cap the Intermediate Heat Exchanger line stub.

29. Connect a nitrogen gas supply to the 1/4-in. bar stock valve on the capping plate on line stub No. 254.

    NOTE: The gas line connecting the inert gas supply to the 1/4-in. bar stock valve shall be equipped with a relief valve set at 5 psig, a flow indicator (range 0 to 10 cfm), and a gas pressure regulator.

30. The inert atmosphere inside the heat exchanger shall be maintained via the inert gas supply connected to line stub No. 254.

31. Cut a section out of line No. 296, Intermediate Heat Exchanger No. 1A Secondary Side Vent Line, where it connects to the heat exchanger. Tape seal or install plastic caps to maintain the inert atmosphere inside the heat exchanger.

32. Remove the temporary seal from the Intermediate Heat Exchanger portion of line No. 296 line stub and install an expandable rubber stopper into the line nozzle.

33. Remove any residual sodium that may be inside line stub No. 296; remove rubber stopper.

34. Install a 2-in. SS capping plate on Intermediate Heat Exchanger line stub No. 296, and perform the specified weldment inspections on the root pass and final pass.

35. Remove the temporary seal in the line portion of vent line No. 296 and install a 2-in. mild steel capping plate as per Steps 32 through 34.

36. Pressurize the secondary or shell side of heat exchanger module 1A to 5 psig with inert gas, close and seal the 1/4-in. bar stock valve on line No. 254 capping plate.

37. Proceed with isolation of Intermediate Heat Exchanger Units 1B, 2A, 2B, 3A, and 3B as per Steps 1 through 36.

    NOTE: Refer to appropriate P and I drawing for valve and line numbers.

38. A daily progress record (log book) of the task shall be kept and delivered to CPPD at the completion of the isolation of the Secondary Side of the Intermediate Heat Exchangers.

PROCEDURE 12.

ISOLATION AND REMOVAL OF THE SECONDARY SODIUM PUMP CASES FROM THEIR RELATED SYSTEMS

I. PURPOSE

This procedure establishes a step-by-step sequence for isolating the three secondary sodium pump cases from their related systems.

II. TASK DESCRIPTION

The activities to be accomplished by this procedure are as follows. Disconnect and remove the electrical service to the pump cases and their associated sodium piping; disconnect and remove the temperature indicating instrumentation serving the pump cases and their associated sodium piping, removing thermal insulation from selected locations to permit working access; cut and cap the sodium piping connecting to the pump cases; cut and cap the vent lines serving the pump cases; and hoist the pump cases from their mountings.

III. CRITERIA

A. REQUIREMENTS AND PREREQUISITES

1. This activity shall be coordinated with other plant activities during the HNPF retirement program.
2. All operation of plant equipment and deenergizing of electrical circuitry required in the performance of this procedure shall be performed only by CPPD plant personnel.
3. Notification shall be made to the CPPD shift supervisor prior to making any penetrations into piping or components which may affect, or require special operation of, any associated plant system or components.
4. Personnel assigned to specific noninterruptable tasks such as pipe cutting and closure welding, and sodium removal and handling shall not be given additional assignments until their particular phase of that task is completed.
5. Tools and equipment required for the work detailed in this procedure are as follows.
   a. Hand tools — wrenches, pliers, hammers, lagging saws, tin snips, wire cutters, screw drivers, spatulas, sodium augers, and scrapers
   b. Pipe cutting equipment — Toledo — electrically driven cutter, Fein pipe saw or other suitable pipe cutting devices
   c. Porter-Cable band saw
   d. Plastic pipe caps — sized for pipe
   e. Hand operated pipe cutter (4-in. throat)
   f. Cloth tape — 2 rolls

g. Supply of dry calcium carbonate

h. Lifting gear — properly sized for task

i. Dynamometer — suitable capacity

j. Welding machine — equipped for inert gas welding

k. Portable O_2 analyzer — equipped with audible alarm

l. Dry ice (if required) — procure as needed

m. Sheet plastic

n. Steel bread pans — 4 each

o. Chain falls — 2 each — 1/2-ton capacity

p. Rope lifting slings — 1/2-in. diameter — 2 each

q. Inflatable seal — 14-in. diameter

r. 14-in. diameter by 1/4-in. thick mild steel plate with 1/4-in. bar stock valve

s. Manometer — use oil with specific gravity of 1.

6. The following prerequisite item shall be completed and signed off prior to initiating performance of the work detailed in Section IV, "Method of Performance and Sequence of Events" of the procedure.

   a. Attach to this procedure a copy of the appropriate P and I drawing marked to show the equipment isolation to be performed

   NOTE

   The P and I drawing shall be marked in accordance with the physical inspection of the system.

   b. Personnel assigned to the task shall study and sign the procedure.

   c. Potential hazards relating to the work to be performed shall be discussed with the personnel involved before the operation is initiated.

   d. Confirm that the secondary sodium service system has been isolated from the secondary main heat transfer system per SOP 3101 (Rev. 1), Draining of Secondary Sodium Loops.

   e. Deenergize and tag out preheat electrical power to the Secondary Pump Cases and the adjacent secondary heat transfer system piping in accordance with normal HNPF tagging procedures.

   f. Check to confirm that the entire secondary main heat transfer system is at ambient temperature prior to initiating the cutting of any of the sodium piping.

B. HEALTH AND SAFETY

1. Special Work Permits are required for all penetrations, cutting, handling, cleaning, and welding of sodium piping, and for sodium handling operations.

2. Sodium protective clothing shall be worn at all times by all personnel while within the working areas of sodium-pipe cutting, handling, cleaning, welding and during handling or disposal of sodium scrap.

3. The buddy system is required for all sodium handling, pipe cutting, and welding operations.

4. Deposit all sodium scrap in metal containers containing calcium carbonate and deliver them to the storage location designated by Health and Safety.

NOTE

No sodium scrap or sodium contaminated component may be left open or unattended except as specifically authorized by Health and Safety.

5. Calcium carbonate and sodium fire fighting equipment must be kept standing by at all times during cutting, cleaning, and welding of sodium piping, and during handling of sodium scrap.

6. Disposal of sodium contaminated equipment and materials shall be performed only as directed by Health and Safety.

7. The portable O_2 analyzer shall be in service to monitor the work area atmosphere.

C. STANDARDS

Work performed in connection with this activity shall be carried out under normal HNPF Health and Safety regulations (including sodium safety regulations) in accordance with Part III of the HNPF Operations Manual (Volume I-a), the CPPD Safety Manual, and specific requirements of this procedure.

IV. METHOD OF ACCOMPLISHMENT AND SEQUENCE OF EVENTS

1. All preparatory work shall be completed before starting the work of this section of the procedure.

NOTE

Procedural steps No. 2 through 31 describe the removal of secondary sodium pump case No. 1; however, they shall also be applicable to the removal of secondary sodium pump cases No. 2 and 3. The proper line numbers will be entered on the field copy of the procedure prior to starting the work on pump cases No. 2 and 3.

2. Confirm that the electrical supply to the pump case and adjacent piping has been deenergized and tagged out per part III, section A, step 6e. Disconnect the pump case and adjacent piping preheat electrical supplies at the breakers by removing fuses and/or lifting the heater supply leads; then remove the leads at the heaters. Electrical leads disconnected at the breaker must be taped with electrical tape.

3. Remove preheat conduits, wire trays, and junction boxes for the pump case and associated piping as required to facilitate removal.

4. Disconnect and remove the temperature indicating instrumentation from the pump cases and their associated sodium lines as necessary to permit removal of the pump cases. Shut off power to all secondary, sodium-pump, indicating devices.

5. Remove thermal insulation from secondary sodium lines No 217 (2-in. vent line), 267 (14-in. suction line), 270 (14-in. pump discharge line), 306 (8-in. overflow line), and 324 (2-in. bypass line) as required to permit access for cutting the lines where they connect to the pump case.

6. Carefully remove the pump case heating furnace. Do not damage the furnace as it will be salvaged with the pump case.

7. Detach and bend the line heaters away from the sodium pipes where the insulation was removed.

8. Remove the flexible cooling duct from the pump case. This is also a reusable item; therefore, do not damage it.

CAUTION

The personnel performing the sodium line cutting operation shall suit in the proper protective clothing as specified by Health and Safety and as detailed in the Special Work Permit for this work.

Nofity CPPD shift supervisor prior to making each penetration into sodium line.

9. Reduce to 1/2 to 1 in. w. g. the gas pressure in the secondary heat transfer loop which pertains to the pump case being removed by opening; and then close the following valves.

Loop No. 1	V-8179	Secondary Sodium Expansion Tank No. 1 Vent
Loop No. 2	V-8185	Secondary Sodium Expansion Tank No. 2 Vent
Loop No. 3	V-8191	Secondary Sodium Expansion Tank No. 3 Vent

10. Maintain 1/2 to 1-in. w. g. inert gas pressure in the heat transfer piping by operating the helium system gas supply valves listed below which pertain to the pump case to be removed. Install an oil manometer containing oil with specific gravity of 1 to monitor the pressure.

Loop No. 1	V-8175	Secondary Sodium Expansion Tank No. 1 Helium Supply Block Valve
	V-8177	Secondary Sodium Expansion Tank No. 1 Helium Supply Throttle Valve

Loop No. 2	V-8181	Secondary Sodium Expansion Tank No. 2 Helium Supply Throttle Valve
	V-8183	Secondary Sodium Expansion Tank No. 2 Helium Supply Throttle Valve
Loop No. 3	V-8187	Secondary Sodium Expansion Tank No. 3 Helium Supply Block Valve
	V-8189	Secondary Sodium Expansion Tank No. 3 Helium Supply Throttle Valve

11. Install an inert gas connection with pressure indicators on the pump case cover plate, for adding inert gas as necessary and to monitor the pump case pressure.

NOTE

The lines connected to the pump case are of different sizes, and some are more accessible than others; therefore, the most suitable pipe cutting mechanism will be used for a particular location. Prior to cutting sodium piping, the pipe shall be checked to determine if it is adequately supported. When removing a section of pipe from the system, the spool being cut loose shall be properly supported before starting to cut the pipe.

12. Cut sodium line No. 324, 2-in. pump bypass, near where it connects to sodium line No. 267, 14-in. pump suction. Tape seal the cut or install plastic caps over the open ends of the pipe.
13. Cut sodium line No. 267, 14-in. pump suction, where it connects to the pump case. Tape seal the cut or install plastic caps over the line openings.
14. Cut sodium line No. 267, 14-in. pump suction, upstream from the previous cut and remove the pipe spool from the work area. Install plastic caps over the line openings.
15. Cut sodium line No. 270, 14-in. pump discharge, where it connects to the pump case. Tape seal the cut or install plastic caps over the line openings, whichever is applicable.
16. Cut sodium line No. 270, 14-in. pump discharge, downstream of sodium line No. 324 connecting nozzle. Install plastic caps over the line openings and remove the pipe spool from the work area.
17. Cut sodium line No. 306, 8-in. overflow, near the pump case nozzle. Tape seal the cut or install plastic caps, whichever is applicable.
18. Cut sodium line No. 306, 8-in. overflow, upstream from the previous cut. Install plactic caps over the line openings and remove the loose section of pipe from the work area.
19. Cut vent line No. 217 near where it connects to the pump case. Tape seal the cut or install plastic caps.

20. Cut vent line No. 217 upstream from the previous cut. Install plastic caps over the line openings and remove the loose section from the work area.

21. Attach lifting slings to the pump case lifting eyes.

NOTE

A dynamometer shall be used for load indication

22. Remove the hold bolts from the pump case mounting flange and carefully hoist the pump case from its position.

23. Place the pump case in an approved storage area or transport it to the steam cleaning facility. Preparation and packaging for off-site shipment shall be per Atomics International packaging and shipment specifications.

NOTE

If the steam generator has not been isolated as of this time, the following Steps 24 through 31 must be performed to provide a means of maintaining an inert atmosphere in the steam generators.

24. Remove the temporary seal covering the end of the steam generator side of line 270.

25. Install a 14-in. inflatable seal, one foot into the line, and pressurize to seal the inert atmosphere.

26. Remove residual sodium in this portion of the heat transfer piping.

27. Determine that a 14-in. diameter steel capping plate equipped with a 1-in. bar stock valve will fit inside this portion of line. It is suggested that a handle be tack welded to the capping plate to permit positioning the plate quickly.

28. Deflate the seal sufficiently to permit sliding it out of the line.

29. Immediately install a cap plate to a depth of 3/8 in.; tack weld in position.

30. Tape seal the annulus between the cap plate and proceed to make a seal weldment.

31. Attach an inert gas supply with a pressure regulator and a flow indicator to the 1/4-in. bar stock valve on the cap plate to provide means of maintaining the inert atmosphere in the steam generators.

32. A daily progress record (log book) of the task shall be kept and delivered to CPPD at the completion of the secondary sodium pump case removal. This record shall include man hours, unusual problems encountered, and any unusual conditions that may have developed in the course of this work.

PROCEDURE 13.
ISOLATION OF STEAM GENERATOR EVAPORATORS AND SUPERHEATERS FROM THE SECONDARY SODIUM SYSTEM

I. PURPOSE

This procedure establishes a step-by-step sequence for isolating the three steam generators (three superheaters - three evaporators) from their related secondary sodium systems as part of the retirement of the Secondary Heat Transfer Systems and to permit the removal of the steam generators for off-site shipment.

II. TASK DESCRIPTION

The activities to be accomplished by this procedure are as follows.

Disconnect and remove the electrical service to the steam generator superheaters, evaporators, and the associated piping; disconnect and remove the temperature indicating instrumentation from the steam generator superheaters and the associated piping; remove thermal insulation from the steam generator superheaters and evaporators as required to provide access for cutting the sodium lines; remove line heaters as necessary; cut and cap the steam generator superheater and evaporator sodium line stubs; and provide a mean s of pressurizing the sodium side of the steam generators with inert gas.

III. CRITERIA

A. REQUIREMENTS AND PREREQUISITES

1. This activity shall be coordinated with other plant activities during the HNPF retirement program.
2. All operation of plant equipment and deenergizing of electrical circuitry required in the performance of this procedure shall be performed by CPPD plant personnel.
3. Notification shall be made to the CPPD shift supervisor prior to initiating this work and also immediately prior to making any penetrations into piping or components which may effect, or require special operation of any associated plant system or component.
4. Personnel assigned to specific noninterruptable tasks (such as pipe cutting, closure welding, and sodium removal and handling shall not be given additional assignments until their particular phase of that task is completed.
5. Air shall not be permitted to enter the steam generator superheaters and evaporators during the cutting and capping operation.
6. Dry ice shall be used as a heat sink when welding capping plates if the particular situation warrants its use

7. Tools and equipment required for the work detailed is listed below.

   a. Hand tools - wrenches, pliers, hammers, lagging saws, tin snips, wire cutters, screw drivers, spatulas, sodium augers, and scrapers

   b. Skill saw - equipped with carborundum blade

   c. Pipe cutting equipment - Fein pipe saw or equivalent

   d. Porter-Cable band saw - spare blades

   e. Hoisting and lifting gear as required to handle the removed pipe spools

   f. Portable 1/2-in. electric drill motor

   g. Holes saws - 4-in. diameter - 12 each

   h. Inflatable bladders - 4 each - sized for sealing 14 and 20-in. pipe when inflated

   i. Supply of dry calcium carbonate

   j. Capping plates - 1/4-in. thick mild steel - 1-in. diameter - 6 each - equipped with 1/4-in. carbon steel bar stock valves

   k. Capping plates - type 304 SS - 1/4-in. thick - 12 each - 14-1/2-in.[2]

   l. Capping plates - mild steel - 1/4-in. thick - 6 each - 20-in. diameter

   m. Dry ice - procure as needed

   n. Ladders and scaffolding as required

   o. Weldment dye penetrant inspection kit

   p. Nitrogen bottles - equipped with regulators, fitting, and hose - 2 each

   q. Relief valve - gas service, set at 10 psig, 2 each

8. The secondary sodium loops have been drained and isolated from the secondary Na service system; therefore, operation of sodium valves is not required for the work detailed in this procedure.

9. The inflatable bladder technique shall be used to maintain an inert atmosphere on the steam generator superheaters and evaporators.

10. Plastic seal caps or equivalent shall be used on exposed sodium piping. Metal seal plates must be welded in place before leaving the work area without surveillance.

11. The sodium pressure sensing elements shall be removed from the heat transfer piping by CPPD personnel prior to initiation of Section IV of this procedure.

12. The following prerequisite items shall be completed and signed off prior to initiating performance of the work detailed in Section IV, "Method of Performance and Sequence of Events", of this procedure.

   a. Attach to this procedure a copy of the appropriate P and I drawing marked to show the equipment and system isolation to be performed.

      NOTE: Drawings shall be marked in accordance with physical inspection of the system.

b. Personnel assigned to the task shall study and sign the procedure.

c. Potential hazards relating to the work to be performed shall be discussed with the personnel involved before the operation is initiated.

d. Electrical power to the steam generators shall be deenergized and tagged "out of service" in accordance with normal HNPF tagging procedures.

e. Confirm that the entire Main Secondary Heat Transfer System is at ambient temperature prior to initiating the cutting of any of the sodium pumping.

B. HEALTH AND SAFETY

1. Special Work Permits are required for all penetrations, cutting, handling, cleaning, and welding of sodium piping, and for sodium handling operations.

2. Sodium protective clothing shall be worn at all times by all personnel while within the working areas.

3. The buddy system is required for all sodium handling, pipe cutting, and welding operations.

4. Deposit all sodium scrap in metal containers containing calcium carbonate and deliver them to the storage location designated by Health and Safety.

   NOTE: No sodium scrap or sodium contaminated component may be left open or unattended except as specifically authorized by Health and Safety.

5. Calcium carbonate and sodium fire equipment must be kept standing by at all times during cutting, cleaning, and welding of sodium piping and during handling of sodium scrap.

6. Disposal of sodium equipment and materials shall be performed only as directed by Health and Safety.

C. STANDARDS

Work performed in connection with the activity will be carried out under normal HNPF Health and Safety regulations (including sodium safety regulations) in accordance with Part III of the HNPF Operations Manual and, specific requirements of this procedure. Inspection of the capping plate weldments shall include dye penetrant testing of each root pass and of the final pass.

IV. METHOD OF ACCOMPLISHMENT AND SEQUENCE OF EVENTS

1. All preparatory work shall be completed before starting the work of this section of the procedure.

2. Confirm that the preheat electrical supply has been deenergized and tagged out per Part III, Section A, Step 12d. Disconnect the steam generator superheater and evaporator electrical supplies at the breakers by moving fuses and/or by lifting the electrical leads feeding the heaters. All electrical leads must be taped with electrical tape.

3. Remove electrical leads at the heaters, associated conduits, and junction boxes as necessary to facilitate cutting and capping of sodium lines. The extent of the pipe heater removal shall be determined by the physical inspection of the system with the objective of removing the steam generators from the facility.

4. Remove the temperature ;indicating instrumentation from the steam generator superheaters and evaporators. These items shall include the thermocouple lead wires, conduits, and junction boxes as is necessary to permit subsequent removal of the steam generators.

5. Remove thermal insulation from secondary lines No. 312 (superheat outlet - evaporator inlet), and No. 315 (evaporator vent and superheater drain) as necessary to provide access for cutting the lines.

6. Remove the line heater reflector oven from the uninsulated areas of sodium lines No's. 270, 273, 312, and 315.

7. Remove the line heaters from the exposed areas of sodium lines No's. 270, 273, 312, and 315.

8. Confirm that the inert gas pressure on secondary heat transfer loop No. 1 has been reduced to 2-in. w.g. pressure or less.

NOTE: Use oil of specific gravity of 1 in the manometer. This step will also be applicable to No. 2 and No. 3 secondary loops when No. 2 and No. 3 steam generators are isolated.

CAUTION

The personnel performing the sodium line cutting operation shall suit in the proper protective clothing as specified by Health and Safety and detailed in the Special Work Permit issued for this work. Notify CPPD shift supervisor prior to making each penetration into sodium lines.

NOTE: An inert gas atmosphere shall be maintained on the steam generator superheaters and evaporators during the isolation operation.

9. Cut sodium line No. 315 at the evaporator nozzle and at the superheater nozzle. This is a 1-in. diameter pipe and can be cut and sealed quickly.

10. Install plastic caps over the line stubs on the superheater and the evaporator. Prevent loss of inert gas atmosphere in the modules.

11. Remove line No. 315 and seal both ends with plastic caps before transferring to an approved storage area.

12. Remove the plastic cap from line stub No. 315 on the steam generator evaporator.

13. Remove any residual sodium from inside the line stub.

14. Position 1/4-in. mild steel capping plate (equipped with a 1/4-in. bar stock valve) over the line stub and tack weld in position.

15. Complete the capping plate weldment and dye penetrant inspect the weld.

16. Remove the plastic cap from line stub No. 315 on the steam generator superheater.

17. Remove any residual sodium from inside the line stub.

18. Position a 1/4-in. mild steel capping plate (equipped with a 1/4-in. bar stock valve) over the line stub and tack weld in position.

19. Complete the capping plate weldment and dye penetrant inspect the weld.

20. Connect a regulator equipped nitrogen bottle to the 1/4-in. bar stock valve on line stub No. 315 on the evaporator.

> NOTE: The line connecting the nitrogen bottle to the bar stock valve shall be equipped with a flow indicator and a pressure regulator. The bar stock valve on line stub No. 315 on the steam generator superheater can be used as a vent valve if it is required during the cutting and capping of sodium lines No. 270 and No. 273.

21. Make a 4-in. hole saw penetration into sodium line No. 273 (14-in. diameter - evaporator outlet) near where it connects to the evaporator. The exact location of the hole penetration shall be determined after the pipe has been exposed.

22. Install inflatable bladders in line No. 273 in both directions from the hole saw penetration. Inflate sufficiently to seal the line. Tape seal the hole saw penetration.

23. Cut sodium line No. 273 near the evaporator nozzle and tape seal the cut.

24. Cut sodium line No. 273 on the other side of the hole saw penetration; seal the pipe spool with plastic caps and remove it from working area. The inflatable bladders must be retained in position during the line cutting operation.

25. Remove the residual sodium from inside the line stub on the evaporation, to the bladder. The sodium must be removed - 10-in. from where the capping plate weldment is made.

> NOTE: Check the 1/4 to 14-1/2-in.^{2} SS capping plate for fit on the line stub. It is suggested that a handle be attached to the capping plate to make it easy to handle.

26. Deflate the bladder in the line stub just enough to permit sliding it out of the line stub. Immediately position the capping plate and tack weld it into position.

27. Seal the annulus between the capping plate and the line stub with tape to prevent undue loss of inert gas.

28. Apply the weldment root pass to the capping plate, removing the tape as necessary to permit welding.

29. Dye penetrant inspect the weldment root pass.

30. Complete the weldment, and dye penetrant inspect the final filler pass.

31. Remove the inflatable seal and install a plastic cap over the open end of line No. 273 (piping system side). If the IHX has not been isolated at this time, it will be necessary to weld metal capping plates on lines No. 270 and 273 to maintain an inert atmosphere in the IHX.

32. Make a 4-in. hole saw penetration into sodium line No. 270 (14-in. diameter - superheater sodium inlet). The location of this penetration will depend on the accessibility of the line. The eliminator may interfere with the pipe cutting machine. This shall be dtermined before making the hole saw penetration.

33. Cut and cap line No. 270 as per procedural Steps 22 through 31.

34. Remove the inflatable seal and install a plastic cap over the open end of sodium line No. 270 (pipe system side).

35. Connect a regulator equipped nitrogen gas bottle to the bar stock valve on the capping plate on line stub No. 315 on the steam generator superheater.

> NOTE: The gas line from the bottle to the bar stock valve must be equipped as specified in step No. 20. An inert atmosphere shall be maintained on both the superheater and the evaporator during the cutting and capping of sodium line No. 312.

36. Make a 4-in. hole saw penetration into sodium line No. 312 (20-in. diameter Cr Mo - superheater to evaporator crossover line) near where it connects to the evaporator inlet nozzle.

37. Install an inflatable bladder into the evaporator nozzle and another upstream from the hole saw penetration. Inflate the bladders sufficiently to seal the pipe.

38. Tape seal the hole saw penetration in line No. 312.

39. Make a 4-in. hole saw penetration in line No. 312 near where it connects to the superheater outlet nozzle.

40. Install an inflatable bladder into the superheater nozzle and another upstream of the hole saw penetration. Inflate the bladders sufficiently to seal the pipe.

41. Cut sodium line No. 312 near where it connects to the evaporator nozzle.

CAUTION

> Line No. 312 is a large section of 20-in. diameter pipe and will have to be properly supported before it is cut.

42. Cut sodium line No. 312 near where it connects to the superheater outlet nozzle. Seal the pipe spool and lower the pipe to the floor.

43. Install 1/4-in. thick mild steel capping plates onto the evaporator nozzle and the superheater nozzle per Steps 26 through 30.

44. Pressurize the superheater and the evaporator with nitrogen gas to 5 psig. Close and seal the 1/4-in. bar stock valves on the capping plates.

45. Remove pipe spool No. 312 from the work area. This pipe spool may have to be cut into two pieces before it can be removed.

46. The isolation of steam generators No. 2 and 3 shall be accomplished as specified by this procedure. The proper line number shall be entered on the field copy of the procedure before starting the task.

> NOTE: Refer to appropriate P and I drawing for valve and line numbers.

47. A daily progress record (log book) of the task shall be kept and delivered to CPPD at the completion of the isolation of the steam generator, superheaters, and evaporators.

PROCEDURE 14.

DISMANTLING AND REMOVAL OF FEEDWATER-AND-STEAM-SYSTEM PIPING-AND-COMPONENTS

I. PURPOSE

This procedure establishes a sequence of events for isolating the three steam generators from their related feedwater and steam piping systems.

II. TASK DESCRIPTION

The activities to be accomplished by this procedure are as follows. Disconnect and remove the electrical service to the various electrically operated components (except lighting and service outlets) within the confines of the steam generator rooms. Shut off and disconnect the instrument air supply to the steam generator rooms. Disconnect and remove the temperature indicating instrumentation from the steam generator evaporators, superheaters, eliminators, and their associated steam and water lines. Remove thermal insulation from the steam generator modules and their associated steam and water lines. Remove steam and feedwater pipes as necessary to permit removal of the steam generators. This procedure in conjunction with the procedure for "Isolation of Steam Generator Evaporators and Superheaters from Secondary Sodium System" prepare the steam generators as a unit for removal to a position where they can be removed as individual components from their rooms by using the high bay bridge crane.

III. CRITERIA

A. REQUIREMENTS AND PREREQUISITES

1. This activity will be coordinated with other plant activities during the HNPF retirement program.
2. Plastic caps or other temporary sealing devices shall be used on exposed steam and water piping connections on the steam generator modules to prevent debris from falling into the interior of the vessels.
3. The sodium pressure instrument must be removed before initiating this procedure.
4. The mechanism used for cutting the feedwater, steam, and other miscellaneous steam generator piping will be determined by physical inspection of the particular line. Distortion of salvageable items by flame cutting shall not be permitted.
5. The physical location of the pipe cutting directly related to the isolation of steam generators will be determined by CPPD or by persons authorized by CPPD.
6. The interconnecting lines which support the unit shall not be cut unless specified per this procedure. It is intended that the generators be relocated as units before the interconnecting lines which support the units are cut.
7. Tools and equipment required for the work detailed in this procedure are as follows.
   a. Hand tools - wrenches, pliers, hammers, lagging saws, tin snips, wire cutters, and screw drivers

b. Pipe cutting equipment - acetylene cutting torch, electrically driven pipe saw - Fein or equivalent

c. Porter-Cable band saw

d. Welding machine - equipped for inert gas welding

e. Hoisting and rigging gear as required for pipe handling

f. Dye penetrant weldment inspection kit

g. Capping plate - 1/4-in. thick - carbon steel - sixed for capping all steam and water lines connecting to the steam generator vessels

h. Bar stock 1/2-in. valves or other acceptable valves - 9 each

i. Ohmmeters - 2 each

j. Scaffolding - as required

k. Wood cribbing - as necessary

l. Loading indicating devices - dynamometers

9. The following prerequisite items shall be completed and signed off prior to initiating performance of the work detailed in Section IV, "Method of Performance and Sequence of Events" of the procedure.

a. Attach to this procedure a copy of the appropraite P and I drawing marked to show the equipment isolation to be performed.

NOTE: Drawings shall be marked in accordance with physical inspection of the system.

b. Potential hazards relating to the work to be performed shall be discussed with the personnel involved before the operation is initiated.

c. Confirm that the steam-water side of the steam generators have been drained and that they are not pressurized with inert gas.

d. Deenergize and tag out the electrical power to the motor operated feedwater stop valves, solenoid valves, temperature readout instruments, and other electrically operated components within the confines of the steam generator rooms.

e. Secure and tag out the instrument air supply to the steam generator rooms.

f. Secure the helium supply to the third fluid system on the steam generators.

B. HEALTH AND SAFETY

Electrical wiring shall be checked for voltage prior to making physical contact with uninsulated connections or wires.

C. STANDARDS

Work performed in connection with this activity will be carried out under normal HNPF Health and Safety regulations and CPPD safety practices. The final pass of the capping plate weldments will be dye penetrant inspected.

IV. METHOD OF ACCOMPLISHMENT AND SEQUENCE OF EVENTS

1. All preparatory work shall be completed before starting the work of this section of the procedure.

2. Confirm that the electrical supply (D.C. and A.C. power) to the various electrically operated components within the confines of the steam generator rooms has been deenergized and tagged out per tag-out procedure. Disconnect the electrical supplies to the components at the breakers by removing fuses and/or lifting the leads. Electrical leads disconnected at the breaker must be taped.

3. Remove electrical conduits, wire trays, and junction boxes only as necessary to permit removal of the steam generator from the facility.

4. Disconnect and remove the temperature indicating instrumentation from the steam generator modules.

5. Confirm that the instrument air supply to the generator rooms has been secured and tagged out and then remove the instrument air supply lines from the various components. Example: air actuated flow control vlaves, pressure transmitting instruments, etc.

6. All items that are to be salvaged will be tagged as salvageable items prior to starting the cutting and capping operation. This is to prevent their being damaged by assuming that they are scrap.

7. Disconnect and remove the steam pressure and water level transmitter instrument panel from the steam generator room.

8. Disconnect and remove the instrument panel, Local Board No. 1, from the steam generator room.

9. Remove the fire wall, including the wall support columns, from the south end of the generator.

10. Remove the air driven sodium throttle valve operator including the air accumulator tank.

11. Remove thermal insulation from the steam generator modules and their associated steam and water piping. Remove insulation from the working area.

12. Remove the automatic flow control valve from the evaporator feedwater inlet line. The valve body is flanged and will not require cutting.

    NOTE: It is not necessary to remove the feedwater system piping upstream from the flow control valve to permit removing the generator however salvageable items will be removed for off-site disposition.

13. Remove the feedwater system piping, valves, and warmup steam heater from the downstream side of the automatic flow control valve. This will be an obvious operation by physically inspecting the system.

14. Remove the blowdown system valves and piping from the steam generator evaporator and superheater.

15. Remove the water column assemblies from the steam generator evaporators.

16. Remove the chemical water treatment lines from the evaporator.

17. Cut and remove the (4-in. diameter - Schedule 80) steam attemperating line including the two automatic valves. This line connects the 10-in. saturated steam line to the 10-in. superheater steam outlet line.

18. Cut out and remove the 10-in. steam outlet valve from the superheater steam line including the valve blowdown line.

19. Disconnect the superheater and eliminator relief valves from their related piping.

20. Cut and remove the 3/4-in. vent line that vents the 10-in. saturated steam line connecting the eliminator to the superheater.

21. Cut and remove the superheater vent line.

22. Cut and remove the eliminator steam warm-up line.

23. Remove the 2000-psig pressure gage from the eliminator.

24. Confirm that the third fluid is vented before cutting the helium piping.

25. Cut and remove the 1/2-in. helium supply lines from the third fluid system of the evaporator and the superheater vessels.

26. Weld a capping plate on vessel nozzles which are open. Removal of the steam generators will be per Atomic International's recommendations.

APPENDIX III
SODIUM STEAM-CLEANING FACILITY

The cleaning facility consisted of two corrugated steel buildings as shown in Figures 1 and 2, each 16 ft square, on concrete pads. The facility was located at the south end of the reactor building. One building was used for actual cleaning of parts; the other for processing the exhaust air before venting it to the stack. This facility was erected for the retirement program, and dismantled after its need was fulfilled.

The cleaning building was provided with steam for cleaning the pipes and some components from the plant heating system. An exhaust duct near the ceiling led to the filter building. In the floor, a drain was installed which led to the radioactive liquid waste system where the washings were monitored. From the liquid waste system, washings were led to the existing leach field. Drain flow could be diverted to liquid waste storage tanks in the event of high radiation levels (this never occurred). Flow to the leach field was diluted with ordinary water in order to maintain a low sodium hydroxide concentration.

The filter building contained a blower, water scrubber, and a filter. The suction side of the blower was connected to the cleaning building exhaust vent. Exhaust air was passed through the scrubber and filter and then vented to the stack where it was diluted with air from the stack blowers.

During operation, pipe sections and components were brought from their temporary storage point in the plant to the cleaning facility. Two men, dressed in protective clothing, used a steam lance to clean the pipe sections. These sections were then checked for radioactivity and visual cleanliness, and were brought to the outdoor scrap yard. All cleaning operations were performed by CPPD personnel.

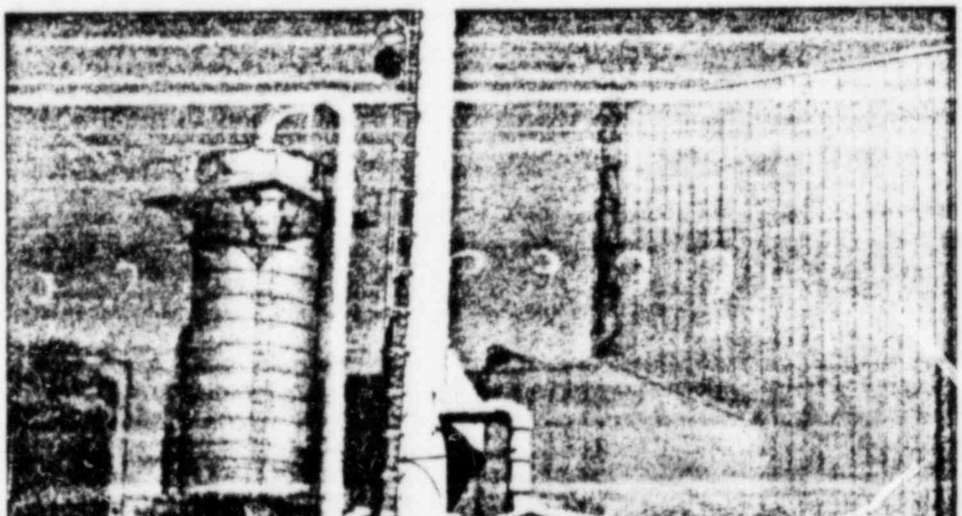

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Figure 1. Sodium Disposal Facility Buildings

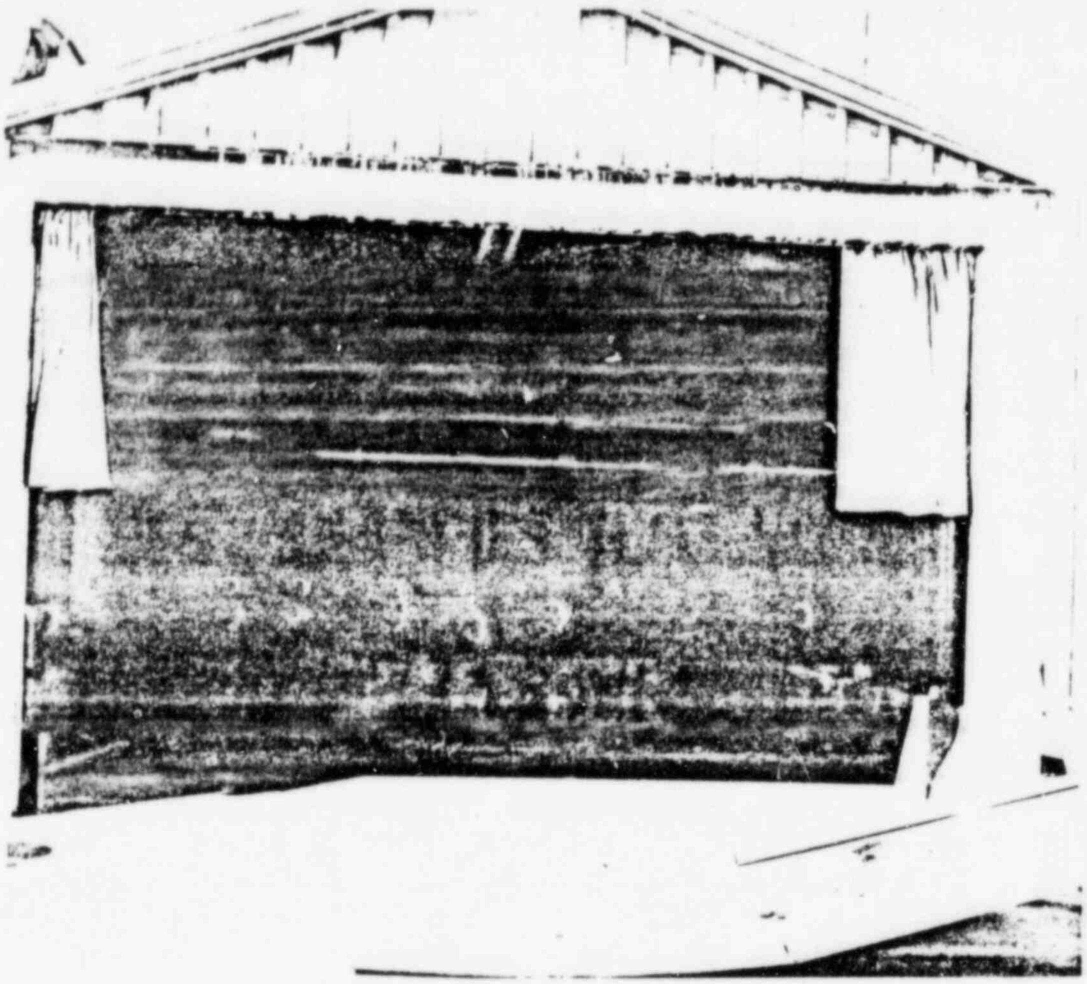

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Figure 2. Interior of Sodium Cleaning Building

APPENDIX IV

EVALUATION OF LONG-TERM HYDROGEN HAZARD

A. ESTIMATE OF MAXIMUM RESIDUAL WATER

In order to evaluate the potential of generating a flammable atmosphere due to radiolytic disassociation of water, it was necessary to estimate the total quantity of water remaining in the reactor vessel. A conservative approach was used in making this estimate in order to avoid an overly optimistic evaluation of any potential hazards.

1. Final Steaming and Drying Conditions

Final steaming of the reactor vessel, after a zero effluent hydrogen concentration was reached, was accomplished by using about 200 cfm of 80% steam (20% nitrogen) at a temperature of 350°F. After the steaming process was completed, 410°F nitrogen was purged through the system at a flow rate of 80 cfm for 24 hr. During the final steaming and drying, the reactor pressure was kept at approximately 15 psia, and the vessel heaters were turned on. Two hours before the drying period ended, it was observed that the condensate flow from the effluent condenser had apparently ceased. Effluent gas temperature from the condenser was about 61°F.

2. Explanation of Phase Diagram

Figure 1 is a partial phase diagram of the NaOH-H_2O system, covering the region between 80 and 460°F for mixtures containing 60 to 100% NaOH by weight. Superimposed on the diagram, are isobars representing the equilibrium partial pressure of water over aqueous NaOH solutions. The heavy line represents the boundary between various equilibrium phases. Below 140°F, mixtures between 6 and 100% NaOH are solid, containing pure NaOH and hydrated crystals of NaOH·H_2O. Above 150°F, the mixture may exist as a pure liquid NaOH solution or as a mixture of liquid NaOH solution and solid NaOH.

The partial phase diagram was obtained from Figure 1.111 referenced.* The superimposed isobars were obtained from the data listed in Table 3-27; see reference.† Extension of the isobars to the liquid-solution-and-solid-NaOH phase region was done by first extrapolating the isobar data from Table 3-27 to

*M. C. Sneed and R. C. Brasted, "Comprehensive Inorganic Chemistry," Vol. 6, Part I – The Alkali Metals by J. F. Suttle, D. Van Nostrand Co., Inc., 1957

†R. H. Perry et al., "Chemical Engineers Handbook," 4th Ed. McGraw-Hill Co., 1963

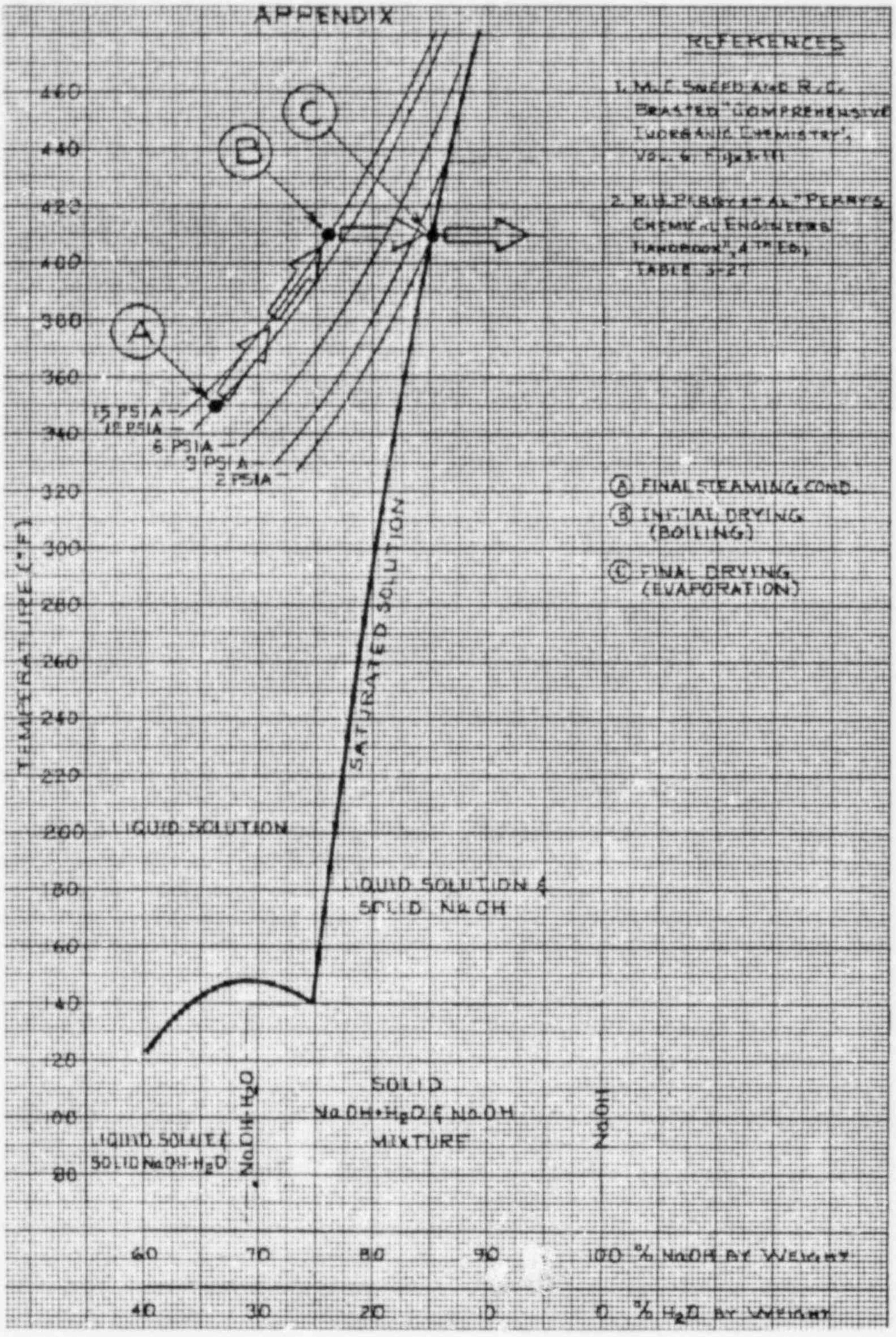


Figure 1. Partial Phase Diagram for Aqueous Sodium Hydroxide Solutions With Lines of Constant Equilibrium H_2O Partial Pressure

The phase region (saturated solution line) and then by following a constant temperature line in the liquid-solution-and-solid-NaOH phase.

3. Calculation of Residual Water

a. Final Steaming Conditions

P Pressure = 15 psia

t_a Temperature = 350°F

f_a Steam fraction = 0.8 (by volume)

P_a Partial pressure of steam (water vapor)

$$P_a = f_a \times P = 0.8 \times 15$$

$$P_a = 12 \text{ psia}$$

Point "A" on Figure 1 shows that at 350°F, a 66% by weight sodium hydroxide solution would remain in equilibrium with an atmosphere having 12 psia partial pressure of water. For HNPF, the assumption of equilibrium is not unreasonable, since these conditions existed for 24 hr. For the calculated 483 lb of sodium reacted,* this corresponds to 433 lb of water and 840 lb of sodium hydroxide.

b. Initial Drying Conditions

P Pressure = 15 psia

t_b Temperature = 410°F

After steaming, nitrogen at approximate 410°F was circulated through the reactor. The initial effects of this was to start the sodium hydroxide solution boiling, thus increasing the NaOH concentration. Boiling continued until the NaOH concentration reached 76%, Point "B" on Figure 1. At this point, the equilibrium partial pressure of water vapor above the NaOH solution was equal to the total pressure in the reactor vessel, 15 psia. Further loss of water from the solution at 410°F lowered the equilibrium partial pressure of H_2O below the total pressure and stopped the boiling process. At point "B" the calculated reactor residue was 266 lb of water and 840 lb of sodium hydroxide.

*T. J. Boardman and T. A. Paulett "HNPF Retirement, Disposition of Sodium," AI-AEC-MEMO-12736, September 30, 1968, p 67

c. Final Drying Conditions

From point "B" evaporation continued at constant temperature toward point "C" on Figure 1 (410°F, 85% NaOH), where solid sodium hydroxide started to precipitate from the solution. For the purpose of a conservative safety analysis, point "C" is taken as the end of the drying process. At point "C," the residue composition would be 149 lb of water and 840 lb of sodium hydroxide.

A conservative estimate of the maximum time required to concentrate the residual NaOH solution from point "A" to point "C" in Figure 1 may be made by assuming that all evaporation took place by diffusion of water vapor from the solution surface into the nitrogen. This assumption neglects the accelerating effects of boiling, from point "A" to point "B," and of the gas velocity sweeping over the solution surface. An approximate equation for diffusion evaporation from a flat surface is:*

$$w = 0.00138(p_w - p) \quad ,$$

where:

w = Evaporate rate (lb/ft^2-hr),

p_w = Partial pressure of H_2O at surface (mm Hg) (Assumed equal to equilibrium partial pressure), and

p = Partial pressure of H_2O at farthest point (mm Hg) (Assumed equal to zero).

The assumption of zero partial pressure at the farthest point is based on the entering nitrogen having practically zero water vapor.

From Figure 1, the minimum value for p_w throughout the process was 2 psia (approximate 100 mm Hg), as indicated at point "C." The vessel bottom had an 18-ft diameter, corresponding to an area (A) of 254 ft^2. Thus, the minimum evaporation rate ($\dot{W}_{min}$) may be calculated from:

$$\dot{W}_{min} = wA = 0.00138(p_w - p)A$$

$$= 0.00138(100 - 0)254$$

$$\dot{W}_{min} = 35 \text{ lb/hr} \quad .$$

*J. H. Perry, "Chemical Engineers Handbook," 3rd Ed., McGraw-Hill, 1950

The maximum time (t_{max}) based on the minimum rate, required to concentrate the NaOH solution from point "A" to point "C" by diffusion was:

$$t_{max} = (W_a - W_c)/\dot{W}_{min}$$

$$= (433 - 149)/35$$

$$t_{max} = 8.1 \text{ hr} \quad ,$$

where:

W_a = Residual water at point "A" (lb), and

W_c = Residual water at point "C" (lb).

During this process, the heat of solution and the latent heat of evaporation were supplied by the heaters on the bottom of the reactor vessel. The total heat required may be approximated by using the latent heat of evaporation at the vapor pressure at which evaporation occurred.* For point "C" (2 psia), this value would be 1022.2 Btu/lb.† Therefore, the maximum heat requirement would have been (433 - 149) x 1022.2 = 2.9×10^5 Btu, or 12,100 Btu/hr over the 24-hr drying period. The connected power available to the bottom heaters was 60 kw,§ equivalent to 205,000 Btu/hr. This power was of course modulated to maintain constant temperature.

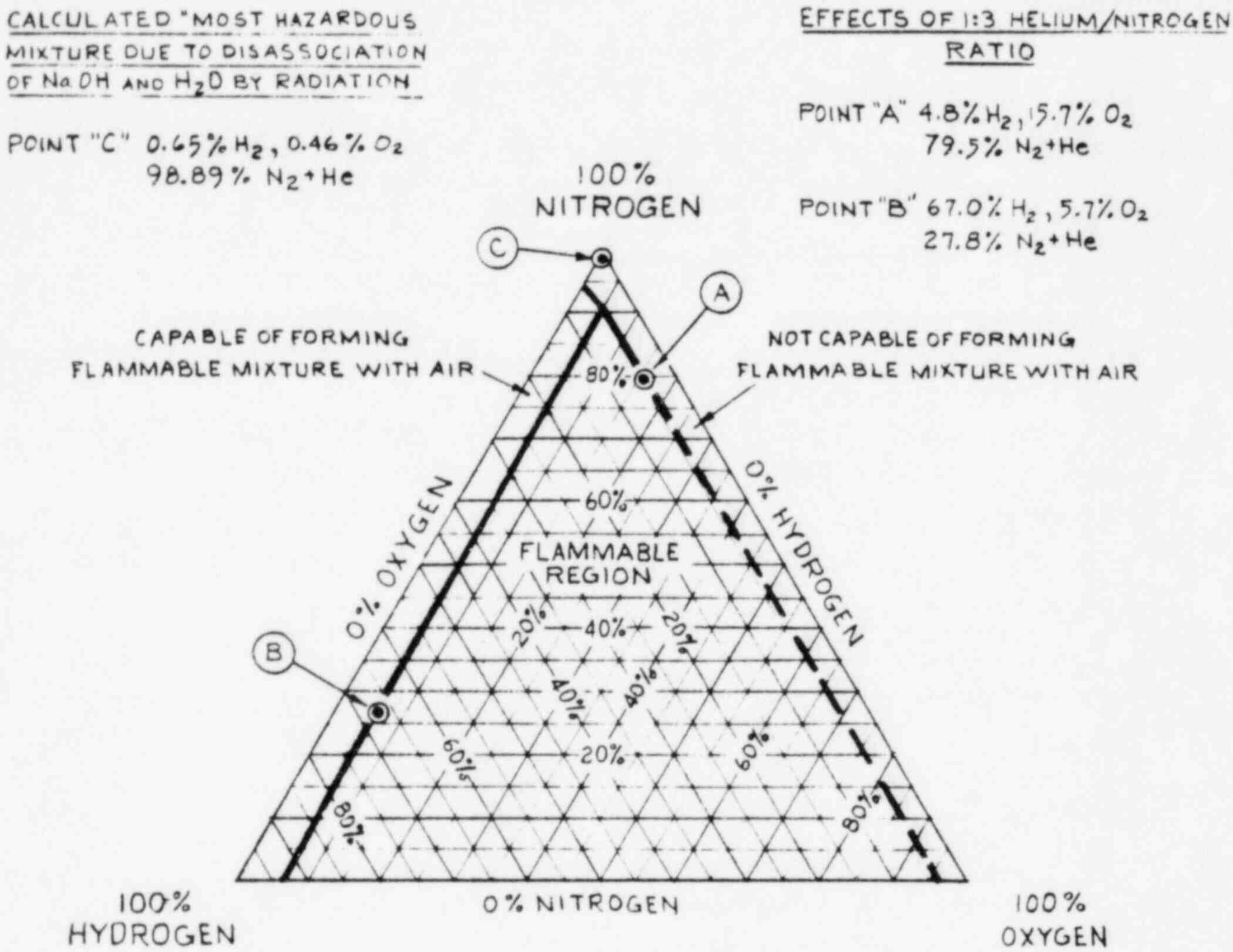
4. Conclusions

From the foregoing analysis, it can be seen that there was more than enough time and power available to concentrate the residual NaOH solution to 15% by weight and that the final residue probably contained even less water. Additional support to these conclusions is evidenced by the observation that no additional condensate was collected from the effluent condenser after twenty-two hours of drying and that the effluent relative humidity at the end of the drying period was less than 2% (0.0083 psia partial pressure of H_2O) when cooled to

*W. L. Badger and W. L. McCabe, "Elements of Chemical Engineering," 2nd Ed., McGraw-Hill, 1936, p 188

†J. H. Keenan and F. G. Keyes, "Thermodynamic Properties of Steam," Wiley and Sons, 1936

§J. E. Mahlmeister et al., "Engineering and Construction of the Hallam Nuclear Power Facility Reactor Structure," NAA-SR-7366, August 15, 1962, p 15



REFERENCE - H.F. COWARD AND G.W. JONES, "LIMITS OF FLAMMABILITY OF GASES AND VAPORS", BUREAU OF MINES BULLETIN 503, 1952, PART III

Figure 2. Flammability Region of Hydrogen-Oxygen-Nitrogen System

74°F. This corresponds to less than 0.002% relative humidity at the actual effluent temperature of 468°F.* When allowed to cool below 140°F, the sodium hydroxide residue solidified to the solid mixture of NaOH and $NaOH \cdot H_2O$ shown at the lower part of Figure 1.

B. ESTIMATE OF FLAMMABILITY HAZARD

After cooling down, the reactor vessel was kept under a positive nitrogen gas pressure in order to permit leak checking. Future radiolytic decomposition of the residue would result in the addition of hydrogen and oxygen to this atmosphere. In order to evaluate the flammability hazard, the combustible limits for hydrogen-oxygen-nitrogen mixtures were obtained from Figure 8 referenced† and plotted on a triangular diagram, see Figure 2. Next, the effects of an approximate 1:3 helium/nitrogen ratio were investigated by using the data of Table 3 referenced.§ This data permitted calculation if two points, one upper limit and one lower limit, for the 1:3 ratio. The helium effects are practically negligible (for this problem) as shown by the plot of the two points, "A" and "B," on Figure 2.

Point "C" in Figure 2 shows the calculated ultimate concentration of hydrogen and oxygen in the reactor vessel. Details of the calculations,§ are shown on the following pages. Since the point is well outside of the flammability range, a flammable mixture can not be formed when mixed with air; therefore, it is concluded that there will be no long term explosion hazard due to disassociation of residual water in the HNPF reactor vessel.

C. RADIOLYTIC DECOMPOSITION CALCULATIONS

Starting with the worst possible condition of 15% water of crystallization in the residual sodium hydroxide, a layer of this mixture $NaOH \cdot NaOH + H_2O$ is assumed to be spread evenly over the bottom of the reactor vessel. At a density of 2.3 and with an area of $2.6 \times 10^5 \, cm^2$, the thickness of such a layer would be 0.69 cm.

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*T. J. Boardman and T. A. Paulett, "HNPF Retirement, Disposition of Sodium," AI-AEC-MEMO-12736, September 30, 1968, p 68

†H. F. Coward and G. W. Jones, "Limits of Flammability of Gases and Vapors," Bureau of Mines Bulletin 503, 1952, Part III

§W. F. Heine, "HNPF Final Status Report," AI-AEC-MEMO-12794, 1968

The source of the radiation energy required for the radiolytic decomposition is almost exclusively the Co^{60} contained in the activated reactor components. Although there are large amounts of Fe^{55} activity in the reactor components, this activity does not contribute significantly to the radiolytic decomposition, due to the fact that Fe^{55} decays only by "K" electron capture; thus, virtually all of its disintegration energy (0.23 Mev) is released in the form of unabsorbable neutrinos. The only absorbable energy released from Fe^{55} is in the form of "K" x-ray photons; this represents about 7.6 kev/dis. Similarly, Ni^{63}, which is the third significant radionuclide in the irradiated reactor components does not contribute significantly to the hydrogen evolution, since it decays by emission of a 67 kev beta particle with no accompanying gamma photon. The low energy beta particle which has a maximum range in steel of less than 0.001 inch will be removed by self-absorption, except for those particles emitted very near the surface of the respective reactor components. In the case of Co^{60}, the disintegration energy is emitted in the form of two gamma photons with energies of 1.17 and 1.33 Mev, and in the form of a 0.23 Mev beta particle. Thus, the energy available for absorption is 2.6 Mev/dis Co^{60}, if it is conservatively assumed that half the beta energy escapes self-absorption.

Examination of the configuration of the irradiated reactor components reveals that only three portions of these components contribute significant energy flux to the NaOH layer: the bottom surface of reactor vessel, the lower grid plate, and the bottom head assemblies of the moderator elements. The portions of the vessel and components located above the bottom moderator head assemblies do not contribute significant fluxes, due to shielding by the moderator elements and the grid plate. The sides of the vessel located below the grid plate do not contribute significant fluxes due to their configuration.

The source geometry for the bottom of the reactor vessel may be conservatively assumed to approximate a 2-in. thick infinite slab with a specific Co^{60} activity of 3.2×10^{-5} Ci/cm^3. This activity assumes that a thermal neutron flux of 1.0×10^{10} n/cm^2-sec was present at the bottom of the vessel during reactor operation. Also conservatively, the grid plate and the lower moderator head assemblies may be combined to approximate the source geometry of a 2-in. thick infinite-slab source with a specific Co^{60} activity of 5.4×10^{-3} Ci/cm^3.

This activity assumes that a thermal neutron flux of 1.7×10^{12} n/cm^2-sec was present at the lower grid plate during reactor operation.

The effective geometry equation which will be used for this combined source to evaluate the energy flux incident on the sodium hydroxide layer is

$$\varphi \frac{BS_v}{2\mu_s}\left[E_2(b_1) - E_2(b_3)\right] \quad , \qquad \ldots(1)$$

where:

φ = Scalar flux (cm^{-2}-sec^{-1}),

μ_s = Linear absorption coefficient of source material (cm^{-1}),

t_1 = Thickness of ith shield (cm),

u_1 = Linear absorption coefficient of shield (cm^{-1}),

$b_1 = \mu_1 t_1$,

h = Thickness of slab source (cm),

$b_3 = b_1 + \mu_s h$,

$E_2(b) = b\int_b^{\infty}(e^{-t}/t^2)\,dt$,

S_v = Source strength (cm^{-1}-sec^{-1}), and

B = Symbolic buildup factor.

In the case at hand, the numerical values applied in the infinite slab equation are as follows:

$S_v = 3.2 \times 10^{-5}$ Ci/cm^3 or 3.1×10^6 Mev/sec/cm^3 (vessel bottom),

$S_v = 5.4 \times 10^{-3}$ Ci/cm^3 or 5.2×10^8 Mev/sec/cm^3 (grid plate),

$\mu_s = 0.40$ cm^{-1},

$t_1 = 0$ cm,

B = 3.0 (dimensionless), and

h = 5.08 cm.

Application of these values to Equation 1, and use of the E_2 function curves from Reference,* reveals that the sum of the energy incident upon the NaOH layer as a result of irradiation by the vessel bottom, the lower grid plate, and the lower moderator head assemblies is 1.8×10^9 Mev/cm^2-sec. Applying a mass absorption coefficient of 0.06 cm^2/gm for the NaOH allows the evaluation of the energy absorption in the NaOH as a function of total mass by the equation

$$E_a = E_i \cdot \mu_m \cdot M \quad , \qquad \ldots(2)$$

where:

E_i = incident energy flux (Mev/sec-cm^2)

E_a = absorbed energy (Mev/sec-cm^2)

μ_m = mass absorption coefficient (cm^2/gm), and

M = mass of absorber (gm).

On the basis of Equation 2, the rate of energy absorption in the NaOH layer is 4.17×10^{13} Mev/sec.

Since we are dealing with a 15% water content hydroxide, it may be shown that the rate of energy absorption in the water present in the NaOH-H_2O in the reactor vessel would be a maximum of 15% of the total energy absorption rate, or a maximum of 6.3×10^{12} Mev/sec. Assuming a "G" value for the radiolytic evolution of hydrogen from water of 0.5 molecules H_2 per 100 ev absorbed reveals that the initial hydrogen evolution rate in the reactor vessel will be 3.1×10^{16} H_2 molecules/sec. This corresponds to a volume rate of 1.2×10^{-3} cm^3/sec at STP.

The rate of hydrogen evolution will decrease with time as the Co^{60} decays. Since the Co^{60} half life is 5.3 years, its mean life is 7.7 years, so that the total hydrogen release possible over an infinite time is:

3.1×16^{16} molecules/sec $\times$ 3.1×10^7 sec/yr $\times$ 7.7 yr = 7.5×10^{24} molecules

This total hydrogen evolution corresponds to 2.8×10^5 cm^3 at STP.

*T. Rockwell III, Reactor Shielding Manual TID 7004, March 1956

The void volume within the reactor vessel, after subtraction of the space occupied by the moderator elements and stored core components, is approximately 6.00×10^3 ft^3 or 1.70×10^8 cm^3. Thus, the release of 2.8×10^5 cm^3 of H_2 into this void would result in a hydrogen concentration of 0.16 vol %. This concentration is far below the lower limit of explosive concentration in an air atmosphere. Further, it may reasonably be assumed that no air is present in the reactor vessel since the vessel was sealed under a nitrogen atmosphere, and that any subsequent in-leakage of air would also provide an avenue of escape for hydrogen. Some oxygen, however, would be provided by the same radiolytic decomposition of water which provides the hydrogen. Since only one molecule of O_2 would be provided for each two molecules of H_2 from this phenomenon, a maximum oxygen concentration of 0.08 vol % could develop in the sealed reactor vessel. In the presence of this O_2 concentration, the calculated H_2 concentration is even further below the lower limit of explosive concentrations.

Additional hydrogen evolution may be expected to result from the radiolytic decomposition of the NaOH. In order to determine the "G" value for the radiolytic evolution of hydrogen from NaOH, an experiment was performed at Atomics International in which 11 grams of anhydrous NaOH was sealed in an evacuated glass apparatus and exposed to 3.1×10^7 roentgens in a high-level Co^{60} gamma irradiation facility. Assuming an energy absorption of 86 ergs/gm-roentgen, this irradiation resulted in the absorption in the NaOH of 2.9×10^{10} ergs or 2.5×10^{22} ev. Analysis by means of gas chromatography of the atmosphere contained in the apparatus, following the irradiation, revealed the presence in the atmosphere of 1.4×10^{19} molecules of hydrogen. Thus, on the basis of this experiment, the "G" value for radiolytic evolution of hydrogen from NaOH is 0.06 molecules per 100 ev absorbed.

Assuming an initial energy absorption rate of 4.2×10^{13} Mev/sec in the NaOH layer, there would result an initial hydrogen evolution rate of 2.5×10^{16} molecules/sec or a total hydrogen evolution of 6.1×10^{24} molecules over an infinite time period. These values correspond to 9.3×10^{-4} cm^3/sec and 2.2×10^5 cm^3 at STP respectively. Thus, the total maximum radiolytic hydrogen evolution from the NaOH and its water of crystallization over an infinite time period is 5.1×10^5 cm^3 at STP. This volume released into the reactor vessel void volume would result in a hydrogen concentration of 0.30 vol %. It is difficult to postulate conditions which would bring about the release of O_2 as a result of radiolytic decomposition of the NaOH, since the residue would tend to oxidize to Na_2O_2 following release of the hydrogen. However, if it is conservatively assumed that one atom of oxygen is released for each atom of hydrogen released radiolytically from the NaOH, there results, upon addition of the resulting O_2 to that evolved from the H_2O, a total of 3.6×10^5 cm of oxygen in the vessel atmosphere at STP, or 0.21 vol %. Since these values are also below the lower limit for explosive hydrogen-oxygen mixtures, as shown in Figure 2, it may be concluded that no hydrogen explosion will occur in the reactor vessel as a result of radiolytic decomposition of the NaOH and its contained water.

APPENDIX V

DISPOSITION OF MATERIAL AND EQUIPMENT

A. E. C. Case No.	Description	Quantity	HNPF Item Number	AEC Recipient	Date Shipped	Acquisition Cost
1	Graphite Logs	6	66-3-841	Ames Lab.	11/16/66	821
2	Graphite Logs	38	66-3-841	Nat. Lead	1/5/67	5,198
3	Plant Protective Panel	1	66-3-1	K. A. P. L.	3/17/67	2,900
	Amplifier	58	66-3-1-1			16,000
	Amplifier	48	66-3-1-2			7,900
	Amplifier	5	66-3-1-3			3,050
	Bl-Stable Unit	262	66-3-1-4			4,900
	Power Supply	20	66-3-1-5			1,500
	Power Supply	2	66-3-1-6			100
	Dual Relay Drive	46	66-3-1-7			750
	DC Logic Plug In Unit	19	66-3-1-8			480
	SFM Dual Plug In Unit	26	66-3-1-9			430
	Emitter	60	66-3-1-10			800
	DC Logic Plug In Unit	29	66-3-1-11			575
	DC Logic Plug In Unit	9	66-3-1-12			200
	Power Supply	1	66-3-1-13			150
	Transpac	7	66-3-1-14			530
	Transpac	1	66-3-1-15			80
	Power Supply	1	66-3-1-16			540
	Transpac	1	66-3-1-17			80
	Transformer	1	66-3-1-18			30
	Fan	5	66-3-1-19			1,000
	Fan	5	66-3-1-20			190
	Regulator	7	66-3-1-21			11,150
	Panel	1	66-3-1-22			480
	Transpac	2	66-3-1-23			150
	Transpac	3	66-3-1-24			230
	Emitter	18	66-3-1-25			480
	DC Logic Plug In Unit	2	66-3-1-26			50
	Relay	32	66-3-1-27			675
	Relay	7	66-3-1-28			100
	Relay	16	66-3-1-29			345
	Relay	2	66-3-1-30			100
	Ammeter	2	66-3-15			100
	Voltmeter	2	66-3-16			100
	Transpac	1	66-3-27			160
	Transpac	1	66-3-28			80
	Bl-Stable Unit	36	66-3-35			690
	Logic DC Plug In Unit	15	66-3-36			280

A.E.C. Case No.	Description	Quantity	HNPF Item Number	AEC Recipient	Date Shipped	Acquisition Cost
3 (Cont'd)	R. D. Dual Plug in Unit	23	66-3-37	K. A. P. L.	3/17/67	380
	Voltmeter	2	66-3-41			8,600
	Helipot	7	66-3-53			480
	Gear Train	7	66-3-54			670
	Servo Motor	7	66-3-55			860
	Amplifier	2	66-3-56			430
	Amplifier	2	66-3-57			340
	Amplifier	2	66-3-58			335
	Amplifier	2	66-3-59			290
	Milli-amp Meter	19	66-3-60			1,000
	Milli-amp Meter	6	66-3-61			530
	Amplifier	18	66-3-83			5,100
	Logic, D.C.	57	66-3-84			1,270
	Emitter	42	66-3-85			570
	Logic, D.C.	45	66-3-86			1,050
	SEM, Dual	7	66-3-87			140
	Total – Case 3					79,400
4	Circuit Breakers	1	66-3-111-18	Argonne	5/25/67	2,915
5	Pri. & Sec. Na Pumps	5	66-1-1-2	NRTS Strge	12/26/67	624,330
6	Pri. Pump Cases	3	66-1-1	NRTS Strge	7/17/68	120,000
7	Pri. Fill Tanks	5	66-2-1	AEC-Richland	12/16/67	235,695
8	Loading Face Shield Comp.	1	66-1-56	NRTS Strge	8/2/67	15,000
	L. F. S. Snubbers	2	66-1-56	NRTS	8/2/67	500
	Loading Face Heat Exchanger	1	66-1-57	NRTS	8/2/67	1,137
	Total – Case 8					16,637
9	Turbine Feedwater Pump	1	66-1-35	NRTS Strge	9/13/67	16,017
	Motor Driven Feed-water Pump	1	66-1-36	NRTS	9/13/67	2,530
	Total – Case 9					18,547
10	Cold Trap Heater Oven	1	66-1-4	NRTS Strge	1/5/68	8,500
11	Case Cancelled	-	-	-	-	-
12	Intermediate Heat Exchangers	6	66-1-11	NRTS Strge	6/19/68	652,176
13	Steam Generators	3	66-1-13	NRTS Strge	6/5/68	3,083,847
14	Secondary Sodium Pump	1	66-1-2	NRTS Strge	10/9/67	109,956
15	Single Element Fuel Ship. Cask	1	66-3-367	Gen. Atomics	9/11/67	50,473
16	Pulse Generator	1	66-3-45	Sandia Corp.	9/13/67	325

A. E. C. Case No.	Description	Quantity	HNPF Item Number	AEC Recipient	Date Shipped	Acquisition Cost
17	Timer/Counter	1	66-3-39	Ames Lab.	9/8/67	475
	Recorder	1	66-3-74			1,200
	Vacuum Pump	1	66-3-76			135
	Pump, Valve, Motor	1	66-3-138			1,735
	Centrifugal Pump	1	66-3-139			
	Valve	2	66-3-140			} 1,735
	Induction Motor	1	66-3-141			
	Basket Strainer	1	66-3-142			100
	Calculator	1	66-3-936			1,146
	Count Ratemeter	1	66-3-99	Ames Lab.	9/8/67	775
	Total – Case 17					7,301
18	Battery Operated Source	1	66-3-63	K. A. P. L.	9/13/67	275
	Power Supply	1	66-3-89	K. A. P. L.	9/13/67	300
	Total – Case 18					575
19	Power Supply	1	66-3-40	Los Alamos	9/15/67	230
20	Calibrator	1	66-3-51	NRTS	9/13/67	350
21	Secondary Fill Tanks	3	66-2-2	AEC-LMEC	10/24/67	131,703
22	Amplifier	1	66-3-91	Los Alamos	9/15/67	4,950
	Analyzer	1	66-3-95	Los Alamos	9/15/67	4,950
	Total – Case 22					9,900
23	Case Cancelled	-	-	-	-	-
24	Cold Trap Heater Ovens	2	66-1-4	PNL	7/7/67 10/31/67	15,300
	Cold Trap Cooling Blower	1	66-1-5		9/20/67	500
	8-in. Steam Block Valves	3	66-1-16		7/25/68	9,590
	6-in. Feedwater Block Valves	3	66-1-18		1/5/68 1/19/68 2/2/68	} 7,300
	6-in. Feedwater Block Valves	3	66-1-19		2/2/68	5,500
	Loading Face Shield Comp.	1	66-1-55		9/20/67	13,495
	Snubbers	2	66-1-56		9/20/67	500
	Loading Face Heat Exchangers	1	66-1-57		9/20/67	1,045
	Feedwater Steam Panel	1	66-3-13		12/7/67	500
	Annunciator	1	66-3-13-3		12/7/67	1,175
	Flashers	4	66-3-13-5		12/7/67	140
	Voltmeter	7	66-3-13-6		12/7/67	700
	Pressure Control	2	66-3-13-10		12/7/67	40
	Switch	8	66-3-13-14		12/7/67	528
	Switch	3	66-3-13-15		12/7/67	90
	Switch	5	66-3-13-16		12/7/67	120
	Switch	3	66-3-13-17		12/7/67	39
	Recorder/Controller	4	66-3-13-18		12/7/67	2,540
	Recorder	13	66-3-13-19		12/7/67	4,680

A. E. C. Case No.	Description	Quantity	HNPF Item Number	AEC Recipient	Date Shipped	Acquisition Cost
24	Controller	2	66-3-13-20	PNL	12/7/67	850
(Cont'd)	Hand Control Station	14	66-2-12-21			1,410
	Gauge, Pressure	4	66-3-13-22			110
	Gauge	11	66-2-13-23			138
	Gauge	4	66-3-13-24			50
	Receiver	3	66-3-13-25			600
	Controller	6	66-3-13-26			2,830
	Relay	6	66-3-13-27			1,000
	Air Switch	3	66-3-13-28			66
	Flow Computer	3	66-3-13-29			1,220
	Rack	1	66-3-13-30			720
	Rack	1	66-3-13-31			860
	Matrix Control Panel	1	66-3-18			2,250
	Recorder	2	66-3-18-1			37,011
	Recorder	6	66-3-18-2			1,620
	Annunciator	2	66-3-18-3			2,700
	Horn Relay	2	66-3-18-4			100
	Flasher	2	66-3-18-5			70
	EMF Converter	6	66-3-18-6			2,270
	Alarm Unit	6	66-3-18-7		12/7/67	1,000
	Recorder	1	66-3-19		7/25/68	670
	Recorder	8	66-3-19-1		7/25/68	6,500
	Recorder	1	66-3-20		7/25/68	500
	Pyr-A-Larm	2	66-3-21		10/16/68	2,800
	Recorder	1	66-3-23		8/14/68	1,480
	Regulator	1	66-3-26		10/21/68	70
	Oscillograph	1	66-3-29		10/31/67	4,700
	Preamplifier	4	66-3-30		10/31/67	600
	Power Supply	4	66-3-31		10/31/67	2,000
	Oscillograph	1	66-3-32		10/31/67	4,700
	Preamplifier	4	66-3-33		10/31/67	600
	Power Supply	4	66-3-34		10/31/67	2,000
	Oscilloscope	1	66-3-38		10/21/68	1,635
	Plug-In Unit	1	66-3-38-1		10/21/68	190
	Plug-In Unit	1	66-3-38-2		10/21/68	170
	Plug-In Unit	1	66-3-38-3		10/21/68	170
	Oscilloscope	1	66-3-38-4		10/21/68	1,620
	Scopemobile	1	66-3-38-5		10/21/68	180
	Micro-Ammeter	1	66-3-50		10/31/67	630
	Oscilloscope	1	66-3-66		10/21/68	450
	Potentiometer	1	66-3-67		10/21/68	332
	Potentiometer	1	66-3-69		10/21/68	300
	Potentiometer	1	66-3-70		10/21/68	300
	Recorder	1	66-3-72		10/31/67	610

A. E. C. Case No.	Description	Quantity	HNPF Item Number	AEC Recipient	Date Shipped	Acquisition Cost
24 (Cont'd)	Chromatograph	1	66-3-75	PNL	10/31/67	1,500
	Na Level Probe	2	66-3-98		10/31/67	540
	Cold Traps	3	66-3-629		9/20/67	10,000
	Cooling Fan	1	66-3-671		10/31/67	500
	Motor	1	66-3-671-1		10/31/67	700
	Cooling Fan	1	66-3-672		10/31/67	500
	Cooling Fan	1	66-3-673		10/31/67	500
	Freeze Seal Valves	1	66-3-701		7/25/68	850
	Valve	2	66-3-701A		7/25/68	500
	Valve	3	66-3-702		7/25/68	850
	Valve	8	66-3-703		7/25/68	2,300
	Valve	4	66-3-704		7/25/68	944
	Total – Case 24					173,548
25	Cold Trap Panel	1	66-3-693	PNL	9/20/67	500
	Transformer	6	66-3-693-1			108
	Transformer	1	66-3-693-2			70
	Transmitter	1	66-3-693-3			800
	Indicator	2	66-3-693-4			1,600
	Transmitter	2	66-3-693-5			2,000
	Transmitter	1	66-3-693-6			800
	Control Unit	1	66-3-693-7			1,500
	Relay	1	66-3-693-8			24
	Relay	1	66-3-693-9			56
	Timer	1	66-3-693-10			60
	Positioner	1	66-3-693-11			250
	Indicator	1	66-3-693-13			1,400
	Variac	1	66-3-693-14			125
	Relay	2	66-3-693-15			112
	PanAlarm	1	66-3-693-16			35
	Level Alarm	7	66-3-693-17			4,900
	Rectifier	1	66-3-693-18			600
	Oscillator	4	66-3-693-19			3,000
	Indicator	5	66-3-693-20			500
	Indicator	1	66-3-693-21			750
	Control Box	1	66-3-693-22			40
	Total – Case 25					19,230
26	Recorder	1	66-3-71	Argonne Lab.	10/16/67	725
	Megohmeter	1	66-3-80			145
	Recorder	1	66-3-870			1,115
	Pump	1	66-3-875			133
	Total – Case 26					2,118
27	Meter	1	66-3-77	Ames Lab.		475

A. E. C. Case No.	Description	Quantity	HNPF Item Number	AEC Recipient	Date Shipped	Acquisition Cost
28	Dotco Air Wrench	2	66-3-422	Los Alamos	8/20/68	590
29	Counter	1	66-3-92	Lucius Pitkin	10/16/67	895
30	Counter	1	66-3-78	K. A. P. L.	11/8/67	1,044
	Counter	1	66-3-79	K. A. P. L.	11/8/67	644
	Recorder	1	66-3-38	K. A. P. L.	11/8/67	161
	Total – Case 30					1,849
31	Power Supply	1	66-3-93	MIT	11/8/67	149
	Converter	2	66-3-97	MIT	11/8/67	840
	Total – Case 31					989
32	Storage Batteries	120	66-1-62	Bur. of Rec.	10/31/67 4/19/68	7,804
33	Induction Pumps	2	66-1-3	PNL	7/25/68	58,722
	Freeze Traps	6	66-1-24		7/25/68	24,000
	Sod. Filter	2	66-1-27		4/19/68	16,258
	Flowmeter, 14 in.	1	66-1-30		6/19/68	6,400
	Flowmeter, 3 in.	3	66-1-32		6/19/68	4,300
	Flowmeter, 2 in.	1	66-1-33		6/19/68	1,800
	Flowmeter, 1 in.	3	66-1-34		6/19/68	4,800
	Plugging Control	1	66-1-38		1/19/68	6,158
	Heat Exchangers	2	66-1-43		5/7/68 9/6/68	5,318
	Battery Charger	1	66-1-63		1/3/68	4,895
	Panel, Process	1	66-3-3		1/19/68	33,322
	Panel	1	66-3-4		8/14/68 7/25/68	36,208
	Electrometer	1	66- -7		8/14/68	488
	Truck, Ca. C.	1	66-3-333		8/14/68	600
	Panel	1	66-3-594		1/19/68	5,260
	Panel	1	66-3-598		1/5/68	2,552
	Total – Case 33					211,681
34	Cold Traps	2	66-3-629	AEC-LMEC	9/20/67	7,000
35	Hydryer	6	66-3-847-1-5	AEC-LMEC	2/27/68	1,860
36	Assigned to Idaho					
37	Assigned to Idaho					
38	Mask, Air	1	66-3-164	Ames Lab.	10/21/68	274
	SourcesSet	1	66-3-169			175
	Absorber Set	1	66-3-170			100
	Sample Changer	1	66-3-176			1,460
	Meter, Count Rate	1	66-3-177			1,465
	Sample, Changer	1	66-3-179			1,460
	Timer, Printer	1	66-3-180			770
	Meter, Count Rate	1	66-3-195			1,465

A.E.C. Case No.	Description	Quantity	HNPF Item Number	AEC Recipient	Date Shipped	Acquisition Cost
38 (Cont'd)	Mask, Air	1	66-3-209	Ames Lab.	10/21/68	274
	Mask, Air	1	66-3-329			274
	Mask, Sling	1	66-3-330			274
	Mask, Air	1	66-3-361			274
	Mask, Sling	1	66-3-362			274
	Mask, Sling	1	66-3-419			274
	Mask, Sling	1	66-3-428			274
	Mask, Sling	1	66-3-433			274
	Mask, Sling	1	66-3-501			274
	Mask, Air	1	66-3-558			274
	Mask, Sling	1	66-3-559			274
	Mask, Air	1	66-3-630			274
	Mask	1	66-3-679			274
	Gauge	1	66-3-245			50
	Multipointer	1	66-3-246			250
	Dynamometer	1	66-3-341			500
	Dynamometer	11	66-3-809			500
	Total – Case 38					12,031
39	Indicator Unit	1	66-3-52	Los Alamos		235
	Welder	7	66-3-814-6			6,500
	Saw, Electric	1	66-3-931			200
	File Cabinet	2	66-3-952			194
	Total – Case 39					7,129
40	Cabinet	1	66-3-243	Bendix Corp.	10/16/68	125
	Cabinet	1	66-3-244		10/16/68	85
	Oven	1	66-3-247		10/16/68	195
	Total – Case 40					405
41	Transformer	3	66-3-172	NRTS	10/21/68	135
42	Desiccator	1	66-3-264	Puerto Rico N.C.	10/14/68	175
43	Meter	1	66-3-174	General Electric	10/14/68	325
	Scaler	1	66-3-178			1,250
	Monitor	1	66-3-156			2,500
	Monitor	1	66-3-158			2,500
	Set, Source	1	66-3-173			200
	Total – Case 43					6,450
44	Secondary Pump Cases	1	66-1-2	AEC-LMEC	6/12/68	120,000
45	Analytic Balance	1	66-3-257		10/9/68	35
	Centrifuge	1	66-3-248		10/9/68	105
	Total – Case 45					140
46	Panel, Helium Vent & Supply	1	66-3-490		9/5/68	1,000
	Switch	2	66-3-490-1			150
	Indicator	1	66-3-490-2			35
	Controller/Indicator	2	66-3-490-3			900

A. E. C. Case No.	Description	Quantity	HNPF Item Number	AEC Recipient	Date Shipped	Acquisition Cost
46 (Cont'd)	Controller/Indicator	2	66-3-490-4	AEC-LMEC	9/5/68	800
	Control Valve	3	66-3-490-5			825
	Indicator, Flow	1	66-3-490-6			85
	Indicator, Flow	3	66-3-490-7			250
	Indicator, Pressure	3	66-3-490-8			105
	Indicator, Pressure	5	66-3-490-9			175
	Indicator, Flow	4	66-3-490-10			335
	Indicator, Flow	1	66-3-490-11			85
	Indicator, Flow	2	66-3-490-12			170
	Indicator, Flow	1	66-3-490-13			85
	Total – Case 46					5,000
47	Panel, Nitrogen Vent Supply	1	66-3-489			1,340
	Controller	6	66-3-489-1			1,650
	Valve	6	66-3-489-2			1,200
	Set, Air w/Gauge	6	66-3-489-3			420
	Indicator, Flow	2	66-3-489-4			130
	Indicator, Flow	4	66-3-489-5			260
	Total – Case 47					5,000
48	Containment Drain Tank	1	HNPF-2-4		5/10/68	3,000
	All Valves Not Radio-active	63	No Number		8/7/68	100,000
	Secondary Na Serv. Drain Tank	1	HNPF-2-6		5/10/68	3,000
	Total – Case 48					106,000
49	Circuit Breakers	3	66-3-111-18		10/9/68	8,745
50	Extinguisher, Fire	1	66-3-136	K. A. P. L.	10/14/68	83
	Extinguisher, Fire	1	66-3-202			83
	Extinguisher, Fire	1	66-3-221			83
	Extinguisher, Fire	1	66-3-256			83
	Extinguisher, Fire	1	66-3-268			83
	Extinguisher, Fire	1	66-3-270			83
	Extinguisher, Fire	1	66-3-320			83
	Extinguisher, Fire	1	66-3-324			83
	Extinguisher, Cart	1	66-3-331			575
	Extinguisher, Fire	1	66-3-359			83
	Extinguisher, Fire	1	66-3-360			83
	Extinguisher, Cart	1	66-3-363			800
	Extinguisher, Fire	1	66-3-370			83
	Extinguisher, Fire	1	66-3-372			83
	Extinguisher, Fire	1	66-3-393			83
	Extinguisher, Fire	1	66-3-418			83

A.E.C. Case No.	Description	Quantity	HNPF Item Number	AEC Recipient	Date Shipped	Acquisition Cost
50 (Cont'd)	Extinguisher, Fire	1	66-3-429	K.A.P.L.	10/14/68	83
	Extinguisher, Fire	1	66-3-503			83
	Extinguisher, Fire	1	66-3-551			83
	Extinguisher, Cart	1	66-3-554			800
	Extinguisher, Fire	1	66-3-557			83
	Extinguisher, Fire	1	66-3-569			83
	Extinguisher, Fire	1	66-3-570			83
	Extinguisher, Fire	1	66-3-632			83
	Extinguisher, Fire	1	66-3-633			83
	Extinguisher, Cart	1	66-3-634			575
	Extinguisher, Fire	1	66-3-639			83
	Extinguisher, Cart	1	66-3-662			575
	Extinguisher, Fire	1	66-3-666			83
	Extinguisher, Fire	1	66-3-667			83
	Extinguisher, Fire	1	66-3-668			83
	Extinguisher, Fire	1	66-3-721			83
	Extinguisher, Cart	1	66-3-722			575
	Extinguisher, Fire	1	66-3-738			83
	Extinguisher, Cart	1	66-3-739			575
	Extinguisher, Fire	1	66-3-753			83
	Extinguisher, Fire	1	66-3-755			83
	Extinguisher, Cart	1	66-3-756			575
	Extinguisher, Fire	1	66-3-761			83
	Extinguisher, Fire	1	66-3-850			83
	Cabinet, Tray	1	66-3-185			65
	Scaler, Slave	1	66-3-192			640
	Timer, Dual	1	66-3-200			200
	Detection, Victoreen	1	66-3-224			225
	Hot Plate	1	66-3-259			105
	Unit, Air Sampling	1	66-3-345			4,985
	Monitor, Remote Area	1	66-3-404			500
	Speaker, University Model	1	66-3-387			20
	Wall Periscope	1	66-3-407		8/22/68	7,589
	Meter, Roentgen	1	66-3-175		10/14/68	125
	Unit, Air Sampling	1	66-3-343		10/14/68	4,985
	Sampler, Air	1	66-3-612		10/14/68	4,985
	Leak Detector Panel	28	66-3-5 to -9		8/2/68	23,845
	Counter	2	66-3-8		8/22/68	1,600
	Cabinet, Storage	1	66-3-344		10/14/68	150
	Unit Recorders	3	66-3-412		8/22/68	1,011
	Cabinet, Storage	1	66-3-815		10/14/68	150
	Total – Case 50					58,886
51	Plugging Meter	1	66-1-37	PNL	4/19/68	25,000
	Bridge Crane, 25 Ton	1	66-1-52	PNL	4/19/68	41,873

A. E. C. Case No.	Description	Quantity	HNPF Item Number	AEC Recipient	Date Shipped	Acquisition Cost
51 (Cont'd)	Roller Shutter Door	1	66-1-66	PNL	9/6/68	13,007
	Printer	1	66-3-10		9/6/68	1,995
	Radiation Detection Panel	1	66-3-12-1		10/16/68	3,000
	Basic Control Unit	1	66-3-12-1			1,600
	Basic Control Unit	1	66-3-12-3			1,600
	Basic Control Unit	1	66-3-12-4			1,600
	Radiation Level Read Out	2	66-3-12-5			4,340
	Radiation Level Alarm	1	66-3-12-6			1,700
	Fan, Cooling	1	66-3-12-7			152
	Recorder, Strip	2	66-3-12-8			3,400
	Meter, Count Rate	1	66-3-12-9			765
	Meter, Count Rate	1	66-3-12-10			765
	Amplifier	1	66-3-12-11			920
	Power Supply	1	66-3-12-12			400
	Fan, Cooling	1	66-3-12-13			184
	Scaler	1	66-3-12-14			1,500
	Time Switch	1	66-3-12-15			150
	Recorder	1	66-3-12-16			2,500
	Annunciator	1	66-3-12-17			750
	Chamber	3	66-3-12-18			675
	Chamber	3	66-3-12-19			675
	Chamber	1	66-3-12-20			225
	Chamber	1	66-3-12-21			225
	Probe	10	66-3-12-22			2,200
	Chamber	4	66-3-12-23			540
	Detector	1	66-3-12-24			220
	Detector	1	66-3-12-25			250
	Alarm Unit	3	66-3-12-26			1,585
	Converter	3	66-3-12-27			2,265
	Meter	1	66-3-187			2,445
	Detector	1	66-3-190		10/21/68	400
	Timer	1	66-3-188		10/21/68	60
	Cabinets w/s. s top	10	66-3-215 to 215-6		8/14/68	2,590
	Hood Wash Pan	1	66-3-217		8/9/68	15,000
	Hood, s. s	1	66-3-249		10/21/68	2,549
	Cabinet	3	66-3-250		10/21/68	375
	Cabinet	1	66-3-251		10/21/68	150
	Cabinet	1	66-3-252		10/21/68	290
	Cabinet	1	66-3-261		10/21/68	750
	Sampler, Air	6	66-3-277 to 277-5		9/6/68	6,985
	Extinguisher, Fire	1	66-3-319		8/9/68	83

A. E. C. Case No.	Description	Quantity	HNPF Item Number	AEC Recipient	Date Shipped	Acquisition Cost
51 (Cont'd)	Extinguisher, Fire	1	66-3-323	PNL	8/9/68	83
	Extinguisher, Fire	6	66-3-326		8/9/68	586
	Extinguisher, Cart	1	66-3-332		8/9/68	800
	Scale	1	66-3-339		10/21/68	935
	Dynamometer	1	66-3-340		10/21/68	2,250
	Truck, CaCo2	1	66-3-364		8/14/68	800
	Detector, Smoke	1	66-3-366		9/6/68	110
	Transformer	1	66-3-366-1			130
	Notifier	1	66-3-366-2			145
	Detector, Smoke	1	66-3-374			110
	Transformer	1	66-3-374-1			130
	Notifier	1	66-3-374-2			145
	Radiation Window	2	66-3-409			40,000
	Cart, Instrument	1	66-3-415		8/14/68	21,505
	Cabinet	1	66-3-415-1			70
	Recorder	1	66-3-415-2			1,700
	Ratemeter	1	66-3-415-3			650
	Power Supply	1	66-3-415-4			400
	Amplifier	1	66-3-415-5			400
	Fan	1	66-3-415-6			75
	Probe	1	66-3-415-7			250
	Radiation Window	1	66-3-439		9/6/68	20,000
	Detector, Smoke	1	66-3-463		9/6/68	110
	Transformer	1	66-3-463-1		9/6/68	130
	Notifier	1	66-3-463-2		9/6/68	145
	Meter, Gas	1	66-3-485		10/21/68	600
	Gas Sampler	1	66-3-539		9/6/68	300
	Motor	1	66-3-539-1			142
	Powerstat	1	66-3-539-2			150
	Switch	1	66-3-539-3			100
	Detector, Smoke	1	66-3-553			110
	Transformer	1	66-3-553-1			130
	Notifier	1	66-3-553-2			145
	Totals – Case 51					240,074
52	Motor Control Center	48	66-3-582 to 582-45		8/14/68	5,114
	Variac	2	66-3-583		8/9/68	100
	Operator	2	66-3-584		8/9/68	200
	Detector, Leak	1	66-3-600		9/6/68	6,570
	DeWar, Liq. Ni.	1	66-3-617		9/6/68	150
	Emergency Fire Cabinet	1	66-3-670		9/6/68	500
	Emergency Shower	1	66-3-690		8/14/68	485

A.E.C. Case No.	Description	Quantity	HNPF Item Number	AEC Recipient	Date Shipped	Acquisition Cost
52 (Cont'd)	Secondary Sodium Exp. Panel	8	66-3-698 to 698-7	PNL	5/17/68	4,010
	Secondary Sodium Exp. Panel	8	66-3-699 to 699-7		5/17/68	4,010
	Secondary Sodium Exp. Panel	8	66-3-700 to 700-7		5/17/68	4,010
	Metal Truck $CaCo^2$	1	66-3-724		8/14/68	900
	Metal Truck $CaCo^2$	1	66-3-758		8/9/68	900
	Detector, Leak	5	66-2-794 to 794-4		8/14/68	6,570
	Filters, Absolute	86	66-3-835		8/9/68 8/14/68	2,580
	Total – Case 52					36,099
53	Motor Control Center	40	66-3-110	Argonne Lab.	3/26/68	3,200
	Master Slave Manipulators	4	66-1-48		8/22/68	18,032
	Beckman Printer	1	66-3-0		8/22/68	995
	X, Y Recorder	1	66-3-194		10/14/68	900
	Hydra Set	1	66-3-301		10/14/68	1,449
	Camera	1	66-3-149		10/14/68	315
	Scale	1	66-3-260		10/14/68	650
	Compressor	1	66-3-891		8/22/68	2,100
	Total – Case 53					27,641
54	Control Center	13	66-3-125 to 125-12		8/22/68	3,214
	Switch Gear	11	66-3-127 to 127-7		8/22/68	5,945
	Panel Lighting	1	66-3-128		10/21/68	70
	Indicator, Oxygen	1	66-3-166		10/14/68	350
	Transformer	1	66-3-399		10/14/68	33
	Control Center	9	66-3-581 to 581-8		8/22/68	3,895
	Blower	1	66-3-752-1		10/14/68	500
	Motor	1	66-3-752-2		10/14/68	25
	Total – Case 54					14,032
55	Pinhole Camera	1	66-3-405	AEC-LMEC	6/13/68	207
	Pinhole Camera	1	66-3-406	AEC-LMEC	6/13/68	207
	Total – Case 55					414
56	Extinguisher, Cart	1	66-3-669	PNL	8/9/68	792
57	Carrier Air Conditioner	1	66-3-165	Westinghouse	10/14/68	700
	Water Heater	1	66-3-203			400
	Am. Standard Air Conditioner	1	66-3-236			400
	Transformer	1	66-3-316			500
	Total – Case 57					2,000
58	Secondary Pressure Transducers	12	None	AEC-LMEC	5/28/68	30,000

A.E.C. Case No.	Description	Quantity	HNPF Item Number	AEC Recipient	Date Shipped	Acquisition Cost
59	File Cabinet	1	66-3-152	Natl. Accel. Lab.	10/16/68	100
	Storage Cabinet	1	66-3-159			90
	Storage Cabinet	1	66-3-216			85
	Storage Cabinet	1	66-3-229			185
	Storage Cabinet		66-3-238			85
	Storage Cabinet	1	66-3-338			330
	Storage Cabinet	1	66-3-420			70
	Cabinet, Storage	1	66-3-831			100
	File, Card	2	66-3-934			30
	Cabinet, Metal	4	66-3-948			360
	Cabinet, Metal	1	66-3-949			90
	Cabinet, Metal	3	66-3-950			432
	Cabinet, Metal	1	66-3-951			175
	Cabinet, Metal	5	66-3-954			490
	Cabinet, Metal	1	66-3-955			98
	File, Card	1	66-3-956			5
	Cabinet, Metal	1	66-3-969			98
	Holder, Blueprint	1	66-3-294			200
	Intercom	1	66-3-435			50
	Intercom	1	66-3-436			75
	Intercom	2	66-3-237			400
	Total – Case 59					3,548
60	Cabinet, Fire Equipment	1	66-3-365			400
	Cabinet, Fire Equipment	1	66-3-560			400
	Total – Case 60					800
61	14 in. Flowmeters	4	66-1-30	NRTS	7/17/68	12,800
62	Fuel Shipping Boxes	2	66-3-804	AEC-LMEC	6/13/68	2,000
63	Sodium Pipe & Pipetees	8	-	AEC-LMEC	8/6/68	4,000
64	Moderator Cask and Spares	11	66-3-290	Savannah River	8/6/68	50,000
65	Canister Welder	1	-	AEC-LMEC	8/2/68	22,500
66	Coil Cap. Bank E. M. Pump	4	-	AEC-LMEC	9/5/68	40,000
67	60-Ton Bridge Crane	1	66-1-51	AEC-LMEC		93,173
68	Nuclear Instruments	6	None	Argonne Lab.	9/17/68	7,500
69	Miscellaneous Parts					16,163
						6,518,465

APPENDIX VI

DIRECTIVES AND SIGNIFICANT CORRESPONDENCE

The decision to retire the HNPF and the authorizations to proceed with deactivation are documented. Also included, are letters in which the hazards and cost of dismantling are discussed.

COPY

UNITED STATES
ATOMIC ENERGY COMMISSION
Washington, D.C. 20545

Mr. D. W. Hill, General Manager
Consumers Public Power District
Columbus, Nebraska 68601

Dear Mr. Hill:

Upon receipt of your May 24, 1966 notification that the CPPD Board of Directors had decided not to exercise the option to purchase the Hallam Nuclear Power Facility, the Commission authorized the enclosed plan for making the premises safe from a radiation standpoint in accordance with the applicable provisions of the lease between AEC and CPPD.

Pursuant to this plan, the premises would be placed in such a condition that no continuing AEC license would be required from a health and safety standpoint: access to any area of the premises that would remain exposed would not entail an AEC license. Following execution of this plan, periodiceenvironmental monitoring would be conducted to assure the continuing radiation safe condition of the premises. You will note that this plan is consistent in principle with the views expressed in Emerson Jones' letter to K. A. Dunbar of March 11, 1966 as they relate to the attainment of a radiation safe condition that is free of AEC licensing.

Beyond the actions contemplated by the enclosed plan to make the premises safe from a radiation standpoint, the plant will be further dismantled for scrap or salvage or for use of equipment and components elsewhere, to the extent advantageous to the Government.

We appreciate your Board's resolution to cooperate with AEC in the decommissioning of the reactor plant. You may expect to be contacted very shortly by Mr. Dunbar to develop in conjunction with your organization the procedures for implementing the enclosed plan, and to work out any related contractual arrangements with CPPD.

Sincerely yours,

(S) R. E. Hollingsworth
General Manager

Enclosure:
Plan for Decontamination of HNPF Premises

COPY

UNITED STATES
ATOMIC ENERGY COMMISSION
Washington, D.C. 20545

November 3, 1967

Docket No. 115-3

Consumers Public Power District
Columbus, Nebraska 68601

Attention: R. S. Kamber
Power Supply Manager

THRU: K. A. Dunbar, Manager
Chicago Operations Office

Gentlemen:

An Order is enclosed authorizing you to dismantle and decontaminate the Hallam Nuclear Power Facility in accordance with your application dated November 14, 1966, with supplements thereto submitted June 2 and June 7, 1967. Copies of the Notice of Issuance which is being filed with the Office of the Federal Register and the staff safety evaluation in this matter are also enclosed.

After the completion of dismantling and decontamination of the facility, the submission of a report describing the condition of the remaining structures, and an inspection by representatives of the Commission, consideration will be given to whether a further order should be issued terminating Operating Authorization No. DPRA-1.

Sincerely yours,

Peter A. Morris, Director
Division of Reactor Licensing

1. Order
2. Notice of Issuance
3. Safety Evaluation

COPY

UNITED STATES
ATOMIC ENERGY COMMISSION
Washington, D.C. 20545

UNITED STATES ATOMIC ENERGY COMMISSION

CONSUMERS PUBLIC POWER DISTRICT

DOCKET NO. 115-3

ORDER AUTHORIZING DISMANTLING OF FACILITY

By application dated November 14, 1966, the Consumers Public Power District (CPPD) requested authorization to dismantle and decontaminate the Hallam Nuclear Power Facility (HNPF), located in Hallam, Nebraska, in accordance with the HNPF Retirement Plan, Revision 4, enclosed with the application. A revised HNPF Retirement Plan was submitted by letter dated June 2, 1967, and Supplement 5 to the Final Summary Safeguards Report was submitted by letter dated June 7, 1967.

Operation of the HNPF has been discontinued and it is being deactivated by removing all the fuel and the sodium coolant used in operation of the reactor from the site.

We have reviewed the application in accordance with the provisions of the Commission's regulations and have found that the dismantling of the facility and its decontamination will be accomplished in accordance with the regulations in this chapter and will not be inimical to the common defense and security or to the health and safety of the public.

Accordingly, it is hereby ordered that CPPD may proceed with dismantling of the HNPF covered by Operating Authorization No. DPRA-1, as amended, in accordance with its application dated November 14, 1966, and supplements dated June 2 and June 7, 1967.

After the completion of dismantling and decontamination of the facility, the submission of a report describing the condition of the remaining structures, and an inspection by representatives of the Commission, consideration will be given to whether a further order should be issued terminating Operating Authorization No. DPRA-1.

FOR THE ATOMIC ENERGY COMMISSION

Peter A. Morris, Director
Division of Reactor Licensing

Date: November 3, 1967

COPY

UNITED STATES ATOMIC ENERGY COMMISSION

DOCKET NO. 115-3

CONSUMERS PUBLIC POWER DISTRICT

NOTICE OF ISSUANCE OF ORDER

AUTHORIZING DISMANTLING OF FACILITY

The Atomic Energy Commission has issued an Order, set forth below, authorizing the Consumers Public Power District, Columbus, Nebraska, to dismantle the Hallam Nuclear Power Facility, located in Hallam, Nebraska, and covered by AEC Operating Authorization No. DPRA-1, as amended.

Copies of the application dated November 14, 1966, with supplements thereto dated June 2 and June 7, 1967, and the related staff safety evaluation are available for public inspection at the Commission's Public Document Room, 1717 H Street, N.W., Washington, D.C. A copy of the staff safety evaluation may be obtained at the Public Document Room or upon request addressed to the Atomic Energy Commission, Washington, D.C., 20545, Attention: Director, Division of Reactor Licensing.

FOR THE ATOMIC ENERGY COMMISSION

Peter A. Morris, Director
Division of Reactor Licensing

Dated at Bethesda, Maryland
this 3rd day of November, 1967

SAFETY EVALUATION BY THE DIVISION OF REACTOR LICENSING

DOCKET NO. 115-3

CONSUMERS PUBLIC POWER DISTRICT

HALLAM NUCLEAR POWER FACILITY RETIREMENT

The Hallam Nuclear Power Facility (HNPF) is a sodium-cooled, graphite moderated power Demonstration Reactor which was operated by Consumers Public Power District (CPPD) at its Sheldon Station near Hallam, Nebraska. The nuclear-steam generating portion of the facility is owned by the Atomic Energy Commission and the conventional facilities at the Sheldon Station, including the turbine generator, are owned by CPPD. As a Commission owned facility, HNPF is not subject to licensing, but construction and operation were authorized in accordance with the Commission's regulation, "Procedures for Review of Certain Nuclear Reactors Exempted from Licensing Requirements", 10 CFR Part 115. The HNPF was shut down on September 27, 1964, for replacement of moderator elements which had developed cladding leaks. After extensive review, the Commission determined that the HNPF had fulfilled its basic objectives in the power demonstration program and announced in June 1966 that it would decommission the facility.

Under its contractual relationship with CPPD, the Commission is obligated, in decommissioning the facility, to make the premises safe from a radiation standpoint. To this end, the Commission requested CPPD and Atomics International, the designer of the facility, to draw up a plan for retirement of the facility which would meet this objective and which would put the facility in such a condition that no continuing AEC license would be required.

This plan was submitted to the Division of Reactor Licensing for regulatory review and approval. The review made by the regulatory staff is equivalent to the review which would be made pursuant to Section 50.82, 10 CFR 50, if HNPF were a licensed facility.

By letter dated November 14, 1966, CPPD submitted "Hallam Nuclear Power Facility Retirement Plan", NAA-SR-MEMO-12340, to the Division of Reactor Licensing and requested authorization to proceed with dismantling of the facility. Additional information to complete the application was required, and in response to questions from the regulatory staff, CPPD submitted by letter dated June 7, 1967, "Final Summary Safeguards Report for the Hallam Nuclear Power Facility, Supplement 5", NAA-SR-5700, which assesses the basic safety considerations related to the retirement plan. This document, together with facility design information already on record and a revised facility retirement plan (NAA-SR-MEMO-12340, Revised 5/20/67), constitute the information considered in the regulatory review summarized herein.

DISCUSSION

The retirement plan proposed entails the following basic elements:

(1) All reactor fuel will be shipped off-site.
(2) All bulk sodium will be shipped off-site and residual sodium will either be removed or reacted to remove any chemical hazard.
(3) Contaminated or irradiated equipment will be packaged and shipped off-site or placed in the reactor vessel, Fuel Storage Pit No. 3 or one of the moderator storage cells. These areas will be sealed to provide final disposal of the radioactive material.
(4) Accessible portions of the plant will be decontaminated and plant volumes used for disposal of contaminated or irradiated materials will be sealed to prevent personnel access and to assure weather-tightness. The plant will be left in such condition that radiation levels at all accessible portions would be essentially at background and in such condition that access to contaminated and irradiated material left at the site would require deliberate planned action and preliminary approval by the appropriate licensing authority.

Safety aspects related to the dismantling process and the long-term storage of radioactive materials at the site are discussed in detailed below.

Dismantling Activities

To dismantle the plant, all systems which were necessary to operate the facility will have been inactivated and disassembled to the extent that the reactor could not be utilized without major reconstruction. Sodium pumps will be removed and primary system piping will be cut and sealed at the seal membranes to the reactor cavity. Components such as primary system piping and valves will be removed if they have salvage or scrap value; otherwise they will be decontaminated and left in place in areas of the plant which will be inaccessible. All auxiliary systems necessary to operation of the plant will be permanently deactivated or removed.

CPPD will be responsible for all activities involved in dismantling and retiring the plant, and will have available personnel who are intimately familiar with the plant. Overall direction of the retirement activities will be done by the CPPD health-physics supervisor. Engineering services will be supplied by CPPD and Atomics International, as necessary. The basic procedures and methods for the dismantling and retirement of the HNPF have been outlined in the plan. Our review indicates that these basic procedures are adequate in terms of carrying out each of the required steps in a safe manner. For each task, CPPD will develop and utilize detailed procedures to augment the basic procedures set forth in the plan. On the basis of our review, we have concluded that the CPPD staff is qualified to undertake the retirement activities, and that the procedures governing the activities will be adequately developed and sufficiently comprehensive to assure negligible risk to personnel and the general public.

COPY

Irradiated fuel at the site has decayed to insignificant heat generation levels, and procedures previously developed for handling and shipment of irradiated fuel will be utilized for its removal in sealed shipping casks. Since decay heat generation levels are too low to cause fuel damage from overheating, the only possible release that could occur due to fuel handling would be the result of mechanical damage during handling. The amount of activity that could be released would be negligible, particularly when compared to the amounts previously considered and found acceptable for the plant when it was approved for operation.

The only activities that could potentially involve release of significant amounts of activity are associated with (1) removal of the primary sodium fill tanks, which contain a small amount of residual sodium activity, and (2) removal and reaction of residual sodium in the reactor vessel. There are approximately 9662 ft^3 of primary sodium stored in the five primary sodium fill tanks in a below-grade vault of the HNPF reactor building. The frozen sodium in the fill tanks will be removed from the vault, placed on railroad cars, and shipped to Hanford, Washington, for storage and eventual reuse. The procedures for filling and sealing the tanks are satisfactory and, we conclude, provide adequate assurance against release of radioactive sodium.

The removal and transfer of the tanks filled with frozen sodium presents a possibility of damage to the tanks due to improper handling, equipment failure, or operator error. If reasonable care is exercised, the possibility of tank damage can be minimized and we feel that even in the unlikely event of a massive rupture of one of the tanks there will be no significant release of radioactivity. The activity of the sodium is due to Na^{22} and when measured in February 1967 was 0.025 uc/cc. Since all water will be excluded from the handling area, the possibility of a sodium-water reaction is eliminated. Further, there is ample evidence to establish that the atmospheric oxidation of frozen sodium which might be exposed in the event of tank rupture will not cause a sodium fire. We have concluded, therefore, that removal and transfer of the tanks as proposed will not lead to conditions that can cause release of significant amounts of sodium activity.

Since the reactor vessel will be used for disposal of contaminated and irradiated components, residual sodium must be removed or chemically reacted to remove any potential for a future chemical reaction that might impair the integrity of the enclosure. After draining and pumping operations are completed, it is estimated that about 100 lbs of sodium will remain in the bottom of the vessel and on internal structures. This sodium will be reacted with steam to assure that the potential for a chemical reaction is removed. While the amount of activity in this residual sodium is sufficiently low that release of all of it would not create a radiological safety problem off-site, a gross structural failure during these operations could conceivably release radioactive materials in the form of contaminants other than sodium.

The hazards of the proposed method of reacting residual sodium have been recognized by the applicant and analyzed in detail. Our review indicates that the procedures and controls proposed for accomplishing the process should eliminate any concern over the possibility of an uncontrolled chemical reaction. The process involves injection of a steam-nitrogen mixture into the preheated vessel at a controlled rate and venting to the existing wash cell vent system which is designed to handle hydrogen-steam inert gas mixtures. Oxygen content will be monitored and limited to a maximum of 1% to preclude the possibility of a hydrogen explosion. Hydrogen content will be monitored to measure the progress of the reaction. Pressure within the vessel during steaming will be limited to 4 psig. All process variables will be monitored and alarms will be provided to assure that operators are alerted to any off-normal condition that would require interruption of the steaming process. Our review of the plans for conducting the operation has convinced us that the possibility of a hydrogen explosion will be eliminated by the strict controls over the oxygen content in the vessel. Conservative calculations by AI, in which we concur, indicate that the maximum pressure potential for an uncontrolled steam-sodium-NaOH reaction would be 33 psig. This eliminates any possibility that the ultimate pressure capability of the loading face shield of 58 psig, which was established previously, could be exceeded. Thus it can be concluded that total loss of control over the steaming operation would not breach the closure. In summary, it is our opinion that the proposed method for reacting residual sodium can be carried out safely.

Facility Status After Retirement

The HNPF retirement plan calculates that after retirement activities have been completed, approximately 310,000 curies of activity will be entombed in three designated locations within the below-grade reinforced concrete structure of the plant. Owing to radioactive decay, the inventory would decrease to about 3000 curies at 40 years and about 1300 curies at 100 years. The disposal areas will be sealed to prevent access and to prevent any possible outleakage of the activity. Radiation levels at all accessible portions of the plant will be within the values for unrestricted areas set forth in 10 CFR 20 of the Commission's regulations. With these objectives satisfactorily fulfilled, members of the general public could be granted unlimited access to the plant vicinity with essentially no possibility of receiving radiation exposures that would endanger health and safety.

We have concluded that sufficient information has been submitted to establish that the retirement plan can be carried out with assurance that the plant will be left in a radiation safe conditions. The reactor vessel, Fuel Storage Pit No. 3 and three of the moderator storage pits have been designed as locations for storage of radioactive materials that cannot be readily removed from the plant. Each of these proposed storage locations is ideally located within the massive reinforced concrete structure constituting the main floor and shielding structure of the plant. From our review of the details of this structure prior to operation of the plant, we are satisfied that there is no credible condition that could cause structural failures which would impair the integrity of the storage enclosures. Shielding for accessible locations at main floor level was designed for full power operation of the reactor, under which the radiation levels within the storage locations were anticipated to be several orders of magnitude greater than the radiation levels of the materials remaining after disposal. Consequently, radiation levels at the main floor will be insignificant.

Several steps will be taken to assure that the below-grade areas of the plant are adequately sealed and put into a condition that would make them inaccessible. The loading face shield over the reactor vessel will be covered by a steel plate which will be seal-welded to the shield, and all penetrations into the reactor cavity will be capped and sealed. The fuel storage pit and the moderator storage pits will be sealed similarly at floor level, before dismantling is complete.

There are a number of below-grade volumes, such as pipe tunnels, intermediate heat exchanger cells and equipment areas, which will not be used for storage of radioactive materials. Nevertheless, each of these volumes will be permanently closed to prevent access, and exposed surfaces of the structure will be waterproofed and overlain by an additional barrier, such as concrete, to further inhibit access.

The information supplied in Supplement 5 to the Final Summary Safeguards Reports indicates that the reactor building might be razed. Whether or not the building is razed is a question that is completely independent of any question related to putting the retired plant in a radiation safe condition, and so long as the decontamination limits are met, we are not concerned with the final disposition of the reactor building.

In the retired condition, the only significant questions that can be raised concerning safety are related to the possibility that at some time in the distant future the storage volumes might be deliberately breached or that stored activity could reach ground water.

To preclude any possibility that the storage volumes would be entered, notification and appropriate descriptions will be placed in the land title records of the Lancaster County Court House before the facility license is terminated to notify any future owners of the property that entry into these volumes would require prior licensing action or other specific governmental approval.

COPY

The potential for transport of significant amounts of activity into ground water at the site is exceedingly remote for several reasons. As mentioned earlier, the storage volumes have been found to be structurally sound. In addition, each of these volumes is lined with waterproof steel plate which would be expected to retain its integrity for an extremely long period even if ground water could enter the concrete structure and provide a corrosive medium. Since essentially all of the activity to be stored is in the form of activation products dispersed in metallic components, release of any activity would require that these components be corroded in turn. About 96% of this activity will be in the reactor vessel which is regarded as the principal storage location. In order to estimate the maximum potential ground water contamination that could occur in the vicinity of the plant, the applicant has estimated on the basis of corrosion data that in the unlikely event that the pipe tunnel adjacent to the reactor vessel were continuously flooded with ground water, approximately 100 years would be required to corrode piping closures and allow water to enter the reactor vessel. It is further estimated that another 160 years would be required to completely corrode the stainless steel moderator sheaths in which the bulk of the remaining activity (about 1290 curies of Ni^{63}) would be located. With release governed by corrosion, diffusion of this activity into ground water would necessarily take place over a period of many years. In this case, the liklihood that concentrations in the ground water at any location adjacent to the retired facility would exceed drinking water tolerance is remote. Nevertheless, in order to establish an upper limit to the possible concentration, it was assumed arbitrarily that the entire inventory of Ni^{63} that would exist after 100 years was transported in water-soluble form into the soil adjacent to the floor structure. Based on conservative estimates of dilution and soil decontamination factors, it was calculated that the concentration in a shallow well immediately adjacent to the facility would not exceed 2.3×10^{-13} curies/ml. This is about a factor of 100 below permissible concentrations set forth in 10 CFR 20 for water in unrestricted areas. Transport of this activity to locations further away would take place at a very low rate because of the flatness of the hydraulic gradient at the site. However, concentrations at more removed locations would be even less than calculated above because of increased dilution and ion exchange with the soil. In view of these analyses, in which we concur, we have concluded that any possible out-leakage of activity from the storage enclosures will not lead to any danger to the health and safety of the public.

After decontamination procedures have been completed, the results of a complete radiation survey will be made available for review together with a final calculated inventory of the radioactive materials to be left at the side. The Division of Compliance will also inspect to determine that the dismantling and decontamination program has been carried out in accordance with the application. In addition, the details relative to the final design of closure, sealants, and protective barriers which are to be provided are expected to be submitted. If the retirement program is completed as presently anticipated, it is not expected that any routine post-retirement surveillance of the retired plant will be necessary.

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CONCLUSION

On the basis of our evaluation of the retirement plan proposed for the Hallam Nuclear Power Facility, we have concluded that the dismantling of the facility and the disposal of the component parts as proposed will be performed in accordance with the Commission's regulations and will not be inimical to the common defense and security or to the health and safety of the public. Upon completion of dismantling and disposal in accordance with the application and with the Commission order, the authorization will be terminated.

Donald J. Skovholt
Assistant Director for Reactor Operations
Division of Reactor Licensing

Date: November 3, 1967

COPY

DEPARTMENT OF TRANSPORTATION
HAZARDOUS MATERIALS REGULATIONS BOARD
Washington, D.C. 20590

SP 5469

Mr. Kenneth A. Dunbar
Manager
United States Atomic Energy
Commission
Chicago Operations Office
9800 South Cass Avenue
Argonne, Illincis 60439

Dear Mr. Dunbar:

As you requested in your letter of December 8, attached is Special Permit No. 5469 authorizing the shipment of certain fissile radioactive material in the HNPF six-element fuel shipping container.

Sincerely,

W. K. Byrd
Chairman

att.

cc:
Bureau of Explosives, AAR
Federal Railroad Administration
Atomic Energy Control Board, Canada
Consumers Public Power District, Lincoln, Nebraska
USAEC, Mr. R. A. Kaye

COPY

DEPARTMENT OF TRANSPORTATION
HAZARDOUS MATERIALS REGULATIONS BOARD
Washington, D.C. 20590

SPECIAL PERMIT NO. 5469

This special permit is issued pursuant to the authority of 49 CFR 173.22(a)(1), Department of Transportation (DOT) Hazardous Materials Regulations, as amended.

1. The U. S. ATOMIC ENERGY COMMISSION and CONSUMERS PUBLIC POWER DISTRICT, Lincoln, Nebraska, are hereby authorized to ship fissile radioactive material, n. o. s., further described as irradiated uranium-235, under the provisions of 49 CFR, 173.393(g)(2) and 173.393(m) of the DOT Regulations, in accordance with the provisions of the U. S. Atomic Energy Commission (USAEC), Chicago Operations Office, approval number AEC-CH-2-67 dated December 8, 1967, and as further provided for herein.

2. The authorized packaging shall consist of a cylindrical lead-filled weldment about 19 feet 10 inches long and about 37 inches in diameter, weighing about 40 tons. The cask is identified as the HNPF Six-Element Irradiated Fuel Cask, and is described in Atomics International's Report NAA-SR-12547, and on Atomics International's drawing numbers N69081051 and N69081052.

3. The closure device must have affixed to it a tamperproof lock wire and seal adequate to prevent inadvertent opening of the container, and of a type that must be broken if the package is opened.

4. The authorized contents of each package shall consist of not more than six HNPF irradiated fuel elements, each containing not more than 7.70 kilograms of uranium-235, in the form of U-C or U-Mo enriched to not more than 4.9% U-235; and containing approximately 175,000 curies of fission products.

5. Shipments are authorized as Fissile Class III, with not more than one package per vehicle.

6. Prior to each shipment authorized by this permit, the consignee shall be notified of the dates of shipment and expected arrival.

COPY

7. The outside of each package shall be plainly and durably marked "DOT SP 5469" and "FISSILE RADIOACTIVE MATERIAL," in connection with and in addition to the other markings and labels prescribed by the DOT Regulations. Each shipping paper issued in connection with shipments made under this permit must bear the notation "DOT SPECIAL PERMIT NO. 5469" in connection with the commodity description thereon.

8. The permit does not relieve the shipper from compliance with any requirement of the DOT Regulations, except as specifically provided for herein.

9. Shipments are authorized only by rail freight.

10. The shipper must furnish a record of experience to the Office of Hazardous Materials if any extension or amendment to the permit is requested. This report must include the approximate number of shipments involved in any loss of contents.

11. This permit shall expire January 15, 1970.

Issued at Washington, D.C., this 3rd day of January 1968.

H. R. Longhurst
For the Administrator
Federal Railroad Administration

Address all inquiries to: Chairman, Hazardous Materilas Regulations Board, U.S. Department of Transportation, Washington, D.C. 20590. Attention: Special Permits.

COPY

UNITED STATES
ATOMIC ENERGY COMMISSION
CHICAGO OPERATIONS OFFICE
9800 South Cass Avenue
Argonne, Illinois 60439

CERTIFICATION OF APPROVAL FOR FISSILE-LARGE QUANTITY SHIPPING CONTAINERS CHICAGO OPERATIONS OFFICE, USAEC

I. Authorization Assignment To

U.S. Atomic Energy Commission
Consumers Public Power District

II. Identification of Shipping Container

HNPF Six-Element Irradiated Fuel Shipping Cask. Shipping Cask Assembly Drawing N69081051. Two Casks (identical). 6-1/2" x 10" Stainless Steel Plate With the Legend: Property of USAEC, Cask No., AI Manual No. NAA-SR-12547, Radioactive Material. DOT No. Tare Wt. 71,000 lbs.
Designed By: Atomics International
Manufactured By: Allied Engineering and Production Corporation
Year Built 1967

III. A. General Information Concerning Container

The container is a cylindrical lead-filled steel weldment weighing approximately 40 tons. Each cask, which will contain six doubly canistered HNPF fuel elements, provides an effective cavity of 18-1/4 inches diameter by 197-1/2 inches long with 8-1/2 inches lead equivalent shielding. The cask body is constructed of two concentric steel shells welded to forged steel flanges at each end. The inner shell is 1 inch thick Type 304 stainless steel. A stainless steel weld overlay is made on the steel flanges from the stainless steel inner shell to the outer diameter of the flanges. The outer shell is 1-1/2 inch thick ASTM A516 steel. The overall length of the cask is 19 feet and 9-3/4 inches, and the outside diameter is 37-1/4 inches. For cask handling, two 6 inch diameter trunnions are set into the cask top flange, and two 7 inch diameter trunnions are set into the lower flange. Bolted steel heads at both ends of the casks are removable, and hoist and grapple mechanisms are provided for loading the baskets of fuel elements into the casks. Three 1 inch diameter "Snaptite" connectors are provided on each cask for purging, venting, and draining. A 1/4 inch toggle valve is provided for hose attachment for sampling cask gaseous or liquid contents at the upper end of the cask body.

Steel baskets, designed to hold six fuel elements, support the fuel in the casks during shipment.

Cradles built of structural steel are provided for supporting each cask for shipment and storage. Sunshades are provided for use with the casks in the summer months to meet SRP's requirements that the sodium bond between the fuel and the cladding be maintained in a solid state.

There are two types of HNPF fuel, and both are clad with 304 stainless steel. The majority of the elements are U-10 wt. % Mo fuel material 3.6% enriched. The active fuel length is 159 inches. Each element contains 220 Kg of U-Mo fuel. The other elements are UC fuel material.

Eight of the UC elements contain 3.7 wt. % ^{235}U and two contain 4.9% ^{235}U. Both enrichments contain 165 Kg UC fuel. The U-10 Mo fuel contains 7.13 Kg of ^{235}U, the 3.7% enriched UC fuel contains 5.81 Kg of ^{235}U, and the 4.9% enriched UC fuel contains 7.70 Kg of ^{235}U per element. The total maximum curie inventory per cask load is 1.75×10^5 curies. Since the cask contains no coolant other than air, the only potential escape of radioactivity is from ^{85}Kr. If 5% of the fission gases born escape from the fuel, through the cladding, through the double canisters around the fuel elements and then out of the cask, only 32 curies would escape.

The maximum thermal output per element is 45 watts.

Each cask will have the exclusive use of a railroad car.

These casks are designed to transport spent fuel as Fissile Class III.

IV. Specific Limitations and Restrictions

A. The containers shall not be used for shipments of fissile materials in solution.

V. Additional Information and/or Limitations

A. Information contained in NAA-SR-12547 shows that the container meets all the evaluation criteria established by AEC Chapter 0529 and 10 CFR 71.

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VI. Certification of Approval

Pursuant to AEC Manual Chapter 0529, this container is approved subject to the limitations and restrictions described above. This certification does not relieve the shipper of this responsibility to obtain a DOT permit and to comply with the requirements of other Federal Regulations as appropriate.

Approved: Dated 12/8/67
AEC-CH-2-67

Certification Official
Kenneth A. Dunbar, Manager
Chicago Operations Office
U. S. Atomic Energy Commission

COPY

CONGRESS OF THE UNITED STATES
JOINT COMMITTEE ON ATOMIC ENERGY

June 15, 1966

Honorable Glenn T. Seaborg
Chairman
U.S. Atomic Energy Commission
Washington, D.C.

Dear Dr. Seaborg:

The National Coal Policy Conference, Inc. has directed the attached letter, dated June 10, 1966, to Chairman Holifield concerning the decommissioning of the Hallam reactor. According to the NCPC

> "The development in the Hallam reactor case highlights a serious problem associated with the construction and operation of nuclear power plants that has received scant, if any, public attention. Now the Nation must begin to face up to just how we are going to handle the problem of securing or disposing of the many large plants now being built, as well as the large numbers forecast during the next decade and beyond, once their useful life is over."

The NCPC has raised nine specific questions in its letter. We would appreciate your comments with respect to these questions and on any other matters which you consider pertinent to this problem.

Thank you for your cooperation.

Sincerely yours,

John T. Conway
Executive Director

Attachment:
cc NCPC ltr of
6/10/66

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NCPC

NATIONAL COAL POLICY CONFERENCE, INC.
1000 Sixteenth Street, N. W.
Washington, D.C. 20036

June 10, 1966

Honorable Chet Holifield, Chairman
Joint Committee on Atomic Energy
The Capitol
Washington, D.C. 20510

Dear Chairman Holifield:

According to reports which I have read in trade and professional journals, the decision of the Consumers Public Power District not to buy the reactor portion of the 82,000 kilowatt developmental atomic power reactor at Hallam, Nebraska, poses some grave questions for the AEC as to how the radioactive portions of the nuclear reactor will be protected or disposed of so as not to represent a hazard to the public. Even though the generating capacity of the Hallam reactor is small and the components are reported to have a low radioactive level because it has operated very little, nevertheless, the present situation presents a real problem and its solution may be very expensive.

The development in the Hallam reactor case highlights a serious problem associated with the construction and operation of nuclear power plants that has received scant, if any public attention. Now the Nation must begin to face up to just how we are going to handle the problem of securing or disposing of the many large plants now being built, as well as the large numbers forecast during the next decade and beyond, once their useful life is over.

As far as I can determine, there is nothing in an initial licensing procedure which requires an applicant to describe just how he proposes to carry out the provisions in the Rules and Regulations of the AEC which apply to disposal of radioactive material, decontamination of the site and other procedures so as to provide reasonable assurance that the dismantling of the facility and disposal of the component parts "will not be inimical to the common defense and security or to the health and safety of the public."

The eventual magnitude of the problem involved in securing or disposing of obsolete or inoperable nuclear plants is such that it demands immediate attention by the Joint Committee. It is credible that any one

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plant may not even last longer than Hallam. In any case, nuclear power plants will not last indefinitely. When their useful economic life is over, some action must be taken, most probably they must be dismantled and disposed of, just as obsolete, uneconomic coal-fired plants are razed. We hope the Joint Committee can schedule hearings during the current session at which the many and varied aspects of this problem can be developed and thoroughly discussed. We must start planning now on how this potentially grave problem will be handled to protect the public safety and welfare and moreover, spell out precisely what the responsibility of the operators of the nuclear plants will be.

We believe that the following questions, among others, should be thoroughly explored at such hearings.

1. What is the estimated cost of dismantling, decontaminating and disposing of a nuclear plant in the 500 to 1,000 MWe range of the type currently being built or planned in increasing numbers?

2. Does the property insurance of the utilities contain sufficient funds for phasing out a nuclear plant, in case the plant had to be shut down prematurely because of a malfunction?

3. Will such plants have to be protectively covered or buried on site or is it possible, with known engineering and transportation techniques, to dismantle the large radioactive components and transport them safely to a permanent storage area?

4. Is the cost of dismantling and storing a nuclear plant, with the many problems uniquely associated with it, included as a part of the capital and operating cost upon which the cost of power from such a plant is now projected?

5. If the dismantling costs are not now included in this over all cost, by roughly how much (mills/kwh) would it affect the projected power costs if it should be?

6. Where will radioactive components, such as the pressure vessel, be buried?

7. Is it anticipated that there will be disposal areas operated by privately owned businesses by then or will government-owned facilities have to be used?

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Mr. Holifield

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8. If the government provides these disposal areas, will the nuclear power plant operators be required to pay a fee which will fully reimburse the government for its expense involved in maintaining in perpetuity such waste disposal areas?

9. Will all of them have to be maintained in perpetuity long after the generation that built and used them are gone?

I realize, of course, that these questions merely scratch the surface. Numerous others no doubt will be raised, many of them of a much more technical nature than the questions I have suggested. The AEC is now considering the Hallam situation and therefore should have some pertinent and useful information to contribute. Surely the utilities building or planning nuclear plants must have considered these ultimate questions. The important thing, it appears to me, is for the Joint Committee to recognize that the Nation does face a tremendous problem. The public has a right to know now, in the early stages, the procedures and responsibilities for the orderly and safe disposal of nuclear power plants as they become obsolete or inoperable in the years ahead.

Your early attention to the request for hearings will be deeply appreciated.

Sincerely,

/s/ Joseph E. Moody

President

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United States
ATOMIC ENERGY COMMISSION
Washington, D.C.

July 14, 1966

Dear Mr. Conway:

Your letter of June 15, 1966 requests our comments on a June 10 letter to Chairman Holifield from the National Coal Policy Conference, Inc. (NCPC) regarding the decommissioning of power reactors.

As a general comment, it appears to us that the NCPC letter reflects a misunderstanding of the extent to which the decommissioning of a large power reactor involves the new or unknown. Consequently, we do not agree with their assessment of the potential risk to the health and safety of the public and the financial impact of such decommissioning on the owning utility. Our reasons are indicated below and amplified in the enclosure.

Decommissioning of a reactor does not involve any operations which from a safety standpoint are significantly different from those that may be required for refueling and maintenance of the reactor. Under the Commission's regulations, procedures for dismantling of the plant would be subject to specific Commission approval and would be required to meet the Commission's standards for protection of the worker and the general public. Actually, after removal of the fuel from the reactor the precautions required for safety are far less than those required for an operating reactor.

Within prescribed safety requirements, the cost to a utility of decommissioning its nuclear plant is dependent upon the extent of decommissioning that the utility decides is most to its economic advantage. The alternatives available include varying degrees of "mothballing" the plant, dismantling and removing the facility in whole or in part, burial in place, and decontamination of the premises. Regardless of the mode of decommissioning selected, the cost will not be substantial in relation to the resources provided by the utility for the construction and operation of the plant. As a basis for measuring the potential cost and decommissioning, it might be noted that at the rate of .01 mill per KWH the utility would accumulate $5 million during the 30-year operation of a 1000 MWe plant.

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The enclosure provides further amplification on the above points as well as comments relating to the nine questions contained in the NCPC letter.

We will be happy to provide any additional information on this matter that you desire.

Cordially,

/s/ Glenn T. Seaborg

Chairman

Mr. John T. Conway, Executive Director
Joint Committee on Atomic Energy
Congress of the United States

Enclosure:
Supplemental Comments

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Supplemental Comments
Related to NCPC Letter and Questions

Under the Commission's regulations set forth in 50.82, 10 CFR 50, no licensee may surrender a license or dismantle or dispose of component parts of a nuclear facility until it has applied for and received authority from the Commission to do so. The application must demonstrate that the proposed course of action will not be inimical to the common defense and security or to the health and safety of the public. The application is evaluated by AEC safety experts to assure that adequate safety measures will be taken in the course of the decommissioning, and also that necessary safeguards will be provided as regards any radiation that may thereafter remain at the site.

The decommissioning of a nuclear reactor plant does not introduce significant new or unknown safety problems. The Commission's experience to date in decommissioning its reactors demonstrates that they can be done safely. A utility, in decommissioning its power reactor, would be motivated by economic considerations not associated with the decommissioning - i.e., recovery of the value of the fuel material - to reprocess the radioactive fuel, the cost of which is a part of the fuel cycle costs charged to operations. Removal of the fuel for reprocessing substantially eliminates the major source of radioactivity at the plant site. Such removal of the fuel is essentially a repetition of the fuel removal routine carried out by the utility cyclically while the plant is in operation, which is required to be done by procedures determined by the Commission to be safe from a radiation standpoint. Similarly, the removal by the utility of components and equipment for decommissioning purposes involves in large part a repetition of measures previously applied by the utility in the course of routine maintenance or replacement during operation of the plant, pursuant to procedures approved by the Commission as providing necessary health and safety protection. In the extreme case where the utility decides to remove the irradiated reactor vessel and containment, this would be done by procedures required to be approved by the Commission. The extent to which removal of the reactor vessel required special engineering techniques would depend upon such factors as the induced radioactivity present, whether the vessel can be removed by means similar to those used in its transport to and installation in the plant, etc. In any event, such engineering techniques do not present unknown problems even for dismemberment of the vessel and they can be derived from the substantial experience already gained in the use of remotely-controlled devices for the cutting and handling of radioactive materials.

The utility has available to it a number of alternatives as to the extent to which it chooses to decommission the reactor plant. These alternatives vary from complete removal of the reactor plant from the site to the other extreme of leaving the plant substantially intact and providing the public safety protection stipulated by the Commission license for the radioactive materials remaining. It is to be expected that the alternative selected by the utility will be that offering the greatest economic advantage, considering such factors as the cost of providing maintenance and safety protection for a plant left in place, as compared to the cost of dismantling all or part of the plant; the extent to which it is profitable to remove individual items of equipment in order to use them elsewhere or recover their salvage value; and the advantage from re-using the structures or the exact location of the plant. In any case, adequate safeguards to the public must be established and submitted to the Commission for approval. If the decommissioning procedures that the utility submits are considered inadequate, the Commission has ample authority to require the utility to provide the additional safeguards necessary.

The financing of the decommissioning of a power plant may come under the cognizance of the utility regulatory authorities, but not AEC. We understand that the anticipated cost to decommission or retire a conventional power plant might be provided for in the power rates through depreciation by reducing the salvage value used in calculating the plant depreciation. This procedure would be equally applicable to nuclear plants, since the utility is subject to the same regulatory requirements. We understand that the regulatory commissions do not anticipate and include in utility rates any provision for the cost of premature abandonment or sudden obsolescence of the plant. If such an event occurs, the utility and the regulatory commissions are faced with the question of whether such extraordinary costs are to be borne by the customers, by the stockholders, or by a combination of the two.

The property insurance carried on insurable risks is a matter for determination by the utility. The Commission's regulations impose no requirements for property insurance. The property insurance on a conventional power plant does not pay, in whole or part, for decommissioning in case it has to be shut down prematurely, unless as a result of an insured risk. The same is true of property insurance on nuclear plants.

While the Commission regulates the public health and safety protection required with respect to any radioactivity that remains buried or stored at the site after decommissioning, it does not dictate the extent to which the plant must be decommissioned and the radioactivity removed. Consequently, the utility may seek authority from the Commission to bury or store radioactive parts of the plant or to transfer them to an authorized disposal site. It is the Commission's policy that radioactive material generated at privately-owned facilities, unless authorized to be buried in place, be buried at privately-operated sites which have been established for the permanent disposal of such material. Four such commercial burial sites are licensed at present by the Commission to operate on federal - or state-owned land under conditions that require safety surveillance in perpetuity. As is true for any transport of radioactive material, the transport of radioactive parts of a power plant to these burial sites is governed by regulations of AEC, ICC, and such other regulatory authorities as may have jurisdiction.

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CONGRESS OF THE UNITED STATES
JOINT COMMITTEE ON ATOMIC ENERGY

July 21, 1966

Mr. Joseph E. Moody
President
National Coal Policy Conference, Inc.
1000 Sixteenth Street, N.W.
Washington, D.C.

Dear Mr. Moody:

I am sorry I haven't been able to communicate with you sooner on the questions you raised concerning the disposal of nuclear power plants. I asked Mr. John Conway to get in touch with the Atomic Energy Commission to give us more information on the points you raised. He did this and the Commission responded by letter dated July 14, 1966, a copy of which is enclosed with out letter asking for the information. When you brought the matter up during your testimony yesterday, I hadn't had a chance to familiarize myself with information we had received from the Atomic Energy Commission and therefore could not discuss it with you.

The Commission in commenting upon this matter expresses the view that the decommissioning of a reactor does not involve any operations which from a safety standpoint are significantly different from those encountered in normal refueling and maintenance of the reactor. If you have any additional comments on this, please send them to me.

In your letter you asked that early hearings be held on the subject of nuclear power plant disposal. At the moment, the Committee does not have any plans to hold hearings specifically on this subject. I would expect that this subject could be adequately covered during other Committee hearings, such as the AEC authorization and "202" hearings.

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Your letter also stated that the public has a right to know about the problems of disposal of nuclear power plant. I concur in your belief. I would expect that, as in the past the public hearings of the Committee will provide the vehicle for such dissemination of information. I also plan to include this exchange of correspondence in the public record of the Price-Anderson amendment hearings at the point in your testimony yesterday when you referred to your letter to me.

Sincerely yours,

Chet Holifield
Chairman

Enclosure:
AEC Comments 7/14/66
Cy ltr to Seaborg from Conway

U. S. ATOMIC ENERGY COMMISSION
CHICAGO OPERATIONS OFFICE
ARGONNE, ILLINOIS

FINAL INSPECTION AND ACCEPTANCE

FOR

Project Demolition of Buildings, Hallam Nuclear Power Facility

Contractor Harrison Iron & Metal Co., Inc.

Contract No. AT(11-1)-1814

On July 24, 1969, a complete inspection was made of the above contract work. The work was found by the inspection party to be complete and in accordance with the contract drawings, specifications, and any modifications thereto.

Accordingly, the demolition work is accepted by the U. S. Atomic Energy Commission on July 24, 1969.

Inspected by:

U. S. Atomic Energy Commission

By Louis C. Lazar

Title Proj. Engineer

Demolition Contractor

By

Title

Consumers Public Power District

By Landon T. Baldwin

Title Plant Supt.

Atomics International

By Bruce T. Ureda

Title Project Manager

Accepted:

By D. Casey
Director, Construction Division
Chicago Operations Office

APPENDIX VII
REACTOR TECHNICAL DESCRIPTION

The design parameters of the HNPF, the specifications for the configuration, and identification of materials are listed.

TABLE 1

GENERAL SPECIFICATIONS

Power (Mwt)	256
Fuel	U - 10 wt % Mo
Moderator	Graphite
Primary coolant	Sodium
Flow rate (lb/hr)	8.4×10^6
Inlet temperature (°F)	610
Outlet temperature (°F)	945
Secondary coolant	Sodium
Flow rate (lb/hr)	8.4×10^6
IHX inlet temperature (°F)	565
IHX outlet temperature (°F)	900
Active core height (ft)	13.5
Active core diameter (ft)	12.9
Moderator and reflector height (ft)	17.0
Moderator and reflector diameter (ft)	16.8

TABLE 2

FUEL

Number of fuel elements	137
Uranium inventory (kg)	27,600
U^{235} enrichment (wt %)	3.6
U^{235} inventory (kg)	994
Maximum fuel burnup (Mwd/MTU)	15,000
Radial peak to average power	1.4
Axial peak to average power (max)	1.65
(avg)	1.5
Average specific power (kw/kg U^{235})	256
Average fission density in fuel (fiss/cm^3-sec)	4.4×10^{12}
Process tube material	Zircaloy 2

TABLE 3

PERIPHERAL FUEL ELEMENTS

Fuel rods per element	18
Fuel slug diameter (in.)	0.590
Sodium bond annulus (in.)	0.025
Stainless steel cladding thickness (in.)	0.010
Fuel rod OD (in.)	0.660
Average fuel rod power to power of hottest rod (nominal)	0.88
(maximum)	0.86
Flow adjustment permitted by variable orifice	4.1

TABLE 4

CENTRAL FUEL ELEMENT CHARACTERISTICS

Design power (Mwt)	2.46
Design maximum fuel temperature (°F)	1250
Nominal maximum fuel temperature (°F)	1184
Maximum surface heat flux (Btu/hr-ft^2)	3.8×10^5
Maximum rod power (kw/ft)	19.4
Coolant flow area (in.)	7.35
Core pressure drop (psi)	9.6
Sodium average velocity (ft/sec)	8.6
Sodium flow rate (lb/sec)	22.5

TABLE 5
MODERATOR AND REFLECTOR ELEMENTS

Moderator	Graphite*
Moderator cladding	304 SS
Distance across flats (in.)	16
Length (ft)	17
Graphite density (gm/cm^3)	1.68
Stainless steel thickness (in.)	0.016
Sodium gap between elements, avg (in.)	0.160

*Hexagonal Bars with edges scalloped to provide process channels

TABLE 6
CONTROL RODS

Number of driven shim-safety-regulating rods	19
Number of stationary rods, temporary	6
Thimble material	Zircaloy-2
Control rod atmosphere (@ 15 psig)	Helium
Control rod travel (ft)	13.0
Poison column OD (in.)	3.4
Poison column length (ft)	12.5
Poison material	45% Gd_2O_3 - 45% Sm_2O_3
Poison cladding material	Hastelloy X
Maximum poison column temperature (°F)	1800
Reactivity worth of driven rods (% Δ k/k)	12
Maximum reactivity rate (%/sec)	0.03

TABLE 7

STEAM CONDITIONS

	@ Design	@ High Power
Throttle steam pressure (psig)	800	850
Throttle steam temperature (°F)	825	865
Throttle steam flow (lb/hr)	710,000	839,000
Net electrical output (Mwe)	75	~ 87
Gross electrical output (Mwe)	81.8	95.0

APPENDIX VIII
CONTAMINATION LIMITS

Use was made of well established irradiation limits for personnel exposure, and of contamination and radiation control during the retirement of the HNPF These are listed together with an inventory of the stored radioactive materials and radiation levels in the structure at the time of closure.

UNRESTRICTED USE RADIOLOGICAL CONTAMINATION LIMITS FOR THE HNPF

ISOTOPE*	TABLE 1		TABLE 2	
	TOTAL§	REMOVABLE†§	TOTAL§††	REMOVABLE†§
U Nat, U^{235}, U^{238} Th Nat, Th^{232}, and associated decay products	10,000 dpm $\alpha/100\ cm^2$	1,000 dpm $\alpha/100\ cm^2$	Average 5,000 dpm $\alpha/100\ cm^2$ Maximum 25,000 dpm $\alpha/100\ cm^2$	1,000 dpm $\alpha/100$ cm
Other isotopes which decay by alpha emission or by spontaneous fission	1,000 dpm $\alpha/100\ cm^2$	100 dpm $\alpha/100\ cm^2$	Average 500 dpm $\alpha/100\ cm^2$ Maximum 2,500 dpm $\alpha/100\ cm^2$	100 dpm $\alpha/100$ cm
Beta-gamma emitters (Isotopes with decay modes other than alpha emission or spontaneous fission)	0.4 mrad/hr at 1 cm**	1,000 dpm $\beta\gamma/100\ cm^2$	Average 0.2 mrad/hr at 1 cm** Maximum 1.0 mrad/hr at 1 cm**	1,000 dpm $\beta\gamma/100$ cm

*Where surface contamination by both alpha and beta-gamma emitting isotopes exists, the limits established for alpha and beta-gamma emitting isotopes shall apply independently.

†The amount of removable radioactive material per 100 cm^2 of surface area shall be determined by wiping that area with dry filter or soft absorbent paper and with the application of moderate pressure, and assessing the amount of radioactive material on the wipe with an appropriate instrument of known efficiency. In determining removable contamination on objects of lesser surface area, the pertinent levels shall be reduced proportionally, and the entire surface shall be wiped.

§As used in this table, dpm (disintegrations per minute) means the rate of emission by radioactive material as determined by correcting the counts per minute observed by an appropriate detector and count rate meter, for background, efficiency, and geometric factors associated with the instrumentation.

**Measured through not more than 7 milligrams per square centimeter of total absorber.

††Measurements of total contaminant shall not be averaged over more than 10 square meters. For objects of lesser surface area, the average shall be derived for each such object.

NOTE: Either Table 1 or Table 2 may be used. For example, if all beta-gamma readings were less than 0.4 mrad/hr at 1 cm, Table 1 could be used; but if the maximum reading were 0.8 mrad/hr, material could be released under Table 2, provided the average was less than 0.2 mrad/hr.

NOTE: Source reference: AEC Memorandum, M. B. Biles to B. J. Snyder, dated April 7, 1967
Subject: Comments on Hallam Scrap Disposal Criteria

RADIOLOGICAL STATUS OF HNPF AS OF FINAL CLOSURE OF ISOLATION STRUCTURE

Item No.	Reactor Building	Specific Location	Radiation		Contamination (dpm/100 cm^2)
			Maximum	General Area	
1.	Reactor Vessel and Contents	In Vessel	Not available	Not available	Not available
2.	Fuel Storage Vault No. 1	In Vault – Ext. to Storage Thimbles	1.0 mr/hr	1.0 mr/hr	<30
3.	Fuel Storage Vault No. 2	In Vault – Ext. to Storage Thimbles	20 mr/hr	20 mr/hr	<30
4.	Fuel Storage Vault No. 3	In Vault Under Access Opening		2.0 R/hr[(1)]	Not available
5.	Moderator Storage Cells	Elevation 1440 ft 6 in.	Background	Background	<30
6.	Moderator Storage Cell No. 1	In Cell		Background[(2)]	Not available
7.	Moderator Storage Cell No. 2	In Cell	Not available	Not available	1.0×10^4
8.	Moderator Storage Cell No. 3	In Cell	900 mr/hr	500 mr/hr	Not available
9.	Moderator Storage Cell No. 4	In Cell	18 R/hr[(3)]	Not available	Not available
10.	Moderator Storage Cell No. 5	In Cell		Background[(2)]	Not available
11.	Moderator Storage Cell No. 6	In Cell	3.0 mr/hr	Not available	3.3×10^3
12.	Moderator Storage Cell No. 7	In Cell	65 mr/hr[(4)]	Not available	1.0×10^5
13.	Moderator Storage Cell No. 8	In Cell	36 R/hr[(5)]	Not available	Not available
14.	Moderator Storage Cell No. 9	In Cell		Background[(2)]	Not available
15.	Moderator Storage Cell No. 10	In Cell	4.5 mr/hr	Not available	2.8×10^3
16.	Moderator Storage Cell No. 11	In Cell	9.0 mr/hr	1.0 mr/hr	5.3×10^4
17.	Moderator Storage Cell No. 12	In Cell	20 R/hr[(6)]	Not available	Not available
18.	R/A Vent Fan Room	General Area	Background	Background	<30
19.	Maintenance Cell Operating Area	All three levels	Background	Background	<30
20.	Steam Generator Rooms	All three rooms	Background	Background	<30
21.	Secondary Sodium Area	Fill tank and Service area	Background	Background	<30
22.	Maintenance Cell Holdup Tank Vault	Maintenance Cell and Liquid Waste Transfer Tanks	42 mr/hr[(7)]	0.5 mr/hr	<30
23.	Fuel Handling Machine Grapple Storage Pit	In Pit No. 19	80 mr/hr[(8)]	1.0 mr/hr	1.44×10^4
24.	Laundry Tank Vault	Maximum – No. 2 Laundry Tank	10 mr/hr	0.04 mr/hr	<30
25.	Primary Sodium Service Vault	Interior of Vault	Background	Background	<30
26.	Building Fan and Filter Room	General Area	Background	Background	<30
27.	Pipeway Behind Helium and Nitrogen Board	General Area	Background	Background	<30
28.	IHX Vault No. 1	Maximum – at contact with Line No. 231	0.75 mr/hr	0.04 mr/hr	<30
29.	IHX Vault No. 2	Maximum – at contact with Line No. 235	0.70 mr/hr	0.04 mr/hr	<30
30.	IHX Vault No. 3	Maximum – at contact with Line No. 236	1.0 mr/hr	0.2 mr/hr	<30[(9)]
31.	Primary Pipe Vault	Maximum – at contact with Line No. 239	0.5 mr/hr	0.1 mr/hr	<30
32.	Wash Cell Drain Valve Area	Maximum – at contact with Line No. 1403	21.0 mr/hr	1.0 mr/hr	<30
33.	Primary Sodium Fill Tank Vault	Maximum – R/A Liquid Waste Tank Sections	6.0 mr/hr[(10)]	Background	<30
34.	Pipeway behind Wash Cell Board	Maximum – Fuel Cells Vent Line	0.8 mr/hr	Background	<30
35.	Nitrogen Cooling Unit No. 6 Room and Operating Area	General Area	Background	Background	<30
36.	Sample Preparation Room	Elevation 1440 ft 6 in.	Background	Background	<30
37.	Moderator Pump Cells	In Cells	0.17 mr/hr	Background	<30
38.	Maintenance Cell	Interior of Cell	250 mr/hr[(11)]	2.0 mr/hr	2.68×10^5
39.	Auxiliary Bay Area	Elevation 1440 ft 6 in.	Background	Background	<30
40.	Reactor Bay Area	Elevation 1440 ft 6 in.	Background	Background	<30
41.	Personnel Area	Elevation 1440 ft 6 in.	Background	Background	<30
42.	Reactor Building Ventilation Fan Room	Elevation 1440 ft 6 in.	Background	Background	<30

(1) Fuel storage pit No. 3 was surveyed remotely by lowering the instrument into the vault through the man access hole. This reading was obtained 6 ft above the vault floor.

(2) Empty moderator storage cells were not entered for survey purposes.

(3) Radiation reading of moderator element SN-142 was taken when elements were transferred during moderator element replacement program. Readings were taken at approximately 5 ft from elements.

(4) Maximum radiation level is from two piston rings, the contamination maximum is from a tool used in the maintenance cell. The largest amount of activity is from solidified residue which was in the normal level and intermediate level radioactive liquid waste storage tanks. The maximum reading of this residue was 1.56 μc/ml.

(5) Radiation reading of moderator element SN-58 was taken during the moderator replacement program. Reading was taken at approximately 5 ft from the element.

(6) Radiation reading of the moderator element SN-45 was taken during the moderator replacement program. This reading was taken at a distance of approximately 5 ft.

(7) Maximum reading was on the lower portion of the liquid waste transfer tank.

(8) The reactor source was placed in a lead container prior to disposal in this pit. Readings of this source are as follows:

Bare source – 7.5 R/hr
Through container at top – 50 mr/hr
Through container at side – 80 mr/hr
Contamination level of bare source prior to containment in shielded cask – 7.42×10^4 dpm/100 cm^2.

General contamination levels of wall areas in Pit No. 19 were less than 100 dpm/100 cm^2.

Maximum contamination level on components placed in cell = 1.44×10^4 dpm/100 cm^2.

(9) IHX Vault No. 3 contains the following primary sodium pipe sections in addition to the heat transfer pipe.

(a) Twelve pieces of 14-in.-diameter pipe, total length of 120 ft. Maximum radiation equal to 0.7 mr/hr at contact. Maximum internal contamination equal to 2.0×10^3 dpm/100 cm^2. Maximum external contamination equal to <30 dpm/100 cm^2.

(b) Eleven pieces of 6-in.-diameter pipe, total length equal to 66 ft. Maximum radiation equal to 0.3 mr/hr at contact. Maximum internal contamination equal to 2.1×10^3 dpm/100 cm^2.

(10) General area reading are background except for the section of the normal level and intermediate level R/A liquid waste storage tanks which are as follows:

Normal Level Tank:	6 mr/hr maximum	
	2 mr/hr general area	
	5.32×10^4 dpm/100 cm^2	
Intermediate Level Tank:	2 mr/hr maximum	
	1 mr/hr general area	
	2.34×10^4 dpm/100 cm^2	

(11) Maximum radiation reading in the maintenance cell was measured 1 in. from a 1-gallon can containing piston rings which were removed from irradiated process tubes. Contamination levels of the interior walls and false floor of the cell varied from 1.2×10^3 dpm/100 cm^2 to 2.68×10^5 dpm/100 cm^2.

Reference:

L. J. Cooper, et al., "Hallam Nuclear Power Facility Monthly Retirement Report No. 27," October 1968, Sheldon Station, Nebraska