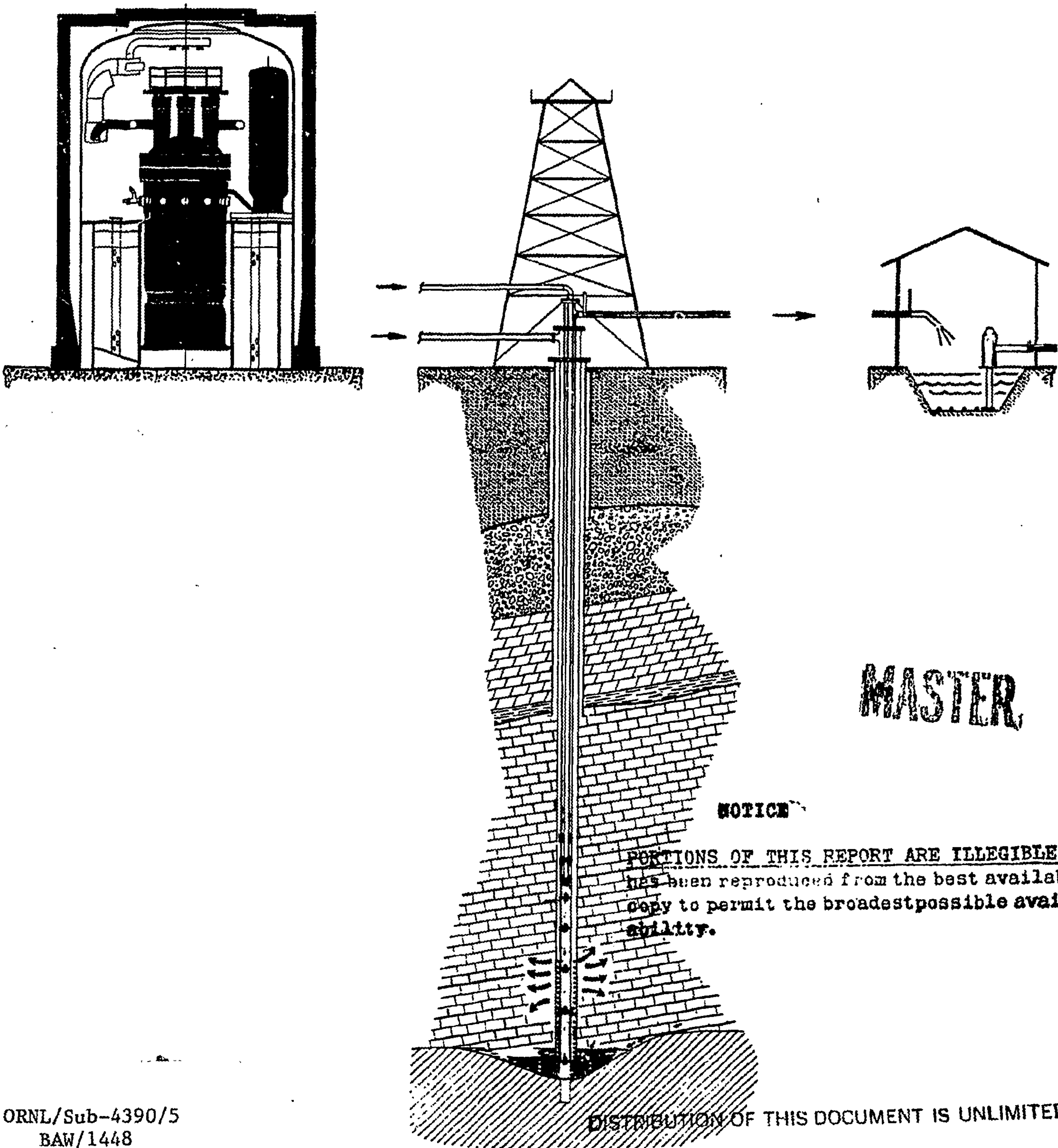
DUVAL CORPORATION APPLICATION STUDY

NUCLEAR PROCESS ENERGY FROM PE-CNSG



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NUCLEAR PROCESS ENERGY FROM PE-CNSG

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FOREWORD

This report describes the work funded by the Department of Energy (DOE), formerly the U. S. Energy Research and Development Administration, and managed for them by the Oak Ridge National Laboratory (ORNL). The technical and economic studies were performed to examine the possible installation of a small, integral pressurized water reactor as an industrial energy source in the Duval Corporation's Frasch Process sulfur mining operation located in Culberson County, Texas. Since this is the first industrial application study attempted for this type of reactor, it has been a learning process on the nuclear plant side as well as the industrial side, particularly in the area of economic analysis. The importance of considering inflationary effects, the significance of plant financing form, and the annualized, after-tax cash flow and incremental rate-of-return methods of comparison in determining energy costs have all been recognized during the course of the study. We recognize that the use of reboilers to provide additional contamination barriers interfacing in nuclear-industrial complexes, while easing regulatory considerations, often imposes an economic penalty. In this study reboilers and existing heat exchangers are incorporated into the interface loop system. If future studies are successful in indicating that a multi-steam-generator nuclear plant (with automatic isolation to ensure negligible radioactivity leakage from the secondary loop) is a promising alternative, it may be worthwhile to reoptimize the tertiary loop design to improve nuclear plant economics.

ACKNOWLEDGMENT

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NOMENCLATURE

APSR	Axial power shaping rods
ASME B&PV	American Society of Mechanical Engineers Boiler & Pressure Vessel Code
ATCF	After-tax cash flow
BOL	Beginning of life
BOP	Balance-of-plant
B&W	Babcock & Wilcox
BTCF	Before-tax cash flow
BWCC	B&W Construction Company
CC	Containment cooling
CCW	Component cooling water
CGC	Combustible gas control
CNSG	Consolidated nuclear steam generator
CR	Control rod
CRA	Control rod assembly
CRDM	Control rod drive mechanism
DH	Decay heat
DNBR	Departure from nucleate boiling ratio
ECCS	Emergency core cooling system
EDH	Emergency decay heat
EFPD	Effective full power day(s)
ERDA	Energy Research and Development Agency
ESFAS	Engineered safety features actuation system
EWST	Emergency water storage tanks
FA	Fuel assembly
FCR	Fixed charge rate
FOAK	First-of-a-kind
FR	Fuel rod
GWD	Gaseous waste disposal
GVW	Gross vehicle weight

HEPA	High-efficiency particulate absorber
ICS	Integrated control system
IDC	Interest during construction
IMS	Incore monitoring system
ITC	Investment tax credit
LBPR	Lumped burnable poison rods
LBPRA	Lumped burnable poison rod assembly
L/D	Length/diameter
LOCA	Loss-of-coolant accident
LPI	Low-pressure injection
LPZ	Low population zone
LWA	Limited work authorization
LWD	Liquid waste disposal
MARR	Minimum attractive rate-of-return
MCNSG	Maritime CNSG
MHA	Maximum hypothetical accident
MU/HPI	Makeup/high-pressure injection
MU&P	Makeup and purification
NDTT	Nil ductility transition temperature
NI	Nuclear instrumentation
NNI	Non-nuclear instrumentation
NOAK	Nth-of-a-kind
NSS	Nuclear steam system
O&M	Operating and maintenance
ORA	Orifice rod assembly
ORNL	Oak Ridge National Laboratory
OTSG	Once through steam generator
PE-CNSG	Process energy CNSG
PWR	Pressurized water reactor
RC	Reactor coolant
RCP	Reactor coolant pump
RCS	Reactor coolant system
RO	Reverse osmosis
ROI	Return on investment
RPA	Reactor protection assembly
RPE	Reactor plant equipment

RPS	Reactor protection system
RPSW	Reactor plant service water
RTD	Resistance temperature detector
RV	Reactor ventilation
RWD	Radwaste disposal
SA	Sampling system
SFAS	Safety features actuation system
SPC	Suppression pool cooling
SRCI	Safety-related control instrumentation
SWD	Solid waste disposal
TG	Turbine generator
TIG	Tungsten inert gas
UE&C	United Engineers and Constructors
USNRC	United States Nuclear Regulatory Commission
VMFT	Vessel model flow test

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1. EXECUTIVE SUMMARY

This study is the first of three studies of the technical interfaces and economic effects of using a small, integral-design pressurized water nuclear reactor to provide process steam in actual industry applications. The work discussed was sponsored by the Department of Energy (formerly the Energy Research and Development Administration) through the Oak Ridge National Laboratory (ORNL) and was undertaken by The Babcock & Wilcox Company (B&W) and United Engineers and Constructors, Inc. (UE&C) with the cooperation of the Duval Corporation (a subsidiary of Pennzoil Co.) as an extension and update of previous work conducted to determine the near-term feasibility of small reactors as replacement industrial energy sources. The major objective was to determine the technical and economic feasibility of the PE-CNSG as a replacement energy source at the Duval Corporation Frasch process sulfur mine.

In the previous work[1] land-based and floating-platform-mounted nuclear plant concepts were studied for electric power and process steam productions. The plant concepts were based on the consolidated nuclear steam generator (CNSG) reactor design developed by B&W for Maritime applications. The results of the earlier study indicated that the most promising nuclear plant concept was the land-based process steam plant. However, because the original study was based on the hypothetical Middletown site[2] and on assumed industrial process conditions, actual technical and economic feasibility was determined only on a hypothetical basis. The major considerations in the present study were to confirm the extension of the CNSG power level from 314 to 365 MWt and to identify site acceptability, process safety and operating restrictions, and site- and application-specific economics for an existing industrial location. This study addresses the considerations noted. Contract arrangements included the support services of UE&C to refine balance of plant (BOP) design and cost estimates. The Duval Corporation agreed to supply data (including process, siting, and economics data) for the study. UE&C participation was provided under Union

Carbide Subcontract 4484 with ERDA; Duval Corporation participation was provided without charge.

This section is an overview of the work accomplished and duplicates in part information contained in the more detailed sections that follow.

1.1. Technical Findings

1.1.1. General

No technical impediments to siting the PE-CNSG at the Duval Corporation's site or to using the PE-CNSG in their sulfur mining operation were identified in the study. Technical study findings are noted below.

1. An extension of the PE-CNSG rated power level capability from 314 to 365 MWt (the original maritime reactor power level) proposed in reference 1 as a result of design modification was re-examined and confirmed. At the 365-MWt power level, the PE-CNSG provides adequate capacity to produce the required 1.7×10^6 long tons of sulfur per year.

2. There are no safety or operational features of the Frasch mining procedure that will be affected adversely by using the PE-CNSG as an energy source. The industrial process had no adverse effects on the safety of the nuclear system as planned for this site.

3. Although no detailed site field examination was made, the site and site data were examined for nuclear plant acceptability. Surface sinkholes, mining subsidence, and possible subsurface subsidence from soluble rock were noted. Some surface faults exist in the general site region — the nearest one within about 1000 feet of the plant site. No evidence was found indicating that the fault zone was an active one. None of these faults appear to be of a nature that would prevent PE-CNSG siting near the mine.

4. A satisfactory reboiler loop with minimized water loss was designed to produce water for mine injection at the desired temperature and pressure.

5. Based on the NRC's published site selection guidelines (10 CFR 100), plant boundary and low population zone distances were set, presenting no problem to the industry.

1.2. Economics

Economic comparisons of alternative energy sources that operate over extended time periods are often ambiguous because economic inflation can produce major changes in important cost components. Inflation will have different effects on the various costs that comprise the price of industrial energy. In the absence of changes in real costs, staff wages, replacement parts, and fuel will tend to increase at prevailing inflation rates; however, the fixed charges for capital obtained with long-term financing will remain relatively unchanged, so that escalation of these costs does not occur. Thus, a given rate of inflation will tend to raise the cost of energy from energy sources with a large fuel cost (such as oil-fired boilers) relative to the cost of energy from other sources with large capital costs (such as nuclear reactors). The effect of inflation on the cost comparison between the CNSG and alternative fossil-fueled systems was considered by assuming fuel and operating/maintenance costs subject to a constant representative rate of inflation (6% per year).

The cost comparison between the nuclear plant and coal for the same service was carried out on an annualized, after-tax, rate-of-return basis over the 22-year plant life considered. For consistency with Duval's evaluation model for such comparisons, 100% of the capital cost financing is assumed to come from corporate equity; this financing assumption tends to increase the cost of energy from capital intensive sources (such as a nuclear plant) relative to the corresponding cost of less capital intensive systems (such as an oil-fired boiler).

The after-tax rate of return on the incremental capital investment required for the nuclear plant in excess of the cost of the coal-fired unit ranged from 12-15% for coal priced from $30 to $40 (1976 dollars) pre ton. The price of $40/ton represents an upper limit on presently perceived coal costs, and thus under present conditions a rate of return (on the nuclear plant incremental capital investment) in excess of 15% seems unlikely.

An 11-15% incremental rate of return is not considered by Duval to be sufficiently high for an investment risk that the corporation would associate with nuclear plant purchase and installation. Conversion to oil firing, even with $14/bbl (1976 dollars) oil prices, may be a comparatively attractive option for Duval since capital investment requirements would be low. Rates of return

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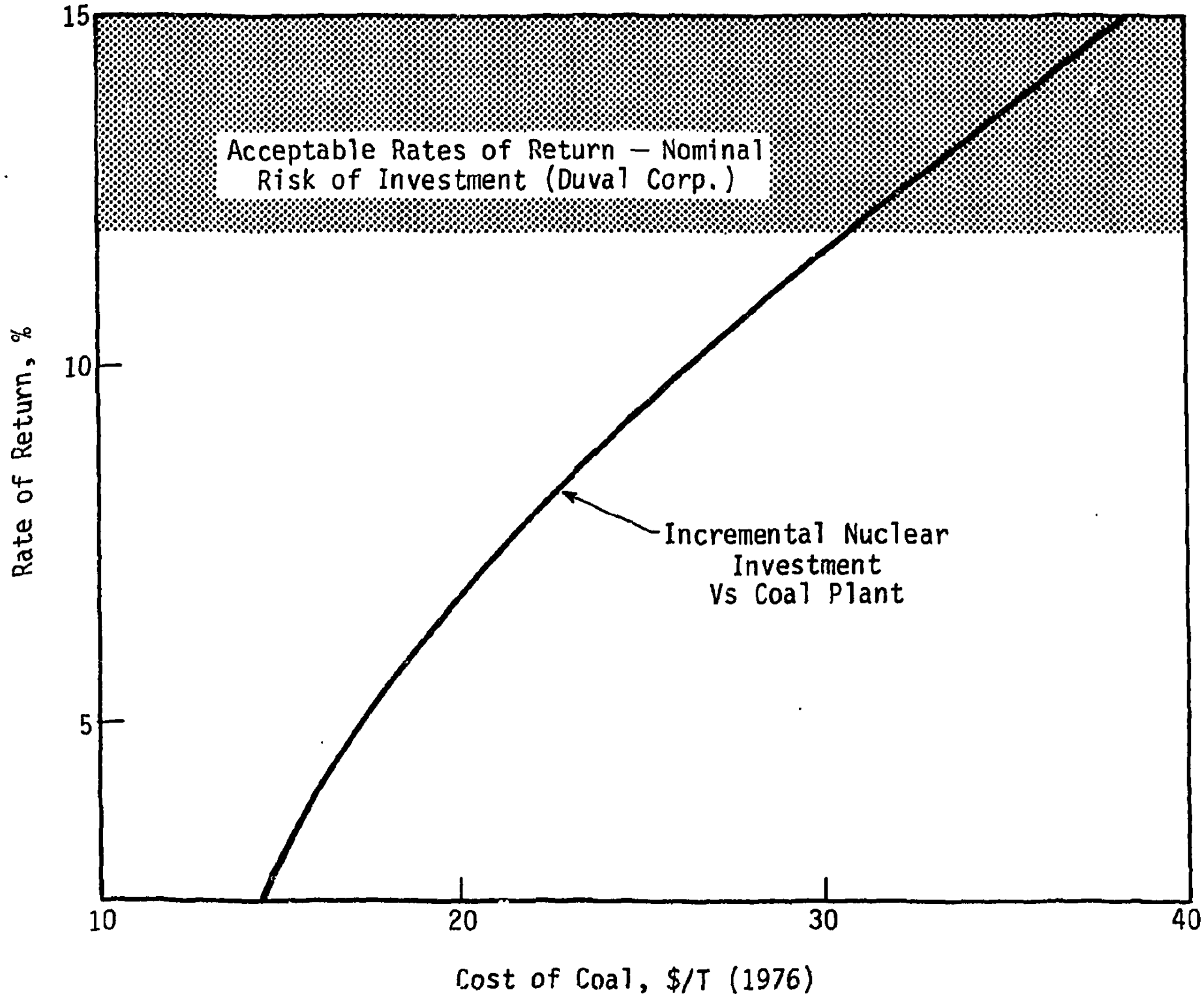
on the incremental nuclear plant investment for this site and application as a function of coal price are shown in Figure 1.2-1.

Figure 1.2-1. Rate of Return on Incremental Nuclear Investment (6%/Year Inflation, 1982-2004 Plant Operation)



1.3. PE-CNSG Plant Description

The PE-CNSG reactor is a small, integral design pressurized water reactor developed by B&W from the CNSG designed for commercial nuclear ship propulsion. The Maritime Administration of the U. S. Deoartment of Commerse has funded design and research and development engineering associated with the maritime reactor since 1969, affording a design and cost analysis basis; without this basis detailed estimates for the present study would not have been possible. In the PE-CNSG (Figure 1.3-1) the core and steam generators are located inside the reactor vessel; the only external portion of the primary system is an electrically heated pressurizer and interconnecting piping. The reactor coolant system is made of the reactor vessel, a set of twelve modular, once-through steam generators, four vertically mounted, wetted-motor reactor coolant pumps, and the pressurizer and interconnecting piping. The steam generators are positioned inside the reactor vessel in an annulus above and radially outside the core. The reactor coolant pumps are mounted in the reactor vessel head above the steam generators with their impellers and diffusers in the steam generator annulus. The reactor coolant travels down through the steam generator tubes, transferring heat to the feedwater; the steam is generated on the shell side of the steam generator. The coolant continues down through the annulus below the steam generator. At the bottom of the reactor vessel, the coolant is directed to the center of the reactor vessel and flows upward through the core. The coolant is heated as it travels through the core; leaving the core, it flows upward to the pump suction and turns downward through the steam generator, beginning another flow pass. The design power level of the PE-CNSG is 365 MWt, an extension of the maritime CNSG power level of 314 MWt made possible by the use of borated coolant for reactivity control. The change in power level was reviewed and confirmed as a part of the Duval application study.

1.3.1. Reactor Vessel

The reactor vessel is a thick-walled, carbon steel vessel clad over the interior surfaces with stainless steel. The inside diameter of the reactor vessel is 3.99 m (157 inches), and the core and steam generators are located within the vessel. Penetrations are provided in the reactor vessel head for the reactor coolant pumps and control rod drive assemblies.

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1.3.2. Reactor Core and Internals

The reactor core consists of 57 fuel assemblies with an active fuel length of 1829 mm (72 inches). On the average each fuel assembly contains 208 Zircaloy-4-clad, helium prepressurized fuel rods 10.922 mm (0.430 inch) in diameter arranged in a 15-by-15 array. The fuel is slightly enriched uranium in the form of uranium dioxide pellets. The PE-CNSG design includes 17 control rods and uses boric acid in the reactor coolant for long-term reactivity control. Rated core thermal power output is 365 MW.

1.3.3. Reactor Coolant Pumps

The four vertically mounted reactor coolant pumps are wetted-motor, single-stage, mixed-flow devices with a capacity of 1.196 m^3/s (18,950 gpm) – each at 32.31 m (106 ft) of head. The design provides sufficient pump inertia to accommodate the loss-of-flow transient and subsequent coastdown.

1.3.4. Steam Generator

The PE-CNSG uses 12 modular OTSGs arranged in a circle inside the reactor vessel. Primary fluid flows down through the steam generator tubes. On the shell side feedwater enters and steam exits through concentric pipes. Feedwater enters the generator through the inner pipe and is directed through a downcomer to the bottom of the steam generator. It then passes up through the tube bundle outside the tubes and exits from the top of each module as superheated steam. At full power the PE-CNSG produces 183 kg/s (1,450,000 lb/h) of steam at 4.83 MPa (700 psia) and 281C (538F) with 19.4C superheat (35F). The feedwater temperature at full power is 204C (400F).

Each steam generator module has 993 Inconel tubes, each 127 mm (0.5 inch) in outer diameter. The feedwater and steam lines, up to and including the first isolation valve, are designed to withstand primary system pressure.

1.3.5. Containment

A compact pressure-suppression system is provided. The shell is a free-standing, bottom-supported steel cylinder 11.58 m (38 feet) in diameter and 19.51 m (64 feet) high overall. The center section of the upper head is removable for construction installation, servicing of the major components, and refueling. A personnel hatch near the main operating floor provides access for routine maintenance and inspection.

1.3.6. Balance-of-Plant Design

The BOP design for this study was based on the land-based PE-CNSG plant concept developed by B&W in conjunction with the UE&C as described in reference 1. For that study the hypothetical site chosen was the standard Middletown site described in reference 2. The PE-CNSG for that study was rated at 313 MWt with two separate conceptual BOP designs with a production of 137 kg/s (1,090,000 lb/h) of process steam or 91 MW of electricity. The objective of this study was to interface the conceptual design with the particular conditions of the Duval process at the augmented power level of 365 MWt.

The plant layout for the PE-CNSG is shown in Figure 4.3-5. The plant has a reactor service building containing the CNSG, its containment and all supporting nuclear auxiliary systems, a control building, a diesel generator building, and an administrative building. The reactor service control and diesel generator buildings (seismic Class 1 structures) are designed to withstand tornado-generated missiles.

1.3.7. BOP Modifications

Some minor changes in the containment have been required, but on the whole the reactor containment, the reactor service building, the control room, and many other BOP features have remained the same as the plant design described in reference 1. The primary changes were in the development of the process energy delivery system (tertiary loop). For the Middletown site two separate plants were addressed to produce steam and electricty; for the Duval site a mixture of steam for existing turbine generators and service requirements, and hot water for mine use is necessary. The design modifications for the Duval site are further restricted by two site constraints. Since the site is both arid and remote, special consideration is required to provide adequate water and backup electricity. The BOP design includes a water treatment facility to provide water at the desired purity level and to supply offsite electrical power from a utility system approximately 14 km (9 mi) from the site.

1.3.8. Tertiary Loop Design

To supply the slightly pressurized 325F water required for the Duval mining operation, either of two alternative design arrangements could have been chosen. The first alternative examined was the most straightforward and inexpensive procedure, whereas reactor steam output would be largely consumed by mixing with process feedwater and then by injecting into the sulfur beds. This process (Figure 4.6-1) had the inherent drawback of requiring once-through processing of steam generator feedwater to high-quality requirements from raw water of poor quality. The second alternative was to provide a tertiary loop with an additional heat exchanger interface between the nuclear system and the process.

The tertiary loop system would transfer energy from the PE-CNSG using reboilers and feedwater heaters. The reboilers would produce saturated steam at 1.13 MPa (164 psia) in the tertiary loop that would be transported to the existing boiler/turbine building. This steam would be used as service steam for heating the mine water to the required 162.8C (325F) final temperature and directly in the existing turbine generators to produce the 11.5 MW of electrical energy required for use at the site. The principal advantage of this system is that the steam and feedwater (secondary loop) system could be controlled closely, minimizing the possibility of contamination of the secondary loop water and subsequent fouling or damaging of the PE-CNSG steam generator modules. The tertiary loop system was chosen as the reference design (Figure 4.6-4).

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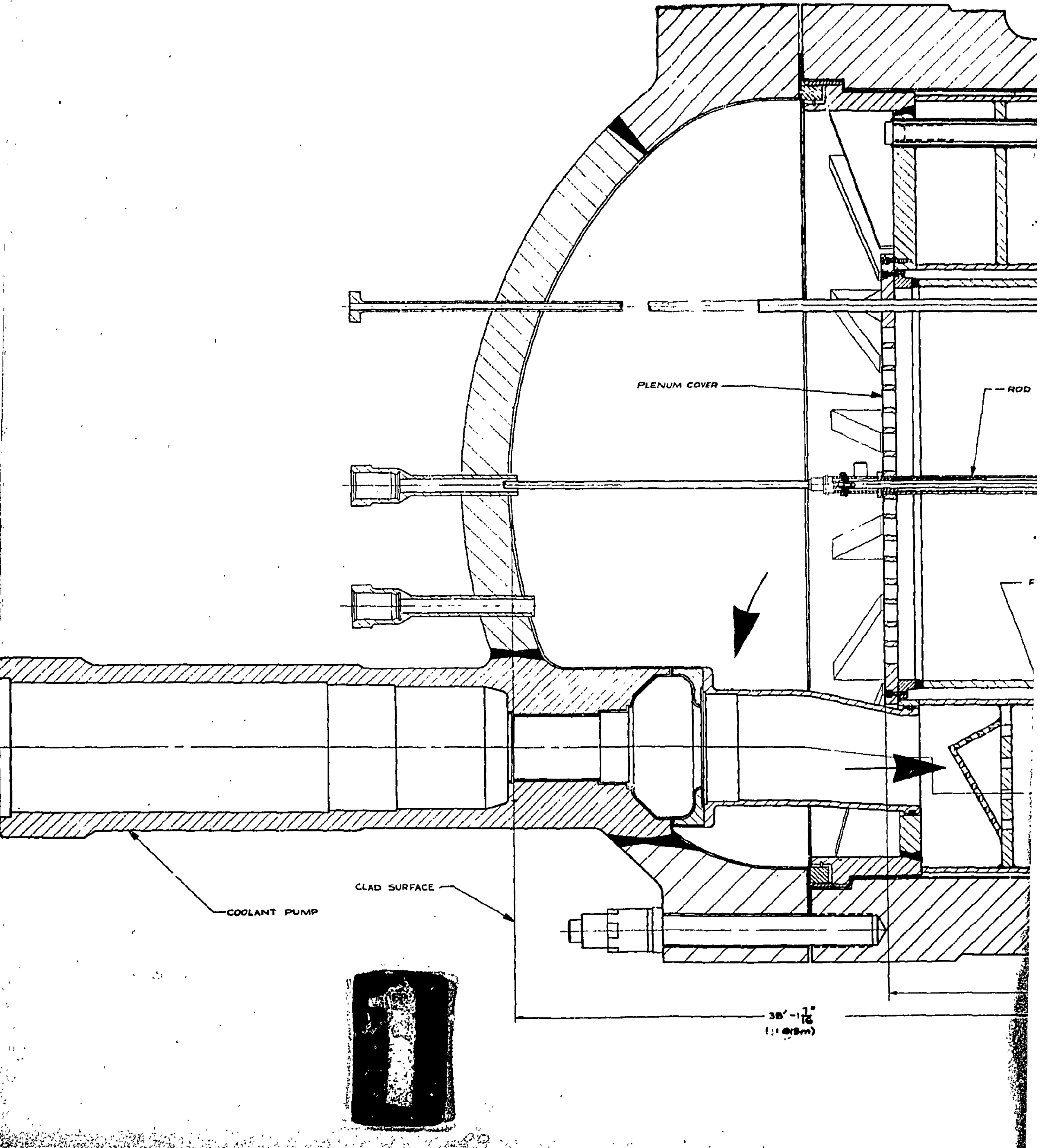
PLENUM COVER
ROD
CLAD SURFACE
COOLANT PUMP
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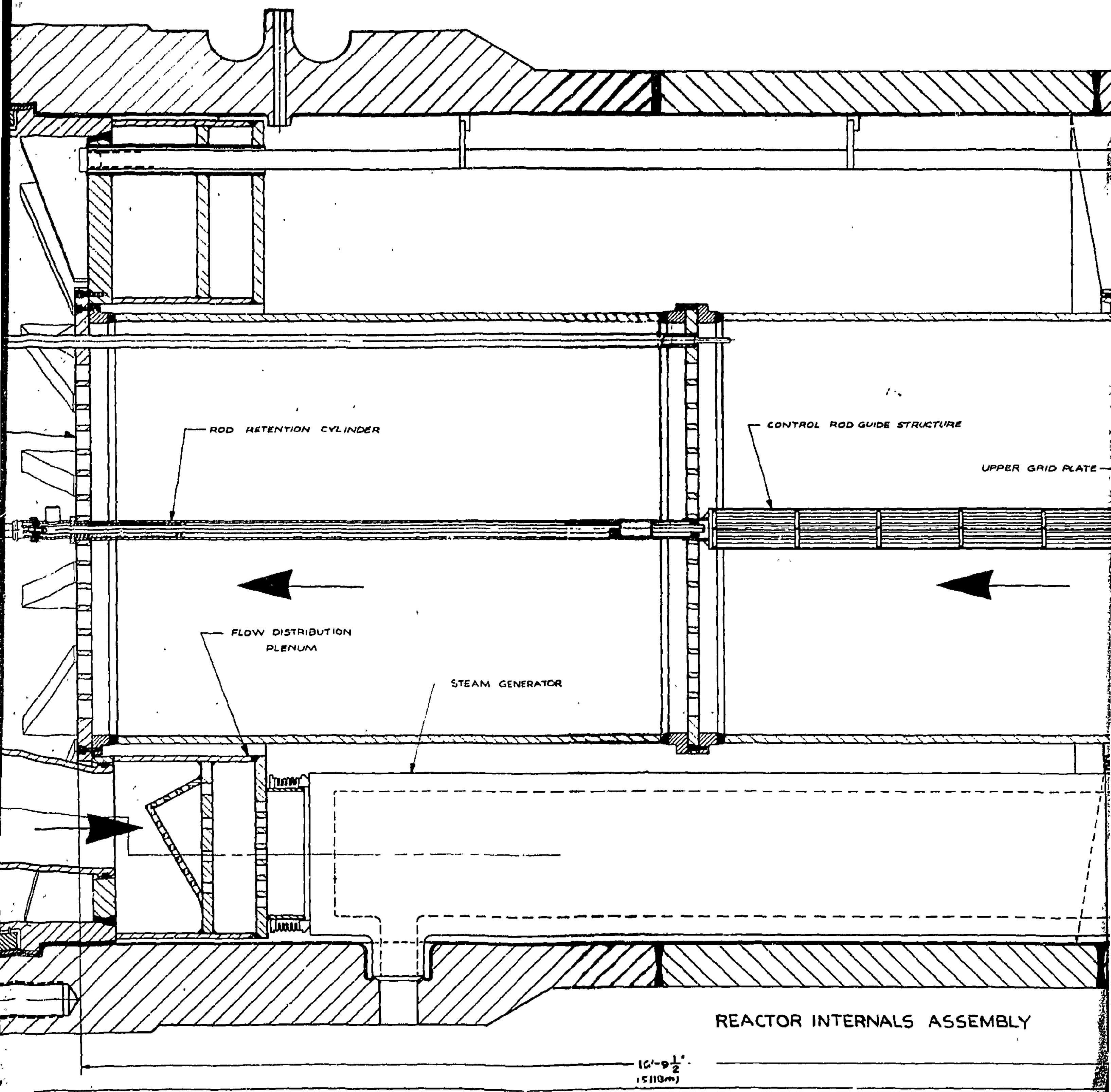

REACTOR INTERNALS ASSEMBLY

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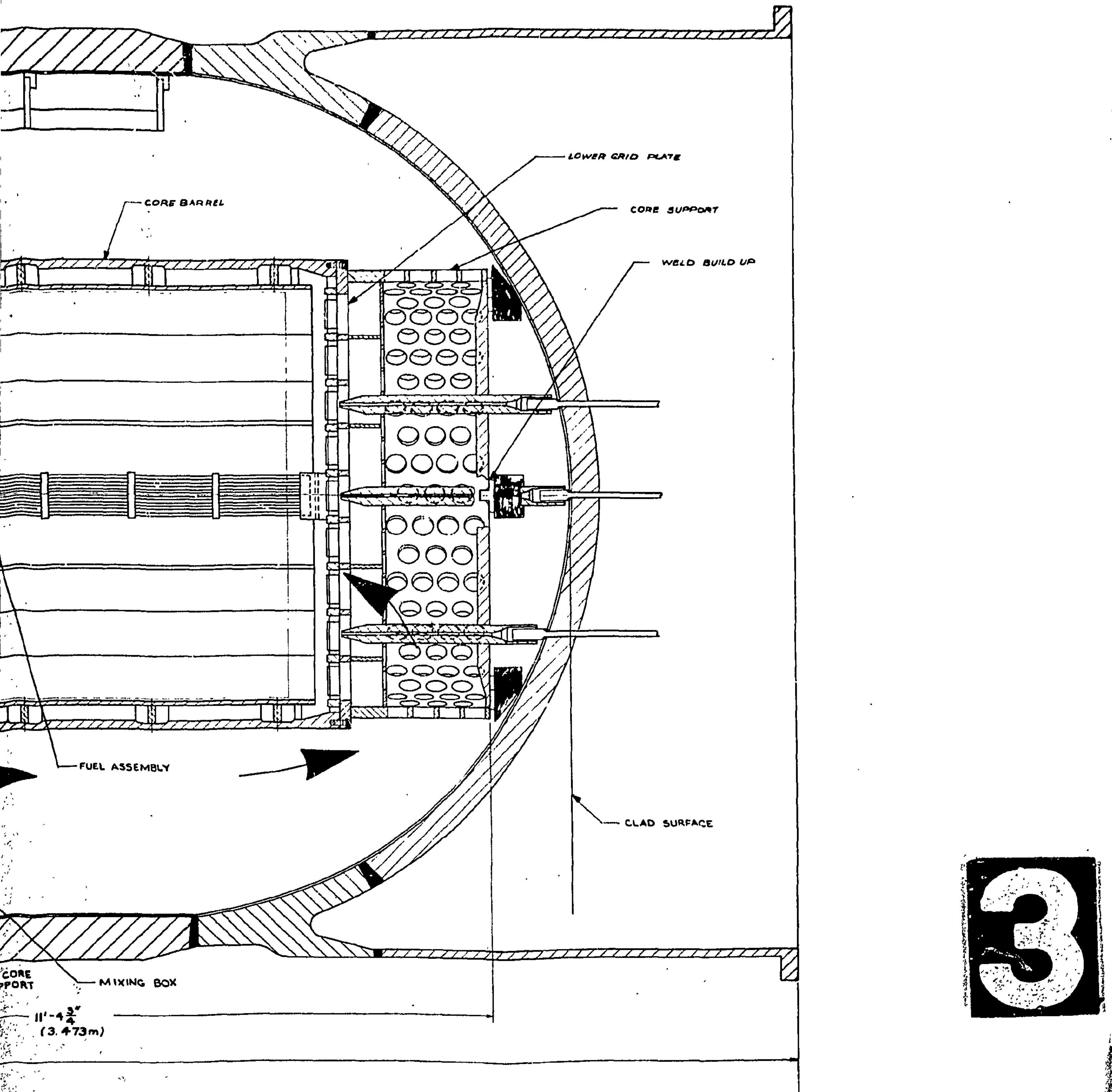
Figure 1.3-1. Elevation View of the CNSG



1.4. Recommendation and Conclusion

Based on study results reported herein, there is no technical reason why a PE-CNSG nuclear plant of the recommended design could not be substituted for the existing natural gas energy source at Duval's sulfur mining operation. The incremental investment for the nuclear plant, when compared with conversion to coal and to coal prices in the $30-40/ton price range, does not produce a rate of return high enough to meet the investment criteria used by Duval.

Therefore, it is concluded that under present circumstances, further study of the CNSG industrial reactor application at the Culberson County site does not appear to be warranted. However, it may become worthwhile to reexamine the prospects for the nuclear option if some of the underlying economic circumstances should change. For example, if new tax incentives become available to encourage capital investment in new industrial energy sources, if the cost of burning coal were to rise (for example due to more strict environmental regulations), or if the prices of fuels were to rise at a rate significantly greater than the prevailing inflation rate, the economics of the nuclear option could become more attractive.

If additional study of the PE-CNSG application at the Duval site is undertaken for these or other reasons, it is recommended that the study areas below receive early additional attention.

Heavy Equipment Transportation — Existing restrictions on weight and dimensions may require shipment of PE-CNSG heavy components by a special transporter from Galveston, Texas to the site. One potential alternate rail route has also been identified. Therefore, it is recommended that a detailed route study be made and the associated transportation cost be estimated.

Construction Schedule — The construction schedule used for this study assumes that adjustments made to accommodate partial field fabrication of the containment base plate and field installation of steam generators will not impact the overall schedule. Potential delays of approximately six months were identified. The remoteness of the Duval site may also impact the construction schedule through a lack of skilled workmen. It is recommended that the overall schedule be reviewed with attention to these considerations.

2. PE-CNSG SITE AND APPLICATION STUDY

2.1. Introduction

In 1974, B&W began a design study for ERDA to determine if the CNSG previously developed for shipboard installation would be suitable for a demonstration process steam plant. The results of the study are presented in BAW-1428.[1] The primary conclusion of the studies was that "The concept of using the CNSG reactor system for supplying process energy to industrial processes appears technically and economically feasible."

Based on the study results for the Middletown site and the ground rules utilized,[2] the land-based CNSG for process energy (PE-CNSG) appeared to be competitive with fossil fuels under some conditions; however, the range of possible energy conditions and process variables for different industrial applications is large. Further, potential sites could impose restrictions on the use of nuclear power. The interfaces between the PE-CNSG and any specific process application could require design modifications not apparent for the Middletown site. For these reasons B&W initiated a follow-on study program for ERDA in conjunction with UE&C to integrate the PE-CNSG and ancilliary equipment into specific process applications.

This study includes generic issues to the PE-CNSG but not related to a specific site or process. In addition, the specific needs of the Duval Corporation for energy at the Culberson County, Texas sulfur mining operation are addressed.

2.2. Scope of Study (Duval Corporation)

The study program comprises the following tasks:

1. Task 1 — Develop Site Data Base: Potential industrial process energy participants were identified by B&W and ERDA. Each potential participant was asked to provide technical and safety interface data, economic evaluation criteria, and site parameters to support the program. In particular, information on process parameters, availability requirements, process hazards, economic model, and site characteristics was requested.

2. Task 2 — PE-CNSG Interface Development: At the conclusion of the initial study,[1] it was determined that an increase in CNSG power level from 314 to 365 MWt would provide a technically and economically superior product for process energy applications. Because detailed design efforts had been based on the lower power level, the CNSG's design capability to achieve 365 MWt was evaluated.

   The detailed process data from the industrial participants were reviewed to determine interface requirements and possible design or operational changes. A design for the process energy delivery system (tertiary loop) was also developed. In addition to interfacing the PE-CNSG with the process, baseline safety analyses for the design basis LOCA were performed, and containment pressure and temperature responses to the design basis LOCA were determined.

3. Task 3 — Siting Constraints: Siting of the PE-CNSG can be constrained by many factors. In the absence of a complete environmental and safety review, site suitability cannot be fully addressed. However, the geologic and seismic characteristics, atmospheric diffusion characteristics, and population density and distribution for each site were used in determining whether the potential site should undergo additional study.

   Dose-to-man due to the design basis LOCA was analyzed. The radiological analysis was based on industrial participant meteorology data, when available, or on the assumptions of USNRC Regulatory Guide 1.4, "Assumptions Used for Evaluating the Potential Radiological Consequences of a Loss-of-Coolant Accident for Pressurized Water Reactors." Present and projected site demography was used for this study. Recommendations on site potential based on the analysis and evaluations above were prepared, and possible regulatory relief was identified where necessary.

4. Task 4 – Economic Evaluation: This task compared the economics of using an on-site nuclear energy source with other currently available replacement energy sources at the industrial participant's specific sites, locations, and conditions. In addition, the effects of various factors at different sites were compared with a uniformly applied economic base.

   A standard analytical economic evaluation model was developed in conjunction with the Duval Corporation and ORNL. This model incorporates specific process plant considerations such as availability, construction time differential, and alternate fuel cycles. Economic input data for the process plant were provided to B&W by the Duval Corporation. Where specific data were considered proprietary by the industrial participant, a range of data was employed.

   Economic data on the BOP as defined for each site were provided for the economic model by UE&C. This data reflected changes made to the BOP to accommodate the specific process.

5. Task 5 – Report: Included in this task is the preparation of a summary report for all tasks for the application/site studied. B&W worked in conjunction with United Engineers and Constructors, Inc., on all sites. This report presents the results, conclusions, recommendations from the generic tasks (as noted), and the specific considerations for the PE-CNSG as the replacement energy source for the Duval Corporation's sulfur mining facilities located in Culberson County, Texas.

2.3. Study Basis

The PE-CNSG design used for this study is essentially an updated version of the design reported in BAW-1428.[1] Design modifications have been limited to those resulting from the parallel nuclear merchant ship program or required as a result of the increased power level. Modifications to the BOP are limited to those imposed by Duval's sulfur mining process and attendant site conditions. The tertiary loop design was developed to conform to the parameters provided by the Duval Corporation.

Ground rules for the economic evaluation base case were provided by the Duval Corporation and ERDA to B&W.

3. SITE IDENTIFICATION AND EVALUATION

3.1. Introduction

Section 3 treats the results of the preliminary evaluation of the suitability of the Duval site for siting a nuclear power plant such as the PE-CNSG. The objective of this preliminary evaluation was not to study site suitability, per se, including a specific boring program and related field studies to assess foundation adequacy, nor evaluate the various detailed site acceptability aspects ordinarily considered even in a brief study of this kind. This type of evaluation would have required consideration of such aspects as meteorology, hydrology, siting guidelines, demography, land use, biota and ecological systems, seismology, water quality, flooding, tornadoes, exclusion area, security, licensing, etc. However, because of the importance of geological suitability in the siting of nuclear power plants, the intent was to provide an indication as to whether the economic and technical feasibility study of the PE-CNSG for the Duval site could proceed with low risk of being rendered meaningless later by major geologic inadequacies such as major seismic faults or unacceptable subsidence conditions. Such inadequacies as possible active faults and extensive subsidence could lead to almost insurmountable licensing problems or high costs to make the site acceptable.

Although proper investigation of such potential inadequacies would require extensive field studies, it was felt that the stated objective could be met by a site visit and by examination of Duval's applicable geologic data. This has been accomplished, but the costs associated with site suitability studies and preparations or remedial work (that may be revealed later by detailed field investigations) have not been factored into the economic assessment. Should economic advantages of the PE-CNSG for Duval be evident from this preliminary phase, these costs should be addressed in subsequent refinements to the study.

3.2. Site Geology

Two studies prepared by Woodward-Clyde & Associates for the plant foundations, water reservoir, and plant grading (1968-1969) were made available.[3,4] A written part of a preliminary geologic report on the sulfur deposits was obtained from the mine management during the site visits.

The geology of the area consists of thick marine and evaporitic sediments of Permian age known as Rustler and Castile formations. Petrographically, the Rustler formation contains calcerous and siliceous rocks and varieties of gypsum, while the Castile formation contains massive anhydrite (the principal sulfur body) and calcite. The general structure of the area west of the Delaware Basin consists of a monocline dipping about 2° 30' east.

The Culberson property shows a particularly complicated geology because of the presence of near-vertical strike-slip faults and because of the rock alteration or solution phenomena. A correct interpretation of the geology is difficult because (except for producing zones such as sulfur) no cores were removed from borings and the classification of the rock was based on the drilled material brought to the surface by the drilling fluid. It appears that there is no definite opinion on the detailed structural geology of the site. The study was oriented toward finding the mineralized area more than disclosing the tectonic or engineering geology associated with the siting of a nuclear plant.

Surface geologic mapping of the outcrops and typical geological features, borings results, and stereo photography have been used to prepare the preliminary geologic report. No information was available concerning the age of the suspected faults discussed in the report nor was any movement recorded or presented in the documentation.

Unless there is definite evidence that movement has occurred at least once within the past 35,000 years or recurrently within the past 500,000 years, geologic faults are not defined in Appendix A of 10 CFR 100[3] as capable. The presence or absence of a capable fault in the vicinity of a site is a geological feature that would have a major impact on any nuclear power plant siting; its presence could require design for surface faulting or if discovered in the immediate vicinity, could eliminate the proposed site.

3.3. Site Visit and Evaluation of Available Data

The previous examination of site conditions resulting from studies made by Woodward-Clyde & Associates[3,4] shows that the borings were drilled to a shallow depth into surficial overburden and upper rock deposits (dolomite and gypsum interlayered with claystone and sandstone). Since that exploratory program was performed only to define the foundation conditions for the structures of the sulfur processing plant and the water storage reservoir, no attempt was made to describe or clarify the details of the intricate site geology. The presence of sink holes and the type of rock required a number of water tests, which produced a wide variation of data.

A second report prepared by Woodward-Clyde & Associates during the construction of the reservoir shows the presence of a zone where claystone, siltstone, gypsum-anhydrite, and dolomite appeared as faulted rock.[5] The trace of a probable fault mentioned by the Duval geologists is indicated on the sketch showing the reservoir location and exploratory work that was performed (Figure 3.3-1). The depth investigated by seismic profiles was less than 9.1 m (30 ft) and showed no channels or voids in the rock; the air-track drilled holes ran into competent rock 4.6 m (15 ft) below the bottom of the reservoir. Except for mentioning the location of the fault compartments (down and up), no other information was available on this probable fault.

The mine geologist of the Duval Corporation believes that what have been interpreted by other geologists as faults are actually collapses along solution channels lacking any structural or tectonic significance.

The mine area where subsidence resulting from sulfur extraction follows the extraction process was toured during the visit. The man-made subsidence accounts for movements of up to 7.6-10.7 m (25-35 ft) near the center and tapering off the edge of the subsided area. Mine personnel have kept accurate records of the subsidence phenomenon and its development in time and space.

The proposed site area is relatively flat with no visible outcrops. Southwest of the site, the topography becomes hilly, and dolomite outcrops occur frequently. The proposed reactor site is southwest of the water storage reservoir and the suspected faulted area.

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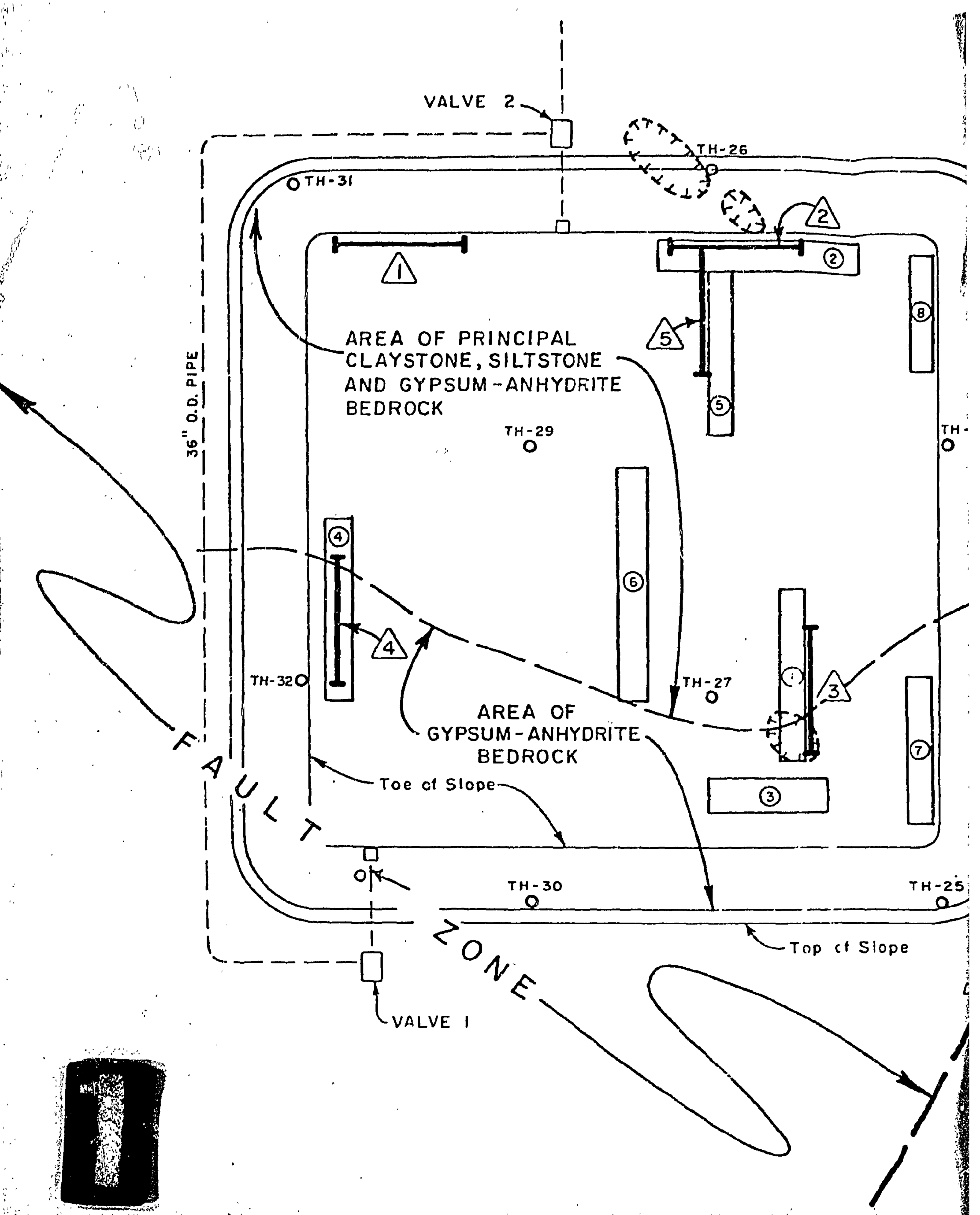
VALVE 2
TH-26
TH-31
2
1
5
8
AREA OF PRINCIPAL
CLAYSTONE, SILTSTONE
AND GYPSUM-ANHYDRITE
BEDROCK
36" O.D. PIPE
TH-29
4
6
TH-32
TH-27
3
AREA OF
GYPSUM-ANHYDRITE
BEDROCK
7
Toe of Slope
FAULT
TH-30
TH-25
Top of Slope
ZONE
VALVE 1

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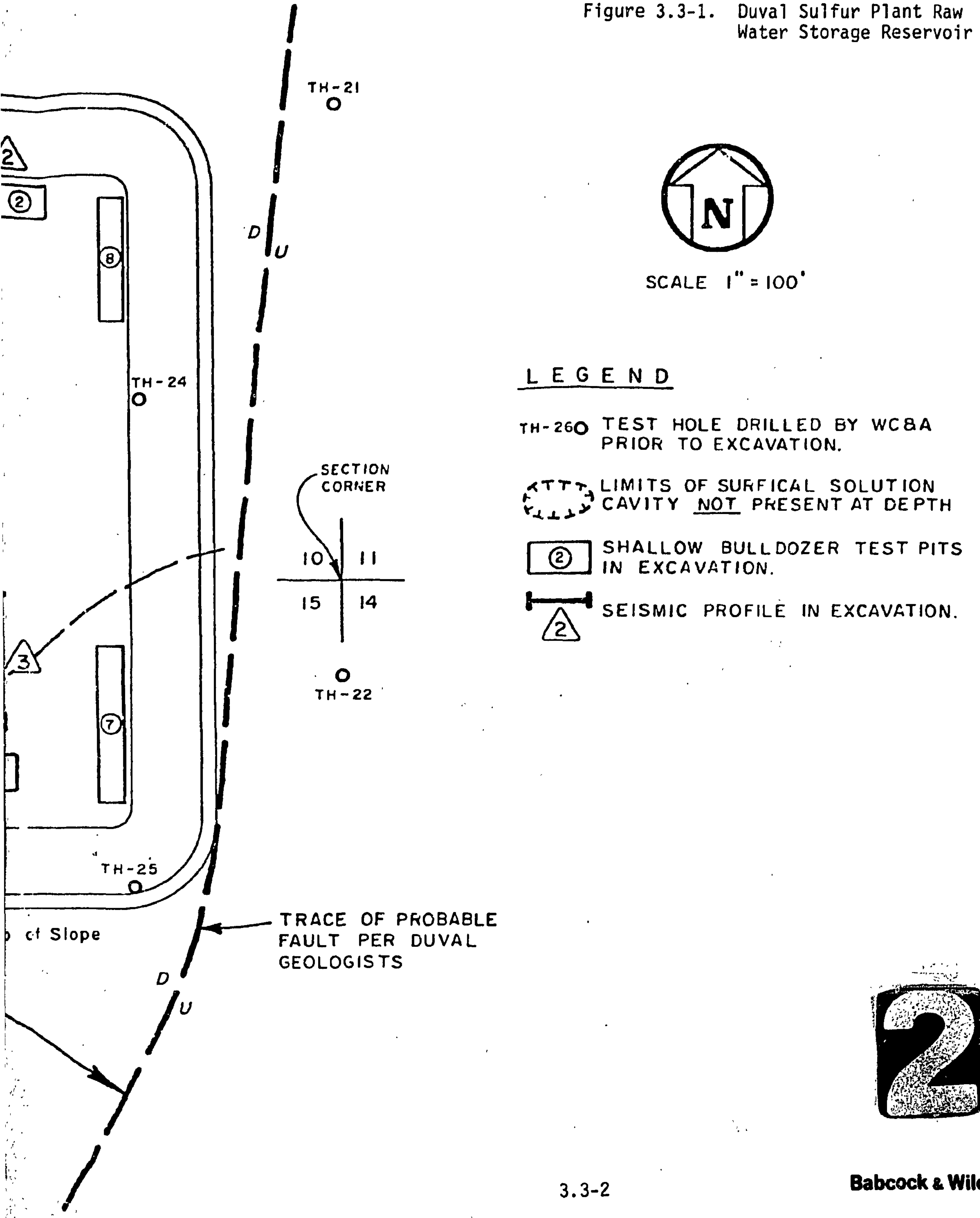
Figure 3.3-1. Duval Sulfur Plant Raw Water Storage Reservoir



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3.4. Conclusions and Recommendations

1. The limited time and amount of information available did not permit any field geology work.

2. The existence of surface sink holes, occurrences of voids in the drilled borings, and the type and solubility of the rock indicate that subsurface subsidence could exist or that collapse could affect the foundations of the nuclear facility. Defining rock characteristics below the foundations and establishing the most secure and economical methods to provide a safe foundation will require a thorough engineering-geology investigation.

3. Because of sulfur mining, the influence of ground subsidence due to the mining of sulfur can be avoided by keeping the site at a safe distance beyond the subsidence funnel and continuous monitoring of the subsidence phenomena.

4. The preliminary geologic report on the mine area was difficult to interpret without mapping and cross sections. The report contained no features or comments to indicate that any of the faults discussed were either capable (USNRC) or active.

Since economic considerations are involved in this phase of study, it is recommended that the conceptual work be based on a location in the general area proposed by the Duval Corporation. Although there are no regulations for the location of a nuclear site in the vicinity of an inactive, healed fault, it is advisable to keep a distance of 152-305 m (500-1000 ft) between the facility and the surface expression of the fault. The selected site location conforms to this guideline to the extent site conditions are known based on limited study.

4. REPLACEMENT ENERGY SOURCE

4.1. Plant Design

This section presents the design of the PE-CNSG for use as the replacement energy source at the Duval Corporation site. Specific topics include the industrial process, plant siting and arrangement, the nuclear plant design, and the design of the nuclear BOP.

4.2. Industrial Process

The descriptions of the mining operation and those characteristics impacting the nuclear plant are discussed below.

4.2.1. Frasch Mining Process

In 1891 Herman Frasch secured his first patents on a process for mining sulfur from limestone and sulfur deposits, but it was not until 1903 that his method became an assured commercial success. Crystalline sulfur has a fairly definite melting point at about 115.6C (240F). The Frasch process profits from this fact and fuses the sulfur underground by pumping water heated above this temperature into the deposit. The melted sulfur, which is almost chemically pure, flows away from the gangue and is pumped to the surface where it may be allowed to solidify.

The sulfur mining method used at the Duval's Culberson mine is largely the same as that developed by Dr. Frasch in 1891. The process involves four basic operations:

1. Drilling a well to the sulfur formation.
2. Treating and heating large quantities of water in a power plant.
3. Pumping superheated water into the deposit to melt the sulfur.
4. Raising the melted sulfur to the surface.

Drilling the Well — To reach the sulfur formation, rotary rigs similar to those employed by the petroleum industry are used. After the hole is drilled, it is equipped with five pipes of various sizes, placed concentrically, which extend from the surface of the ground through the sulfur-bearing stratum. A 340-mm (13.375-inch) OD pipe is set through the upper sediments to an approximate depth of 14 m (150 ft), and then a 244-mm (9.625-inch) OD pipe is set through the remaining strata to the top of the sulfur-bearing formation. The production equipment consists of three strings of pipe — 178-mm (7-inch) OD, 89-mm (3.5-inch) OD, and 19-mm (0.75-inch) OD in size. The 178-mm (7-inch) pipe extends through the sulfur-bearing formation and rests in the upper portion of the underlying calcium sulfate or anhydrite. The 89-mm (3.5-inch) pipe is placed inside the 178-mm (7-inch) pipe and reaches nearly to the bottom of the sulfur-bearing rock but rests on a collar set within the outer 178-mm (7-inch) pipe to seal the annular space between the two. Finally, a 19-mm (0.75-inch) air pipe is set lowered inside the 89-mm (3.5-inch) pipe to a depth slightly

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above the collar in the 178-mm (7-inch) pipe. The 178-mm (7-inch) pipe is perforated at two intervals separated by the collar. The upper set of holes permits the escape of the hot water pumped down the 178-mm (7-inch) pipe, while the lower holes admit the molten sulfur. Figure 4.2-1 is a diagram of a sulfur well operation.

Water – Treating and Heating – The first step in production is the pumping of heated water into the sulfur formation. The amount of water needed to melt a ton of sulfur varies from 3.8-45.4 m^3 (1,000-12,000 gal) depending on the characteristics of the deposit. It takes a number of wells to constitute a mine. Large-scale production requires about 10 million gallons of water each day. To ensure an adequate water supply, it is necessary to develop a reservoir system to store water drawn from wells.

However, before this water can be used in boilers and pipes, it has to be treated and softened to remove harmful scale-forming salts and corrosive substances. From the water treating facilities a portion of water moves to the boilers in the power plant where it is converted into steam. The remainder of the water is fed into heaters where its temperature is increased to 163C (325F) through direct contact with steam. Insulated hot water lines then carry the heated water to field control stations where it is metered into wells.

Steaming the Well – Pumping hot water down the space between the 178- and 89-mm (7- and 3.5-inch) pipes is known to sulfur miners as "steaming the well." Discharging into the porous sulfur formation through the upper set of perforations, the water heats the region through which it circulates, melting the sulfur. Since it is heavier than water, the liquid sulfur sinks and forms a pool at the bottom of the well. Here it enters the well column through the lower set of perforations and rises in the 178-mm (7-inch) pipe until the seal blocking it from the down-rushing hot water forces it into the inner 89-mm (3.5-inch) pipe.

Raising the Sulfur to the Surface – The level to which the sulfur will rise in the well is determined by its specific gravity and the pressure on the hot water in the formation – perhaps half or two-thirds of the distance to the surface. The weight of the sulfur column is reduced by aeration with compressed air jetted down the central 19-mm (0.75-inch) pipe and surges to the surface in what is virtually an airlift.

The water percolating through the underground formation cools as it does its work and must be removed to make room for in-flowing hot water. This is accomplished by "bleed wells," usually located around the outskirts of the producing area, which drain the cold water from the formation. The used water is treated to remove soluble sulfides and other impurities and then reheated for injection into other sulfur wells.

The molten sulfur from the wells is gathered into a heated collection pan and stored in insulated and steam-heated tanks at a temperature of 135C (275F). The molten sulfur is loaded from the storage tanks into insulated tank cars for shipment to consumers or storage terminals.

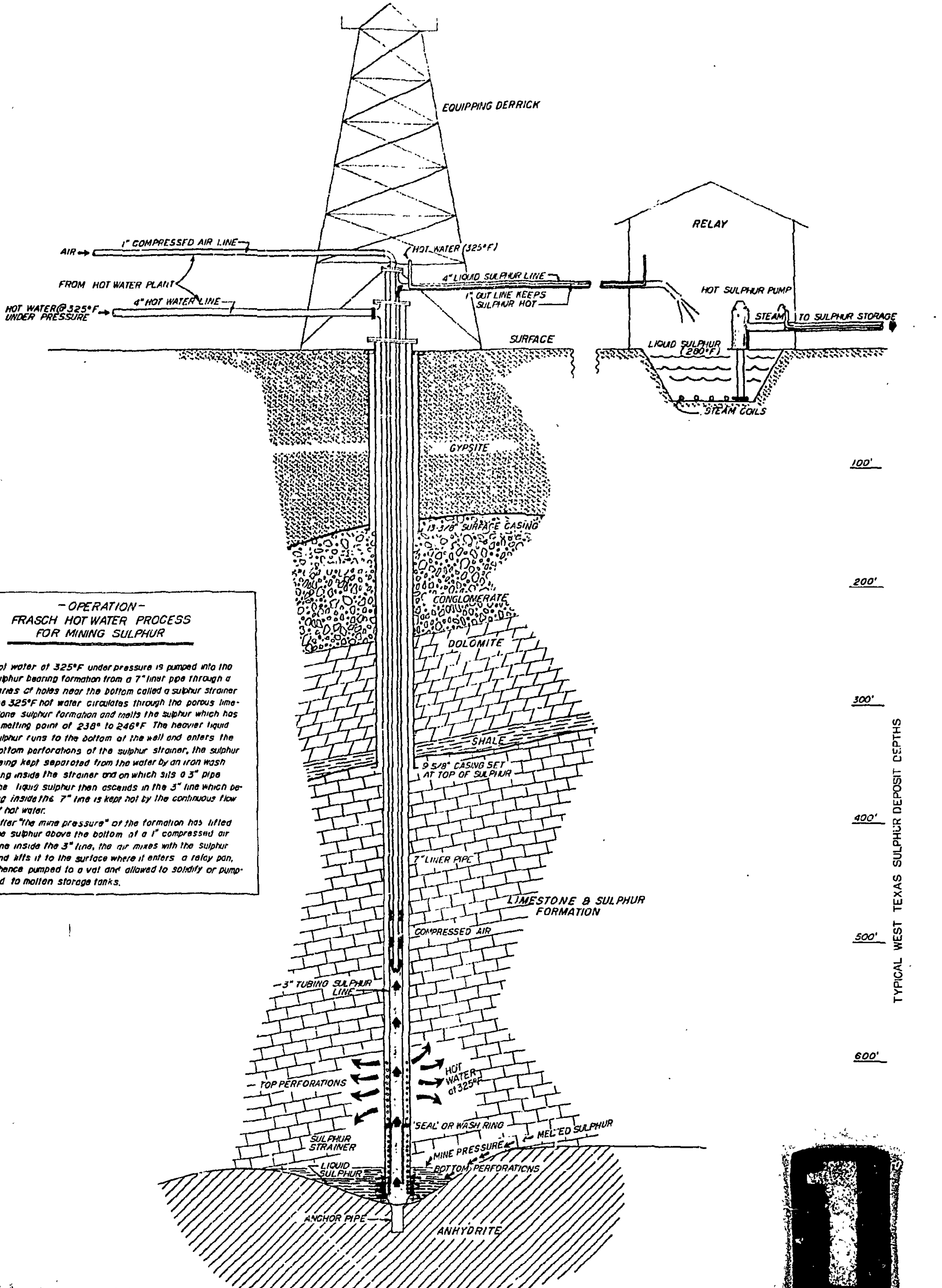
4.2.2. Energy Requirements

The energy required by the Duval Corporation's sulfur mining operation was determined based on the maximum annual sulfur production of 2.54×10^9 kg (2.5×10^6 L-T). The average annual sulfur production expected over the nuclear plant life is 1.73×10^9 kg (1.7×10^6 L-T). Assuming a capacity factor for the PE-CNSG of 0.8 and a 30-day refueling period, the energy required is available with adequate margin. The total energy requirement at maximum production is 11.5 MW for electricity and 348 MW for service steam and process heating.

4.2.3. Special Requirements

The source of raw water for the sulfur mine is deep wells located approximately 67 km (36 miles) from the site. The water is brought by pipeline and stored in a 60,566 m^3 (16 million-gal) reservoir. Because the total hardness and dissolved solids of the raw water are high, water treatment for the PE-CNSG tertiary loop is required. The reference system selected uses reverse osmosis for treatment. The RO system effluent will be piped to existing settlement tanks. Additional analysis of tank capacity is required.

Sufficient electric power is produced onsite for process and facility needs. The primary source of electric power for the reactor and BOP (see above) will be from an off-site source through redundant transmission lines. Standby and emergency power will be provided onsite.



- OPERATION -
FRASCH HOT WATER PROCESS FOR MINING SULPHUR

1. Hot water of 325°F under pressure is pumped into the sulphur bearing formation from a 7" liner pipe through a series of holes near the bottom called a sulphur strainer
2. The 325°F hot water circulates through the porous limestone sulphur formation and melts the sulphur which has a melting point of 238° to 246°F The heavier liquid sulphur runs to the bottom of the well and enters the bottom perforations of the sulphur strainer, the sulphur being kept separated from the water by an iron wash ring inside the strainer and on which sits a 3" pipe
3. The liquid sulphur then ascends in the 3" line which being inside the 7" line is kept hot by the continuous flow of hot water.
4. After "the mine pressure" of the formation has lifted the sulphur above the bottom of a 1" compressed air line inside the 3" line, the air mixes with the sulphur and lifts it to the surface where it enters a relay pan, thence pumped to a vat and allowed to solidify or pumped to molten storage tanks.

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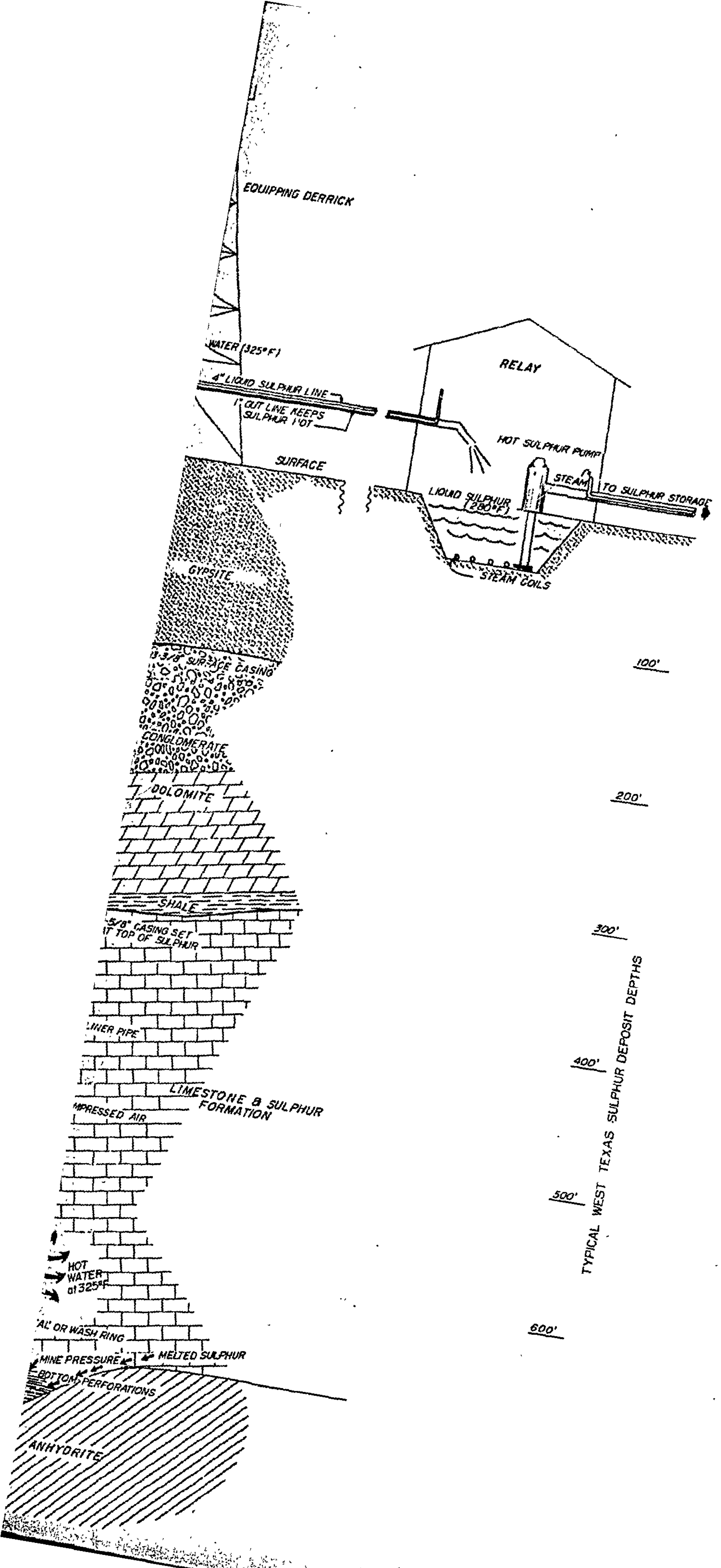

Figure 4.2-1. Frasch Hot Water Process for Mining Sulfur

4.3. Plant Siting

Culberson County, Texas, where the plant site is located is surrounded by Eddy County of New Mexico on the north, Reeves County, Texas on the south, and Ward and Hudspeth Counties, Texas, to the east and west, respectively. Within a 48.3-km (30-mile) radius from the plant site are located Red Bluff Lake to the northeast and scattered ranches with an estimated eight to ten persons each. The population density is 0.017 persons/km^2 (0.043 persons/square mile).

Outside the 48.3 km (30-mile) radius, the nearest concentrations of population are Pecos, Texas and Carlsbad, New Mexico. Pecos has 12,682 people and is 66 km (41 miles) from the site. Carlsbad has 20,600 people and is 81 km (50 miles) from the site. Figure 4.3-1 is a map relating the Culberson property and the 48.3 km (30-mile) radius with the surrounding area. Site topography prior to sulfur production and the current topography are shown in Figures 4.3-2 and 4.3-3.

4.3.1. Site Development

The proposed plant site (Figure 4.3-4) is located southeast of the existing plant and raw water reservoirs where the existing grade slopes toward the northeast at approximately 3.5%. Dolomite outcrops are visible both near and south of the proposed site. Earthwork is proposed to create a plateau at elevation +1010 m (+3315 ft) consisting of approximately 15,300 m^3 (20,000 cubic yards) of cut. It is expected that this excavation can be done by bulldozers with some ripping. In addition, local excavation is proposed for each building. Dewatering will probably not be necessary.

4.3.2. Site Access

Shipment of the PE-CNSG from the fabrication shops to the Duval Corporation site has been studied briefly. Equipment weights and envelope dimensions were determined. The containment base plate and the PE-CNSG reactor vessel were identified as possible limiting items due to size and weight. The B&W Construction Company was consulted to determine potential routes, shipping methods, and cost. Barge shipment to East Texas was recommended, followed by either rail or road shipment to the site. Both rail and road shipment will require the use of special vehicles. Because road shipment is lengthy, additional investigation into rail shipment was initiated. The Santa Fe Railroad identified the following limits for the direct route to the site:

Maximum weight	119,000 kg/car	263,000 lb/car
	30,4000 kg/axle	67,000 lb/axle
Maximum height	5.94 m	19.5 ft
Maximum overhang	2.29 m	7.5 ft

The weight limits are based on the standard eight-axle rail car. Use of B&W's tandem 12-axle cars was acceptable, but no upper weight limit was set.

A further possibility of combined shipment was also identified, but no time was available to pursue the alternative. This alternative would require a siding on the Texas and Pacific Railroad tracks near U.S. 80 and shipment of the material to the site by overland transporter on an east/west access road.

The trend of transporting construction materials by special transporter has been increasing in the past decade because of greater reliability. Also, fuel and materials required during plant operation can be transported by truck. Access to State Route 652 is available by 17.7 km (11 miles) of 7.3 m- (24-foot) wide, double penetration private road. The roads allow up to 36,300 kg (40 T) of GVW. Hauling when low road areas are wet would be restricted.

Table 4.3-1 lists the weights of major PE-CNSG components. The envelope dimensions are marginal for rail shipment. Due to the lack of information on the specific portion(s) of the rail route, which limits the shipping dimensions, the BWCC believes that rail shipment may be possible by modifying the rail line.

To determine the actual shipping method and transportation cost, the BWCC has recommended the following actions:

1. Ship the reactor vessel and steam generators together if the Santa Fe Railroad approves.
2. Investigate schedule effect for field installation of steam generators, if required.
3. Perform a detailed study of the existing rail facilities to confirm suitability and/or recommended rail facility improvements.

4.3.3. Construction Facilities

A plateau at an elevation of 1010 m (3315 ft) with an area of approximately 200,000 m^2 (50 acres) is required onsite for construction storage facilities.

The construction facilities are as follows:

1. Change house
2. Main office
3. Construction parking
4. Subcontractor trailer area
5. Toilet and wash house
6. Sewage treatment plant
7. Pipe shop
8. Electrical shop
9. Temporary power substation.

The construction material storage facilities are as follows:

1. Batch plant
2. Rebar laydown
3. Lumber yard
4. Pipe storage
5. Warehouse
6. Gas, diesel pumps, and tanks
7. Cable yard

4.3.4. Foundations

This study assumes that the main plant building and structures will be constructed on rock. Additional site investigation must be carried out to ensure this.

4.3.5. Facility Locations

The plot plan for the PE-CNSG based on Figure 4.3-4 is shown in Figure 4.3-5. The buildings and equipment are located to minimize construction time and to provide for efficient plant operation. Figure 4.3-6 is the Civil Plot Plan of the existing facilities and Figure 4.3-7 is an aerial photo of the existing facilities.

Table 4.3-1. PE-CNSG Weights

	Weight	
	Metric, kg	English, lb
Nuclear Steam System		
Reactor vessel	248,100	546,500
Closure head	99,340	218,800
Head studs and nuts	6,350	14,000
Steam generator modules	93,440	205,800
Internals	60,780	133,900
CRDMs	14,970	33,000
CRD service structure	9,530	21,000
Pumps and diffusers	18,600	41,000
Pressurizer	39,000	85,900
SG lateral support, mixing box	7,710	17,000
FW inlet/steam outlet adapters	7,260	16,000
NSP dry weight, total	605,080	1,332,900
Water weight, 21C (70F)	95,350	210,000
Total weight	700,430	1,542,900

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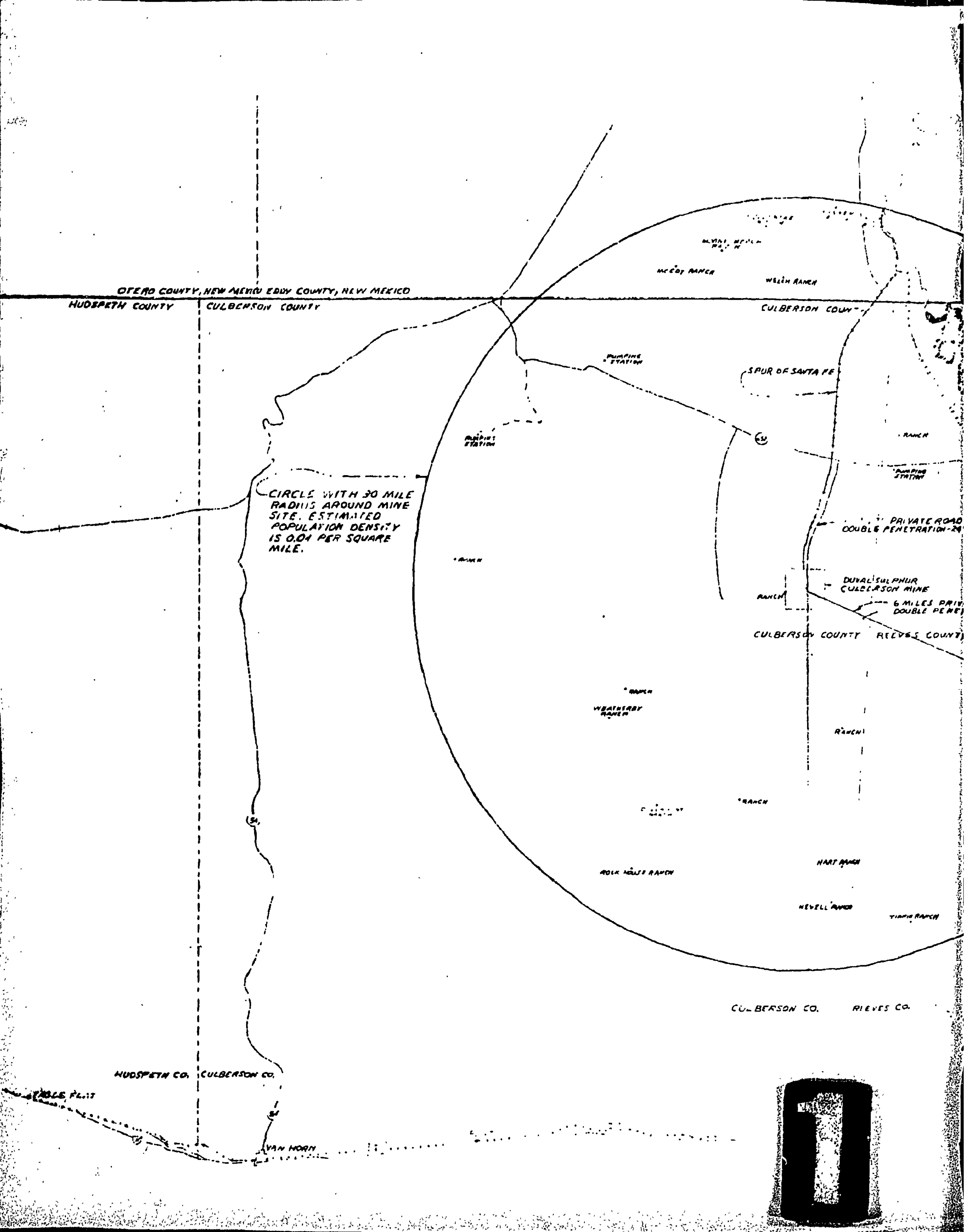

OTERO COUNTY, NEW MEXICO EDDY COUNTY, NEW MEXICO
HUDSPETH COUNTY
CULBERSON COUNTY
McCOY RANCH
WELCH RANCH
PUMPING STATION
SPUR OF SANTA FE
PUMPING STATION
RANCH
PUMPING STATION
CIRCLE WITH 30 MILE RADIUS AROUND MINE SITE. ESTIMATED POPULATION DENSITY IS 0.04 PER SQUARE MILE.
PRIVATE ROAD
DUVAL SULPHUR CULBERSON MINE
RANCH
RANCH
CULBERSON COUNTY REEVES COUNTY
RANCH
WEATHERBY RANCH
RANCH
RANCH
ROCK HOUSE RANCH
HART RANCH
NEWELL RANCH
CULBERSON CO. REEVES CO.
HUDSPETH CO. CULBERSON CO.
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VAN HORN
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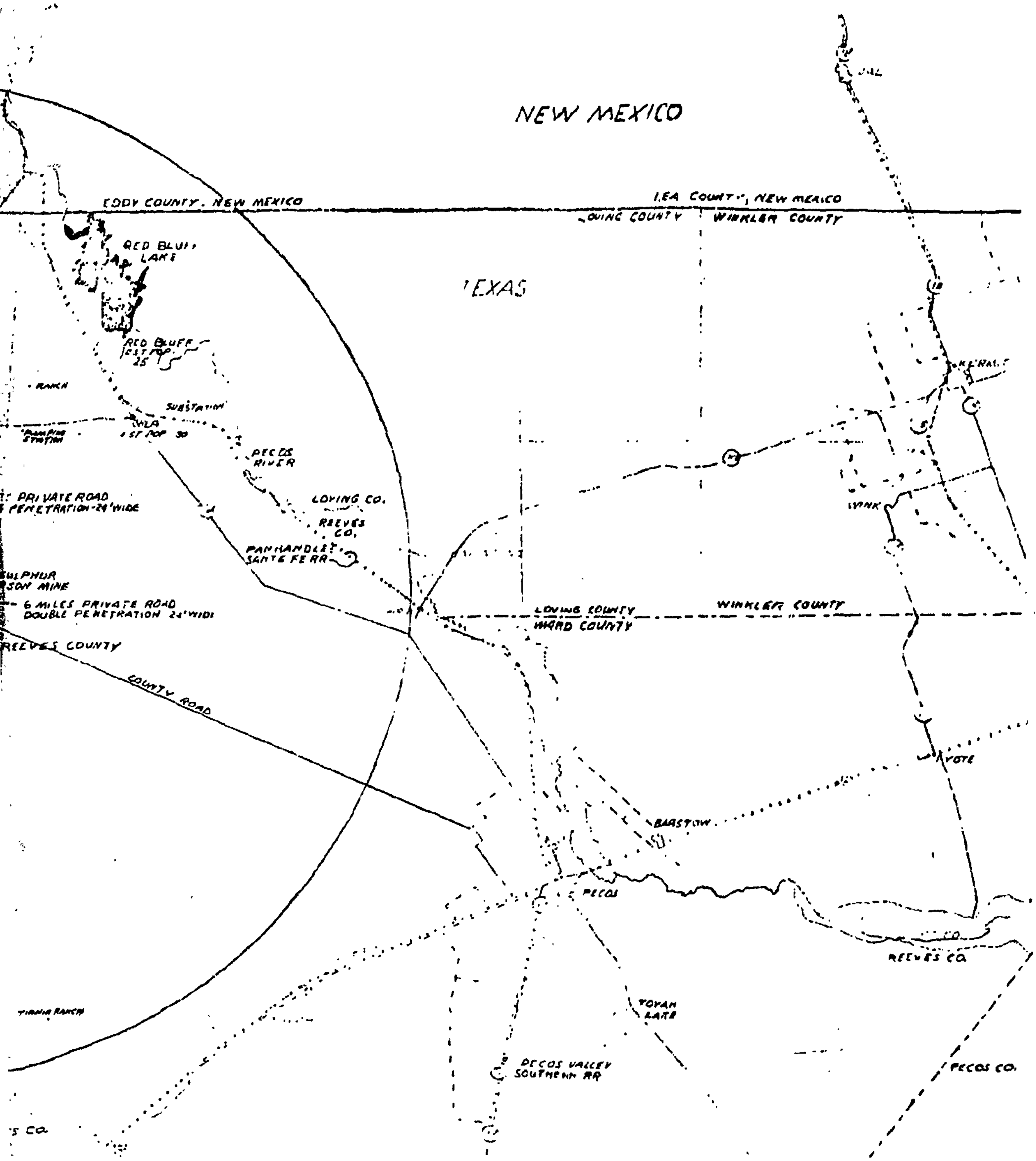
Figure 4.3-1. Duval Sulfur Mine and Surrounding Area



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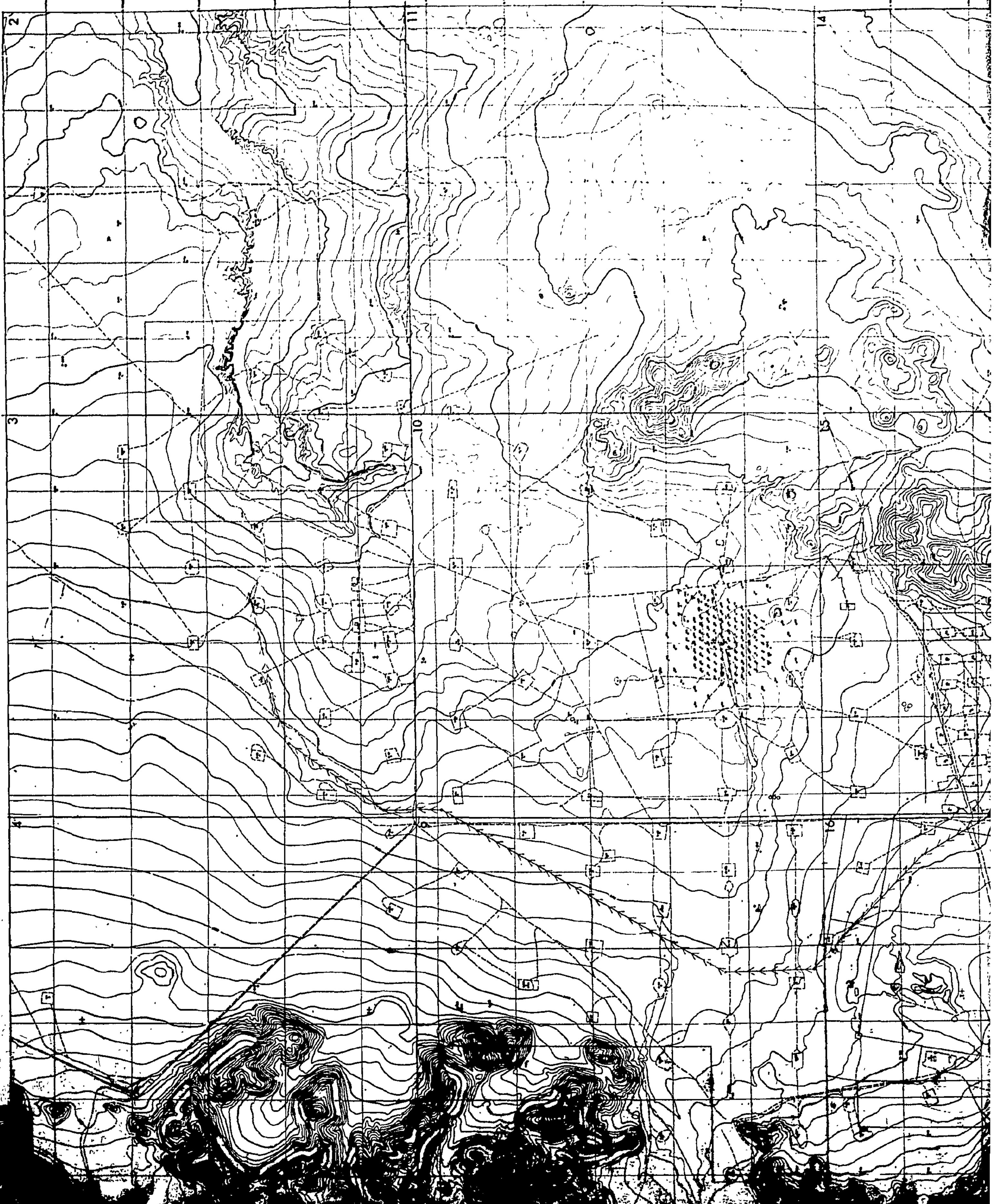

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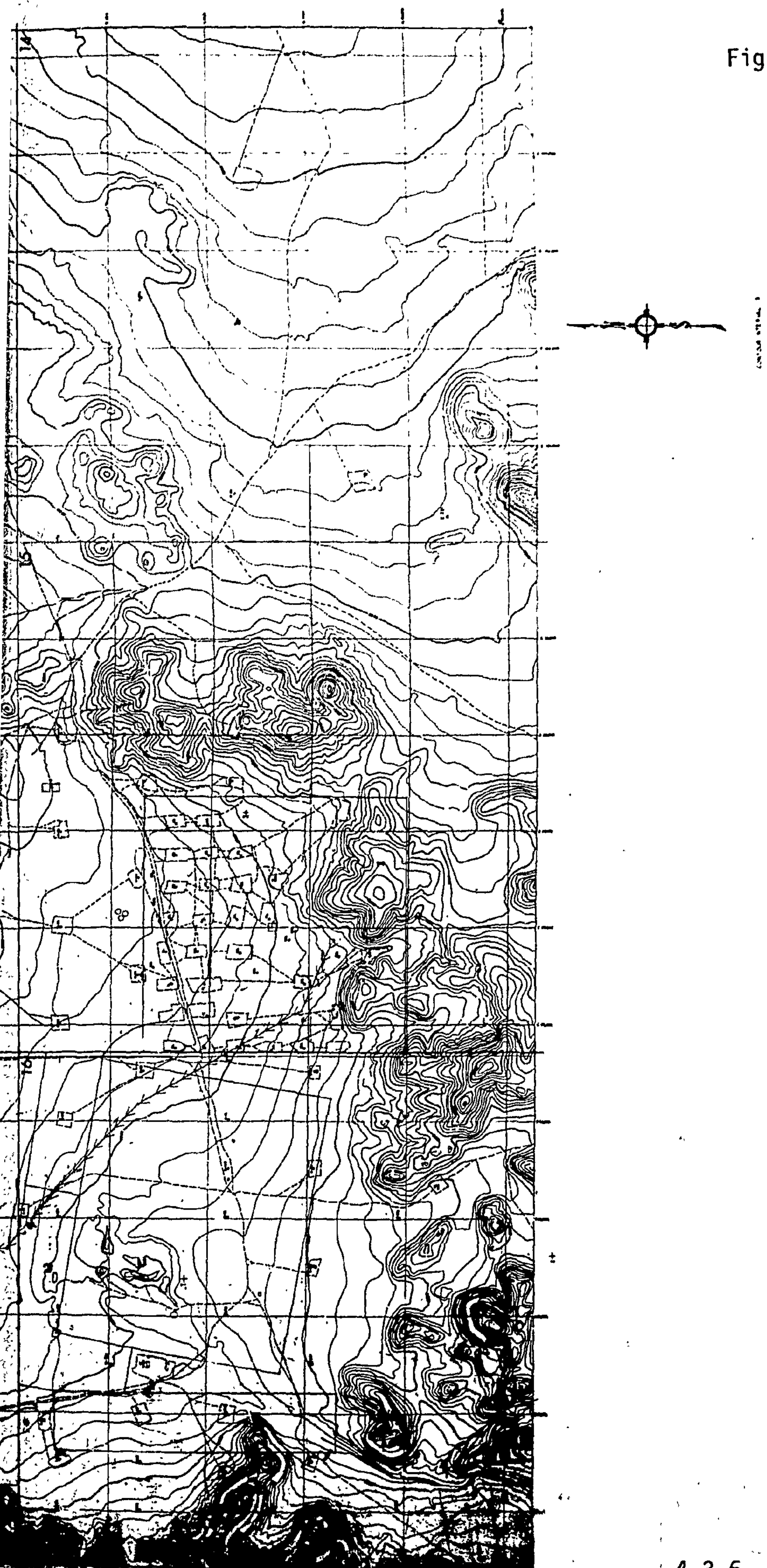
Figure 4.3-2. Topographic Map of Culberson Property – 1969



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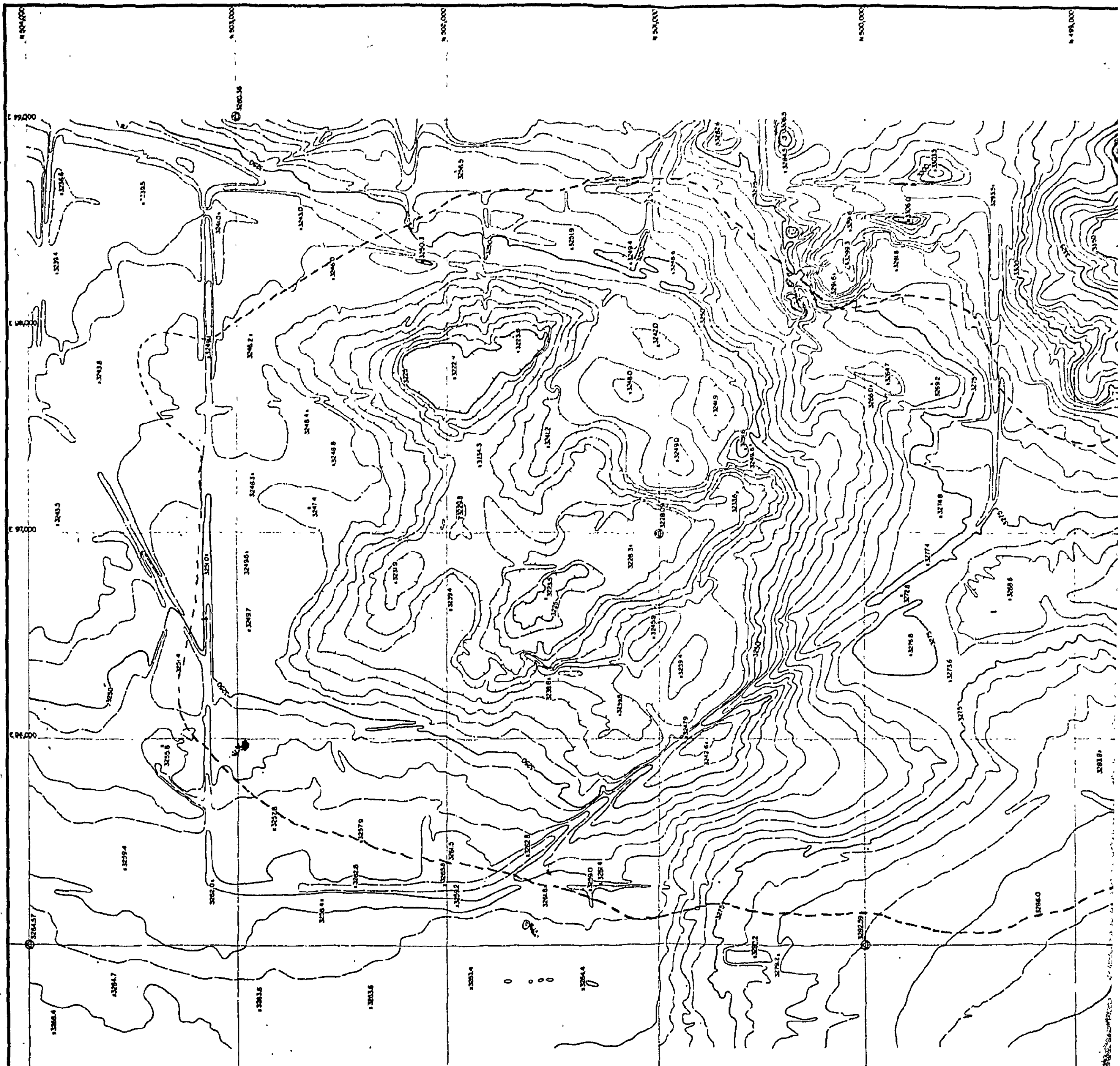

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Figure 4.3-3. Topographic Map of Culberson Property – August 1976

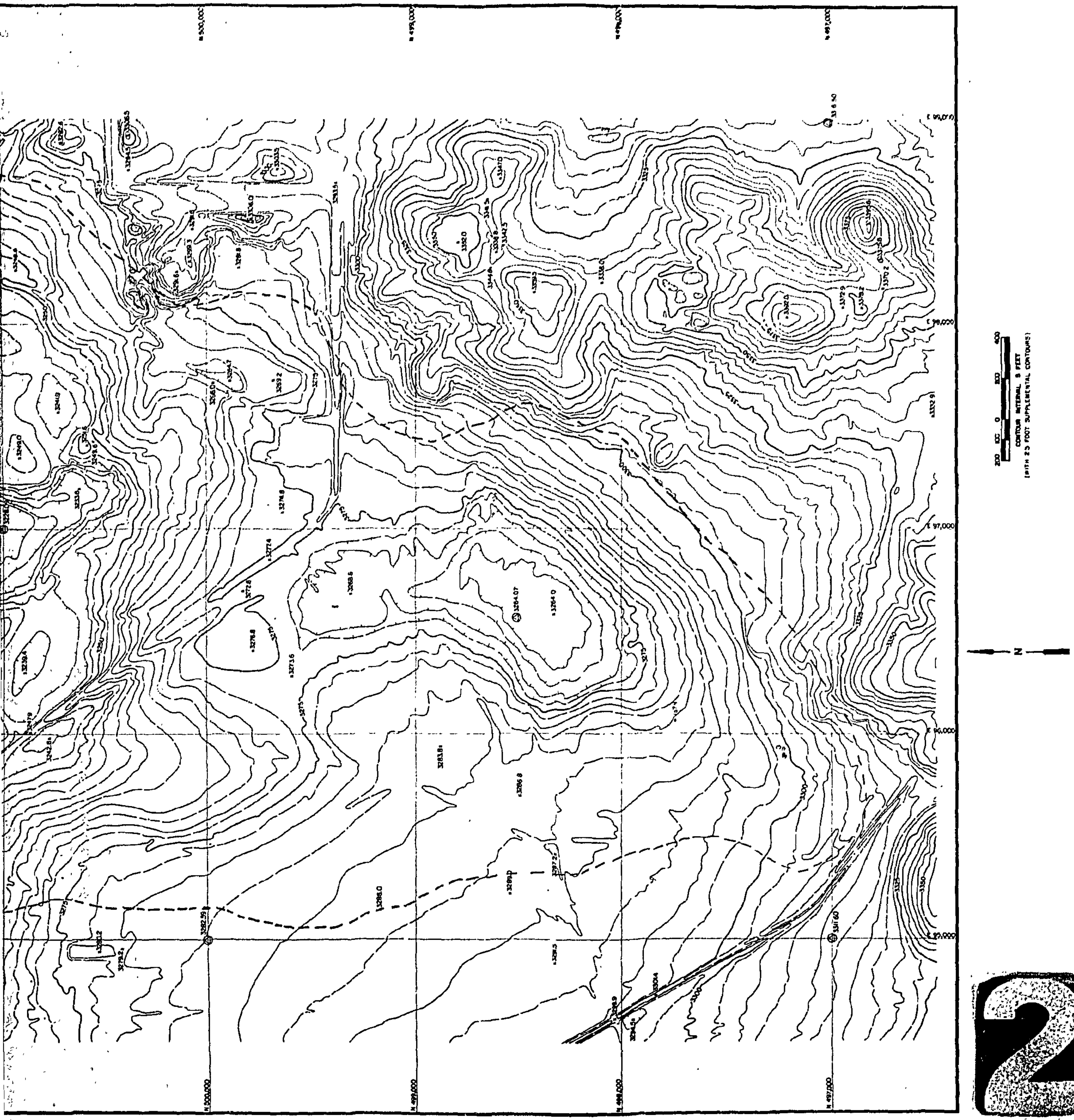

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Figure 4.3-4. 365-MWt CNSG for Process Heat — Site Plan

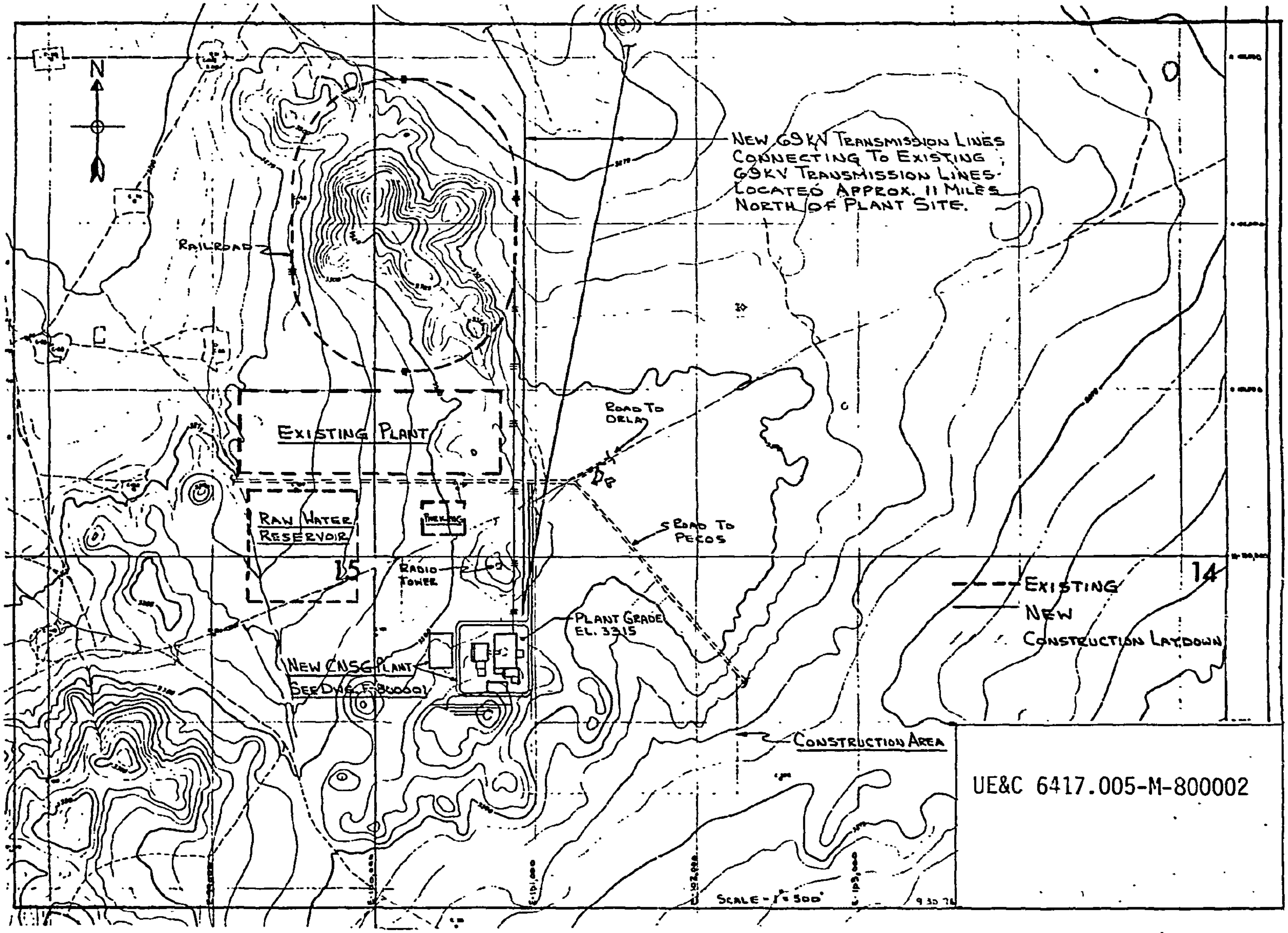

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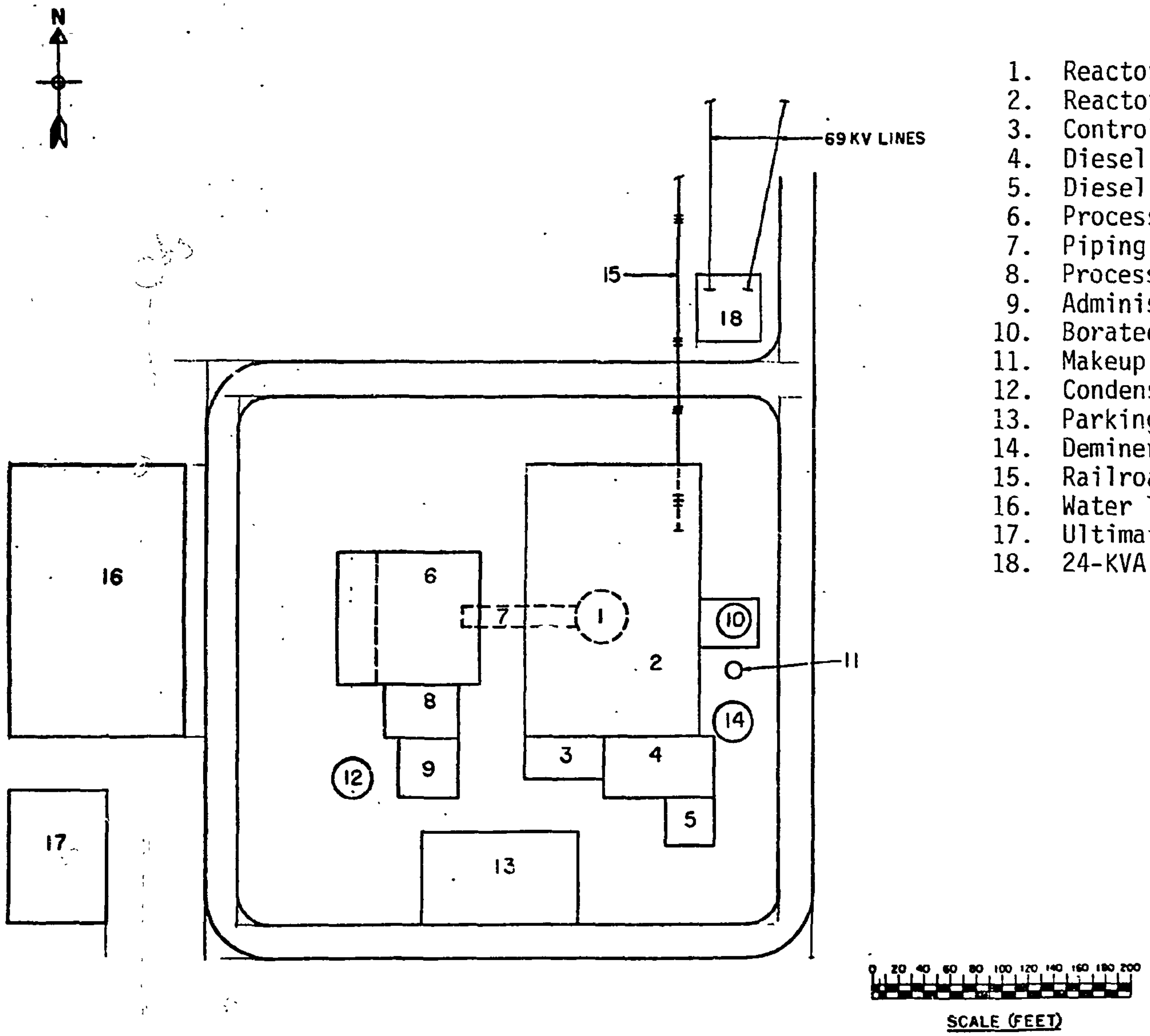
Figure 4.3-5. 365-MWt CNSG for Process Heat — Plot Plan



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1. Reactor Containment Vessel
2. Reactor Service Building
3. Control Building
4. Diesel Generator Building
5. Diesel Fuel Oil Building
6. Process Heat Building
7. Piping Tunnel
8. Process Heat Service Building
9. Administration Building
10. Borated Water Storage Tank
11. Makeup Tank
12. Condensate Storage Tank
13. Parking area
14. Demineralized Water Storage Tank
15. Railroad Siding — Fuel Car
16. Water Treatment Area
17. Ultimate Heat Sink Cooling Tower
18. 24-KVA Substation

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Figure 4.3-6. Civil Plot Plan

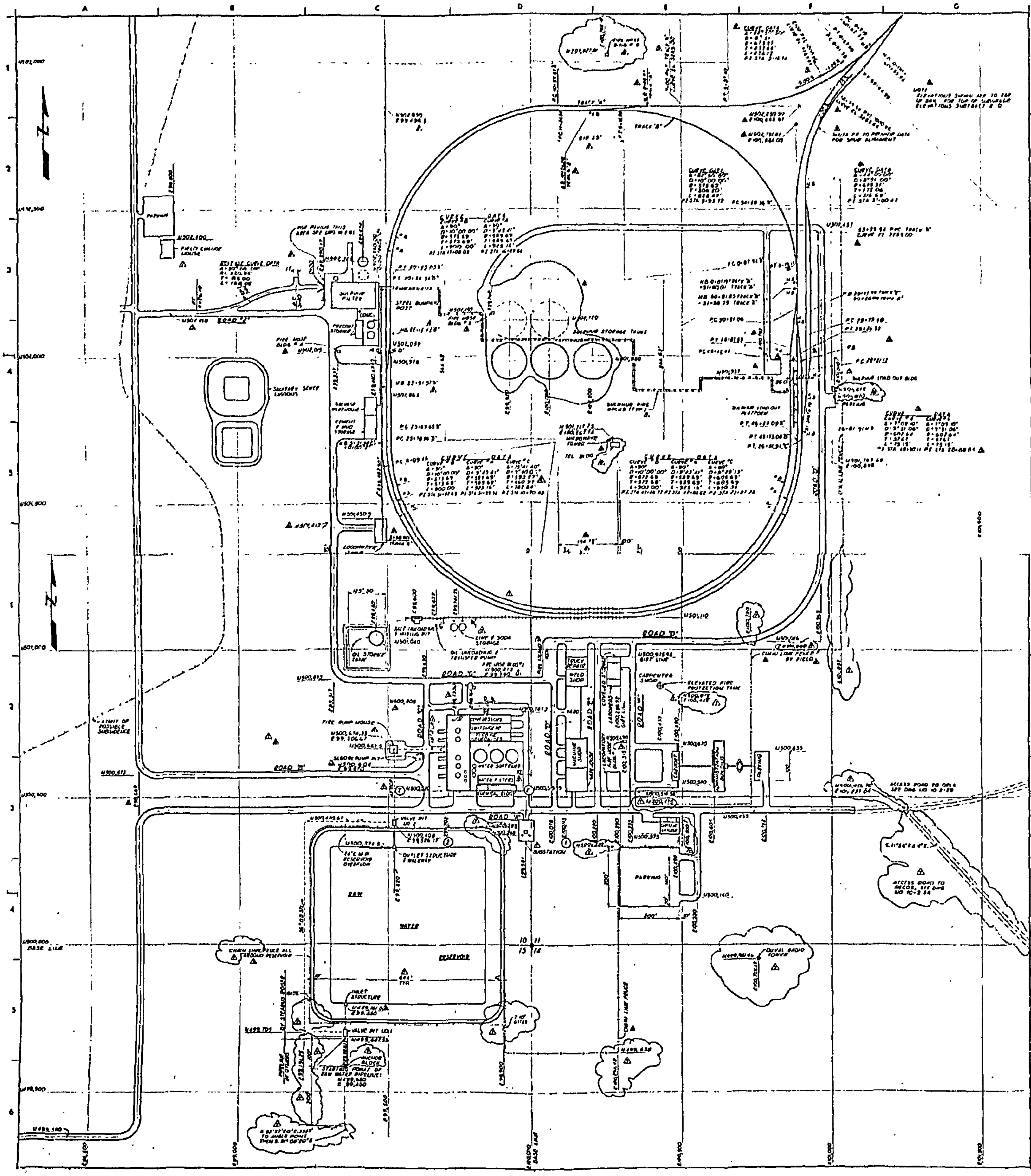
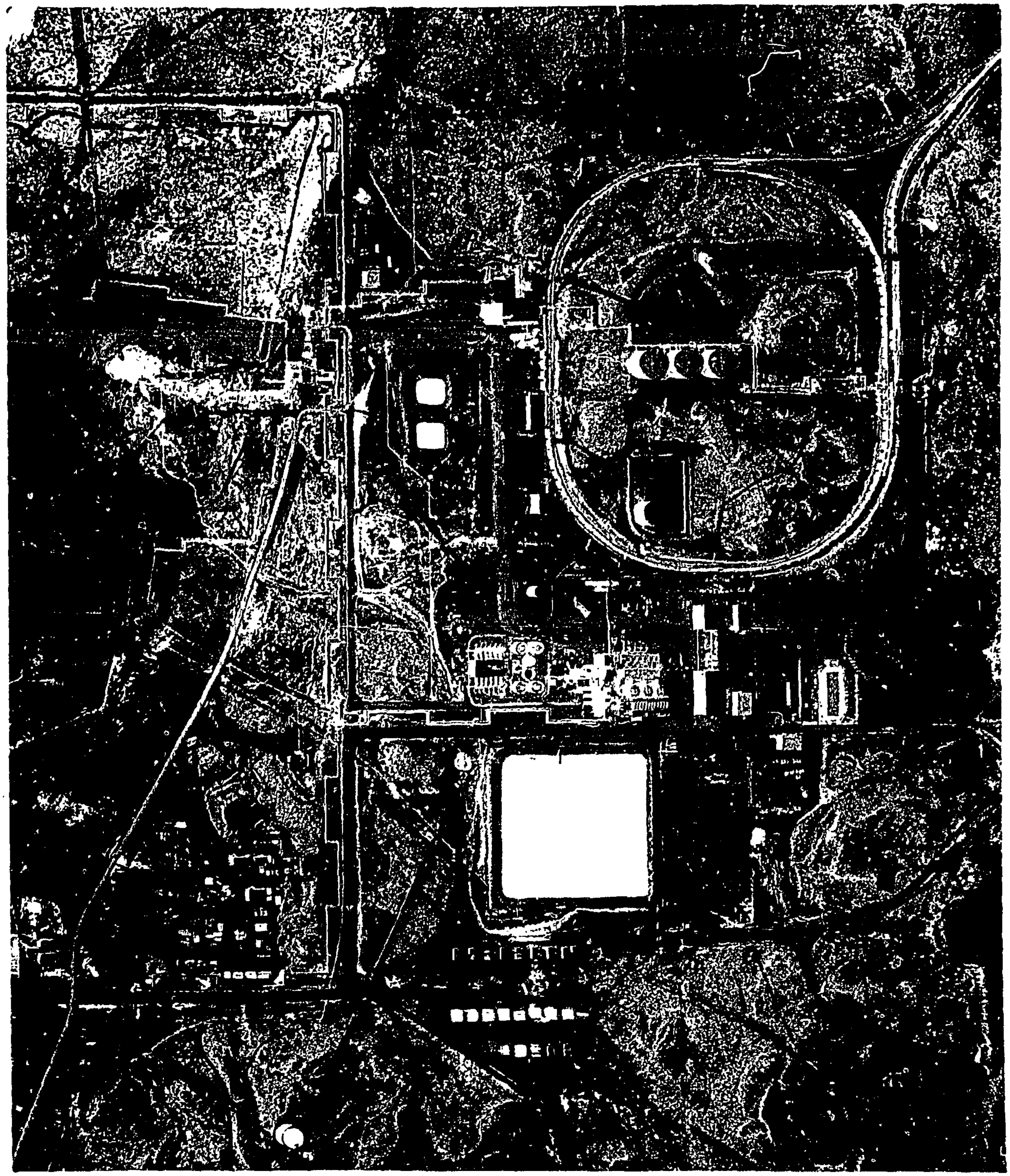


Figure 4.3-7. Aerial View — Duval Corporation Sulfur Mine



4.4. Nuclear Plant Design

The PE-CNSG is an integral, externally pressurized water reactor (PWR), which was originally developed to provide propulsion power for commercial nuclear ships. The core, steam generator, and RC pumps are located within the reactor pressure vessel. The nuclear plant also includes the pressure-suppression containment system. Modifications to the marine CNSG design were made only to meet necessary process requirements.

4.4.1. Power Level Selection

The marine CNSG design, which forms the basis for the PE-CNSG, is rated at 314 MWt. At this power level the CNSG produces 1.57×10^2 kg/s (1.25×10^6 lb/h) total steam flow with conditions of 1.82×10^6 Pa (700 psia) and 281C (538F) with 19.4C (35F) superheat at the steam generator outlet nozzles. Review of several potential process applications at the conclusion of the Phase 1 study[1] indicated that many applications could require more energy than was available. In addition, an increase in power level could make the PE-CNSG more attractive economically when compared with coal. Further studies of the market indicate that a power increase of 15 to 20% would be appropriate. Accordingly, the marine CNSG was evaluated to determine the feasibility of increasing the power level. No technical issues were found that would prohibit increasing the power level approximately 16% to 365 MWt. Therefore, the power level was set at 365 MWt for this study.

4.4.2. PE-CNSG Power Level Evaluation

To verify the conclusion that 365 MWt was technically achievable, a more detailed technical review of design requirements from the preliminary evaluation was conducted. The study provided a more detailed basis for the technical judgement of design capability. The areas studied were (1) adequacy of RC flow with respect to DNBR margin, (2) steam generator secondary side pressure drop, (3) impact of power increase on reactor auxiliaries, plant protection and control, core, steam generator, and transient and safety analyses, (4) compatibility of RC pump material with boric acid in the reactor coolant, (5) impact of wet refueling on containment analysis (potential reduction in containment volume), and (6) required test and evaluation programs. The results of the study are summarized below.

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Reactor Coolant Flow Adequacy — The reference total RCS flow of 3283 kg/s (26.06×10^6 lb/h) was the basis used for determining flow adequacy. The evaluation was conducted for the maximum design conditions at design overpower based on 365 MWt. The results were as follows:

	Metric	English
Core power, MWt	365	365
T_{in}/T_{avg}	300/309C	571.6/589F
Min DNBR (predicted)	1.91	1.91
Min DNBR (design basis)	1.42	1.42
System pressure	1.55×10^7 Pa	2250 psia

For the PE-CNSG at 365 MWt, the reference flow is adequate and provides a satisfactory minimum DNBR of 1.91 and a design minimum DNBR of 1.42.

Steam Generator Secondary Pressure Drop — The major concern in this area is flow stability. Current test and evaluation programs for the CNSG IVA steam generator are expected to demonstrate design adequacy. No additional problems due to increased power level are expected.

Power Increase Impact — Design changes required for PE-CNSG reactor auxiliaries will be in design parameters rather than in design concept. The same is true of plant protection and control systems; i.e., process variables and setpoints may change but the design concept will not.

Phase I studies on the PE-CNSG core indicated capability for increased power, which was verified qualitatively. The modular steam generator is adequate for 365 MWt with only a slight reduction in superheat capability. The qualitative assessment of PE-CNSG transient and safety analyses at 365 MWt indicated that power operation at this level will not adversely impact operating and safety margins. The analytical work performed in other portions of this work effort provides additional verification (section 5.2.2).

Materials Compatability — The reference, wet-rotor RC pump was designed for service in the CNSG IVA for maritime service in which chemical shim reactivity control (boric acid) is not used. In the PE-CNSG at 365 MWt, chemical shim reactivity control is used; therefore, the materials of the RC pump must be compatible with boric acid in solution in the reactor coolant. A materials investigation revealed that the reference pump materials had been changed for compatibility since boric acid is used in the CNSG IVA reactor coolant during refueling outages. Therefore, the PE-CNSG will utilize the reference pumps.

Wet Refueling Impact — The influence of wet refueling on the PE-CNSG was in the potential for reduction in containment dry well air volume, which would tend to increase containment design pressure. The plant arrangement for refueling was reviewed to assess the magnitude of the problem. This concern proved groundless, since no reduction in dry well volume was found.

Test and Evaluation Programs — While much analytical work remains to fully develop the PE-CNSG at 365 MWt, no major problem areas in the design are expected. However, the fabrication development and the test and evaluation programs for the maritime CNSG may reveal additional program needs for the PE-CNSG. Several additional test programs specific to the PE-CNSG were identified during the present design evaluation. The test and evaluation programs necessary are as follows:

1. FA prototype detail design and fabrication — fabrication modifications to the B&W Mark B FA for use in the CRDM drop time test.
2. VMFT — testing with modified reactor internals using M-CNSG test facilities.
3. CRDM prototype test — modifications to the M-CNSG or central power station CRDM require prototype testing.
4. CRDM life test — Only the drop time portion of the life testing is necessary.

Based on the foregoing evaluations of the PE-CNSG, there are apparently no technical obstacles in achieving design conditions at 365 MWt. If the process supplied with energy requires special design considerations or detailed studies indicate the need, there is flexibility in the design to modify design parameters to some extent.

4.4.3. PE-CNSG Arrangement

The CNSG for process energy application (PE-CNSG) is based on the marine nuclear propulsion system that has evolved in response to an increasing emphasis on size and weight reduction and reactor safety. The CNSG has been designed utilizing years of experience beginning in the late 1950's when B&W began its activities in the marine nuclear propulsion field with the NS Savannah program. Design modifications have been made as a result of recent design reviews by both governmental and industrial groups.

The RCS flow path has been designed to minimize the existence of complex flow obstructions. Vessel model flow tests will verify inlet core flow distribution. The straight-tube OTSG design incorporates proven features of the B&W utility PWR steam generators. The straight-tube design greatly simplifies the problem of in-service inspection as required by the USNRC Regulatory Guide 1.83, i.e., eddy current inspection of steam generator tubing.

The PE-CNSG is an integral PWR system in which the core, steam generator, and RC pumps are located within the reactor pressure vessel (Figure 4.4-1). An electrically heated pressurizer is externally connected to the pressure vessel. The steam generator comprises 12 modular once-through units located above the top level of the core in the annulus between the core and pressure vessel. Each steam generator module can be isolated in the unlikely event of tube failure. A steam generator module incorporates counterflow heat transfer with shell-side boiling to produce steam at a constant pressure. The RCS operates at a constant average temperature over the load range from 20 to 100% of rated power and a varying average temperature from 0 to 20% of load.

The reactor core consists of 57 FAs with Zircaloy-4 tubes containing slightly enriched UO_2 pellets enclosed by welded end plugs. The tubes are supported in assemblies by a spring-clip grid structure. The 17 control rods, which control reactor power, are clusters of neutron absorber rods that move in guide tubes within the FAs.

During operation the reactor coolant is pumped downward through the steam generator tubes where the coolant transfers heat to the secondary side feedwater, thereby producing superheated steam. Leaving the steam generator modules, the coolant flows downward over mixing vanes and then turns at the bottom of the reactor vessel. Heat generated by the nuclear fuel raises the coolant temperature as it passes upward through the core. The coolant continues to flow upward until it reaches the RCP suctions.

The RCS pressure boundary is designed and fabricated in accordance with the codes listed in Table 4.4-1. The design considers steady-state and transient conditions expected to occur during the design life including preoperational testing, normal operation, abnormal operation, and postulated accident conditions.

The auxiliary service requirements and engineered safety features of the PE-CNSG plant have been evaluated and combined for maximum system simplification consistent with reliability, operability, safety, and function. All auxiliary systems are based on current PWR technology and consideration of duty and environment. All systems and components have been designed in accordance with nuclear service codes and standards or their equivalent. Components are selected and systems arranged to meet operational safety requirements.

4.4.4. PE-CNSG Parameters

The PE-CNSG is designed to produce 1.83×10^2 kg/s (1.45×10^6 lb/h) total steam flow with conditions of 1.82×10^6 Pa (700 psia) and 281C (538F) with a small degree of superheat at the steam generator outlet nozzles. Table 4.4-2 lists the major design parameters.

Table 4.4-1. Reactor Coolant System Pressure Boundary Codes and Classifications

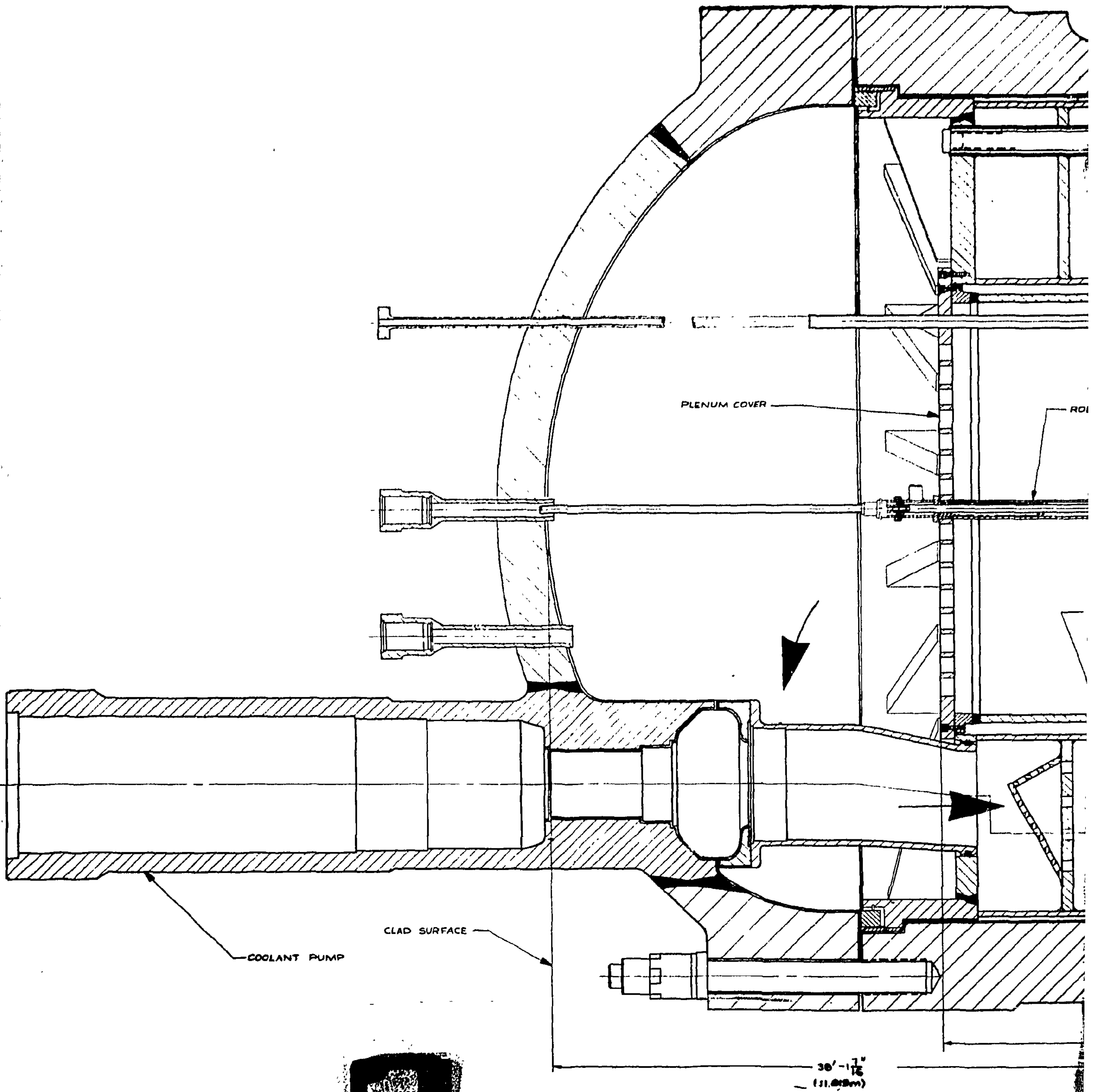
Component	Code	Classification	USNRC quality group
Reactor vessel	ASME III	Class 1	A
Steam generator	ASME III	Class 1	A
Pressurizer	ASME III	Class 1	A
Reactor coolant pump	ASME III	Class 1	A
Piping	ASME III	Class 1	A
Valves	ASME III	Class 1	A
CR drive motor tubes	ASME III	Class 1	A

Table 4.4-2. Reactor Coolant System Parameters

	Metric	English
Design[a] Performance Summary		
Power	365 MWt	365 MWt
Steam pressure at SG outlet	4.83 MPa	700 psia
Steam temp at SG outlet	281C	538F
Superheat	19.4C	35F
Steam flow	183 kg/s	1.45×10^6 lb/h
Feedwater inlet temp	204C	400F
Feedwater inlet pressure	5.17 MPa	750 psig
Feedwater inlet enthalpy	0.874 MJ/kg	376 Btu/lb
Nominal core inlet temp	300C	571.6F
Nominal core outlet temp	319C	606.4F
Reactor vessel avg temp	309.4C	589F
RCS flow	3289 kg/s	26.06×10^6 lb/h
Equipment Data/Design Performance Data		
No. of SG modules	12	12
RCS total primary volume	106.9 m^3	3775 ft^3
Primary water volume	99.8 m^3	3525 ft^3
Pressurizer gas volume	7.08 m^3	250 ft^3
Reactor vessel ID	3.99 m	157 in.
No. of CRAs	17	17
No. of fuel assemblies	57	57
RC pump flow (four used)	1.196 m^3/s	18,950 gpm
RC pump head	32.31 m	106 ft
RC pump expected power, hot	0.326 MW	437 hp
Pressurizer[b]		
Overall length	8.969 m	29 ft, 5.125 in.
Shell OD	2.013 m	79.25 in.

[a] Steam conditions are as per feedwater temperatures listed. Any other feedwater temperature selected by the Purchaser may not be applied without further evaluation.

[b] All dimensions nominal.


PLENUM COVER
COOLANT PUMP
CLAD SURFACE
38'-1 7/16"

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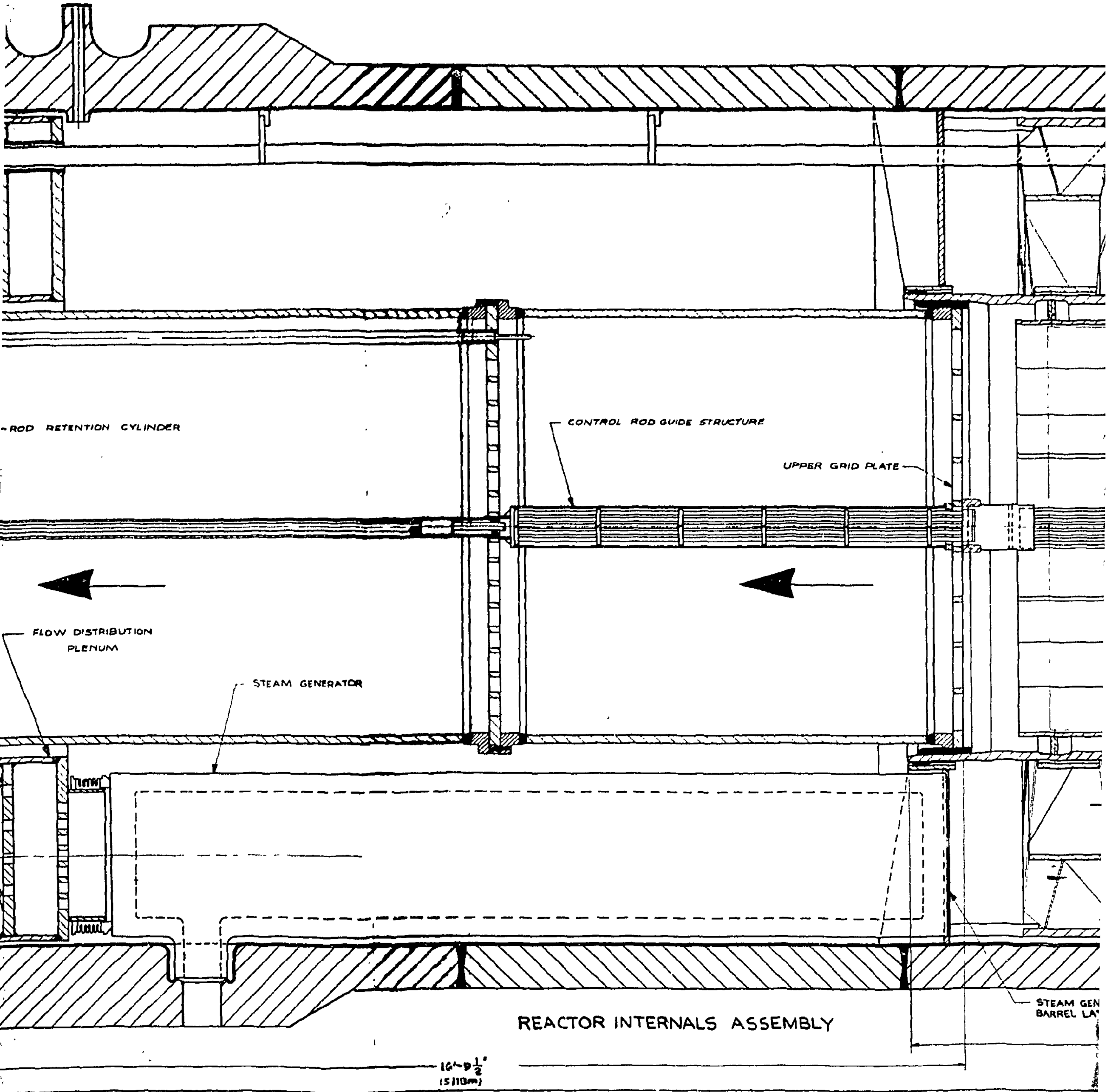
ROD RETENTION CYLINDER
CONTROL ROD GUIDE STRUCTURE
UPPER GRID PLATE
FLOW DISTRIBUTION PLENUM
STEAM GENERATOR
REACTOR INTERNALS ASSEMBLY
16'-9 1/2"
(5.113m)

2

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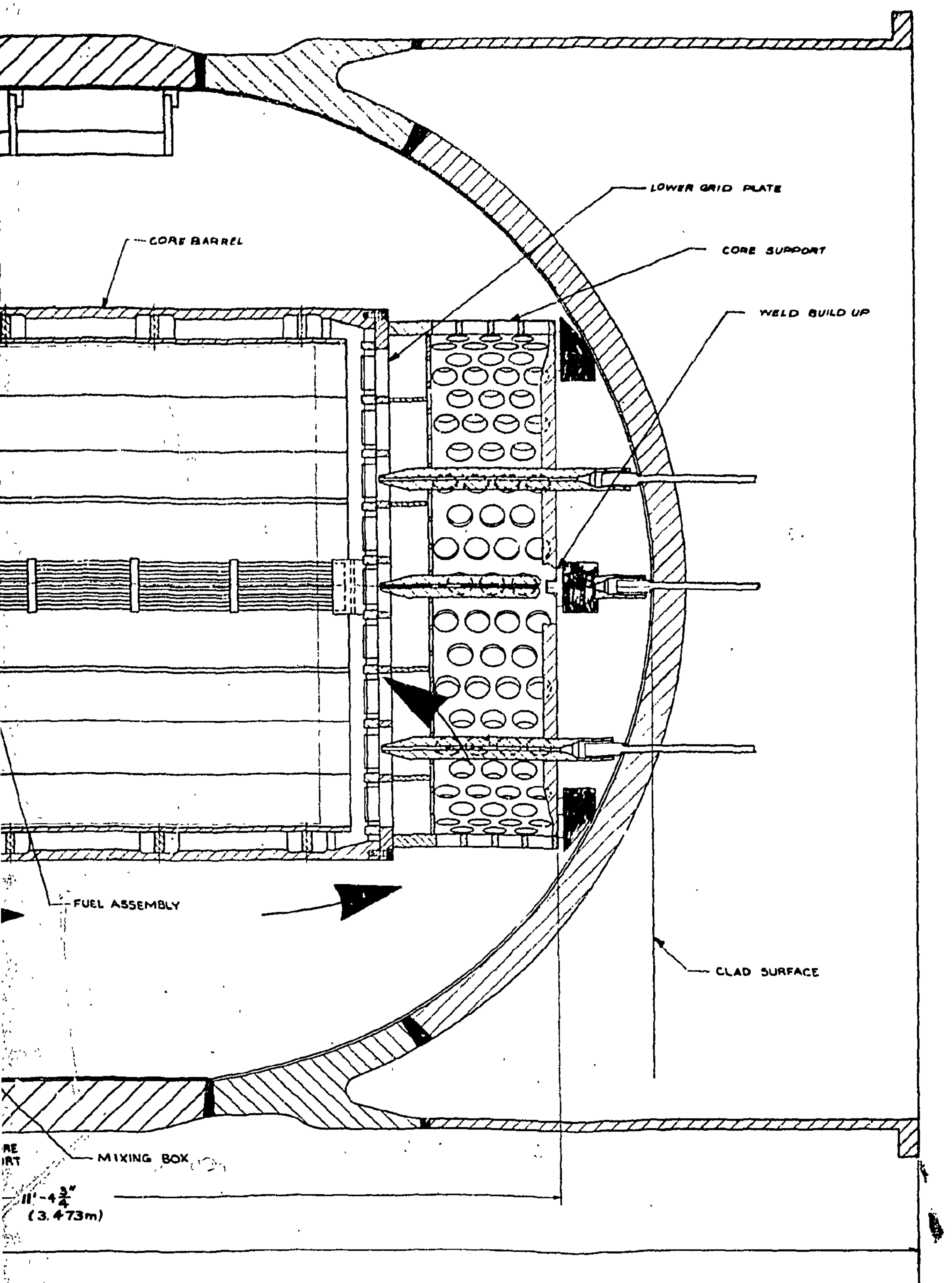
Figure 4.4-2. Elevation View of the CNSG



4.5. Balance-of-Plant Design

The BOP design for this study was based on the land-based PE-CNSG plant concept developed by B&W in conjunction with the UE&C as described in BAW-1428.[1] For that study the hypothetical site chosen was the standard Middletown site described in NUS-531.[2] The PE-CNSG for that study was rated at 313 MWt with two separate conceptual BOP designs — production of 137 kg/s (1,090,000 lb/h) of process steam, and 91 MW of electricity in the other case. The intent of this study was to modify the conceptual design to meet the particular conditions of the Duval site and the augmented power level of 365 MWt.

4.5.1. Plant Layout

The plant layout for the PE-CNSG is shown in Figure 4.3-5. The plant has a reactor service building containing the CNSG, its containment and all supporting nuclear auxiliary systems, a control building, a diesel generator building, and an administration building. The reactor service, control, and diesel generator buildings (seismic Class 1 structures) are designed to withstand tornado-generated missiles.

The administration and process heat service buildings provide space for offices, change rooms, maintenance shop, and spare-parts storage. These structures are standard industrial structures.

All facilities are located in two main sections with a piping tunnel between the sections allowing the installation of an access road through the middle of the plant layout. The access road permits easy movement of equipment and personnel around the site.

The reactor containment vessel is located inside the reactor service building (see Figure 4.5-1). Normal personnel access to the containment is through an air lock at the 14-m (46-ft) elevation. For refueling access the top of the containment is opened at an elevation of 20 m (66 ft). Access to the area under the reactor vessel is provided by an access tunnel and bolted containment closure.

The diesel fuel oil storage tank is positioned so that it has minimal exposure to potential plant accidents. The tank is enclosed in a concrete bunker so that in the event of an oil spill and/or oil fire, the spill/fire can be contained.

Very large equipment such as large tanks (borated water storage tanks, demineralized water storage tank, condensate storage tank) and plant electrical equipment (transformers) are located outside the main buildings, thus minimizing the size of these buildings. Minimal weather protection is provided for this outside equipment.

4.5.2. Systems Layout

The layout of systems in the reactor service building is based on past experience with large nuclear power plants. The location of major equipment is shown in Figures 4.5-1 through 4.5-6. Shielded cubicles are provided for each piece of equipment that contains radioactivity. Plant structural layouts are designed to minimize construction interferences and to shorten construction schedules.

The following systems are housed in the reactor service building:

1. Chemical addition and boron recovery.
2. Heating, ventilation, and air conditioning.
3. Reactor compartment ventilation.
4. Post-LOCA (loss-of-coolant accident) combustible gas.
5. Reactor plant water supply.
6. Pressurizer spray.
7. Containment dry well cooling.
8. Emergency DH removal.
9. Decay heat removal.
10. Makeup and purification.
11. Equipment and floor drainage.
12. Sampling.
13. Component cooling water.
14. Waste management.
15. Fuel storage and handling (wet refueling is used).

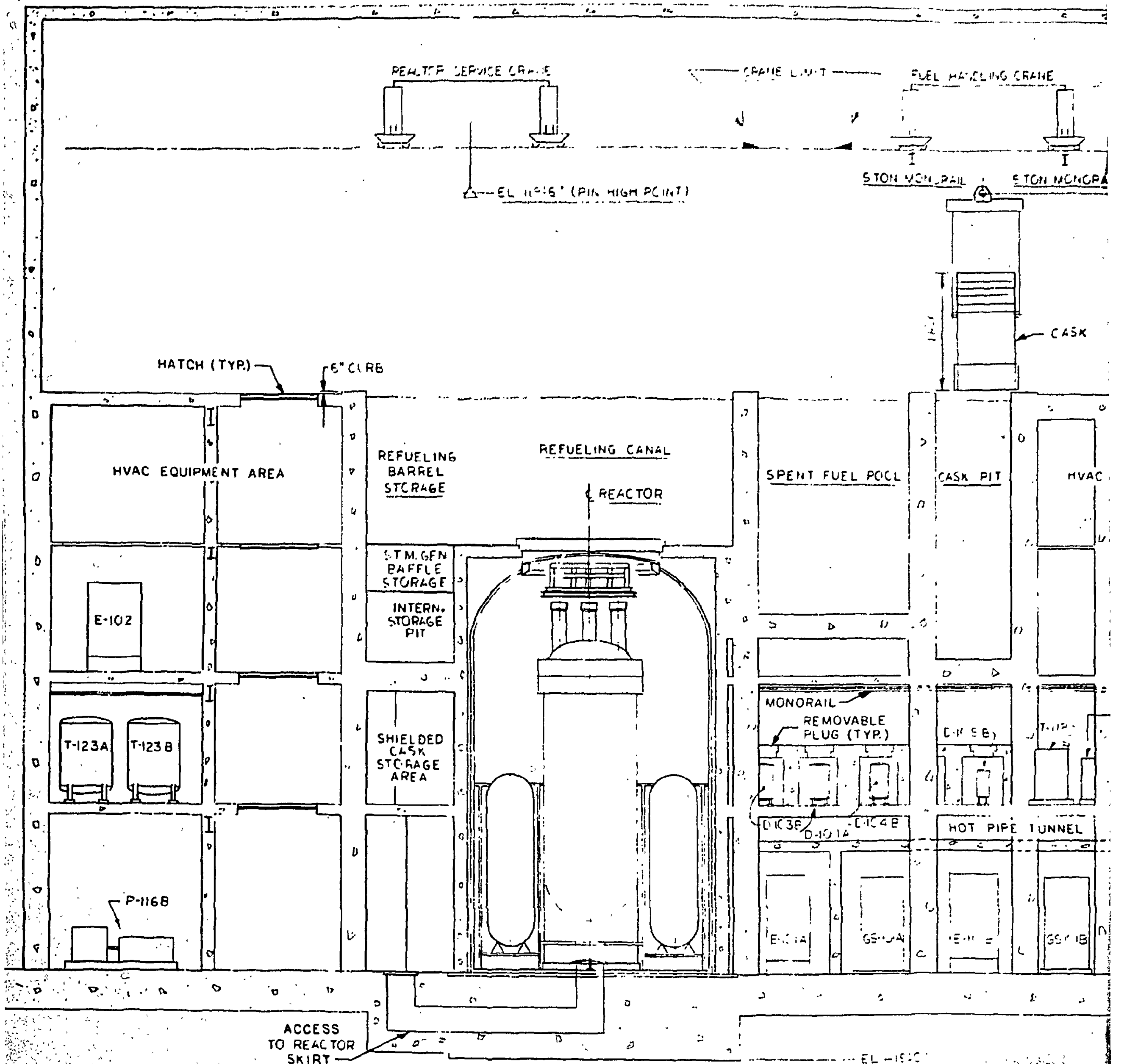
4.5.3. BOP Modifications

Some minor changes in the containment have been required, but (as a whole) the reactor containment, the reactor service building, the control room, and many other BOP features have remained the same. The primary area of changes was in the development of the process energy delivery system (tertiary loop). For the Middletown site two separate plants were addressed to produce steam and electricity. For the Duval site a mixture of steam for existing turbine generators, service requirements, and hot water for mine use is necessary.

The design modifications for the Duval site are further restricted by two site constraints. Because the site is both arid and remote, special consideration is required to provide adequate water (quantity and quality) and electricity. To meet these constraints, the BOP design includes a water treatment facility to provide water at the desired purity level. Further, the design of an electrical supply system from a utility source approximately 14 km (9 mi) from the site was necessary.

4.5.4. Process Steam System

Development of the process steam system is discussed in section 4.6. Figure 4.5-7 is the plant layout showing the tertiary loop connections to the existing facilities. Figure 4.5-8 shows the process building layout.


REACTOR SERVICE CRANE
CRANE LIMIT
FUEL HANDLING CRANE
5 TON MONORAIL
HATCH (TYP.)
6" CURB
CASK
HVAC EQUIPMENT AREA
REFUELING BARREL STORAGE
REFUELING CANAL
℄ REACTOR
SPENT FUEL POOL
CASK PIT
HVAC
STM. GEN BAFFLE STORAGE
INTERN. STORAGE PIT
E-102
T-123A
T-123B
SHIELDED CASK STORAGE AREA
MONORAIL
REMOVABLE PLUG (TYP.)
HOT PIPE TUNNEL
P-1168
ACCESS TO REACTOR SKIRT

SECTION Z-Z"

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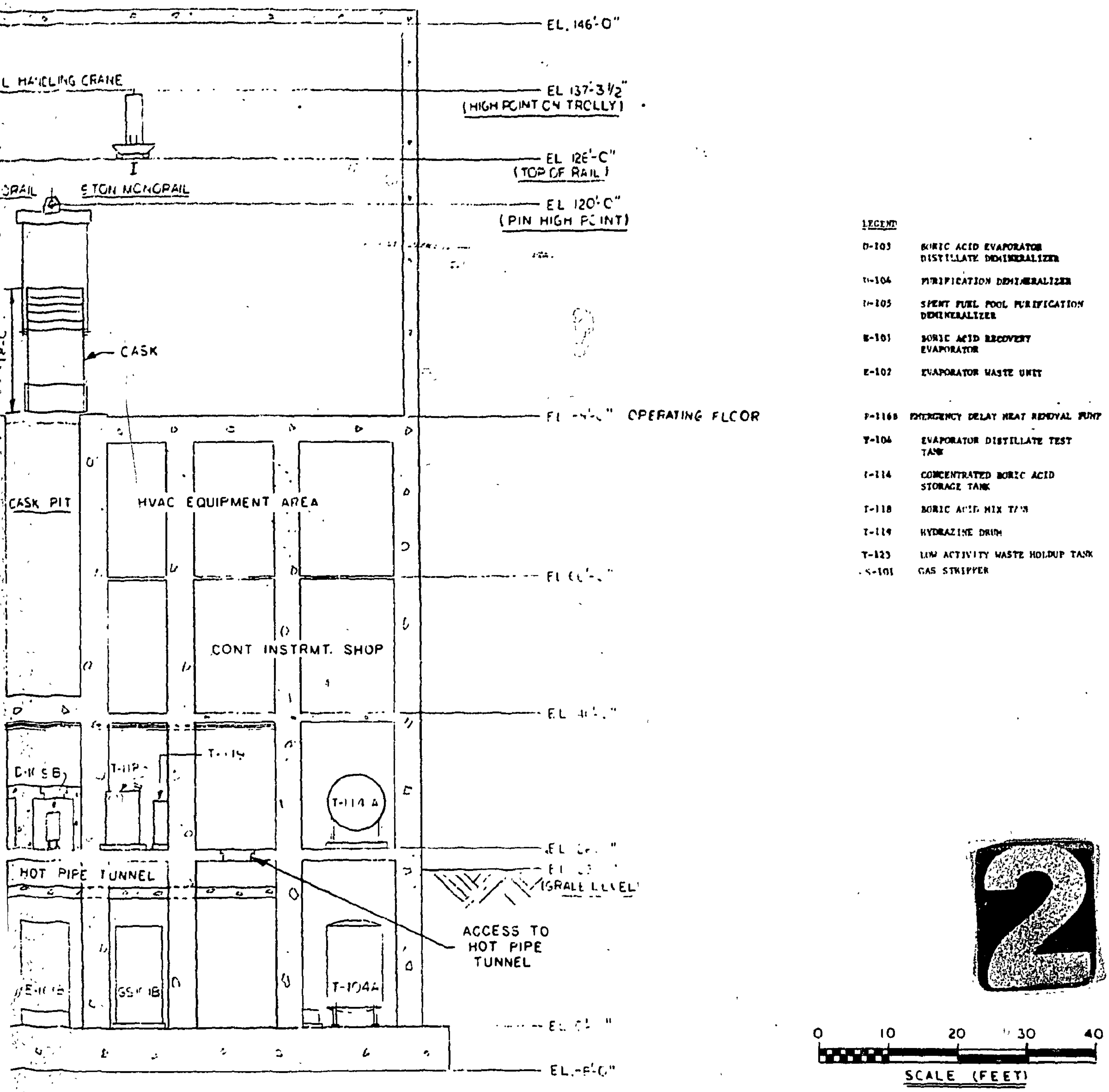
Figure 4.5-1. Reactor Service Building, General Arrangement — Section View

(UE&C 6255.005-D-008)



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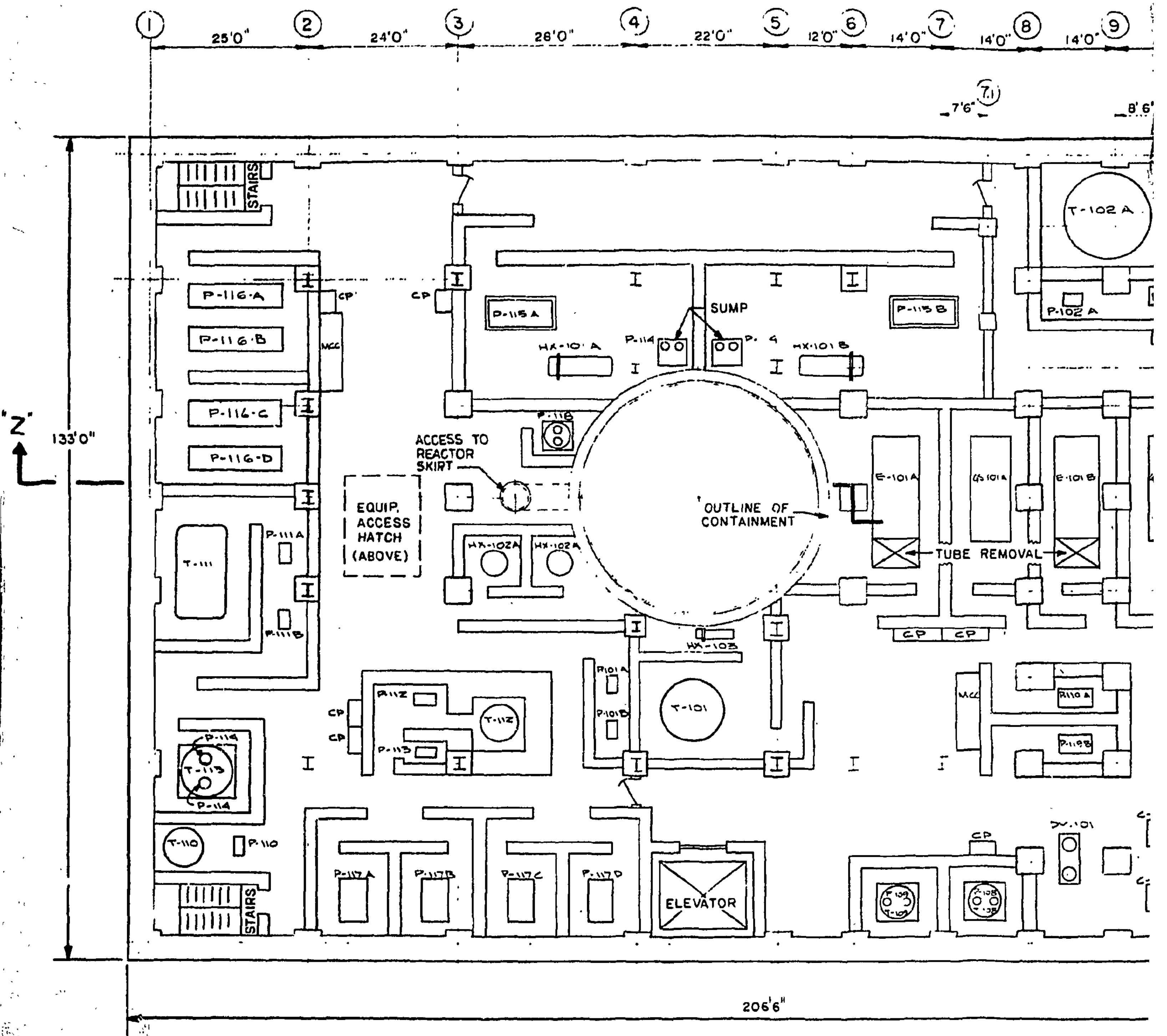

25'0"
24'0"
26'0"
22'0"
12'0"
14'0"
14'0"
14'0"
7'6"
8'6"
133'0"
206'6"
"Z"
STAIRS
P-116-A
P-116-B
P-116-C
P-116-D
MCC
CP
P-115 A
P-115 B
SUMP
HX-101 B
T-102 A
P-102 A
ACCESS TO REACTOR SKIRT
EQUIP. ACCESS HATCH (ABOVE)
OUTLINE OF CONTAINMENT
TUBE REMOVAL
E-101 A
E-101 B
T-111
P-111 A
P-111 B
HX-102 A
HX-103
T-101
T-112
T-113
P-114
T-110
P-110
P-117 A
P-117 B
P-117 C
P-117 D
ELEVATOR

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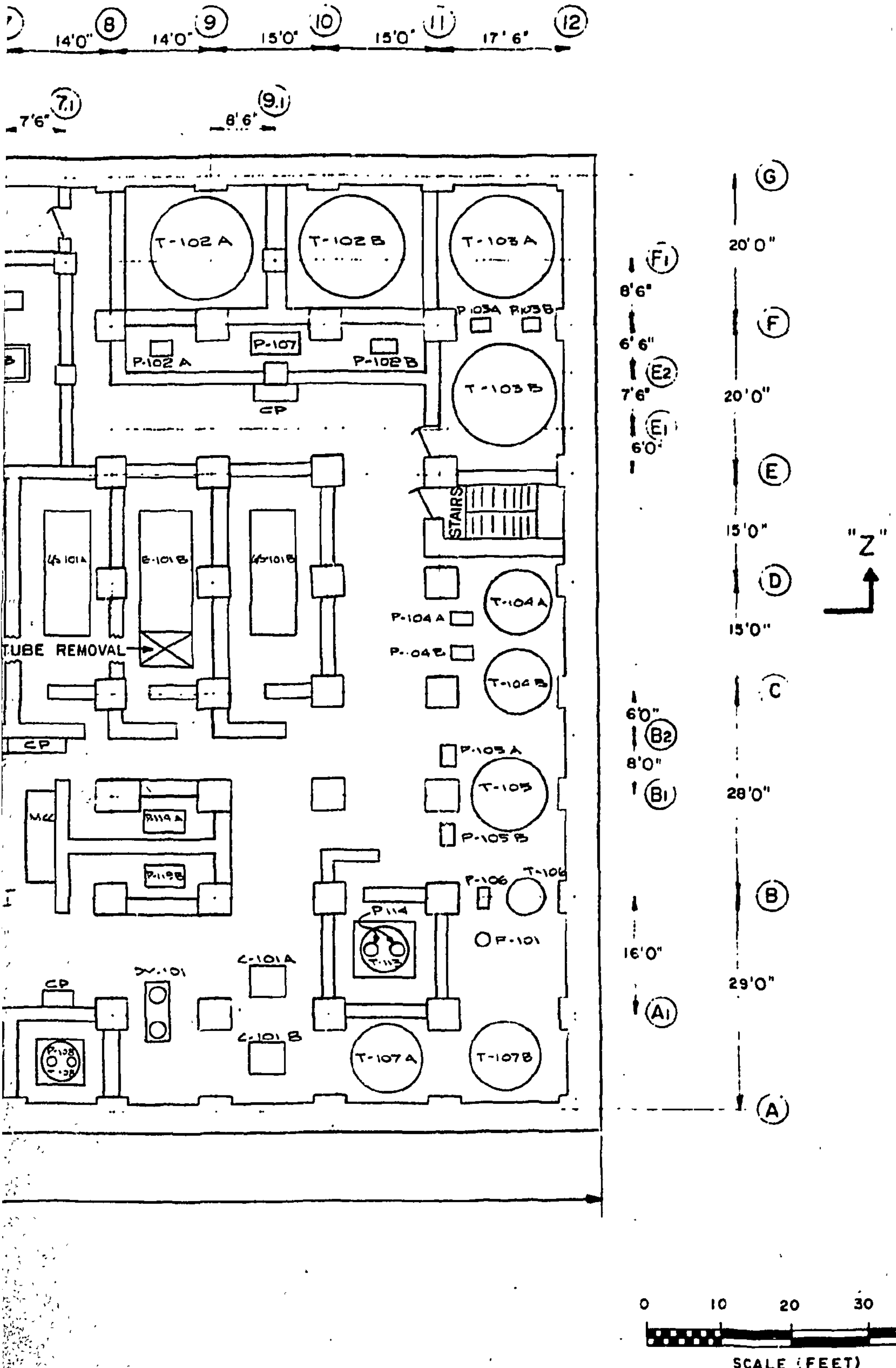
Figure 4.5-2. Reactor Service Building, Plan at Elevation 0'0"

(UE&C 6255.005-D-003)



LEGEND

C-101	NUCLEAR PLANT INSTRUMENT AIR COMPRESSOR
CP	CONTROL PANEL
DY-101	NUCLEAR PLANT INSTRUMENT AIR DRYER
E-101	BORIC ACID RECOVERY EVAPORATOR
F-101	CASK DECONTAMINATION DRAIN COLLECTOR FILTER
GS-101	DEGASIFIER
HX-101	DECAY HEAT REMOVAL HEAT EXCHANGER
HX-102	LETDOWN COOLER
MCC	MOTOR CONTROL CENTER
P-101	REACTOR COOLANT DRAIN TANK PUMP
P-102	REACTOR COOLANT BLEED EVAPORATOR FEED PUMP
P-103	REACTOR COOLANT DISTILLATE TRANSFER PUMP
P-104	EVAPORATOR DISTILLATE TEST TANK PUMP
P-105	PURIFICATION PUMP
P-106	CASK DECONTAMINATION DRAIN PUMP
P-107	REACTOR COOLANT BLEED RECIRCULATING PUMP
P-108	LOW ACTIVITY SUMP PUMP
P-109	HIGH ACTIVITY SUMP PUMP
P-110	LAUNDRY & HOT SHOWER TANK PUMP
P-111	DEMINERALIZER FLUSH PUMP
P-112	[illegible] TRANSFER PUMP
P-113	SPENT RESIN SLUICE PUMP
P-114	EQUIPMENT & FLOOR DRAINS PUMP, SUMP
P-115	DECAY HEAT REMOVAL PUMP
P-116	EMERGENCY DECAY HEAT REMOVAL PUMP (AUX. FEEDWATER PUMP)
P-117	MAKEUP PUMP
P-118	COMPONENT SUMP PUMP
P-119	PRESSURIZER SPRAY PUMP
T-101	REACTOR COOLANT DRAIN TANK
T-102	REACTOR COOLANT BLEED HOLDUP TANK
T-103	REACTOR COOLANT DISTILLATE STORAGE TANK
T-104	EVAPORATOR DISTILLATE TEST TANK
T-105	SPENT FUEL POOL SURGE TANK
T-106	CASK DECONTAMINATION DRAIN COLLECTION TANK
T-107	NUCLEAR PLANT INSTRUMENT AIR RECEIVER
T-108	LOW ACTIVITY SUMP TANK
T-109	HIGH ACTIVITY SUMP TANK
T-110	LAUNDRY & HOT SHOWER DRAIN TANK
T-111	DEMINERALIZER FLUSH TANK
T-112	SPENT RESIN STORAGE TANK
T-113	EQUIPMENT & FLOOR DRAINS COLLECTION TANK

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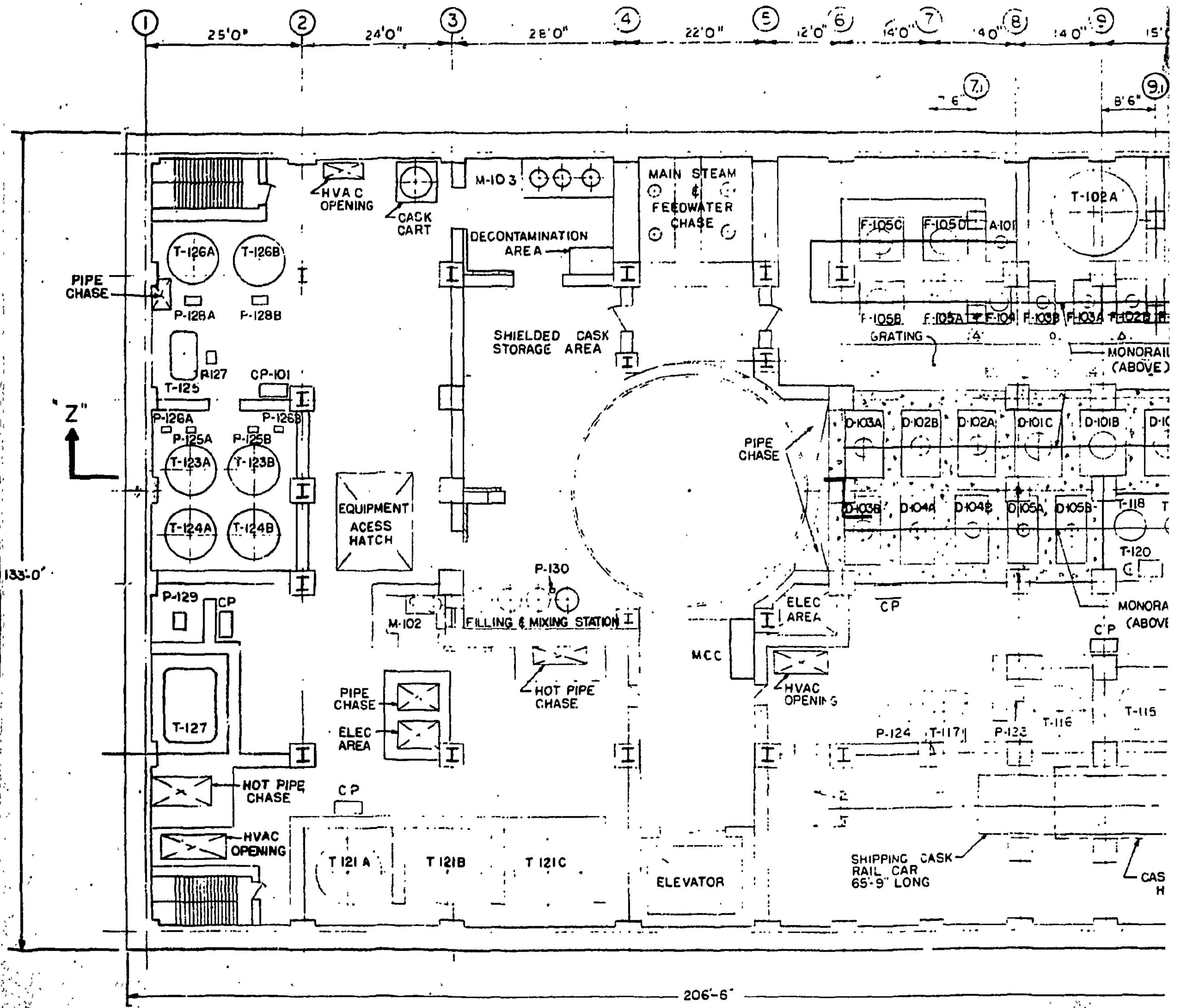

25'0"
24'0"
28'0"
22'0"
12'0"
14'0"
14'0"
14'0"
8'6"
HVAC OPENING
CASK CART
M-103
MAIN STEAM & FEEDWATER CHASE
T-102A
F-105C
F-105D
A-101
DECONTAMINATION AREA
T-126A
T-126B
PIPE CHASE
P-128A
P-128B
F-105B
F-105A
F-104
F-103B
F-103A
SHIELDED CASK STORAGE AREA
GRATING
MONORAIL (ABOVE)
P127
CP-101
T-125
"Z"
P-126A
P-126B
P-125A
P-125B
T-123A
T-123B
D-103A
D-102B
D-102A
D-101C
D-101B
PIPE CHASE
EQUIPMENT ACESS HATCH
T-124A
T-124B
D-103B
D-104A
D-104B
D-105A
D-105B
T-118
T-120
133'-0"
P-130
P-129
CP
M-102
FILLING & MIXING STATION
ELEC AREA
CP
MCC
HOT PIPE CHASE
HVAC OPENING
PIPE CHASE
ELEC AREA
T-127
P-124
T-117
P-123
T-116
T-115
HOT PIPE CHASE
CP
HVAC OPENING
T 121 A
T 121B
T 121C
ELEVATOR
SHIPPING CASK RAIL CAR 65'-9" LONG
206'-6"

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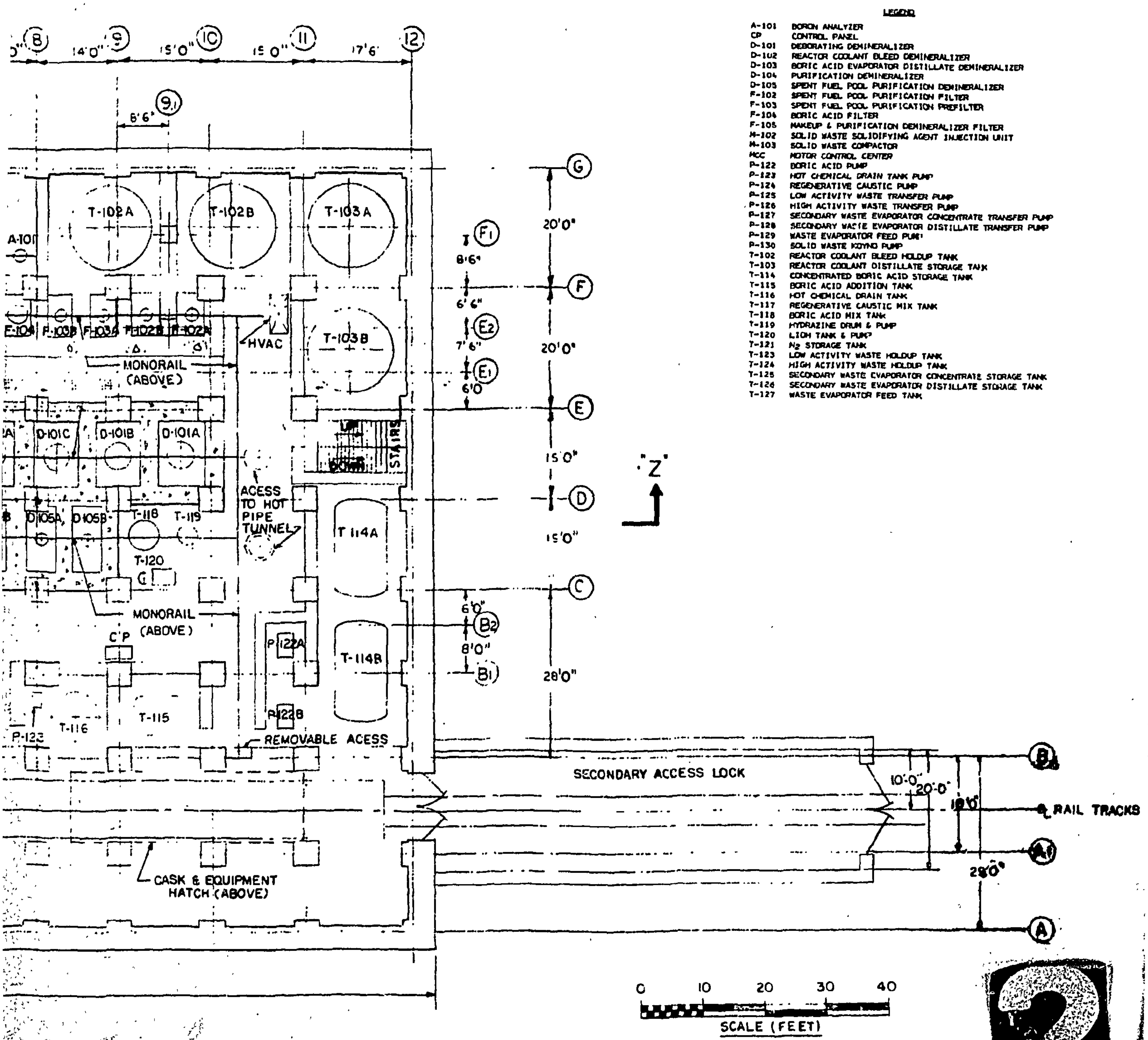
Figure 4.5-3. Reactor Service Building, Plan at Elevation 26'0"

(UE&C 6255.005-D-004)



Babcock & Wilcox

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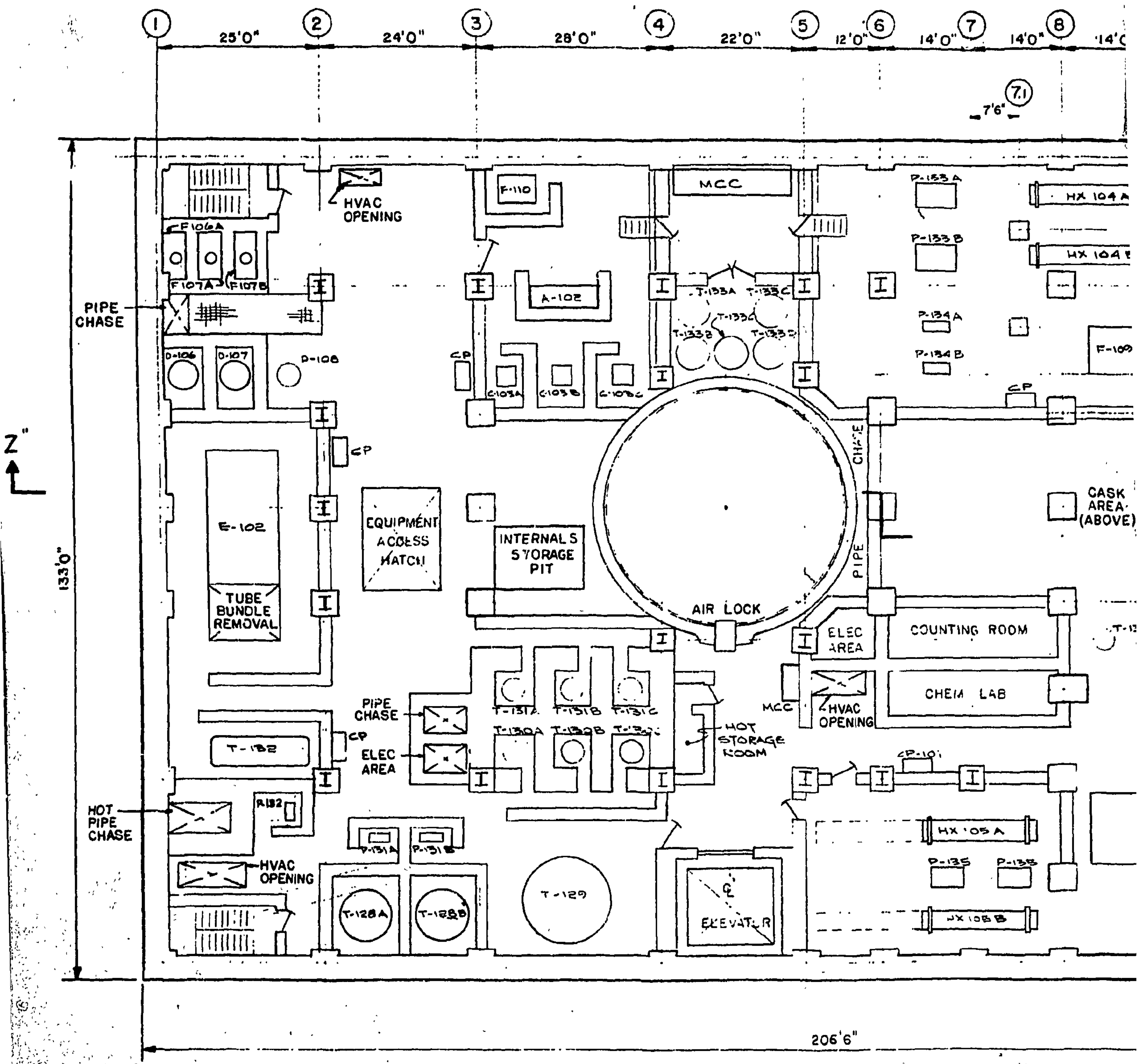

25'0"
24'0"
28'0"
22'0"
12'0"
14'0"
14'0"
7'6"
133'0"
206'6"
HVAC OPENING
MCC
F-110
A-102
PIPE CHASE
F106A
F107A
F107B
D-106
D-107
P-108
CP
C-103A
C-103B
C-103C
T-133A
T-133C
T-133B
T-133D
P-133A
P-133B
P-134A
P-134B
HX 104A
HX 104B
F-109
E-102
TUBE BUNDLE REMOVAL
EQUIPMENT ACCESS HATCH
INTERNALS STORAGE PIT
AIR LOCK
PIPE CHASE
CASK AREA (ABOVE)
ELEC AREA
COUNTING ROOM
CHEM LAB
MCC
HVAC OPENING
HOT STORAGE ROOM
PIPE CHASE
ELEC AREA
T-131A
T-131B
T-131C
T-130A
T-130B
T-132
HOT PIPE CHASE
R132
HVAC OPENING
P-131A
P-131B
T-128A
T-128B
T-129
ELEVATOR
CP-101
HX 105A
P-135
HX 105B

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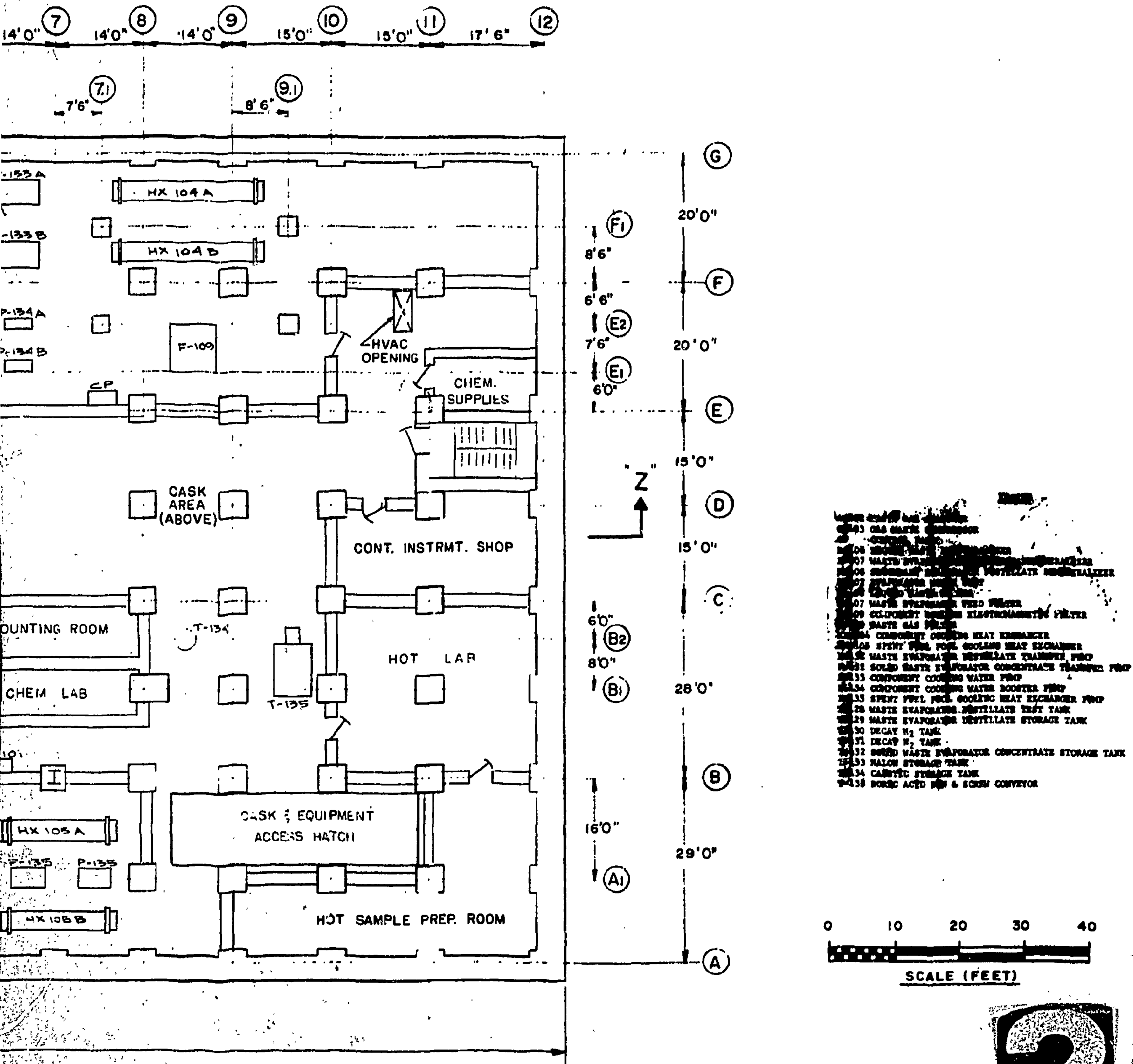
Figure 4.5-4. Reactor Service Building, Plan at Elevation 46'0"

(UE&C 6255.005-D-005)



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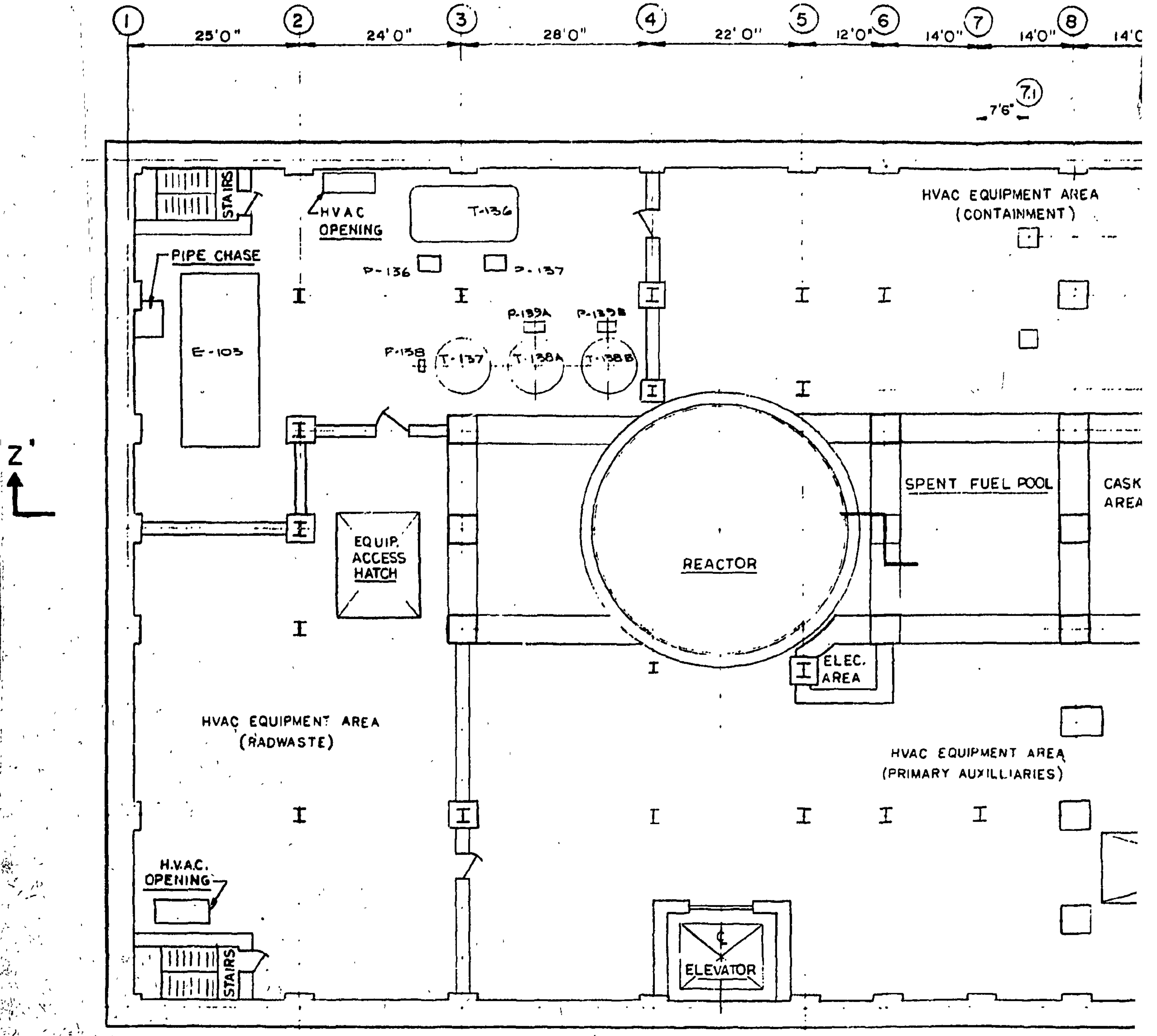

1
2
3
4
5
6
7
8
25'0"
24'0"
28'0"
22'0"
12'0"
14'0"
14'0"
7.1
7'6"
HVAC EQUIPMENT AREA
(CONTAINMENT)
STAIRS
HVAC
OPENING
T-136
PIPE CHASE
P-136
P-137
P-139A
P-139B
P-138
T-137
T-138A
T-138B
E-105
Z'
SPENT FUEL POOL
CASK
AREA
EQUIP.
ACCESS
HATCH
REACTOR
ELEC.
AREA
HVAC EQUIPMENT AREA
(RADWASTE)
HVAC EQUIPMENT AREA
(PRIMARY AUXILLIARIES)
H.V.A.C.
OPENING
STAIRS
ELEVATOR

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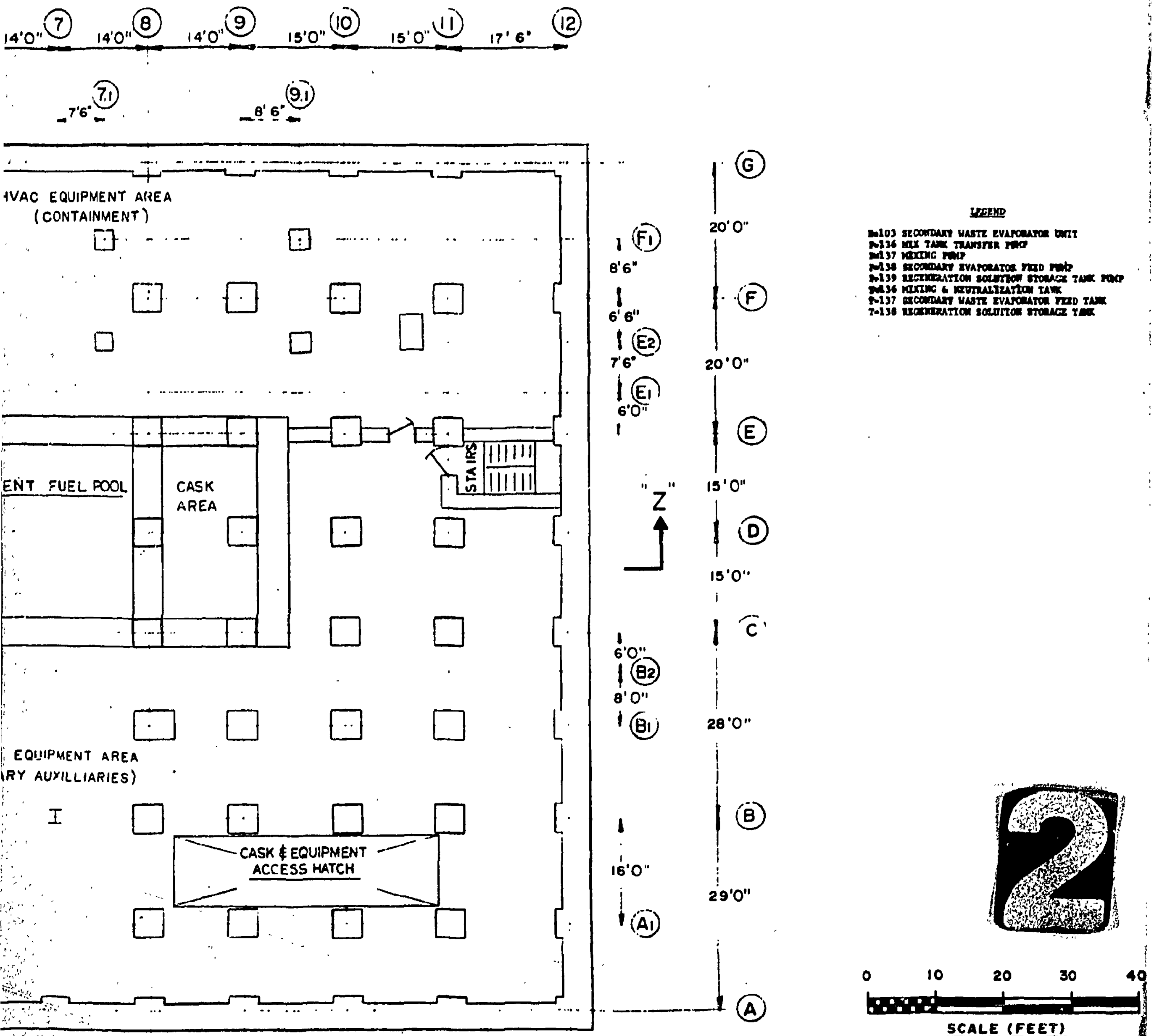
Figure 4.5-5. Reactor Service Building, Plan at Elevation 66'0"

(UE&C 6255.005-D-006)



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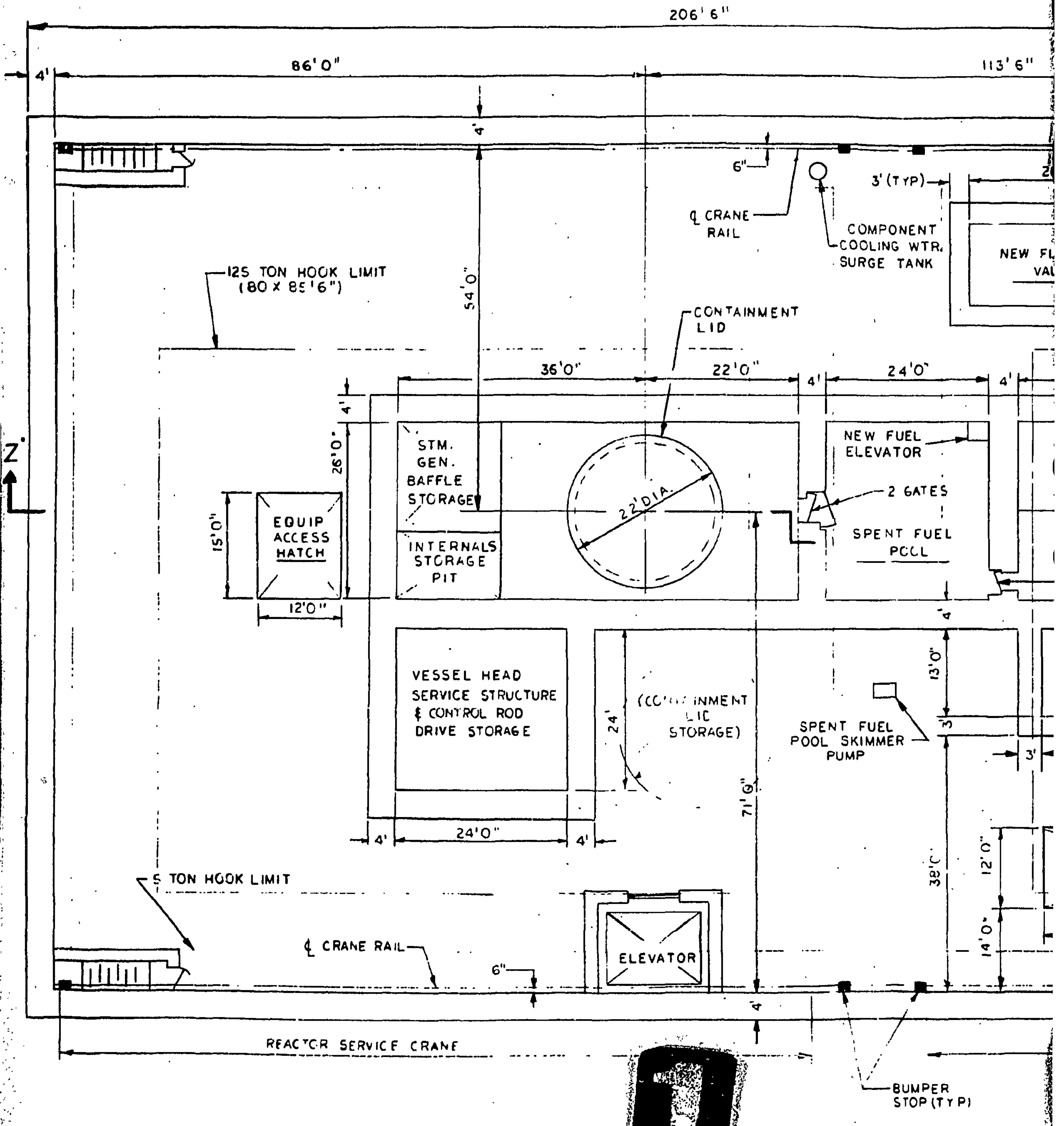

206' 6"
4'
86' 0"
113' 6"
4'
6"
℄ CRANE RAIL
COMPONENT COOLING WTR. SURGE TANK
3' (TYP)
NEW FU
VAU
125 TON HOOK LIMIT (80 X 85'6")
54'0"
CONTAINMENT LID
36'0"
22'0"
4'
24'0"
4'
4'
26'0"
STM. GEN. BAFFLE STORAGE
NEW FUEL ELEVATOR
2 GATES
Z
15'0"
EQUIP ACCESS HATCH
22' DIA.
SPENT FUEL POOL
INTERNALS STORAGE PIT
12'0"
4'
VESSEL HEAD SERVICE STRUCTURE ℄ CONTROL ROD DRIVE STORAGE
24'
STORAGE)
13'0"
SPENT FUEL POOL SKIMMER PUMP
3'
3'
71'0"
4'
24'0"
4'
38'0"
12'0"
5 TON HOOK LIMIT
14'0"
℄ CRANE RAIL
6"
ELEVATOR
4'
REACTOR SERVICE CRANE
BUMPER STOP (TYP)

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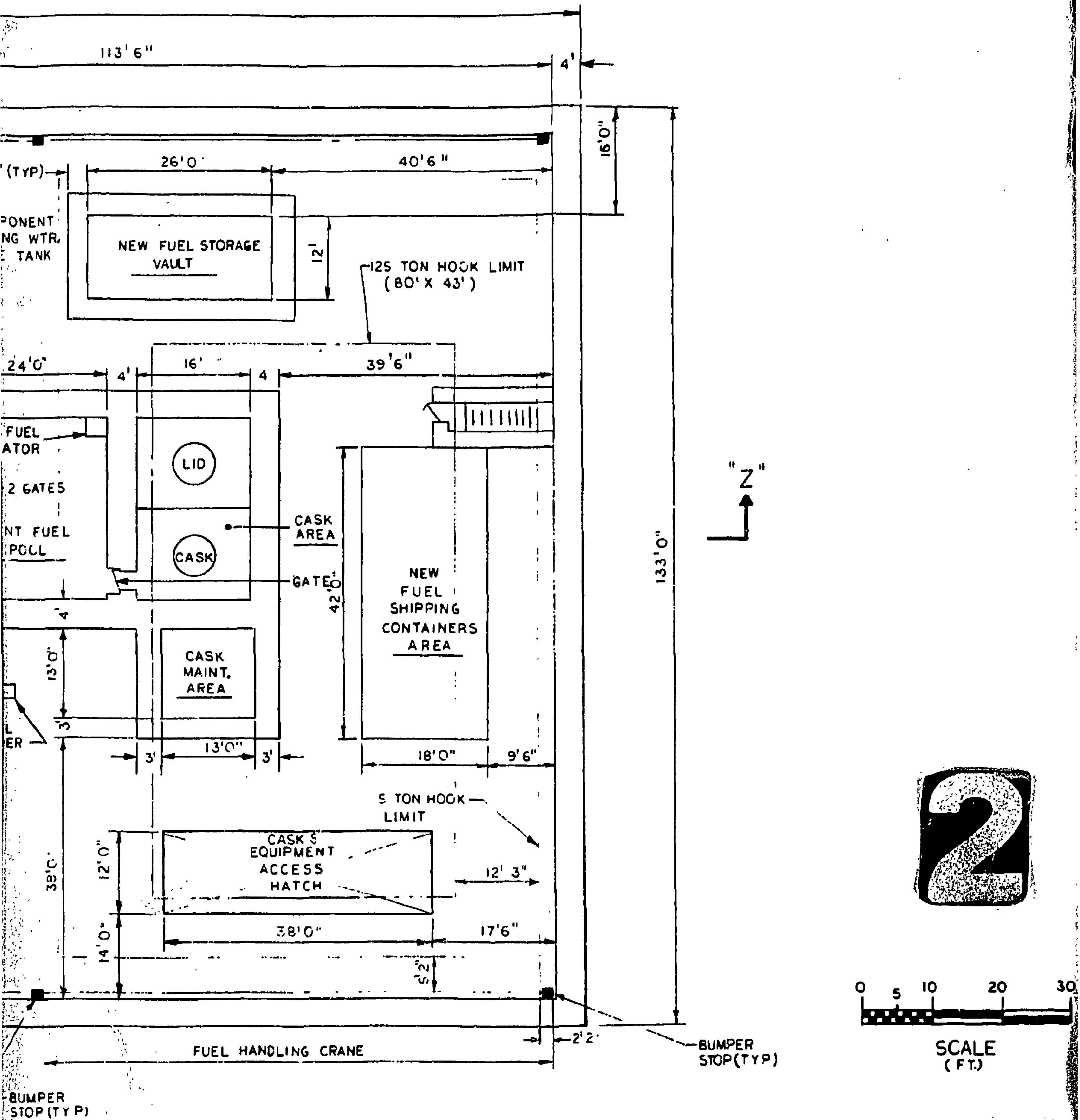
Figure 4.5-6. Reactor Service Building, Plan at Elevation 89'0"

(UE&C 6255.005-D-007)



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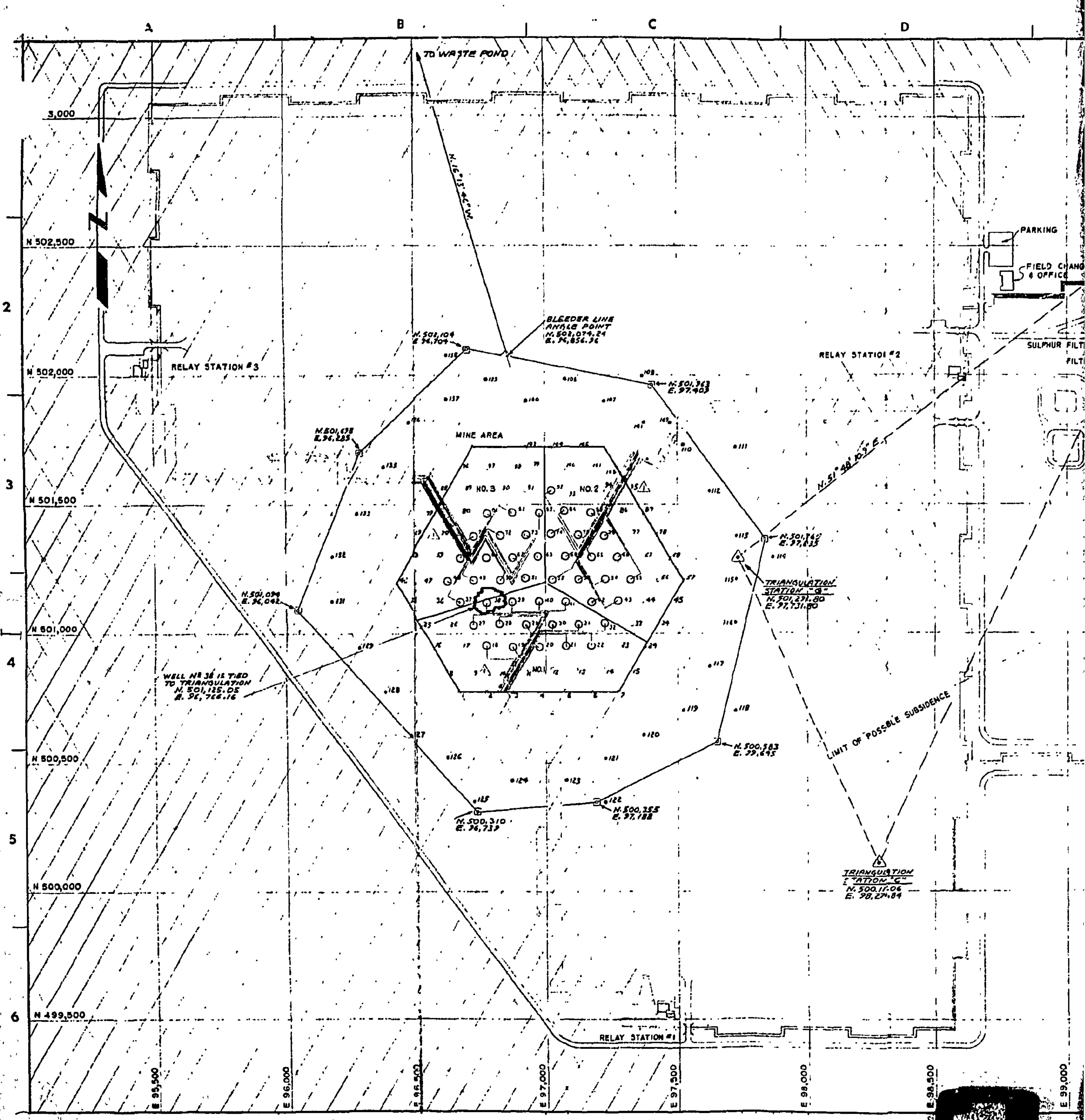

A
B
C
D
TO WASTE POND
3,000
N 502,500
N 502,000
N 501,500
N 501,000
N 500,500
N 500,000
N 499,500
2
3
4
5
6
PARKING
SULPHUR FILT
RELAY STATION #3
RELAY STATION #2
RELAY STATION #1
BLEEDER LINE
ANGLE POINT
N.502,074.24
E.96,856.96
MINE AREA
NO.3
NO.2
NO.1
N.501,363
E.97,403
N.501,698
E.96,285
N.501,094
E.96,042
N.501,740
E.97,815
TRIANGULATION
STATION "G"
N.501,291.80
E.97,731.80
WELL Nº 36 IS TIED
TO TRIANGULATION
N. 501,125.05
E. 96,766.16
LIMIT OF POSSIBLE SUBSIDENCE
N.500,583
E.97,645
N.500,255
E.97,188
N.500,310
E.96,737
TRIANGULATION
STATION "C"
N.500,11.06
E. 98,274.84
E 95,500
E 96,000
E 96,500
E 97,000
E 97,500
E 98,000
E 98,500
E 99,000

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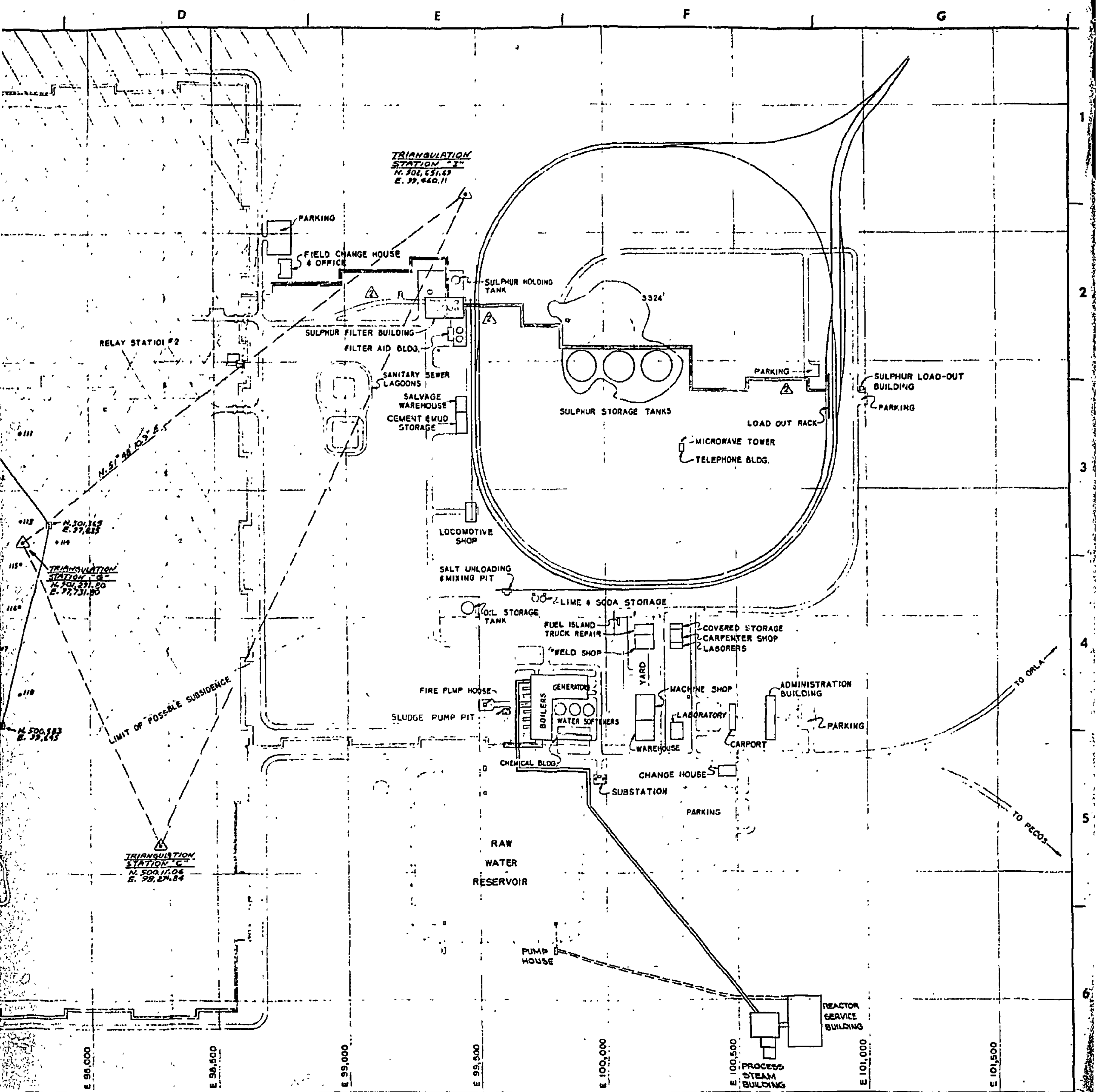
Figure 4.5-7. Main Piping Layout, General Arrangement



Babcock & Wilcox

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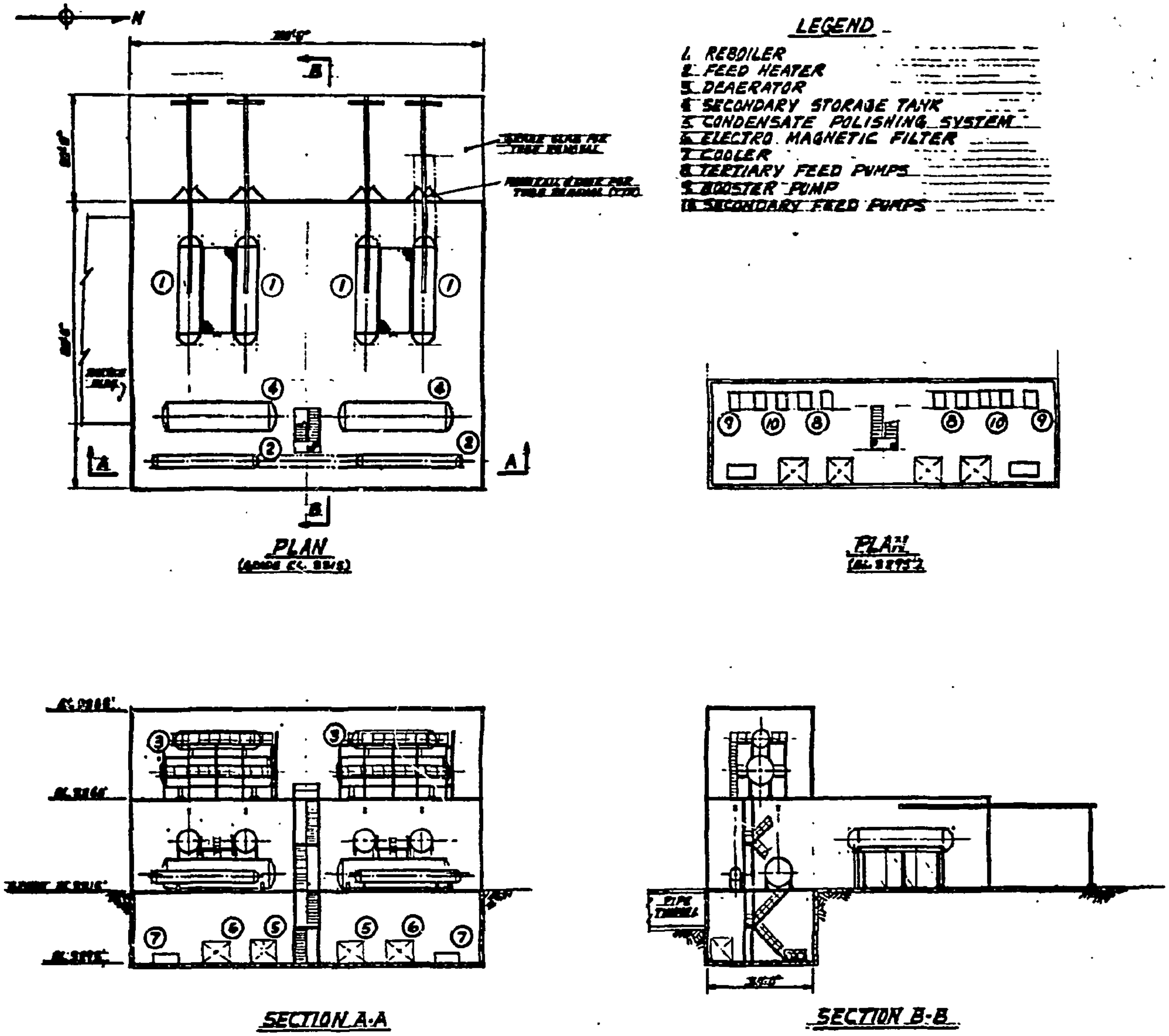
Figure 4.5-8. Process Building Layout — 365-MWt CNSG for Process Heat



4.6. Tertiary Loop Design

The energy requirements of the Duval Corporation's Culberson sulfur mine have been identified as:

1. 11.5 MW of electricity produced by non-condensing turbine-generators.
2. 16.25 kg/s (129,000 lb/h) of service steam for miscellaneous heating requirements throughout the mining operation.
3. Hot water at 162.78C (325F) — the capacity dependent on the rate of sulfur mining but equal to approximately 7.57 m^3 (2000 gal) per long ton (1016 kg) of sulfur mined. The hot water requirement is about 9.9×10^6 lbs/day.

Raw material is hard well water at 24C (75F) and water returned from the mines at 46C (115F) — about half from each source but variable depending on the type of operation at the particular time under consideration.

The CNSG is capable of producing 365 MWt of energy in the form of slightly superheated steam at 4.826 MPa (700 psia) using feedwater at 204C (400F).

A tertiary loop design was devised based on the energy available from the CNSG, the requirements of the mining operation, and making optimal use of the raw materials and existing facilities.

4.6.1. Design Selection

Prior to establishing the tertiary loop design, the energy requirements for process operations were compared to the available energy to determine plant capability. The PE-CNSG is capable of producing sufficient energy for all process needs. However, offsite supply of electrical energy is required for reactor and BOP.

The second concern in the selection of the tertiary loop design is the unavailability of high-quality water. The raw well water supply is very high in hardness and dissolved solids. Typical water chemistry (in ppm) and acidity as provided by the Duval Corporation are listed below (6/18/76).

$CaCO_3$	Ca	Mg	Na	HCO_3	SO_4	Cl	SiO_2	pH
916	270	62	210	281	600	375	15	7.4

The following list of requirements and assumptions for tertiary loop design was developed based on this water chemistry:

1. Total solids in the secondary loop must not exceed 0.050 ppm.

2. Total solids in the boiler water of the reboilers must not exceed 2000 ppm.
3. Feedwater entering the reboiler must be softened and deaerated.
4. Water to heat exchangers where no boiling occurs must be softened and deaerated.
5. Water returned from the mines is assumed to be softened and suitable for use in heat exchangers, pumps, pipes, and valves.
6. Contact of hard, aerated water with pipes, pumps, and valves must be kept to a minimum.

Two concepts for supplying the energy requirements to the mining operation are technically possible. One concept passes all of the PE-CNSG steam through reboilers to generate tertiary steam for the process. The second concept utilizes the PE-CNSG steam directly in the process with possibly some amount being used to generate tertiary steam in reboilers (Figure 4.6-1).

From an equipment standpoint the system where 4.826 MPa (700 psia) steam produced by the CNSG would be used directly in heat exchangers to provide process water at 162.78C (325F) is the simplest. However, this concept has potential difficulties in maintaining the PE-CNSG water chemistry. Further, this concept alone, without some provision for preventing carryover of radioactive contaminants due to heat exchanger tube failures, could increase operational and licensing problems. Since there was not sufficient time to investigate these potential problems, the concept utilizing the total PE-CNSG steam in reboilers for tertiary steam production was used in this study.

The indirect transfer or tertiary loop system would transfer energy from the PE-CNSG using reboilers and feedwater heaters. The reboilers would produce saturated steam at 1.13 MP (164 psia) in the tertiary loop. This steam would be transported to the existing boiler/turbine building. It would be used as service steam, for heating the mine water to the required 162.8C (325F) final temperature, and directly in the existing turbine generators to produce the required 11.5 MW of electrical energy. The principal advantage of this system is that the steam and feedwater (secondary loop) system can be controlled closely, minimizing the possibility of contamination of the secondary loop water and subsequent fouling or damaging of the PE-CNSG steam generator modules. The tertiary loop system was chosen as the reference design.

Several variations of the tertiary loop concept were next devised and evaluated (Figures 4.6-2 through 4.6-4).

1. An open loop design (Figure 4.6-2) uses the 1.13 MPa (164 psia) tertiary loop steam in direct-contact water heaters to raise the water to the required final temperature. Blowdown from the reboilers and the turbine exhaust is fed to the existing hot process softeners. No attempt is made to recover the condensate from the service steam. This arrangement results in a minimum capital investment in heat exchangers but requires the greatest investment in water treatment equipment for the feedwater to the reboilers.

2. A closed loop design (Figure 4.6-3) uses the 1.13 MPa (164 psia) tertiary steam in tubular heat exchangers to raise the water to the required final temperature and also uses the turbine exhaust steam in tubular exchangers for preheating the water that eventually goes to the mine.

   With the closed loop practically the entire feedwater supply to the reboilers is condensate so minimum feedwater treatment is required and reboiler blowdown is kept to a minimum. There is a maximum investment in tubular heat exchangers. However, this arrangement precludes the use of the hot process softeners from a practical viewpoint. Since the hot process softeners are eliminated, high capacity process softeners (zeolite softeners) must be used. The hard well water (∿1000 ppm total hardness) must be softened to prolong the life of the piping, valves, pumps, and heat exchangers.

3. The design chosen (Figure 4.6-4) is a compromise between the arrangements presented on Figures 4.6-2 and 4.6-3. As before, the secondary loop is a closed loop and all of the energy is supplied as 1.13 MPa (164 psia) steam or reboiler blowdown. In this concept the blowdown and turbine exhaust steam are fed directly to the hot process softeners, but the water to the mines is raised to the final temperature by tubular heat exchangers. The condensate from the service steam and water heaters is returned to be used as feedwater to the reboilers. The turbine exhaust steam and blowdown are lost. These flows must be made up by a feedwater treatment system consisting of a RO system and zeolite softeners.

In addition to the technical merits of the reference tertiary loop (Figure 4.6-4), the economic worth of each variation was determined. Capital and O&M costs were obtained from equipment vendors. The reference design was approximately 50% less costly than the closed loop and 10% less costly than the open loop.

4.6.2. Design Description

The flows, pressures, and temperatures shown in Figure 4.6-3 are based on the CNSG supplying 365 MWt to the tertiary system. As energy requirements are decreased, the flows will decrease proportionately. Pressures in the tertiary loop will remain approximately constant, but the pressure of the secondary steam entering the reboilers will decrease with decreasing load.

The recommended design is described in more detail below. The 1.13 MPa (164 psia) steam is fed to (1) the turbine-generators (TGs), (2) the tubular heat exchangers, (3) the service steam, and (4) the deaerator and tertiary storage tank.

The condensate from the service steam and from the tubular heat exchangers is returned to the tertiary storage tank. The exhaust from the TGs and the blowdown is fed to the hot process softeners and mixed with the water that goes to the mine. This amount of water must be treated by the RO system and zeolite softener.

The tertiary storage tank collects condensate return, tertiary makeup water, and water from the tertiary water cooler, deaerates the water, and feeds water to the tertiary feedwater pump. From here the feedwater goes through a feedwater heater to the reboiler, receiving heat from the secondary loop. Well water is pumped through the RO system removing undesirable salts, thence to the hot process softener, to a water heater and to the mine. Mine return water is divided between the hot process softener and a water heater from which water is returned directly to the mine, maintaining required water purity.

Control

Feedwater flow to the CNSG steam generator is controlled by an ICS but receives its main signal from steam generator pressure automatically increasing flow if pressure decreases blow 4.83 MPa (700 psia) and vice versa.

The secondary steam flow to the reboiler is controlled by tertiary steam discharge pressure from the reboiler. Secondary steam flow will automatically increase if the tertiary steam pressure drops below 1.13 MPa (164 psia) and vice versa. Secondary steam flow to the turbine generator will be controlled by frequency, a part of the Duval Corporation system. Secondary steam flow to the water heaters will be controlled by the temperature of the water leaving the heaters. Steam flow will automatically increase (increasing pressure) if the temperature drops below 162.8C (325F) and vice versa.

Condensate flow from the water heaters will be controlled by the water level in the water heater and will automatically increase if the level rises above the set point and vice versa.

Feedwater flow to the reboilers will be controlled by the water level in the tertiary side of the reboiler. Flow will automatically increase if the level drops below a set point and vice versa. The feedwater flow controller will probably be a three-element controller using water level, steam flow, and feedwater flow to generate a feedwater flow control signal. Makeup to the tertiary storage tank will be controlled by a water level in the tank.

Blowdown will be continuous and manually adjusted to maintain boiler water solids concentration within specified limits. The flow of well water to the softener is assumed to be adjusted manually to maintain a desired water level in the softener; the flow of mine return water to the softener will then be adjusted to maintain the proper temperature of the water in the softener.

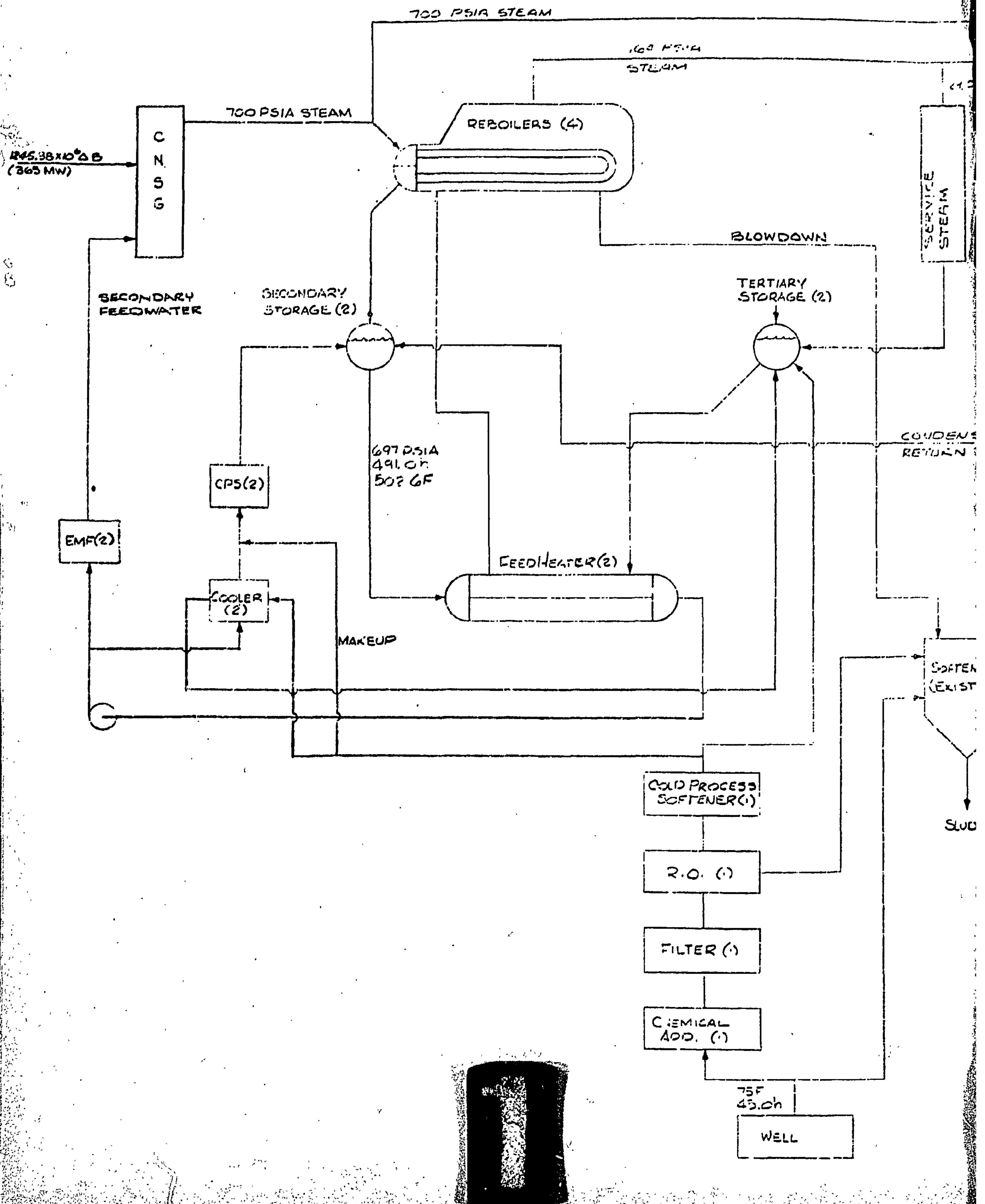
4.6.3. Interface With Existing Equipment

It is assmumed that the new Process Heat Transfer Building and the Water Treatment Building will be separated from the existing boiler/turbine building by at least 402 m (0.25 mile).

The reboilers, feedheaters, secondary and tertiary water storage tanks, the secondary and tertiary water treatment facilities, and associated piping, pumps, and control valves, will be housed in a new building. The water heaters will be located in the existing boiler turbine building convenient to the existing turbine generators, hot process water softeners, pumps, etc.

The connections between two groups of facilities will consist of the following:

1. Two 610-mm (25-inch), 1.13 MPa (164 psia) tertiary steam lines.
2. Two 203-mm (8-inch) condensate return lines.
3. Two 51-mm (2-inch) reboiler blowdown lines.
4. One raw water (well water) to the water treatment building.
5. One concentrated raw water from the water treatment building instrumentation and control lines.



700 PSIA STEAM
STEAM
700 PSIA STEAM
REBOILERS (4)
C N S G
(365 MW)
SERVICE STEAM
BLOWDOWN
SECONDARY FEEDWATER
SECONDARY STORAGE (2)
TERTIARY STORAGE (2)
697 PSIA
CPS(2)
EMF(2)
FEEDHEATER(2)
COOLER (2)
MAKEUP
COLD PROCESS SOFTENER(1)
R.O. (1)
FILTER (1)
CHEMICAL ADD. (1)
75F
WELL

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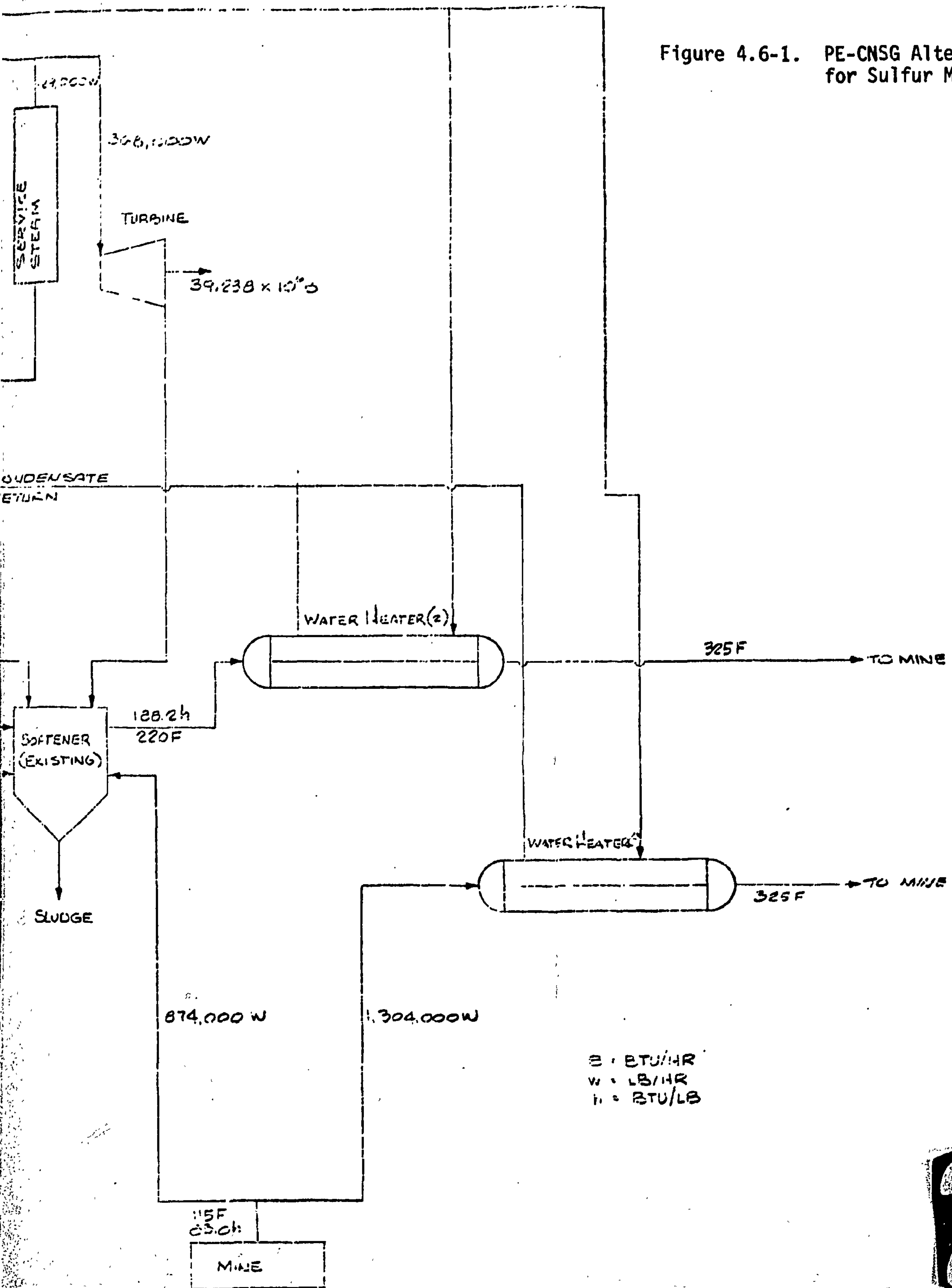
Figure 4.6-1. PE-CNSG Alternate No. 1 for Sulfur Mine



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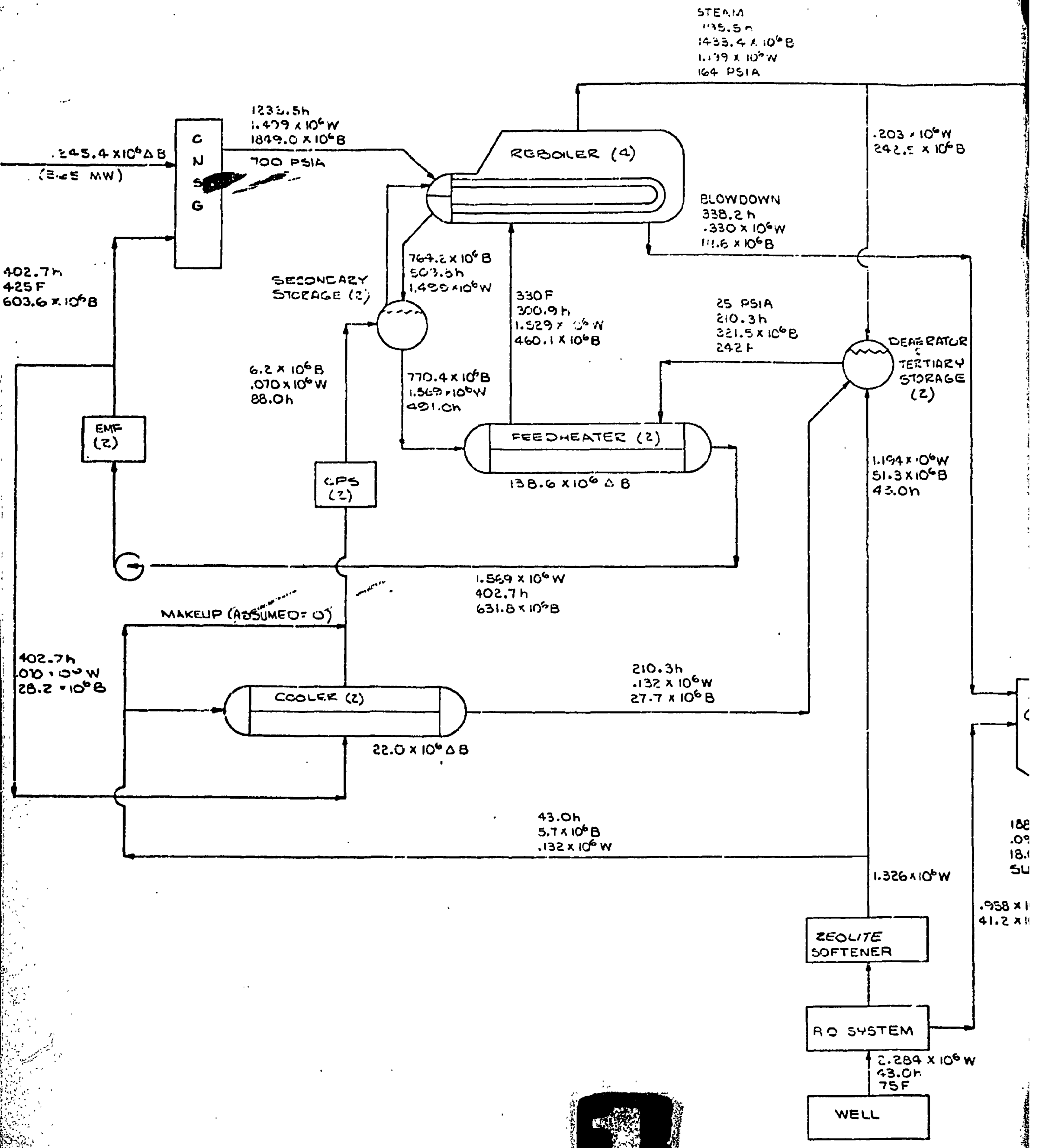

STEAM
195.5 h
1433.4 x 10^6 B
1.139 x 10^6 W
164 PSIA
1233.5h
1.429 x 10^6 W
1849.0 x 10^6 B
700 PSIA
245.4 x 10^6 Δ B
(3.65 MW)
C
N
S
G
REBOILER (4)
.203 x 10^6 W
242.5 x 10^6 B
BLOWDOWN
338.2 h
.330 x 10^6 W
111.6 x 10^6 B
402.7h
425 F
603.6 x 10^6 B
SECONDARY STORAGE (2)
764.2 x 10^6 B
503.6h
1.499 x 10^6 W
330 F
300.9 h
1.529 x 10^6 W
460.1 x 10^6 B
25 PSIA
210.3h
321.5 x 10^6 B
242 F
DEAERATOR & TERTIARY STORAGE (2)
6.2 x 10^6 B
.070 x 10^6 W
88.0h
770.4 x 10^6 B
1.569 x 10^6 W
491.0h
EMF (2)
FEEDHEATER (2)
138.6 x 10^6 Δ B
CPS (2)
1.194 x 10^6 W
51.3 x 10^6 B
43.0h
1.569 x 10^6 W
402.7 h
631.8 x 10^6 B
MAKEUP (ASSUMED = 0)
402.7h
.070 x 10^6 W
28.2 x 10^6 B
COOLER (2)
22.0 x 10^6 Δ B
210.3h
.132 x 10^6 W
27.7 x 10^6 B
43.0h
5.7 x 10^6 B
.132 x 10^6 W
1.326 x 10^6 W
.958 x 1
41.2 x 1
ZEOLITE SOFTENER
RO SYSTEM
2.284 x 10^6 W
43.0h
75F
WELL

1

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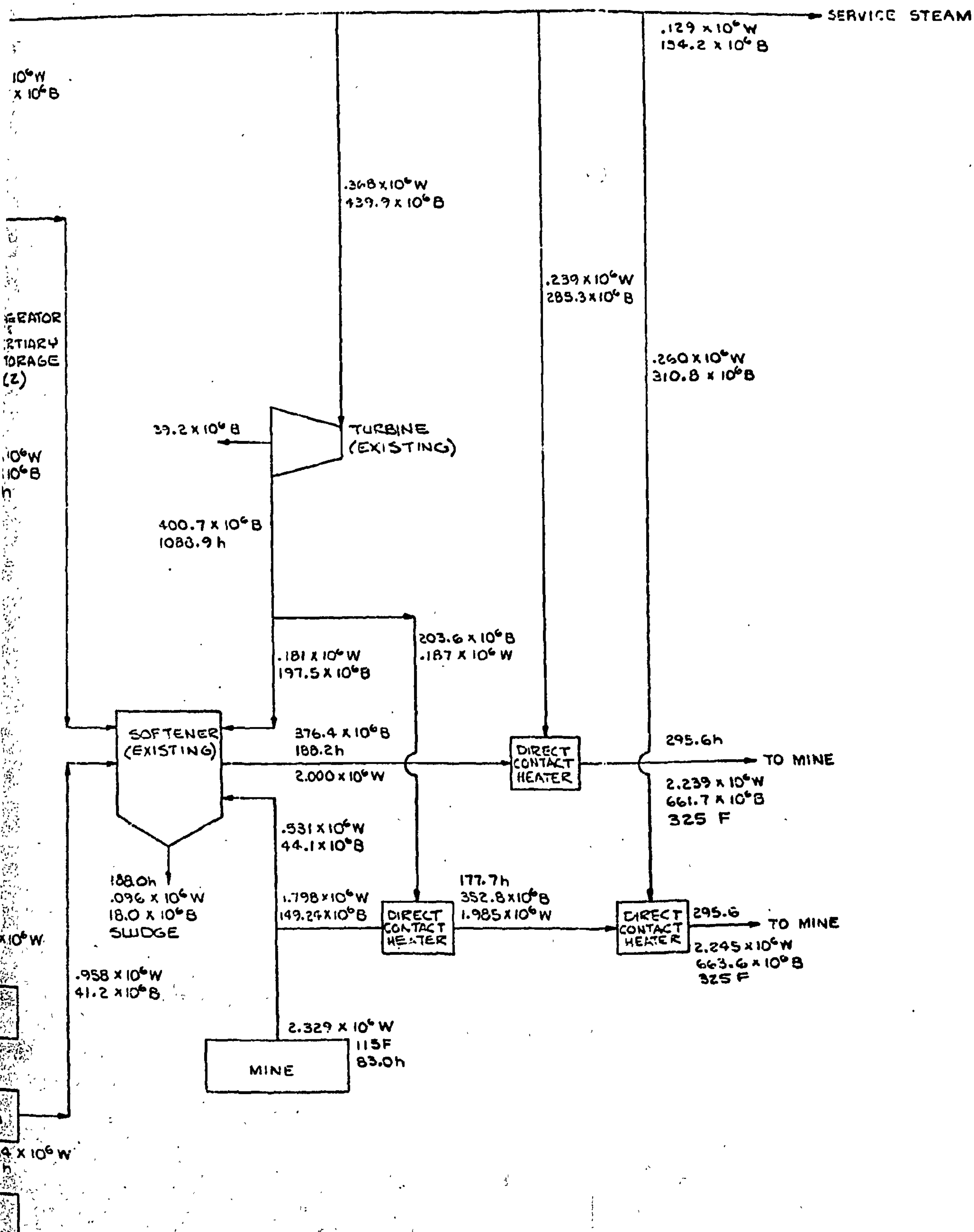
Figure 4.6-2. Open Loop — ~25% Blowdown PE-CNSG for Sulfur Mine



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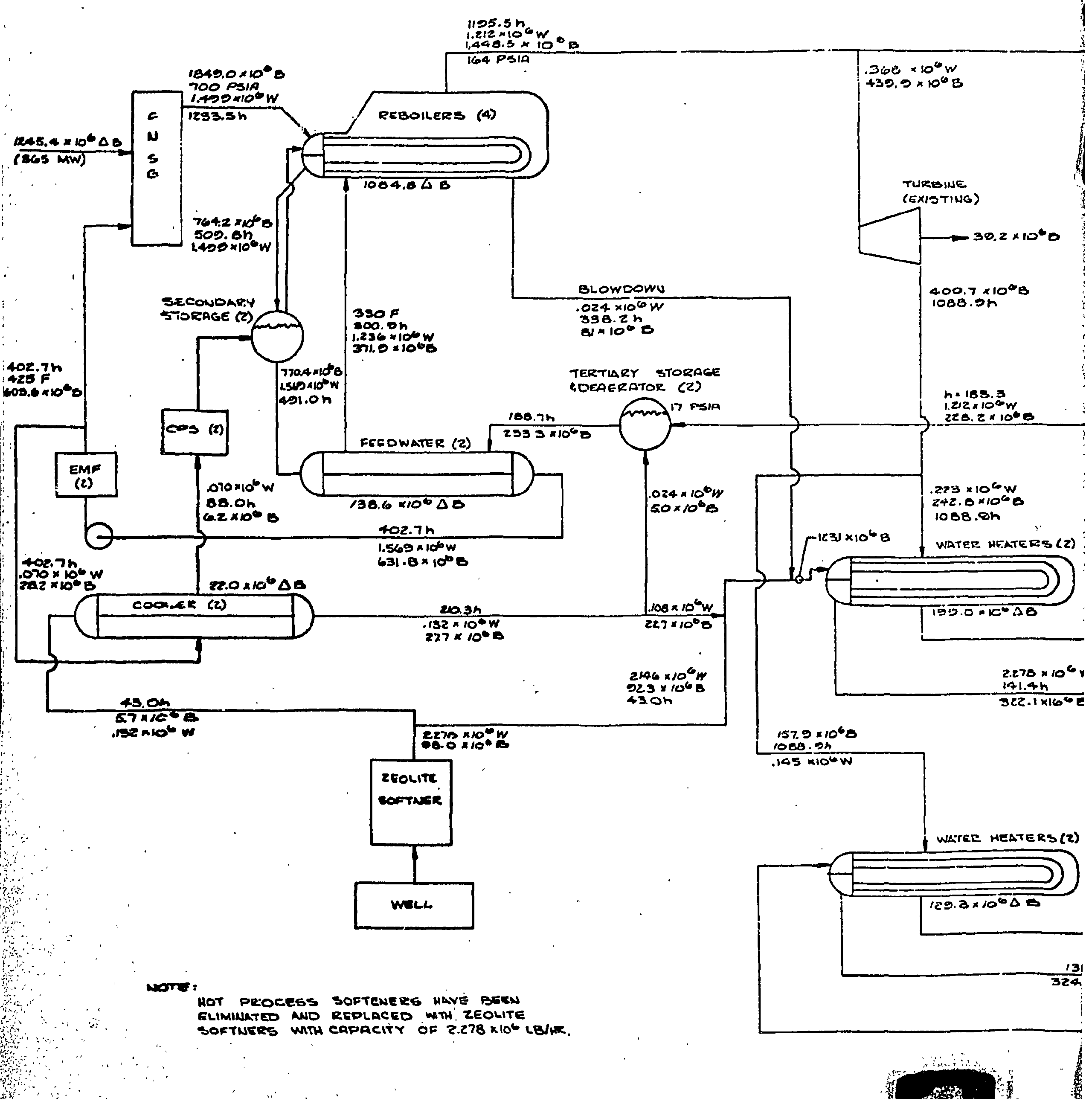

NOTE:

HOT PROCESS SOFTENERS HAVE BEEN ELIMINATED AND REPLACED WITH ZEOLITE SOFTNERS WITH CAPACITY OF 2.278 × 10^6 LB/HR.

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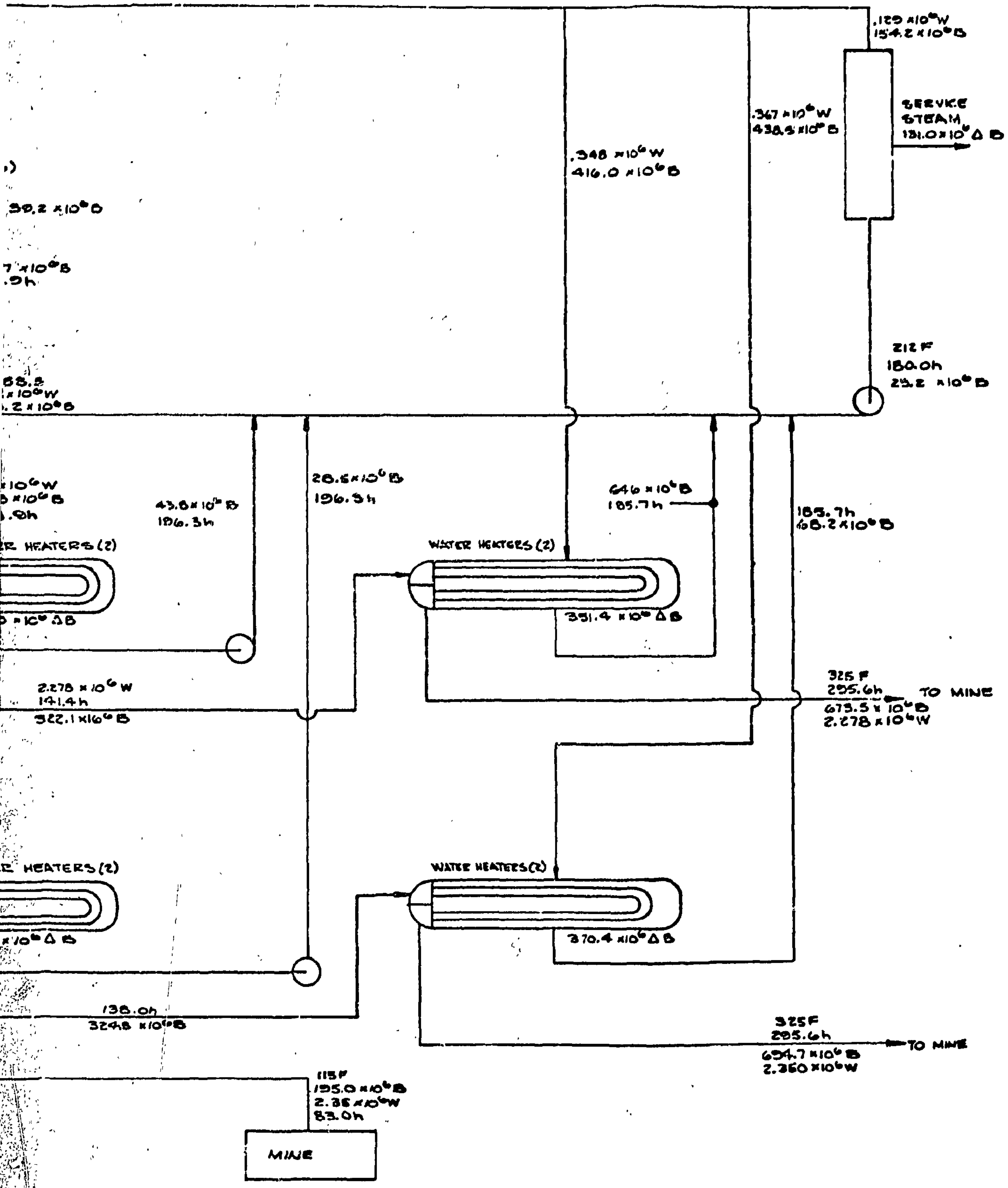
Figure 4.6-3. Closed Loop — ~2% Blowdown PE-CNSG for Sulfur Mine



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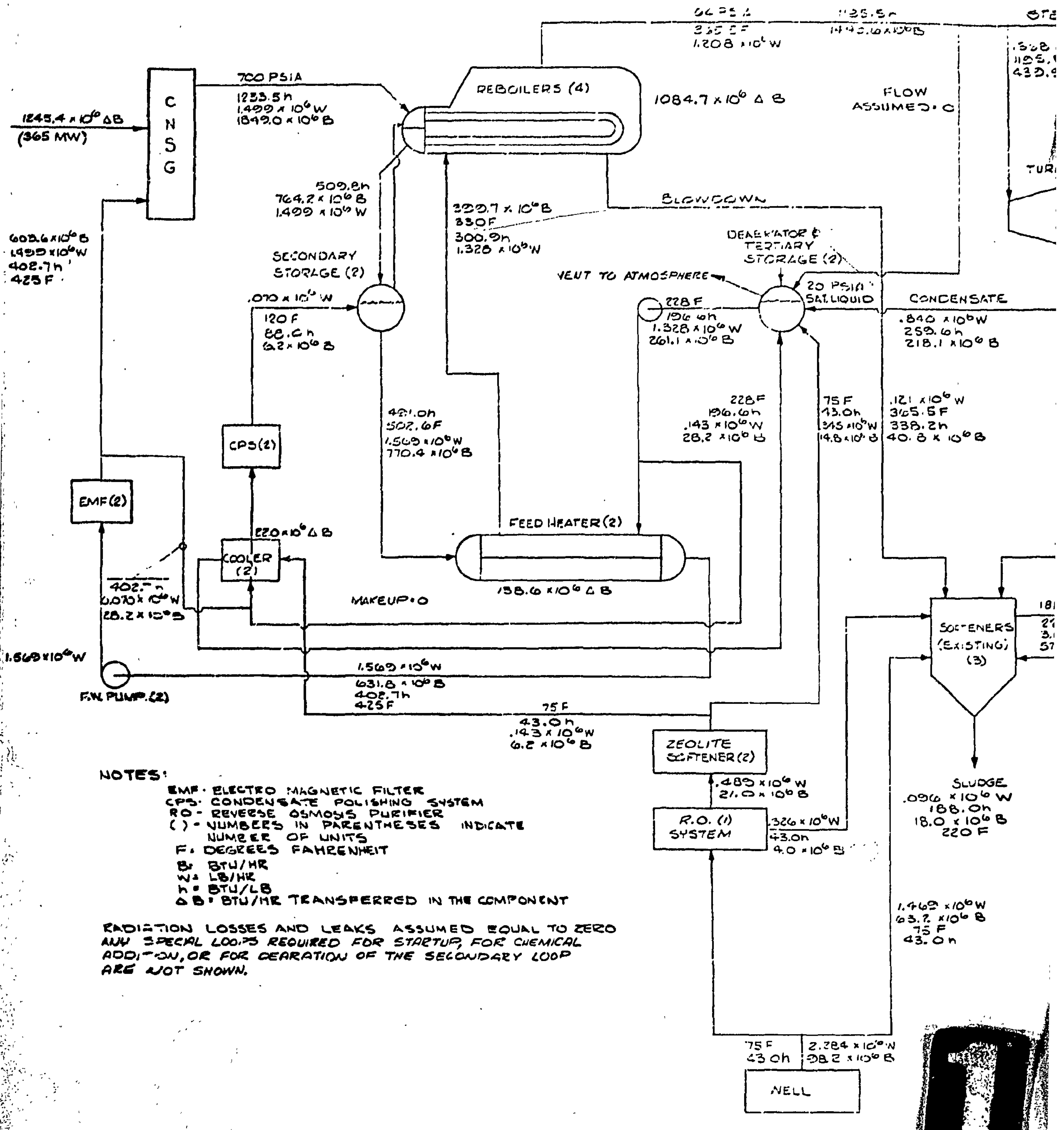

NOTES:

EMF - ELECTRO MAGNETIC FILTER
CPS - CONDENSATE POLISHING SYSTEM
RO - REVERSE OSMOSIS PURIFIER
( ) - NUMBERS IN PARENTHESES INDICATE NUMBER OF UNITS
F - DEGREES FAHRENHEIT
B - BTU/HR
W - LB/HR
h - BTU/LB
ΔB - BTU/HR TRANSFERRED IN THE COMPONENT

RADIATION LOSSES AND LEAKS ASSUMED EQUAL TO ZERO
ANY SPECIAL LOOPS REQUIRED FOR STARTUP, FOR CHEMICAL ADDITION, OR FOR DEARATION OF THE SECONDARY LOOP ARE NOT SHOWN.

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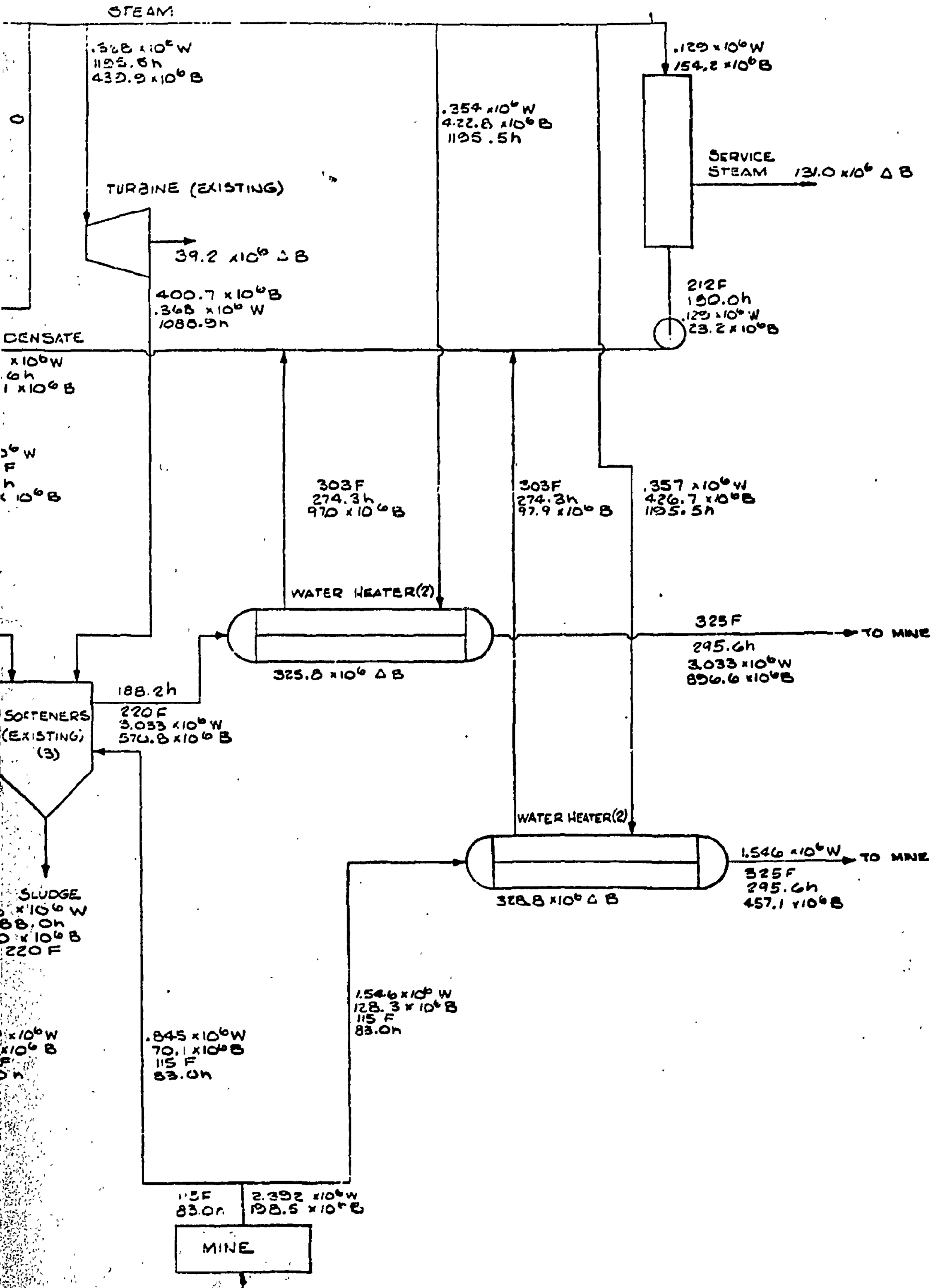
Figure 4.6-4. PE-CNSG for Sulfur Mine — ~10% Blowdown



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5. SAFETY ASSESSMENT

The safety assessment of the PE-CNSG for use as a replacement energy source was concentrated in two areas — process safety interfaces and reactor accidents. The purpose of the assessment was to identify necessary design and/or operational changes.

5.1. Safety Interfaces

The principal safety interfaces identified were process hazards (fire, subsidence) use of existing facilities, and radiation effects on the process. Each of these interfaces was evaluated qualitatively.

5.1.1. Process Hazards

The Frasch mining process uses hot water to melt the sulfur deposit and force the molten sulfur to the surface. At the Duval site the sulfur is kept molten and shipped by rail in that state. The potential hazards identified for this study were (1) sulfur fires, (2) process line breaks, and (3) fuel oil fires. The mining operation itself causes the additional hazard of subsidence.

Sulfur Fires

The potential for sulfur fires at the Duval site stated by Duval Corporation is low because the sulfur is kept molten rather than solid. The natural gas currently used for the energy source is brought in by pipeline and is not stored onsite. Fuel oil stored onsite is approximately 305 m (1000 ft) from the sulfur storage area. New storage facilities for fuel oil for standby operation will also be located a sufficient distance from the sulfur storage. If a sulfur fire occurs, it can be readily extinguished with a water mist. Onsite facilities exist to combat sulfur fires.

Fuel Oil Fires

As previously stated, the existing oil storage is located well away from the sulfur storage tanks. The proposed location of the PE-CNSG plant is approximately 305 to 460 m (1000 to 1500 ft) from the existing storage facility.

This separation is expected to be sufficient to mitigate the consequences of potential fires. Because of the remote location of the site, all fire service must be self-contained. Review of the existing facilities should be conducted to determine the need for additional training and equipment.

Subsidence

The subsidence present at the Duval site is primarily the result of the mining operation. Figure 5.1-1 is a topographic map of the mine area prior to the start of mining. Figure 5.1-2 reflects the current topography. The closest portion of the mine area is approximately 762 m (2500 ft) from the proposed plant location. This distance is much greater than the expected subsidence cone from the mining operation, which is not expected to contribute hazards to the PE-CNSG plant.

Process Line Breaks

Rupture of a process energy delivery line from the tertiary loop to the mining operation was postulated as a process hazard. The transient caused by such a pipe failure would affect the tertiary loop and could feed back into the PE-CNSG primary circuit through the steam and feedwater system. A qualitative assessment indicates that such feedback would be a minor disturbance to the primary system.

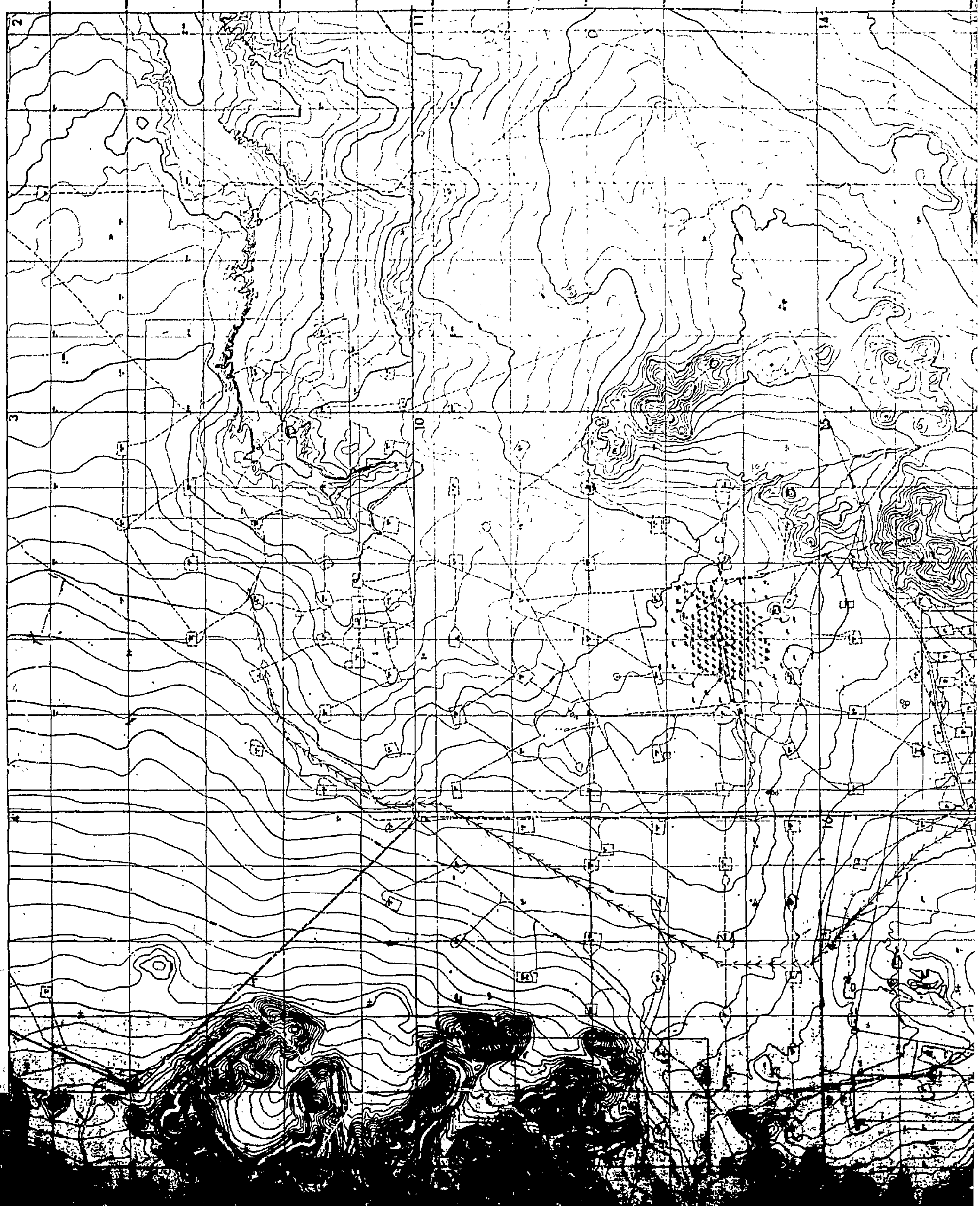
5.1.2. Utilization of Existing Process Facilities

The basic criterion used in developing the PE-CNSG as a replacement energy source for the Duval Corporation site was to utilize the existing equipment where possible. This is reflected in the tertiary loop design, which uses much of the existing water treatment facilities. Electric power for the process is provided by the in-place, non-condensing turbines. Electric power for the nuclear plant is brought from offsite by two 69-kV transmission lines. Although a cooling tower would be provided for safety service, the existing 60,600-m^5 (16-million gal) reservoir will be used for cooling as well. The heat exchangers designed for the final stage of process heating will be placed in the existing plant buildings. Backup process energy for use during refueling and other reactor outages will be provided by existing boilers using fuel oil instead of natural gas.

The interfaces with existing process equipment identified in the study were reviewed to determine impact on design and operation. No major impacts were identified except in the tertiary loop design. For the tertiary loop the existing equipment imposes temperature and pressure limits. A detailed evaluation of all interfaces would be performed in future work.

5.1.3. Radiation Effects

The use of nuclear energy as the replacement energy source presents a minor hazard to the process, i.e., the possible contamination of the sulfur by trace quantities of activated corrosion products or fission products. However, there are multiple barriers between the sources of contamination and the mining operation. In addition, the water quality in the heat exchange circuits is maintained at very low levels of particulates by the use of filters and demineralizers. Radiation monitors are expected to be used in the heat transfer loops, and appropriate limits for trace contaminants will be set and controlled by periodic analyses of the sulfur.



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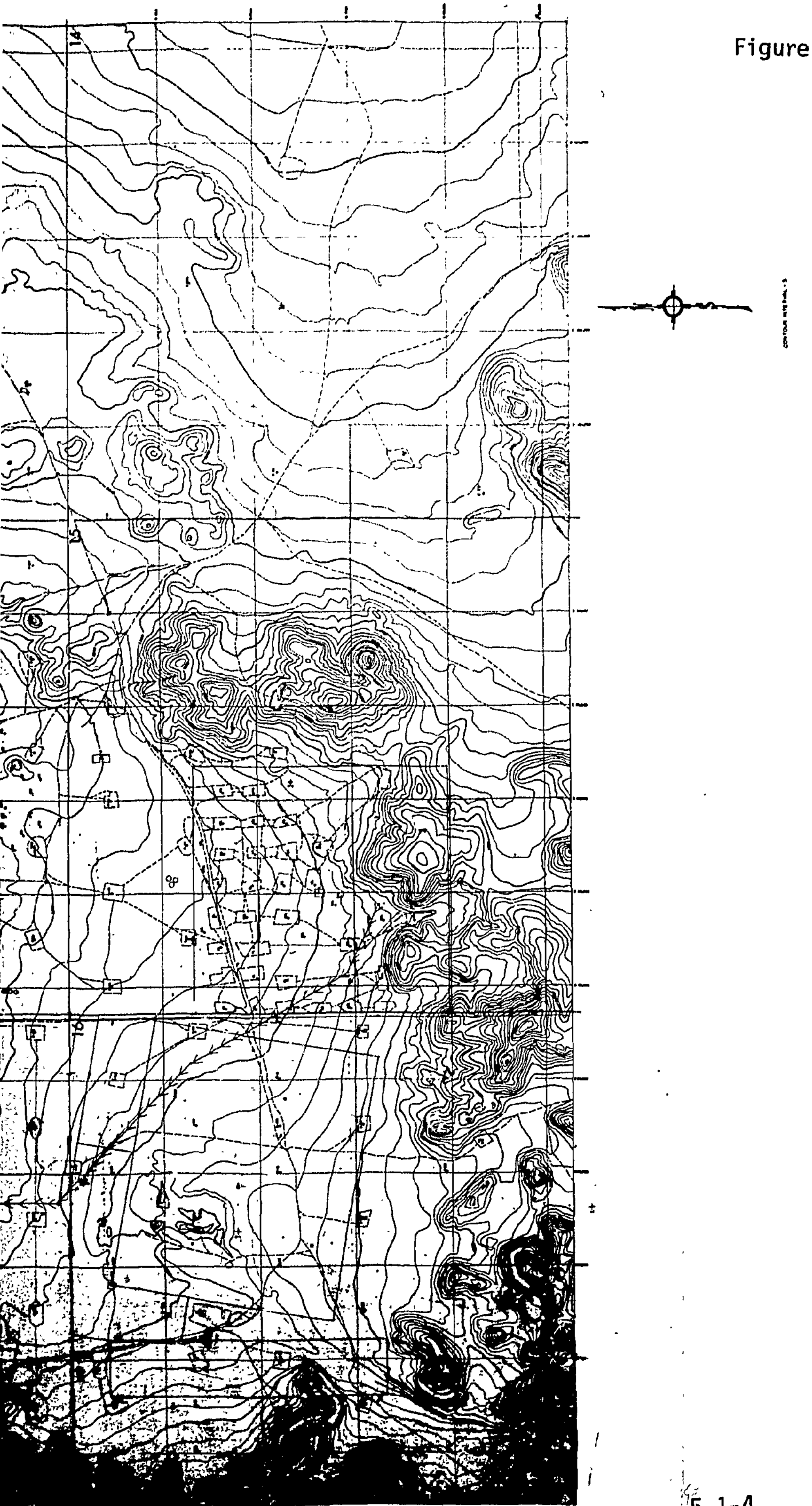

Figure 5.1-1. Topographic Map of Culberson Property - 1969

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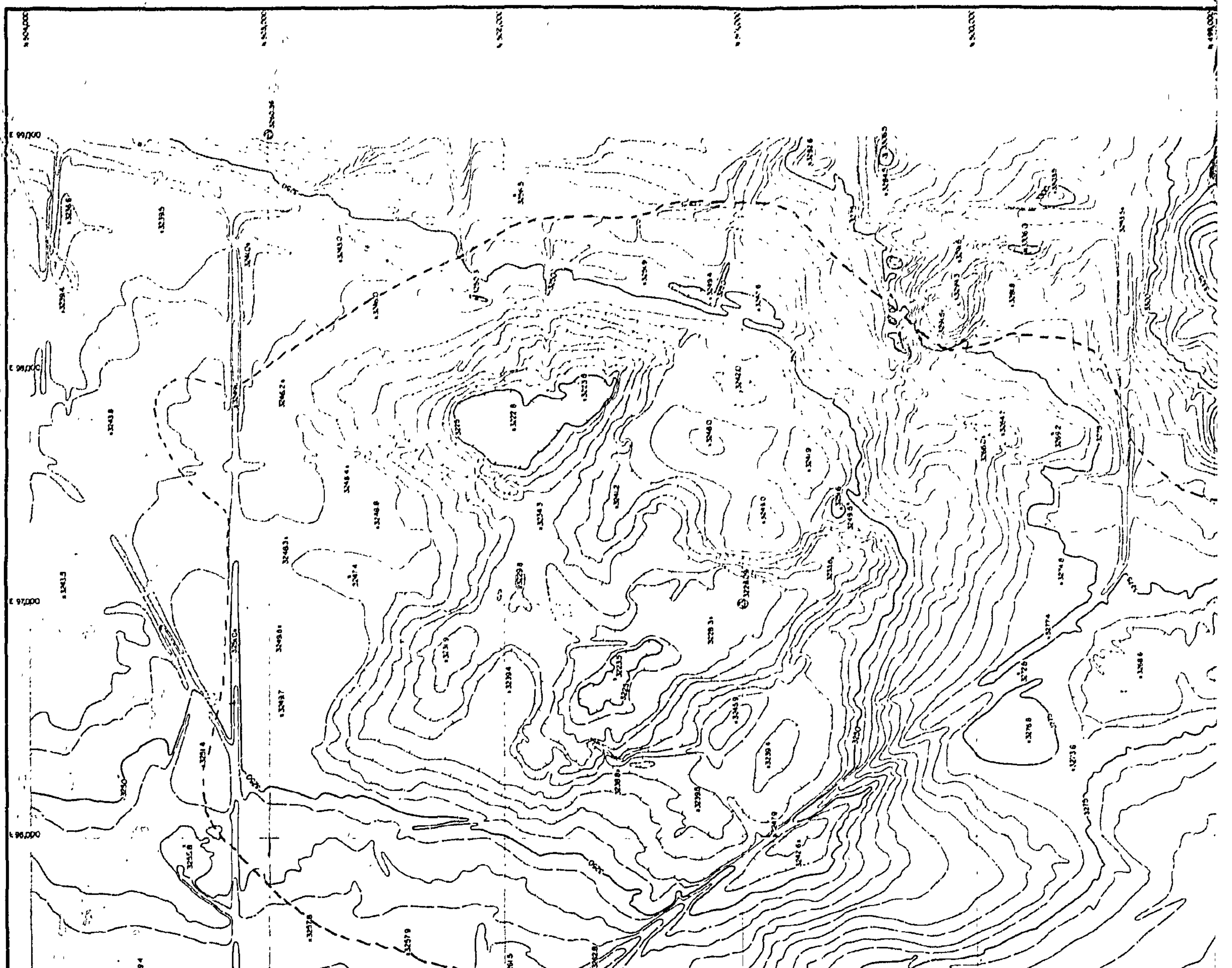
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E 96,000

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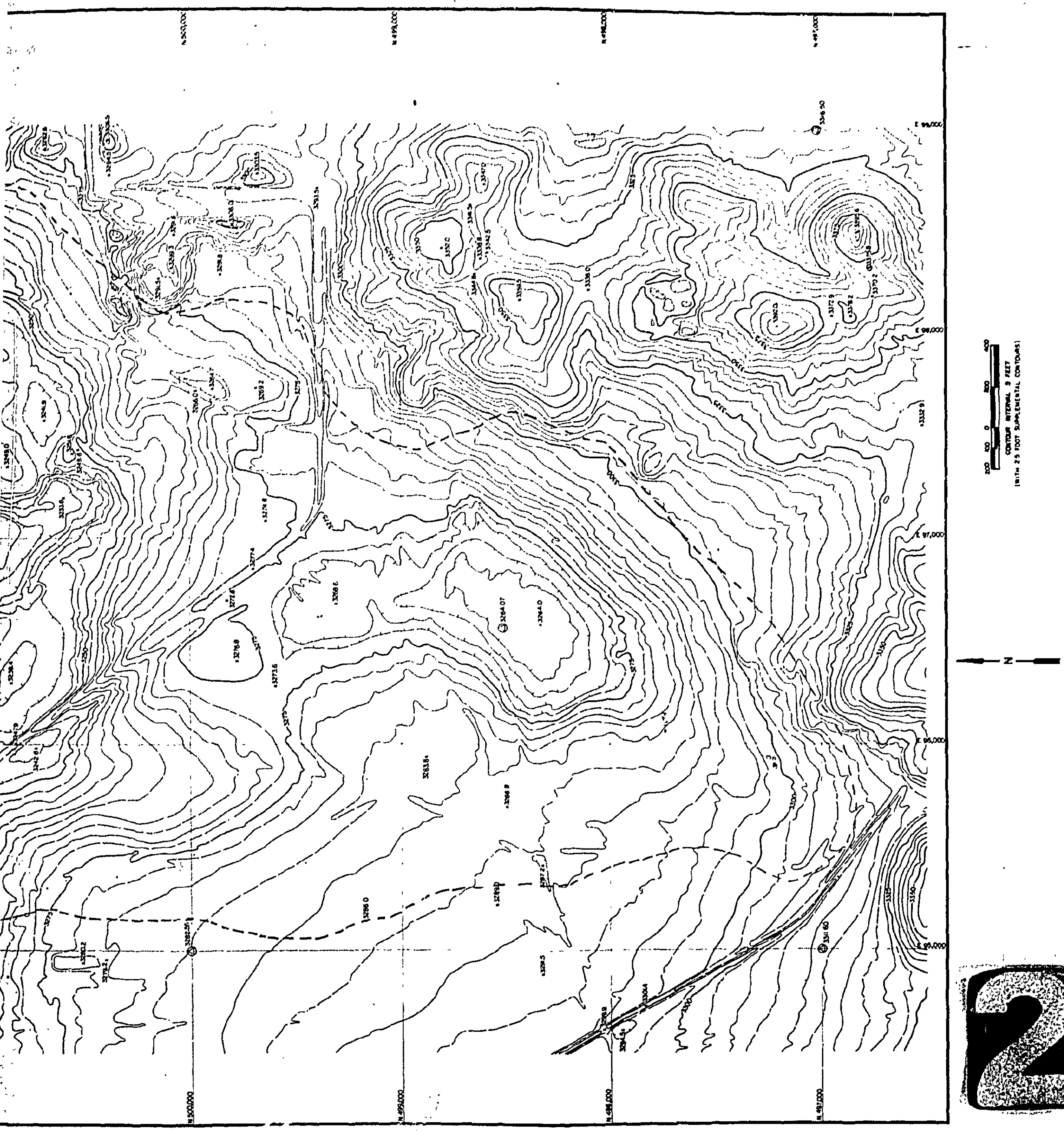
Figure 5.1-2. Topographic Map of Culberson Property — August 1976



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5.2. Transient/Safety Analysis

A wide range of transient and safety analyses must be performed for the PE-CNSG at 365 MWt to ensure that design limits and limiting conditions of operation can be met. The preliminary transient/safety analysis work performed for this study was intended to confirm design margins and identify potential problems.

5.2.1. Process Transients

The transients of concern to the PE-CNSG plant from the process are those involving changes in energy demand. Current operating practice at the Duval Corporation sulfur mines is based on sulfur demand forecasts over a six month period with monthly updates. For each month of operations, energy demand is essentially constant. Process energy is supplied to 20-40 wells continuously, and changes in demand are infrequent and slow. Further, the loss of load to the process will be absorbed by the tertiary loop and feedback to the reactor circuit thereby minimized. Loss-of-load studies will be required to be performed to confirm operation practice and design margins. It is noted that backup steam will be required and will be available from existing sources in case of reactor trip to maintain sulfur in the molten condition.

5.2.2. Design Basis LOCAs

For the PE-CNSG there is one design basis LOCA for the ECCS and one for the containment. Since the piping openings to the PE-CNSG RCS are relatively small, the core can be assumed to be adequately cooled if it remains covered by water or by a water-steam mixture over the blowdown period of the LOCAs. The ECCS design basis LOCA results in the smallest water volume remaining in the reactor vessel to cool the core. This is the injection line LOCA. The design basis LOCA for the containment is the LOCA that produces the highest peak pressure and temperature in the dry well. From the work done on the M-CNSG, the design basis containment LOCA is the surge line LOCA, which produces peak pressures and temperatures slightly greater than those for the letdown line LOCA.

5.2.2.1. ECCS Design Basis LOCA

As previously described, the injection line LOCA is the design basis accident for the ECCS. The injection line (3-inch Sch 160) passes through the reactor vessel above the steam generator modules and turns downward to open below the steam generators in the downcomer annulus placing the height of the end of the injection line just above the active fuel height.

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This accident is analyzed using the CRAFT[7] computer code. The RCS is described as a group of fluid nodes and connecting flow paths. The model used for this analysis (Figure 5.2-1) is based on the M-CNSG work. The core power and steam generator heat removal rate were changed to reflect the 365 MWt power level for the PE-CNSG. For this analysis the following assumptions are made:

1. The reactor is operating at a steady-state power level of 102% before the rupture.
2. The break occurs instantaneously, and the leak flow is calculated by the Moody leak correlation with a discharge coefficient of 1.0.
3. The control rods begin entering the core 0.5 seconds after the trip signal is generated.
4. The reactor trips on low RCS pressure of 13.9 MPag (2015 psig).
5. The RC pumps trip and coast down coincident with the reactor trip.
6. Secondary system trips occur coincident with the reactor trip.
7. One HPI system and one LPI system provide cooling water.
8. Half of the available cooling water is injected into the reactor vessel, while the other half is spilled into the containment through the broken injection line.
9. The auxiliary feedwater system removes no energy from the reactor vessel.

Using the CRAFT model shown in Figure 5.2-1 and the above assumptions, an analysis of the injection line LOCA for the PE-CNSG was performed. The volume of water remaining in the reactor vessel during the blowdown is shown in Figure 5.2-2. That volume of water reaches a minimum of 24.8 m^3 (875 ft^3) at 750 seconds. At that time the injection flow equals the flow through the break. This is the end of the blowdown period and the smallest reactor vessel water volume that will occur for this LOCA. Although this is less than the 26.9 m^3 (950 ft^3) of water that would completely cover the core, the core will be covered by a two-phase mixture. Since the heat transfer within the mixture is by a nucleate boiling process, the cladding temperature will remain within a few degrees of the saturated water temperature.

As a result of this anlaysis, the ECCS provides adequate cooling of the PE-CNSG core during the design basis injection line LOCA.

5.2.2.2. Containment Design Basis LOCA

For containment design the pressurizer surge line LOCA produces the peak containment dry well pressures and temperatures. The pressurizer surge line (6-inch sch 160 pipe) connects the pressurizer and the reactor vessel. A flow restriction has been placed in this line where it passes through the reactor vessel. With the flow area of a 4 sch 160 pipe, this restriction limits the flow during surge line breaks to help protect the core.

The analysis for the containment design basis LOCA begins with the determination of the mass and energy released through that break. This is determined using a CRAFT model similar to that used for the ECCS design basis LOCA analysis. The major differences are the type, location, size of the break, and that all cooling water from one ECCS string is injected into the reactor vessel. This provides the mass and energy released through the break during the LOCA, which is then input to the containment analysis.

Containment response to a LOCA is analyzed using the MVRBA[2] code. The PE-CNSG pressure suppression containment is described as a dry well volume and a wet well volume connected by a series of parallel vent pipes. The containment volumes and other pertinent information are given in Table 5.2-1. For this analysis the following assumptions were made:

1. One containment dry well cooler operates to remove heat from the dry well atmosphere.
2. One suppression pool cooler operates to remove heat from the wet well water.
3. Initial containment pressure is 68,950 Pa (10 psia). The normal operating range is 34,470-48,260 Pa (5 to 7 psia).
4. 100% of the steam flow through the vent pipe is condensed.

Limits have been set for the maximum dry well pressure and temperature and the maximum wet well water temperature. The design pressure for this containment is 0.724 MPag (105 psig), which includes a margin of 50% of the maximum allowable calculated pressure yielding a maximum allowable calculated pressure of 0.483 MPag (70 psig). The maximum dry well temperature limit was set at 152C (305F), which is the temperature limit set by the instrumentation. A limit of 71C (160F) has been set for the wet well water temperature due to several experiences with excessive vibrations in other suppression systems at water temperatures above this point.

An analysis of the containment response to a surge line LOCA has been made using the model and assumptions described above. The dry well pressure as shown in Figure 5.2-3 reached a peak of 0.407 MPag (59 psig), and the dry well temperature shown in Figure 5.2-4 reached a peak of 153C (307F). The wet well water temperature reached a peak of 71C (160F). The dry well atmosphere temperature exceeded the limit by 1.1C (2F), and the wet well water temperature has peaked at its limit.

With the current conditions the containment design basis surge line LOCA does exceed the dry well temperature limit set by instrumentation requirements. In addition, there is no margin in the wet well water temperature. It will be necessary to either remove some of the conservatism from the analysis or make e changes in the design of the containment and the auxiliary systems that remove heat from the containment during a LOCA. Any reduction of the margin of conservatism would have to be justified as would the effects of changes to the containment or the auxiliary systems.

The changes necessary to lower the dry well temperature below the 152C (305F) limit are not expected to present a major problem. Thus, the containment should handle the design basis LOCA with minor changes to the systems involved.

5.2.3. PE-CNSG Dose Analysis

The USNRC, in their investigation of potential reactor sites, considers the environmental effects of certain accidents. Their decision on site suitability is based in part on these considerations. Accordingly, analyses were performed to determine the radiological consequences of the MHA as defined below. The results of this analysis according to the NRC practice are used to establish minimum distances to the site boundary and to the LPZ boundary. The dose limits at these boundaries are set in 10 CFR 100[5] plants which have been granted a construction permit, and in USNRC Regulatory Guide 1.4[7] for reactor sites under USNRC review. Regulatory Guide 1.4 states:

> "It should be shown that the off-site dose consequences will be within the guidelines of 10 CFR, Part 100. (During the construction permit review, guideline exposures of 20 rem whole body and 150 rem thyroid should be used...)"

The site suitability with respect to accidental exposure to the public is then based on three criteria:

1. An exclusion area of such size that an individual located at any point on its boundary for two hours immedately following the MHA would not receive any more than 150 rem thyroid or 20 rem whole body.
2. A LPZ of such size that an individual located at any point on its outer boundary for 30 days immediately following the MHA would not receive more than 150 rem thyroid or 20 rem whole body.
3. A population center distance of at least 1-1/3 times the distance from the reactor to the outer boundary of the LPZ.

Definition of the Maximum Hypothetical Accident

The MHA is defined for radiological dose analysis considerations as a release to the containment atmosphere of 100% of total core noble gas inventory and 50% of the total core iodine inventory late in core life. The containment leaks at the design leak rate (as set in the Technical Specifications) for the first 24 hours and at half that rate thereafter. The following assumptions are used in the analysis:

1. At t = 0, the following activity is released to the containment atmosphere:
   a. 100% of the core equilibrium noble gas inventory.
   b. 50% of the core equilibrium iodine inventory.
2. Half of the airborne iodine plates out on the containment inner wall and equipment resulting in 25% of the total containment iodine inventory available for leakage.
3. The annular area surrounding the containment is considered a secondary containment; i.e., activity can be removed under controlled conditions.
4. The primary containment leaks to the secondary containment at the following rate:
   a. 0.2%/day for first 24 hours.
   b. 0.1%/day, thereafter.
5. The secondary containment (air-gap) exhaust rate is three air changes per hour.
6. Exhaust filter efficiency
   a. Elemental iodine, 95%
   b. Particulate idodine, 99%
   c. Organic iodine, 95%
7. No credit is taken for decay or deposition of effluents after release from the secondary containment.
8. One percent of the secondary containment exhaust is assumed to bypass the filters.
9. The atmospheric dispersion factors (X/Q) were determined by use of USNRC Regulatory Guide 1.4[7], which employs the following assumptions:
   a. Ground level release
   b. Building wake effects
   c. The following atmospheric conditions and wind speeds are assumed:

Time after accident	Atmospheric conditions ($\bar{u}$ = 1 m/speed)
0-8 hours	Pasquill type F, $\bar{u}$ = 1 m/s Uniform dir.
8-24 hours	Pasquill type F, $\bar{u}$ = 1 m/s Dir. variable within 22.5° sector
1-4 days	(a) 40% Pasquill type D, $\bar{u}$ = 3 m/s (b) 60% Pasquill type F, $\bar{u}$ = 2 m/s (c) Dir. variable within 22.5° sector
4-30 days	(a) 33.3% Pasquill type C, $\bar{u}$ = 3 m/s (b) 33.3% Pasquill type D, $\bar{u}$ = 3 m/s (c) 33.3% Pasquill type F, $\bar{u}$ = 2 m/s (d) Wind dir. 33% freq. in 22.5° sector

10. Power: 365 MWt, two-batch fuel cycle with first batch irradiated for 450 days and second batch for 185 days.

11. Breathing rates:

Time after accident	Breathing rate, 10^{-4} m^3/s
0-8 hours	3.47
8-24 hours	1.75
1-30 days	2.32

The following tabulation summarizes the results:

Distance, m (mi.)	2-hour site dose, rem		30-day LPZ dose, rem	
	Thyroid	Whole body	Thyroid	Whole body
100(1/16)	480.6	13.6	NA	NA
201(1/8)	138.1	3.9	1439.4	15.9
402(1/4)	41.8	1.2	423.0	4.7
604(3/8)	30.1	0.85	247.4	3.03
805(1/2)	22.6	0.64	166.4	2.14
1006(5/8)	17.8	0.51	124.7	1.65
1207(3/4)	14.5	0.41	97.4	1.32
1408(7/8)	12.1	0.34	79.0	1.08
1609(1.0)	10.8	0.31	68.4	0.95

The thyroid dose dominates at all distances at both boundaries; the whole body doses are always below the 20 rem limit.

These data are plotted on Figures 5.2-5 and 5.2-6. The minimum distances are (from the graphs):

Site boundary minimum distance = 201 m (660 ft)
LPZ outer boundary (minimum) = 1006 m (3300 ft)

The nearest population center (Orla, Texas) is more than 2,250 m (14 mi) from the reactor, which satisfies Criterion 3. Land controlled by the Duval Corp. is shown in Figure 5.2-7.

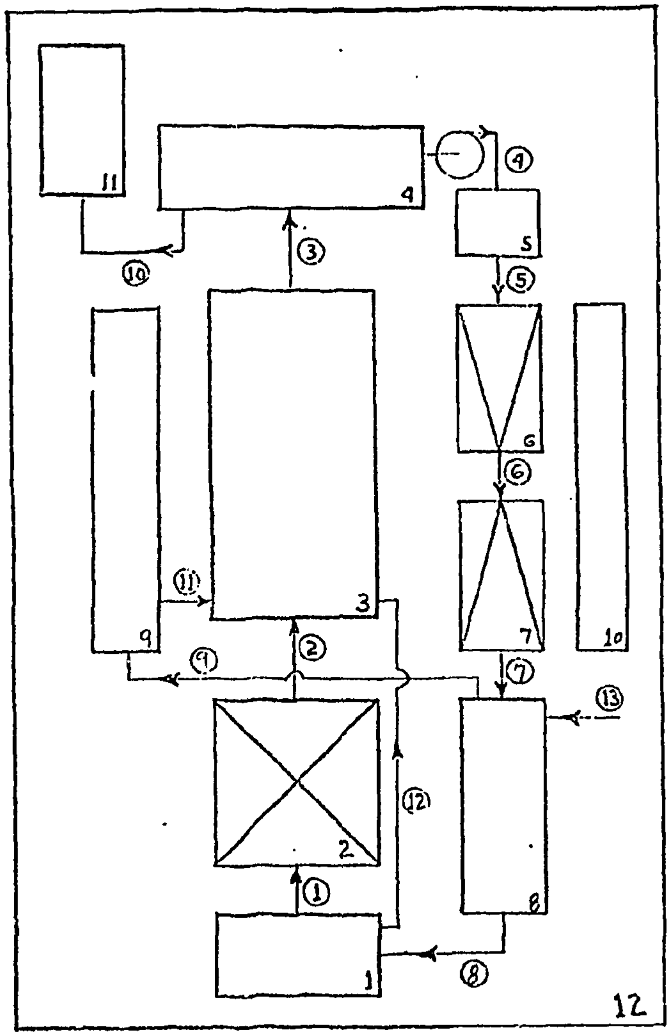
Emergency procedures must be prepared and operational status must be assessed during licensing following detailed study of all emergency conditions.

Table 5.2-1. Assumptions for Containment Response

	Metric	English
Containment dry well free volume	1062 m^3	37,520 ft^3
Containment wet well water volume	442 m^3	15,610 ft^3
Containment wet well air volume	268 m^3	9464 ft^3
Number of vent pipes	12	12
Vent pipe rupture disc setpoint	0.14 MPad	20 psid
Containment dry well cooling	One containment cooler — emergency operation	
Wet well initial water temperature	43.3C	110F
Computer code used for containment response	MVRBA	MVRBA
LOCA	Double-ended surge line break	
Discharge coefficient	1.0	1.0
Power level, % of 365 MWt	102	102
Decay power, % of recommended CNSG decay curve	120	120
Initial RCS mass	150,000 lbm	68,000 kg
ECCS	1 HPI, 1 LPI	1 HPI, 1 LPI
Flow of one HPI at 1600 psia	0.02 m^3/s	350 gpm
Flow of one LPI at 200 psia	0.03 m^3/s	500 gpm
Primary metal heat transfer coefficient:		
Primary metal to froth	28,370 $W/m^2 \cdot K$	5000 Btu/h-ft^2-°F
Primary metal to steam	11.35 $W/m^2 \cdot K$	2 Btu/h-ft-°F

Computer code used for mass and energy release: CRAFT

Figure 5.2-1. Model of CNSG Reactor Coolant System for LOCA



Node No.

1. Lower Plenum
2. Core
3. Upper Plenum
4. Closure Head
5. Pump Outlet and Steam Generator Inlet Distribution Annulus
6. Top Half of Steam Generator-Primary Side
7. Bottom Half of Steam Generator-Primary Side
8. Downcomer Annulus
9. Annulus Surrounding Steam Generator Modules
10. Secondary Side of Steam Generator
11. Pressurizer
12. Containment

Flow Path No.

1. Core
2. Core
3. From Upper Plenum to Upper Head
4. Pump
5. From Pump Outlet to Steam Generator
6. Steam Generator
7. From Steam Generator to Downcomer
8. From Downcomer Annulus to Lower Plenum
9. Bypass From Downcomer Annulus to Annulus Around Steam Generator Modules
10. Pressurizer Surge Line
11. Bypass From Annulus Around Steam Generator Modules to Upper Plenum
12. Core Bypass
13. High- and Low-Pressure Injection

Figure 5.2-2. Water Volume in Reactor Vessel During Injection Line LOCA

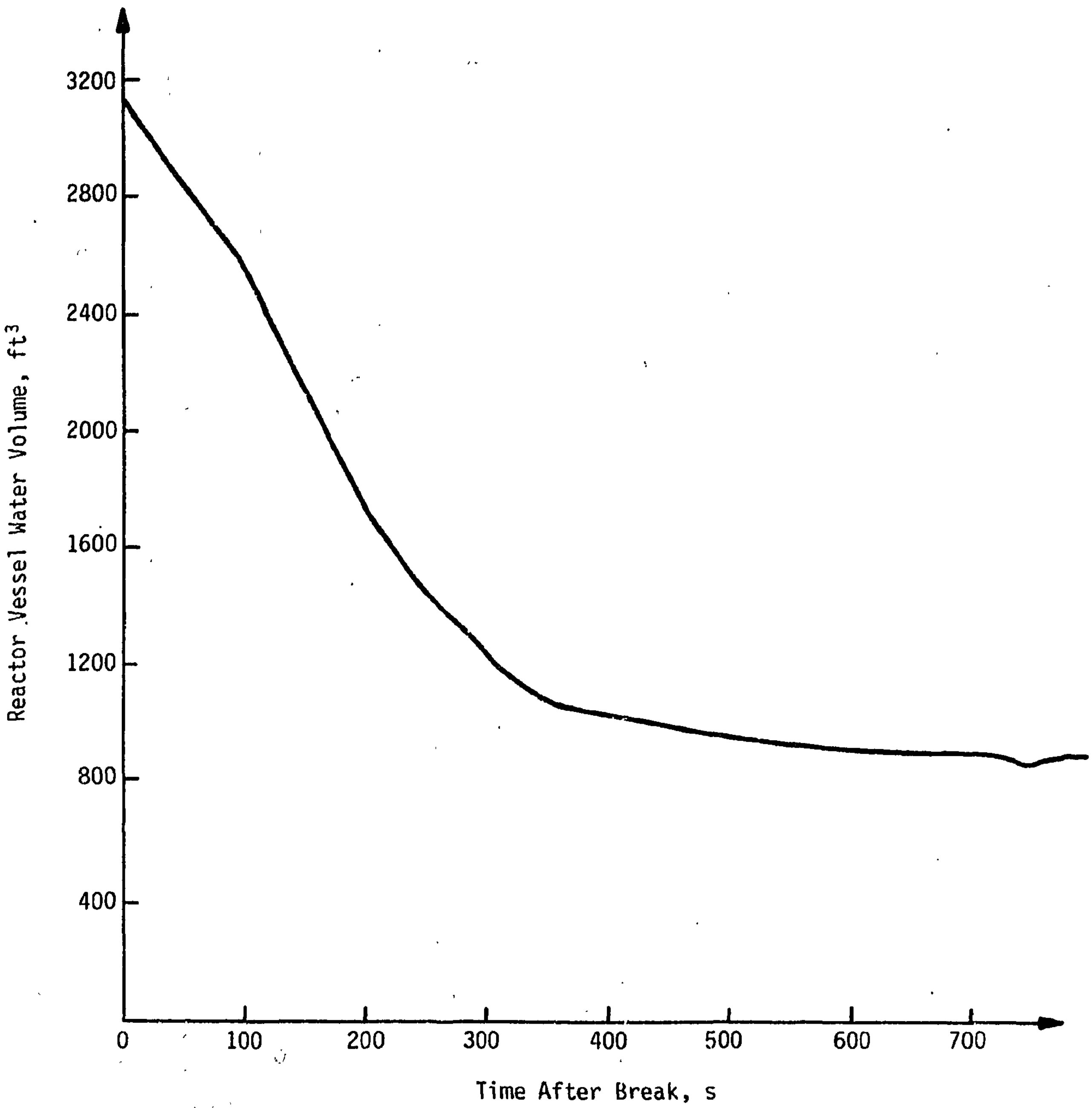


Figure 5.2-3. Pressure Vs Time After Rupture

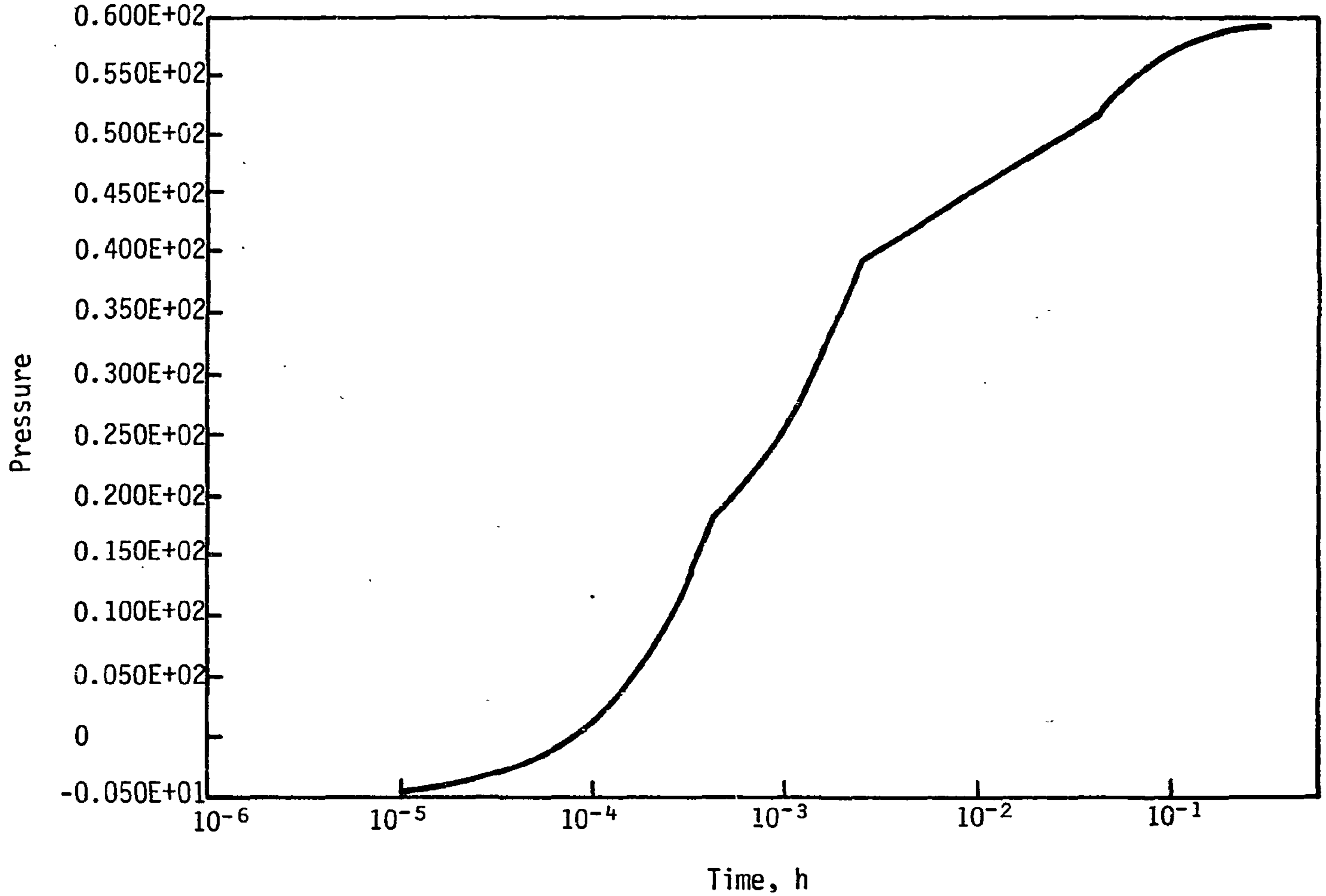


Figure 5.2-4. Temperature Vs Time After Rupture

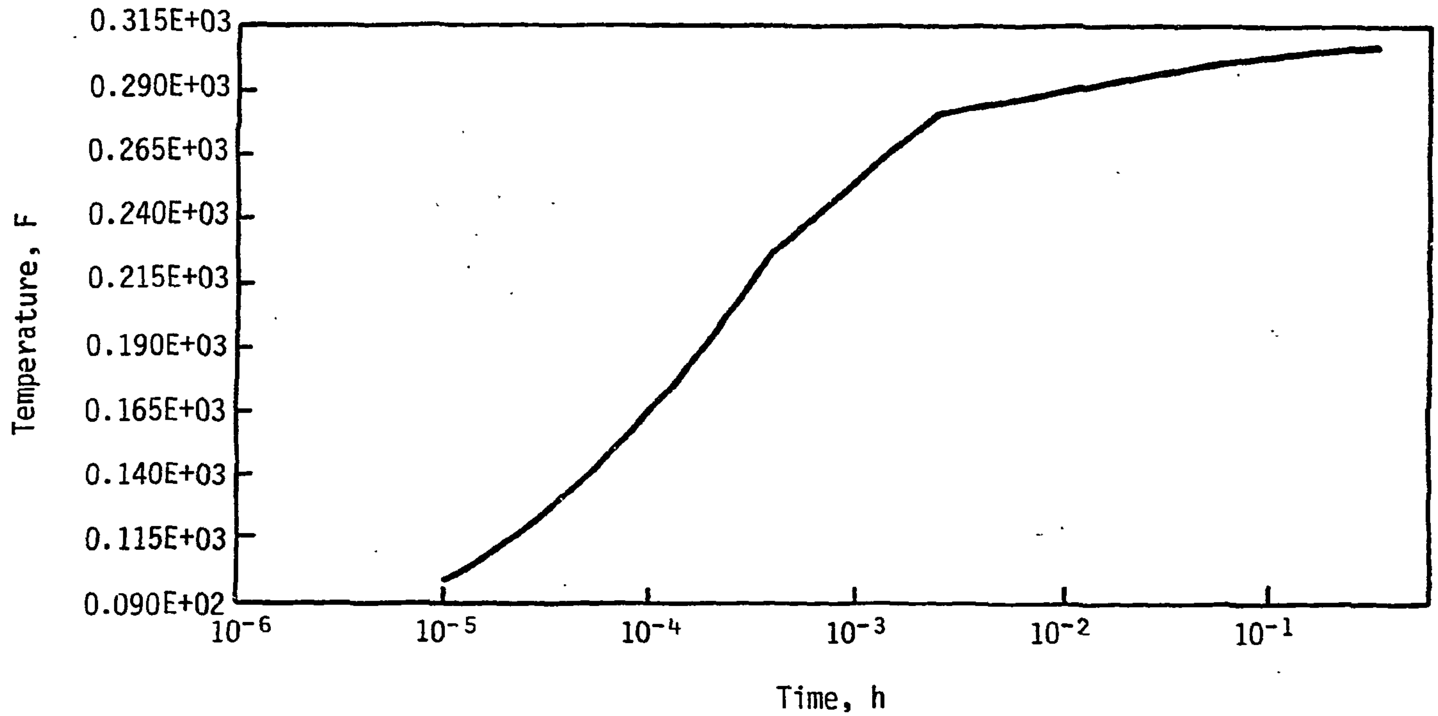


Figure 5.2-5. Two-Hour Integrated Thyroid Dose at Site Boundary Vs Distance From Reactor Building to Site Boundary Following MHA

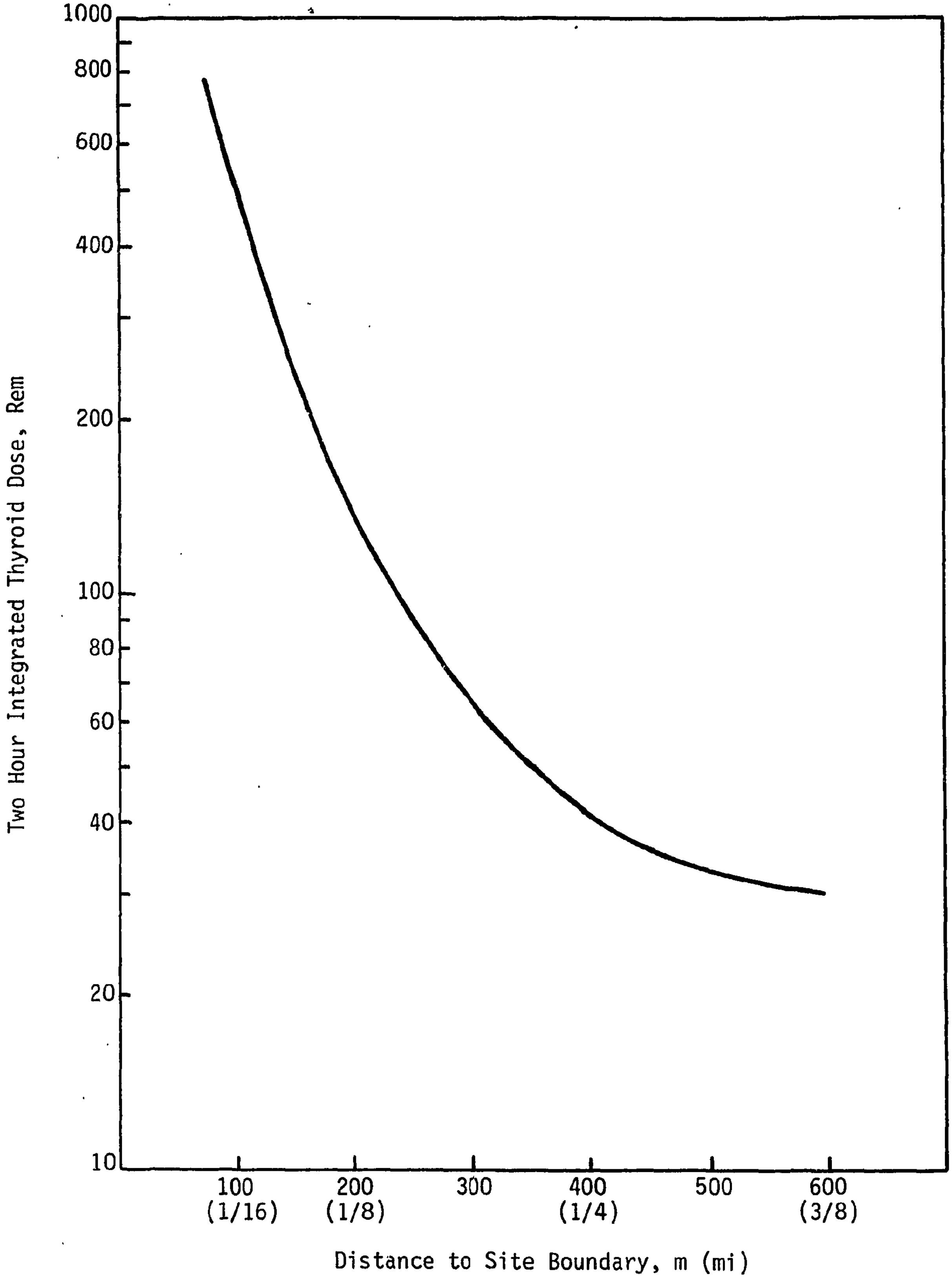


Figure 5.2-6. 30-Day Integrated Thyroid Dose at LPZ Boundary Vs Distance From Reactor Building to LPZ Boundary Following MHA

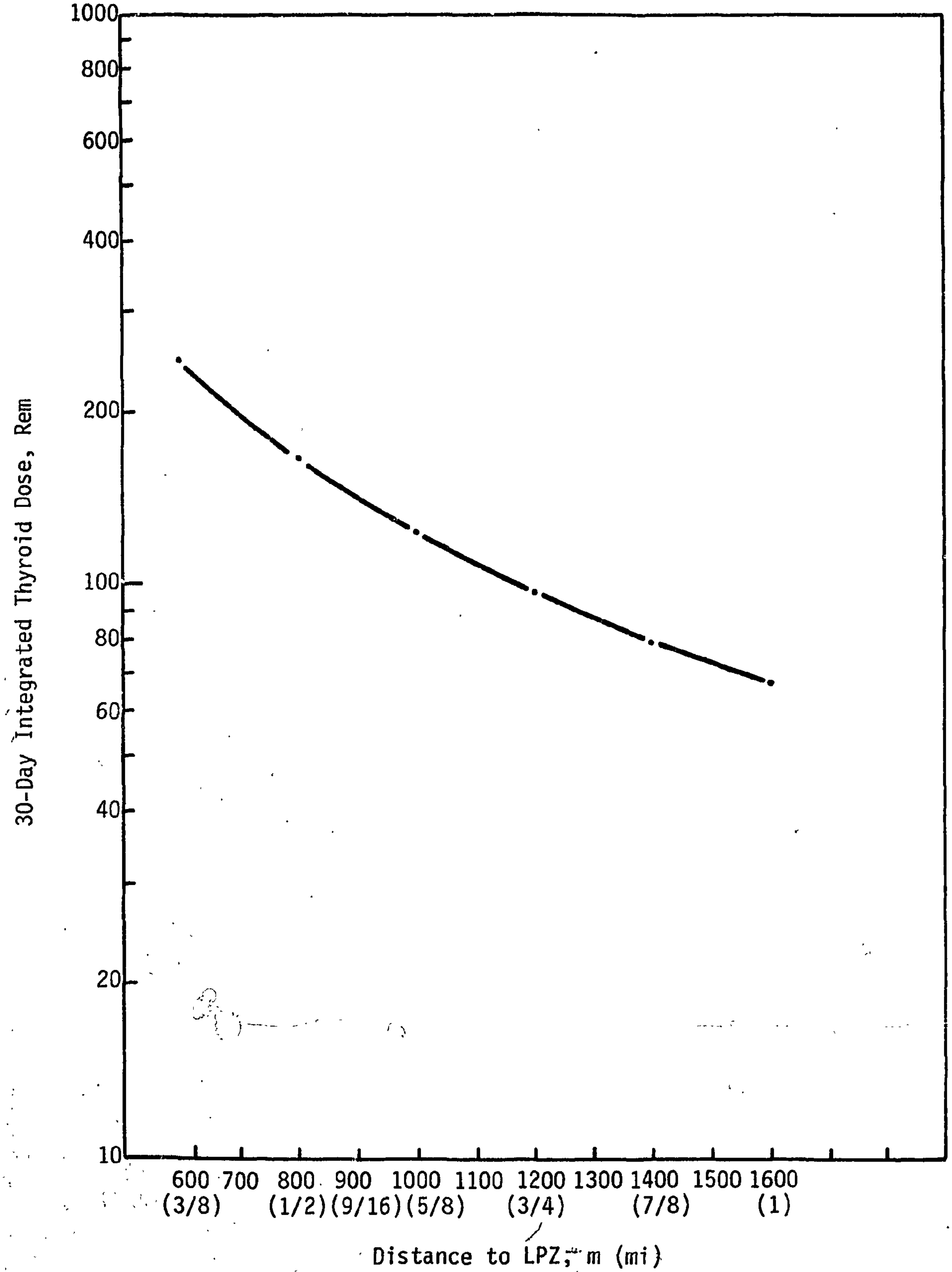


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Figure 5.

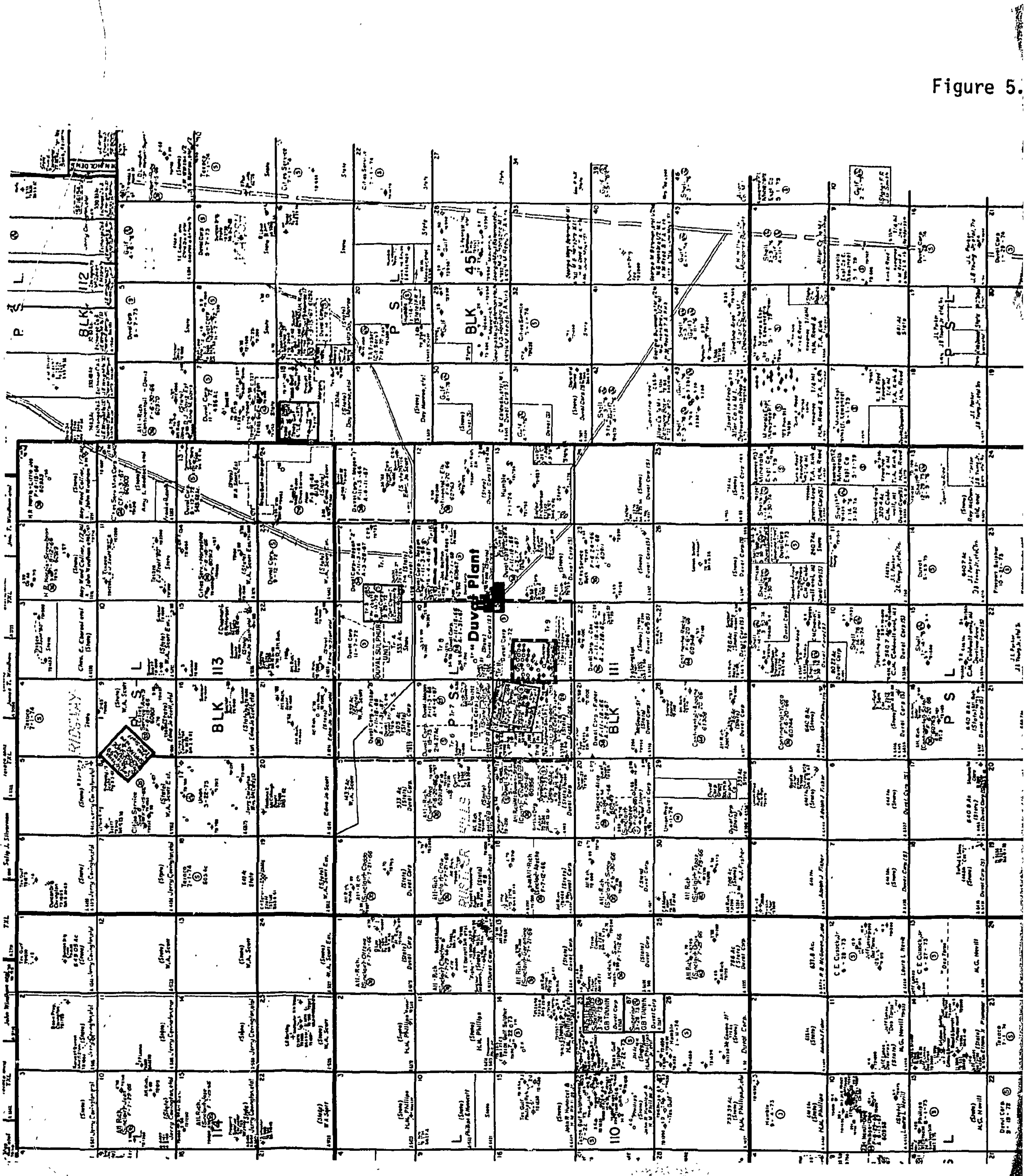

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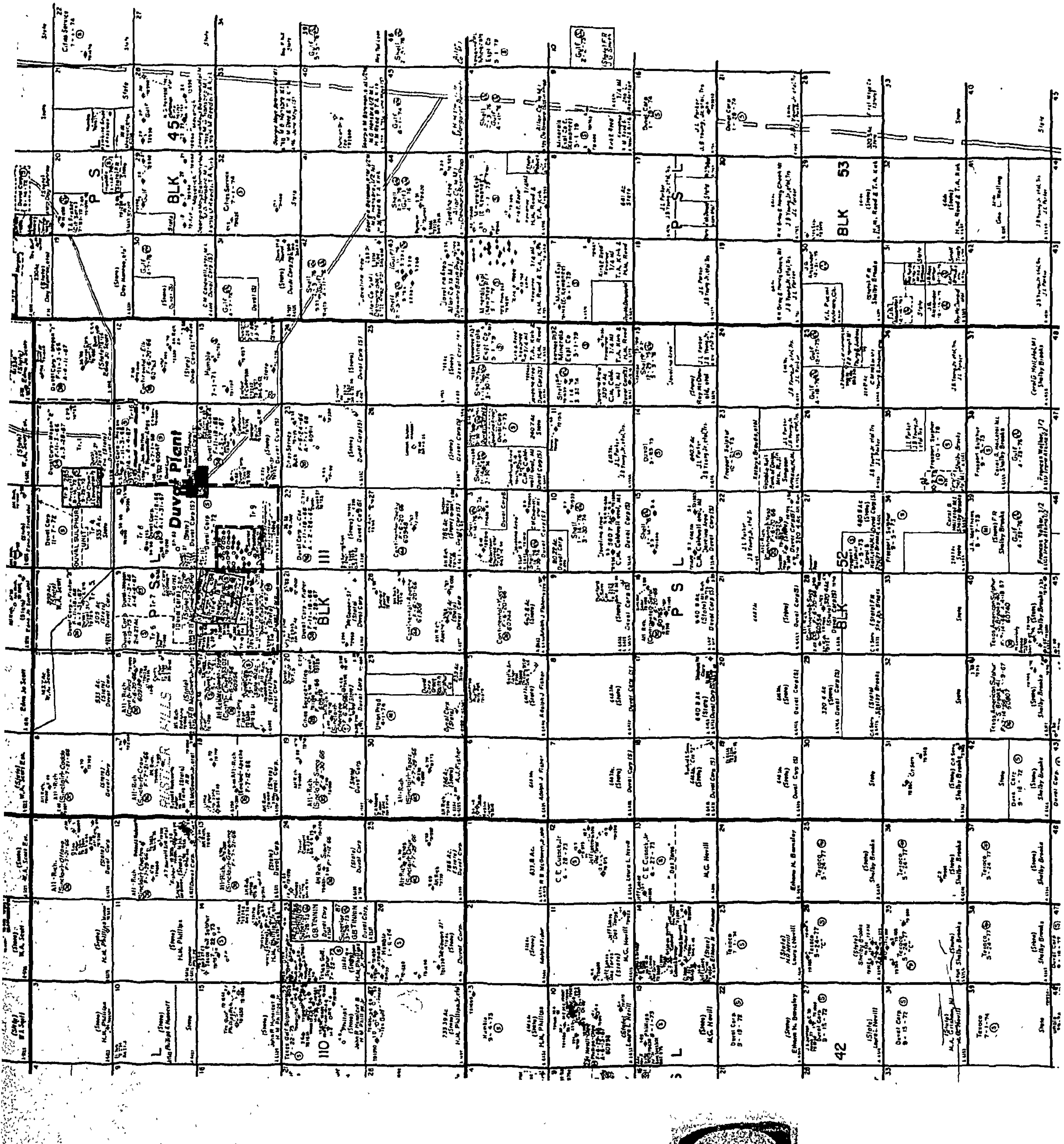
Figure 5.2-7. Land Controlled by Duval Corporation



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6. ECONOMIC EVALUATION

The objective of the economic evaluation was to compare capital and operating costs of the PE-CNSG for the particular sites identified. The earlier economic studies were based on the USNRC hypothetical Middletown site.[2] Further, specific details of owner financing were not available.

6.1. Evaluation Ground Rules

To evaluate the PE-CNSG for application to the Duval Corporation sulfur mining operation, two sets of ground rules were established. The first areas considered were the mining operation and the expected means of financing; the second area was fuel cycle costs. Table 6.1-1 presents the ground rules applicable to both capital and fuel cycle costs. Table 6.1-2 presents fuel cycle assumptions utilized.

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Table 6.1-1. Economic Evaluation Ground Rules

1. All costs based on third-quarter 1976 dollars.
2. Financing based on 100%-equity capital.
3. Interest rate during construction — 7.25%.
4. Minimum attractive rate of return (MARR) 12-15% (after tax).
5. The USNRC Account System[a] will be used for the identification and tabulation of cost estimates.
6. The evaluation will be based on the Nth-of-a-kind (NOAK) costs. First-of-a-kind (FOAK) costs will be reported separately (section 6.3.2).
7. For comparisons including escalation, the escalation rate is 6%.

[a]From reference 2.

Table 6.1-2. B&W Fuel Cycle Assumptions

Basic Costs	English	Metric
U_3O_8 price	\$40/lb	\$88.18/kg
Conversion cost	\$1.59/lb	\$3.50/kg
Enrichment cost \$/SWU	61.30	61.30
Tails concentration, % ^{235}U	0.25	0.25
Fuel recovery cost	\$90.72/lb	\$200/kg
Fissile plutonium value	\$13,608/lb	\$30/g
Fuel discount rate, %	9.575	9.575
Lead and Lag Times, months		
Prior to batch insertion		
U_3O_8 purchase and conversion	9	
Uranium enrichment	6	
Fuel fabrication	3	
Subsequent to batch withdrawal		
Reprocessing	6	
Fissile material credit	9	
Losses of Fertile and Fissile Materials, %		
New fuel		
Conversion	1.0	
Fabrication	0.5	
Recycled fuel		
Reprocessing	1.0	
Refabrication	1.0	

6.2. Cost Estimate/Analysis — PE-CNSG

Total plant capital costs have been developed for the PE-CNSG plant. Costs of NSS components, plant equipment, structures, material, and labor for installation were developed. Cost estimates were based on past experience, studies by others, shop material and labor estimates, and equipment vendor estimates. Previous estimates[1] were revised to reflect the reference power level of 365 MWt.

A uniform system of accounts for reporting construction and operating costs for nuclear power plants and related transmission and general plant facilities was used in deriving direct plant costs. This system of accounts is presented in detail in NUS-531[2]. First-of-a-kind (FOAK) and Nth-of-a-kind (NOAK) costs were determined and are discussed in section 6.2.2 and 6.2.1, respectively.

6.2.1. NOAK Costs

In general, NOAK costs are lower than FOAK costs due to the elimination of non-repetitive, first-time engineering and to labor learning experience. B&W NOAK equipment costs are lower due to a B&W shop labor learning curve and to the elimination of first-time engineering. These improvement factors are based on past B&W experience with central station plant engineering and equipment. A reduction in UE&C equipment scope costs results from field labor learning where there is a carryover of supervisor personnel from one project to another.

The scope of responsibility for cost estimates is presented in Table 6.2-1 for B&W and UE&C. Certain data are listed as the responsibility of a particular organization for review.

A summary of PE-CNSG, NOAK plant costs is presented in Table 6.2-2. Costs are broken down into major accounts as suggested in NUS-531[2]. The cost tabs are further separated in Table 6.3-3 to identify depreciable and nondepreciable items and the portion of depreciable items subject to ITC. Total capital cost for the nuclear process steam supply is 142 million dollars (1976 dollars).

Interest during construction was calculated using the typical progress payment schedules of B&W and UE&C and the construction schedule presented in Appendix B. The DON computer code,[8] which computes interest as a percentage of total cost for the selected construction period, was used to obtain interest during construction (IDC). Tables 6.2-4 and 6.2-5 are the progress payment and IDC data for B&W and UE&C equipment, respectively.

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6.2.2. First-of-a-Kind Costs

The nuclear plant costs (Table 6.2-2) and the overall economic comparison (section 6.4) are based on Nth-of-a-Kind costs and thus include no FOAK expenses. For this study it has been assumed that the FOAK costs would be borne by others (U. S. government, EPRI, etc.). This type of arrangement for the first plant is necessary if nuclear power is to become a viable energy source for progress energy use.

FOAK is defined as follows: those work activities that are nonrepetitive for follow-on units, including engineering, licensing, and test and evaluation efforts required to develop design parameters, to demonstrate safety to the regulatory authorities, and to verify design adequacy.

Generic activities as defined herein include fabrication processes, baseline component and system design, resolution of generic licensing issues, and FOAK engineering proof tests and evaluation programs. Detailed FOAK work items for the first unit include (but are not limited to) the following tasks.

6.2.2.1. Engineering

1. Development of reactor baseline design (component and system specifications and drawings).
2. Design and stress analyses required to satisfy regulatory agencies.
3. Development of reactor plant checkout, startup, and operating procedures.
4. Support of test and evaluation programs.
5. Architectural and construction design.
6. Development of BOP systems and equipment designs.
7. Development of BOP checkout, startup, and operating procedures.

6.2.2.2. Fabrication and Construction

1. Manufacturing development for welding procedures, special fixtures, and ASME Code cases.
2. Development of special fabrication processes for shop and field construction.
3. Preparation of detailed shop processing and construction schedules.

6.2.2.3. Licensing

1. Resolution of generic issues related to the class of reactors.
2. Determination of necessary supportive, environmental monitoring programs.
3. Prepare generic parts of PSAR and FSAR.

6.2.2.4. In-service and Initial Operational Tests and Inspections

1. Baseline techniques for code in-service inspection.
2. Flow-induced vibration evaluation of reactor internals.
3. Hot-functional test programs (field).

6.2.2.5. Hardware

1. Design and manufacture of special tools and handling equipment for major components.
2. Design and construction of fueling and refueling equipment and special tools.

In addition to the FOAK work items discussed above, certain test and evaluation programs are required to verify design adequacy or to demonstrate the margin of conservatism of the design. The test and evaluation programs also support the licensing processes.

6.2.2.6. Test and Evaluation

1. Steam generator functional performance, secondary side flow distribution, and downcomer performance.
2. Steam generator fouling and chemical cleaning.
3. Upper internals vibration.
4. Control rod guide structure.
5. Fuel assembly prototype detail design and fabrication.
6. Fuel assembly life test.
7. Primary pump prototype.
8. Pressure suppression containment.
9. Reactor coolant temperature sensor.
10. Containment pressure suppression tests (not included as part of the cost estimate pro[illegible]

The total plant FOAK cost is estimated to be approximately 25 to 50 million dollars, including all FOAK costs in the areas of engineering, shop and field construction, licensing, in-service and initial operational tests, and inspections and hardware design and manufacture. This total plant FOAK estimate assumes that all FOAK costs are applied to a single program and concept. In reality many are common to three programs involving integral nuclear steam systems of similar or identical design: the PE-CNSG, the Maritime M-CNSG, and the higher-power-level CNSS concept. All three program studies and design activities have been supported, at least in part, by federal agency funding. A construction project involving any one of these programs would give impetus to the others, so that some sharing of these FOAK costs over a period of six to eight years between programs can be considered. In this respect FOAK government support of these programs should be especially cost effective. If FOAK costs could be shared between programs, the estimated range of FOAK costs above for the PE-CNSG could be reduced correspondingly.

The FOAK cost estimates discussed above are not based on an extensive investigation in this study but rather on work previously done in the Phase I study and in M-CNSG program activities. If an industrial process energy user should decide to proceed further with this study, the BOP FOAK costs should be determined in more detail.

The previous estimates do not include consideration of government legislation to provide nuclear accident liability insurance similar to Price-Anderson legislation. This would be required to cover industrial organizations as an incentive to establish nuclear plants for initial industrial installations. The estimates also exclude the cost of longer, first-time construction schedules and resultant increases in interest during construction for pioneer plants.

Table 6.2-1. Cost Estimate Scope of Responsibility

Item	Responsibility B&W	Responsibility UE&C
Land and Land Rights*	X	
Structures and Improvements		X
Yard work		
Containment structure		
Reactor service building		
Control building		
Diesel generator and fuel oil building		
Administration building		
Process building		
Process heat service building		
Water treatment building		
Service water intake		
Reactor Plant Equipment		✓
Nuclear steam supply equipment	✓	
Reactor equipment		✓
Reactor coolant system		✓
Safeguards cooling system		✓
Radioactive waste treatment and disposal systems		✓
Nuclear fuel handling and storage systems		✓
Nitrogen and hydrogen gas system		✓
Coolant purification and chemical treatment system		✓
Component cooling system		✓
Miscellaneous plant equipment		✓
Miscellaneous items		✓
Ultimate heat sink		✓
Water treatment system		✓
Instruments and controls		✓

*Data supplied by Duval Corporation.

Table 6.2-1. (Cont'd)

Item	Responsibility B&W	Responsibility UE&C
Process Energy Equipment		
Secondary system		X
Tertiary system		X
Reboilers	X	
Other process steam equipment		✓
Electric Plant Equipment		✓
Switchgear		
Station service equipment		
Switchboards		
Protective equipment		
Electrical structures and wiring containers		
Power and control wiring		
69 kV transmission ans substation		
Miscellaneous Plant Equipment		✓
Transportation and lifting equipment		
Air, water, and steam service system		
Communication equipment		
Furnishings and fixtures		
Undistributed Cost		
Engineering and home office services		✓
Field supervision, quality control and job office expense		✓
Temporary facilities		✓
Construction equipment		✓
Construction services		✓
NSS engineering/project management	✓	
NSS services	✓	
Other Plant Cost		
Licensing and public relations expense, operator training, and spare parts		✓
NSS licensing	✓	

Table 6.2-2. Nuclear Plant Capital Investment Summary (Budgetary Estimate)

Account No.	Title	Total cost, 10^3 \$
Direct cost		
20	Land and land rights	80
26	Special materials	0
21	Structures and improvements	24,245
22	Reactor plant equipment	34,937
23	Turbine plant equipment	0
24	Electric plant equipment	10,902
25	Miscellaneous plant equipment	2,968
27	Process energy equipment	8,946
Total direct cost		82,078
Indirect cost		
91	Construction facilities equipment and services	8,903
92	Engineering services	13,246
93	Other costs	5,400
94	Interest during construction	13,143
95	Contingency	18,750
Total indirect cost		59,442
Plant capital investment (1976)		141,520
Plant capital investment (1982)		200,746

Table 6.2-3. Depreciable and Nondepreciable Costs

Depreciable costs	10^3 \$
Structures and improvements	
Reactor plant equipment	
Electric plant equipment	
Miscellaneous equipment	
Process energy equipment	
Spare parts	
Construction facilities and equipment	
Total depreciating	87,968
Nondepreciable costs	**10^3 \$**
Construction services	
Land and land rights	
Engineering services	
Contingency/gross margin	
Interest during construction	
Other costs	
Working capital	
Total nondepreciating	54,291
Portion of depreciable items subject to ITC	65,481

Table 6.2-4. Progress Payment and IDC Rates — B&W Equipment

		Annual interest, 7.25%	
Percent time	Percent cost	Construction period, months	Interest during construction, %
0.0	0.000	30	6.51
5.0	0.313	33	7.19
10.0	0.625	36	7.88
15.0	0.625	39	8.58
20.0	2.083	42	9.28
25.0	4.688	45	9.98
30.0	8.750	48	10.70
35.0	12.500	51	11.42
40.0	14.938	54	12.14
45.0	17.500	57	12.87
50.0	21.313	60	13.61
55.0	27.375	63	14.36
60.0	33.938	66	15.11
65.0	41.250	69	15.87
70.0	49.063	72	16.63
75.0	57.438	75	17.41
80.0	73.594	78	18.18
85.0	79.375	81	18.97
90.0	92.875	84	19.76
95.0	99.313	87	20.56
100.0	100.000	90	21.37
		93	22.19
		96	23.01
		99	23.84
		102	24.52
		105	25.52
		108	26.37
		111	27.23
		114	28.10
		117	28.97
		120	29.86

Table 6.2-5. Progress Payment and IDC Rates — UE&C Equipment

Percent time	Percent cost	Annual interest, 7.25% Construction period, months	Interest during construction, %
0.0	0.000	30	8.69
5.0	0.540	33	9.61
10.0	1.030	36	10.53
15.0	2.700	39	11.47
20.0	4.728	42	12.41
25.0	7.060	45	13.86
30.0	10.600	48	14.33
35.0	14.780	51	15.30
40.0	19.890	54	16.28
45.0	27.710	57	17.27
50.0	36.410	60	18.27
55.0	50.540	63	19.28
60.0	62.500	66	20.30
65.0	72.550	69	21.33
70.0	80.980	72	22.37
75.0	87.170	75	23.43
80.0	91.840	78	24.49
85.0	95.100	81	25.56
90.0	97.820	84	26.64
95.0	99.460	87	27.74
100.0	100.000	90	28.84
		93	29.96
		96	31.08
		99	32.22
		102	33.37
		105	34.53
		108	35.71
		111	36.89
		114	38.09
		117	39.30
		120	48.52

6.3. Cost Estimate/Analysis — Fossil Plants

6.3.1. Coal-Fired Plant

Capital costs for the coal-fired alternative were adapted from data supplied by the Duval Corporation and from boiler costs reported by United Engineers and Constructors in reference 9. The steam generating system comprises three 400,000-lb/h boilers and auxiliary equipment, coal and ash handling equipment, a fuel oil system, and a feedwater treatment system. The plant produces steam at 150 psig (saturated), matching the output of existing gas-fired boilers. The boilers are capable of burning western low-sulfur, sub-bituminous coal as produced in the Powder River Basin (Wyoming, Montana). This type of coal is believed to be representative of western coal since the Powder River Basin is a major source of coal with a long-term resource base. Ash content is 6%; sulfur amounts to 0.4%, so that stack gas scrubbers were not provided for. The high heat value ranged from 7500 to 8600 Btu/lb; a nominal value of 8000 Btu/lb was selected for the economic analysis.

Table 6.3-1 summarizes the capital costs for the coal-fired plant. The table does not include the cost of existing facilities that would be retained and utilized after the conversion to coal fuel. Included in this category are the feedwater treatment system, fuel oil system, electrical system, etc. The total capital cost amounts to 49 million dollars in 1976 dollars.

6.3.2. Oil-Fired Plant

The present gas-fired boilers are already equipped for burning No. 2 oil; however, the Struthers water heaters would have to be modified if natural gas fuel were replaced with oil. Table 6.3-2 summarizes the relatively minor capital costs for the conversion to oil fuel, which amounts to $589,000 in 1976 dollars.

Table 6.3-1. Capital Costs — Coal-Fired Alternative (Three 400,000-lb/h Boilers)

Account No.	Title	Total cost, $\$ \times 10^3$
11	Structural improvements	
	Coal storage pile and yard work	170
	Railroad (12,000 lineal ft)	540
	Structure — foundations and super structure	1,800
	Total	2,510
12	Boiler plant equipment	
	Boiler accessories, pulverizers, and fans	21,300
	Precipitators	4,500
	Draft flue and breeching	300
	Stack (400') and foundation	1,240
	Coal fuel equipment	3,540
	Fly ash and dust handling	575
	Instrumentation and control	1,300
	Feedwater and treatment system (tie into existing)	200
	Total	32,995
15	Electric accessories — coal plant	1,095
	Direct total	36,560
	Indirects	
	Licensing	300
	Construction site costs	695
	Spare parts (2% direct materials)	523
	Engineering and construction management	2,900
	Contingency (10% direct costs)	3,650
	Indirect total	8,068
	Total capital cost	44,628
	Interest during construction	4,861
	Total in 1976 (1976 dollars)	49,489
	Total in 1982 (1982 dollars)[a]	70,200

[a] Escalation 1976-1982 is 6%/year.

Table 6.3-2. Capital Costs — Oil-Fired Alternative

Account No.	Title	Total cost, $ × 10^3	
12	Boiler plant equipment		
	Existing oil storage increased from 7 to 14 days at 10¢/gal or $4.20/bbl	72	7 days
	Dyke transfer pumps, piping, etc.	72	--
	Day tank — 60,000 gal at $0.25	15	--
	Struthers heater conversion from gas to oil firing — 12 units at $15,000 each	180	--
	Instrumentation and controls (including SO_2 and NO_x monitoring)	50	--
15	Accessory electric — allowance	50	--
	Total directs	439	
	Contingency — 15%	50	
	Owner's indirect (engineering, management, etc.)	100	∿20%
	Total in 1976 (1976 dollars)	589	
	Total in 1982 (1982 dollars)[a]	835	

[a]Escalation 1976-1982 is 6%/year.

6.4. Fuel Costs

The ground rules used to calculate the nuclear fuel cycle costs are given in Table 6.1-2; the method used to obtain unit fuel cost is shown in Figure 6.4-1. The fuel cycle considered is shown in Figure 6.4-2; the CYCO computer code[10] is used to calculate fuel costs. After the levelized fuel cycle costs are obtained, the following method is used to determine the effect of investment tax credit:

1. Obtain levelized fuel cycle cost from CYCO.
2. Determine the present worth, incore capital charges, and total expenses for an equilibrium cycle, and average these values for the equilibrium cycle.
3. Multiply the levelized fuel cycle costs by the ratio of the incore capital charges to the expenses determined in step 2; this yields the portion of the levelized fuel costs attributable to incore capital charges.
4. Subtract the result of step 3 from the levelized cost to determine the depreciable cost.
5. Multiply the depreciable cost by the investment tax credit rate adjusted for the total number of fuel assemblies not subject to the credit (yields the levelized credit).
6. Subtract the levelized credit from the levelized fuel cost to obtain the levelized fuel cost including the investment tax credit.

The levelized fuel cost for the reference annual refueling cycle amounted to 57.4¢/10^6 Btu in 1976 dollars. This assumes PE-CNSG operation at 365 MWt and a plant factor of 0.8. The ground rules for the fuel cycle were established when the reprocessing of spent fuel was anticipated; thus, the fuel costs reflect this assumption. Fuel costs might increase by 5 to 10¢/10^6 Btu if spent fuel is not reprocessed; however, this slight cost increment would not alter the conclusions of this study.

The fuel costs for the nuclear and fossil-fired alternatives are summarized in Table 6.4-1. In this comparison the coal-fired, oil-fired, and nuclear plants were assumed to deliver equal, annual amounts of energy to the industrial process. The thermal efficiency of the fossil-fired boilers/heaters was taken to be 85%. A range of coal prices from about $15 to $40 per ton was selected to account for the uncertainty in market conditions and coal transportation costs. The coal costs are equivalent to 0.91 to 0.245 (1976 dollars) per million Btu delivered to the industrial process. Oil costs are based on

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a price of 33¢/gal (1976 dollars — No. 2 oil) suggested by the Duval Corporation.

Table 6.4-1. Fuel Costs[a]

	Coal[b]	Oil[c]	Nuclear
Dollars in 1976			
Unit cost	14.50-39.17 \$/T	13.85 \$/bbl	--
10^6 \$/yr	9.3-25.1	23.7	5.0
¢/10^6 Btu del.	91-245	272	57
Dollars in 1982[d]			
Unit cost	20.57-55.56 \$/T	19.65 \$/bbl	--
10^6 \$/yr	13.2-35.6	33.6	7.1
¢/10^6 Btu del.	128-348	385	81

[a] Annual process energy requirement — 8.73×10^{12} Btu/yr equivalent to 365 MWt; 0.80 plant factor; fossil boiler efficiency — 85%.

[b] Low sulfur coal — 8000 Btu/lb.

[c] 33¢/gal — 6×10^6 Btu/bbl.

[d] Escalation 1976-1982 is 6%/year.

Figure 6.4-1. Fuel Cost Summary

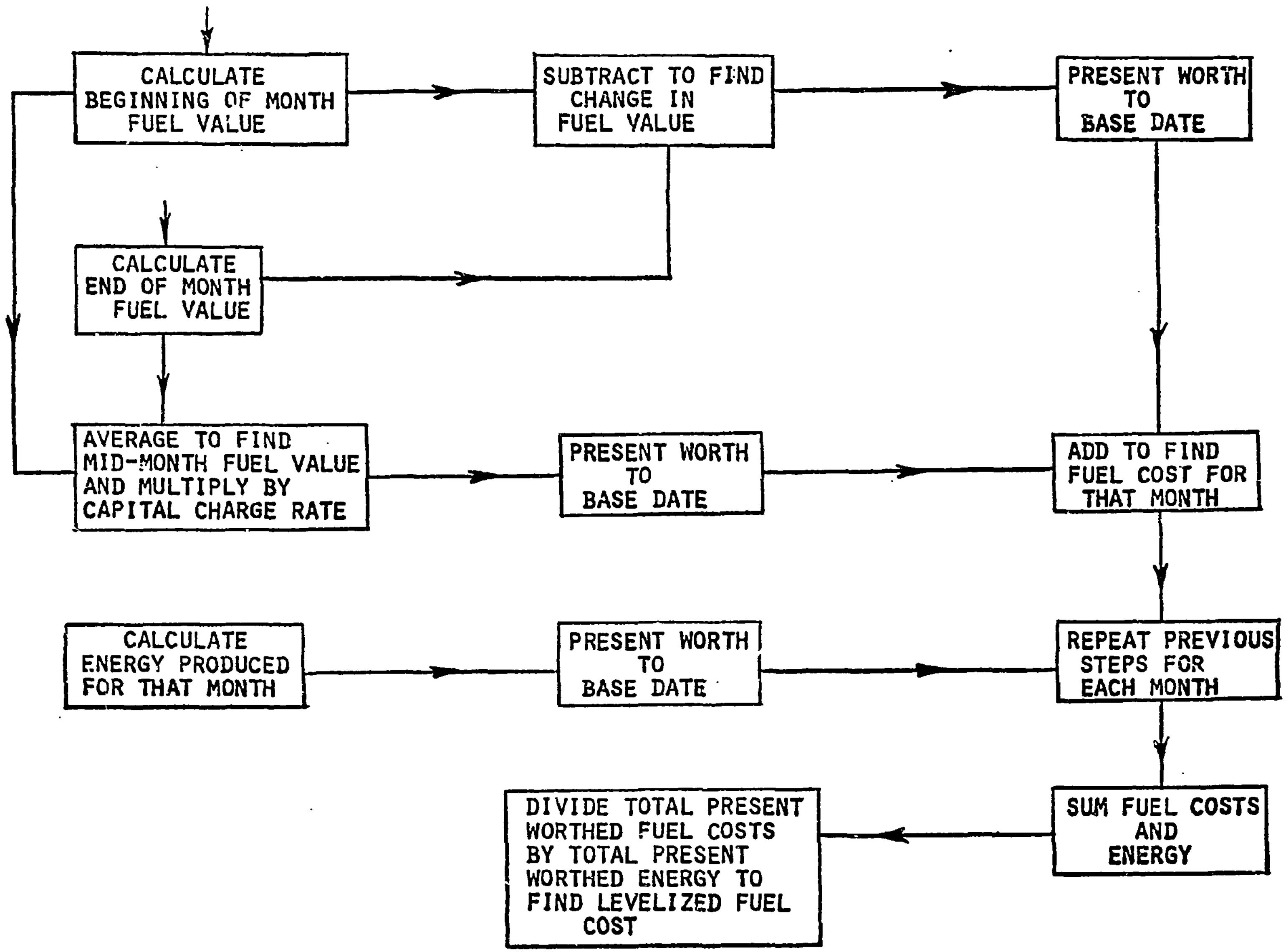
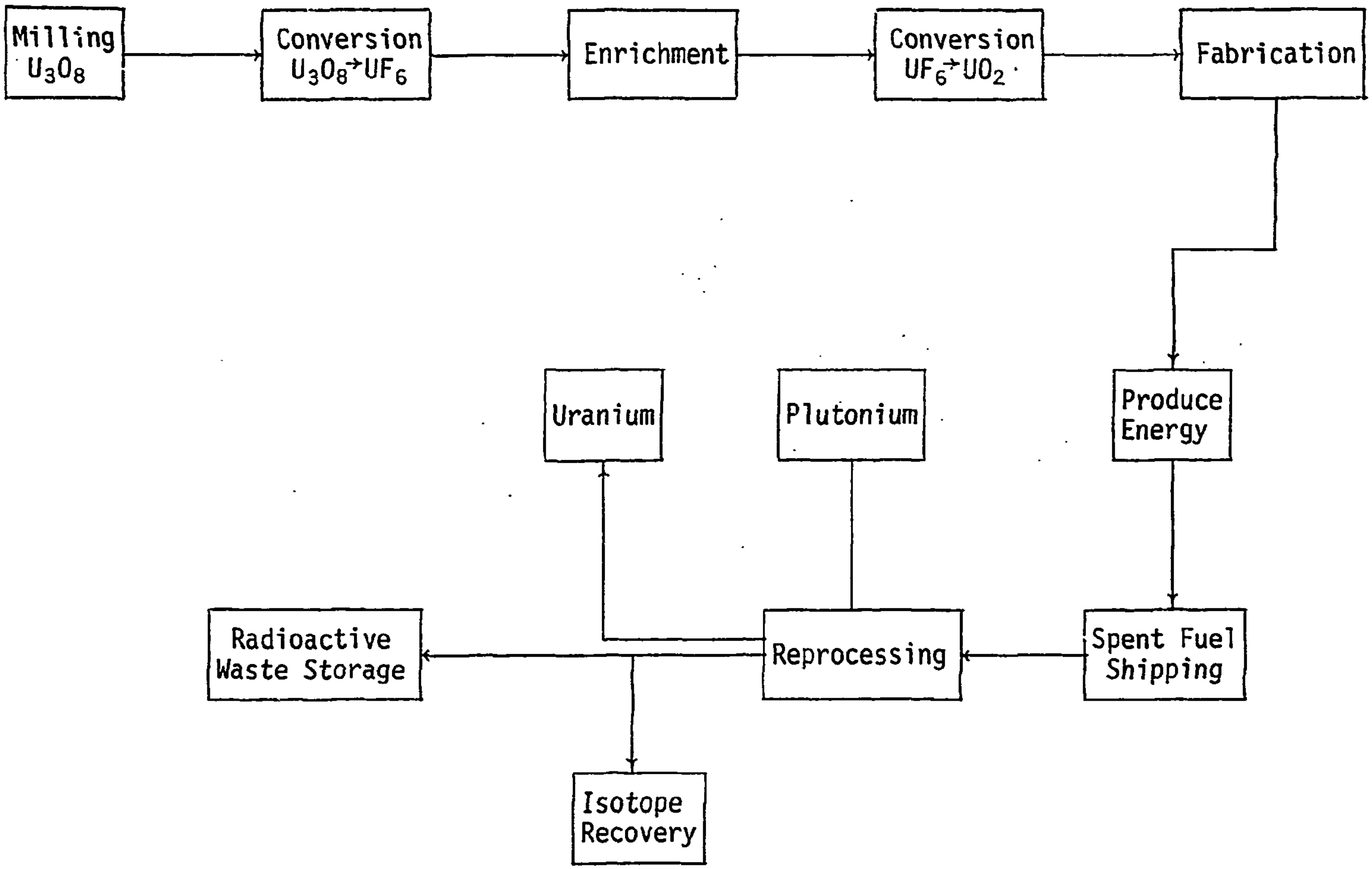


Figure 6.4-2. Nuclear Fuel Cycle



6.5. Operating and Maintenance Costs

Annual operating and maintenance costs are summarized in Table 6.5-1. Values are shown in terms of 1976 and 1982 dollars. The cost for the nuclear plant cost does not include an allowance for augmented security forces that may be required by regulatory agencies for some reactor sites.

Table 6.5-1. Annual Operating and Maintenance Costs ($\$ \times 10^3$) — Fossil and Nuclear Plants

	Coal	Oil	Nuclear
Staff	853	700	1,000
Maintenance (less staff)	500	325	600
Supplies and expenses	190	175	215
Administration and general	120	100	140
Nuclear insurance			
Commercial			277
Governmental			13
Operating fee			24
Total in 1976 dollars	1,663	1,300	2,269
Total in 1982 dollars[a]	2,359	1,844	3,218
Ash disposal			
1976	533		
1982[a]	756		
Total in 1976 (1976 dollars)	2,196	1,300	2,269
Total in 1982 (1982 dollars)	3,115	1,844	3,218

[a] Escalation 1976-1982 is 6%/year.

6.6. Economic Analysis of Alternatives

The relative economic advantages (or disadvantages) for process energy alternatives being investigated at Duval's sulfur mine are computed by using the after-tax, incremental, rate-of-return approach, a form of discounted cash flow analysis. This method of economic evaluation is that used by the Duval Corporation in its capital budgeting studies.

Year-by-year, after-tax cash flows for each alternative over a 22-year period are prepared from estimates of capital costs, operations and maintenance expenses, and fuel costs presented in previous sections. Next, these cash flows are analyzed in terms of the then-current dollars. From these tables of year-by-year, after-tax cash flows, it is possible to examine the relative economics of each pair of alternatives and to compute the rate of return being realized (if any) on the more capital intensive system.

The basic cost tradeoff in generating process energy at the Culberson County mine lies in lower fuel costs associated with the more capital intensive nuclear system (PE-CNSG) when compared to fossil-fired alternatives. These savings in annual costs make possible a positive rate of return on the extra investment required by the PE-CNSG system, and the magnitude of this return is heavily dependent on future coal and oil prices, the inflation rate, and the cost of money.

Current expectations for future inflation range from about 5 to 7%; in the present analysis, an average rate of 6% inflation per year was assumed. The corresponding current yields for the chemicals and petroleum refining industries are (in anticipation of the inflation rates above) about 8.3% for debt and 15.3% for equity (after taxes). In the present calculation incremental rates of return are computed (assuming a prevailing inflation rate of 6% per year) for a range of coal prices. The resultant, after-tax rates of return can then be compared with Duval's minimum attractive rate of return (MARR), which is customarily 12-15% on new capital outlays to determine the financial attractiveness of the nuclear and fossil alternatives.

Note that in many "must-have" types of projects (as opposed to the "would-like-to-have" types of projects) the MARR is often not stringently maintained, thus permitting considerations (other than economic) to determine the final, recommended system.

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6.6.1. PE-CNSG Cash Flows

From capital cost estimates given in section 6.2.1, the approximate, end-of-year cash flow, including interest during construction, has been calculated and is presented in Table 6.6-1 based on data given in Table 6.2-2. Also show in Table 6.6-1 are investment tax credits that could be taken by Duval over an assumed, 72-month construction period. These estimates have been placed in the appropriate locations in Table 6.6-2 where after-tax cash flows are computed at a 6% annual inflation rate.

The BTCF column in Table 6.2-2 indicates before-tax cash flows consisting of inflated (or then-current) costs of nuclear fuel, O&M, and miscellaneous items. Depreciation is calculated by the straight-line method (with no assumed salvage value) over the 22-year plant life. Accelerated depreciation methods, such as the sum-of-the-years digits method, could have been used, but the straight-line method is customary practice for Duval Corporation. (A faster write-off method would favor the more capital-intensive alternative in a relative sense; Table 6.6-2 offers a conservative analysis of the PE-CNSG system.)

A combined federal and state effective income tax rate was assumed to be 0.50, and resultant tax credits are calculated in column 5 based on reductions to taxable income shown in column 4. Actually, these credits would be employed to offset tax liabilities created by annual revenues arising from the sale of sulfur.

Finally, the after-tax cash flows (ATCF) are estimated in column 6. It is this column of numbers that will be compared with ATCFs of fossil alternatives in determining an incremental rate of return on the extra investment required by the CNSG system. Note that 100% equity financing has been assumed here to simplify the analysis; this corresponds to Duval's method of making preliminary economic evaluations.

6.6.2. Fossil-Fired Plant Cash Flows

From capital cost estimates given in section 6.3.1, cash flows during construction of the coal-fired boilers, as well as the allowable investment tax credits have been computed. These estimates are shown in Table 6.6-3, and progress payments over the assumed 48-month construction period were based on Table 6.2-4. After transferring numbers from Table 6.6-3 to the BTCF column of Table 6.6-4, the process of determining after-tax cash flows can proceed in the same manner as for the nuclear alternative (Table 6.6-2). Again, it was assumed that the plant would have a zero salvage value at the end of its 22-year life.

Three sets of coal prices (1976 dollars) were analyzed: $14.50, $29.62, and $39.17/ton. This was done to provide some feel for sensitivity of the incremental rate of return to changes in this very important factor. Complete calculations for only one case, namely $14.50/ton, are given (Table 6.6-4) in the interest of brevity. An identical procedure to that of Table 6.6-3 was followed in analyzing the oil-fired alternative, and its after-tax cash flows at 33¢/gal were also computed in the same manner as that indicated in Table 6.6-4.

6.6.3. Cost Comparison Between Nuclear and Fossil-Fired Alternatives

After tables of ATCFs for the five alternatives (1 nuclear, 3 coal, and 1 oil) were developed, incremental, after-tax cash flows were determined on which a rate of return could be calculated. To illustrate one of the four incremental comparisons made, Table 6.6-5 is presented. The year-by-year entries in this table are found by subtracting the ATCF entries in Table 6.6-4 (coal at $14.50/ton) from the corresponding, annual ATCF amounts in Table 6.6-2. The nuclear alternative involves rather large cash outflows during its 6-year construction period followed by a modest fuel savings during subsequent operation of the facility (Table 6.6-5).

A computer program was used to calculate the incremental rate of return for the cash flow in Table 6.6-5; it was found that the incremental return for this case was 2.4% in terms of the then-current dollars. (This rate of return was determined by solving for the interest rate at which the present-worth of cash outflows exactly equaled the present-worth of cash inflows.) A summary of calculated, incremental rates of return for fossil-fired alternatives relative to the CNSG system is given in Table 6.6-6. Column 1 shows the incremental investment rates of return in the nuclear versus fossil comparison for coal prices in the range from $14.50-39.17/ton (1976 dollars), assuming 6%-per-year inflation. These results are depicted in Figure 6.6-1 indicating that for coal at ~$40/ton (1976 dollars), a 15% return might be realized on the incremental capital of the nuclear plant relative to the coal-fired alternative. At about $30/ton a return of about 12% would be realized, which is at the lower end of the 12-15% desired range. For comparison a 10.0% rate of return was calculated on the incremented capital of the nuclear plant relative to the oil-fired alternative.

Column 2 of Table 6.6-6 shows the incremental investment rates of return, assuming zero inflation beyond 1976. This analysis was performed in terms of capital, operating/maintenance, and fuel costs held fixed at 1976 levels. The rates of return on the incremental investment for the nuclear plant ranged from less than zero to 8.2%. The yields in Column 3 are consistent with the higher yields shown in Column 2 for a real-world inflating economy. The magnitude of the interest rate with inflation (Column 1) relative to the interest rate without inflation (Column 2) is consistent with the following formulization that has been proposed by a number of economists[11] and that takes the following form:

1 + interest rate in an inflating economy = (1 + interest rate in a noninflating economy) × (1 + annual inflation rate).

The cost comparisons between the nuclear and fossil-fired steam supplies include a number of assumptions designed to simplify the analysis. The cost of providing fossil-based backup steam to prevent sulfur solidification during reactor refueling and maintenance was not included. Only 2 to 4% of the total annual energy requirement might be used for this purpose. Another simplifying step based the cost comparisons on a common time of plant startup. Under actual conditions the coal-fired plant could start operation about two years earlier since its construction time would be about two years shorter than for the nuclear plant.

The results shown, namely the rates of return on investment, are intended to provide only a crude indication of economic potential to the decision makers. A more detailed evaluation would be required to derive more accurate economic projections.

Table 6.6-1. Calculation of Capital Cost Cash Flow and Investment Tax Credits for Nuclear Alternative (1976 Dollars)

Total direct cost, $	82,078,000
Total indirect cost (includes IDC)	59,442,000
	141,520,000
Working capital (100% salvage value assumed)	739,000
Total initial investment	$142,259,000

Depreciable costs — $87,968,000.

Depreciable costs subject to investment tax credit (ITC) — $65,481,000.

Nondepreciable costs — $54,291,000.

No. years before plant startup	Construction -- $ spread (includes interest during construction)					72 months assumed
5	EOY 1[a] 0.20 FC[b]	=	0.02	(142,259,000)	=	2,845,180
4	EOY 2 +0.10 FC	=	0.10	(142,259,000)	=	14,225,900
3	EOY 3 +0.10 FC	=	0.10	(142,259,000)	=	14,225,900
2	EOY 4 +0.25 FC	=	0.25	(142,259,000)	=	35,564,750
1	EOY 5 +0.33 FC	=	0.33	(142,259,000)	=	46,945,470
0	EOY 6 +0.20 FC	=	0.20	(142,259,000)	=	28,451,800
						$142,259,000

	10% investment tax credit (during construction)				
5	EOY 1	0.002	(65,481,000)	=	130,962
4	EOY 2	0.010	(65,481,000)	=	654,810
3	EOY 3	0.010	(65,481,000)	=	654,810
2	EOY 4	0.025	(65,481,000)	=	1,637,025
1	EOY 5	0.033	(65,481,000)	=	2,160,873
0	EOY 6	0.020	(65,481,000)	=	1,309,620

[a]EOY: end of year.

[b]FC: total initial investment; end of year convention is being used.

Table 6.6-2. Calculation of After-Tax Cash Flow[a] (ATCF) for Nuclear Alternative at 6% Annual Inflation, $\$ \times 10^6$

	End of year	BTCF	Depreciation[b]	Taxable income	Fed and state taxes at 50%		ATCF
	-5	-3.016	0	0	+0.139		-2.877
	-4	-15.984	0	0	+0.736		-15.248
	-3	-16.943	0	0	+0.780		-16.163
	-2	-44.899	0	0	+2.067	ITC	-42.832
	-1	-62.824	0	0	+2.892		-59.932
Startup (1982)	0	-40.359	0	0	+1.858		-38.501
	1	-10.939	-4.000	-14.939	+7.469		-3.470
	2	-11.596	-4.000	-15.596	+7.798		-3.798
	3	-12.291	-4.000	-16.291	+8.146		-4.145
	4	-13.029	-4.000	-17.029	+8.515		-4.514
	5	-13.810	-4.000	-17.810	+8.905		-4.905
	6	-14.639	-4.000	-18.639	+9.320		-5.319
	7	-15.517	-4.000	-19.517	+9.759		-5.758
	8	-16.448	-4.000	-20.448	+10.224		-6.224
	9	-17.436	-4.000	-21.436	+10.718		-6.718
	10	-18.481	-4.000	-22.481	+11.241		-7.240
	11	-19.591	-4.000	-23.591	+11.796		-7.795
	12	-20.766	-4.000	-24.766	+12.383		-8.383
	13	-22.012	-4.000	-26.012	+13.006		-9.006
	14	-23.333	-4.000	-27.333	+13.667		-9.666
	15	-24.733	-4.000	-28.733	+14.367		-10.366
	16	-26.217	-4.000	-30.217	+15.109		-11.108
	17	-27.790	-4.000	-31.790	+15.895		-11.895
	18	-29.456	-4.000	-33.456	+16.728		-12.728
	19	-31.224	-4.000	-35.224	+17.612		-13.612
	20	-33.097	-4.000	-37.097	+18.549		-14.548
	21	-35.084	-4.000	-39.084	+19.542		-15.542
	22	-37.188	-4.000	-41.188	+20.594		-16.594
	22	+2.6667[c]	0	+1.90	-0.95		+1.72

[a]Values are expressed in terms of then-current dollars.

[b]Annual depreciation amount = $\frac{\$87,968,000}{22}$ = \$3,998,545.

[c]Return of working capital.

Table 6.6-3. Calculation of Capital Cost Cash Flow and Investment Tax Credits for the Coal-Fired Alternative (1976 Dollars)

Total direct cost	$36,560,000
Total indirect cost (includes IDC)	12,929,000
	$49,489,000

Working capital
(100% salvage value assumed)

1. 25% first year's est. coal cost @ 14.50/T = $2,831,084
2. 2.5 months work of first year's operating cost (not including depreciation)

$$\frac{2.5}{12}(1,663,000) = \$346,458$$

Total initial investment	$52,666,542
Depreciable costs	$36,390,000
Depreciable costs subject to investment tax credit (ITC)	$32,810,000
Nondepreciable costs	$16,276,552

No. years before plant startup	Construction — $ spread (includes interest during construction)						48 months assumed
3	EOY 1	0.05 FC	=	0.05	(52,666,542)	=	2,633,327
2	EOY 2	0.16 FC	=	0.16	(52,666,542)	=	8,426,647
1	EOY 3	0.36 FC	=	0.36	(52,666,542)	=	18,959,955
0	EOY 4	0.43	=	0.43	(52,666,542)	=	22,646,613
							$52,666,542

10% investment tax credit (during construction)				
EOY 1	0.005	(32,810,000)	=	164,050
EOY 2	0.016	(32,810,000)	=	524,960
EOY 3	0.036	(32,810,000)	=	1,181,160
EOY 4	0.043	(32,810,000)	=	1,410,830

Table 6.6-4. Calculation of After-Tax Cash Flows (ATCF) for the Coal Alternative ($14.50/Ton) at 6% Annual Inflation, $ × 10^6

EOY	BTCF	Depreciation[a]	Taxable income	Fed and state taxes at 50%	ATCF
-3	-2.791	0	0	+0.174	-2.617
-2	-9.469	0	0	+0.590	-8.879
-1	-22.582	0	0	+1.408	-21.174
0	-28.591	0	0	+1.781	-26.810
1	-17.306	-1.654	-18.96	+9.48	-7.826
2	-18.344	-1.654	-20.00	+10.00	-8.344
3	-19.444	-1.654	-21.10	+10.55	-8.894
4	-20.612	-1.654	-22.27	+11.14	-9.472
5	-21.848	-1.654	-23.50	+11.75	-10.098
6	-23.158	-1.654	-24.81	+12.40	-10.758
7	-24.548	-1.654	-26.20	+13.10	-11.448
8	-26.020	-1.654	-27.67	+13.84	-12.180
9	-27.583	-1.654	-29.24	+14.62	-12.963
10	-29.237	-1.654	-30.89	+15.45	-13.787
11	-30.992	-1.654	-32.65	+16.33	-14.662
12	-32.851	-1.654	-34.51	+17.25	-15.601
13	-34.822	-1.654	-36.48	+18.24	-16.582
14	-36.912	-1.654	-38.57	+19.28	-17.632
15	-39.127	-1.654	-40.78	+20.39	-18.737
16	-41.475	-1.654	-43.13	+21.57	-19.905
17	-43.963	-1.654	-45.62	+22.81	-21.153
18	-46.599	-1.654	-48.25	+24.13	-22.469
19	-49.396	-1.654	-51.05	+25.53	-23.866
20	-52.359	-1.654	-54.01	+27.00	-25.359
21	-55.502	-1.654	-57.16	+28.58	-26.922
22	-58.831	-1.654	-60.49	+30.25	-28.581
22	+11.452[b]	0	+8.27	-4.14	+7.312

[a]Annual depreciation amount = $\frac{\$36,390,000}{22}$ = $1,654,090.

[b]Return of working capital.

Table 6.6-5. Incremental, After-Tax Cash Flow[a] for Nuclear Relative to Coal at $14.50/Ton, $\$ \times 10^6$

	End of year	Nuclear − coal		End of year	Nuclear − coal
	-5	-2.877	Unfavorable to nuclear	10	6.547
	-4	-15.248		11	6.867
	-3	-13.546		12	7.218
	-2	-33.953		13	7.576
	-1	-38.758		14	7.966
Startup	0	-11.691		15	8.371
	1	-4.356	Favorable to nuclear	16	8.797
	2	4.546		17	9.258
	3	4.749		18	9.741
	4	4.958		19	10.254
	5	5.193		20	10.811
	6	5.439		21	11.380
	7	5.690		22	11.987
	8	5.956		22	-5.592
	9	6.245		--	Return of working capital

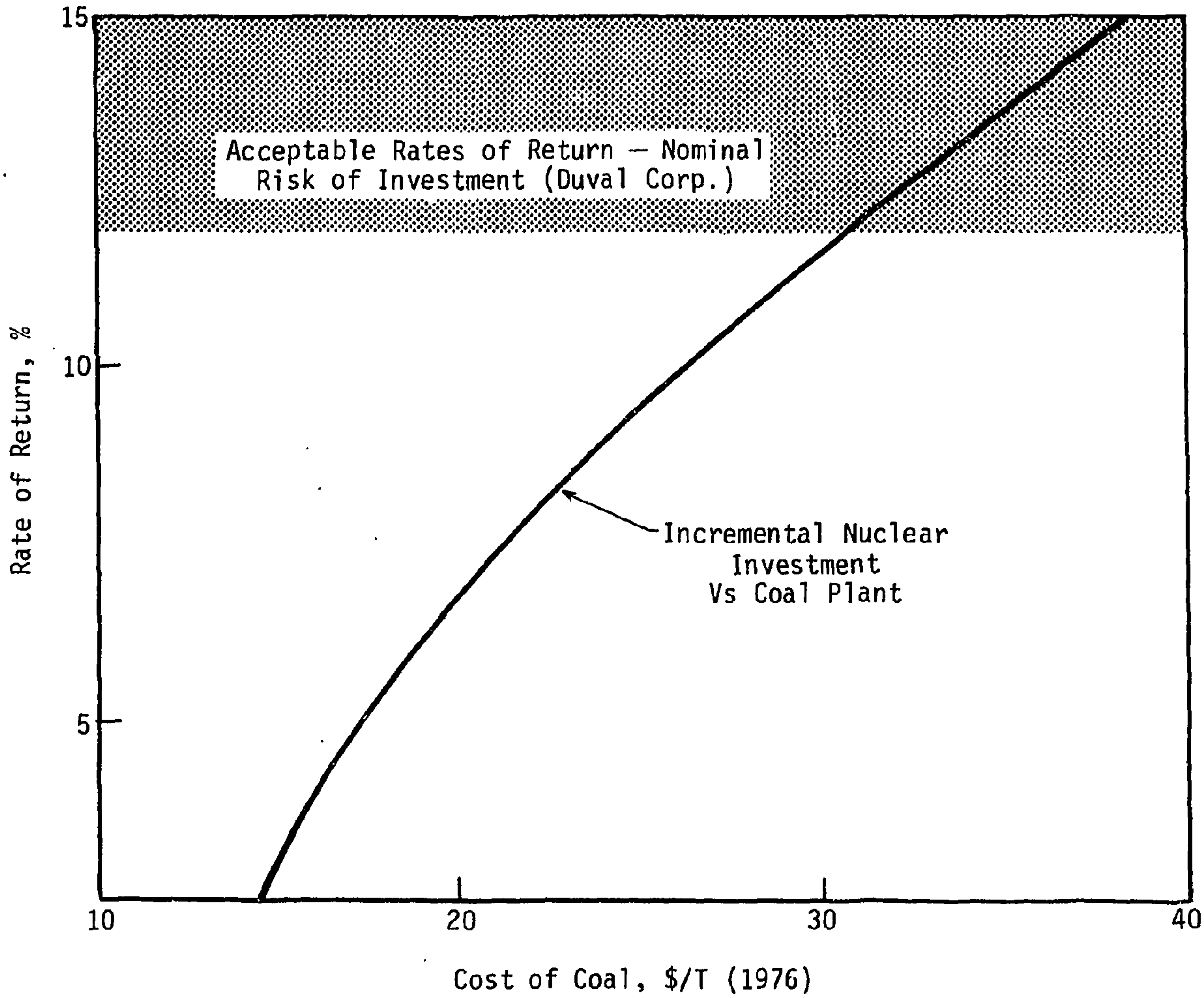
[a]Values expressed in terms of then-current-dollars. Incremental rate of return on extra investment in nuclear = 2.1%.

Table 6.6-6. Summary of Incremental Investment Rates of Return for Nuclear Alternative Relative to Fossil Alternatives

Comparison	6%/year inflation, %	0%/year inflation, %
Nuclear – Coal* (at $14.50/ton)	2.4	Negative
Nuclear – Coal* (at $29.62/ton)	11.6	4.6
Nuclear – Coal* (at $39.17/ton)	15.1	8.3
Nuclear – Oil* (at $0.33/gal.)	10.0	3.4

[a]Fossil fuel prices given in 1976 dollars.

Figure 6.6-1. Rate of Return on Incremental Nuclear Investment (6%/Year Inflation, 1982-2004 Plant Operation)



7. RESULTS AND CONCLUSIONS

The siting/application work undertaken to study the use of the PE-CNSG plant at the Duval sulfur mine incorporated the following tasks:

1. Confirm extension of reactor power level from 314 to 365 MWt.
2. Assess site acceptability.
3. Identify process safety and operating restrictions.
4. Evaluate reactor plant economics for a specific process application at an existing industrial site.

The results and conclusions are summarized below.

7.1. Siting

Section 3 covers site identification and evaluation, and section 4 discusses some aspects of plant relationship to the site. Seismically, the plant would be located in a region of relatively inactive seismic activity (the Permian cap). There is no clear detail as to the structural geology of the site, and a thorough, engineering-geological investigation would be necessary before siting approval. Type and solubility of rock indicate that some subsurface solution subsidence could exist in which case site stabilization work would be required. Mining subsidence exists in the sulfur extraction area, but the selected plant location is well removed from possible mining subsidence effects (Figure 4.3-6). Although some indications of faults have been observed, there is no known indication of active or capable faults. No population density or accident dose-to-man site limitations were identified.

7.2. Interface Design

Plant cooling water and water for mining process consumption is supplied by pipeline from deep wells located about 36 miles from the site and is stored in a reservoir at the site. Since the raw water has high, total dissolved solids and hardness characteristics, it must be treated prior to use in the tertiary system. A reverse osmosis (RO) treatment system was incorporated to accomplish this. In addition, a standby heat sink must be available for nuclear regulatory considerations. To ensure seismic acceptability, a small cooling tower was incorporated into a plant design.

Onsite, full-capacity, electric power requirements are estimated at 11.5 MW. Power for onsite process and facility needs will be produced from the reactor, but capacity for the reactor and nuclear balance of plant will be supplied from an offsite utility grid for necessary reliability and redundancy. Onsite power will be used for backup. New transmission lines and switchyard are provided in the design to supply offsite power needs.

The nuclear and process systems are interfaced through a tertiary loop designed with reboilers, as well as existing heat exchanger units, to provide the required 325F water. Separation of the nuclear and process systems and operational flexibility has been provided to maintain appropriate separation under differing failure modes. Barriers to the release of radioactivity include fuel cladding, each of twelve steam generators (any of which can be isolated to eliminate tube leakage sources), reboilers, and finally feedwater heaters to supply heated process water. The design has provided sufficient reboiler surface to permit operation to continue with a reboiler unit out for repair. Since radioactivity leakage can be monitored and isolated through either the nuclear steam system steam generators or the tertiary loop reboilers, sufficient operational flexibility exists to ensure clean, nonradioactive water to the process.

Tertiary loop design provides for excellent water chemistry control to minimize tube corrosion in the steam generator modules and the reheaters, isolating both from poor-quality water and the possibility of inleakage of such water to the systems protected by these barriers. The tertiary loop design represents a conservative design known to be adequate and satisfactory for nuclear plant operation but not necessarily optimized economically. In particular, the possibility of replacing reboilers with an automated control and isolation system might influence positively overall nuclear economics.

7.3. Economics

Nuclear plant economics must compete with available fossil fuels (in this case coal) or possibly oil as a replacement energy source for natural gas. Nuclear fuel costs are much lower than comparable fossil fuel costs, but the capital investment required in plant is much greater. In the interval between the start of the initial PE-CNSG economics study and this application study, nuclear fuel costs rose due to high uranium oxide ore cost and higher separative work unit (enrichment) costs. The higher values are reflected herein. Reestimates of nuclear plant and balance-of-plant capital costs were also raised somewhat.

The cost comparison between the nuclear and coal-fired options has projected an 11 to 15% rate of return on equity for the incremental capital investment required by the nuclear plant for coal prices ranging from approximately $30-$40/ton ($1976). The return on equity would be greater if some of the investments were financed with bonds rather than with equity only; however, Duval's budgeting criteria customarily consider only equity financing. Coal prices are difficult to forecast due to uncertainties in demand, supply, and transportation costs; however, under present conditions it appears unlikely that coal prices would rise sufficiently to produce a rate of return in excess of 15% for the nuclear plant incremental capital investment. From the viewpoint of the Duval Corporation, a 12-15% rate of return might be appropriate for an investment involving average risks and uncertainties. Duval feels that a higher yield would be required to make a nuclear project attractive (a high-risk investment in Duval's eyes). The uncertainties relate primarily to licensing and project schedule issues rather than to technical questions. Numerous nuclear generating projects have experienced unanticipated delays, and it is to be expected that this experience would be a source of major concern to potential industrial users of nuclear energy.

Oil is another alternative for fuel. This option requires minimal capital expenditures in comparison to coal or nuclear. If industrial fuel oil remains available at about $14/barrel ($1976), the oil-fired energy supply is attractive relative to nuclear power.

Therefore, it is concluded that under present circumstances, further study of the CNSG industrial reactor application at the Culberson County site does not appear to be warranted. However, it may become worthwhile to reexamine the prospects for the nuclear option if some of the underlying economic circumstances should change. For example, if new tax incentives become available to encourage capital investment in new industrial energy sources, if the cost of burning coal was to rise (for example due to more strict environmental regulations), or if the prices of fuels were to rise at a rate significantly greater than the prevailing inflation rate, the economics of the nuclear option could become more attractive.

APPENDIX A

PE-CNSG Component and System Descriptions

1. Component Design

Reactor Vessel and Closure Head — The reactor vessel (Figure 4.4-1) contains the pressurized, moderator-coolant water and supports and houses the reactor core support assembly, control rod guide and core holddown assembly, FAs, upper plenum assembly, vertical primary coolant pumps, pump diffusers, steam generator and steam generator mixer box assembly, control rod assemblies, and reactor control and safety-related instrumentation.

The reactor vessel consists of a cylindrical shell, a spherically dished bottom head, and a ring flange to which the spherically dished reactor closure head with four vertical pump nozzles is bolted. The vessel shell material is protected from fast neutron flux and gamma heating effects by a water annulus and a stainless steel shell located between the reactor core and the vessel wall.

The reactor coolant is contained within the reactor vessel, the pressurizer, and the MU&P. The coolant passes up through the core, turns at the top of the upper plenum, flows through the primary pump into the steam generator tubes, moves down the annulus between the core barrel assembly and vessel inside wall, and then turns up through the core support assembly.

The vessel has 12 coaxial steam/feedwater nozzles located on the vessel side wall just below the vessel flange. A pipe passes through each nozzle in which feedwater passes to the bottom of the steam generator. As feedwater passes up over the outside surface of the steam generator tubes, slightly superheated steam is produced, which then exits the reactor vessel through the annulus provided by the coaxial steam/feedwater nozzle. Smaller nozzles located between the vessel flange and the steam/feedwater nozzles serve as inlets for MU&P, DH cooling, and emergency cooling water injection. Two small nozzles are provided between the vessel flange and the steam/feedwater nozzles to serve as the surge and spray connections to the pressurizer. In addition to the four primary coolant pumps, the closure head is penetrated by 17 flanged nozzles to which are attached the CRDMs. There are also six RTD nozzles for instrumentation for core outlet temperature measurement. The bottom head of the vessel is provided with instrumentation nozzles.

The reactor vessel is supported vertically by a skirt welded to a forged dutchman between the lower shell course and head.

Lugs are welded (Inconel weld buildup) to the vessel inside wall near the bottom of the steam generator to provide support for the steam generator outlet mixer box assembly. Keys are welded (Inconel weld buildup) to the vessel inside wall at the same elevation as the lugs to orient the mixer box assembly and to provide lateral support for the core support structures. The core support assembly is supported vertically from weld pads on the bottom head of the reactor vessel.

Two concentric metallic seals provide the pressure integrity seal between the reactor vessel and closure head flanges. The CRDM flanges are sealed by concentric metallic seals. A high-pressure leakoff and drain tap is provided at the annulus between all concentric seals.

Surveillance specimens made from the reactor vessel shell material are located between the reactor vessel wall and the core barrel. These specimens will be examined periodically to evaluate NDTT. The surveillance program will be used for the first CNSG unit to confirm that there are no NDTT problems.

Steam Generator — The steam generator (Figure A-1) consists of 12 cylindrical modules (operating in parallel) installed in the annular space between the control rod guide and core holddown structures and the inner surface of the reactor vessel. Each module comprises a cylindrical shell with tubesheets welded to each end. The heat transfer surface is the tubes welded between the tubesheets. The steam generator uses the once-through concept with shell-side boiling producing superheated steam. Steam generator performance characteristics are shown in Table 4.4-2.

Reactor coolant is circulated by pumps down through the tubes of each module and up through the reactor core. Feedwater enters and steam leaves each module through a coaxial pipe arrangement near the top of each module.

Feedwater enters the steam generator modules through the inner coaxial pipe and sprays into the top end of a triangular downcomer, which is open at each end and extends to near the bottom tubesheet of each module. In the downcomer steam is mixed with the subcooled feedwater to raise its temperature to saturation before it enters the tube bundle. Saturated feedwater flows up through the bundle, boils to dryness, and is superheated before leaving through the outer pipe of the coaxial arrangement. A portion of the steam, the amount depending on load and feedwater temperature, is recirculated through the downcomer.

Spacer grids and baffles inside each module support the tubes. They are designed to provide optimum heat transfer and pressure drop performance and to prevent tube vibration that could damage the tubes.

Each module is supported by the steam nozzle pipe that is welded between it and the wall of the reactor vessel. Lateral support is provided at the bottom end of each module. This support and an auxiliary support at the top end also serve to restrain the modules in the event of steam nozzle failure.

During normal operation the shell side of each module is filled with feedwater and steam. The inside of the tubes and the external surface of each module are in contact with primary coolant. The steam generator is located above the core to reduce the radiation from the reactor core, thereby reducing N-16 generated in the secondary coolant. This also minimizes the effects of irradiation on material properties.

The steam generator modules consist of shells, tubesheets, tubes, spacer grids and baffles, vent and drain lines, and a downcomer. The feedwater spray nozzle, although functionally a part of the steam generator, is considered a part of the feedwater inlet/steam outlet adapter. This adapter piece is bolted to the outside surface of the reactor vessel and forms the connections and flow paths between the steam generator and the reactor vessel and the feedwater, steam, vent, and drain lines.

Pressurizer and Pressurizer Surge and Spray Lines – The pressurizer (Figure A-2) is a vertical, cylindrical pressure vessel connected to the reactor vessel by a surge line from its bottom head and a spray line from its top head. The functions of the pressurizer are to establish and maintain RCS pressure and to provide the surge volume and reserve water for accommodating volume fluctuations in the system during operation, startup, and shutdown.

The pressurizer contains steam and water maintained at saturated conditions by electric immersion heaters located in the side section of the vessel and the water spray nozzle located in the vessel upper section. Pressurizer outsurges caused by contraction of the reactor coolant decrease the pressurizer pressure; some of the saturated water in the pressurizer flashes to steam, thus limiting the pressure decrease. The electric heaters are actuated to restore normal operating pressure. Pressurizer insurges caused by expansion of the reactor coolant increase the pressurizer pressure. The pressure increase is limited by water spray, which condenses some of the steam in the pressurizer. Spray flow and the electric heaters are controlled automatically by a pressure controller during normal operation.

The surge line connects the pressurizer and the reactor vessel. It contains water at the system pressure during normal operating conditions. By connecting the pressurizer and the reactor vessel, the line conveys the pressurizer pressure to the coolant system to account for fluctuations in the reactor vessel primary coolant operating pressure and transmits the pressurizer insurges and outsurges when the reactor coolant expands and contracts.

The spray line, connected between the pressurizer and the reactor vessel, transports reactor coolant to the pressurizer steam space as necessary for RCS pressure control.

2. Core Configuration

The core comprises 57 FAs containing slightly enriched UO_2. It is arranged on a square lattice pitch to approximate the shape of a cylinder (Figure A-3).

Fuel assemblies consist of fuel rods positioned in a square pitch array by a structural cage made up of end fittings, control rod guide tubes, incore instrumentation tubes, and spacer grids. The fuel rods are held in place laterally by spacer grids. The control rod guide tubes provide the structural tie between the upper and lower end fittings, and the spacer sleeves on the instrumentation tube provide the vertical location of the spacer grids.

Cluster-type CRAs in conjunction with boric acid are used for reactivity control. Each CRA has stainless steel-clad neutron absorber rods coupled by a stainless steel spider to a common drive shaft. The cladding carries operating loads caused by control rod motion.

The CRA is inserted through the upper end fitting of the FA. The CR guide tubes included in the fuel assembly and guides in the plenum assembly provide full-length guidance of the CR.

Fuel assemblies at core positions not requiring CRs will contain an LBP rod assembly or an ORA according to requirements of the fuel management scheme. Neutron sources are inserted at appropriate locations in vacant CR guide tubes.

Fuel Assembly — The fuel assembly (Figure A-4) proposed for the PE-CNSG design is basically a shortened central station Mark B FA. Table A-1 shows a comparison between the Mark B FA and the PE-CNSG FA. Table A-2 lists pertinent FA components, materials, and dimensions for the CNSG-FA.

The FA consists of pressurized FRs and LBPRs, 16 control rod guide tubes, an instrumentation tube, six spacer grids, and two end fittings. The guide tubes, spacer grids, and end fittings form a structural cage to arrange the rods and tubes in a 15 by 15 square pitch array. The center position in the assembly is reserved for the instrumentation tube.

The FRs are held in place laterally by spacer grids. The CR guide tubes provide the structural tie between the upper and lower end fittings, and the spacer sleeves on the instrumentation tubes provide the vertical location of the spacer grids. Any FA can accommodate either a CRA, LBPRA, or ORA.

The guide tubes are rigidly attached to the upper and lower end fittings. To accommodate axial length changes in the FR caused by irradiation growth and potential pellet-cladding interaction, adequate clearance has been allowed between the FRs and the FA end fittings. The use of similar material in the guide tubes and FRs also results in minimum differential thermal expansion.

The FA is indexed to the upper grid plate and the lower grid using dowel indexing. The upper end fitting receives the two dowels in the upper core grid plate, and the lower end fitting receives the two dowels in the lower core grid plate.

Upper End Fitting – The upper end fitting positions the upper end of the FA in the upper core grid plate structure and provides a means for coupling the handling equipment. Penetrations in the upper end fitting grid are provided for the guide tubes and to permit coolant flow directly out of the upper end of the FA.

Lower End Fitting – The lower end fitting positions the FA in the lower core grid plate. Its grillage provides a support surface for the bottom end of the FRs. Penetrations in the lower end fitting are provided for attaching the CR guide tubes, for access to the instrumentation tube, and to permit coolant flow directly into the bottom of the FA.

Spacer Grids – Spacer grids are constructed from strips that are slotted and fitted together in an "egg-crate" fashion to form the 15 by 15 lattice. TIG welding is used to join the punched strips together to form a high-strength grid capable of withstanding normal and accident loads. The square cells formed by the interlaced strips provide support for the fuel rods in two perpendicular directions through contact points on each wall on each cell. The contacts are in the form of protruding dimples, which are integrally punched from the strips on the walls of each square opening. On each of the two end spacer grids, the peripheral strip is extended and mechanically attached to the respective end fitting.

A unique feature of the B&W spacer grid is that the cell fuel rod contact points are held open with removable keys during fuel rod loading into the grids. Thus, the rods can be freely inserted into the assembly without detrimental scratching of the cladding surface. An important characteristic of the spacer grid is that it is not mechanically attached to the CRA guide tube as is the case with other designs. Consequently, the grid is free to axially adjust as the FRs and guide tubes undergo relatively different rates of thermal expansion and irradiation growth. This reduces scratching of the FR surface. The end grids are rigidly attached to the end fittings through a peripheral skirt, providing a structurally sound assembly.

Control Rod Guide Tube – The Zircaloy guide tubes provide continuous guidance for the CRAs when inserted in the FA and provide the structural continuity for the FA. These tubes also serve as a channel for ORAs where LBPRAs are not inserted. The guide tubes are mechanically fastened to both the upper and lower end fittings. Transverse location of the guide tubes is provided by the spacer grids.

Instrumentation Tube — The Zircaloy instrumentation tube serves as a channel to guide, position, and contain the incore instrumentation with the FA. The instrumentation probe is guided up through the lower end fitting to the desired core elevation. It is retained axially at the lower end fitting. Spacer sleeves on the instrumentation tube provide for vertical location of the spacer grids.

Spacer Sleeves — The spacer tubes fit around the instrument tube between spacer grids and restrict axial movement of the spacer grids.

Fuel Rod — The fuel rod (Figure A-5) consists of fuel pellets, the cladding, fuel support components, and end caps. Table A-2 lists pertinent fuel rod components, materials, and dimensions. A unique feature of the FR is that there are flexible spring spacers located both above and below the pellet stack. This becomes particularly important in the event that a pellet becomes lodged somewhere along the stack height. Should this occur, the lower spring will accommodate fuel expansion in the downward direction, thus significantly reducing cladding strain. Under normal operating conditions fuel expansion will be in the upward direction because of the more flexible nature of the upper spring.

The spring spacers further serve to separate the fuel stack from the end caps and maintain the pellet stack in place during shipping and handling.

Zircaloy spacers are located between the fuel pellets and the spring spacers to thermally insulate and separate fuel pellets from the spring spacers and end caps.

Fission gas generated in the fuel is released into voids, the radial gap between the pellets and the cladding, and into void spaces at the top and bottom ends of the fuel rods.

1. Fuel Pellet

The pellets are manufactured by cold-pressing, enriched UO_2 powder into cylinders with end dishes and chamfers and then sintering to obtain the desired density and microstructure. After sintering, the pellets are centerless-ground to the required diametral dimensions.

Principal design parameters are as follows:

Pellet diameter, mm (in.)	9.347 (0.368)
Density, % TD	94
Active fuel stack height, mm (in.)	1829 (72)

The design parameters are selected to minimize cladding strains that could result from the following causes:

a. Axial strain of the cladding due to thermal expansion of the pellet column.

b. Radial pellet-cladding interactions resulting from "ridging" or "hourglassing" due to thermal expansion of individual pellets.

c. The effect of irradiation-induced swelling superimposed on the thermal effects of a and b above, particularly toward the end of life of the fuel.

The dish, chamfer, and L/D dimensions are optimized with respect to thermal, stress, and manufacturing requirements. Both the dish and L/D are optimized between the need to reduce axial expansion (larger dish, larger L/D) and the need to minimize ridging (limited dish, smaller L/D). The chamfer and L/D are set to minimize ridging and to allow free axial motion of pellet column during both rod loading and reactor operation.

The density and microstructure are designed to minimize the effect of densification while limiting the subsequent effects of both radial and axial fuel swelling. The irradiation swelling data used were generated by Bettis Atomic Power Laboratory.

2. Cladding

The cladding is Zircaloy-4 tubing. The principal design features are as follows:

Outside diameter, mm (in.)	10.922 (0.43)
Thickness, mm (in.)	0.673 (0.026)
Hydride orientation	Predominantly circumferential

The cladding OD is a primary parameter determined by the overall thermal-hydraulic design of the reactor. The cladding is subject to a variety of sustained and cyclic stresses resulting from the following:

a. External hydrostatic pressure.

b. Increasing internal pressure.

c. Differential thermal expansion and irradiation growth between the cladding and the pellet column.

Stress analyses conservatively incorporate the effects of thermal gradients across the cladding. The FR is designed so that the stress intensity does not exceed two-thirds of the minimum specified yield strength. The total cladding strain is limited to 1% plastic strain and 0.4% elastic strain.

Cumulative fatigue damage is determined for conservative estimates of power cycling of the material.

The cladding thickness is optimized between the need for good heat transfer characteristics and the requirements for adequate strength. The FR is pressurized internally to retard the cladding from creeping down into the fuel and to help prevent creep collapse. To set this pressurization level, several analyses are performed to determine the internal pressure at the end of the FR lifetime and the collapse characteristics of the internally pressurized FR. The retardation of cladding creep helps to increase the cladding fatigue life.

The cladding hydride orientation, being predominantly circumferential, provides increased strength and ductility in the circumferential and axial directions. Thus, cladding resistance to pellet-cladding interaction strains is substantially enhanced.

3. Reactivity Control

Reactivity is controlled through the use of CRAs, soluble boron, and LBPRs. In FAs that contain neither a CRA nor a LBPRA, ORAs are used to restrict bypass flow.

Control Rod Assembly

The CRA has 16 CRs, a stainless steel spider, and a female coupling. The 16 CRs are attached to the spider by a nut threaded to the upper shank of each rod. After assembly, all nuts are lock-welded. The CRDM is coupled to the CRA by a bayonet-type connection. Full-length guidance is provided by the CR guide tube of the upper plenum assembly and by the FA guide tubes. The CRAs and guide tubes are designed with adequate flexibility and clearances to permit freedom of motion within the FA guide tubes throughout the stroke.

Each individual CR has a section of neutron absorber material composed of B_4C pellets. These pellets are clad with cold-worked type 304 stainless steel. End pieces are welded to the tubing to form a water- and pressure-tight container for the absorber material. The stainless steel tubing provides the structural strength of the CRs and prevents corrosion of the absorber material. A spring spacer similar to that in the fuel rod is used to prevent absorber motion within the cladding during the shipping and handling and to permit differential expansion in service.

Principal data pertaining to the CRAs are shown in Table A-3.

Lumped Burnable Poison Rod Assembly

Each LBPRA has burnable poison rods and a stainless steel spider. The LBPRs are attached to the spider. The assembly is inserted into the FA guide tubes through the upper end fitting.

Each LBPR has a section of sintered Al_2O_3-B_4C pellets which serve as burnable poison. The poison is clad with cold-worked Zircaloy-4 tubing with Zircaloy-4 upper and lower end pieces. The end pieces are welded to the tubing to form a water- and pressure-tight container for the absorber material. The Zircaloy-4 tubing provides the structural strength of the BPRs. Pertinent data on the BPRAs are shown in Table A-4. Except for the substitution of Al_2O_3-B_4C pellets for B_4C pellets and Zircaloy cladding for stainless steel cladding, the BPRA is geometrically similar to the CRA.

Orifice Rod Assembly

The ORA limits bypass flow through empty guide tubes. Pertinent data on the ORAs are shown in Table A-5. Each ORA has 16 orifice rods and a stainless steel spider and is similar to the CRA except that the stub stainless steel rods replace the CRs.

Neutron Sources

Primary neutron sources encapsulated in stainless steel are provided for initial startup. The present design is based on the use of two 2-Ci californium-252 source capsules. However, alternate types of source materials having more desirable nuclear characteristics may be used if they are available. The primary startup sources are inserted into vacant CR guide tubes.

4. Fuel Cycle

The PE-CNSG fuel cycle is based on four-batch refueling. On the average, one fourth of the core is replaced every refueling, which is once every 12 months. Therefore, except for those in the first core, every FA is irradiated for four cycles. The initial fuel loadings are given in Table A-6.

There are four enrichment zones in the initial core. Those FAs in the lowest enrichment zone are irradiated for only one cycle, while those in the next-to-the-lowest zone are irradiated for two cycles, etc. At each refueling the FAs are shuffled to different positions in the core so that each assembly will be subjected to about the same amount of irradiation in order to achieve a flatter power distribution. The LBPRAs are replaced every refueling and are used to improve fuel performance by helping shape the core power distributions and ensure a negative moderator coefficient at BOL.

The subsequent cycles require only one feed enrichment. This fresh fuel is put in the outer portion of the core where it helps to increase the power produced in those assemblies. The reload enrichments are given in Table A-6, assuming that each subsequent cycle will be the same. Sufficient flexibility exists to accommodate an unscheduled change in the capacity factor. A cycle that is longer than scheduled can be achieved by coastdown, which is a process where the core is operated past EOL but is forced to operate at an ever-decreasing power level. The extra burnup extracted by coastdown will shorten the subsequent cycle by approximately two thirds of the full-power-day extension.

5. Reactor Internals

Reactor vessel internals include the core support assembly, the core barrel assembly, the upper plenum assembly, and the control rod guides and core hold-down assembly. The reactor internals arrangement in the vessel is shown in Figure 4.4-1.

The internals support the core both vertically and horizontally and direct and distribute coolant through the core and steam generator. They also maintain FA alignment, limit FA movement, and maintain control rod alignment. A rod retention cylinder is included in the internals to limit the rate of control rod withdrawal in the unlikely event of a complete CRDM housing failure. The internals are designed to accomplish their functions without applying excessive stress concentrations to the vessel wall and without producing vibrations that could cause system damage. All internal components can be removed for inspection.

The PE-CNSG utilizes a bottom-supported core. The weight of the core and core barrel and the FA spring reaction force are transmitted through the core support cylinder to the lower head of the reactor vessel. The control rod guide and core holddown assembly are used to counteract the upward force of the FA springs. The difference in thermal expansion between the reactor internals, the steam generator, and the reactor vessel is provided for by an expansion joint between the upper grid plate and the core barrel. The upper grid plate is free to move axially against the FA springs, and the core barrel is free to move axially relative to the steam generator support plate. An additional expansion joint is provided at the pump inlet to allow the internals holddown plate to move relative to the pumps during transient operation.

The PE-CNSG internals support the static loads of the core and the added loads imposed by the effect of seismic excitation. These loads are transmitted to the pressure vessel at three locations: the lower head, the steam generator support, and the upper vessel flange. The internals are also designed to withstand the dynamic loads resulting from the coolant flow and the vibration loads imposed by such external sources as the auxiliary equipment.

Core Support Assembly

The core support assembly consists of a lower grid plate, support ribs, a flow distributor plate, and a support cylinder. The weight of the core is transmitted through the grid plate to a support cylinder and onto the lower head. The design provides for thermal expansion between the cylinder and lower head. The flow distributor plate has holes of various sizes to ensure that flow is properly distributed to each FA.

Core Barrel Assembly

The core barrel assembly surrounds the core, directs the coolant through it, and takes side loads due to FA deflection. These side loads are transmitted to the vessel wall through the previously discussed horizontal supports.

Control Rod Guide and Core Holddown Assembly

This assembly comprises the CR extension guide assemblies and the support structure necessary to transfer core loads to the upper head of the reactor vessel. Also included in the CR guide and core holddown assembly are the upper grid plate, the CR extension shafts, the upper plenum cover, the CR retention cylinders, and the trip springs.

The upper internals are loaded in compression by the force of the FA holddown springs. This force is counteracted by the upper internals and the differential created by coolant flow. The vertical loads are transmitted between the upper grid plate and the upper plenum cover by the cylinders. Horizontal loads are transmitted to the vessel wall at the lower steam generator support and at the upper head flange.

Flow Distributor

A flow distributor plenum located at the top of the steam generators distributes the output of the four coolant pumps to the twelve steam generator modules. A flow mixer located below the steam generators prevents hot streaks of coolant from reaching the core in the event that one or more steam generator modules are shut down.

6. Reactor Coolant Pumps

Four primary coolant pumps are mounted on the reactor vessel head. The RC pumps circulate the primary coolant between the core and the steam generator contained in the reactor vessel. The RC pumps are vertical, single-stage, single-suction, constant-speed mixed-flow pumps. They are glandless devices with no mechanical seal between the pump and motor. A view of the pump/motor is shown in Figure A-6. The pump consists of the following components: motor vessel, motor, shaft, impeller, thermal barrier, and diffuser. The pump is cooled by an external heat exchanger.

The pump motor is a three-phase, four-pole, squirrel-cage induction motor nominally rated at 1800 rpm when supplied with 2300-V, 60-Hz ac. The motor, which is an integral part of the pump, incorporates a number of special features. The primary components of the motor are a stator assembly, a terminal assembly, a rotor assembly, one radial bearing, and one radial bearing-thrust bearing combination.

The main impeller is a conventional mixed-flow device and has seven equally spaced vanes. It is designed for counterclockwise rotation as viewed from the stator cap end.

The thermal barrier located between the pump/motor and hydraulic parts reduces heat flow from the diffuser into the pump/motor cavity. The thermal barrier separates the primary water system from the lower-temperature internal water system.

The pump diffuser contains the main impeller and vanes designed to straighten and slow the flow. B&W is developing a diffuser system design as part of the pump test program for the maritime CNSG. This type of pump is used in small reactors in England and is being built for future use in Swedish reactors.

7. Control Rod Drive Mechanism

The CRDM comprises a motor tube assembly, a lower mechanism assembly, and a leadscrew and extension assembly (Figure A-7). The motor tube houses a rotor assembly and is closed on the upper end with a closure and vent assembly. It is part of the primary pressure boundary. An external motor stator surrounds the motor tube. Position indicators are arranged outside the upper extension of the tube. The CRD output element is a non-rotating, translating leadscrew coupled to the control rod. The screw is driven by separating, antifriction roller nut assemblies attached to segment arms and magnetically rotated by the motor stator located outside the pressure boundary. Stator current causes the separating roller nut assembly halves to close and engage the leadscrew. Mechanical springs disengage the roller nut halves from the screw in the absence of current. For rapid insertion the nut halves separate and release the leadscrew and control rod. A hydraulic snubber at the end of the housing decelerates the CRA to a slow speed near the full-in position.

Design features of the drive include the following:

1. A fast trip response whenever electrical power to the stator is interrupted.
2. Exclusive use of static-type seals. No rotary or dynamic seals are used.
3. A leadscrew coupling design that provides easy attachment of the leadscrew to the extension rod from a position external to the reactor.
4. Electrical components mounted outside the primary pressure boundary for protection and ease of maintenance.

During normal operation the drive is used to raise, lower, or maintain the position of the CR within the reactor in response to electrical signals from the CRDM control system. The control system provides sequentially programmed d-c input to six stator windings. Three of the six windings are energized at a time to produce a rotating magnetic field. This magnetic field is coupled to a four-pole rotor, forming a four-pole, reluctance-type drive motor.

The rotor assembly is split so that when the stator is energized, the rotor assembly segment arms pivot to mechanically engage the roller nuts with the leadscrew threads. As the stator coils are sequenced, the rotor rotates in steps to new positions. There are six mechanical steps for each electrical revolution. Two electrical revolutions result in one mechanical revolution.

The drive is designed to trip when called for by the RPS or on any loss of electrical power. The trip signal from the protection system interrupts power to all CRDM stators. During such a power loss the rotor assembly segment arms pivot releasing the mechanical contact between the roller nut and the leadscrew. The CRs are then inserted into the core to the full-in position by the force of gravity. A hydraulic snubber is located in the bottom to decelerate the CRs before the mechanical stop is reached. A trip may be initiated manually at the control console or the hot and cold shutdown panel, automatically by the reactor protection system or inherently through a loss of power to the stator. The latter feature makes the system fail-safe in the event of total power loss.

Motor Tube

The motor tube assembly is the main structural support for all components of the mechanism except the lower mechanism assembly. The motor tube, which is also the primary pressure boundary, will be designed and fabricated in accordance with the requirements of the ASME Code, Section III.

Closure and Vent Assembly

The upper end of the motor tube is closed by an insert closure assembly containing a vapor bleed port and vent valve. The vent valve and the insert closure have double seals. The insert closure is retained by a closure nut threaded to the inside of the motor tube. The sealing load for the closure is applied by jackscrews threaded through the closure nut.

Lower Mechanism Assembly

The lower mechanism assembly is assembled as a unit, lowered into the reactor vessel head, and bolted to the reactor head nozzle. The major components of this assembly are the assembly housing and a thermal barrier and leadscrew guide bushing to reduce the temperature in the roller nut area and to support the lower end of the leadscrew.

Rotor Assembly

The rotor assembly operates completely immersed in primary water within the pressure boundary and is designed to operate with no other lubricant. The principal parts of the rotor assembly are the rotor tube, two segment arms, four pivot pins, four roller nut assemblies, thrust bearings, a radial bearing, a synchronizing bearing, four compression springs, a rotational stop, a thrust bearing retainer, and two rod retention pawls.

The central structural support of the rotor assembly is the rotor tube, which is hollow to allow axial passage of the leadscrew. It also provides the supports for the pivots of the rotor segment arms. A thrust bearing is mounted at the lower end, and the radial and synchronizing bearings are mounted on the upper end of this tube. The thrust bearing and radial bearing support the rotor nut assembly when it is installed in the motor tube.

The rotor segment arms are attached by pivot pins to the rotor tube. The arms rotate with and pivot on the rotor tube. The upper portions of the two arms are designed to form a four-pole collapsible rotor. The lower portions of the arms each contain two roller nut assemblies which, when the stator is energized, engage with the leadscrew threads.

Energizing the stator causes the upper portions of the segment arms to be pulled radially outward. The lower portions of the arms move radially inward, and the four roller nuts engage the leadscrew.

Four compression springs below the pivots of the arms act to keep the roller nuts disengaged from the leadscrew. To engage the leadscrew, a greater force is applied to the arms above the pivot point by the stator magnetic field. The top of each arm has a stop to maintain clearance between the arms and the motor tube wall.

Stator Assembly

The stator assembly is mounted on the motor tube with three bolts and cup-washers and, together with the rotor nut assembly, forms a reluctance-type step or synchropulse motor. If the stator is energized with a continuous direct current (HOLD command), the electromagnetic field attracts the upper ends of the rotor assembly segment arms outward and causes the roller nut assemblies in the lower ends of the arms to engage and hold the leadscrew. If an IN or OUT command current is applied to the stator, the rotor assembly will remain engaged with the leadscrew and rotate axially about the leadscrews causing it to be raised or lowered. The direction of the rotation depends on the sequence in which the stator windings are energized.

The stator is a six-winding, opposing wound star, which permits reversal of the magnetic field without reversing polarity of any windings. By alternately energizing the opposing wound winding in proper sequence, a rotating magnetic field is developed within the stator. The stator leads are terminated in one electrical connector receptacle mounted on a standpipe above the stator assembly.

Silicone O-rings protect the stator windings and laminations from the adverse effects of the environment adjacent to the stator. Two thermocouples are mounted within the stator adjacent to the end coils to monitor the stator winding temperature. The thermocouple leads terminate in receptacles adjacent to the stator power receptacle.

The stator is provided with a stainless steel water cooling jacket on its outside diameter. Heat conducted to the stator core from the primary water and from the energized stator windings is carried away by water circulating through the helical passage of the cooling water jacket.

Position Indicator

The position indicator assembly mounted vertically on the motor tube extension is used to determine the position of the control rod. It consists of a series of equally spaced resistors and reed switches. These reed switches close as a magnet mounted on top of the leadscrew extension travels vertically into their immediate vicinity. When the magnet passes the reed switch, the switch returns to its normally open position. This provides a set potential for the particular reed switch, which can then be converted to indicate control rod position.

8. Insulation

Insulation for use in the PE-CNSG system is designed to limit heat loss to a maximum of 252.2 W/m^2 (80 Btu/h-ft^2). The all-metal insulation is the thermal reflective type.

The insulation forms a system comprising prefabricated units engineered as integrated assemblies to fit the surface to be insulated with due allowance for expansion and contraction. Each insulation unit consists basically of a casing, end closures, supports, inner reflective sheets, and a means for securing each reflective sheet. All parts are arranged to form isolated air chambers between the sheets and end closures. The parts are easily removable for inspection of welds. They are supported in such a way that no load is transmitted to the component.

Metal reflective insulation is placed off the reactor vessel wall below the top of the pressure suppression pool (below the steam/feedwater nozzles) and attached to the inner wall of the pressure-suppression pool. Insulation is placed under the reactor vessel lower head and is attached to the reactor vessel above the pressure-suppression pool, including the closure head dome. Insulation encloses and is attached to the pressurizer and surge and spray line piping.

9. Service Support Structure

The service support structure is positioned on top of the reactor vessel head and performs the following functions:

1. Restrains the CRDM and accessories against deflection resulting from seismic excitation and/or induced vibrations during normal operation or vibrations and/or forces due to emergency operations.
2. Provides a work platform for convenient access to install, maintain, and remove the CRDs.
3. Supports the power and service cables and the cooling water manifolds that serve the CRDMs.
4. Supports the rigging used to install and handle the reactor vessel closure studs, guide studs, and stud tensioners.
5. Provides air flow baffles to direct cooling air past the CRDMs.
6. Provides clearance for removal of the RC pumps.
7. Restricts axial motion of the CRDMs during a postulated rod ejection accident.

The service support structure is designed with vertical support beams attached to the reactor vessel head. A steel plate is welded to these beams to provide air flow baffles for cooling the CRDMs. Horizontal beams are positioned at the top of the service structure to restrict axial movement of the CRDMs. A working platform is provided at the top level of the service structure to allow CRDM maintenance. The carbon steel used throughout the service structure is designed to meet the requirements of ASME Section III, Subsection NF.

Water manifolds are provided to supply stator cooling water, and provisions are made to support the electrical lines that service the CRDMs. Electrical power is supplied from a terminal box located in the containment adjacent to the working platform.

10. Reactor Auxiliaries

Makeup and Purification System

The MU&P system regulates the inventory of the RC system during all modes of operation (Figure A-8a). Figures A-8b and -8c are the P&I symbols. Interconnections with other systems have been designed into the system to enable inventory regulation when the RC system is depressurized. The MU&P system and the pressurizer accommodate all inventory changes due to temperature changes during the course of normal operation. The MU&P system maintains the pressurizer inventory at the level selected by the operator and removes corrosion products, fission products, and other impurities from the letdown fluid. The fluid is cooled to an acceptable temperature before purification. When the reactor is secured, the decay heat removal system can be crossed over to the MU&P system thus providing purification under all modes of RCS operation. The MU&P system is used as the vehicle to inject and adjust the concentration of corrosion control chemicals into the RCS. The MU&P system during emergency operation is used to inject HPI water into the RCS. Component design data for the MU system are shown in Table A-7.

Makeup and Purification Function

The design basis letdown flow for the normal operation of purifying the reactor coolant is one RCS volume per day measured at 48.8C (120F) in the letdown line while operating at normal temperature and pressure. Maximum letdown flow is the peak rate of letdown associated with plant heatup measured at 48.8C (120F) in the letdown line. The letdown portion of the MU system is sized and designed to operate with the normal letdown flow as well as with the maximum letdown flow. The use of multiple flow paths for greater than the normal flow rate is considered an acceptable design. The system is designed to reduce the temperature and pressure of the letdown fluid from RC design conditions 343.3C at 17.24 MPag (650F at 2500 psig) to temperature and pressure levels that will not damage components in the flow path 48.8C at 0.517 MPag (120F at 75 psig). The system is designed to regulate the letdown flow rate from the RCS. Purification demineralizer equipment is provided to remove corrosion products and contaminants from the reactor coolant. Ion-exchange resins are selected to remove a minimum amount of lithium. Filter units are installed to eliminate resin carryover and to collect any foreign particulate matter upstream of the makeup tank. The makeup tank is sized to contain the entire expansion inventory associated with plant heatup plus 2.832 m^3 (100 ft^3) minimum water volume and 2.832 m^3 (100 ft^3) minimum air space. The makeup tank is elevated sufficiently above the makeup pump(s) suction so that the required NPSH is available for all operating conditions. Relief valves are provided on the letdown piping, the MU pump recirculation piping, and the makeup tank outlet to protect the system from overpressurization. Hydrogen is added to the makeup tank for oxygen scavenging during hot power operation. The RCS concentration is maintained at 15 to 40 std. CC H_2 per kg reactor coolant. Normal makeup from a makeup pump taking suction on the makeup tank is controlled by the makeup control valve that is regulated by the pressure level controller. Provisions are made to permit a purge flow of RC makeup to be injected into each of the four RC pump motors. A line is provided from the discharge of the MU pumps to supply an alternate high pressure source of pressurizer spray flow. The flow capability is sufficient to provide spray flow in the event that all RC pumps are inoperative during the first phase of cooldown. Interconnection with the DH system is made in a manner that precludes any possibility of overpressurizing any portion of the DH system.

Makeup for Small Leaks

The MU system operating in its normal mode is capable of compensating for small leaks within the RCS to maintain reactor coolant inventory so that an orderly shutdown may be effected. Each MU/HPI pump is designed to supply 0.01104 m^3/s (175 gpm) to the reactor coolant at a RC system pressure of 15.5 MPa (2250 psia). A single active failure need not be considered in the design of the normal makeup system since the HPI function is designed to provide makeup control if a failure occurs in the normal makeup system. The normal makeup system is capable of manual transfer to emergency power supplies.

High-Pressure Injection Function

The HPI function of the MU system is accomplished employing sufficient redundancy in the number of components to provide the required flow rate assuming a single failure. Each MU/HPI pump is capable of providing 0.0221 m^3/s (350 gpm) at a reactor vessel pressure of 1.103×10^7 Pa gage (1600 psig). The

HPI function is activated automatically by a low RCS pressure of 1.103×10^7 Pa gage (1600 psig) or a high containment pressure of 34.5 kPa gage (5 psig). All pumps and valves are capable of being operated from the control room. Remote flow indication for total purge flow is provided. Local indication is provided for each of the four purge lines. A remote alarm is provided for high and low flow conditions for each line. The injection water source and piping to the suction of the MU/HPI pumps is arranged so that adequate NPSH is available for all ranges of pump operation. Missile protection is provided for each HPI line either by physical separation or by appropriate enclosures or shielding.

Decay Heat Removal System

The DH system operates in two functional modes. In normal operation this system removes fission product decay heat from the core during the latter stages of cooldown and shutdown. Under emergency operation this system supplies emergency LPI fluid to the reactor vessel and recirculates fluid from the containment ECCS sump through coolers back to the reactor vessel. The DH system is shown in Figure A-9 and component design data are given in Table A-8.

Normal Operation

The DH function of the DH/LPI system is designed to reduce the RC temperature to 60C (140F) after the steam generators have reduced coolant from 317.8C (604F) to the DH system startup temperature of 148.8C (300F). During initial cooldown heat from the steam generator is dumped to the service water pond. The reactor coolant is reduced from the DH startup temperature to 60C (140F) in 14 hours by maximum utilization of system equipment, i.e., two pumps and two coolers. Each DH cooler is sized for the decay heat load at 20 hours after shutdown. The DH letdown lines from the reactor vessel to the DH pumps are supplied with two motor operated valves inside the containment and one motor operated valve outside the containment. The two valves inside containment are interlocked with RCS pressure instrumentation. This interlock is to close or prevent the opening of these valves when the RCS is at a pressure that, when added to the shutoff head of the pump, exceeds the design pressure of the system.

Separate and redundant flow paths and equipment are provided to prevent a single component failure from reducing the system performance capability.

The DH system is designed so that the RCS cooldown rate can be controlled to keep within the reactor NDT temperature limits.

For normal cooldown depressurization of the pressurizer is effected by spray from the RCS while the RC pumps are still operating. When the RC pumps are shut down, pressurizer spray is supplied from an auxiliary spray line from the DH system. In addition an auxiliary spray line is provided for the MU system for spray capability in the event that the RC pumps are not operating during the first part of the cooldown.

Cross-over lines are provided between the discharge of each DHR cooler and the MU&P system. The flow through these cross-over lines does not exceed the maximum letdown flow and the discharge temperature from the DH cooler is less than or equal to 57.2C (135F). Provision is also made for piping connections from the boric acid addition system.

Emergency Operation — Low Pressure Injection

The LPI function of the DH/LPI system is actuated upon receipt of a SFAS signal. The DH/LPI system is connected to the emergency water storage tanks that are independent of the RC system. This water is used during initial LPI. After depletion of water in the emergency water storage tanks, the DH/LPI system is aligned to take suction from the containment ECCS sump. The DH coolers are then used to remove the heat from the containment sump water.

The DH coolers have reactor coolant on the tube side and component cooling water on the shell side. Separate and redundant flow paths and equipment are provided to prevent a single component failure from reducing the system performance capability. Each flow path has the capability of maintaining core cooling following the LOCA. Appropriate piping and valving are provided for testing the DH/LPI system while the plant is in normal operation. Interconnections with other systems are designed in such a manner that LPI fluid will not deviate from the normal injection path. No pump priming is required under any mode of operation.

Emergency Decay Heat System

The EDH system is designed to remove heat from the RC system in the event of a LOCA, loss of all feedwater flow accident, and/or loss of all electric power (Figure A-10). The heat transfer takes place in the steam generators utilizing the large normal primary to secondary heat transfer surface. During the accidents mentioned above the steam generators are isolated from the normal feed and steam systems and the EDH operation is initiated. The steam trapped in the steam generators when they are isolated is admitted to the steam turbine driven EDH pumps. The EDH pumps which act as feed pumps, take suction on the EWSTs and pump this water into the steam generators. The steam generated therein continues to drive the EDH pumps. Steam produced above the EDH pump turbine requirements is discharged to the cooling system. The amount of feedwater required is determined by safety analysis techniques. The requirement is tentatively set at 0.0473 m^3/s (750 gpm) but may change as a result of further analysis work. The turbine driver design may also be affected if the worst case steam conditions change as a result of changing EDH system flow requirements.

The EDH system is initiated either manually or by the SFAS and operates until steam production ceases or manual control is established.

Connections are provided to allow testing of the EDH turbines and pumps. The feedwater used during the test is taken from the emergency water storage tank and returned to this source. The steam used by the pump turbines during the test is discharged through the EDH piping.

The system is capable of operation independent of electrical power.

The EDH system is designed to meet the requirements of the ASME B&PV Code, Section III, Class 2. The EDH pumps are located such that sufficient NPSH is available to maintain adequate injection for all ranges of pump operation during post-LOCA operations. The steam and feedwater lines of the EDH system connect to the main steam and feedwater lines between the containment wall and the isolation valves. EDH system design parameters and performance data are shown in Table A-9.

Chemical Addition and Boron Recovery System

The chemical addition and boron recovery system (Figures A-11a through A-11c) serve the nuclear plant during normal operations in the following manner:

1. Recovers boric acid from reactor coolant for reuse.
2. Degasifies reactor coolant.
3. Removes boric acid from reactor coolant by demineralization.
4. Prepares, stores, and transfers chemical solutions (boric acid, hydrazine, lithium hydroxide, sodium hydroxide).

5. Stores RC bleed and evaporator distillate.
6. Stores and transfers the quantities of concentrated boric acid needed to achieve cold shutdown of the reactor with a stuck control rod of greatest worth at any time during core life.
7. Changes the boron concentration of the reactor coolant as required during normal operation.

Reactor Plant Service Water System

The RPSW system is designed to supply cooling water to the reactor plant from two sources – the ultimate heat sink (cooling tower) and the existing raw water reservoir. Cooling water for all engineered safety features and for normal operation will use the cooling tower system, which includes a 30-day supply of water in the basin. Service water for other reactor plant cooling needs will be supplied from the raw water reservoir. A separate pump house is provided to furnish 0.15 m^3/s (2400 gpm) to the heat exchangers. The hot water then returns to the pond for atmospheric cooling.

Component Cooling Water System

The function of the CCW system is to transfer heat from the sources listed below to the RPSW system, via the CCW heat exchangers during normal operation of the reactor plant, during scheduled and unscheduled shutdowns of the reactor plant including periods of hot and cold maintenance and during refueling.

1. DH system heat exchangers.
2. MU&P system letdown heat exchangers.
3. RC pumps heat exchangers.
4. CRDM cooling jackets.
5. Suppression pool heat exchangers.
6. Containment dry well cooling system coils.
7. Sampling system heat exchanger.
8. Quench tank heat exchanger.
9. MU pump recirculation heat exchangers.

The CCW transfers heat from the sources listed below during emergency operation of the reactor plant:

1. DH system heat exchangers.
2. Suppression pool heat exchangers.
3. Containment dry well cooling system coils.

The entire CCW system (Figure A-12) is designed to function as a closed-loop cooling system during all phases of operation including shutdowns and emergencies and to operate with a CCW heat exchanger outlet temperature of 35C (95F) with a service water design temperature of 29.4C (85F).

The suppression pool heat exchangers maintain the suppression pool water at a maximum bulk average temperature of 43.3C (110F) with CCW inlet water temperature of 35C (95F). The CRDM cooling jackets are supplied with a minimum of 0.000126 m^3/s (2 gpm) per drive and a maximum of 0.000189 m^3/s (3 gpm) per drive. The design assumes cooling for 41 drives even though current PE-CNSG design is 17 CRDMs.

The CCW system is designed as an ASME Section III, Class 3, USNRC Quality Group C system except for the following which are designed to ASME Section III, Class 2, USNRC Quality Group B: Containment penetration piping and associated isolation valves, containment piping, valves, and equipment serving safety related components cooled normally by the CCW system.

The system has sufficient redundancy so that if maintenance is required or failure occurs on a pump or exchanger, a spare unit is available. The safety related portions of the CCW system have sufficient redundancy so that a single failure of either an active or passive component therein will not prevent the required cooling function from being performed. Normal electric power is supplied to those components whose function is not safety-related. Those safety-related components in the CCW system are supplied by the normal and emergency electric power systems. The CCW system is provided with surge tanks to accommodate volume changes and provide pump suction head and NPSH. The surge tanks are located such that the tank's surge line nozzle is above the elevation of the highest component served by the CCW system. The CCW system pumps, heat exchangers, and surge tanks are located in the reactor compartment. A magnetic filter is installed in the system upstream of the control drive cooling jackets. This filter is used to remove ferrous particles in the CCW fluid which could cause flow blockage and to reduce the concentration of undissolved solids in the entire system. The filter removes all particles larger than 23 microns. Sampling connections are provided downstream of the magnetic filter for verification of water quality. Cooling for the control rod drives is mandatory whenever RC temperature is 93.3C (200F) and above, and at all times that an electric current is imposed upon the stator.

Heatup and possible damage may result if loss of cooling exists for more than approximately 20 minutes. To protect against loss of a pump during operation, provisions are made to insure continued cooling.

The CCW system contains three main circulating pumps, one of which is required to circulate fresh chromated water through selected combinations of the CNSG reactor auxiliary systems. This system transfers heat from the control rod drive stator cooling jackets, the letdown coolers, the RC pump cooler, the containment drywell coolers, the MU pump recirculation coolers, the suppression pool coolers, the sample coolers, quench tank coolers, and the DH coolers to the RPSW system. The CCW fluid is circulated through the shell side of one of two CCW heat exchangers which are cooled by the RPSW system. The component served by this cooling system are arranged in parallel, thus minimizing pumping power requirements and facilitating individual flow regulation. Since the control rod drive stator cooling jackets and RC pump motor heat exchangers are high pressure drop devices, two booster pumps are installed to supply the higher head requirements.

The CCW system is a closed loop cooling system consisting of three full-capacity main circulating pumps, two cooling booster pumps, two full-capacity heat exchangers, a surge tank, a magnetic filter, and the necessary piping, valves, instrumentation, and controls. The CCW system is to be operated during all stages of normal and emergency operation. System data are given in Table A-10.

Cooling During Normal Operation

The conditions that are expected to occur for this operation are cold shutdown, cooldown operation below 149C (300F), cooldown operation above 149C (300F), and power operation. Startup and refueling are also considered under this condition.

Power Operation

During normal power operation of the CNSG, the CCW system will remove heat from (1) one letdown cooler, (2) one containment drywell cooler, (3) one suppression pool cooler, (4) the control rod drive stator cooling jackets, (5) the RC pump motor coolers, (6) the quench tank cooler, (7) one MU pump recirculation cooler, and (8) the sample cooler. For power operation the CCW system requires the use of one main circulating pump, one cooling booster pump, and one heat exchanger. During normal operation one main circulating pump discharges into the header supplying flow to a CCW heat exchanger. The flow from the heat exchanger discharges into the main supply header. All of the components served by the CCW system are arranged in parallel with redundant safety-related components connected separately across the supply and return headers. The outlet cooling water from each component is returned to the main circulating pump via the main return header. The booster pump takes suction on the main supply header and, utilizing the positive suction head provided by the header pressure, furnishes the additional head necessary to provide the required flow through the RC pump motor coolers and CRD cooling jackets. The RCP/CRD cooling loop will be in operation whenever the RC temperature is above 93.3C (200F) and/or the drive stators are electrically energized.

Cooldown Operation Above 149C (300F)

The cooldown above 149C (300F) for the CNSG uses the steam generator and the MU system to reduce the RCS temperature and pressure to a point to allow use of the DHR system. In this cooldown mode the following components require cooling: one makeup pump recirculation cooler, one letdown cooler, one containment drywell cooler, one suppression pool cooler, the CRD cooling jackets, the RCP motor coolers, and possibly the sample cooler and quench tank cooler. The heat load at the beginning of this cooldown condition is essentially the same as for normal power operation. The line up of equipment is the same as described above.

Cooldown Operation Below 149C (300F)

Using the steam generator and MU system takes approximately 6 hours to reach a temperature and pressure that will allow DH system operation. When the DH system is placed in operation, the heat load on the CCW takes a step increase since the heat load becomes equal to the core decay heat. During this stage of cooldown heat removal is required from two DH coolers, a containment drywell cooler, one suppression pool cooler, and possibly the sample cooler. One CCW main circulating pump and two CCW heat exchangers will handle the load. The CRD/RCP loop would be operating, however, the RCP loop would have a minimal heat load since the RC pumps are not operating. When the RC temperature is lowered below 93.3C (200F), the booster pumps can be stopped.

Cold Shutdown

During cold shutdown the CCW system will remove heat from one containment drywell cooler, one suppression pool cooler, one DH cooler, and possibly the sample cooler.

Startup

During the startup operation the CCW system operates the same as described in Power Operation except that the heat loads vary.

Refueling

During the refueling operation heat removal is required from one containment drywell cooler, one suppression pool cooler, and one DH cooler.

Cooling During Emergency Operation

During emergency conditions following a LOCA, the non-safety related heat exchangers (MU pump recirculation coolers, RCP/CRD coolers, quench tank cooler, sample cooler, and letdown coolers) are isolated from the system and the CCW main circulating pumps switch to the emergency mode. The booster pumps are de-energized. Even though both sides of the CCW system are operating, one side is sufficient to remove the safety-related heat loads. Each circuit of the system removes heat from (1) one containment drywell cooler, (2) one suppression pool cooler, and (3) one DH cooler. The RPSW system still serves as the cooling system for the CCW system during emergency operation.

Containment Drywell Cooling System

The CC system removes heat from the containment drywell atmosphere during both normal and emergency LOCA operating conditions (Figure A-13). During normal operation the system removes the heat given off by all hot surfaces within the containment as well as the heat equivalent to the electrical inefficiency in all continuously or intermittently running motors and instrumentation. During LOCA operation the system removes heat from the drywell region and condenses steam generated by the effluent from the primary system break, thereby serving to limit peak pressure and temperature and to return the containment to nearly pre-accident conditions.

The system consists of redundant air handlers (fan/coil units) and ducting. Each air handler comprises a cooling coil, a normal service fan, and an emergency service fan in parallel. The fans are in series with and downstream from the cooling coil.

Design parameters and component data are given in Table A-11.

Reactor Building Ventilation System

The RV system serves the following four primary functions for the PE-CNSG reactor plant:

1. Controlling radioactive gaseous release to the environs during both normal and emergency (post-LOCA) operation and maintenance of a low level of airborne radioactivity in the reactor building and auxiliary spaces to permit entry during normal operation or during both scheduled and unscheduled shutdowns.
2. Removing heat to the environs from sources within the reactor building.
3. Containment purging prior to manned entry for maintenance and/or inspection. Containment purging would be accomplished only when the reactor is at or below hot shutdown conditions; i.e., the decay heat system is operating.
4. Providing a source of clean air to the control areas if high radioactivity levels are present.

A subsystem of the RV system, the containment vacuum system, maintains a subatmospheric pressure in the containment during all normal reactor power operation (Figure A-14).

Three single-speed exhaust fans and two exhaust filter trains (HEPA/activated charcoal/HEPA) are provided. The fans and filters are cross-connected and discharge to two redundant exhaust stacks. One fan, one filter train, and one stack are sufficient to provide the required ventilation and gaseous effluent processing capacity for both normal and emergency operation. Small, easily replaceable roughing filters are provided at the inlets from all auxiliary spaces being served by the RV system. A small HEPA/activated charcoal/HEPA filter train is provided on each reactor compartment fresh air intake to serve the control area air supply fans.

RV system design data are given in Table A-12.

Radwaste Disposal System

The RWD system is comprised of the SWD, LWD, and GWD systems. These systems are designed to collect, store, and dispose of all solid, liquid, and gaseous wastes generated by normal operation of the nuclear steam plant.

Solid Waste Disposal System

The SWD system processes and stores solid waste from the following sources:

1. Demineralizer spent resins.
2. Rags, paper, plastic sheeting, glass, etc.
3. Filter cartridges.

No solid radwaste from the PE-CNSG plant will be dumped onsite. The PE-CNSG utilizes the portland cement solidification system developed by ATCOR Inc. of Peekskill, New York. This system mixes the radioactive waste products with a portland cement-water mixture and fills a 0.208 m^3 (55 U.S. gal) drum container.

The PE-CNSG is estimated to produce 5.66 m^3 (200 ft^3) of spent demineralizer resins from the MU system and the LWD system. The spent resins are transferred from each system to the SWD system spent resin collection/storage tank as shown in Figure A-14. When a sufficient amount of spent resins collects in the tank to warrant operation of the system, the mixing and filling operation begins. The system has the capacity to drum 2.124 m^3 of spent resin per day. Since the filling and capping operation is fully automated, the radiation danger to personnel is minimal.

The system also contains a compactor for the rags, glass, etc. An operator can mix the compacted solids with the spent resins at any time to permit filling the drums completely. The control panel is located in the vicinity of the system but shielded adequately. Filled drums are stored within the ATCOR system on a continuous conveyor until transferred to a certified radwaste storage contractor. The CNSG SWD system has the capacity to store all the solid waste generated during one year's operation of the system.

The concrete surfaces and floors in the SWD system area are finished to prevent the absorption of activity in the event of spillage. Water connections are provided for washdown of all components and the floors slope to a drain for discharge to the LWD system. Adequate area ventilation and atmospheric control are provided to control vapors, gases, and dust resulting from packaging operations. Radiation monitoring equipment is provided to measure the radiation level in each of the packaging areas. Readouts are provided locally and in the main control room.

Component data for the SWD system are shown in Table A-14.

Liquid Waste Disposal System

The LWD system is designed to collect, process, store, and discharge all liquid effluents that have the potential of being radioactive (Figure A-16). The dose and release rates from the system will be as low as is reasonably achievable in accordance with established regulating requirements.

The LWD system is designed to handle 0.757 m^3 (200 gal) per day of radwaste with 0.125% failed fuel. This system also has the capability to process 0.000315 m^3/s (5.0 gpm) RC leak for a specified period of time. The length of time that the LWD system can process this leak is a function of the percent failed fuel.

Additional system data are given in Table A-14.

Gaseous Waste Disposal System

The GWD system is designed to accommodate all gases evolving from the operation of the PE-CNSG system (Figure A-17). The waste gases are collected through a network of vent lines throughout the PE-CNSG. Once collected, the gases are then compressed, stored and discharged from limited access, adequately shielded, and radiologically monitored compartments, each having controlled ventilation to mitigate the spread of contaminated waste gas in the event of radioactive gas leaks.

Ultimate disposition of the waste gases following filtration is to the atmosphere via the reactor compartment ventilation system only when the radioactivity levels are less than specified by the code of Federal Regulations (10 CFR 20 and 10 CFR 50, Appendix I), and only when the wind conditions will ensure that the diluted effluent cannot endanger the operating personnel, populated land areas, or ecology.

Tanks are sized assuming 0.125% defective fuel with 700 EFPD. All planned phases of plant operation are considered. System parameters are given in Table A-15.

As previously stated, all waste gases generated or evolving from operation of the nuclear steam supply system are collected and stored to allow natural decay of the short-lived radioisotopes prior to their release to the atmosphere.

The waste gases are scavenged via a network of vent headers that feed into one of two basic type of waste gas manifolds. One manifold handles waste gases that are potentially hydrogenated and the other handles waste gases that are aerated or have been diluted with air and/or nitrogen.

In addition, every precaution is to be used during equipment design, selection, and location to mitigate the chance of a hydrogen explosion, and the consequences of missile (shrapnel) damage.

Explosions due to oxygen-hydrogen mixing are protected against by maintaining the hydrogenated tanks under positive pressure so that oxygen cannot enter. Any leakage from the system is diluted with fresh ventilation air so that the resulting hydrogen concentrations are below 4% by volume. An oxygen-hydrogen analyzer monitors the gas spaces in the two hydrogenated waste tanks and in other locations in the waste disposal system where hydrogen is most likely to concentrate. The analyzer is set to warn the operator if the oxygen content approaches 2.0% by volume.

<u>Sources of Gaseous Wastes</u>

Potentially hydrogen rich or hydrogenated waste gases are collected from equipment in direct contact with the reactor coolant. Major equipment pieces venting into this portion of the gaseous waste system are as follows:

1. Pressure vessel degassing vent(s).
2. Drywell vacuum line.
3. MU tank vent.
4. Purification demineralizer vent.
5. MU filter vents.
6. MU pump vents.
7. MU pump recirculation cooler vents.
8. DH pump vents.
9. DH cooler vents.
10. Quench tank vent.

The aerated or air-nitrogen diluted gaseous wastes are collected separately from the hydrogenated gaseous waste because of their non-explosive nature and if the aerated waste activity is found to be diluted within the allowable regulatory limits are discharged directly to the atmosphere (following filtration) via the RV system. These waste gases are primarily obtained from displaced void spaces found in tanks, pumps, filters, etc. Examples of larger equipment venting into this manifold are (1) liquid waste storage tanks, (2) liquid waste collection tank, and (3) low activity waste storage tank.

Post-LOCA Combustible Gas Control System

The post-LOCA CGC system is designed to prevent ignition of the hydrogen generated within the containment following a postulated LOCA (Figure A-18). The system includes redundant (125% total capacity) Halon 1301 storage tanks, a distribution system, and associated instrumentation and controls. The SPC system and RV system interface with the CGC system following a LOCA to provide part of the required CGC system function. The SPC system pumps transfer water from the emergency water storage tanks into the suppression pools to force all non-condensables, including hydrogen, back into the containment drywell.

The RV system provides containment purging 1600 hours after the LOCA. For the PE-CNSG design hydrogen is considered to come primarily from radiolysis of water (primary and suppression pools) since there are no chemical sprays, very little aluminum, copper, or zinc, and no appreciable metal-water reaction.

CGC system design data are given in Table A-16.

Sampling System

The SA system provides a means of remotely sampling primary coolant, key auxiliary system effluents, and all waste gases within the GWD system (see Figure A-19). Further, the gas samples may be analyzed for hydrogen and oxygen at any time by the automatic gas analyzer or may be separately analyzed utilizing a gas sample container located at the sample sink. The complete sampling station provides the health physicist or water chemist with samples to monitor RC chemistry, purification demineralizer effectiveness, pH, and other water chemistry criteria; oxygen and hydrogen concentration in gas samples, and the necessary gas samples to maintain radiological waste management. Liquid sample purge effluents are returned to the reactor via the MU system or are discharged to the LWD system. Similarly, gaseous sample purge effluents are discharged to the GWD system.

Suppression Pool Cooling System

The SPC system maintains the suppression pool water temperature and chemistry at the levels which permit safe operation of the PE-CNSG plant (Figure A-20).

The SPC system serves both a normal plant function and an emergency containment heat removal function. During normal plant operation the system:

1. Cools the water in the suppression pool to maintain the water temperature at 43.3C (110F).
2. Provides adequate circulation and mixing of pool water to ensure that all chemicals added to inhibit corrosion and algae growth remain in solution.
3. Provides a source of water to the DH system for long-term core cooling.

During emergency operation following a LOCA, the SPC system:

1. Cools the water in the suppression pool (containment heat removal function).
2. Provides a means to entirely fill the suppression pool wet well spaces to remove any trapped hydrogen so that the combustible gas control system might function properly.

The SPC system has three centrifugal pumps and two shell and tube heat exchangers installed. No normal or accident operating mode requires regulation of system flow other than for on/off capability. No throttling is required. Only one pump and cooler are required to perform the system functions. The equipment and components of the system are located in the reactor building and are designed to meet the requirements of ASME Section III, Class 2.

SPC system parameters and design data are given in Table A-17.

11. Reactor Plant Instrumentation and Control Systems

This section describes the major instrumentation and control systems that support the operation of the PE-CNSG. Component identification numbers are given for the convenience of descriptions and are not for scope definition or establishing total system requirements.

Figure A-21 illustrates typical interfaces of the instrumentation and control systems, the RCS, secondary and tertiary plant systems, and the auxiliary systems.

Integrated Control System

The ICS automatically controls the NSS to achieve the optimum dynamic and steady-state operation of the reactor steam generator complex with a main objective being to fulfill the demand for generated load in response to requests established by the unit operator or the automatic dispatch system.

The ICS incorporates the following basic functions to meet the design objectives:

1. Full integration and coordination of the reactor, the steam bypass system, and the steam generator feedwater system to provide plant control requiring minimum operator participation.
2. The system provides completely automatic control in response to operator or dispatcher demand functions above 15% of rated reactor power.
3. Automatic power runback and limiting action during abnormal operating conditions.
4. Manual control of the unit during startup and abnormal operation.

The plant is controlled by the ICS utilizing a basic feedforward load reference (Figure A-22) developed from the demand for load as conditioned by plant status indicative of the plant's ability to meet that demand. The four major control variables are as follows:

1. Load (generated megawatts).
2. RC average temperature (T_{avg}).
3. Secondary system outlet pressure.
4. Steam generator water inventory.

Three of the above variables are independent, while the fourth is fixed by the values of the other three.

The ICS coordinates the parallel interaction of the RC average temperature, steam generator water inventory, and secondary system outlet pressure, thereby fixing the load at the required value.

Nuclear Instrumentation

The NI is designed to supply the following:

1. The reactor operator with measurement of reactor core leakage information over the full operating range of the reactor.
2. Reactor power signals to the RPS.
3. Reactor power signals to the ICS.
4. Monitoring signals to the plant computer.

In addition, the NI is designed to provide CR withdrawal inhibit signals for acceptable plant startup rates.

Reactor Protection System

The RPS monitors reactor variables related to safe operation of the NSS and trips the reactor when one or more of the reactor variables exceed a setpoint value to drop all CR assemblies except APSRs. Protection is provided with respect to the following plant conditions:

1. RCS overpressurization — the system pressure must be limited to the transient pressure allowable under the ASME Code, Section III (110% of design pressure).
2. DNBR must be maintained at a value to preserve the integrity of the fuel cladding.
3. The maximum linear heat rate (kW/ft) in the core must be limited to avoid fuel melt.

Non-Nuclear Instrumentation and Control Systems

The NNI consists of process-variable sensors, sensor signal conditioning equipment, and certain control elements for non-safety-related systems.

NNI provides sensor inputs to non-safety-related control systems to the plant computer and to indication and alarm readout devices in the main control room. The equipment in this system comprises instrumentation and controls for the reactor auxiliary systems, e.g., DH, MU&P, CCW, etc.

Engineered Safety Features Actuation System

The ESFAS (Figure A-23) is designed to monitor critical plant parameters for indication of a loss of reactor coolant or a secondary system accident and to actuate systems to mitigate the effects of these conditions.

The ESFAS monitors the following system parameters:

1. RC pressure.
2. Containment pressure.
3. Steam generator pressure.

The system issues actuation signals to the pumps, valves, and fans, (as appropriate) for the following protective actions:

1. Protection of the fuel cladding by emergency injection of cooling water (HPI from the MU&P system and LPI from the DH system) into the RCS.
2. Isolation of reactor containment fluid system penetrations that are not vital to accident control.
3. Initiation of secondary system cooling for DH (actuation of auxiliary feedwater valves and pumps).

4. Removal of fission products from the reactor containment atmosphere (initiation of reactor building spray) and reduction of reactor building pressure buildup (initiation of reactor building cooling system).

In addition, the ESFAS provides the interlock signals to the DHR to close and prevent opening that system's letdown isolation valve during RCS pressurization and normal reactor operation.

Safety-Related Control and Instrumentation System

The SRCI system is designed to provide the following:

1. Sufficient indications to the operator in the main control room for monitoring critical parameters during and after design basis accidents to ascertain that an accident has taken place, to determine which accident has occurred, and to perform required control functions after automatic protection system action has taken place.
2. Indicators and controls to enable the operator to shutdown and maintain the reactor in a hot shutdown condition either from the inside or outside control room.
3. Electrical controls and interlocks to the operation of the DH letdown valve to ensure that these valves are positioned properly for normal plant operations.

Control Rod Drive Control System

The CRDCS is designed for two basic functions: (1) to provide normal reactor control (reactivity) through the movement of groups of control rods either automatically in response to signals from the ICS or manually by the operator and (2) to release CRAs into the core upon receipt of a trip signal from the RPS. A typical system is diagrammatically shown in Figure A-24.

Incore Monitoring System

The IMS provides the local neutron flux measurements for monitoring core performance. The flux measurements (as processed by the plant computer) provide information for the following:

1. Reactor operator assistance.
2. Plant physics verification tests.
3. Fuel management decisions.

Information is available to the reactor operator for determining xenon oscillation and core power imbalance (axial flux imbalance and radial flux tilt).

12. Fuel Handling and Storage

The refueling method for the PE-CNSG utilizes wet refueling typical of central power station reactors. Both new and spent fuel storage are housed in the reactor service building.

Fuel Handling Systems

The PE-CNSG fuel handling systems provide a safe, effective means of transporting and handling fuel from arrival at the plant in an unirradiated condition until departure from the plant after post-irradiation cooling. The systems have been designed to minimize the possibility of mishandling, which could cause fuel damage and potential release of fission products.

The land-based PE-CNSG uses a conventional method of "wet" refueling where all operations are performed underwater (see Figures A-25 and A-26). Underwater transfer of spent FAs will provide an effective, transparent radiation shield as well as a reliable cooling medium for removal of decay heat. The use of borated water provides an added safety margin that will ensure subcritical conditions during refueling.

Refueling Procedure and Schedule

A step-by-step refueling procedure has been developed. The time span required for each step has been estimated and a critical path schedule prepared. The step-by-step procedure is shown in Table A-18, and the critical path schedule is shown in Figure A-27.

After the plant is prepared for refueling, the reactor is shut down. A refueling outage commences with the shutdown of the reactor. Following cooldown and depressurization of the RCS, the biological shielding lid, the containment vessel lid, and missile shielding are removed. During these operations the refueling canal is washed down, the control rod drive electrical cables are disconnected, and the control rod drives are uncoupled from their respective control component extensions. These operations permit the initiation of closure head removal. Removal of the reactor closure head thermal insulation, control rod drives, and service structure is begun while the RC level is lowered below the flange. Two stud tensioners used to minimize the time required to remove the head loosen one stud at a time but can be used simultaneously at stud locations 180° apart. Following removal of the studs from the reactor vessel, the studs and nuts are supported in the closure head by specially designed spacers. Removal of the studs with the reactor closure head minimizes handling time and reduces the possibility of thread damage.

Before the reactor closure head is removed, the refueling barrel is installed. The reactor closure head assembly is then maneuvered by a handling fixture supported from the reactor service building crane and is lifted out of the containment onto a head storage stand. The stand is designed to protect the gasket surface of the closure head. The lift is guided by two closure head alignment studs installed in two of the stud holes. These studs also provide proper alignment of the reactor closure head with the reactor vessel and internals when the closure head is replaced after refueling. At the head storage pit special stud and nut handling fixtures can be used to remove the studs and nuts from the reactor closure head for in-service inspection and cleaning. A stud storage rack is provided. The underside of the head is washed down with demineralized water.

After the head is removed, the control rod drive extensions are disconnected from the control rod assembly couplings, and the internals lifting fixture is installed. The refueling barrel and canal are filled with borated water after these operations. The refueling canal is flooded with the two decay heat removal pumps 2.52×10^{-2} m^3/s (400 gpm), which take suction from the borated water storage tank.

The containment crane and an internals handling adapter are used to remove the reactor internals and store them underwater on a stand provided on the floor of the internals storage pit.

Next, the steam generator baffle lifting fixture is installed, and the steam generator baffle is removed underwater so that direct access to the steam generator upper tubesheets is available. The steam generator baffle assembly is stored with the internals in a pit in the floor of the refueling canal to avoid release of airborne activity and for shielding during subsequent draining of the canal for steam generator tube inspection. For steam generator in-service inspection, the water level in the reactor vessel must be lowered to below the steam generator lower tubesheet. Due to radiation effects and shielding requirements, the water level cannot be lowered this much with the FAs in the reactor; therefore, the core is completely removed to permit lowering the water level in the reactor vessel.

Refueling operations are performed underwater using a fuel handling bridge that operates on floor-mounted rails and spans the refueling canal and the spent fuel pool. The main fuel handling bridge is used to shuttle spent fuel assemblies from the core to the spent fuel pool. All FAs are handled underwater by a pneumatically operated fuel grapple attached to a telescoping mast mounted on a trolley that moves laterally on each bridge. Control and orifice rod assemblies are handled by a grapple attached to a second mast. Portable neutron detectors are installed in the reactor, and all fuel is unloaded from the core and transferred to the spent fuel pool.

The fuel and control rod handling mechanisms are designed so that the fuel and rod assemblies are withdrawn into the mast tube for protection before transfer. Interlocks are provided to prevent operation of the bridges or trolleys until the assemblies have been hoisted to the upper limit in the mast tube. Mandatory slow zones are provided for the hoisting mechanisms during insertion of fuel and control assemblies. The slow zones are in effect during insertion into and removal from the reactor core and fuel storage racks. The controls are appropriately interlocked to prevent simultaneous movement of the bridge, trolley, or hoists. The grapple mechanisms are interlocked with the hoists to prevent vertical movement unless they are either fully opened or fully closed. The fuel grapple is designed so that when loaded with the FA, it cannot be opened as a result of operator error or electrical or pneumatic failure. Hard stops are provided to prevent raising an assembly above the minimum shielding depth in the event of an uplimit failure.

All spent FA transfer operations are conducted underwater. The depth of water over the spent FAs during transit through the refueling canal and spent fuel pool provides adequate shielding over the top of the active fuel in the spent FAs during movement from the core into storage. The thickness of the concrete walls and floors of these areas is sufficient to limit the maximum radiation levels in these and adjacent working areas to a safe working level.

Water in the refueling canal, storage pool, and the reactor vessel has a minimum boron concentration of 1800 ppm. Although not required for safe storage of spent FAs, the spent fuel storage pool water is borated so that water in the refueling canal and the reactor vessel is not diluted during fuel transfer operation. The boron concentration of the reactor vessel water is sufficient to maintain core shutdown, if all of the CRAs were removed from the core, even though only a few control rods will be removed at any one time during the fuel shuffling and replacement.

The refueling canal in the containment and the fuel transfer canal, new and spent fuel storage area, cask loading area, and the cask decontainment area in the reactor building are lined with stainless steel for leak tightness and ease of decontamination. The spent fuel storage pool is designed to preclude accidental drainage.

Once all fuel is transferred to the spent fuel pool, the transfer gates between the spent fuel pool and the refueling canal are closed and the neutron detectors are removed. The water in the refueling canal, refueling barrel, and the reactor vessel is drained to an elevation just below the bottom of the steam generator tubes to permit eddy current inspection of the tubes. Water is then drained by the DH removal pumps down to the outlet nozzle of the reactor vessel. The remainder is removed with a 3.15×10^{-2} m^3/s (500 gpm) centrifugal pump suspended from the reactor service crane. This pump discharges to the internals storage pit, which in turn provides suction to the DH pumps.

Once the water level is lowered to the proper level, canal walls, floor, and the refueling barrel are washed down, then inspection equipment is installed and 5% of the steam generator tubes are eddy current-inspected to satisfy in-service inspection requirements. Following inspection, the inspection equipment is removed, and the reactor vessel, refueling barrel, and the refueling canal are refilled with borated water.

During tube inspection, control components are handled underwater by the fuel handling bridge in the spent fuel pool. CRAs are shuffled, and BPRAs are replaced as necessary to obtain proper control positions and even burnup. Fuel assembly shuffling is accomplished as the new FAs and partially spent FAs are reloaded into the reactor vessel.

When flooding of the canal and the control component handling is complete, portable auxiliary neutron detectors are installed; the transfer gates are reopened; and the FAs are reloaded into the reactor.

The steam generator baffle assembly and the internals are replaced in the reactor vessel with the appropriate lifting fixtures. The water in the refueling canal and the refueling barrel is drained, and the canal and barrel are washed down.

After the CRD extensions are connected to the control rod couplings, the refueling barrel is removed and placed in storage. The reactor vessel closure head is replaced using the handling fixture and the alignment studs. When the head is properly positioned, the studs are installed and tensioned simultaneously using two stud tensioners 180° apart.

Following connection of the control rod drives to the control rod drive extensions and reinstallation of electrical service, the reactor vessel closure head leak test is performed. When the leak test results are satisfactory, the head thermal insulation is replaced and the RCS is refilled with coolant; the missile shielding, containment lid, and biological shielding are replaced; and appropriate systems testing including CRD and containment integrity tests are performed. Heatup and pressurization of the reactor plant is then accomplished in accordance with instructions.

Fuel Handling Equipment List

A fuel handling equipment list is presented in Table A-20.

13. Containment and Shielding Design

The containment for the CNSG (Figure A-28) is a compact, vertical cylindrical pressure-suppression containment system similar to those used for some land-based power plants. The steel containment cylinder is 11.58 m (38 ft) in diameter, 19.51 m (64 ft) high, and has a nominal wall thickness of 38.1 mm (1.5 in.). The operating floor, which divides the containment into two separate regions (the dry and wet well chambers), is at approximately the mid-elevation of the containment. The reactor vessel is located off the containment centerline 0.457 m (1.5 ft). The offset of the reactor vessel from the containment centerline provides adequate space for placing the pressurizer on the containment operating floor and as near to the containment centerline as possible.

The reactor vessel is given axial support by attaching the vessel support skirt to the containment floor. Lateral support is provided at the base by the support skirt and at the operating floor by 10 reactor vessel snubbers. The operating floor provides the axial support for the pressurizer.

The region above the operating floor and inside the inner suppression pool wall is the dry well. The wet well is located below the operating floor and outside the inner suppression pool wall.

Access to the containment is provided by two major penetrations — the equipment hatch and the personnel air lock. The equipment hatch is a circular plate located in the containment upper head and fitted to the top of the containment with double seals and bolts. This 5.49-m-diameter (18 ft) opening provides access for initial installation of the major components and for installation and removal of the equipment used during refueling. The personnel air lock, located above the operating floor level provides access to the containment vessel for routine maintenance and inspections. The lock has two double-gasketed doors in series with a mechanical interlock to ensure that one door cannot be opened until the second is sealed. Under controlled conditions, such as during maintenance when the reactor is cold, the interlock system can be overridden to provide unrestricted entry to the containment. Safety equipment includes local and control room door "open" and "closed" visual indicators, pressure equalization valves, remote-closing latching and lighting systems and communication systems. The air locks also have provisions for pressure testing at any time.

There are approximately 17 cartridge-type electrical penetrations through the containment vessel; Figure A-29 shows a typical electrical penetration. Each cartridge has a pressure connection to allow pressurization for testing. The cartridges are seal-welded into penetration sleeves, which are welded into the wall of the containment vessel. The penetration headers through which the cables pass are hermetically sealed.

There are approximately 50 piping penetrations through the containment vessel ranging from 50.8 mm (2 in.) or less to 254 mm (10 in.) in diameter. Penetrations are divided into hot piping penetrations (those which must accommodate thermal movement) and cold piping penetrations (those subjected only to small thermal movement or stress). The penetrations for hot piping, such as the steam lines, allow for both axial and lateral movement during normal operations (Figure A-29). The penetration sleeve is welded to the containment vessel, and a bellows expansion joint is provided to accommodate any movement. The bellows are designed to withstand containment design pressure and have connections to permit testing. The cold piping is welded directly to the penetration sleeves. Bellows are not necessary since the thermal stresses are small and are accounted for in the design of the weld joints.

Containment isolation is maintained with closed piping systems and isolation valves. To minimize leakage of radioactive material to the reactor compartment, a safety isolation signal closes all valved penetrations not required for operation of the engineered safety system. All remotely operated isolation valves have control switches and position indicators in the control room.

Pressure Suppression Design Description

The pressure-suppression system is divided into two separate regions. The inner region contains the suppression pool water, and the outer region is a segmented cylindrical ai, space.

The suppression pool water region comprises an inner cylindrical shell below the operating floor, which forms an annular wet well water space eccentric to the containment center line. The pool is located as close to the reactor vessel as feasible. The eccentric annular wet well contains 12 slosh baffles, which extend downward approximately 1.83 m (6 ft) into the water space from the containment operating floor with one vent pipe discharging between each pair of baffles. These baffles divide the wet well surface into 12 equal areas. Low-pressure rupture discs placed in the vent pipes separate the dry and wet wells to prevent water vapor from entering the dry well. A segmented annular wet well air space between the annular wet well and the containment shell provides an air space for accumulation of non-condensibles and water carried over from the wet well. A low-pressure rupture disc normally separates the wet well water and air spaces. During normal operation, the atmosphere within the containment is maintained at subatmospheric pressure to improve the performance of the pressure-suppression system. The containment dry well cooling system aids in controlling containment pressure during a LOCA.

Biological Shielding

Biological shielding limits radiation exposures to operating personnel and to the public in the following manner:

1. Normal operation and anticipated operational occurrences — to ensure that radiation doses to operating personnel and to the public are within the absorbed dose limits set forth in 10 CFR 20 and 10 CFR 50 and as low as reasonably achievable.
2. Emergency conditions — to ensure that operating personnel are adequately protected and to preclude undue hazards to the public. For operating personnel the applicable regulation is 10 CFR 50, Appendix A,[12] Criterion 19, and for the public, 10 CFR 100.[5]

Access to a PE-CNSG will be supervised at all times. The plant is divided into radiation zones according to the radiation environment and plant operational status. These zones are characterized by the maximum design allowable dose rates permitted and are designated I, II, III, IV, and V in order of increasing intensity of exposure as defined below. Zone V would normally be restricted to areas within the containment and biological shield. Other areas would be determined by detailed shield design and by radiation survey.

Zone	Maximum dose rate, mRem/h	Maximum occupancy
I	0.25	Continuous
II	1.0	12 h/day or 1250 h/qtr
III	12	1 h/day or 12.5 h/qtr
IV	100	8 min/day or 12.5 h/qtr
V	>100	Normally inaccessible

1. Uncontrolled access area (zone I) — an area that can be occupied by plant personnel or visitors on an unlimited time basis with a minimum probability of health hazard from radiation exposure.
2. Controlled access areas (zones II-V) — areas in which radiation levels and/or radioactive contamination can be expected. In general, only plant personnel directly involved in the operation of the plant will be allowed to enter these areas. Maximum occupancy time is determined by the zone design dose rate.

Controlled access areas are identified by radiation caution signs at all points of access and other strategic locations. Access restrictions may be enforced. No inadvertent crossover from an uncontrolled area into a controlled area or vice versa is possible under these conditions.

Uncontrolled access areas are those that will receive radiation dose rates of less than 0.25 mRem/h. These areas can be entered by individuals who have received authorization to enter the plant.

In an emergency personnel in controlled areas will be able to use escape routes to minimize exit time.

High-radiation areas are those in which dose rates of more than 100 mRem/h can be expected. These areas are normally inaccessible since they are blocked off completely or can be entered only through locked or controlled doors.

Shielding materials are selected on the basis of shielding effectiveness, ease of installation, and invariance of shielding properties with respect to time and radiation environment. The biological shielding consists of four distinct categories of shields, which are discussed in turn under the following headings.

Primary Shielding

The primary shielding is defined as all shielding material and structural material inside the containment that serves (1) to limit the radiation level inside the containment during shutdown and (2) to limit the radiation levels outside the containment during normal operation. The major components of the primary shielding are as follows:

Suppression Pool

The suppression pool, in addition to its primary design function, also serves as an effective neutron shield, which, taken in aggregate with the containment shield, limits the neutron levels [and to some extent, the gamma (flux) outside the containment to within the design criteria].

Reactor Cavity Shield

The annular gap between the pressure vessel and the inner suppression pool wall will be fitted with an annular shield plug to limit the radiation streaming up the cavity. This shield will be designed (1) to restrict neutron and gamma fluences to equipment within the containment, (2) to minimize dose rates to refueling and maintenance personnel, and (3) to aid in limiting the dose rates outside the containment during operation.

To facilitate the design of this shield, a test program has been established for a B&W central station power plant, whereby measurements of the neutron and gamma flux profiles will be taken in and around the reactor cavity. These measurements will then be used to establish the calculational methods for the cavity shield design on the PE-CNSG.

Pressure Vessel and Internals

These components provide a large fraction of the required shielding for neutron and gamma core radiation and for fission and activation products in the reactor coolant.

Containment Shield

The containment shield is designed to limit the radiation field outside the containment for both accident conditions and normal operation. In addition, it ensures that dose rates are within the prescribed radiation zones during normal operation due to sources in the containment.

System and Component Shielding

This category of shielding consists of walls, ceilings, floors or pipe chases surrounding process equipment, and piping that may contain radioactive materials. The reactor building is divided into various subcompartments containing specific components, e.g., the makeup demineralizer room. These subcompartments are zoned according to their desired occupancy. The shielding around the subcompartments is based on the radiation zones outside and the gamma source strength inside.

Maximum shield thicknesses required around the major components in the reactor building have been calculated on the basis of the following conservative assumptions:

1. 0.25% of the fuel cladding becomes defective allowing isotopic leakage from the fuel to the coolant.
2. Full power (365 MWt) operation for 700 days.
3. Purification demineralizer resins are flushed to the spent resin tank once per quarter.
4. Liquid waste system demineralizer resins are flushed to the spent resin tank once per year.
5. A demineralizer decontamination factor of 10 exists for all isotopes except noble gases, Y, Mo, and Cs, which are not removed in the demineralizers.
6. 0.38 m^3 (100 gal) of primary coolant leakage per day.
7. 0.38 m^3 (100 gal) of secondary coolant leakage per day.
8. 100% of the leaking noble gases is stored in three gas decay tanks.

Table A-20 lists the resultant shielding thicknesses required around various components.

Table A-1. Dimensional Comparison

	Mark B (15 by 15)		PE-CNSG (15 by 15)	
	Metric	English	Metric	English
No. of FRs per FA	208	208	208 (avg)	208 (avg)
No. of LBPRs per assembly	16	16	16 (avg)	16 (avg)
No. of guide tubes per assembly	16	16	16	16
No. of instr tubes per assembly	1	1	1	1
Fuel rod OD	10.922 mm	0.43 in.	10.922 mm	0.43 in.
Cladding thickness	0.673 mm	0.0265 in.	0.673 mm	0.0265 in.
Fuel rod pitch	14.427 mm	0.568 in.	14.427 mm	0.568 in.
FA pitch spacing	218.11 mm	8.59 in.	218.11 mm	8.59 in.
Guide tube OD	13.462 mm	0.53 in.	13.462 mm	0.53 in.
Instr tube spacer sleeve OD	14.072 mm	0.554 in.	14.072 mm	0.554 in.
Fuel pellet OD	9.398 mm	0.37 in.	9.347 mm	0.368 in.
Fuel pellet length	17.780 mm	0.7 in.	17.780 mm	0.7 in.
Fuel stack length	3658 mm	144 in.	1829 mm	72 in.

Table A-2. Fuel Assembly Components, Materials, and Dimensions

Item	Material[a]	Dimensions, mm	
		Metric, mm	English, in.
Fuel Rod			
Fuel	94% TD UO_2 sintered pellets	9.347 OD	0.368 OD
Cladding	Zircaloy-4	10.922 OD × 9.576 ID	0.43 OD × 0.377 ID
Fuel rod pitch	--	14.427	0.568
Active fuel length	--	1829	72
Nominal fuel-cladding gap (BOL)	--	0.229	0.009
Fuel Assembly			
Fuel assembly pitch	--	218.11	8.59
Overall dimensions	--	2454.28 axial 216.814 × 216.814 lateral	96.62 axial 8.54 × 8.54 lateral
Guide tube (24)	Zircaloy-4	13.462 OD × 12.649 ID	0.53 OD × 0.5 ID
Instr tube (1)	Zircaloy-4	12.522 OD × 11.201 ID	0.493 OD × 0.44 ID
End fitting (2)	Stainless steel	--	--
Spacer grid (8)	Inconel-718	--	--
Spacer sleeves (7)	Zircaloy-4	14.072 OD × 12.751 ID	0.554 OD × 0.502 ID

[a] Mark B and CNSG fuel assemblies use the same materials.

Table A-3. Control Rod Assembly Data

	Metric	English
Number of CRs per assembly	16	16
OD of CR	11.176 mm	0.44 in.
Cladding thickness	0.762 mm	0.03 in.
Cladding material	304 SS, cold-worked	304 SS, cold-worked
End plug material	304 SS, annealed	304 SS, annealed
Spider material	SS, Gr CF3M	SS, Gr CF3M
Poison material	B_4C	B_4C
B_4C pellet OD	9.271 mm	0.365 in.
Length of poison section	1727 mm	68 in.
Stroke of CR	1829 mm	72 in.
Number of assemblies	17	17

Table A-4. Burnable Poison Rod Assembly Data

	Metric (English)
No. of BPRs per assembly	16
OD of BPR	10.922 mm (0.43 in.)
Cladding material	Zircaloy-4
End cap material	Zircaloy-4
Poison material	B_4C in Al_2O_3
Length of poison section	1829 mm (72 in.)
Spider material	SS, Gr CF3M
Pellet OD	5.08 mm (0.20 in.)

Table A-5. Orifice Rod Assembly Data

	Metric (English)
No. of ORAs per assembly	16
OD of orifice rod	12.192 mm (0.48 in.)
Orifice rod material	304 SS, annealed
Spider material	SS, Gr CF3M
Length of orifice rod	406.4 mm (16 in.)

Table A-6. Fuel Loadings

Batch No.	No. of assemblies	Residence time, EFPD	Initial enrichment, % ^{235}U
1A[a]	12	291.5	2.03
1B	16	583.0	2.44
1C	13	874.5	3.01
1D	16	1166.0	3.57
2	12	1166.0	3.57
3	16	1166.0	3.57
4	13	1166.0	3.57
5	16	1166.0	3.57
6	12	1166.0	3.57
7	16	1166.0	3.57
8	13	1166.0	3.57
9	16	1166.0	3.57
10	12	1166.0	3.57
11	16	1166.0	3.57
12	13	1166.0	3.57
13	16	1166.0	3.57

[a] 1A-D, initial fuel loadings.

Table A-7. Makeup and Purification System Design Data

	Metric	English
Letdown Coolers		
No. installed	2	2
Type	Shell/spiral tube	Shell/spiral tube
Heat transferred per cooler	2.29 MW	7.81×10^6 Btu/h
Shell-side flow (CCW)	15.12 kg/s	1.2×10^5 lb/h
Tube-side flow	1.86 kg/s	1.48×10^4 lb/h
Shell-side ΔT	35-71C	95-160F
Tube-side ΔT	93.3-343.3C	200-650F
Material, shell/tube	CS/SS	CS/SS
Design temp, shell/tube	93.3/343.3C	200/650F
Design pressure, shell/tube	1551/17,240 kPag	225/2500 psig
Purification Demineralizers		
No. installed	2	2
Material	SS	SS
Volume, resin	0.85 m^3	30 ft^3
Design temperature	79.4C	175F
Operating temperature	48.8C	120F
Design pressure	1379 kPag	200 psig
Operating pressure	103-241 kPag	15-35 psig
Maximum flow	0.00189 m^3/s	30 gpm
Operating flow	0.00158 m^3/s	25 gpm
Purification Filter		
No. installed	2	2
Type	Disp cartridge	Disp cartridge
Material		
Size, ABS	23 μ	23 μ
Design temperature	79.4C	175F
Operating temperature	48.8C	120F
Design pressure	1379 kPag	200 psig
Operating pressure	103-241 kPag	15-35 psig

Table A-7. (Cont'd)

	Metric	English
Maximum flow	0.00189 m^3/s	30 gpm
Operating flow	0.00158 m^3/s	25 gpm
Makeup Tank		
No. installed	1	1
Material	SS or CS/SS	SS or CS/SS
Volume	39.2 m^3	1385 ft^3
Design temperature	79.4C	175F
Operating temperature	48.8C	120F
Design pressure	689 kPag	100 psig
Operating pressure	103-241 kPag	15-35 psig
Makeup Pumps		
Number	3	3
Type	Vertical, multi-stage, centrifugal	
Capacity/head	0.011/1646	175/5400
Pump material	SS wetted parts	SS wetted parts
Design temperature	176.6C	350F
Operating temperature	48.8C	120F
Design pressure	17,240 kPag	2500 psig
Operating pressure	15,860 kPag	2300 psig
Makeup Pump Recirculation Cooler		
Number	2	2
Type	Shell/spiral tube	Shell/spiral tube
Heat transferred per cooler	0.15 MW	5×10^5 Btu/h
Shell-side flow (CCW)	3.15 kg/s	2.5×10^4 lb/h
Tube-side flow	3.15 kg/s	2.5×10^4 lb/h
Shell-side ΔT	35-46.1C	95-115F
Tube-side ΔT	60-48.8C	140-120F
Material, shell/tube	CS/SS	CS/SS
Design temp, shell/tube	93.3/79.4C	200/175F
Design pressure, shell/tube	1551/1379 kPag	225/200 psig

Table A-8. Decay Heat Removal System Design Data

	Metric	English
Decay Heat Exchangers		
Quantity	2	2
Type	Shell/tube	Shell/tube
Design heat load @ 60C (140F)	1.05 MW	3.6×10^6 Btu/h
Shell-side flow (CCW)	3.15 kg/s	25,000 lb/h
Tube-side flow	3.15 kg/s	25,000 lb/h
Design pressure, shell/tube	1034/5516 kPa	150/800 psig
Design temperature, shell/tube	107.2/176.6C	225/350F
Material, shell/tube	CS/SS	CS/SS
Decay Heat Pumps		
Quantity	2	2
Type	Centrifugal	Centrifugal
Capacity/head		
Decay heat removal	$0.0315\ m^3/s/73.8\ m$	500 gpm/242 ft
Low-pressure injection	$0.0252\ m^3/s/160\ m$	400 gpm/525 ft
Material	SS	SS

Table A-9. Emergency Decay Heat System Design Data

	Metric	English
System Parameters		
Steam system design pressure	7584 kPag	1100 psig
Steam system design temp	321C	610F
FW system design pressure	Later	Later
FW system design temperature	Later	Later
Total system dry weight	22,700 kg	50,000 lb
ASME Safety Class, USNRC Quality Class	III, 2/B	III, 2/B
Feedwater Pump		
Quantity	2	2
Type	Centrifugal	Centrifugal
Design (rated) flow rate (each)	0.0473 m^3/s	750 gpm
Total developed head	Later	Later
Feed temperature	35C	95F
Brake horsepower (each)	Later	Later
Material, casing/impeller/shaft	CS/SS/SS	CS/SS/SS
Emergency Decay Heat Turbine		
Quantity	2	2
Type	Steam, Terry Steam Turbine Co. Model Zr-4 or equivalent	
Turbine steam pressure (max)	4620 kPag	670 psig
Turbine steam temp (max)	279C	535F
Turbine exhaust pressure	Later	Later
Turbine horsepower	Later	Later
Turbine steam consumption	Later	Later
Material	CS	CS

Table A-10. Component Cooling Water System Design Data

	Metric	English
System Parameters		
Design temp	107.2C	225F
Design pressure	1034 kPag	150 psig
Component Coolers		
Quantity	2	2
Type	Shell/tube	Shell/tube
Design heat load (each)	2.93 MW	10×10^6 Btu/h
Tube side (service water)		
Flow	0.1136 m^3/s	1800 gpm
Design inlet temp	29.4C	85F
Normal oper. inlet temp range	4.4-29.4C	40-85F
Material	90/10 CuNi	90/10 CuNi
Shell side (chromated)		
Flow	0.0946 m^3/s	1500 gpm
Normal oper. temp, inlet/outlet	46.1-35C	115-95F
Material	CS	CS
Main Circulating Pumps		
Quantity/service	3/chromated water	3/chromated water
Type	Centrifugal	Centrifugal
Design (rated) flow rate (each)	0.0946 m^3/s	1500 gpm
Total dev. head	559 kPag	187 ft H_2O
Motor power (each)	93.25 kW	125 hp
Electrical requirements	440 Vac, 60 Hz, 3ϕ	440 Vac, 60 Hz, 3ϕ
Material	CS	CS
Booster Pumps		
Quantity/service	2/chromated water	2/chromated water
Type	Centrifugal	Centrifugal
Design (rated) flow rate (each)	0.025 m^3/s	400 gpm
Total dev. head	747 kPag	250 ft H_2O
Motor power (each)	22.4 kW	30 hp
Electrical requirements	440 Vac, 60 Hz, 3ϕ	440 Vac, 60 Hz, 3ϕ
Material	SS	SS

Table A-10. (Cont'd)

	Metric	English
Magnetic Filter		
Quantity	1	1
Design flow	0.0252 m^3/s	400 gpm
Removal rating (nominal)	23 μ	23 μ
Surge Tank		
Quantity	2	2
Capacity	0.566 m^3	20 ft^3
Operating pressure	101.3 kPa	Atmospheric
Operating temp	35C	95F
Material	CS	CS
CRD Cooling Jackets		
Quantity (1/CRDM)	17	17
Heat load (per CRDM)	7.62 kW	2.6×10^4 Btu/h
Flow per jacket (min/max)	0.0013/0.00019 m^3/s	2/3 gpm
Cooling water inlet temp (min/max)	15.6/48.8C	60/120F
Material (wetted surface)	304 SS or equiv.	304 SS or equiv.

Table A-11. Containment Dry Well Cooling System Design Data

	Metric	English
Normal Operation		
Design heat load		
Hot surfaces within containment excluding CRDMs	56.9 kW	194,300 Btu/h
Control rod drives	66.3 kW	226,400 Btu/h
Total	123.2 kW	420,700 Btu/h
Air flow distribution		
Flow through CRD, service structure (downflow)	3.077 m^3/s	6520 cfm
Air drawn through dry well (excluding flow through CRDM service structure)	6.60 m^3/s	13,980 cfm
Total	9.68 m^3/s	20,500 cfm
Air temperatures		
At distribution points	37.7C	100F
Design bulk average in containment	48.8C	120F
Inlet to mixing box from dry well atmosphere	60.0C	140F
Inlet to CRDM service structure (top)	37.7C	100F
Average temperature inside CRDM service structure	65.5C	150F
Outlet from CRDM service structure (bottom) into mix box	93.3C	200F
Average temperature inside mixing box (coil inlet)	70.5C	159F
Normal operating humidity range	60-100%	60-100%
Emergency Operation (Post-LOCA)		
Air flow,	4.84 m^3/s	10,250 scfm
Peak temperature	165.5C	330F
Peak ambient pressure	724 kPag	105 psig

Table A-11. (Cont'd)

	Metric	English
Normal Service Fans		
Quantity	2	2
Type	Single-speed, variable pitch, vaneaxial, draw through	
Capacity per fan, flow/rpm/pressure	9.68 m^3/s/2400/1.99 kPag	20500 scfm/ 2400/8 in. WG
Estimated motor power per fan	29.84 kW	40 hp
Electrical requirements	440 V ac, 60 Hz, 3ϕ	440 V ac, 60 Hz, 3ϕ
Emergency Service Fans		
Quantity	2	2
Type	Single-speed, variable-pitch, vaneaxial, draw through	
Capacity, flow/rpm/pressure	0.484 m^3/s/1200/0.498 kPag	10,250 scfm/1200/ 2.0 in. WG
Estimated motor power (each)	29.84 kW	40 hp
Electrical requirements	440 V ac, 60 Hz, 3 ϕ	440 V ac, 60 Hz, 3ϕ
Containment Cooling Coils		
Quantity	2	2
Type	Finned tube	Finned tube
Total head load (sensible) normal operation	123.2 kW	420,700 Btu/h
Material		
Tubes	90/10 CuNi	90/10 CuNi
Fins	Cu	Cu
Air side normal operation		
Air flow velocity over coils	4.17 m/s	820 fpm

Table A-11. (Cont'd)

	Metric	English
Pressure drop	0.323 kPag	1.3 in. WG
Air temperature (in/out)	70.5/37.7C	159/100F
Air side emergency operation		
Peak flow velocity over coils	Later	Later
Inlet temperature	48.8-165.5C	120-330F
Peak heat removal capacity	Later	Later
Air side design pressure	861.8 kPa	125 psia
Air side design temperature	176.6	350F
Water side normal operation		
Water flow	0.011 m^3/s	175 gpm
Normal inlet temperature	35C	95F
Normal outlet temperature	37.6C	99.8F
Pressure drop	17.91 kPa	6 ft H_2O
Water side emergency operation		
Water flow	0.0221 m^3/s	350 gpm
Normal inlet temperature	35C	95F
Peak outlet temperature	Later	Later
Pressure drop	71.66 kPa	24 ft H_2O
Ducting, Fittings, Cabinets, Piping		
Ducting		
Design pressure in either direction	2.49 kPag	10 in. WG
Design temperature	1766C	350
Material	CS, galv.	CS, galv.
Piping		
Design pressure	Sch 40, CS	Sch 40, CS
Design temperature	176.6 C	350 F

Table A-12. Reactor Compartment Ventilation System Design Data

Exhaust Filter Trains

Particulate filters	
Quantity	4
Type	HEPA, dry
Media	Glass fiber (waterproof, fire-retardant)
Rated flow (each)	6.61 m^3/s (14,000 scfm)
Separators	Asbestos (waterproof)
Cell side or [illegible]	Cadmium-plated CS
Face guards	4-mesh galv hardware cloth
Gasketing	ASTM D-1056, Grade SCE-43
Efficiency	99.97% with 0.3 μ dia. DOP
Rating basis	MIL-STD-282
Rated pressure drop unloaded	248.9 Pa (1.0 in. WG)
Codes	Health and Safety Bulletin 306, March 31, 1971; UL-586, MIL-STD 282, May 28, 1965
Charcoal filters	
Quantity	2
Rated flow per charcoal unit	6.61 m^3/s (14,000 scfm)
Type	Activated coconut shell-impreg
Granule size	8-16 mesh
Ignition temperature	360C (680F)
Maximum moisture content	3%
Gasketing material	ASTM D-1056, Fr. SCE-43
Casing	Type 304 SS
Efficiency, %	99.9 elem. iodine @ 0.2032 m/s (40 fpm), 25C, 90% RH 99.5 methyl iodine @ 0.2032 m/s (40 fpm), 25C, 90% RH 99.9 freon-112 based on AEC-DP-1082 (July 1967)
Residence time	0.50 s

Rated pressure drop	498 Pa (2.0 in. WG)
Codes	AEC-DP-1082 (July 1967)

Fresh Air Inlet Filters to Control Room

Particulate filters

Quantity	2
Type	HEPA, dry
Media	Glass fiber (waterprrof, fire-retardant)
Rated flow (each)	2.360 m^3/s (5000 scfm)
Separators	Asbestos (waterproof)
Cell-side material	Cadmium-plated carbon steel
Face guards	4-mesh galvanized hardward cloth
Gasketing	ASME D-1056, Grade SCE-43
Efficiency	99.97% with 0.3 μ dia. DOP
Rating basis	MIL-STD-282
Rated pressure drop unloaded	249 Pa (1.0 in. WG)
Codes	Health and Safety Bulletin 306, March 31, 1971; UL-586, MIL-STD-282, May 28, 1965

Charcoal filter

Quantity	1
Rated flow per charcoal unit	2.360 m^3/s (5000 scfm)
Type	Activated coconut shell-impregnated
Granule size	8-16 mesh
Ignition temperature	360C (680F)
Maximum moisture content, %	3
Gasketing material	ASTM D-1056, GR. SCE-43
Casing	Type 304 stainless steel
Efficiency, %	99.9 elem. iodine at 0.2032 m/s (40 fpm), 25C and 90% RH 99.5, methyl iodine 0.2032 m/s (40 fpm @ 25C), and 90% RH 99.9 freon-112 based on AEC-DP-1082 (July 1967)

Table A-12. Cont'd)

Residence time, seconds	0.25
Rated pressure drop	249 Pa (1.0 in. WG)
Codes	AEC-DP-1082 (July 1967)

Prefilters (Roughing Filters)

Reactor compartment fresh air intake filters

Quantity	2
Type	Later
Media	Glass fiber (waterproof, fire-retardant)
Rated flow	5.19 m^3/s (11,000 scfm)
Efficiency	50% NBS (dust spot) min
Rated pressure drop unloaded	24.9-125 Pa (0.1-0.5 in. WG)

Exhaust duct intake filters

Quantity	Later
Type	Later
Media	Glass fiber (waterproof, fire-retardant)
Rated flow	Later
Efficiency	50% NBS (dust spot) min
Rated pressure drop unloaded	24.9-125 Pa (0.1-0.5 in. WG)

Exhaust Fans

Quantity	3
Rated flow, (each)	6.56 m^3/s (13,900 scfm)
Discharge pressure	4980 Pa (20 in. WG)
Estimated brake horsepower (each)	47 kW (63 hp)
Estimated motor horsepower (each)	56 kW (75 hp)
Material	Carbon steel, galvanized
Electrical requirements	440 Vac, 60 Hz, 3ϕ

Table A-12. (Cont'd)

Vacuum Pumps

Quantity	2
Type	Rotary, water seal
Capacity	0.0708 m^3/s (150 scfm)
Estimated brake horsepower	7.46 kW (10 hp)
Normal operating pressure range (suction/discharge)	34-48 kPa/atm (5-7 psia/atm)
Estimated motor horsepower	11.2 kW (15 hp)
Electrical requirements	440 Vac, 60 Hz, 3ϕ

Table A-13. Solid Waste Disposal System Component Data

	Metric	English
Spent Resin Storage Tank		
Volume	7.079 m^3	250 ft^3
Design pressure	2.758 kPag	4 psig
Design temperature	65.5C	150F
Material	SS	--
Code	ASME III, Class 3	--
Spent Resin Transfer Pump		
Quantity	2	--
Type	Centrifugal	--
Capacity/head	0.003155 m^3/s/42.67 m	50 gpm/140 ft
Design temperature	93.3C	200F
Code	ASME VIII	--

Table A-14. Liquid Waste Disposal System Design Data

	Metric	English
System Parameters		
Design pressure	1034 kPag	150 psig
Design temperature	93.3C	200F
Liquid Waste Storage Tank		
Quantity	2	2
Volume	17 m^3	600 ft^3
Design pressure	138 kPag	20 psig
Operating pressure	20.7-34.5 kPag	3-5 psig
Design temperature	93.3C	200F
Material	304 SS	304 SS
Liquid Waste Collection Tank		
Quantity	1	1
Volume	17 m^3	600 ft^3
Design pressure	138 kPag	20 psig
Operating pressure	20.7-34.5 kPag	3-5 psig
Design temperature	93.3C	200F
Material	304 SS	304 SS
Low-Activity Waste Storage Tank		
Quantity	1	1
Volume	7.08 m^3	250 ft^3
Design pressure	138 kPag	20 psig
Operating pressure	20.7-34.5 kPag	3-5 psig
Design temperature	93.3C	200F
Material	304 SS	304 SS
Purification Demineralizer		
Quantity	3	3
Type	Mixed bed	Mixed bed
Vessel material	304 SS	304 SS

Table A-14. (Cont'd)

	Metric	English
Resin volume per unit	0.425 m^3	15 ft^3
Flow	0.00189 m^3/s	30 gpm
Design pressure	1034 kPag	150 psig
Design temp	93.3C	200F
Demineralizer Filter		
Quantity	1	1
Type	Disposable element	Disposable element
Vessel material	304 SS	304 SS
Micron size (nominal)	75 μ	75 μ
Flow	0.00189 m^3/s	30 gpm
Design pressure	1034 kPag	150 psig
Design temperature	93.3C	200F
Demineralizer Filter		
Quantity	1	1
Type	Disposable element	Disposable element
Vessel material	304 SS	304 SS
Micron size (abs)	10	10 μ
Flow	0.00189 m^3/s	30 gpm
Design pressure	1034 kPag	150 psig
Design temperature	93.3C	200F
Waste Transfer Pumps		
Quantity	4	4
Type	Centrifugal	Centrifugal
Material	SS wetted parts	SS wetted parts
Capacity	0.00189 m^3/s	30 gpm
Head	289.3 kPa	120 ft H_2O
Motor power	1.12 kW	1.5 hp
Design pressure	1034 kPag	150 psig
Design temperature	93.3C	200F
Electrical requirements	440 Vac, 60 Hz, 3ϕ	440 Vac, 60 Hz, 3ϕ

Table A-15. Gaseous Waste Disposal System Design Data

Design pressure	1379 kPag (200 psig)
Design temperature	149C (300F)
Waste Gas Compressor	
No. of units	3
Type	Diaphragm
Operating discharge pressure range	atm-1207 kPag (atm-175 psig)
Operating suction pressure range	69-241 kPag (10-35 psig)
Approximate compressor brake power	2.16kW (2.9 hp)
Approximate motor power	2.24kW (3.0 hp)
Material (wetted surfaces)	304SS
Electrical requirements	440 V ac, 60 Hz, 3ϕ
Gaseous Waste Decay Tank	
No. of units	5
Usable volume	2.12 m^3 (75 ft^3)
Design pressure	1379 kPag (200 psig)
Design temperature	149C (300F)
Material	SS, CS w/paint, or fiberglass
Hydrogen-Oxygen Gas Analyzer	
No. of units	1
Capacity	10 samples
Electrical requirements	110 V ac, 60 Hz
Exhaust Filters	
Particulate filters	
Quantity	2
Type	HEPA (high efficiency particulate air), dry
Media	Glass fiber (waterproof, fire retardant)
Rated flow (each)	0.00944 m^3/s (20 scfm)
Separators	Asbestos (waterproof)
Cell side material	304SS
Face guards	4-mesh stainless steel hardware cloth
Gasketing	ASTM D-1056 Grade SCE-43
Efficiency	99.97% with 0.3 micron-diameter DOP
Rating basis	MIL-STD-282
Codes:	Health and safety Bulletin 306, dated March 31, 1971 UL-586 MIL-STD 282, dated May 28, 1965

Table A-15. (Cont'd)

Charcoal Filters

Rated flow per charcoal unit	0.00944 m^3/s (20 scfm)
Type	Activated coconut shell, impregnated
Granule size	10-14 mesh
Ignition temperature	360C (680F)
Maximum moisture content (%)	3
Gasketing material	ASTM-D1056, Gr SCE-43
Casing	304SS
Efficiency (%)	99.9, elemental iodine, at 0.2032 m/s (40 fpm) 25C and 90 percent relative humidity 99.5, methyl iodide, at 0.2032 m/s (40 fpm) 25C and 90 percent relative humidity. 99.9, freon 112, based on AEC-DP-10S2, July 1967
Retentivity, seconds	3.0
Codes	AEC-DP-1082 (July 1967)

Table A-16. Post-LOCA Combustible Gas Control System Design Data

Data description		Parameter
Post-LOCA Combustible Gas Control System		
(12050/A2.5)		
1.0	Inerting agent	
	Type	Gaseous, chemical flame inhibitor
	Generic name	Halon
	Chemical composition	CF_3Br (Halon 1301)
	Quality	In compliance with Mil-M-12218
2.0	Required concentration, vol % (peak)	~64
3.0	Volume to be inerted	~1274 m^3 (~4.5 × 10^4 ft^3)
4.0	Maximum allowable un-inerted H_2 concentration in 5 volume % O_2, vol %	4
5.0	Maximum allowable un-inerted O_2 concentration in 4 volume % H_2, vol %	5
6.0	Halon 1301 injection	
	H_2 concentration at time of injection, vol % (maximum)	3.5
	Time after LOCA, h	~10
7.0	Quantity of inertant	
	Required for inerting	~8384 Kg (~18484 lbm)
	Available (125% of required)	~10480 Kg (~23105 lbm)
8.0	Tankage	
	Quantity	5
	Required capacity (each)	1.9 m^3 (67 ft^3)
	Halon 1301 storage density (maximum)	1121 Kg/m^3 (70 lbm/ft^3)
	Storage pressure @ 21.1C (70F)	2482 kPag (360 ± 5% psig)
	Pressurizing agent	Nitrogen
	Design temperature	93.3C (200F)
	Design pressure	10860 kPag (1575 psig)
	Material	Carbon steel

Table A-16. (Cont'd)

9.0	Halon 1301 distribution nozzles	
	Size	Later
	Material	SS
10.0	Suppression pool fill pumps	
	Suppression pool cooling system Pumps perform this function	
	Quantity	3
	Time to fill suppression pools with 1 pump operating, hr	Later
	Flow (each)	Later
	Total developed head	Later
11.0	System code class/quality group	
	ASME	Section III, Class 2
	USNRC	Quality group B

Table A-17. Suppression Pool Cooling System Design Data

	Metric	English
Design pressure	1378 kPag	200 psig
Design temperature	93.3C	200F
Suppression-pool heat exchangers		
Number	2	2
Type	Shell & tube	Shell & tube
Design heat load	3.52 kW	12,000 Btu/h
Flow (shell/tube)	1.52/1.52 kg/s	12,000/12,000 lb/h
Material	CS shell & tube	CS shell & tube
Suppression-pool cooling pumps		
Number	3	3
Type	Centrifugal	Centrifugal
Capacity	0.00631 m^3/s	100 gpm
Head	45.7 m	150 ft
Material	CS	CS

Table A-18. Typical Refueling Procedure

Step	Description
1	Checkout building services
2	Checkout fuel handling equipment
3	Inspect and store new fuel
4	Inspect and store new control rod and burnable poison rod assemblies
5	Reduce power, shut down reactor and borate coolant
6	Cooldown and depressurize reactor coolant system
7	Ventilate containment vessel
8	Disconnect control rod drive service and electrical lines
9	Remove control rod drive housing caps
10	Disconnect CRDs from extensions
11	Remove biological shielding
12	Wash refueling area
13	Detension and remove containment vessel lid studs
14	Remove containment vessel lid
15	Replace CV lid O-rings
16	Remove missile shield
17	Remove reactor vessel closure head thermal insulation
18	Lower reactor coolant level to below flange
19	Detension RV closure head studs
20	Remove 2 studs, install alignment studs, install spacers under other studs
21	Remove reactor vessel closure head
22	Inspect studs
23	Replace RV closure head O-rings
24	Control rod drive maintenance and replacement
25	Disconnect CRD extensions from CRA couplings
26	Install protective ring
27	Remove internals with transfer cask
28	Install SG baffle lifting fixture

Table A-18. (Cont'd)

Step	Description
29	Remove steam generator baffle
30	Install manipulator
31	Install portable auxiliary neutron detectors
32	Unload all fuel from core (see Table D-2).
33	Remove portable neutron detectors
34	Control component handling in pool (see Table D-3)
35	Failed fuel detection and handling
36	Remove manipulator
37	Lower reactor coolant to below steam generators
38	Steam generator tube inspection: Set up equipment Inspection Tear down equipment
39	Raise reactor coolant level
40	Install manipulator
41	Install portable auxiliary neutron detectors
42	Load fuel (see Table D-2)
43	Remove portable neutron detectors
44	Remove manipulator
45	Install steam generator baffle and remove lifting fixture
46	Replace internals with transfer cask
47	Remove protective ring
48	Connect CRD extensions to CRA couplings
49	Replace reactor vessel closure head
50	Install RV closure head studs
51	Remove alignment studs
52	Connect CRDs to extensions
53	Install and leak test CRD housing caps
54	Reinstall CRDM service and electrical lines
55	Tension RV closure head studs
56	RV closure head leak test
57	Replace RV closure head thermal insulation

Table A-18. (Cont'd)

Step	Description
58	Refill reactor coolant system (includes venting CRD housings and RC system)
59	Replace missile shield
60	Inspect containment
61	Post-refueling CRD test
62	Replace containment vessel lid, tension studs, O-ring test
63	Containment integrity test
64	Decontaminate refueling area
65	Replace biological shielding
66	Heat up reactor coolant system
67	Leak rate test RC system: Pressurizer level, makeup rate, flange O-rings
68	Let down reactor coolant to adjust boron concentration
69	Post critical physics tests
70	Remove and store refueling equipment

Table A-19. Refueling Fuel Handling Equipment List

Pneumatic wrenches
Leadscrew servicing tools and equipment
Service structure monorail trollies
Stud tensioners
Closure head stud lifting tool
Reactor vessel cover/work platform
CRDM extension servicing equipment
Reactor vessel closure head support stand
New fuel handling tool
Reactor compartment cover handling rig
Stud tensioner handling rig
Closure head handling/leveling rig
Vessel cover/work platform handling rig
New fuel handling rig
CRD extension servicing stand handling rig
Vessel closure head stud handling rig
Reactor service bridge crane
Refueling bridge (with extended reach)
Spent fuel pool
Shipping cask loading pit
Cask maintenance pit
Fuel and cask handling bridge crane
Spent fuel pool cooling and cleanup system
Fuel receiving and inspection area
New fuel storage vault
Reactor closure head storage pit
Refueling barrel

Table A-20. Component Shielding Thickness

Component	Minimum steel thickness Metric, mm	English, in.
Liquid waste demineralizer	150	6
Waste collection and storage tanks	110	4.375
Liquid waste transfer pumps (operating)	90	3.5
Resin flush piping (LWS)	100	4
LWS filters	80	3.125
Spent resin tank	280	11
Drum storage area	260	10.25
Gas decay tank	290	11.375
Waste gas compressor (operating)	270	10.625
Letdown line	80	3.125
Purification demineralizer	300	11.75
Makeup tank	150	6
Makeup pump (operating)	150	6
Resin flush piping (makeup system)	220	8.625
Makeup system filters	280	11

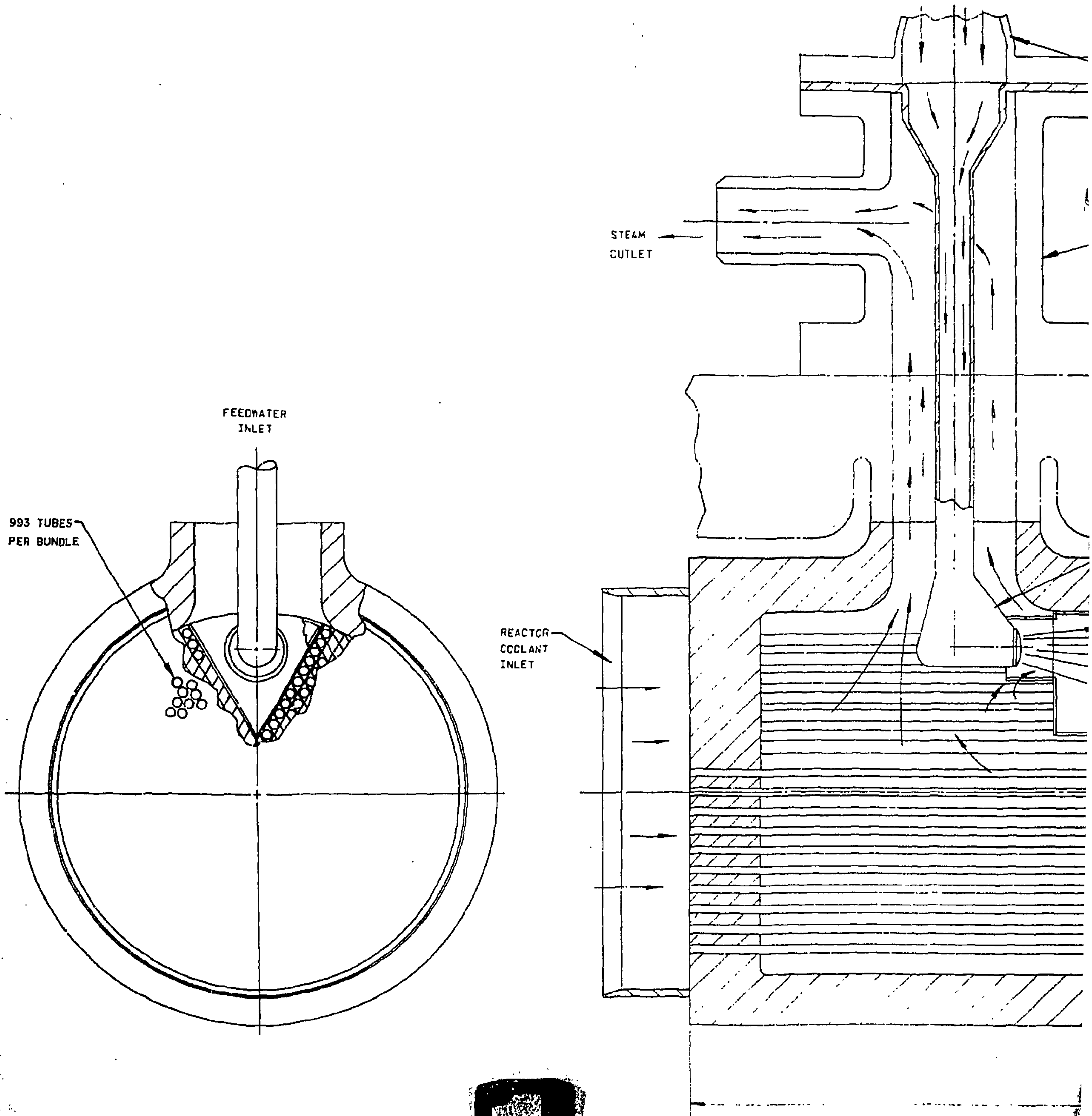

FEEDWATER INLET
993 TUBES PER BUNDLE
STEAM OUTLET
REACTOR COOLANT INLET

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Figure A-1. Steam Generator Module

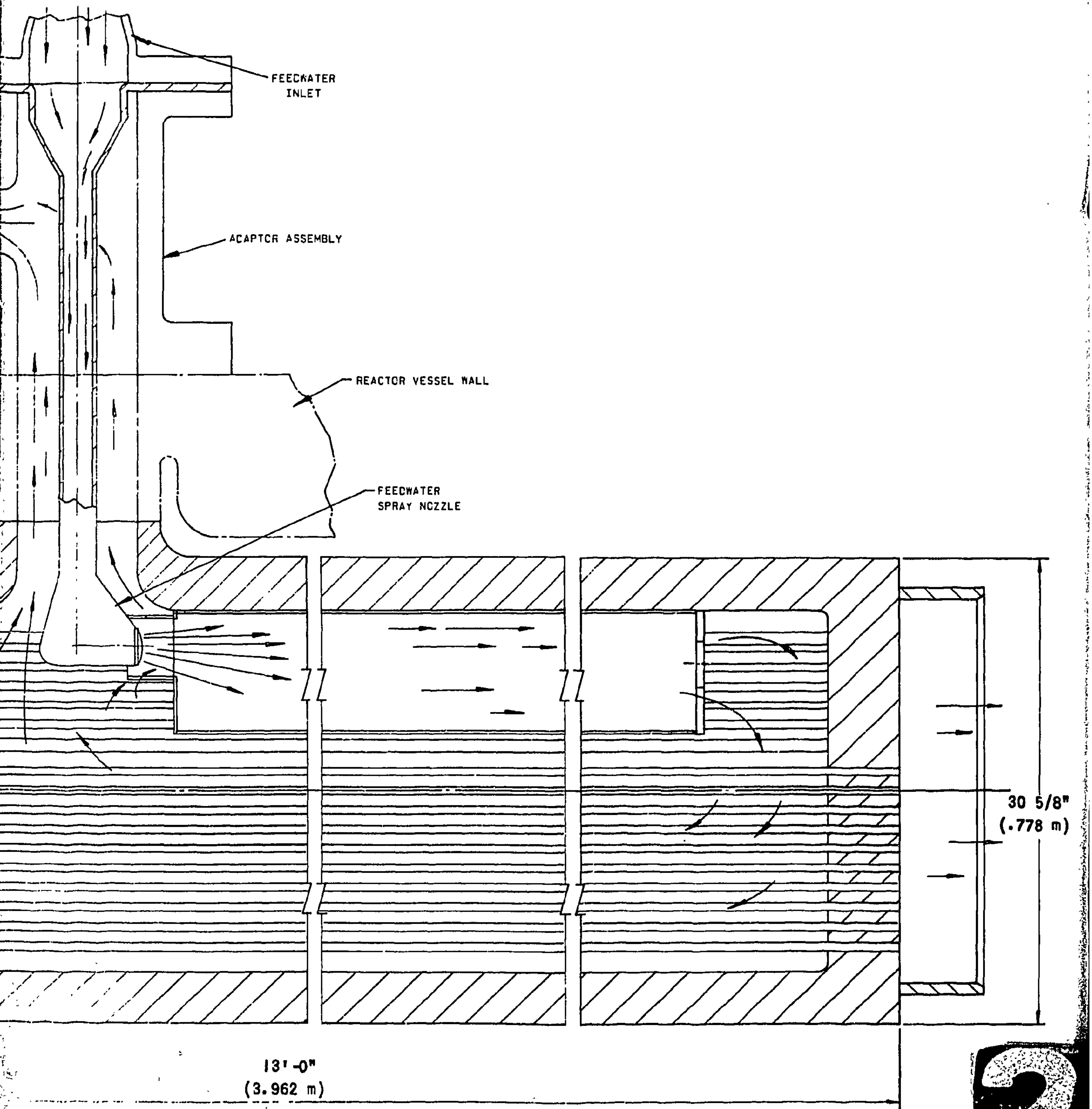

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Figure A-2. Pressurizer

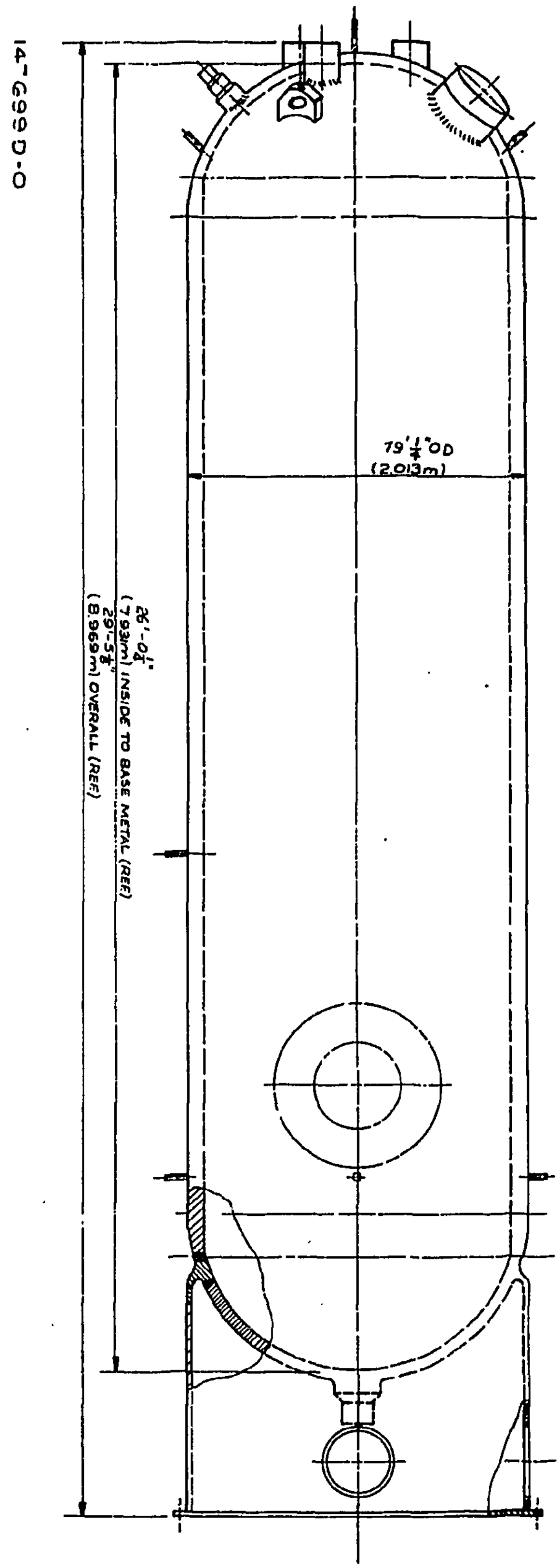

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Figure A-3. Typical LBP Loading Per Fuel Assembly

			4	20	4			
		4	20	20	20	4		
	4	20	24	24	24	20	4	
4	20	24	24	24	24	24	20	4
20	20	24	24	24	24	24	20	20
4	20	24	24	24	24	24	20	4
	4	20	24	24	24	20	4	
		4	20	20	20	4		
			4	20	4			

Number of LBP rods per fuel assembly.

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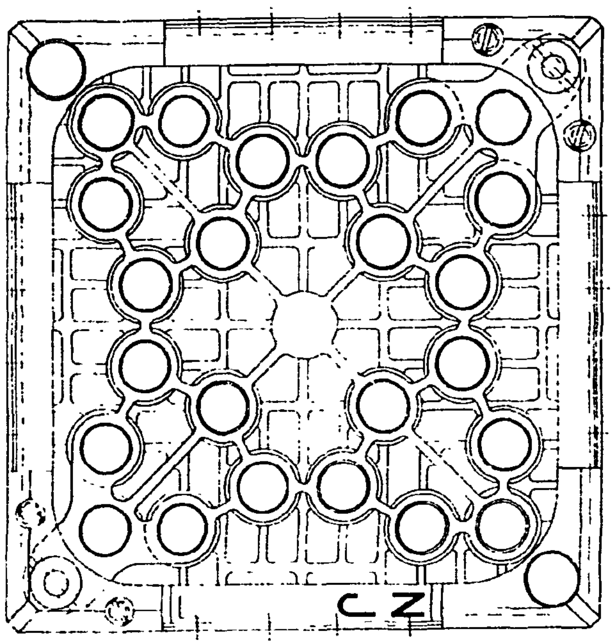
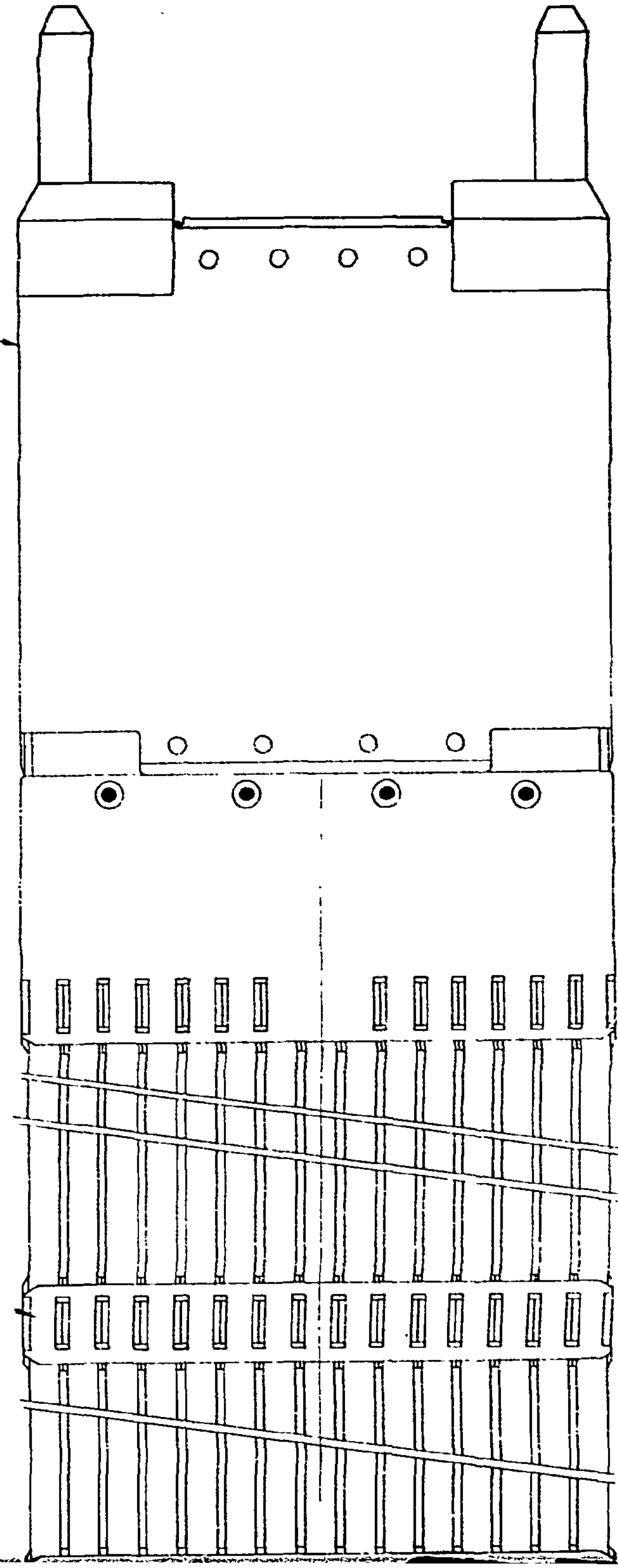


TOP VIEW



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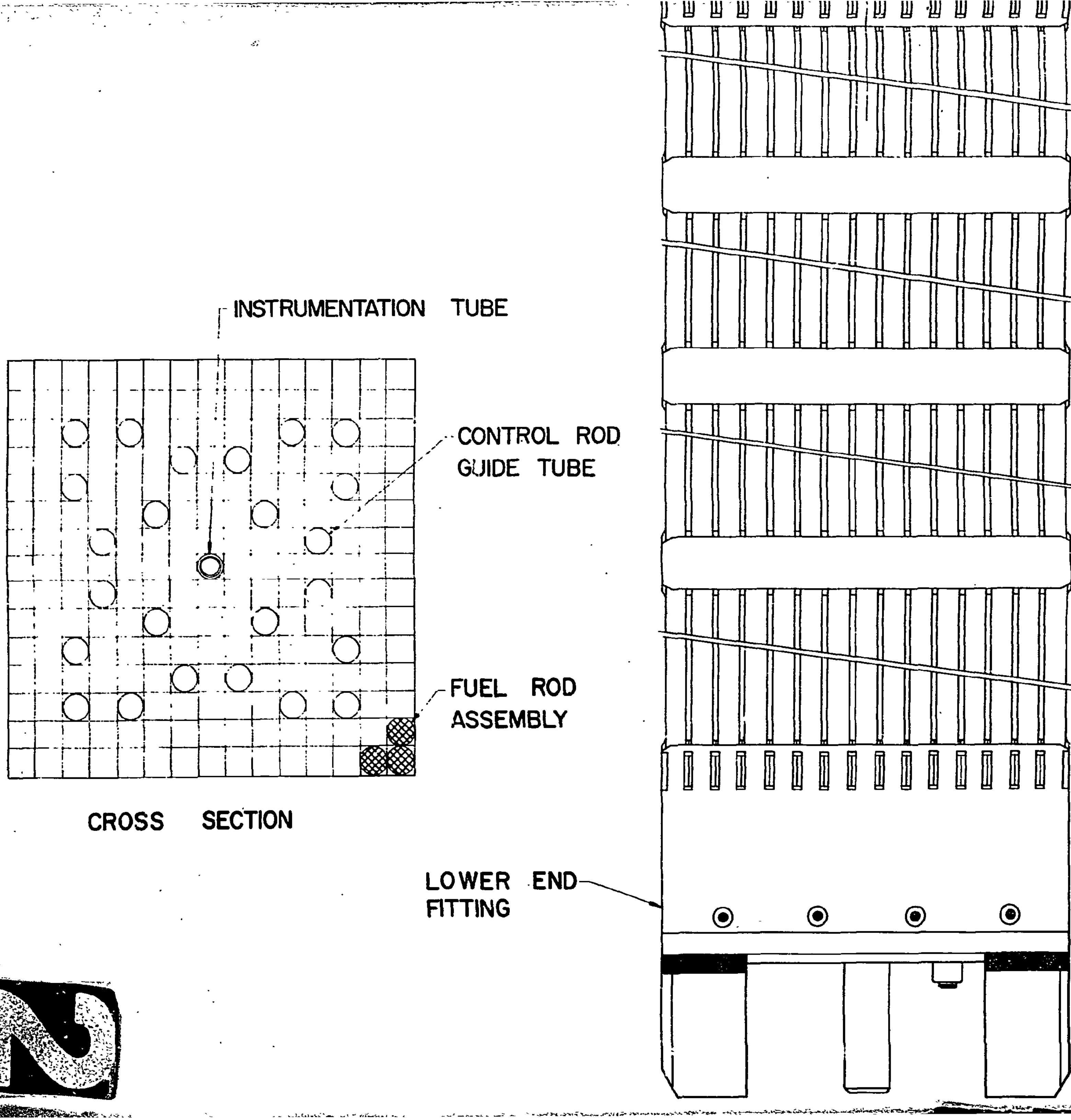

Figure A-4. Fuel Assembly

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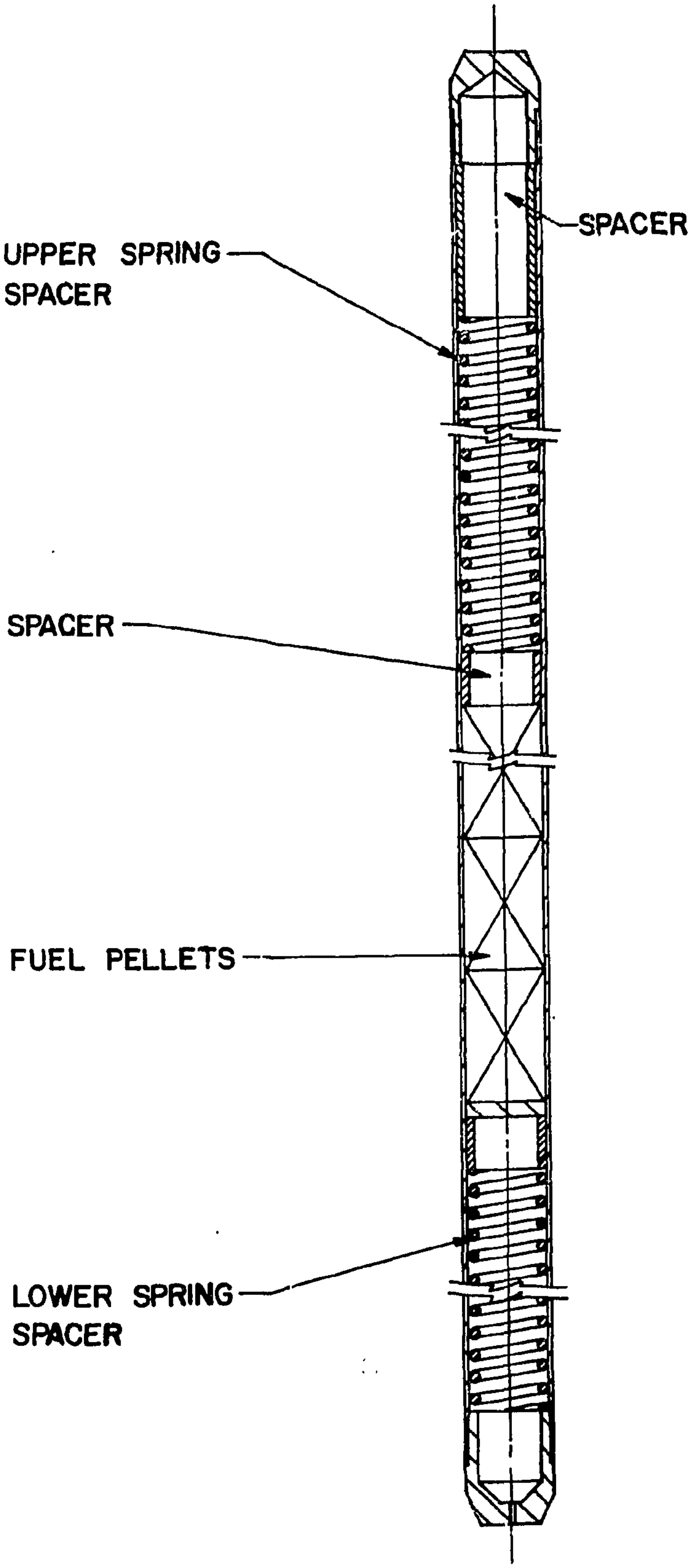
Figure A-5. Fuel Rod



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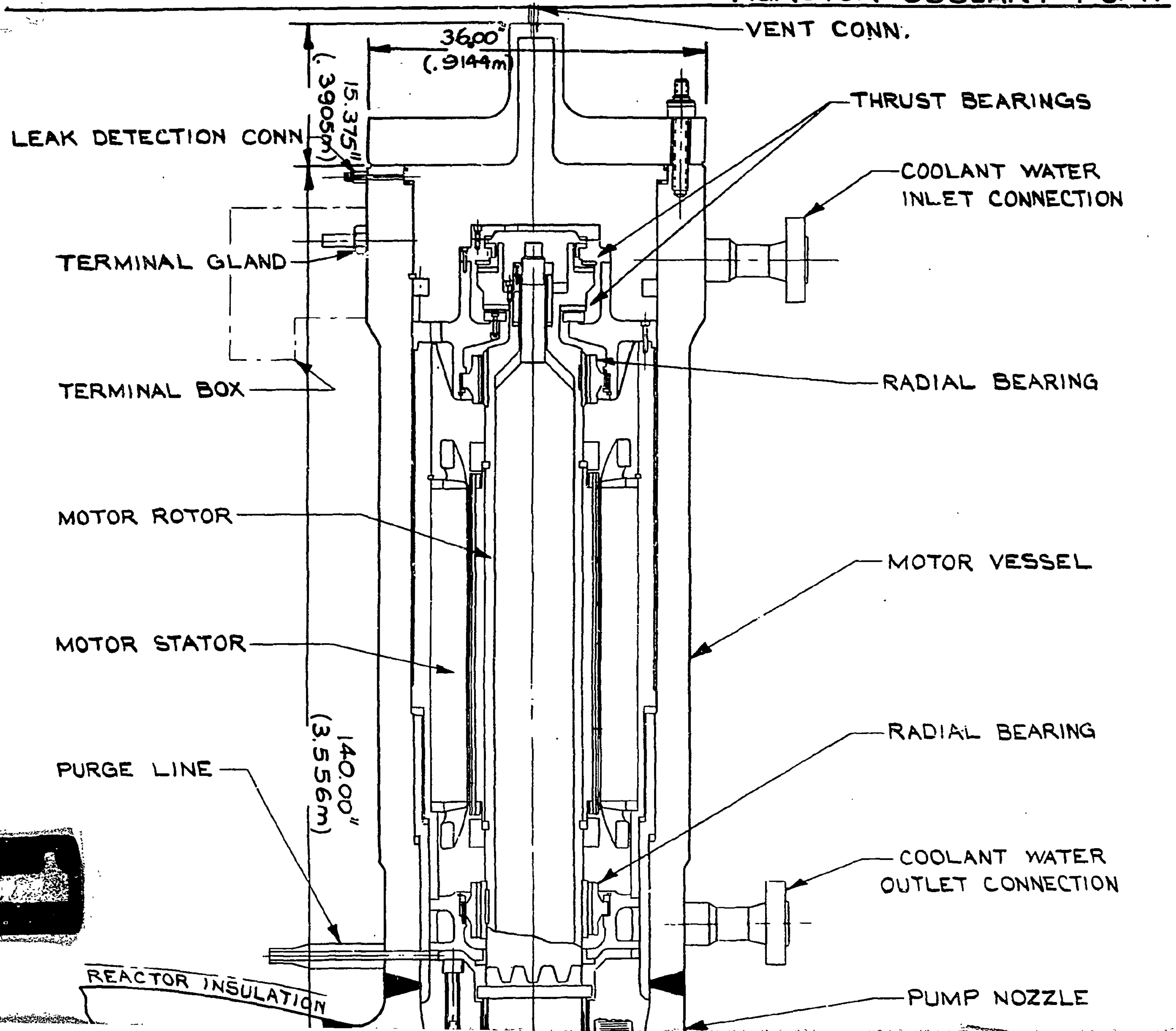
REACTOR COOLANT PUMP



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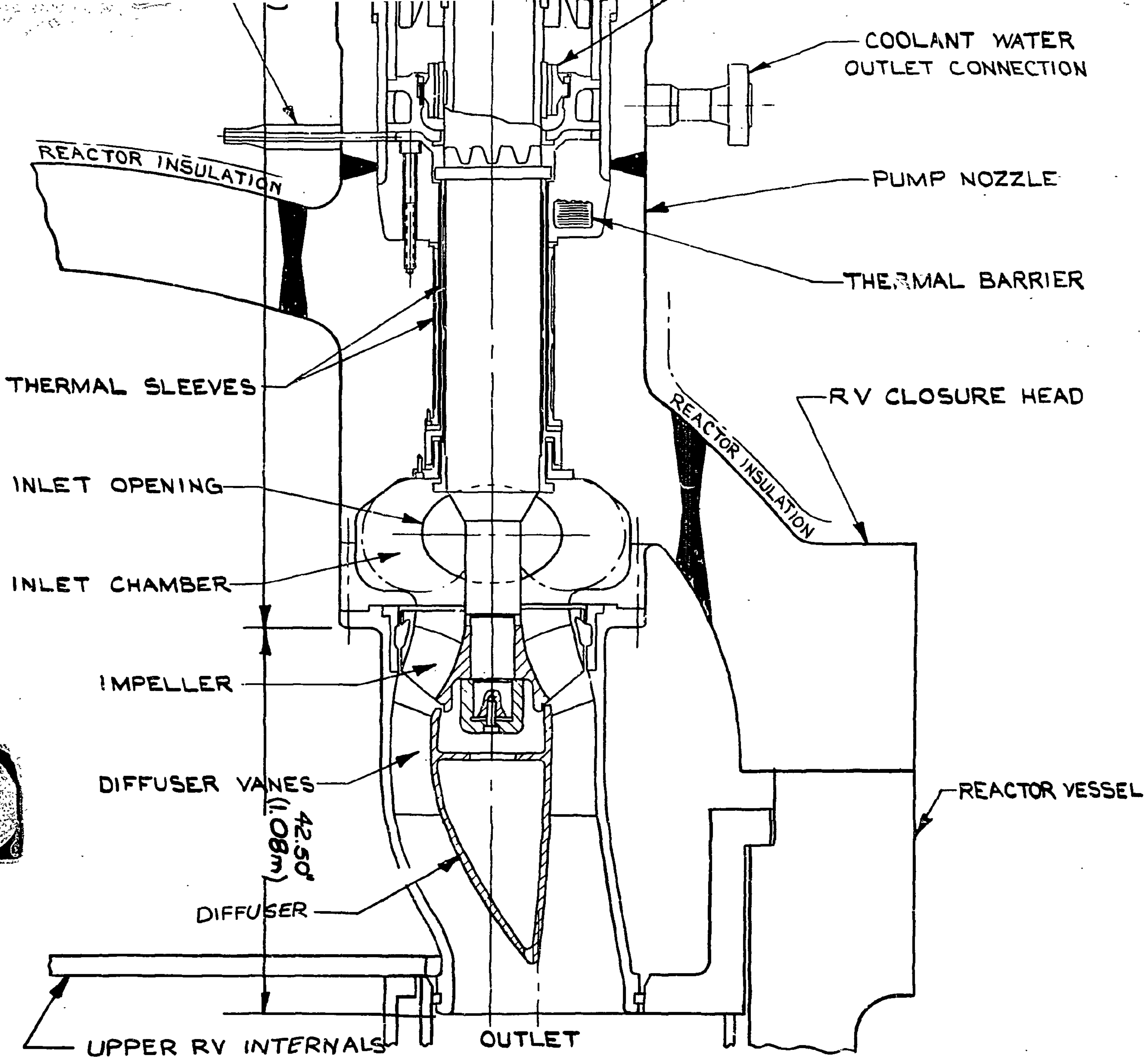

Figure A-6. Reactor Coolant Pumps

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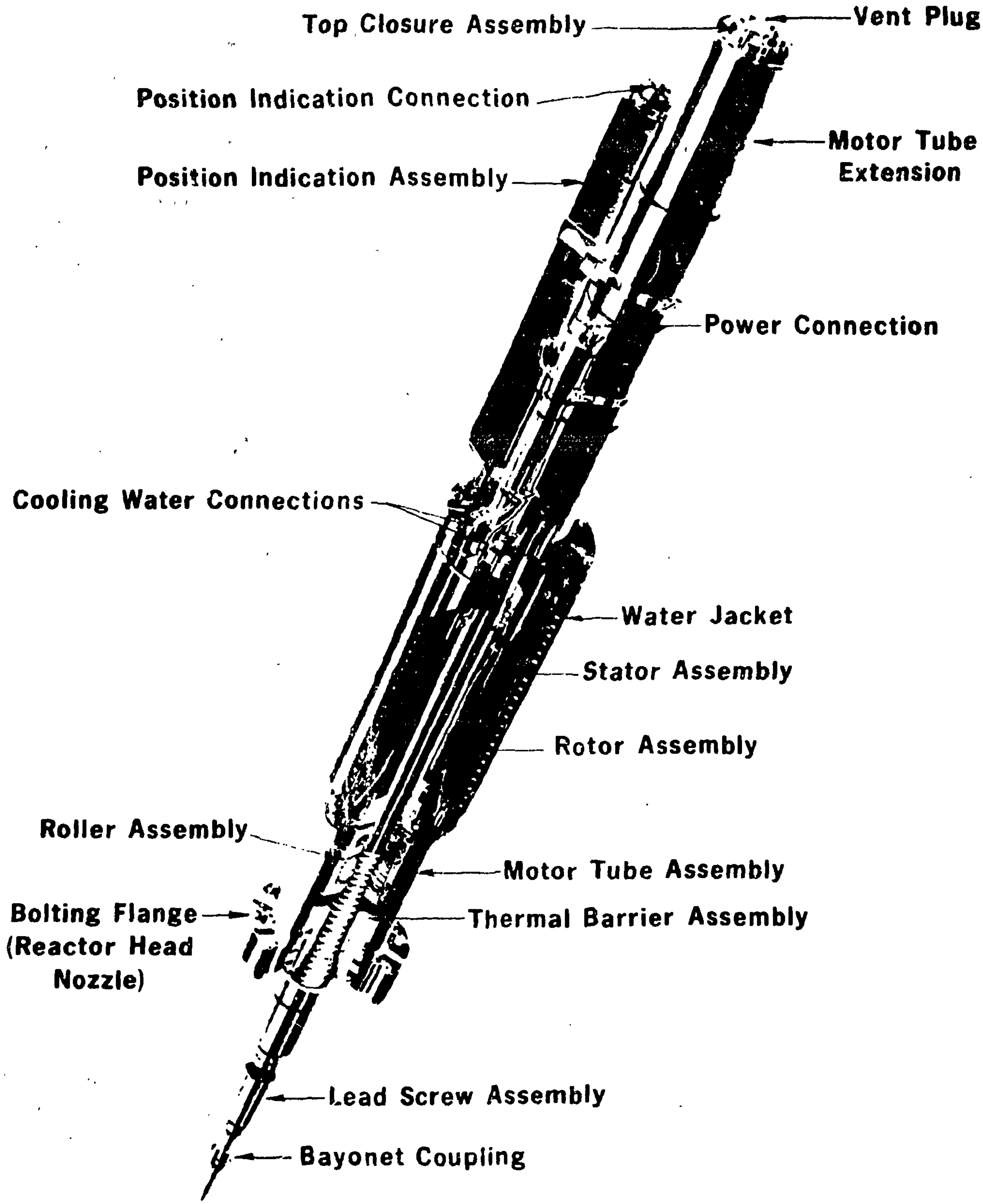
Figure A-7. Control Rod Drive Mechanism



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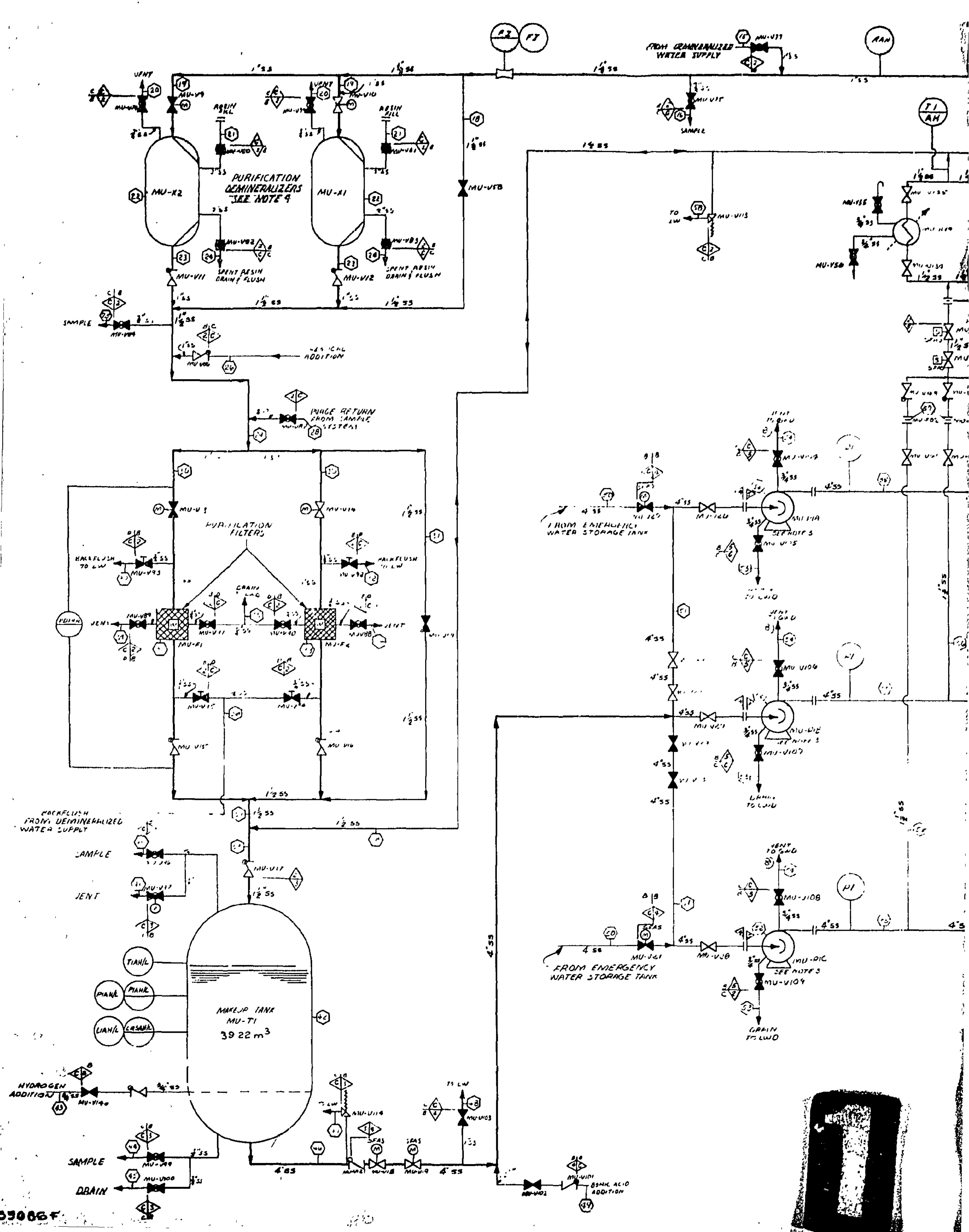
PURIFICATION DEMINERALIZERS SEE NOTE 9
MU-X2
MU-X1
RESIN FILL
SPENT RESIN DRAIN & FLUSH
FROM DEMINERALIZED WATER SUPPLY
SAMPLE
PURGE RETURN FROM SAMPLE SYSTEM
PURIFICATION FILTERS
MU-F1
MU-F2
BACKFLUSH TO LW
FROM EMERGENCY WATER STORAGE TANK
SEE NOTE 3
MU-P1A
MU-P1B
MU-P1C
DRAIN TO LWD
VENT TO GWD
BACKFLUSH FROM DEMINERALIZED WATER SUPPLY
SAMPLE
VENT
MAKEUP TANK
MU-T1
39 22 m³
TIAH/L
PIAH/L
PIAHR
LIAH/L
LASAHL
HYDROGEN ADDITION
SAMPLE
DRAIN
BORIC ACID ADDITION
TO LW
4" SS
4" SS
35086F
1

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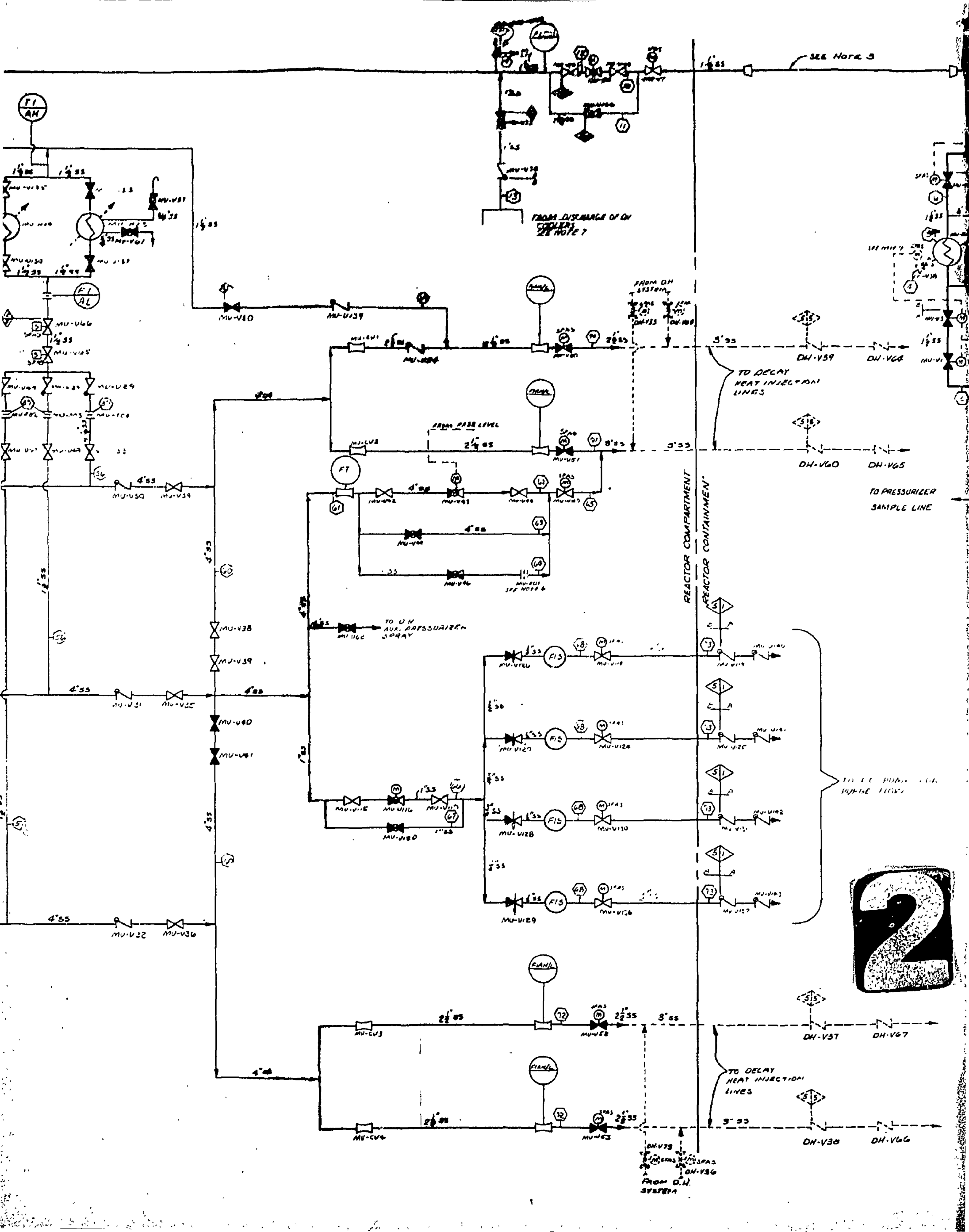

SEE NOTE 5
FROM DISCHARGE OF DH COOLERS SEE NOTE 7
TI AH
FI AL
MU-V66
MU-V139
FROM DH SYSTEM
FT
TO DECAY HEAT INJECTION LINES
DH-V59
DH-V64
DH-V60
DH-V65
TO PRESSURIZER SAMPLE LINE
REACTOR COMPARTMENT
REACTOR CONTAINMENT
MU-V50
MU-V34
MU-V38
MU-V39
MU-V31
MU-V40
MU-V41
MU-V32
MU-V36
TO D H AUX. PRESSURIZER SPRAY
FIS
MU-V128
MU-V129
MU-CV3
MU-CV4
DH-V57
DH-V67
DH-V66
FROM D.H. SYSTEM
2

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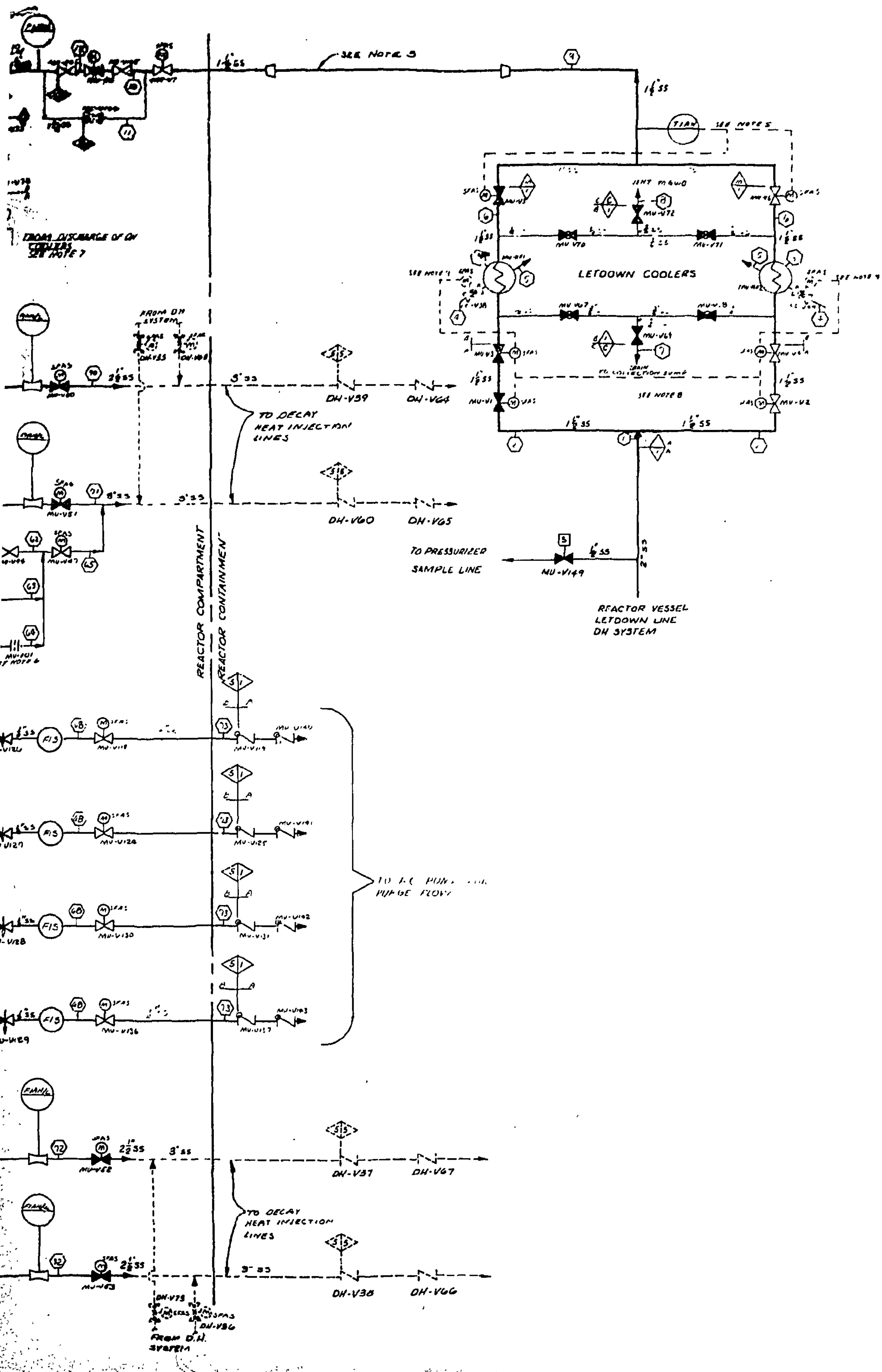

SEE NOTE 5
SEE NOTE 5
LETDOWN COOLERS
FROM DISCHARGE OF DH COOLERS SEE NOTE 7
FROM DH SYSTEM
TO DECAY HEAT INJECTION LINES
DH-V59
DH-V64
DH-V60
DH-V65
TO PRESSURIZER SAMPLE LINE
MU-V149
REACTOR VESSEL LETDOWN LINE DH SYSTEM
REACTOR COMPARTMENT
REACTOR CONTAINMENT
MU-V70
MU-V71
MU-V72
MU-V1
MU-V2
SEE NOTE 8
TO RC PUMP PURGE FLOW
MU-V130
MU-V136
DH-V37
DH-V67
TO DECAY HEAT INJECTION LINES
DH-V38
DH-V66
MU-V62
MU-V63
FROM D.H. SYSTEM

3

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INSTRUMENTATION IDENTIFICATION

AND - AND
OR - OR
I - COMPLEX INTERLOCK

} DIGITAL LOGIC

Y/Z — LOCALLY MOUNTED INSTRUMENT

X-Y/Z-N — REMOTE MOUNTED INSTRUMENT AT LOCATION X

Y/Z — INPUT TO PLANT COMPUTER

NOMENCLATURE FOR LOCATION (X), PROCESS VARIABLE AND FUNCTION (Y), UNIT NUMBER (N) AND STRING NUMBER (Z):

X -
- CA - CHEMICAL ADDITION PANEL (OUTSIDE CONTROL ROOM)
- A - AUXILIARY SHUTDOWN PANEL (OUTSIDE CONTROL ROOM)
- C - CONTROL ROOM
- R - INSTRUMENTATION CABINETS
- S - SAFETY FEATURES PANEL (CONTROL ROOM)
- SFAS - SAFETY FEATURES ACTUATION SYSTEM CABINETS
- WDA - WASTE DISPOSAL AUXILIARY PANEL (OUTSIDE CONTROL ROOM)
- RPS - REACTOR PROTECTION SYSTEM CABINETS

Y - (SEE "INSTRUMENTATION NOMENCLATURE" ON THIS SHEET FOR PROCESS VARIABLE AND FUNCTION)

N - UNIT NUMBERS MAY BE USED WHEN MORE THAN ONE REACTOR PLANT IS CONSTRUCTED AT A SINGLE LOCATION. A SHARED COMPONENT WILL HAVE THE UNIT NUMBERS OF SHARING UNITS. ASTERISK HERE DENOTES CUSTOMER SCOPE OF SUPPLY)

Z - ALL INSTRUMENTATION IS DESIGNATED BY A STRING NUMBER WHICH INCLUDES THE FOLLOWING:

1. THE FIRST NUMBER INDICATES THE PRIMARY STRING NUMBER
2. THE PRIMARY STRING NUMBER WILL BE FOLLOWED BY A LOOP LETTER (A,B, ETC.) WHERE APPLICABLE.
3. IN THE EVENT AN INSTRUMENT IS DUPLICATED ON THE SAME PRIMARY STRING, THE LOOP IS FOLLOWED BY A NUMERICAL NUMBER TO DIFFERENTIATE BETWEEN THE DUPLICATE INSTRUMENTS. WHERE A LOOP LETTER IS NOT APPLICABLE, A HYPHEN WILL PRECEED THIS NUMBER.

EXAMPLE (RC SYSTEM PRESSURIZER HEATER CONTROL SWITCH):

C HIS 2-6

C INDICATES THE INSTRUMENT IS LOCATED ON THE CONTROL ROOM CONSOLE

HIS DESCRIBES THE INSTRUMENT AS A HAND OPERATED SWITCH WITH INDICATING LIGHTS.

2-6 INDICATES THAT THE INSTRUMENT IS ON PRIMARY STRING 2 AND IS THE SIXTH-OF-A-KIND ON THAT STRING

WHEN IDENTIFYING A SPECIFIC INSTRUMENT IN WRITING, THE FOLLOWING ORDER SHOULD BE FOLLOWED:

1. UNIT NUMBER (WHERE APPLICABLE) AND SYSTEM ABREVIATION FOLLOWED BY A HYPHEN.
2. PROCESS VARIABLE AND FUNCTION.
3. INSTRUMENT STRING NUMBER.

EXAMPLE USING THE ABOVE INSTRUMENT: RC-HIS2-6

INSTRUMENTATION NOMENCLATURE

SUCCEEDING LETTER(S) FOR INSTRUMENT FUNCTION(S) (1) / FIRST LETTER FOR PROCESS VARIABLE		DETECTING					CONTROLLING						DISPLAY						
	FUNCTION SYM	T	G	P	E	W	C	S ( ) (4)	D	Y (6)			I	R	J	A ( ) (4)	L		
SYM	MEAS VARIABLE	TRANSMITTER	LOCAL OBSERVATION	TEST CONNECTION	PRIMARY ELEMENT	WELL	CONTROLLER	SWITCH	DELAY	COMPUTATION OR CONVERTER			INDICATING	RECORDING	SCANNING MULTIPOINT	ALARM	LIGHT		
(2) A	ANAL O_2, H_2, PH ETC																		
C	CONDUCTIVITY																		
D	DENSITY																		
F	FLOW	FT	FG		FE		FC	FS ( )		FY			FI	FR		FA ( )			
H	HAND (MANUAL)						HC	HS					HI( )						
K	TIME								KD										
L	LEVEL	LT	LG				LC	LS ( )					LI	LR		LA ( )			
M	MOISTURE HUMIDITY																		
O	TORQUE																		
P	PRESSURE	PT		PP			PC	PS ( )					PI	PR		PA ( )			
Pd	DIFF PRESSURE	PdT						PdS					PdI			PdA( )			
Q	QUANTITY																		
(3) R	RADIATION - γ ETC																		
S	SPEED FREQUENCY																		
T	TEMPERATURE	TT		TP	TE	TW	TC	TS ( )		TY			TI	TR		TA ( )			
Td	DIFF TEMP							TdS( )		TdY			TdI			TdA( )			
(5) X	UNCLASSIFIED																		
Z	POSITION	ZT						ZS					ZI			ZA( )			

NOTES

(1) MORE THAN ONE FUNCTION LETTER MAY SUCCEED THE MEASURED VARIABLE LETTER WHEN THE DEVICE PERFORMS MORE THAN ONE FUNCTION.
EXAMPLES. PIT - PRESSURE INDICATING TRANSMITTER
LRC - LEVEL RECORDING CONTROLLER

(2) THE CHEMICAL FORMULA OR PROCESS WILL BE WRITTEN ON THE P&I DWG NEAR THE PROCESS INSTRUMENT.

(3) THE TYPE OF RADIATION TO BE DETECTED WILL BE WRITTEN ON THE P&ID.

(4) PARENTHESES INDICATE THAT THE FOLLOWING MAY BE ADDED FOR CLARIFICATION OF THE SWITCH OR ALARM FUNCTION:
LL - LOW LOW
L - LOW
HL - HIGH LOW
HH - HIGH HIGH
H - HIGH
EXAMPLE. LAHH - LEVEL ALARM HIGH HIGH

(5) "X" IS USED TO REPRESENT ANY SPECIAL VARIABLES AND MAY BE DEFINED AS REQUIRED.

(6) COMPUTATION OR CONVERTER DESIGNATIONS.

( ) Y $\boxed{X}$

X TYPE OF COMPUTATION OR CONVERSION
- T TEMPERATURE COMPENSATION
- Δ - DIFFERENCE
- Σ - SUMMER
- $\sqrt[N]{\ }$ - ROOT EXTRACTOR, N = 3,4,5 ETC.
- $\int$ - INTEGRATE
- AVG - AVERAGE
- E/P - VOLTAGE TO PNEUMATIC
- I/P - CURRENT TO PNEUMATIC

1

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Figure A-8b. Instrumentation, Piping Symbols and Designations

INSTRUMENTATION NOMENCLATURE

	CONTROLLING						DISPLAY						
R	C	S ( )	D	Y			I	R	J	A ( )	L		
	CONTROLLER	SWITCH (4)	DELAY	COMPUTATION OR CONVERTER (5)			INDICATING	RECORDING	SCANNING, MULTIPOINT	ALARM (4)	LIGHT		
	FC	FS ( )		FY			FI	FR		FA ( )			
	HC	HS					HI( )						
			AD										
	LC	LS ( )					LI	LR		LA ( )			
	PC	PS ( )					PI	PR		PA ( )			
		PdS					PdI			PdA( )			
TW	TC	TS ( )		TY			TI	TR		TA ( )			
		TdS( )		TdY			TdI			TdA( )			
		ZS		.			ZI			ZA( )			

D THE MEASURED VARIABLE
N ONE FUNCTION.
NSMITTER
LER
WRITTEN ON THE P&I DWG

L BE WRITTEN ON THE P&ID.
MAY BE ADDED FOR CLARIFICATION

RIABLES AND MAY BE DEFINED AS REQUIRED.

OR CONVERSION
ATION

3,4,5 ETC.

C
C

COMPONENT IDENTIFICATION

EACH COMPONENT IS DESIGNATED ON THE SYSTEM P&ID BY A SEQUENCE OF LETTERS AND NUMBERS AS FOLLOWS.

1. COMPONENT ABBREVIATION
2. SEQUENTIAL NUMBER DIFFERENTIATING BETWEEN COMPONENTS WHICH HAVE THE SAME ABBREVIATION
3. LETTER INDICATING LOOP (A,B, ETC.) WHERE APPLICABLE

EXAMPLE (MAKEUP FILTER):


F2A

F INDICATES THE COMPONENT IS A FILTER
2 INDICATES ITS THE SECOND FILTER IN THE SYSTEM
A INDICATES THE FLOW PATH

WHEN IDENTIFYING A SPECIFIC COMPONENT IN WRITING, THE FOLLOWING ORDER SHOULD BE FOLLOWED.

1. UNIT NUMBER (WHERE APPLICABLE) AND SYSTEM ABBREVIATION FOLLOWED BY A HYPHEN
2. COMPONENT ABBREVIATION
3. SEQUENCE NUMBER
4. LOOP LETTER (WHERE APPLICABLE)

EXAMPLE USING THE ABOVE COMPONENT : MU-F2A

SYSTEM ABBREVIATIONS

BC - REACTOR BUILDING COOLING
CA & BR - CHEMICAL ADDITION & BORON REC.
BS - REACTOR BUILDING SPRAY
CA - CHEMICAL ADDITION
CC - COMPONENT COOLING
CF - CORE FLOODING
DH - DECAY HEAT REMOVAL
HP - HIGH PRESSURE INJECTION
MU - MAKEUP AND PURIFICATION
RC - REACTOR COOLANT
RW - REACTOR WASTE
SA - SAMPLING
SF - SPENT FUEL COOLING
SP - SECONDARY PLANT
SW - SOLID WASTE
ICS - INTEGRATED CONTROL SYSTEM
NNI - NON-NUCLEAR INSTRUMENTATION
PCS - PLANT COMPUTER SYSTEM
RPS - REACTOR PROTECTION SYSTEM
SFAS - SAFETY FEATURES ACTUATION SYSTEM
LW - LIQUID WASTE
GW - GASEOUS WASTE
WE - WASTE EVAPORATOR
SPW - SECONDARY PLANT WASTE
CW - COOLING WATER
SC - STEAM GENERATOR CIRCULATION SYSTEM
ECI - ESSENTIAL CONTROLS AND INSTRUMENTATION
SRCI - SAFETY RELATED CONTROLS & INSTRUMENTATION

COMPONENT AND FUNCTIONAL ABBREVIATIONS

AB - AUXILIARY BUILDING
B - BLOWER OR FAN
CRD - CONTROL ROD DRIVE
DB - DRY BOX OR SAMPLE SINK
DW - DEMINERALIZED WATER
F - FILTER
FO - FLOW ORIFICE
FW - FEEDWATER
G - GAS ANALYZER
HX - COOLER OR HEAT EXCHANGER
M - MIXER (AGITATOR)
MV - MANUAL VALVE
NS - NITROGEN SUPPLY
P - PUMP
RB REACTOR BUILDING
RD RUPTURE DISC.
RV RELIEF VALVE
S STRAINER
SG STEAM GENERATOR
SP - SAMPLE REMOTE
SL - SAMPLE LOCAL
SH - SHIELDING WALLS
SV - SAFETY VALVE
T - TANK
Ta TEMPERATURE AVERAGE
Tc TEMPERATURE COLD
Th - TEMPERATURE HOT
ΔT - TEMPERATURE DIFFERENCE
V - VALVE
VH VENT HEADER
X - DEMINERALIZER
C - COMPRESSOR
D - DEGASIFIER
EV - EVAPORATOR

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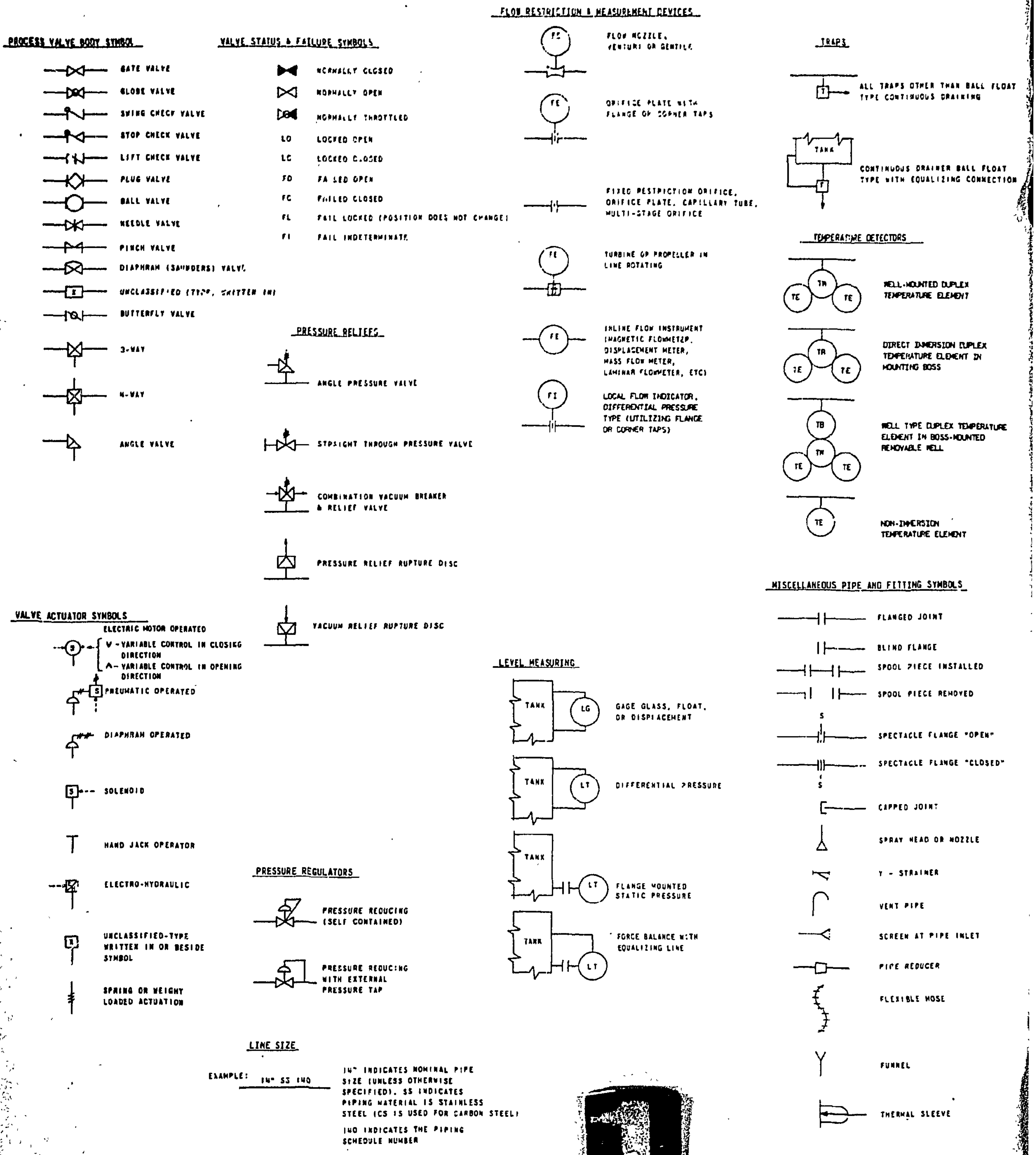
PROCESS VALVE BODY SYMBOL
GATE VALVE
GLOBE VALVE
SWING CHECK VALVE
STOP CHECK VALVE
LIFT CHECK VALVE
PLUG VALVE
BALL VALVE
NEEDLE VALVE
PINCH VALVE
DIAPHRAM (SAUNDERS) VALVE
UNCLASSIFIED (TYPE, WRITTEN IN)
BUTTERFLY VALVE
3-WAY
N-WAY
ANGLE VALVE
VALVE STATUS & FAILURE SYMBOLS
NORMALLY CLOSED
NORMALLY OPEN
NORMALLY THROTTLED
LO
LOCKED OPEN
LC
LOCKED CLOSED
FO
FAILED OPEN
FC
FAILED CLOSED
FL
FAIL LOCKED (POSITION DOES NOT CHANGE)
FI
FAIL INDETERMINATE
PRESSURE RELIEFS
ANGLE PRESSURE VALVE
STRAIGHT THROUGH PRESSURE VALVE
COMBINATION VACUUM BREAKER & RELIEF VALVE
PRESSURE RELIEF RUPTURE DISC
VACUUM RELIEF RUPTURE DISC
VALVE ACTUATOR SYMBOLS
ELECTRIC MOTOR OPERATED
V - VARIABLE CONTROL IN CLOSING DIRECTION
A - VARIABLE CONTROL IN OPENING DIRECTION
S
PNEUMATIC OPERATED
DIAPHRAM OPERATED
S
SOLENOID
HAND JACK OPERATOR
ELECTRO-HYDRAULIC
UNCLASSIFIED-TYPE WRITTEN IN OR BESIDE SYMBOL
SPRING OR WEIGHT LOADED ACTUATION
PRESSURE REGULATORS
PRESSURE REDUCING (SELF CONTAINED)
PRESSURE REDUCING WITH EXTERNAL PRESSURE TAP
LINE SIZE
EXAMPLE:
1N" SS 1NO
1N" INDICATES NOMINAL PIPE SIZE (UNLESS OTHERWISE SPECIFIED). SS INDICATES PIPING MATERIAL IS STAINLESS STEEL (CS IS USED FOR CARBON STEEL)
1NO INDICATES THE PIPING SCHEDULE NUMBER
FLOW RESTRICTION & MEASUREMENT DEVICES
FE
FLOW NOZZLE, VENTURI OR GENTILE
FE
ORIFICE PLATE WITH FLANGE OR CORNER TAPS
FIXED RESTRICTION ORIFICE, ORIFICE PLATE, CAPILLARY TUBE, MULTI-STAGE ORIFICE
FE
TURBINE OR PROPELLER IN LINE ROTATING
FE
INLINE FLOW INSTRUMENT (MAGNETIC FLOWMETER, DISPLACEMENT METER, MASS FLOW METER, LAMINAR FLOWMETER, ETC)
FI
LOCAL FLOW INDICATOR, DIFFERENTIAL PRESSURE TYPE (UTILIZING FLANGE OR CORNER TAPS)
LEVEL MEASURING
TANK
LG
GAGE GLASS, FLOAT, OR DISPLACEMENT
TANK
LT
DIFFERENTIAL PRESSURE
TANK
LT
FLANGE MOUNTED STATIC PRESSURE
TANK
LT
FORCE BALANCE WITH EQUALIZING LINE
TRAPS
T
ALL TRAPS OTHER THAN BALL FLOAT TYPE CONTINUOUS DRAINING
TANK
T
CONTINUOUS DRAINER BALL FLOAT TYPE WITH EQUALIZING CONNECTION
TEMPERATURE DETECTORS
TW
TE
TE
WELL-MOUNTED DUPLEX TEMPERATURE ELEMENT
TR
TE
TE
DIRECT IMMERSION DUPLEX TEMPERATURE ELEMENT IN MOUNTING BOSS
TB
TW
TE
TE
WELL TYPE DUPLEX TEMPERATURE ELEMENT IN BOSS-MOUNTED REMOVABLE WELL
TE
NON-IMMERSION TEMPERATURE ELEMENT
MISCELLANEOUS PIPE AND FITTING SYMBOLS
FLANGED JOINT
BLIND FLANGE
SPOOL PIECE INSTALLED
SPOOL PIECE REMOVED
S
SPECTACLE FLANGE "OPEN"
SPECTACLE FLANGE "CLOSED"
S
CAPPED JOINT
SPRAY HEAD OR NOZZLE
Y - STRAINER
VENT PIPE
SCREEN AT PIPE INLET
PIPE REDUCER
FLEXIBLE HOSE
FUNNEL
THERMAL SLEEVE

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Figure A-8c. Instrumentation, Piping Symbols and Designations

TRAPS

- ALL TRAPS OTHER THAN BALL FLOAT TYPE CONTINUOUS DRAINING
- CONTINUOUS DRAINER BALL FLOAT TYPE WITH EQUALIZING CONNECTION (TANK)

TEMPERATURE DETECTORS

- WELL-MOUNTED DUPLEX TEMPERATURE ELEMENT (TE, TW, TE)
- DIRECT IMMERSION DUPLEX TEMPERATURE ELEMENT IN MOUNTING BOSS (TE, TB, TE)
- WELL TYPE DUPLEX TEMPERATURE ELEMENT IN BOSS-MOUNTED REMOVABLE WELL (TB, TW, TE, TE)
- NON-IMMERSION TEMPERATURE ELEMENT (TE)

MISCELLANEOUS PIPE AND FITTING SYMBOLS

- FLANGED JOINT
- BLIND FLANGE
- SPOOL PIECE INSTALLED
- SPOOL PIECE REMOVED
- SPECTACLE FLANGE "OPEN"
- SPECTACLE FLANGE "CLOSED"
- CAPPED JOINT
- SPRAY HEAD OR NOZZLE
- Y - STRAINER
- VENT PIPE
- SCREEN AT PIPE INLET
- PIPE REDUCER
- FLEXIBLE HOSE
- FUNNEL
- THERMAL SLEEVE

PROCESS EQUIPMENT SYMBOLS

- DEMINERALIZER (DIAGONAL LINES INDICATE INLET AND OUTLET SCREENS)
- ADDITION OR MIX TANK
- FILTER
  - W - WATER
  - G - GAS
  - O - OIL
- DUPLEX STRAINER
- AIR FILTER
  - P - PRE FILTER
  - A - ABSOLUTE FILTER
  - C - CHARCOAL FILTER
- CENTRIFUGAL PUMP
- POSITIVE DISPLACEMENT PUMP WITH INTEGRAL RELIEF VALVE
- VACUUM PUMP OR COMPRESSOR
- CENTRIFUGAL BLOWER OR FAN
- AXIAL BLOWER OR FAN
- CENTRIFUGAL SUMP PUMP
- HELICAL TUBE HEAT EXCHANGER
- STRAIGHT SHELL & TUBE HEAT EXCHANGER
- U-TUBE HEAT EXCHANGER
- COOLING COIL
- AGITATOR
- REACTOR COOLANT PUMP
- QUICK DISCONNECT CONNECTION
- BULLSEYE FLOW INDICATOR
- DIAPHRAGM COMPRESSOR
- IMMERSION HEATERS
- SAMPLING TANK

MISCELLANEOUS SYMBOLS

- INSTRUMENT PROCESS PIPING
- INSTRUMENT CAPILLARY TUBING
- INSTRUMENT AIR LINES
- INSTRUMENT ELECTRICAL LINES
- HEAT TRACED PIPE
- RB —— RB REACTOR BUILDING BOUNDARY
- SH —— SH SECONDARY SHIELDING BOUNDARY
- AB —— AB AUXILIARY BUILDING BOUNDARY

DESIGN CONDITIONS, QUALITY GROUP CLASSIFICATIONS & PROCESS INFORMATION NODES

X - NODE NUMBER REFERRING TO THE PROCESS FLOW INFORMATION SHEETS WHERE DESIGN, NORMAL, AND EMERGENCY FLOW PARAMETERS ARE GIVEN IN THE DESIGN REPORT.

Y - QUALITY GROUP CLASSIFICATIONS AS DEFINED BY AEC DIVISION 1 REGULATORY GUIDE 1.26.

Z - DESIGN FLAGS REFER TO THE FOLLOWING DESIGN PRESSURES AND TEMPERATURES USED THROUGHOUT ALL SYSTEMS TO INDICATE DESIGN CONDITIONS OF PIPING AND COMPONENTS.

ANSI 12.2	AEC SG 26
1	A
2	B
3	C
NNS	D

DESIGN CONDITIONS

1	3200 PSIG @ 300°F
2	3200 PSIG @ 200°F
3	2500 PSIG @ 670°F
4	2500 PSIG @ 650°F
5	2500 PSIG @ 600°F
6	2500 PSIG @ 350°F
7	2500 PSIG @ 300°F
8	2500 PSIG @ 200°F
9	(UNASSIGNED)
10	CUSTOMER FEED WATER DESIGN
11	1235 PSIG @ 630°F
12	(UNASSIGNED)
13	700 PSIG @ 500°F
14	700 PSIG @ 300°F
15	675 PSIG @ 350°F
16	675 PSIG @ 200°F
17	550 PSIG @ 200°F
18	350 PSIG @ 300°F
19	300 PSIG @ 300°F
20	200 PSIG @ 300°F
21	200 PSIG @ 200°F
22	150 PSIG @ 300°F
23	150 PSIG @ 200°F
24	100 PSIG @ 200°F
25	ATMOS @ 150°F
C	CUSTOMER SYSTEM DESIGN CONDITIONS
26	550 PSIG @ 450°F
27	100 PSIG @ 340°F
28	200 PSIG @ 350°F
29	ATMOS @ 200°F

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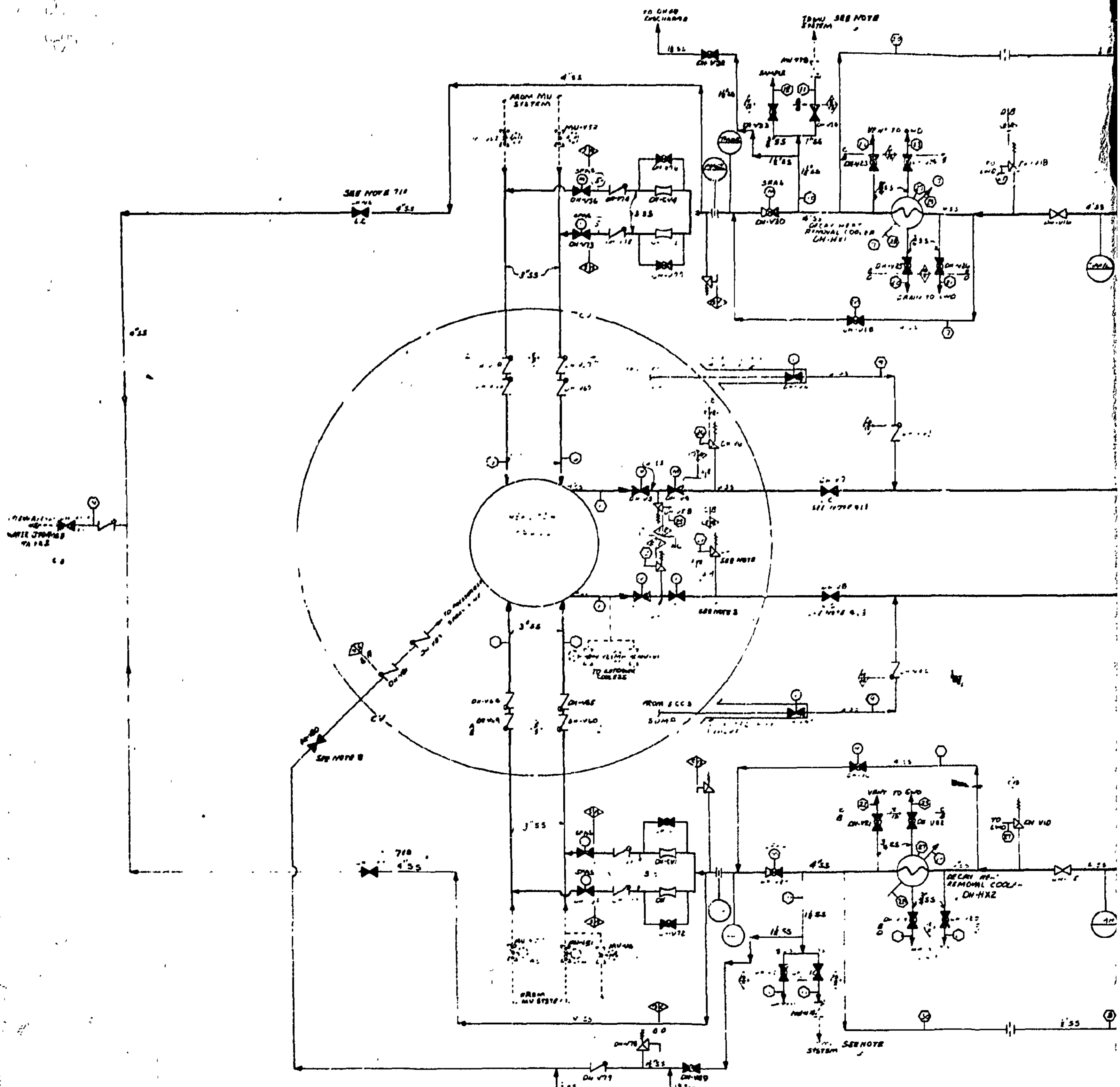

FROM MU SYSTEM
SAMPLE
SEAL
DH-V20
DECAY HEAT REMOVAL COOLER DH-HX1
DRAIN TO LWD
4" SS
3" SS
SEE NOTE 8
FROM ECCS SUMP
TO LETDOWN COOLERS
VENT TO LWD
DECAY HEAT REMOVAL COOLER DH-HX2
FROM MU SYSTEM
SEE NOTE
DH-V77
FROM MU PUMP DISCHARGE
FROM DH-P1 DISCHARGE

39090F-O

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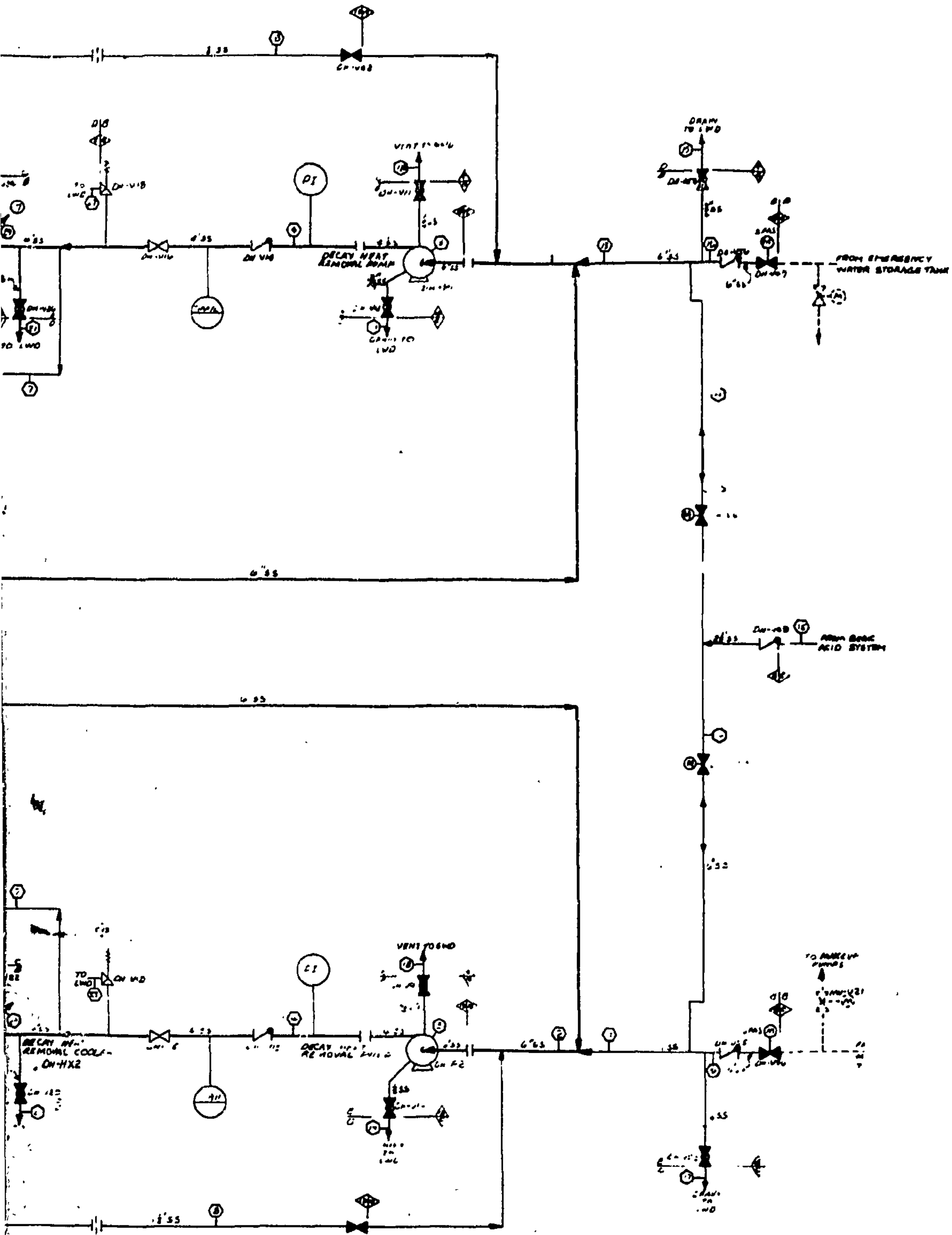
Figure A-9. Decay Heat Removal System



NOTES

1 THIS SYSTEM IS AN ASME III, CLASS 2, QUALITY GROUP CLASS DOWNSTREAM OF VALVES DH-V2 & DH-V4. ALL PIPING & VALVES UPSTREAM OF DH-V2 & DH-V4 AND DOWNSTREAM OF DH-V37, DH-V38, DH-V39 & DH-V60 ARE ASME III, CLASS I, QUALITY GROUP CLASS A

2 ALL RELIEF VALVES INSIDE CONTAINMENT SHALL DISCHARGE INTO (LATER)

3 VALVES DH-V1, DH-V2, DH-V3 & DH-V4 SHALL BE INTERLOCKED WITH R C SYSTEM PRESSURE INSTRUMENTATION TO CLOSE OR TO PREVENT THE OPENING OF THE VALVES WHEN THE RC SYSTEM PRESSURE IS ABOVE THE DH SYSTEM DESIGN PRESSURE.

4 VALVES DH-V7 & DH-V8 SHALL CONTAIN LIMIT SWITCHES TO PREVENT THE OPENING OF VALVES DH-V40 & DH-V41 WHEN VALVE DH-V7 OR DH-V8 RESPECTIVELY IS OPEN

5 LAST VALVE IS V83. LAST NODE IS 30.

6 REACTOR COOLANT WHICH IS SENT TO THE MU SYSTEM FOR PURIFICATION IS RETURNED TO THE REACTOR VESSEL VIA THE MAKEUP SYSTEM

7 THIS PORTION OF SYSTEM IS FOR TESTING ONLY.

8 THIS VALVE MUST BE LOCKED CLOSED DURING NORMAL REACTOR OPERATION

DESIGN CONDITIONS

1	17236 89 KPAG - 343 3°C
2	1378 95 KPAG - 79.4°C
4	689 48 KPAG - 121 1°C
5	17236 89 KPAG - 176.6°C
12	689 48 KPAG - 176.6°C
13	5515 81 KPAG - 176 6°C
C	TO BE DETERMINED BY CUSTOMER
14	723 95 KPAG - 176 6°C

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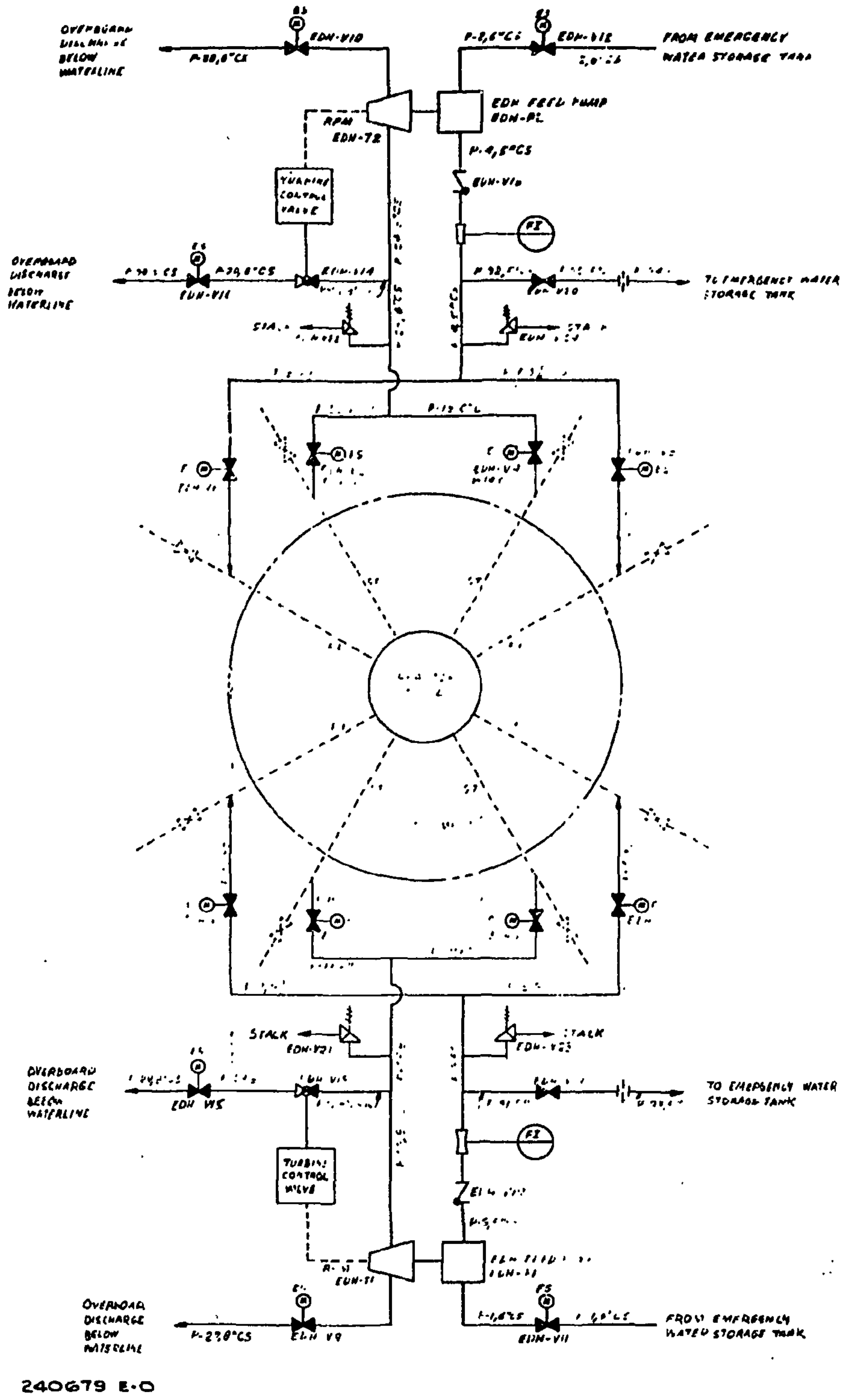
Figure A-10. Emergency Decay Heat System



NOTES:
1 SYSTEM IS ASME III CLASS 2, QUALITY GROUP CLASSIFICATION B
2 LAST VALVE IS V24.

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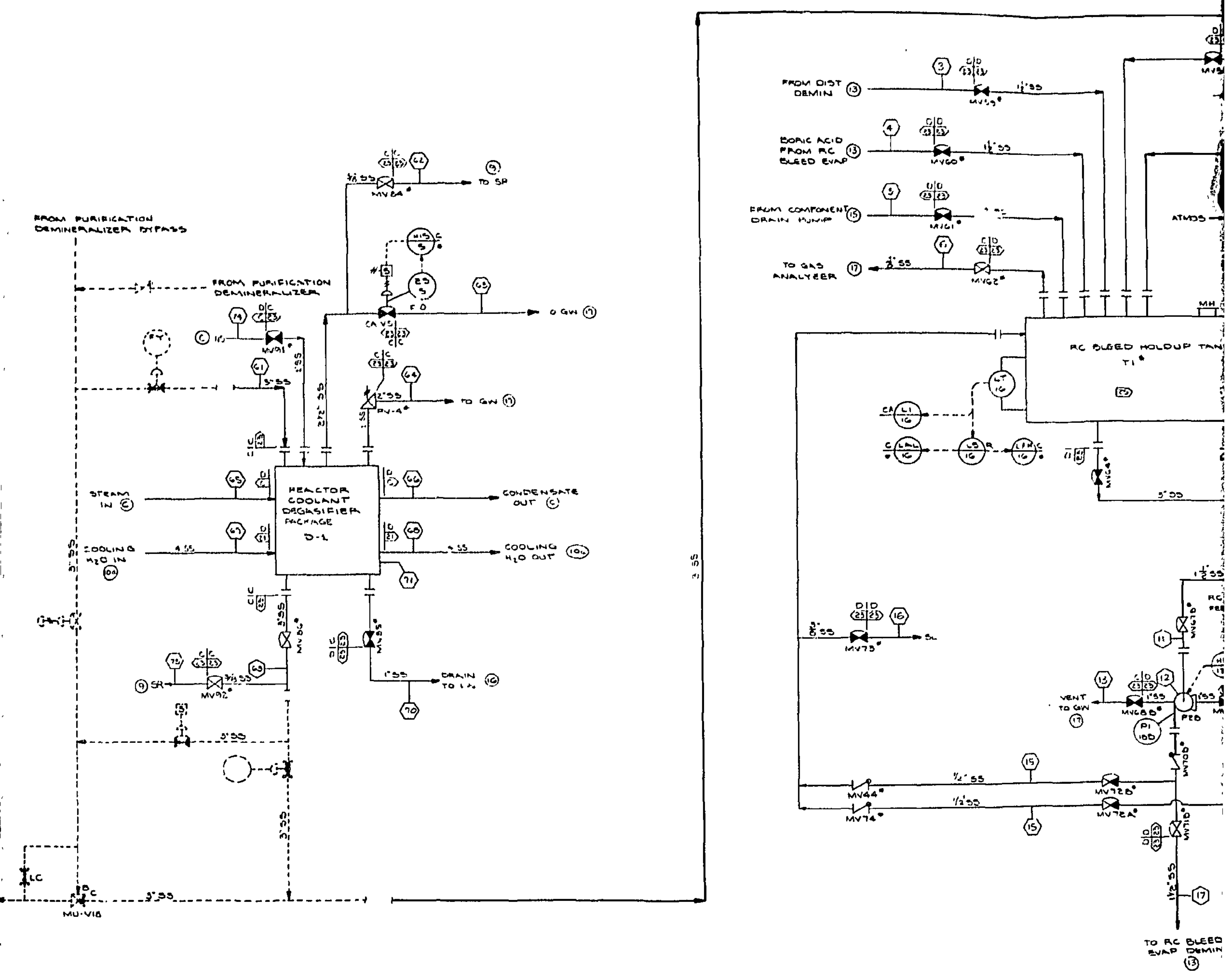

FROM PURIFICATION DEMINERALIZER BYPASS
FROM PURIFICATION DEMINERALIZER
REACTOR COOLANT DEGASIFIER PACKAGE D-1
STEAM IN
CONDENSATE OUT
COOLING H2O IN
COOLING H2O OUT
TO SR
TO GW
DRAIN TO LW
MV84
MV91
MV92
PV-4
MU-V18
FROM DIST DEMIN
BORIC ACID FROM RC BLEED EVAP
FROM COMPONENT DRAIN PUMP
TO GAS ANALYZER
RC BLEED HOLDUP TAN
MV59
MV60
MV61
MV62
MV73
MV44
MV74
VENT TO GW
TO RC BLEED EVAP DEMIN

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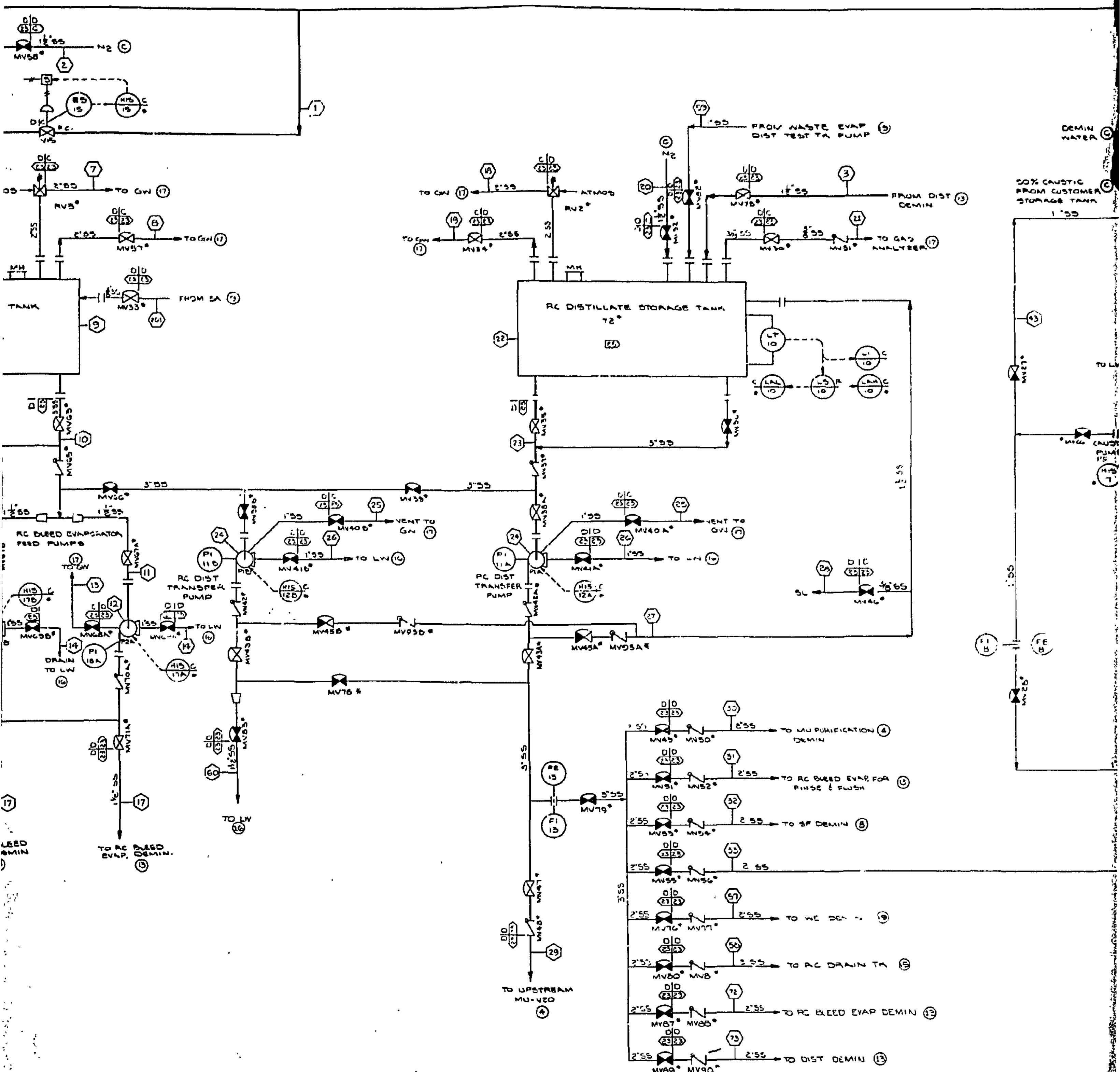

RC DISTILLATE STORAGE TANK
T2
TANK
RC BLEED EVAPORATOR FEED PUMPS
RC DIST TRANSFER PUMP
RC DIST TRANSFER PUMP
FROM WASTE EVAP DIST TEST TK PUMP
FROM DIST DEMIN
TO GAS ANALYZER
FROM SA
TO GW
ATMOS
VENT TO GW
TO LW
DRAIN TO LW
TO RC BLEED EVAP. DEMIN.
TO UPSTREAM MU-V20
TO MU PURIFICATION DEMIN
TO RC BLEED EVAP FOR RINSE & FLUSH
TO SF DEMIN
TO WE DEMIN
TO RC DRAIN TK
TO RC BLEED EVAP DEMIN
TO DIST DEMIN
DEMIN WATER
50% CAUSTIC FROM CUSTOMER STORAGE TANK
N2
MH
MV45
MV50
MV51
MV52
MV55
MV56
MV76
MV77
MV80
MV87
MV88
MV89
MV90
MV79
MV78
MV45B
MV93B
MV45A
MV93A
MV46
MV40B
MV40A
MV41B
MV41A
MV33
MV34
MV57
MV58
RV5
RV2
V15

2

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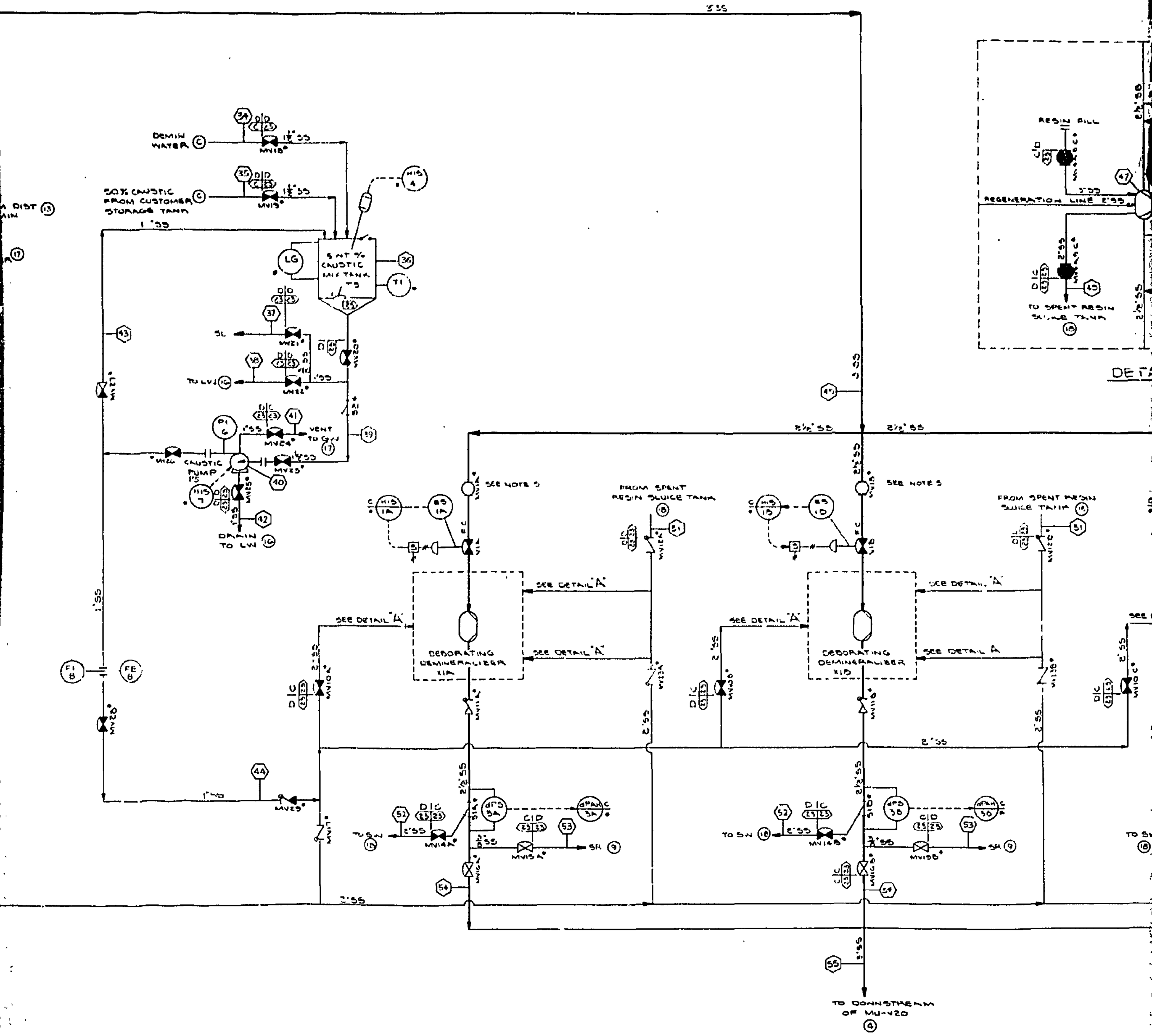
DEMIN WATER
50% CAUSTIC FROM CUSTOMER STORAGE TANK
CAUSTIC MIX TANK
CAUSTIC PUMP
VENT TO GW
DRAIN TO LW
FROM SPENT RESIN SLUICE TANK
SEE NOTE 5
SEE DETAIL "A"
DEBORATING DEMINERALIZER X1A
DEBORATING DEMINERALIZER X1B
TO DOWNSTREAM OF MU-V20
RESIN FILL
REGENERATION LINE
TO SPENT RESIN SLUICE TANK

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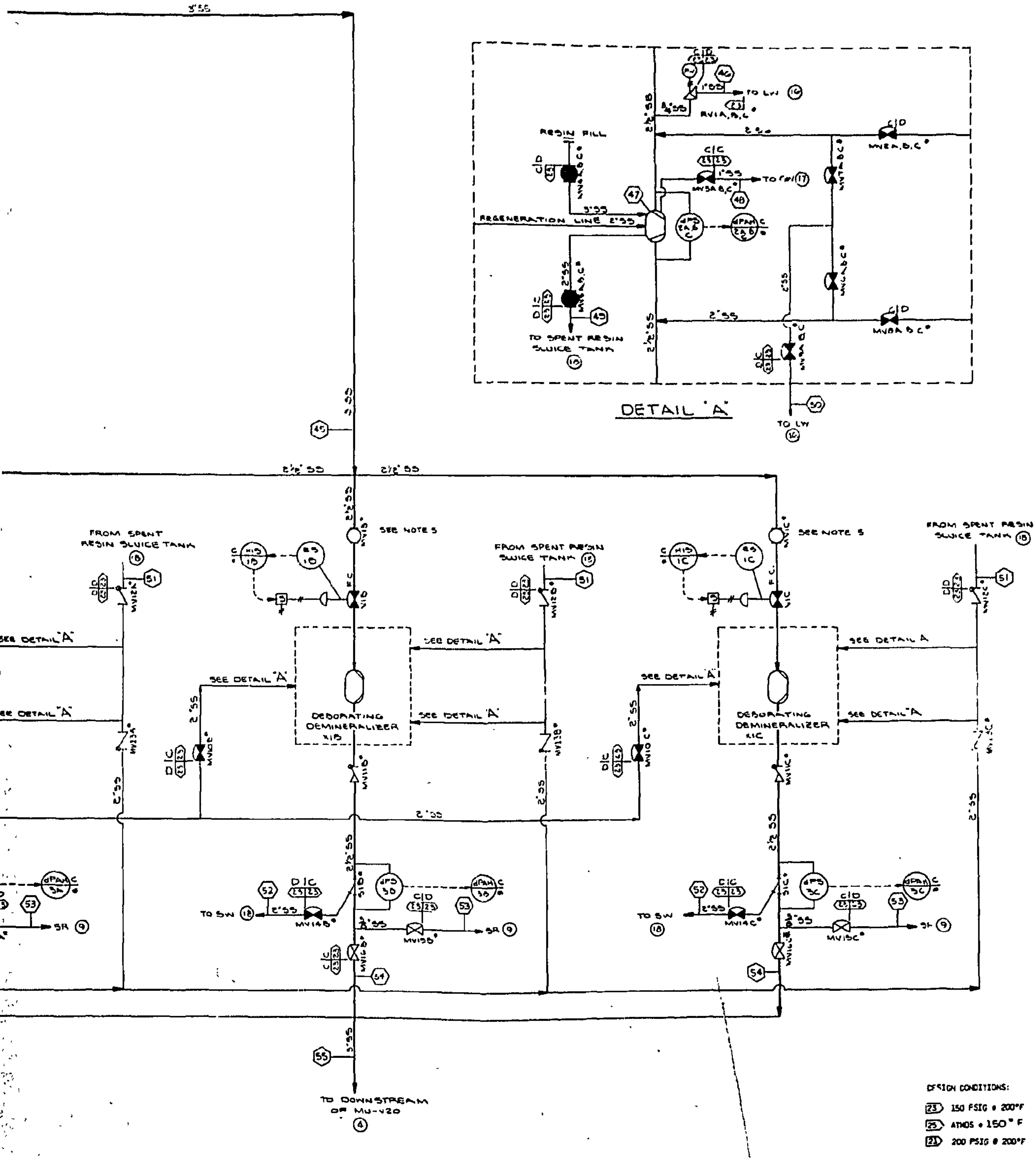
Figure A-11a. Chemical Addition and Boron Recovery System (Sheet 1 of 3)



4

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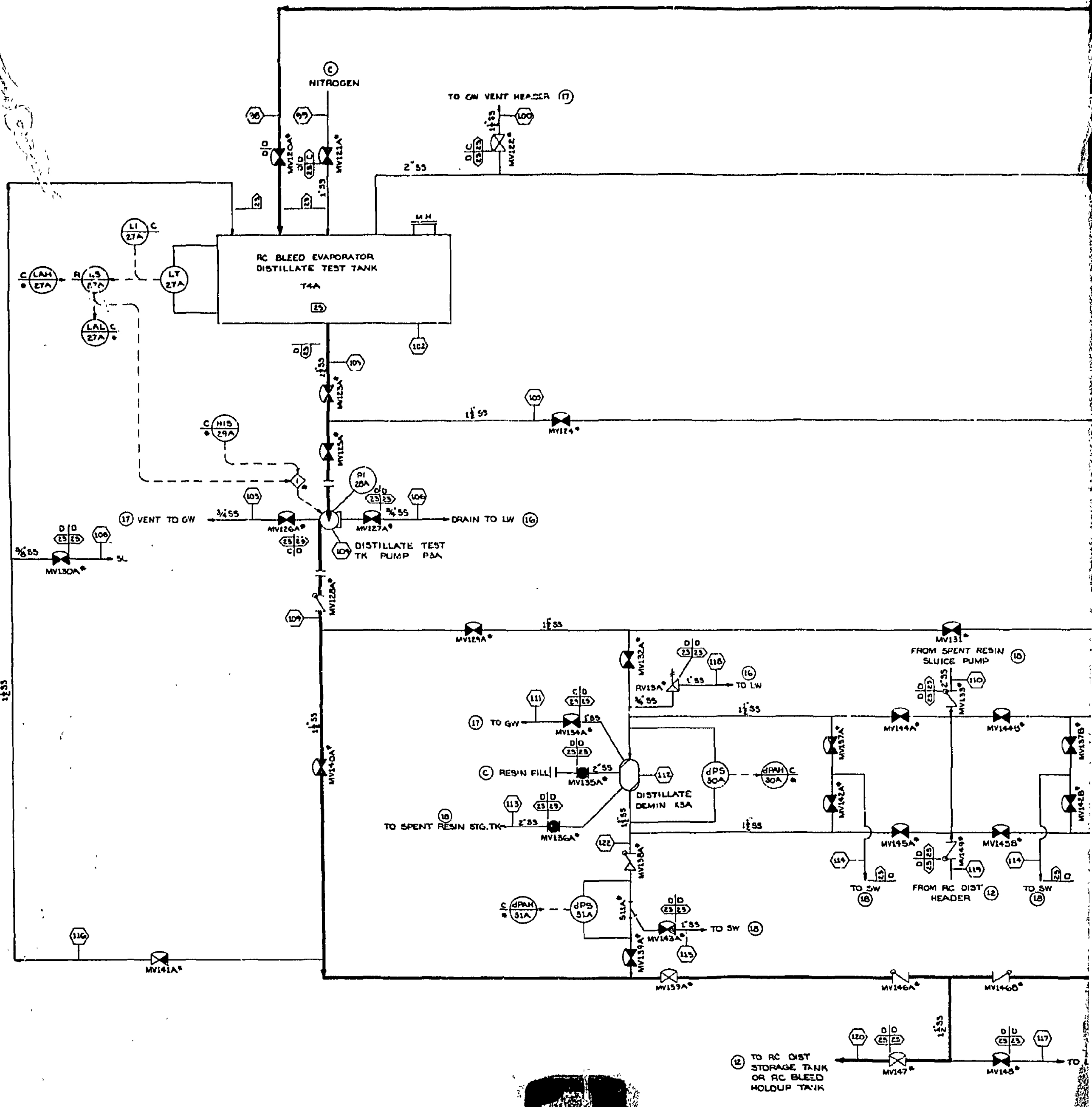
NITROGEN
TO GW VENT HEADER
RC BLEED EVAPORATOR
DISTILLATE TEST TANK
T4A
MH
LI 27A
LT 27A
LS 27A
LAH 27A
LAL 27A
HIS 29A
PI 28A
MV120A
MV121A
MV122
MV123A
MV124
MV125A
MV126A
MV127A
MV128A
MV129A
MV130A
MV131
MV132A
MV133
MV134A
MV135A
MV136A
MV137A
MV137B
MV138A
MV139A
MV140A
MV141A
MV142A
MV142B
MV143A
MV144A
MV144B
MV145A
MV145B
MV146A
MV146B
MV147
MV148
MV149
MV159A
RV13A
S11A
VENT TO GW
DRAIN TO LW
DISTILLATE TEST
TK PUMP P3A
TO LW
TO GW
RESIN FILL
TO SPENT RESIN STG.TK
DISTILLATE
DEMIN X3A
dPS 30A
dPAH 30A
dPS 31A
dPAH 31A
TO SW
FROM SPENT RESIN
SLUICE PUMP
FROM RC DIST
HEADER
TO RC DIST
STORAGE TANK
OR RC BLEED
HOLDUP TANK

1

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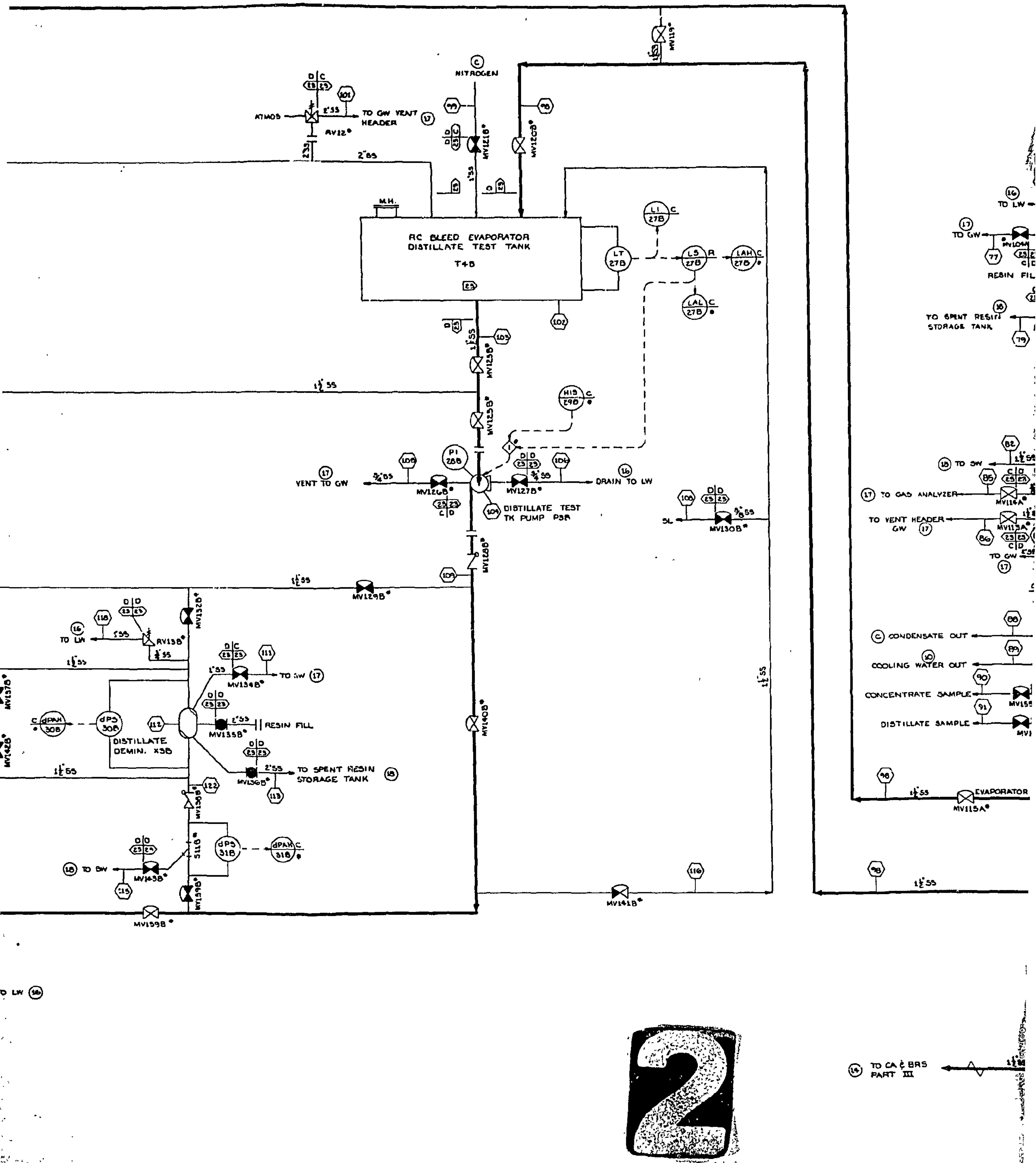
RC BLEED EVAPORATOR DISTILLATE TEST TANK
T4B
NITROGEN
M.H.
ATMOS
TO GW VENT HEADER
RV12
MV121B
MV120B
MV119
LI 27B
LT 27B
LS 27B
LAH 27B
LAL 27B
HIS 29B
PI 28B
MV125B
VENT TO GW
MV126B
MV127B
DRAIN TO LW
DISTILLATE TEST TK PUMP P5B
MV128B
MV130B
MV129B
MV132B
RV13B
TO LW
MV134B
TO GW
dPS 30B
dPAH 30B
DISTILLATE DEMIN. X5B
MV135B
RESIN FILL
MV136B
TO SPENT RESIN STORAGE TANK
MV140B
dPS 31B
dPAH 31B
MV143B
TO SW
MV159B
MV141B
TO LW
TO GW
RESIN FILL
TO SPENT RESIN STORAGE TANK
TO SW
TO GAS ANALYZER
MV114A
TO VENT HEADER GW
MV113A
CONDENSATE OUT
COOLING WATER OUT
CONCENTRATE SAMPLE
DISTILLATE SAMPLE
EVAPORATOR
MV115A
TO CA & BRS PART III
2

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Figure A-11b.

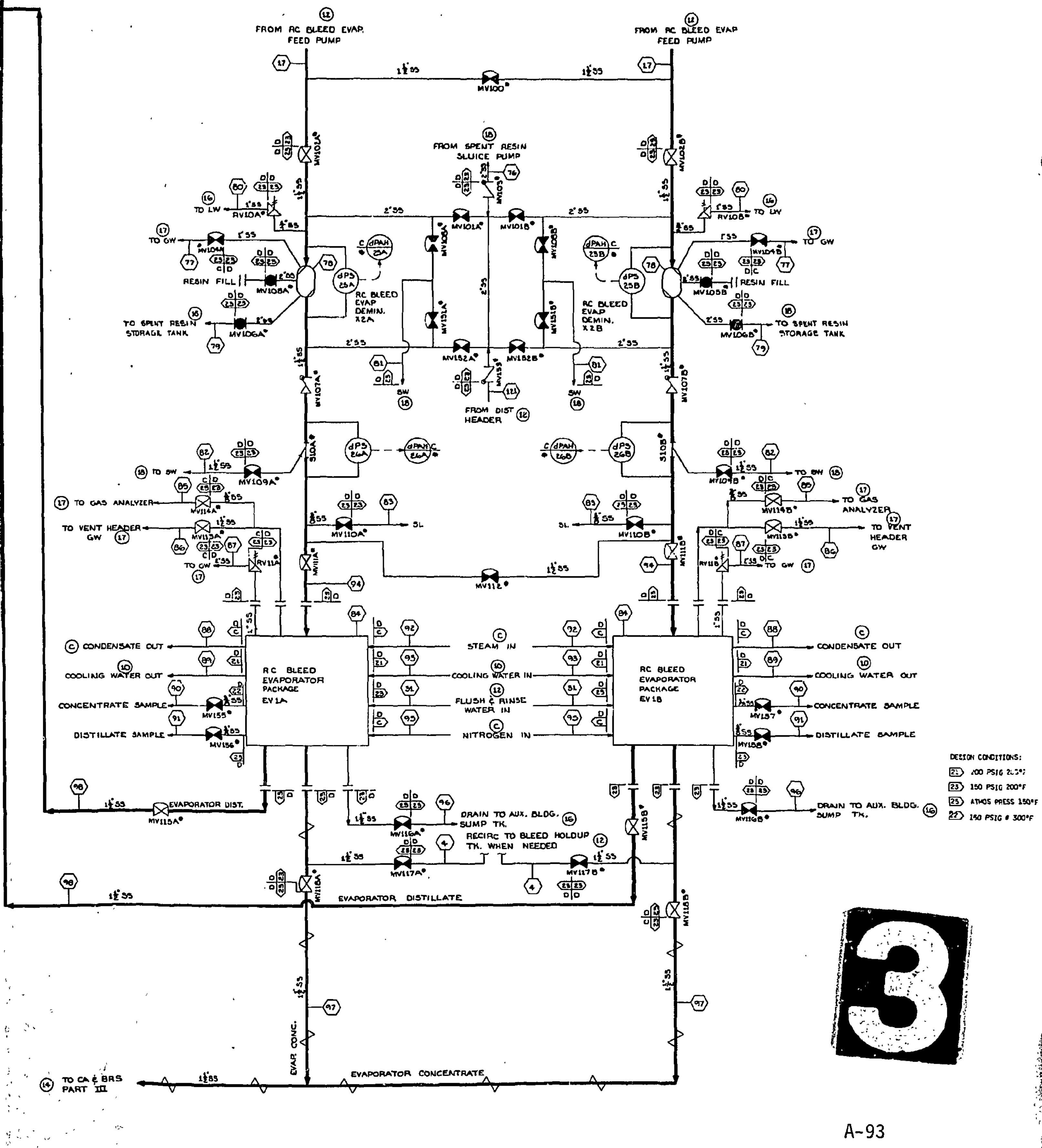

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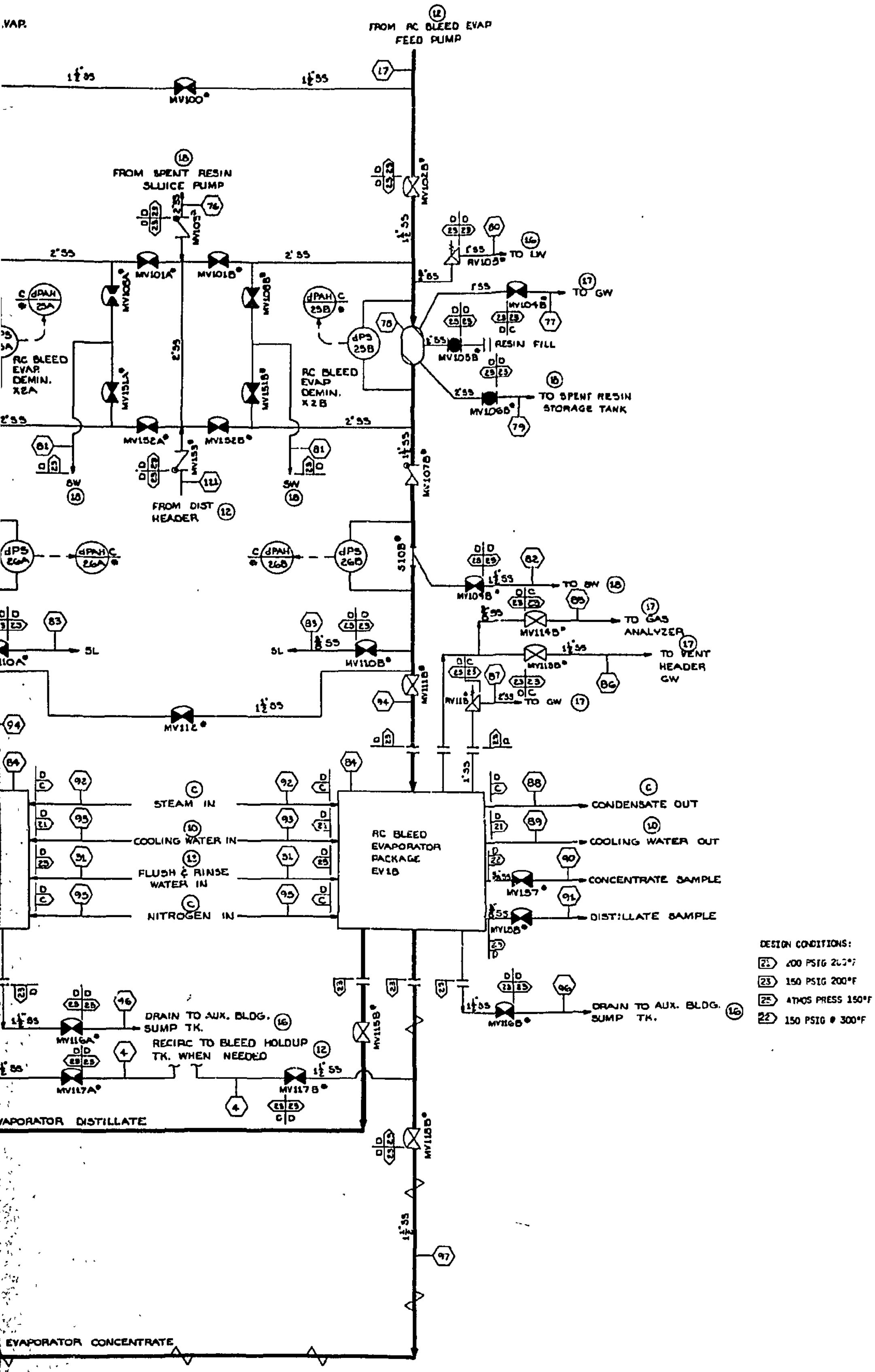
Figure A-11b. Chemical Addition and Boron Recovery System (Sheet 2 of 3)



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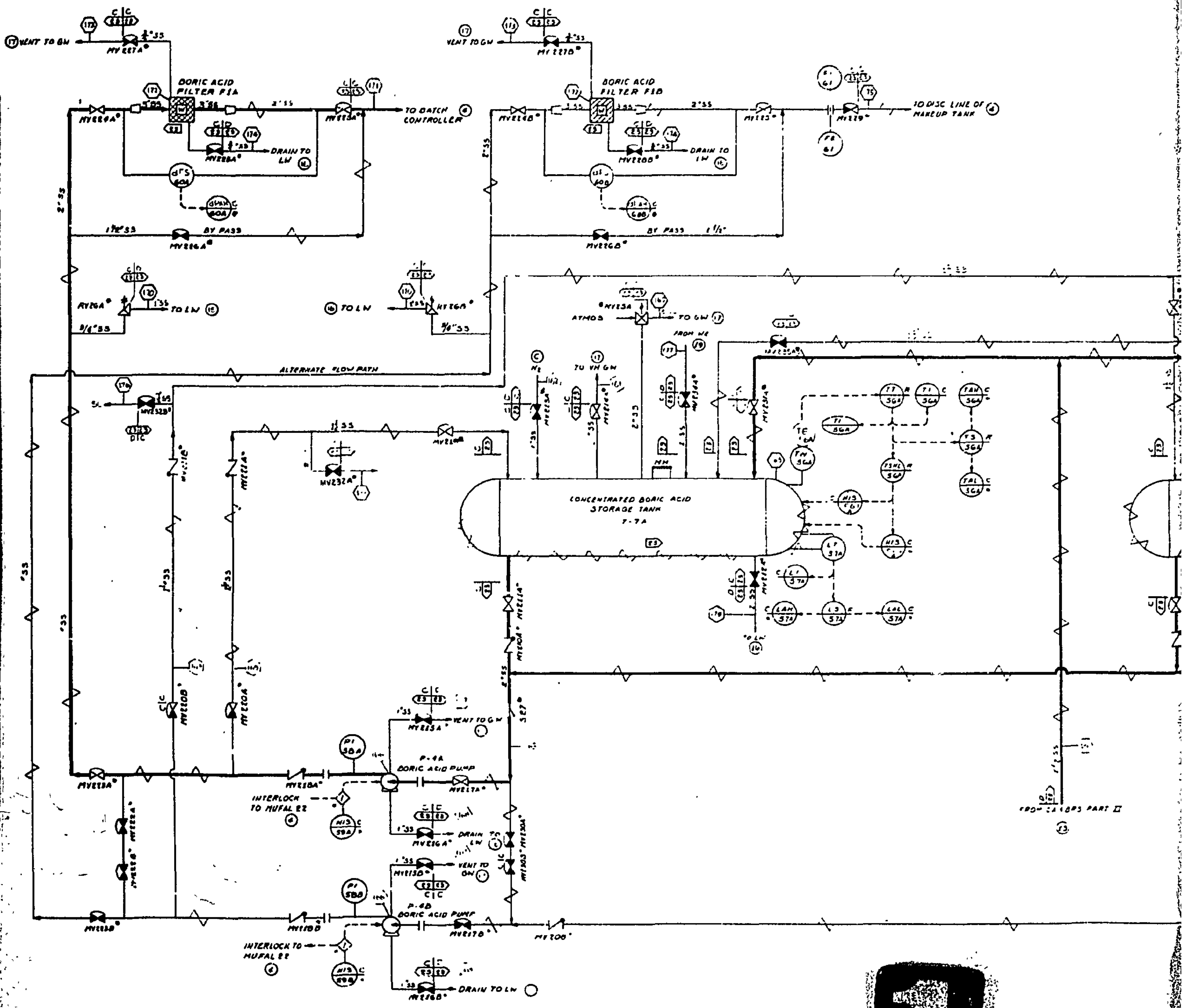
VENT TO GW
BORIC ACID FILTER F1A
TO BATCH CONTROLLER
DRAIN TO LW
BY PASS
BORIC ACID FILTER F1B
TO DISC LINE OF MAKEUP TANK
ALTERNATE FLOW PATH
ATMOS
CONCENTRATED BORIC ACID STORAGE TANK
T-7A
P-4A
BORIC ACID PUMP
P-4B
BORIC ACID PUMP
INTERLOCK TO MUFAL 22
DRAIN TO LW

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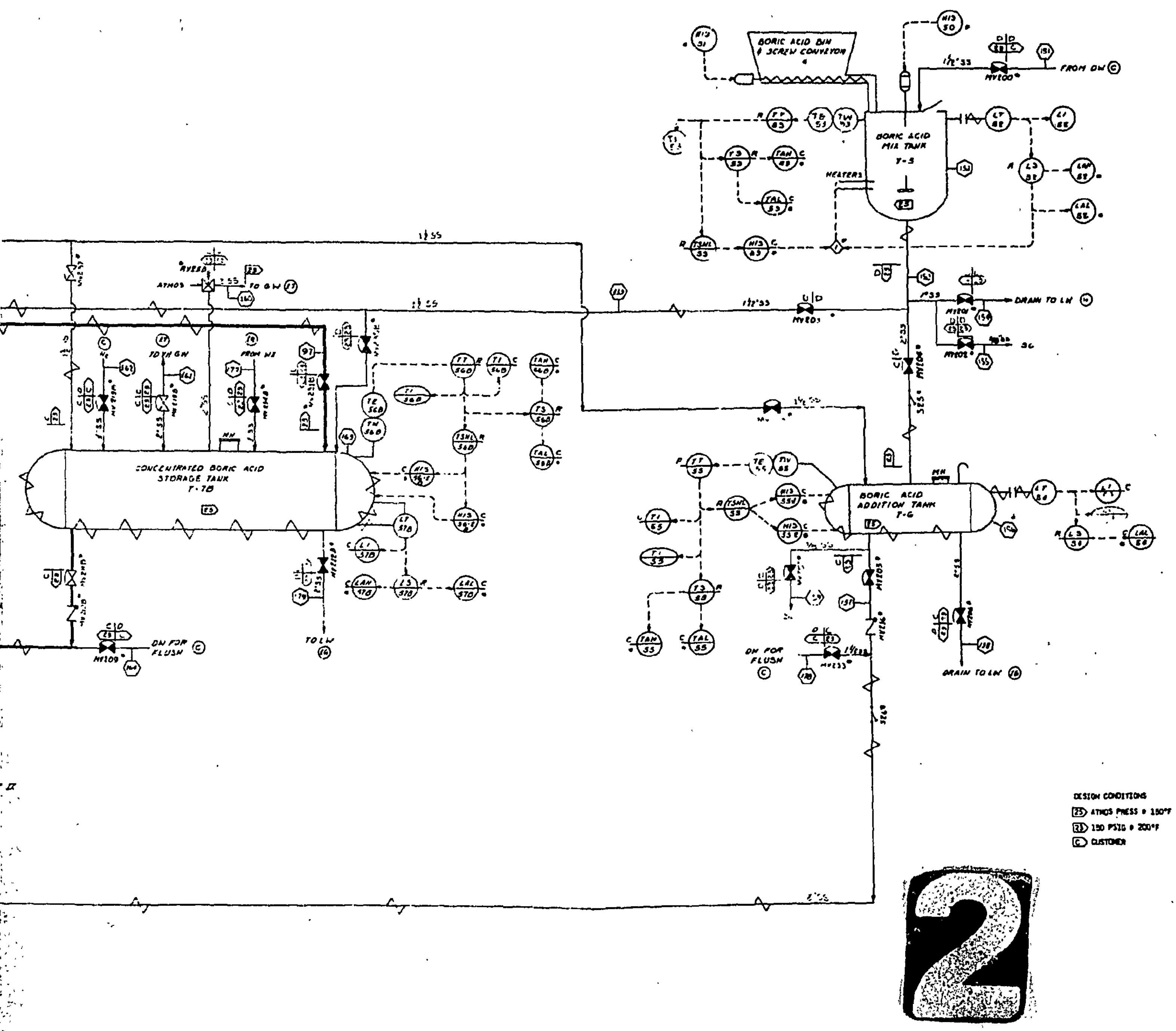
Figure A-11c. Chemical Addition and Boron Recovery System (Sheet 3 of 3)



Babcock & Wilcox

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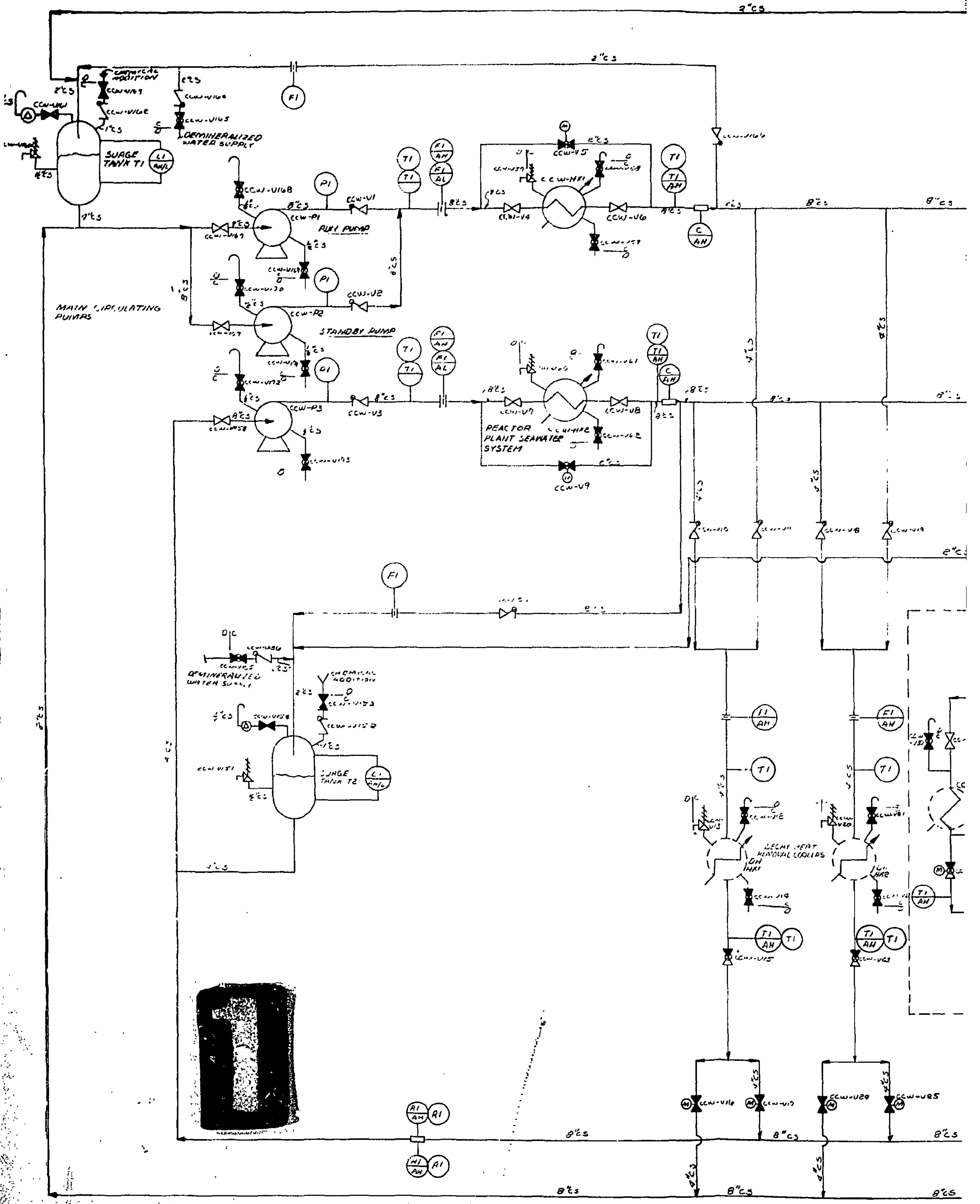

2"CS
FI
CHEMICAL ADDITION
DEMINERALIZED WATER SUPPLY
SURGE TANK T1
LI AH/L
CCW-V168
PI
CCW-P1
RUN PUMP
CCW-V1
TI
FI AH
FI AL
CCW-V5
CCW-HX1
CCW-V4
CCW-V6
TI AH
C AH
8"CS
MAIN CIRCULATING PUMPS
CCW-P2
STANDBY PUMP
CCW-V2
CCW-P3
CCW-V3
REACTOR PLANT SEAWATER SYSTEM
CCW-V7
CCW-V8
CCW-V9
CCW-V18
CCW-V19
DEMINERALIZED WATER SUPPLY
SURGE TANK T2
DECAY HEAT REMOVAL COOLERS
DH HX1
DH HX2
CCW-V15
CCW-V16
CCW-V17
CCW-V24
CCW-V25
RI AH
RI

89082 F-O

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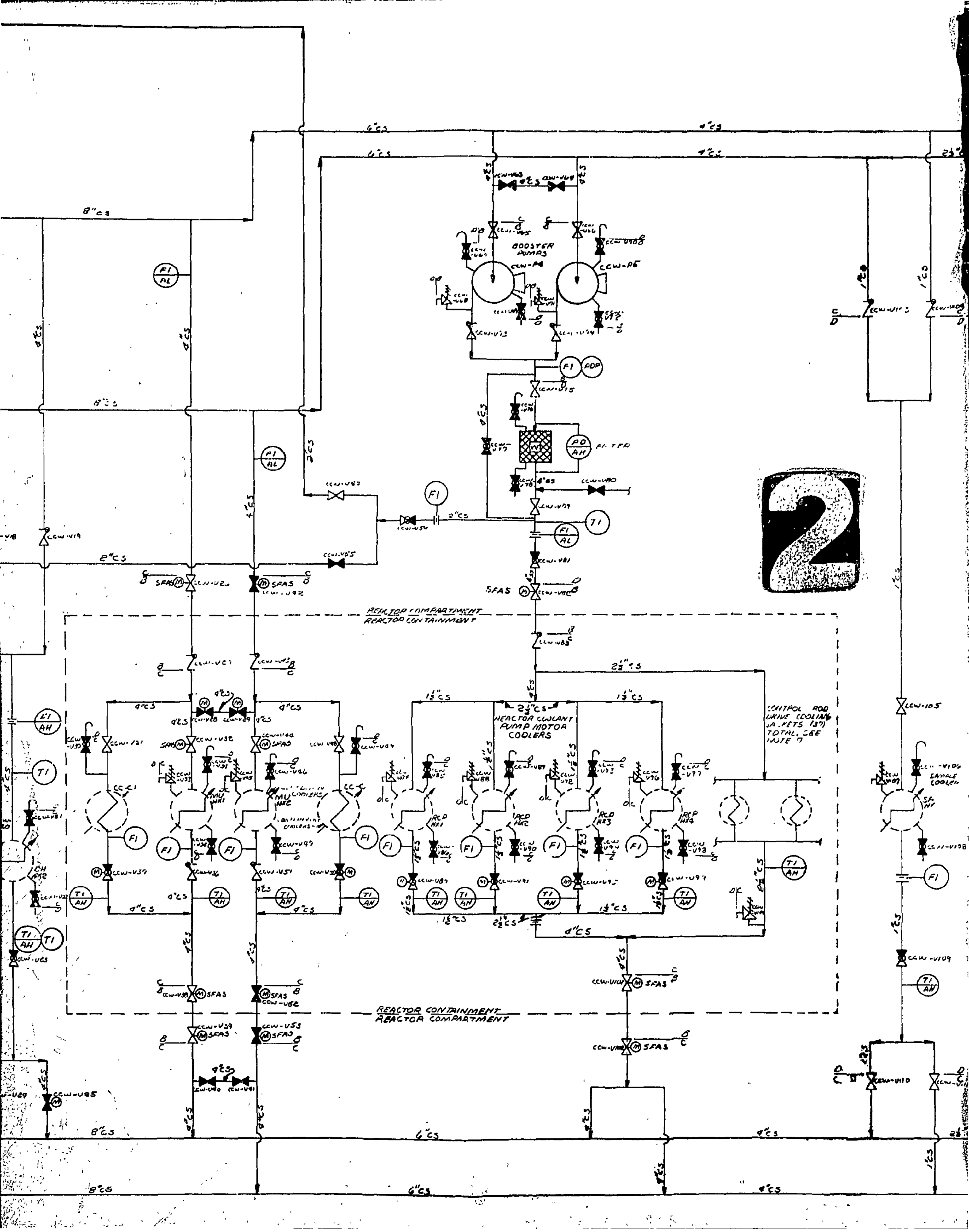

2
8"CS
6"CS
4"CS
2"CS
BOOSTER PUMPS
CCW-P4
CCW-P5
FILTER
SFAS
REACTOR COMPARTMENT
REACTOR CONTAINMENT
REACTOR CONTAINMENT
REACTOR COMPARTMENT
REACTOR COOLANT PUMP MOTOR COOLERS
CONTROL ROD DRIVE COOLING JACKETS (37) TOTAL, SEE NOTE 7
CCW-V105
CCW-V109
CCW-V110
CCW-V85

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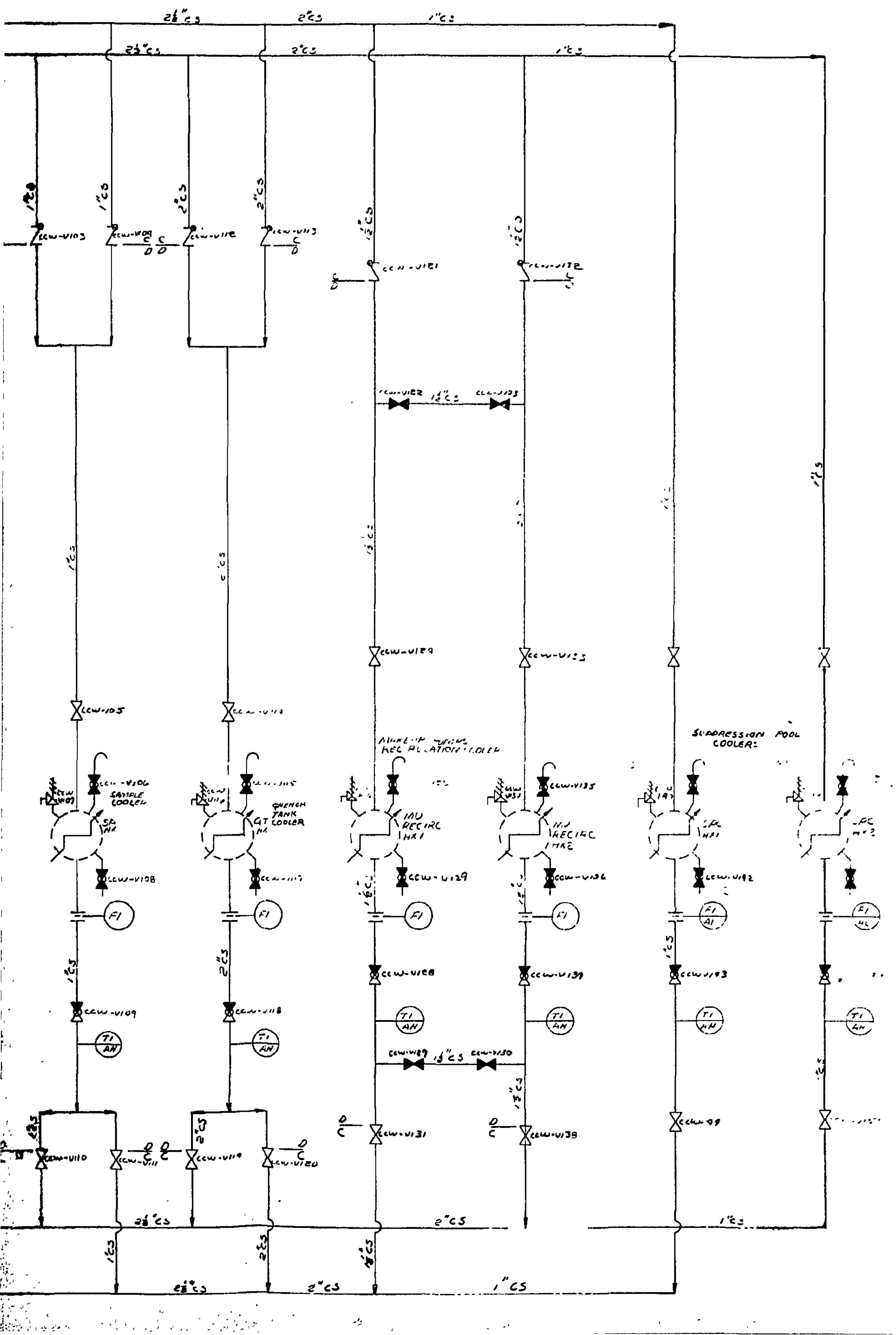
Figure A-



SAMPLE COOLER
SUPPRESSION POOL COOLERS
MU RECIRC HX1
MU RECIRC HX2
CCW-V103
CCW-V129
CCW-V131
CCW-V138
CCW-V110
CCW-V119
CCW-V109
CCW-V108
CCW-V128
CCW-V135
CCW-V143

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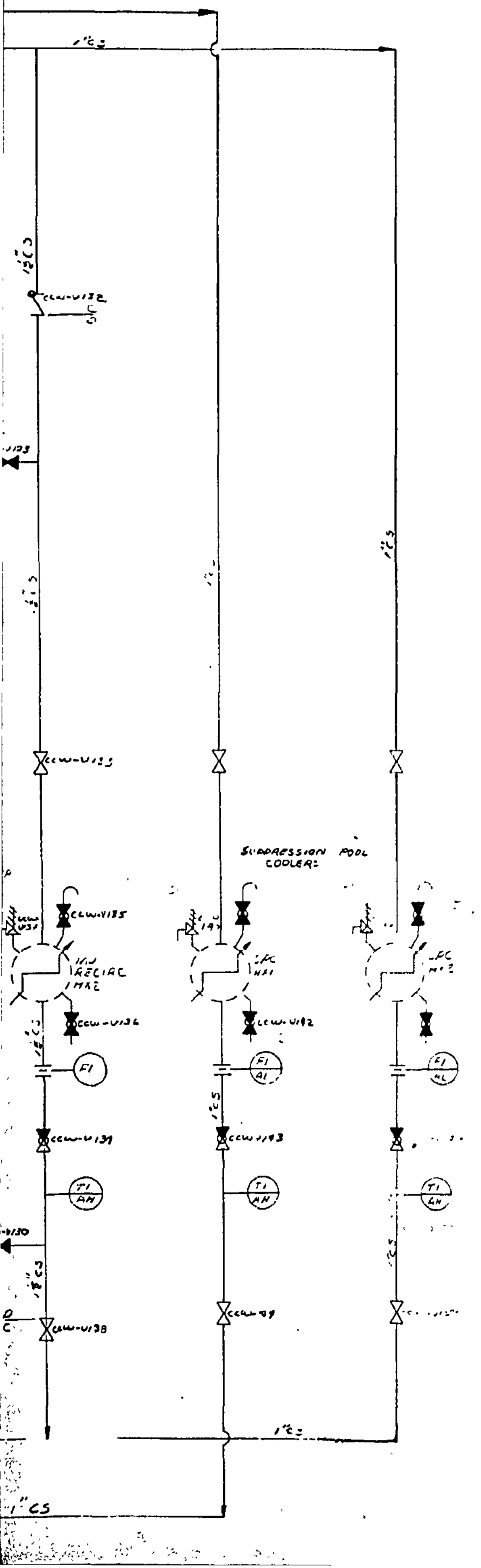
Figure A-12. Component Cooling Water System



NOTES:

1. The system is ASME III Class 3, USNRC Quality Group Classification C except for the following:
   a. Containment penetration piping and associated isolation piping — ASME III Class 2, QG B.
   b. CCW booster pumps and associated isolation valves — ASME III Class 2, QG B.
   c. Piping, valves, and components serving the sample cooler, makeup pump, recirculation coolers, and quench tank cooler — USNRC QG B.
2. Last valve is CCW-1173.
3. Low points in piping shall be provided with screwed brass plugs for drainage.
4. High points shall be provided with valve vents for air purging.
5. The surge tank shall be located at an elevation above the highest point in the system.
6. System lineup is shown with CNSG at 100% rated power.
7. The system is designed for 41 CRDMs.

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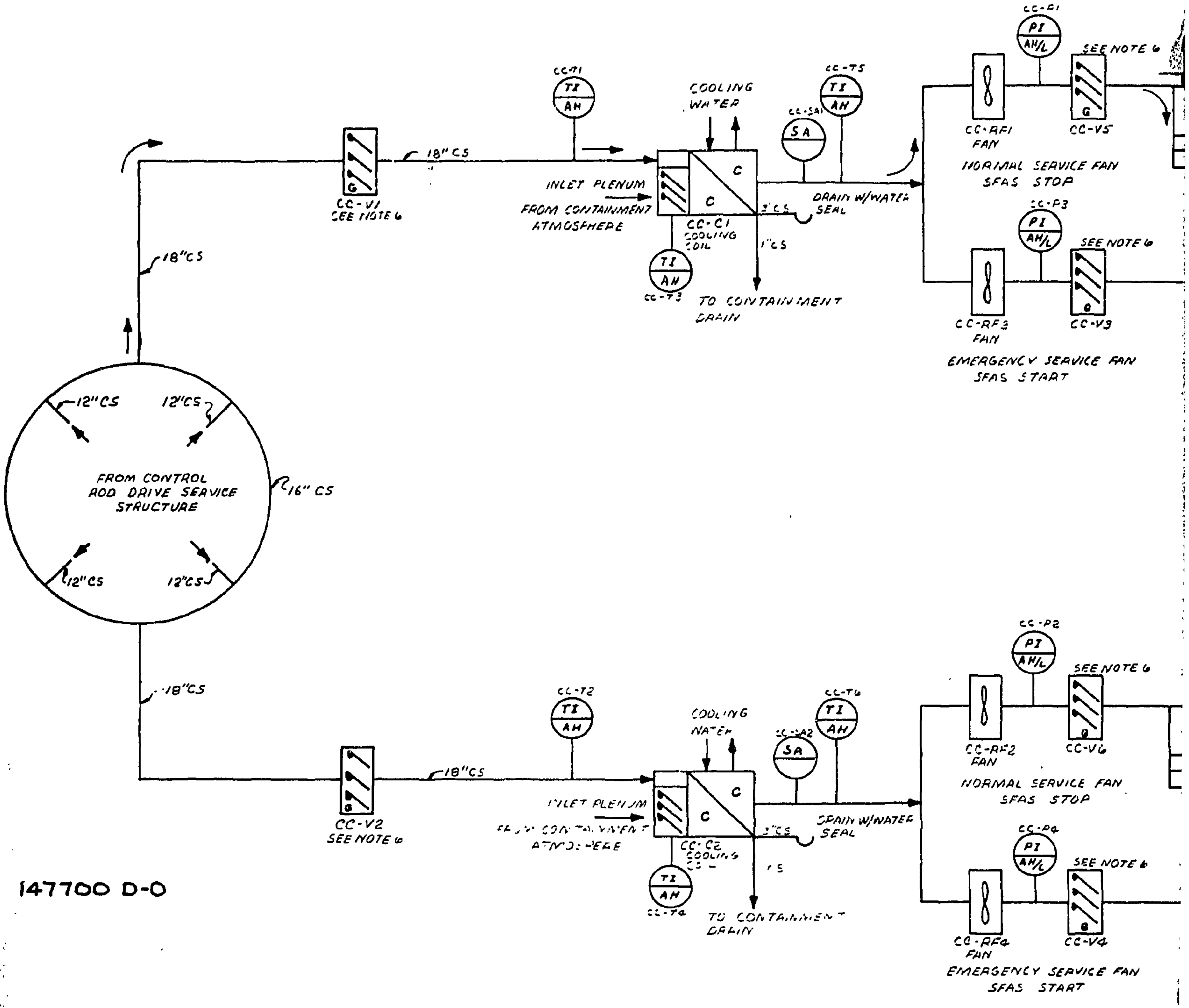

CC-T1
TI
AH
COOLING WATER
CC-SA1
SA
CC-T5
TI
AH
CC-P1
PI
AH/L
SEE NOTE 6
CC-RF1
FAN
CC-V5
NORMAL SERVICE FAN
SFAS STOP
18" CS
CC-V1
SEE NOTE 6
INLET PLENUM
FROM CONTAINMENT ATMOSPHERE
CC-C1
COOLING COIL
DRAIN W/WATER SEAL
3" CS
1" CS
CC-T3
TI
AH
TO CONTAINMENT DRAIN
CC-P3
PI
AH/L
SEE NOTE 6
CC-RF3
FAN
CC-V3
EMERGENCY SERVICE FAN
SFAS START
18" CS
12" CS
12" CS
FROM CONTROL
ROD DRIVE SERVICE
STRUCTURE
16" CS
12" CS
12" CS
18" CS
CC-T2
TI
AH
COOLING WATER
SA
CC-T6
TI
AH
CC-P2
PI
AH/L
SEE NOTE 6
CC-RF2
FAN
CC-V6
NORMAL SERVICE FAN
SFAS STOP
18" CS
CC-V2
SEE NOTE 6
DRAIN W/WATER SEAL
3" CS
CC-C2
COOLING
TI
AH
CC-T4
TO CONTAINMENT
DRAIN
CC-P4
PI
AH/L
SEE NOTE 6
CC-RF4
FAN
CC-V4
EMERGENCY SERVICE FAN
SFAS START
147700 D-0

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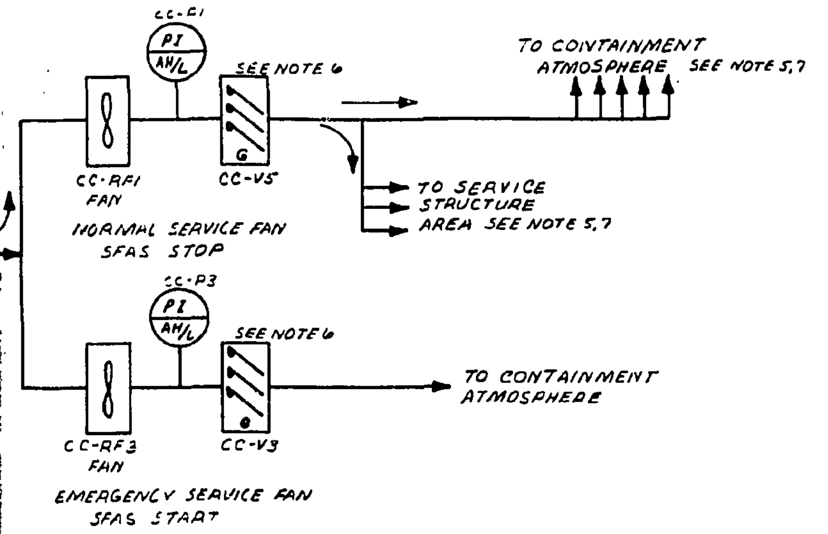
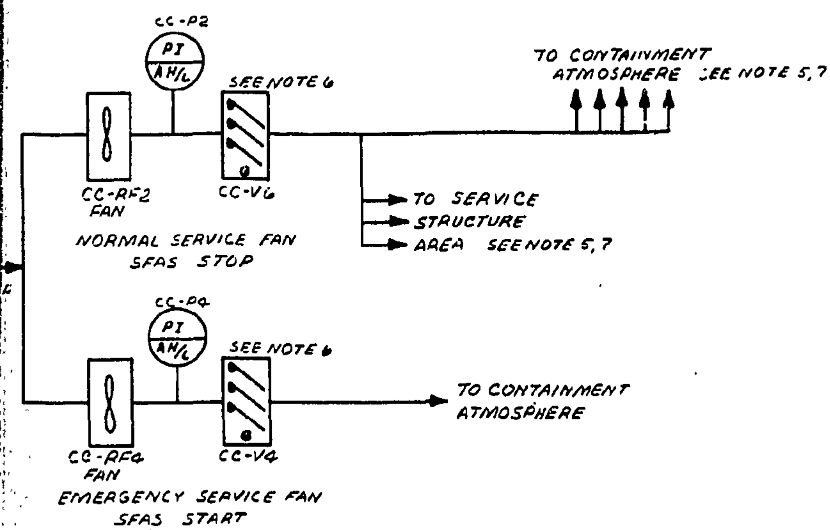
Figure A-13. Containment Dry Well Cooling System



NOTES
1. DUCT SIZES ARE GIVEN FOR ROUND DUCTING. THE EQUIVALENT RECTANGULAR DIMENSIONED DUCTING MAY ALSO BE USED.
2. THIS SYSTEM IS ASME III CLASS 2, QUALITY GROUP CLASSIFICATION B EXCEPT FOR THOSE PORTIONS OF THE SYSTEM THAT PERFORM NO SAFETY FUNCTION. NON SAFETY RELATED PARTS OF THE SYSTEM ARE ASME III CLASS 3, QUALITY GROUP CLASSIFICATION C
3. SYSTEM IS SHOWN IN THE NORMAL OPERATING MODE
4. THE FAN/COIL AND INLET PLENUM HOUSING IS PROTECTED FROM OVER PRESSURIZATION DURING A LOCA BY SPECIAL RELIEF DEVICES THAT WILL RELIEVE IN EITHER DIRECTION.
5. ADJUSTABLE LOUVERS ARE PROVIDED TO BALANCE THE FLOW OF COOL AIR THROUGH THE NORMAL SERVICE DISTRIBUTION DUCTING
6. ELECTRICAL CONTACTS ARE PROVIDED TO INDICATE OPEN AND CLOSED POSITION OF REVERSE FLOW PREVENTING DAMPERS
7. DUCTING SHALL BE PROVIDED TO DIRECT THE COOLED AIR TO KNOWN CONTAINMENT HIGH TEMPERATURE AREAS. FURTHER DUCTING FOR SPOT COOLING WILL BE SIZED AND FIELD RUN AS NEEDED



DESIGN CONDITIONS

15 AIR SIDE OF COILS	861.84 KPag @	176.6°C
14 WATER SIDE OF COILS	1034.24 KPag @	107.2°C

16 DUCTING AND HOUSING ARE DESIGNED FOR 3.73 KPa DIFFERENTIAL IN EITHER DIRECTION, AND 176.6°C

2

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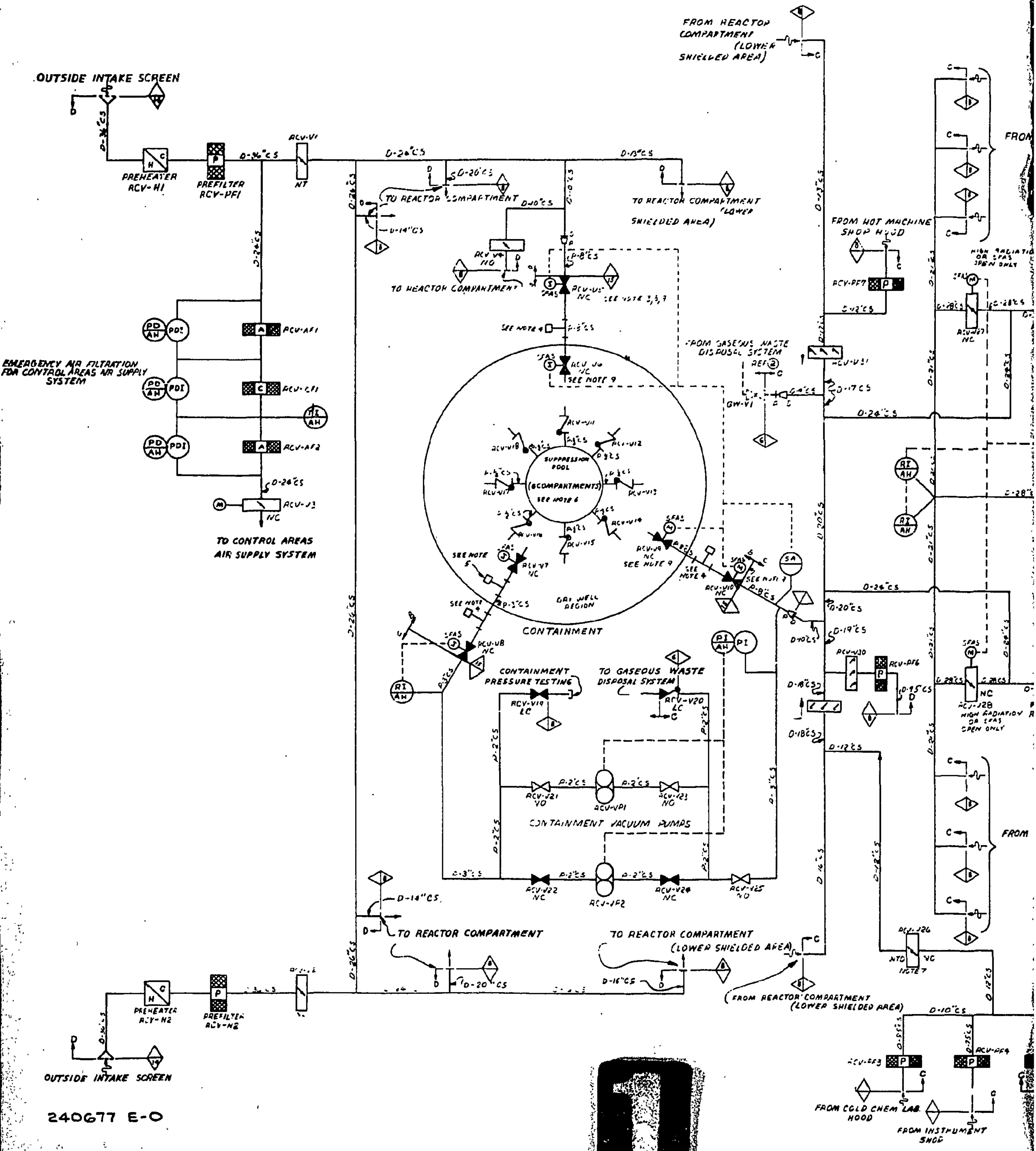
OUTSIDE INTAKE SCREEN
PREHEATER RCV-H1
PREFILTER RCV-PF1
RCV-V1
NT
EMERGENCY AIR FILTRATION FOR CONTROL AREAS AIR SUPPLY SYSTEM
RCV-AF1
RCV-CF1
RCV-AF2
RCV-V3
NC
TO CONTROL AREAS AIR SUPPLY SYSTEM
FROM REACTOR COMPARTMENT (LOWER SHIELDED AREA)
TO REACTOR COMPARTMENT
TO REACTOR COMPARTMENT (LOWER SHIELDED AREA)
SEE NOTE 9
SUPPRESSION POOL
(8 COMPARTMENTS)
SEE NOTE 6
DRY WELL REGION
CONTAINMENT
FROM GASEOUS WASTE DISPOSAL SYSTEM
GW-V1
RCV-V31
FROM HOT MACHINE SHOP HOOD
RCV-PF7
HIGH RADIATION OR SFAS OPEN ONLY
CONTAINMENT PRESSURE TESTING
RCV-V19
LC
TO GASEOUS WASTE DISPOSAL SYSTEM
RCV-V20
LC
RCV-V21
NO
RCV-VP1
RCV-V23
NO
CONTAINMENT VACUUM PUMPS
RCV-V22
NC
RCV-VP2
RCV-V24
NC
RCV-V25
NO
RCV-V30
RCV-PF6
RCV-V28
NC
RCV-V26
NTO
NC
NOTE 7
PREHEATER RCV-H2
PREFILTER RCV-H2
FROM REACTOR COMPARTMENT (LOWER SHIELDED AREA)
RCV-PF3
RCV-PF4
FROM COLD CHEM LAB. HOOD
FROM INSTRUMENT SHOP
FROM

240677 E-0

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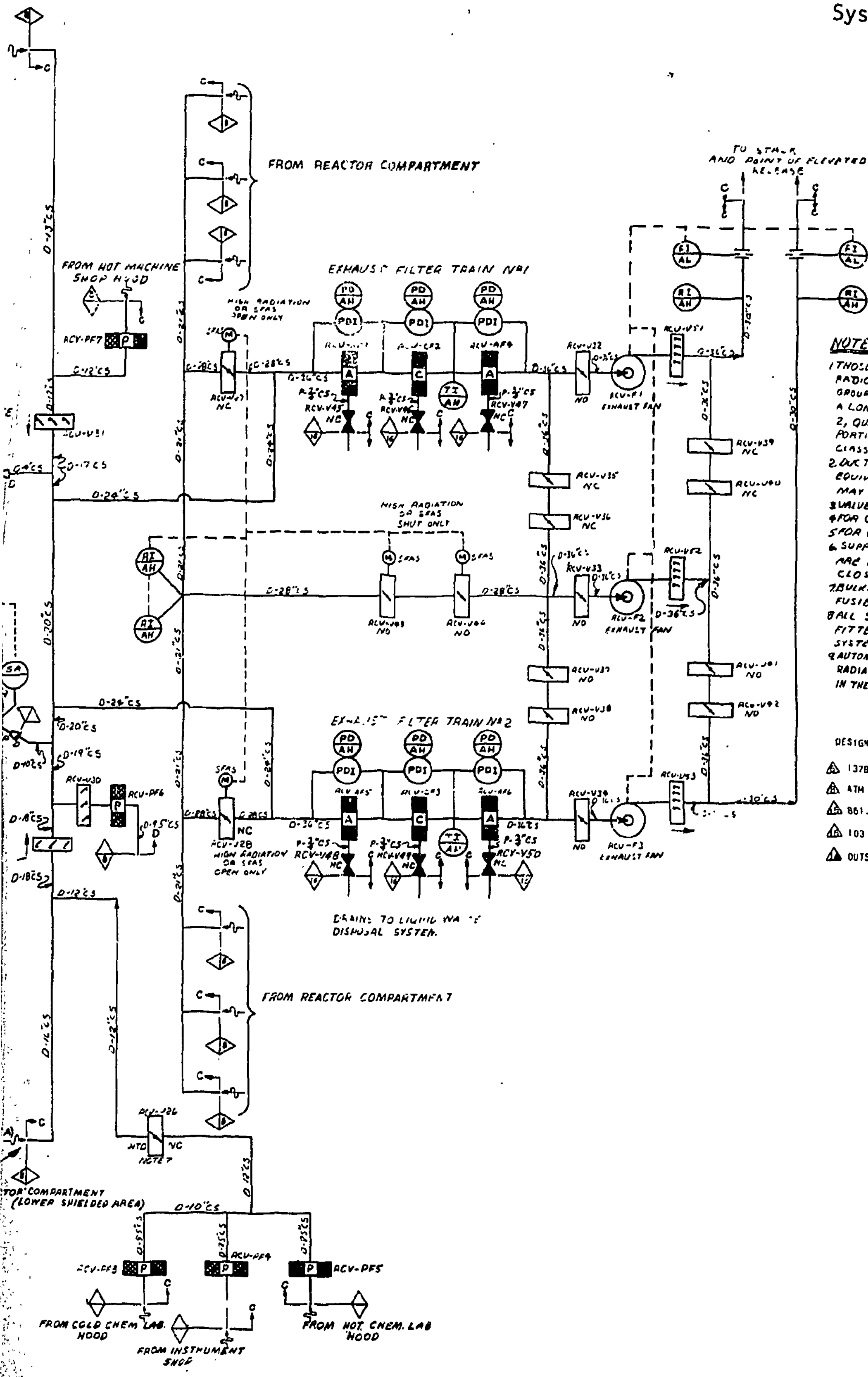
Figure A-14. Reactor Compartment Ventilation System



NOTES

1. THOSE PORTIONS OF THE SYSTEM THAT MAY HANDLE RADIOACTIVE PARTICLES AND/OR GASES ARE QUALITY GROUP CLASSIFICATION C. THOSE PORTIONS THAT CONSTITUTE A CONTAINMENT PENETRATION ARE ASME III, CLASS 2, QUALITY GROUP CLASSIFICATION B. ALL OTHER PORTIONS OF THE SYSTEM ARE QUALITY GROUP CLASSIFICATION D
2. DUCT SIZES ARE GIVEN FOR ROUND DUCTWORK. THE EQUIVALENT RECTANGULAR DIMENSIONED DUCTING MAY ALSO BE USED.
3. VALVE FITTED WITH AN INTERNAL PILOT
4. FOR CONTAINMENT PENETRATION LEAK RATE TESTING
5. FOR VENTING CONTROL ROD DRIVE MOTOR TUBES
6. SUPPRESSION POOL VENT VALVES (V11 THROUGH V18) ARE INSTALLED SO GRAVITY WILL HOLD THEM CLOSED FOR NORMAL SHIPS ATTITUDES
7. BULKHEAD FIRE DAMPERS ARE FITTED WITH FUSIBLE LINKS PER STANDARD MARINE PRACTICE
8. ALL SUPPLY AND EXHAUST CONNECTIONS ARE FITTED WITH ADJUSTABLE LOUVERS FOR SYSTEM FLOW BALANCING.
9. AUTOMATICALLY CLOSE UPON RECEIPT OF SFAS SIGNAL, HIGH RADIATION ALARM INSIDE CONTAINMENT, OR INDICATION OF. SMOKE IN THE PURGE LINES

DESIGN CONDITIONS

1 1378.95 kpag @ 148.8°C
2 ATM @ 93.3°C
3 861.84 kpag @ 176.6°C
4 103.42 kpag @ 93.3°C
5 OUTSIDE AIR CONDITIONS

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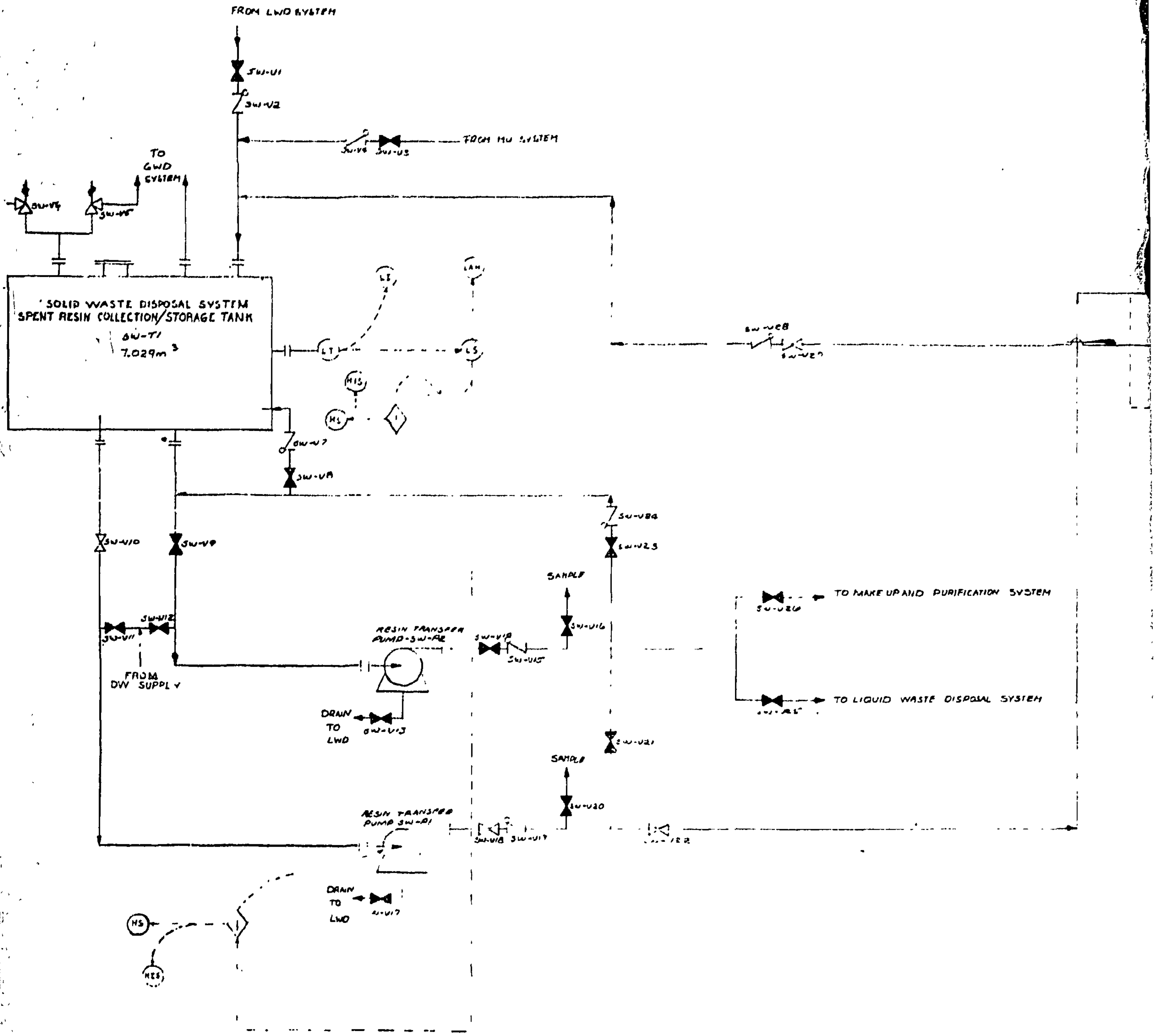
FROM LWD SYSTEM
SW-V1
SW-V2
FROM MU SYSTEM
SW-V3
TO GWD SYSTEM
SW-V5
SOLID WASTE DISPOSAL SYSTEM
SPENT RESIN COLLECTION/STORAGE TANK
SW-T1
7.029m³
LT
LS
HS
SW-V7
SW-V8
SW-V10
SW-V9
SW-V11
SW-V12
FROM DW SUPPLY
RESIN TRANSFER PUMP-SW-P2
DRAIN TO LWD
SW-V13
SAMPLE
SW-V16
SW-V15
SW-V23
SW-V24
SW-V21
SW-V20
TO MAKE UP AND PURIFICATION SYSTEM
TO LIQUID WASTE DISPOSAL SYSTEM
RESIN TRANSFER PUMP SW-P1
SW-V18
SW-V17
HS

39087 F-0

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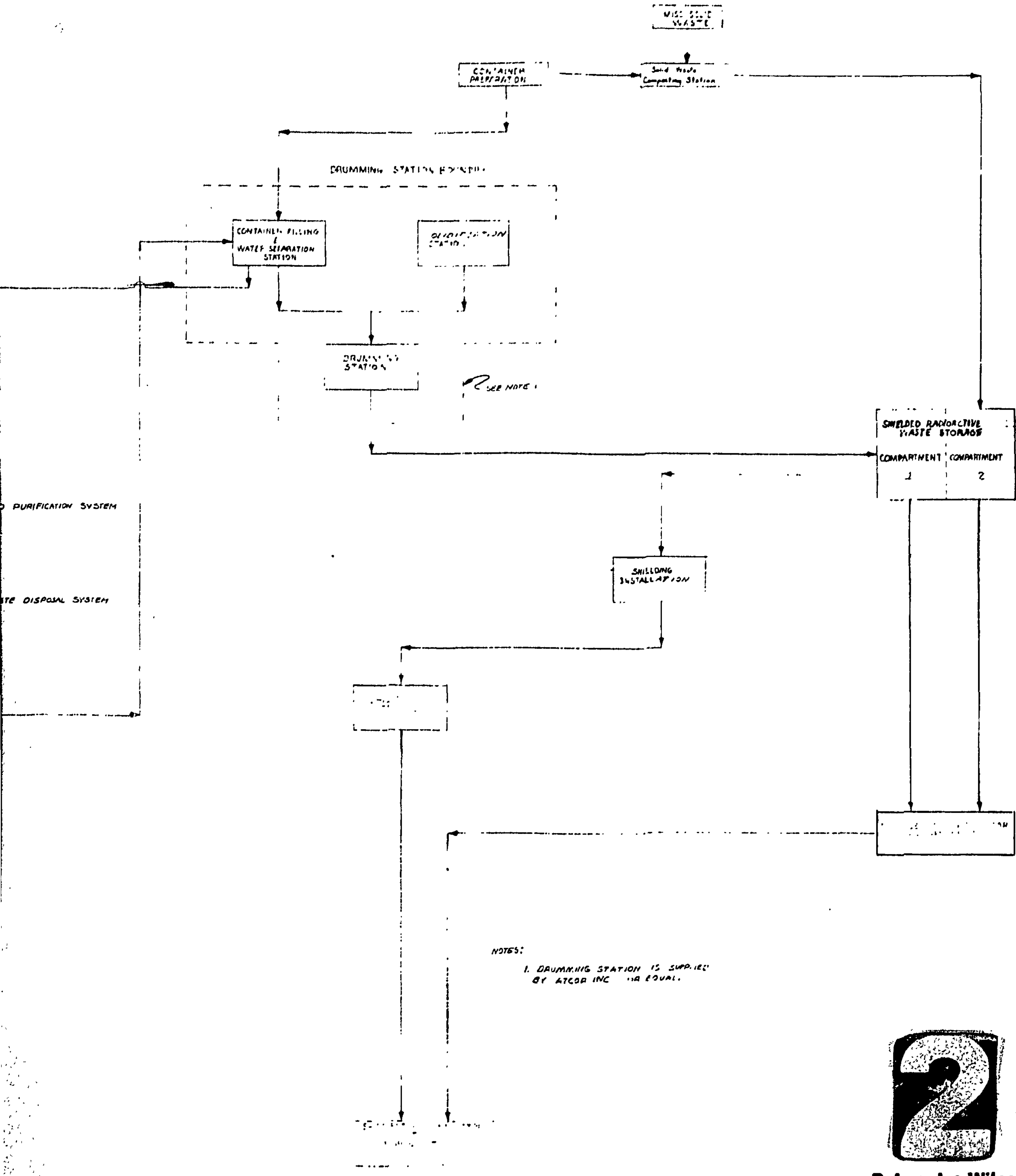
Figure A-15. Solid Waste Disposal System



NOTES:

1. DRUMMING STATION IS SUPPLIED BY ATCOR INC OR EQUAL.

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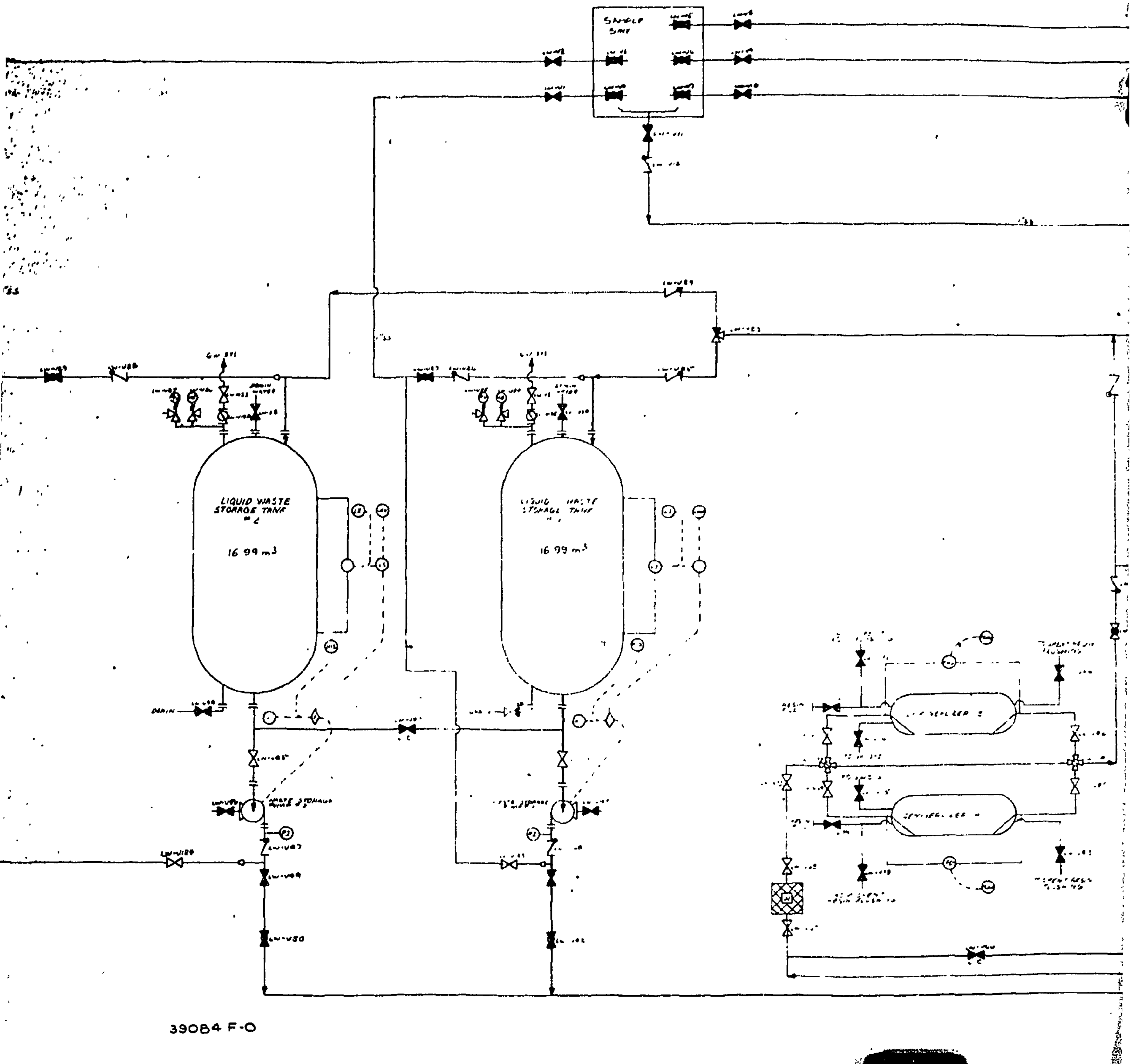

39084 F-O

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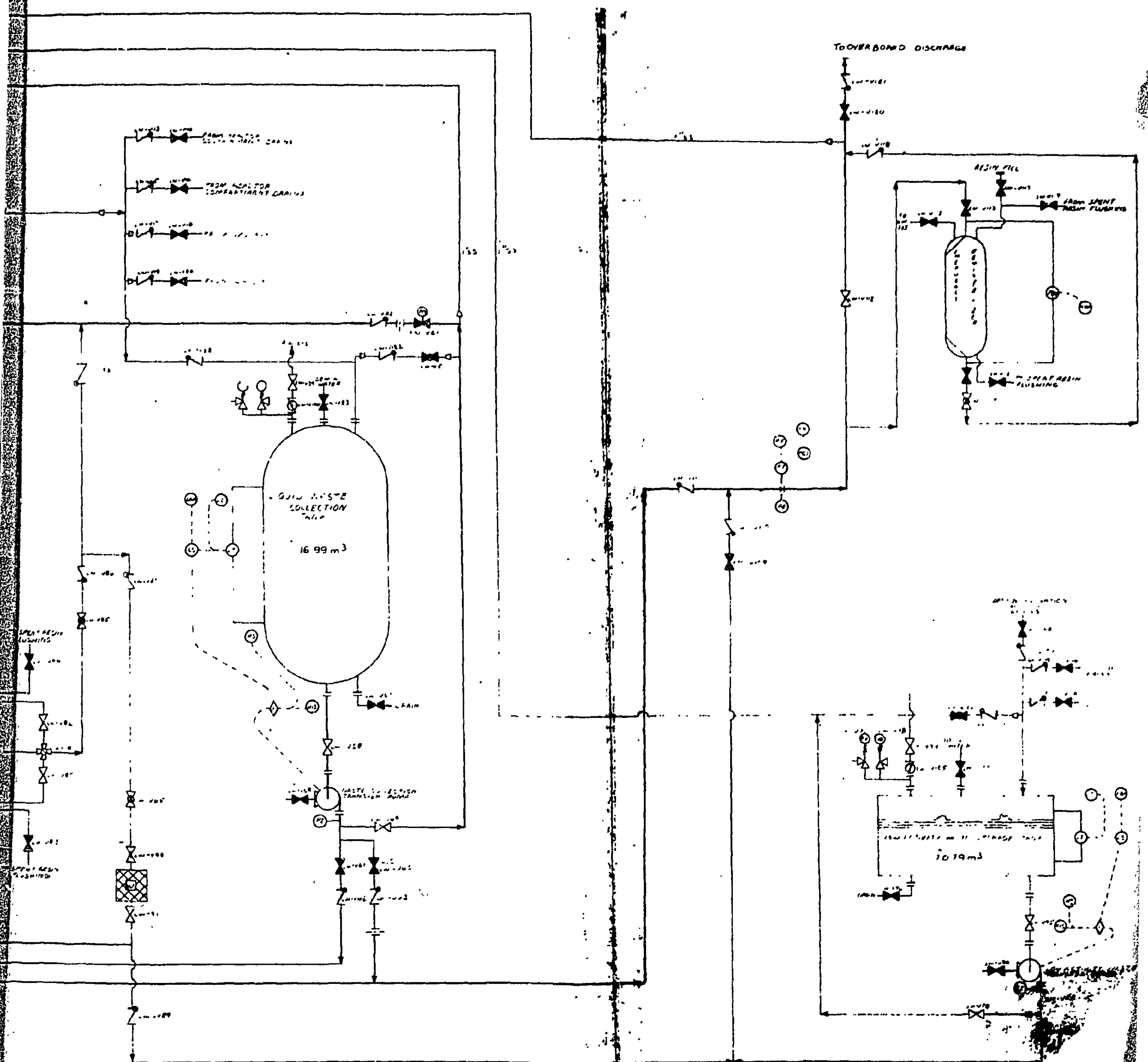
Figure A-16. Liquid Waste Disposal System



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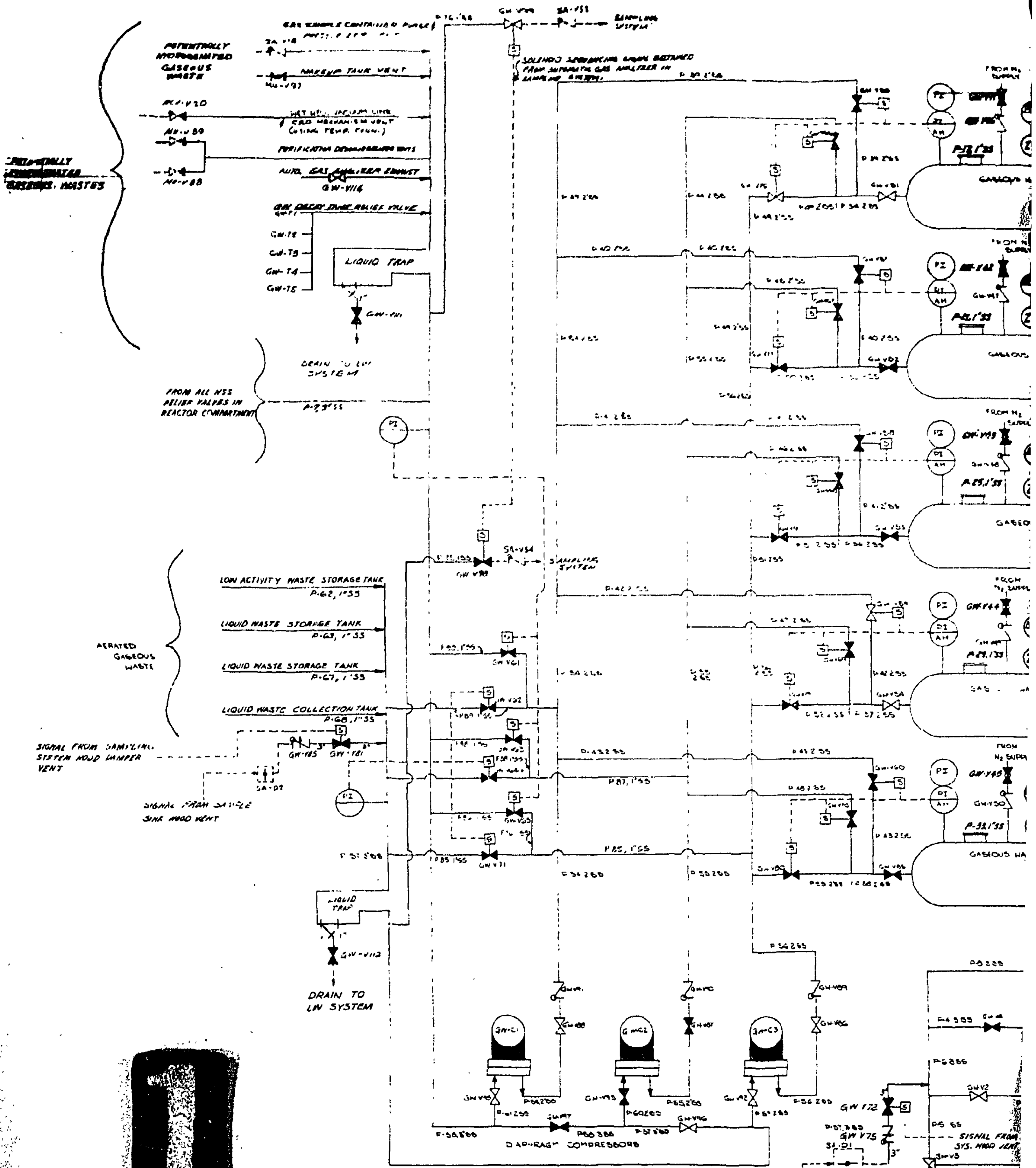
GAS SAMPLE CONTAINER PURGE
SAMPLING SYSTEM
GASEOUS WASTE
MAKEUP TANK VENT
PURIFICATION DEMINERALIZER VENTS
AUTO. GAS ANALYZER EXHAUST
GW-V114
GW-T2
GW-T3
GW-T4
GW-T5
LIQUID TRAP
DRAIN TO LW SYSTEM
FROM ALL NSS RELIEF VALVES IN REACTOR COMPARTMENT
LOW ACTIVITY WASTE STORAGE TANK
P-62, 1"SS
LIQUID WASTE STORAGE TANK
P-63, 1"SS
LIQUID WASTE STORAGE TANK
P-67, 1"SS
LIQUID WASTE COLLECTION TANK
P-68, 1"SS
AERATED GASEOUS WASTE
SIGNAL FROM SAMPLING SYSTEM HOOD DAMPER VENT
SIGNAL FROM SAMPLE SINK HOOD VENT
LIQUID TRAP
GW-V112
DRAIN TO LW SYSTEM
DIAPHRAGM COMPRESSORS
FROM SAMPLE SINK HOOD VENT
SIGNAL FROM SYS. HOOD VENT
GW V72
GW V75
SA-D1

39085 F-O

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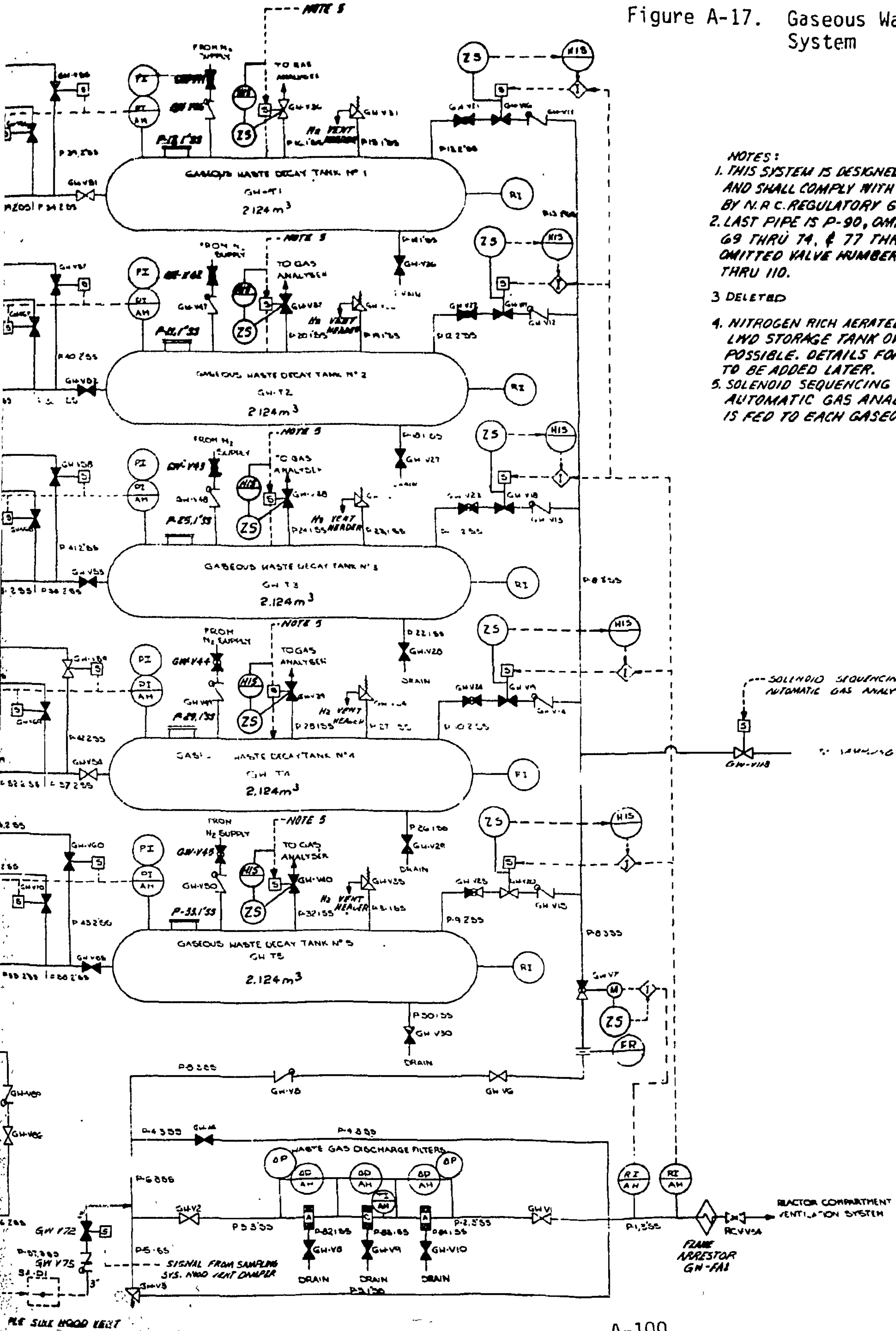
Figure A-17. Gaseous Waste Disposal System



NOTES:

1. THIS SYSTEM IS DESIGNED IN ACCORDANCE WITH ASME III, CLASS 3 AND SHALL COMPLY WITH QUALITY GROUP C AS SPECIFIED BY N.R.C. REGULATORY GUIDE 1.26.
2. LAST PIPE IS P-90, OMITTED PIPE NUMBERS 64, 65, 66, 69 THRU 74, & 77 THRU 84. LAST VALVE IS V-114, OMITTED VALVE NUMBERS 73, 79, 82, 83, 84, 94 & 100 THRU 110.
3. DELETED
4. NITROGEN RICH AERATED GASES ARE TO BE USED FOR LWD STORAGE TANK OVER PRESSURE WHERE POSSIBLE. DETAILS FOR PRESSURE REGULATED LINES TO BE ADDED LATER.
5. SOLENOID SEQUENCING SIGNAL OBTAINED FROM AUTOMATIC GAS ANALYZER IN SAMPLING SYSTEM IS FED TO EACH GASEOUS WASTE DECAY TANK.

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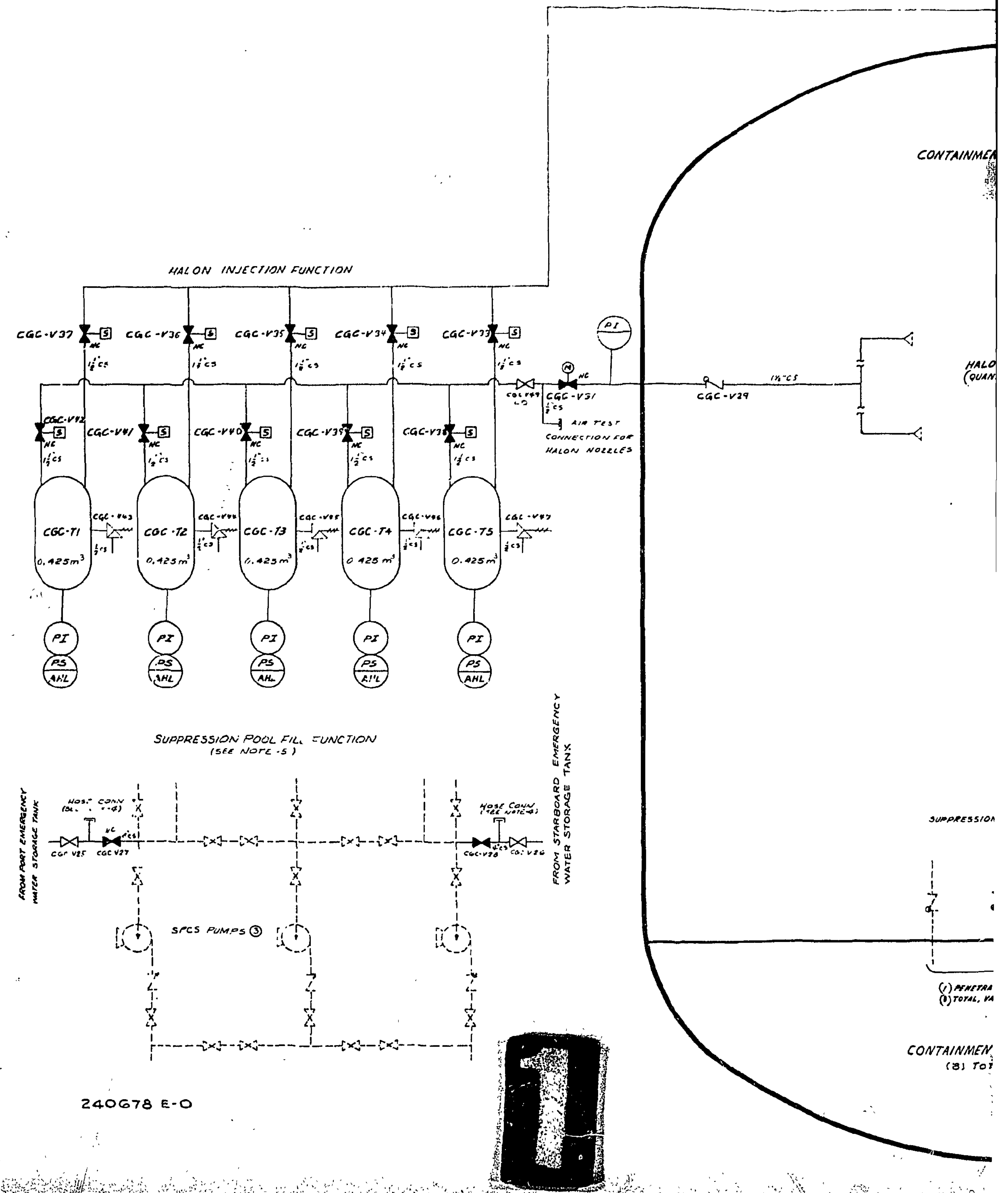

HALON INJECTION FUNCTION
CGC-V37
CGC-V36
CGC-V35
CGC-V34
CGC-V33
CGC-V42
CGC-V41
CGC-V40
CGC-V39
CGC-V38
CGC-T1
CGC-T2
CGC-T3
CGC-T4
CGC-T5
0.425 m³
CGC-V31
CGC-V29
AIR TEST CONNECTION FOR HALON NOZZLES
CONTAINMENT
SUPPRESSION POOL FILL FUNCTION
(SEE NOTE-5)
FROM PORT EMERGENCY WATER STORAGE TANK
FROM STARBOARD EMERGENCY WATER STORAGE TANK
HOSE CONN (SEE NOTE-4)
CGC-V25
CGC-V27
CGC-V28
CGC-V26
SPCS PUMPS ③
240678 E-0
1

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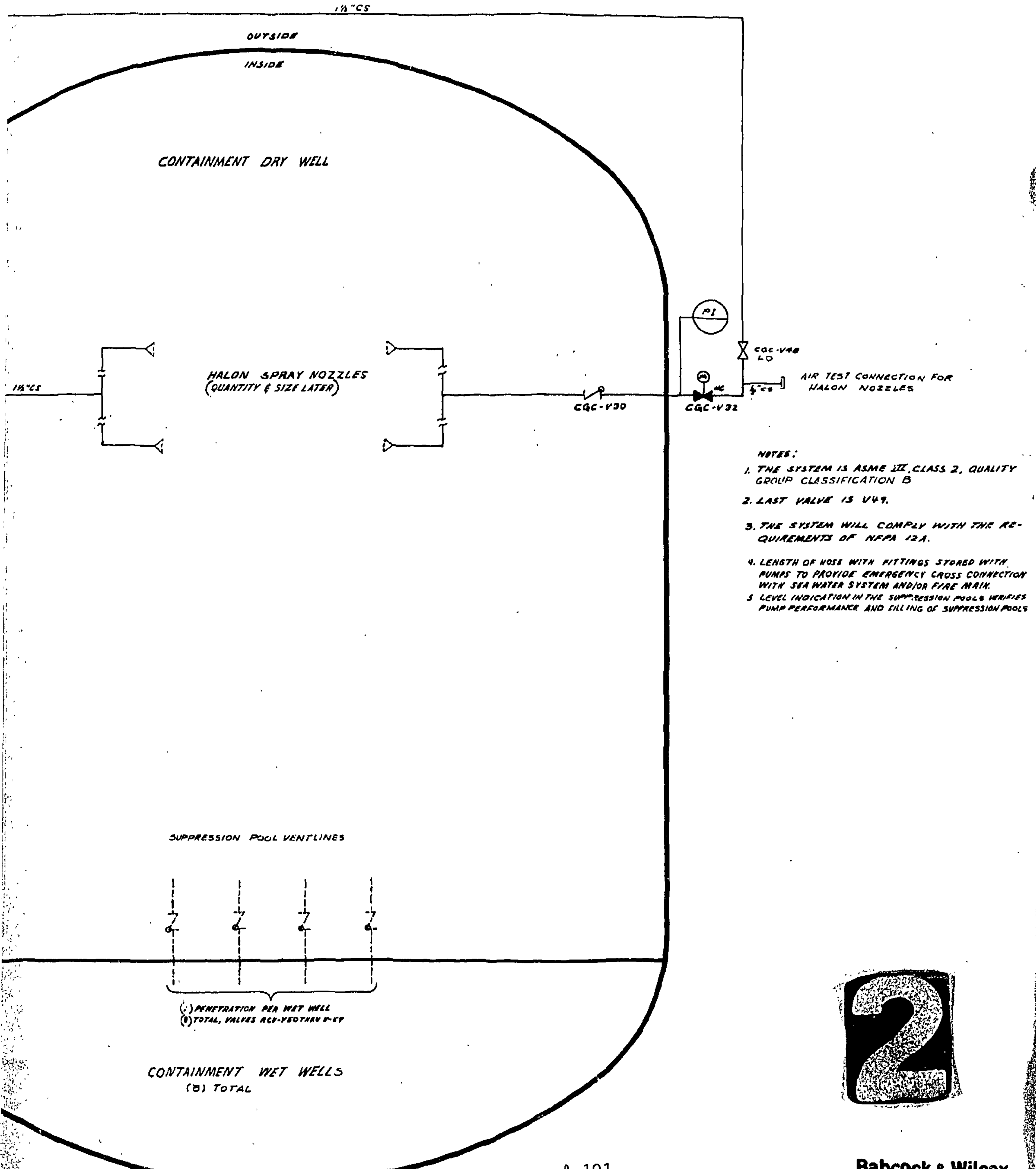
Figure A-18. Post-LOCA Combustible Gas Control System



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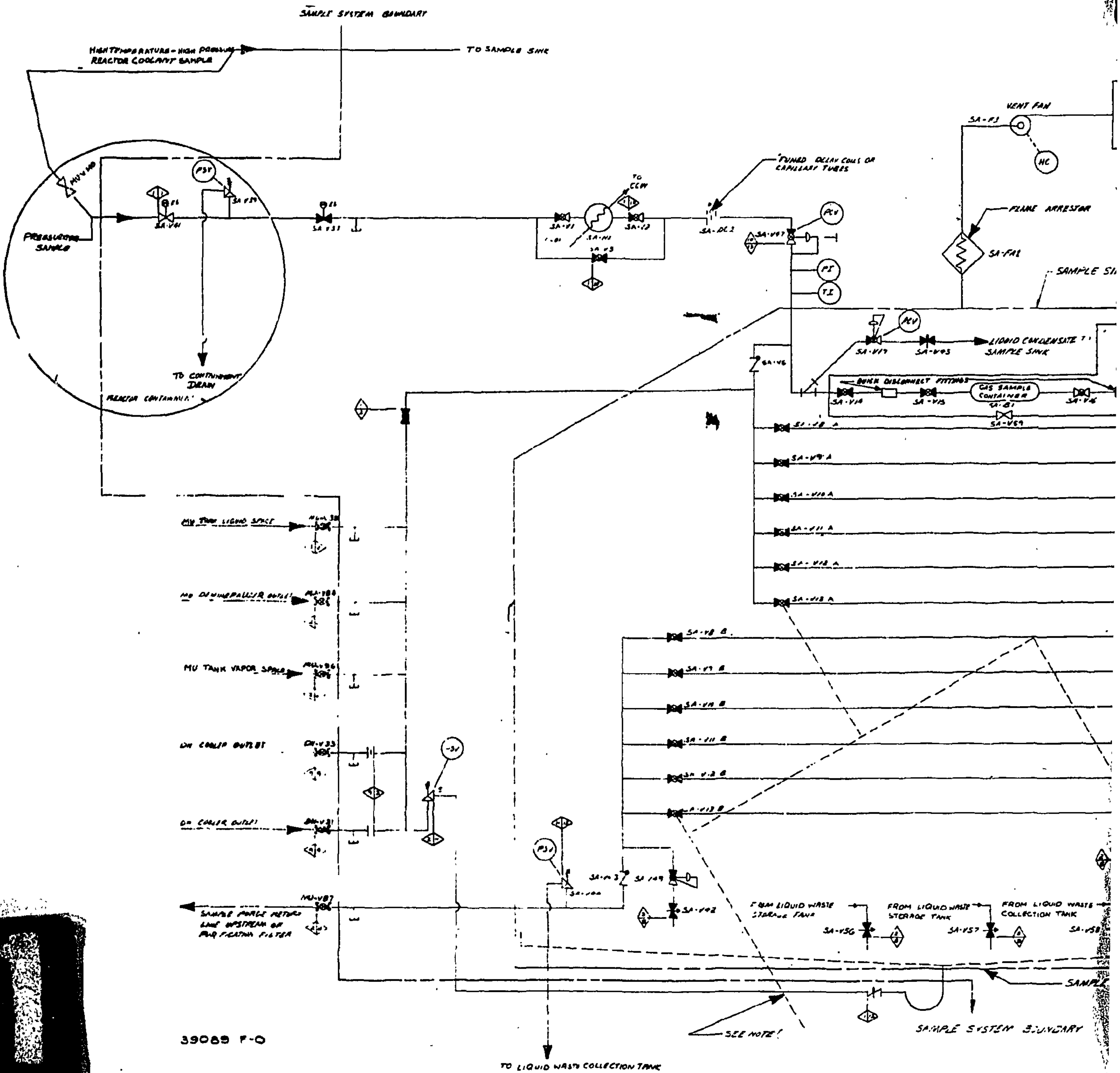
SAMPLE SYSTEM BOUNDARY
HIGH TEMPERATURE - HIGH PRESSURE REACTOR COOLANT SAMPLE
TO SAMPLE SINK
PRESSURIZER SAMPLE
TO CONTAINMENT DRAIN
REACTOR CONTAINMENT
TUNED DELAY COILS OR CAPILLARY TUBES
VENT FAN
FLAME ARRESTOR
LIQUID CONDENSATE SAMPLE SINK
QUICK DISCONNECT FITTINGS
GAS SAMPLE CONTAINER
MU TANK LIQUID SPACE
MU DEMINERALIZER OUTLET
MU TANK VAPOR SPACE
DH COOLER OUTLET
SAMPLE PURGE RETURN LINE UPSTREAM OF PURIFICATION FILTER
FROM LIQUID WASTE STORAGE TANK
FROM LIQUID WASTE STORAGE TANK
FROM LIQUID WASTE COLLECTION TANK
SAMPLE SYSTEM BOUNDARY
SEE NOTE 1
TO LIQUID WASTE COLLECTION TANK
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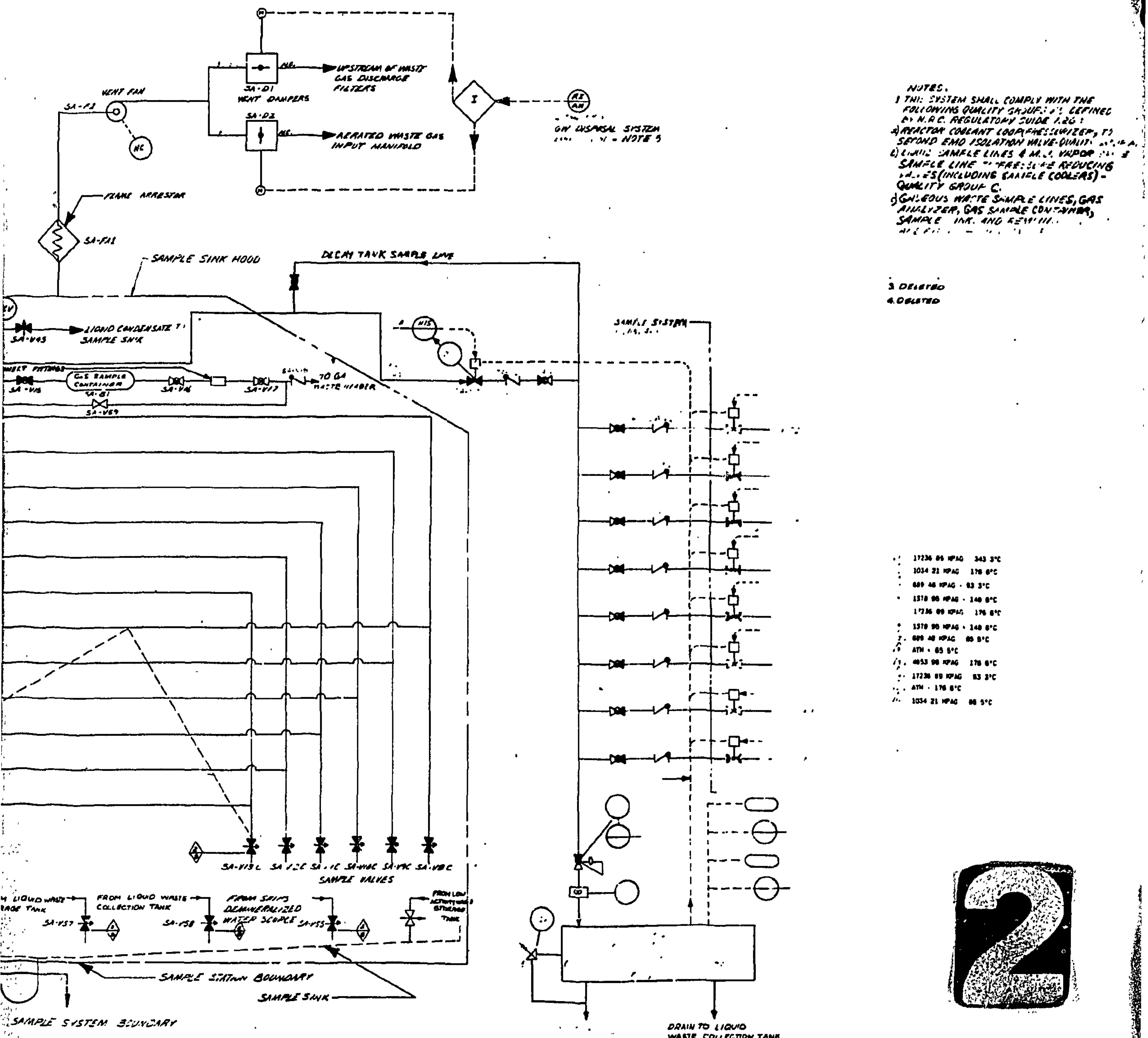
Figure A-19. Sampling System



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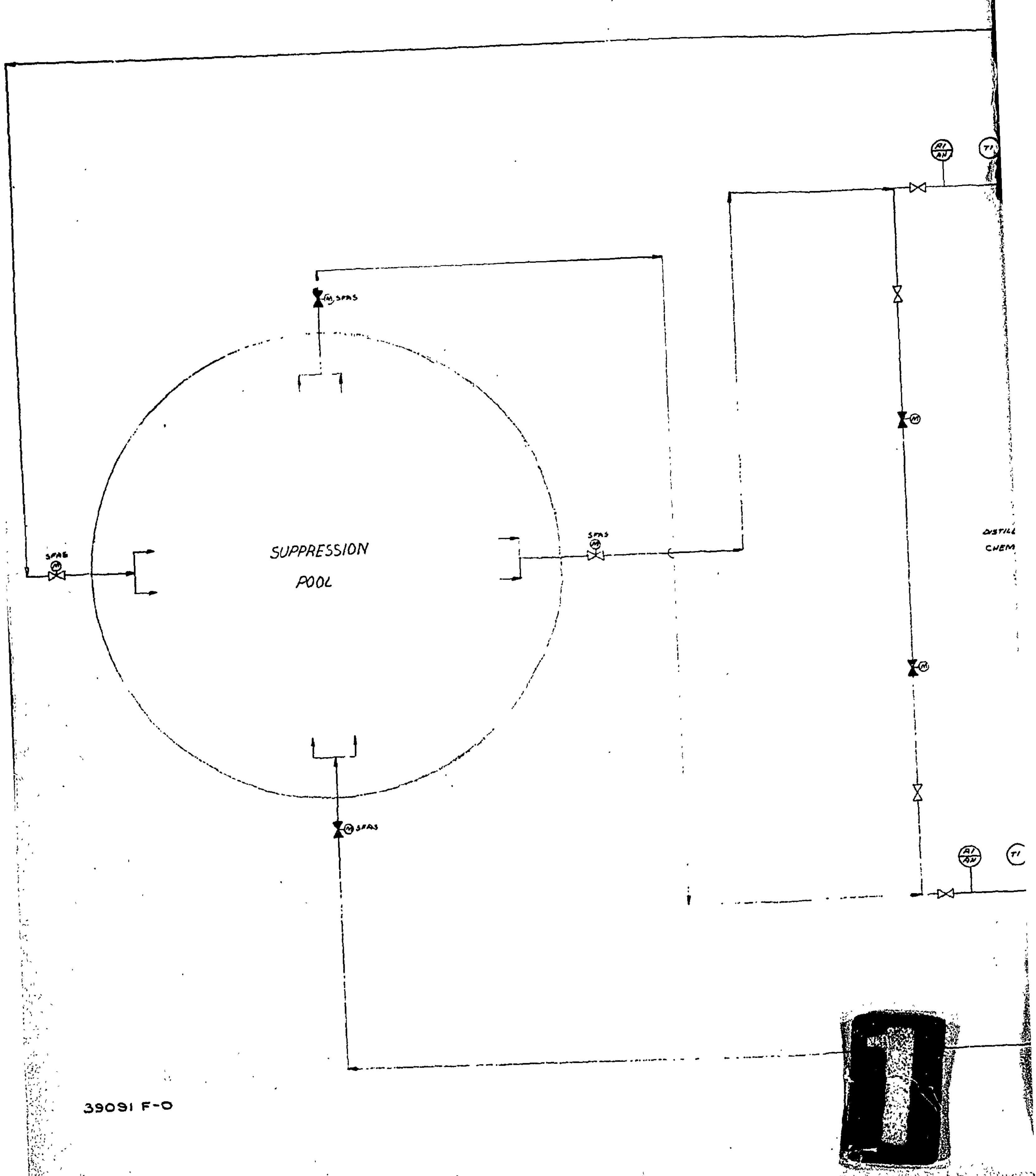

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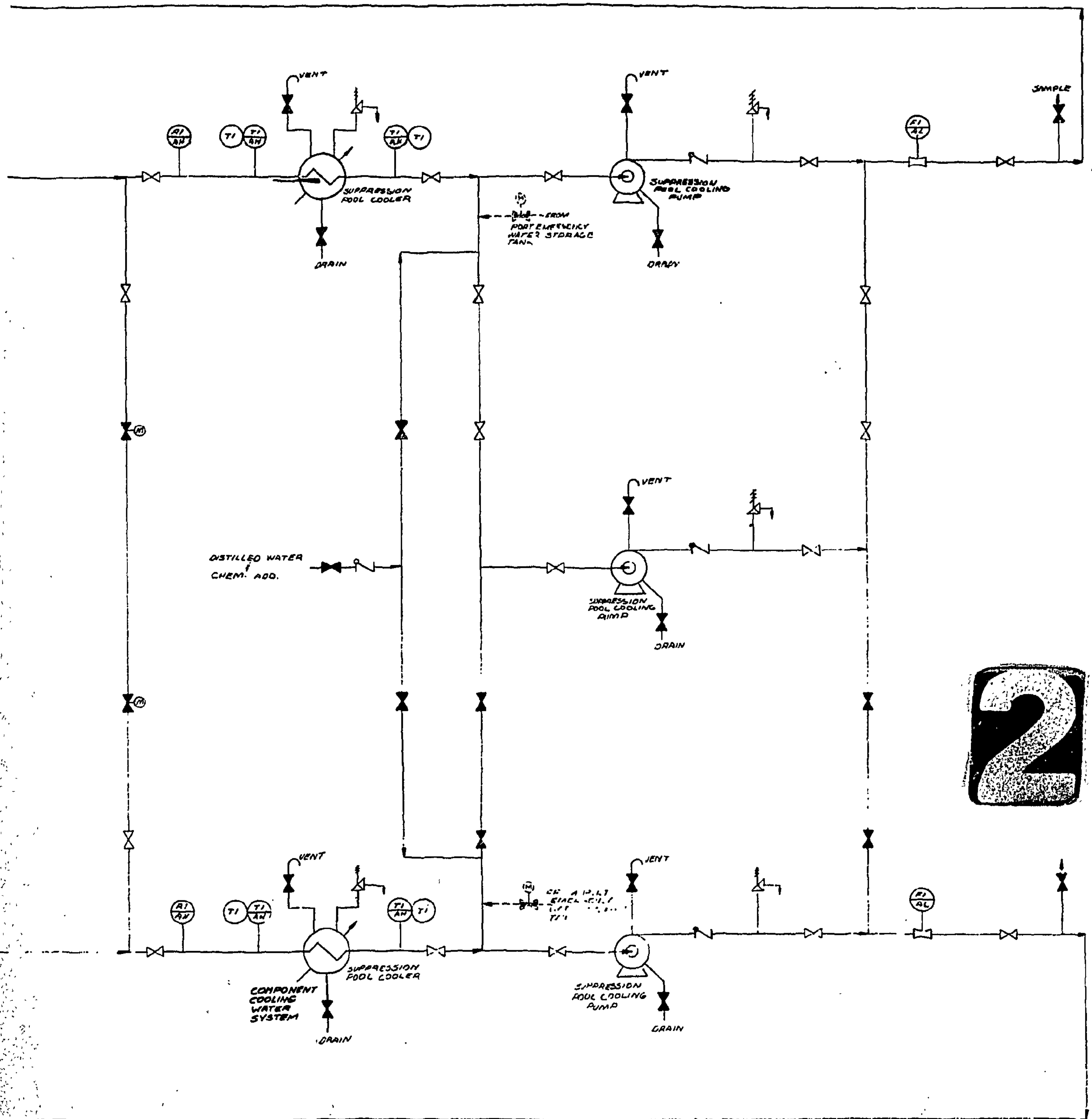
Figure A-20. Suppression Pool Cooling System



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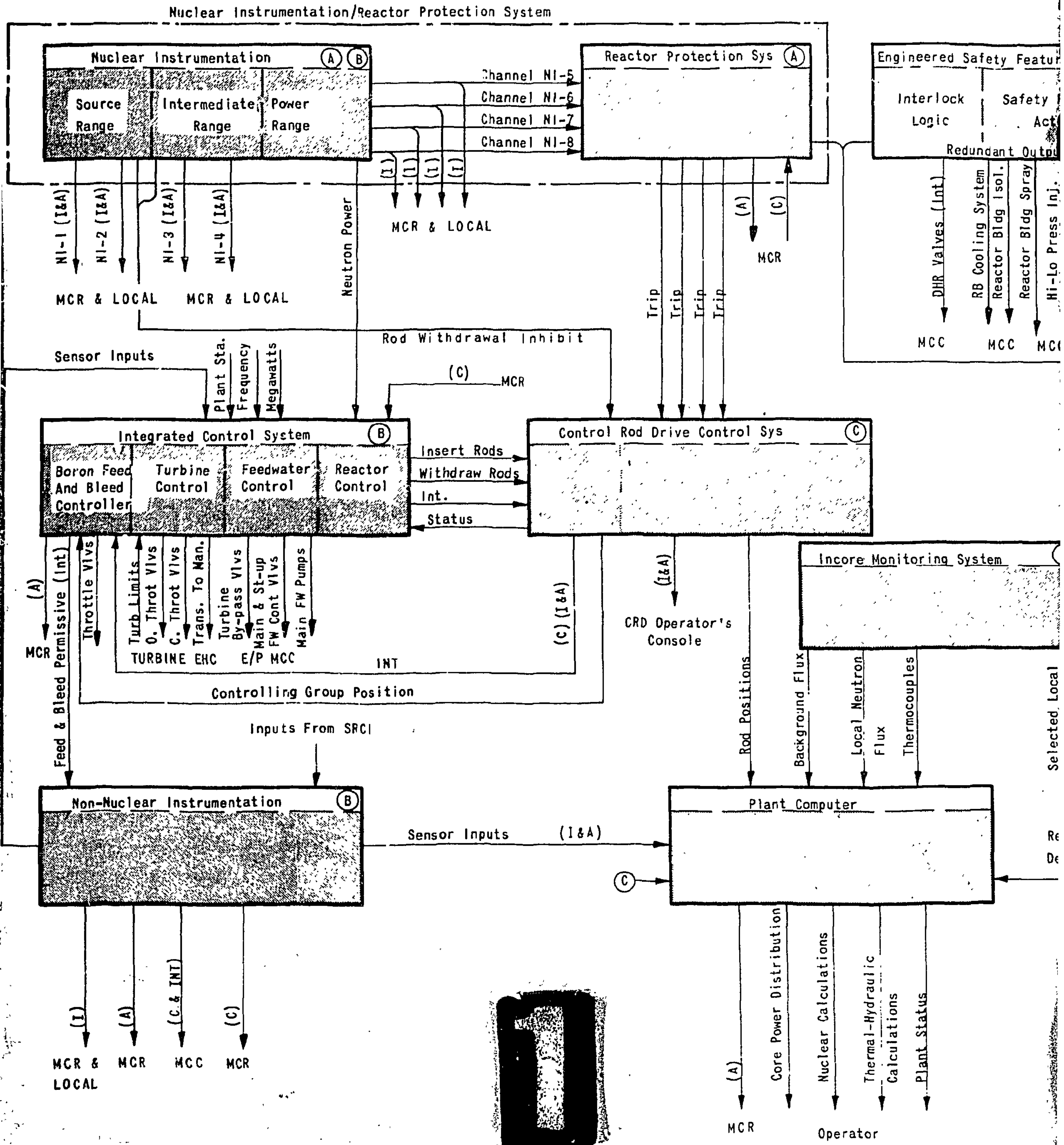

Nuclear Instrumentation/Reactor Protection System
Nuclear Instrumentation A B
Source Range
Intermediate Range
Power Range
Channel NI-5
Channel NI-6
Channel NI-7
Channel NI-8
(I) (I) (I) (I)
MCR & LOCAL
NI-1 (I&A)
NI-2 (I&A)
NI-3 (I&A)
NI-4 (I&A)
MCR & LOCAL
MCR & LOCAL
Neutron Power
Reactor Protection Sys A
(A) (C)
MCR
Trip Trip Trip Trip
Engineered Safety Featur
Interlock Logic
Safety Act
Redundant Outpu
DHR Valves (Int)
RB Cooling System
Reactor Bldg Isol.
Reactor Bldg Spray
Hi-Lo Press Inj.
MCC MCC MCC
Rod Withdrawal Inhibit
Sensor Inputs
Plant Sta.
Frequency
Megawatts
(C) MCR
Integrated Control System B
Boron Feed And Bleed Controller
Turbine Control
Feedwater Control
Reactor Control
Insert Rods
Withdraw Rods
Int.
Status
Control Rod Drive Control Sys C
(A)
MCR
Feed & Bleed Permissive (Int)
Throttle Vlvs
Turb Limits
O. Throt Vlvs
C. Throt Vlvs
Trans. To Man.
Turbine By-pass Vlvs
Main & St-up FW Cont Vlvs
Main FW Pumps
TURBINE EHC
E/P MCC
INT
(C) (I&A)
(I&A)
CRD Operator's Console
Controlling Group Position
Inputs From SRCI
Incore Monitoring System
Rod Positions
Background Flux
Local Neutron Flux
Thermocouples
Selected Local
Non-Nuclear Instrumentation B
Sensor Inputs (I&A)
Plant Computer
C
Re
De
(I) (A) (C & INT) (C)
MCR & LOCAL
MCR
MCC
MCR
(A)
Core Power Distribution
Nuclear Calculations
Thermal-Hydraulic Calculations
Plant Status
MCR
Operator

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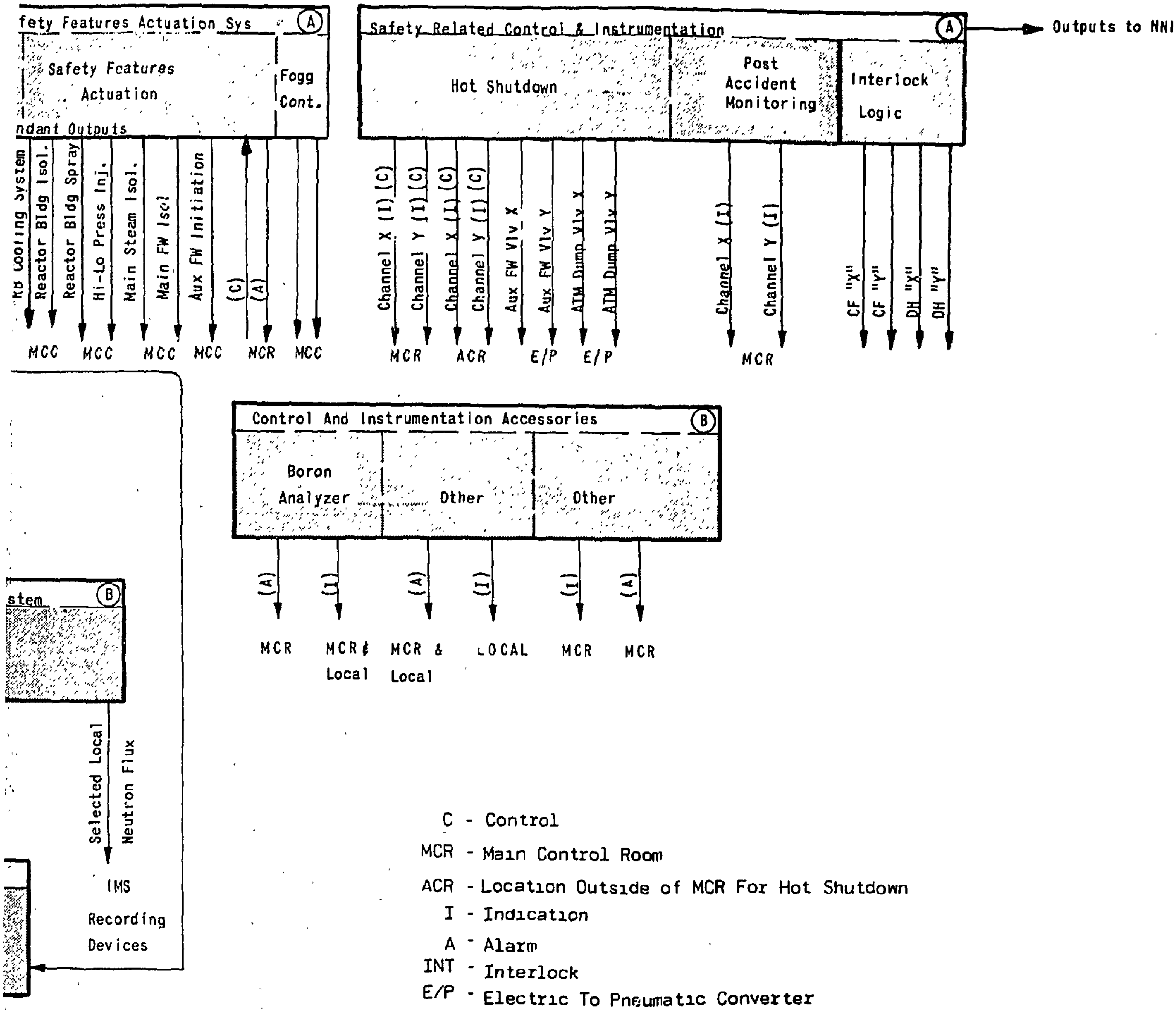
Figure A-21. Typical Interrelation of B&W Instrumentation, Control, and Protection System



C - Control
MCR - Main Control Room
ACR - Location Outside of MCR For Hot Shutdown
I - Indication
A - Alarm
INT - Interlock
E/P - Electric To Pneumatic Converter
MCC - Motor Control Center
(A) - Local/Inline Sensors (Qualified)
(B) - Local/Inline Sensors (Unqualified)
(C) - Pumps, Motors, NSS Component Sensors

2

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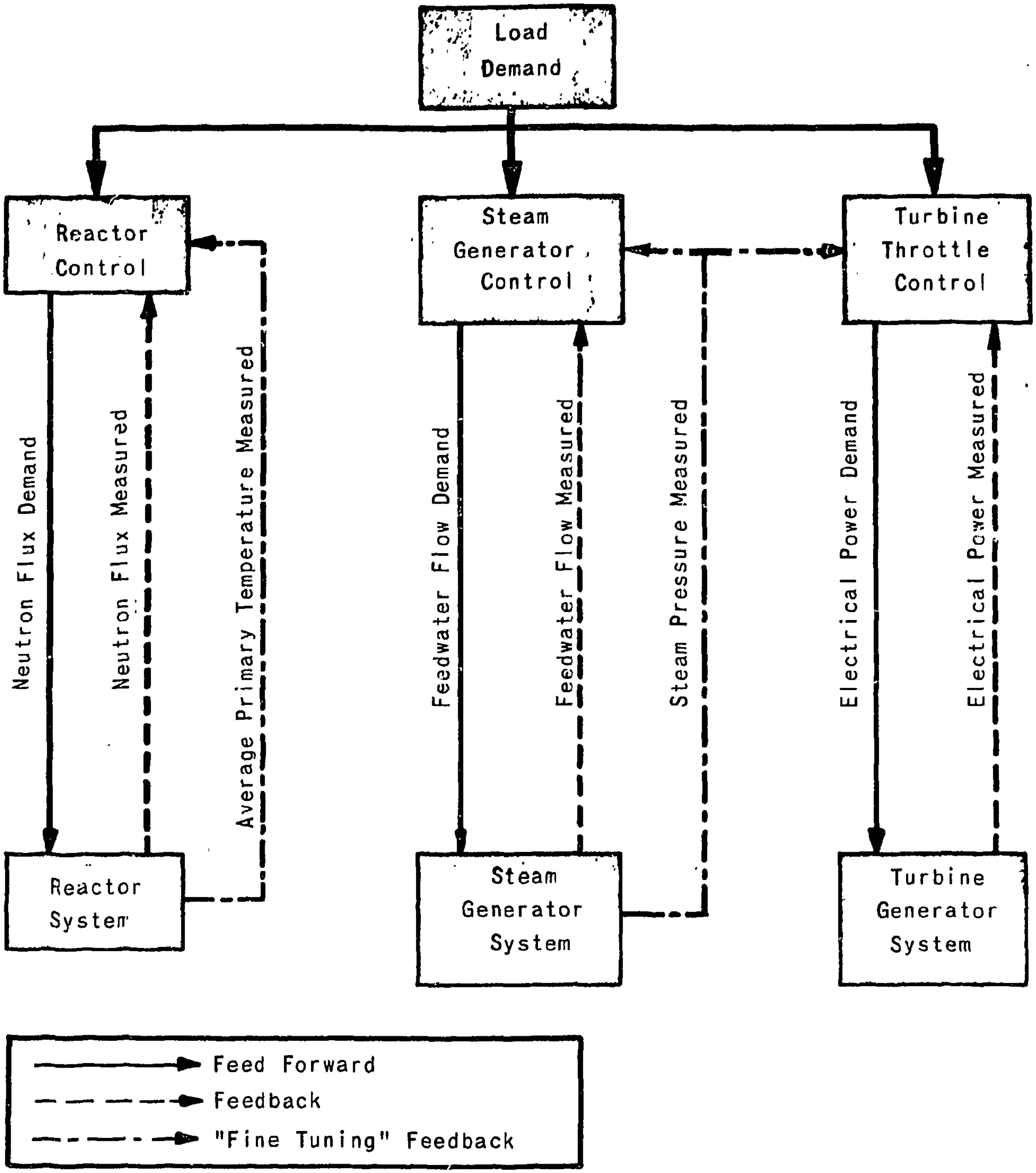
Figure A-22. Integrated Control System Concept



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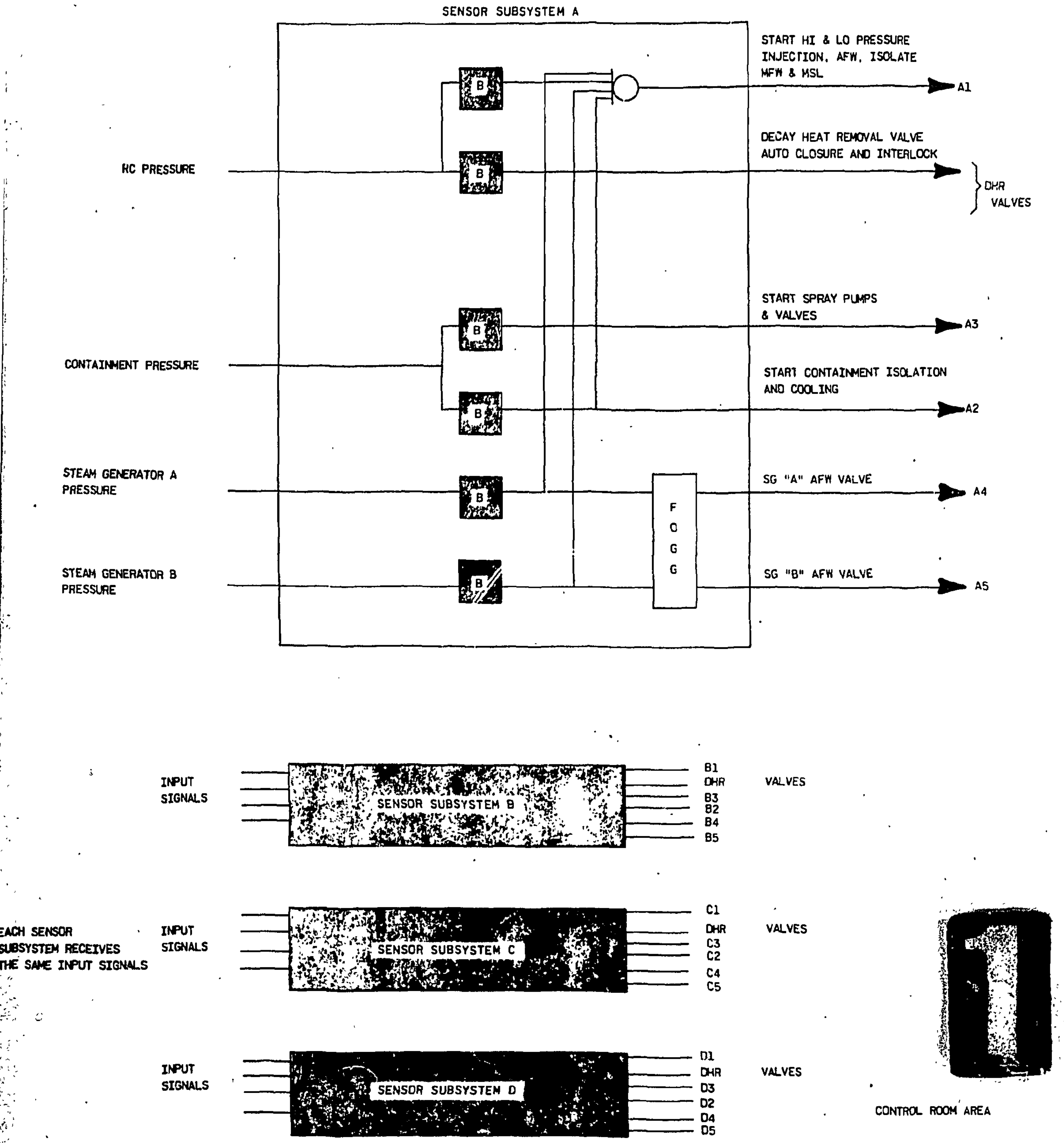
SENSOR SUBSYSTEM A
START HI & LO PRESSURE INJECTION, AFW, ISOLATE MFW & MSL
A1
RC PRESSURE
B
DECAY HEAT REMOVAL VALVE AUTO CLOSURE AND INTERLOCK
DHR VALVES
START SPRAY PUMPS & VALVES
A3
CONTAINMENT PRESSURE
START CONTAINMENT ISOLATION AND COOLING
A2
STEAM GENERATOR A PRESSURE
SG "A" AFW VALVE
A4
F O G G
STEAM GENERATOR B PRESSURE
SG "B" AFW VALVE
A5
INPUT SIGNALS
SENSOR SUBSYSTEM B
B1
DHR
B3
B2
B4
B5
VALVES
EACH SENSOR SUBSYSTEM RECEIVES THE SAME INPUT SIGNALS
INPUT SIGNALS
SENSOR SUBSYSTEM C
C1
DHR
C3
C2
C4
C5
VALVES
INPUT SIGNALS
SENSOR SUBSYSTEM D
D1
DHR
D3
D2
D4
D5
VALVES
CONTROL ROOM AREA

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Figure A-23. Engineered Safety Feature Actuation System – Block Diagram

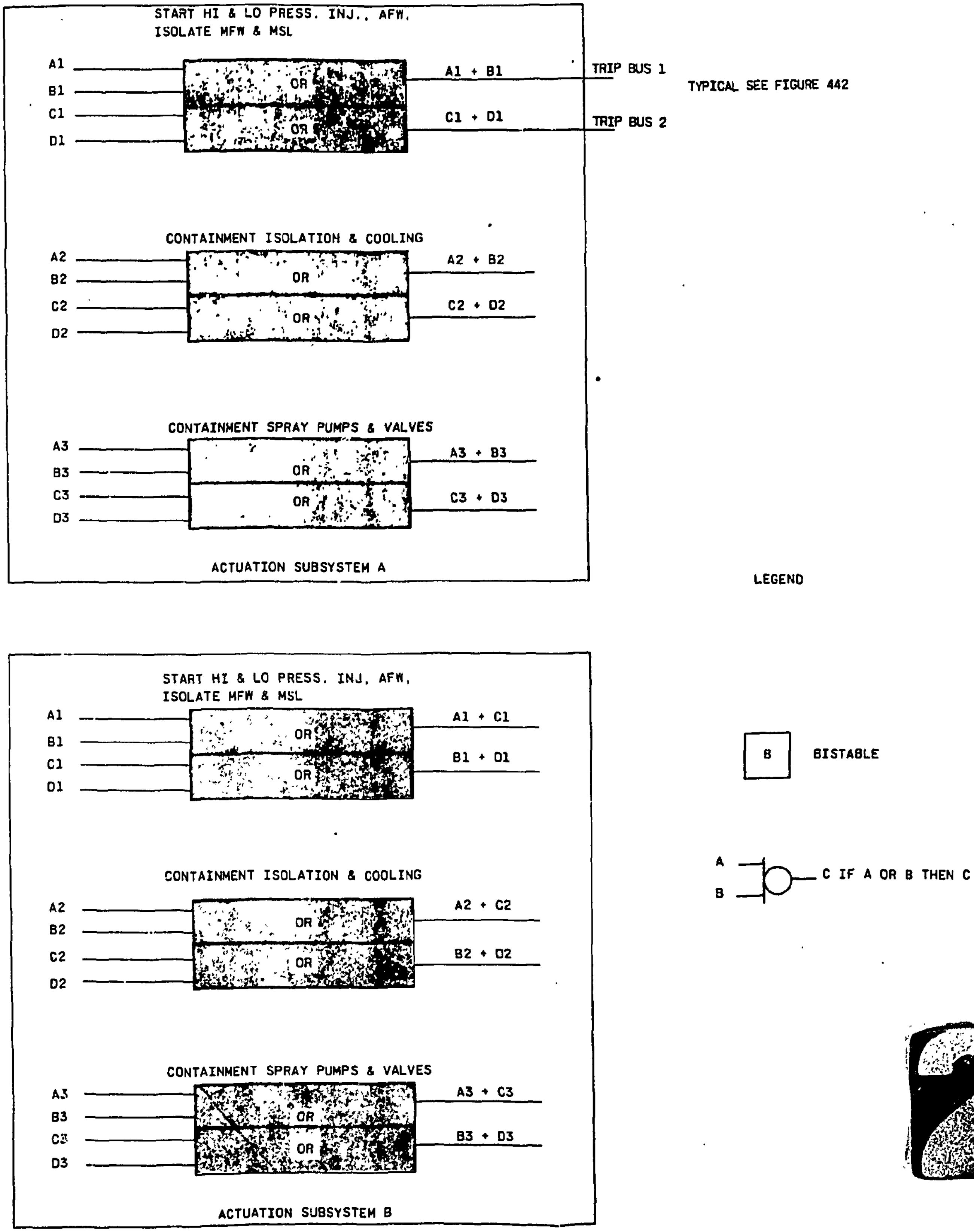

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Figure A-24. Control Rod Drive Control System Diagram

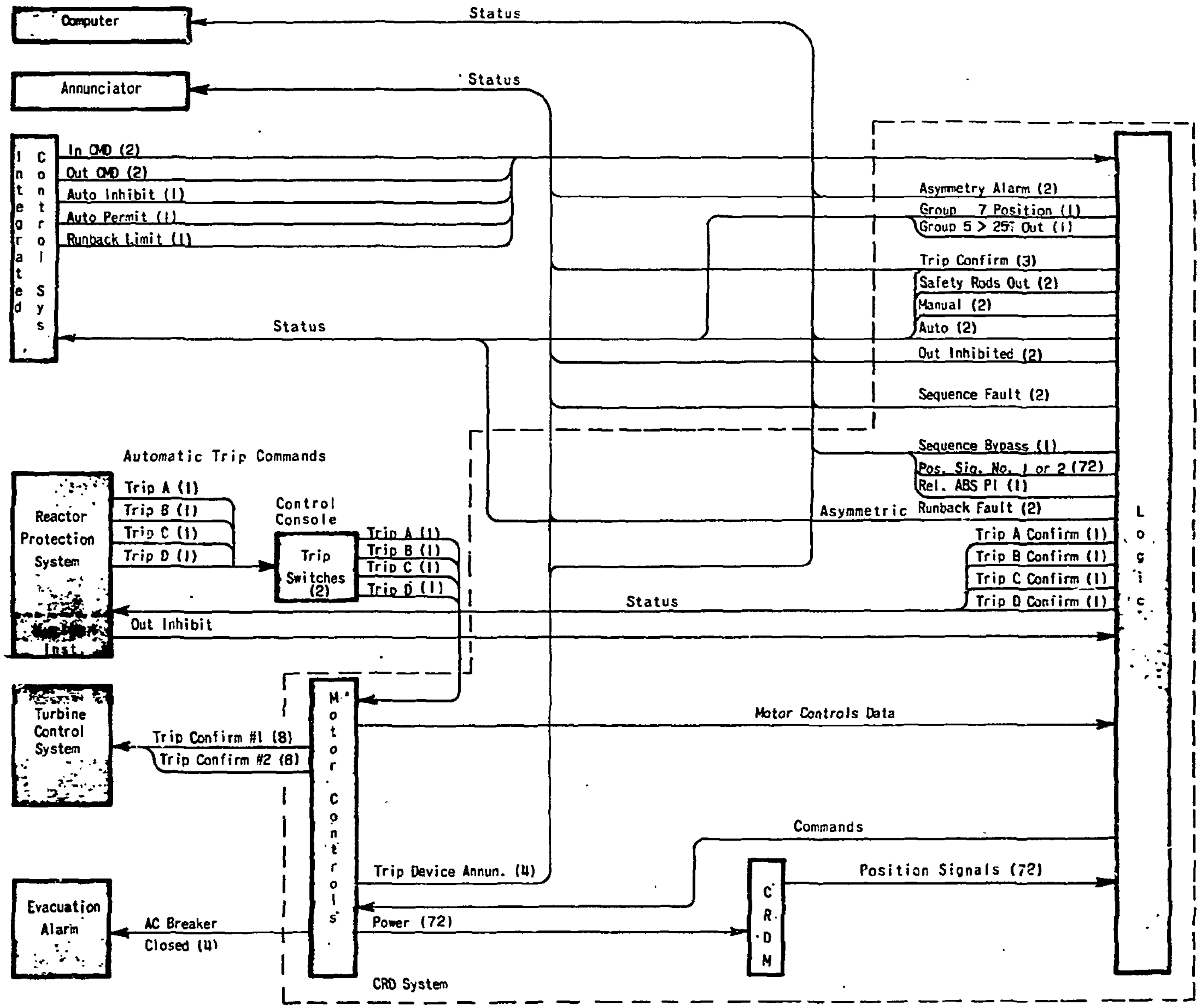

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Figure A-25. Fuel Handling System — Plan View

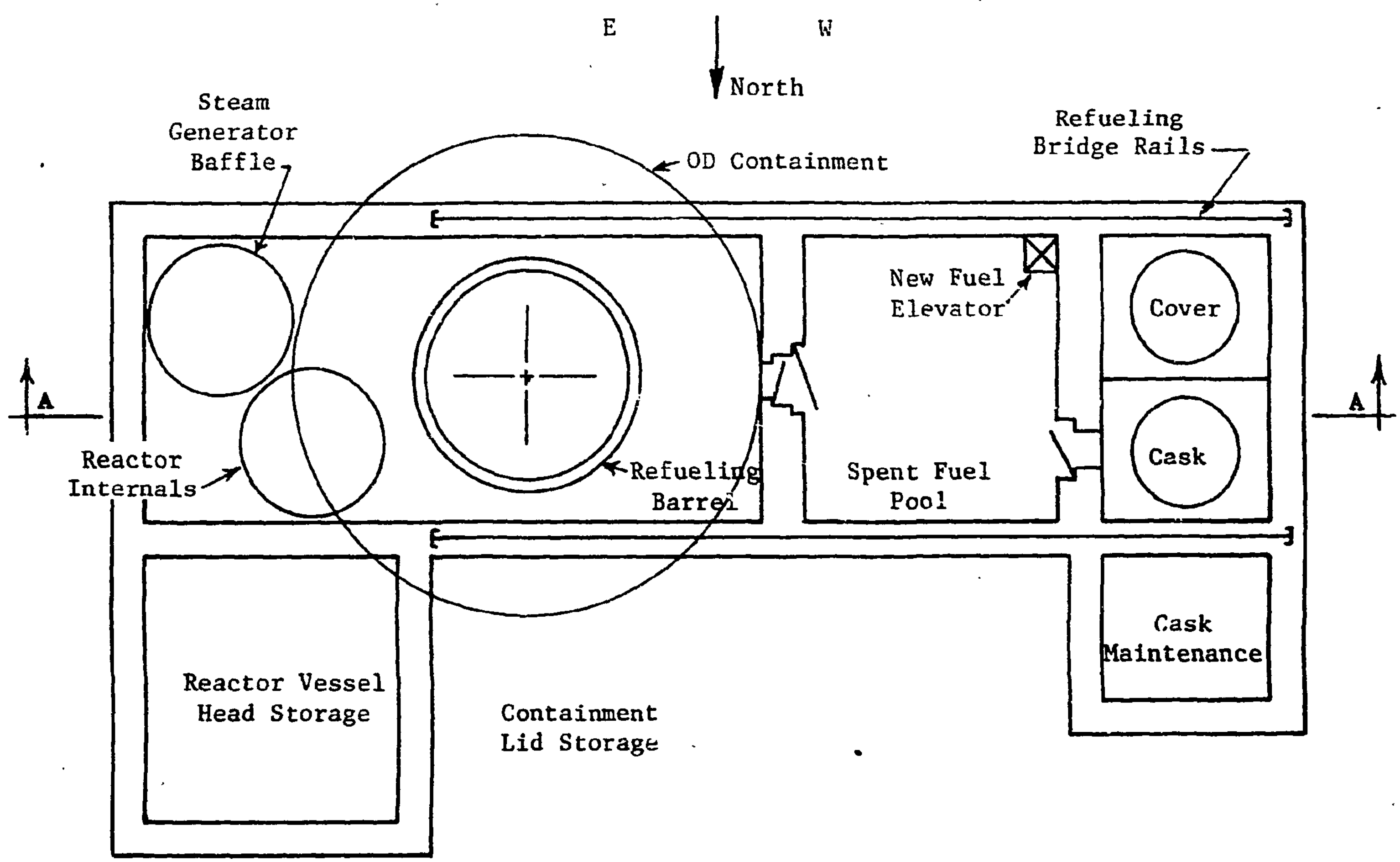


Figure A-26. Fuel Handling System — Section AA

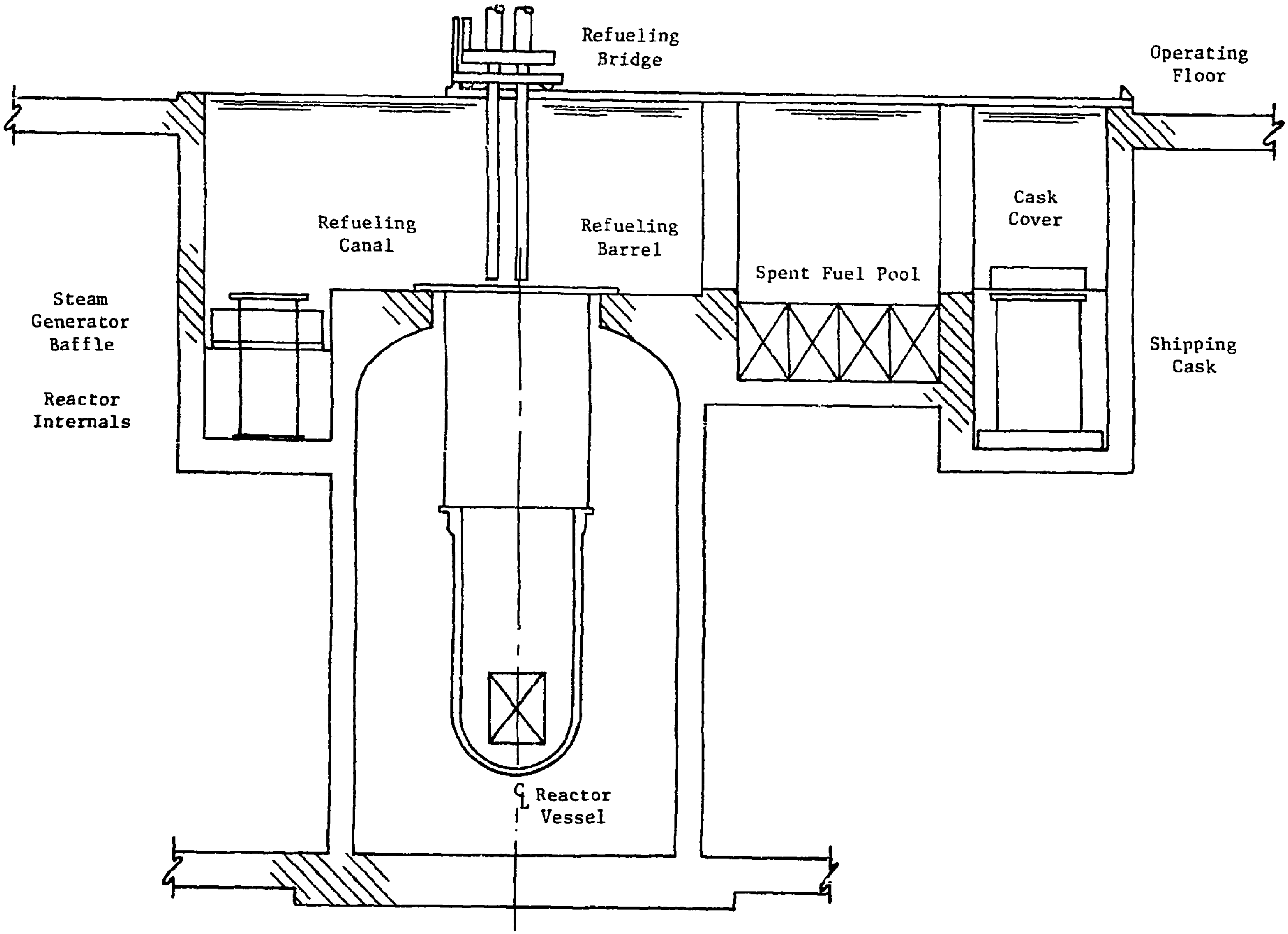
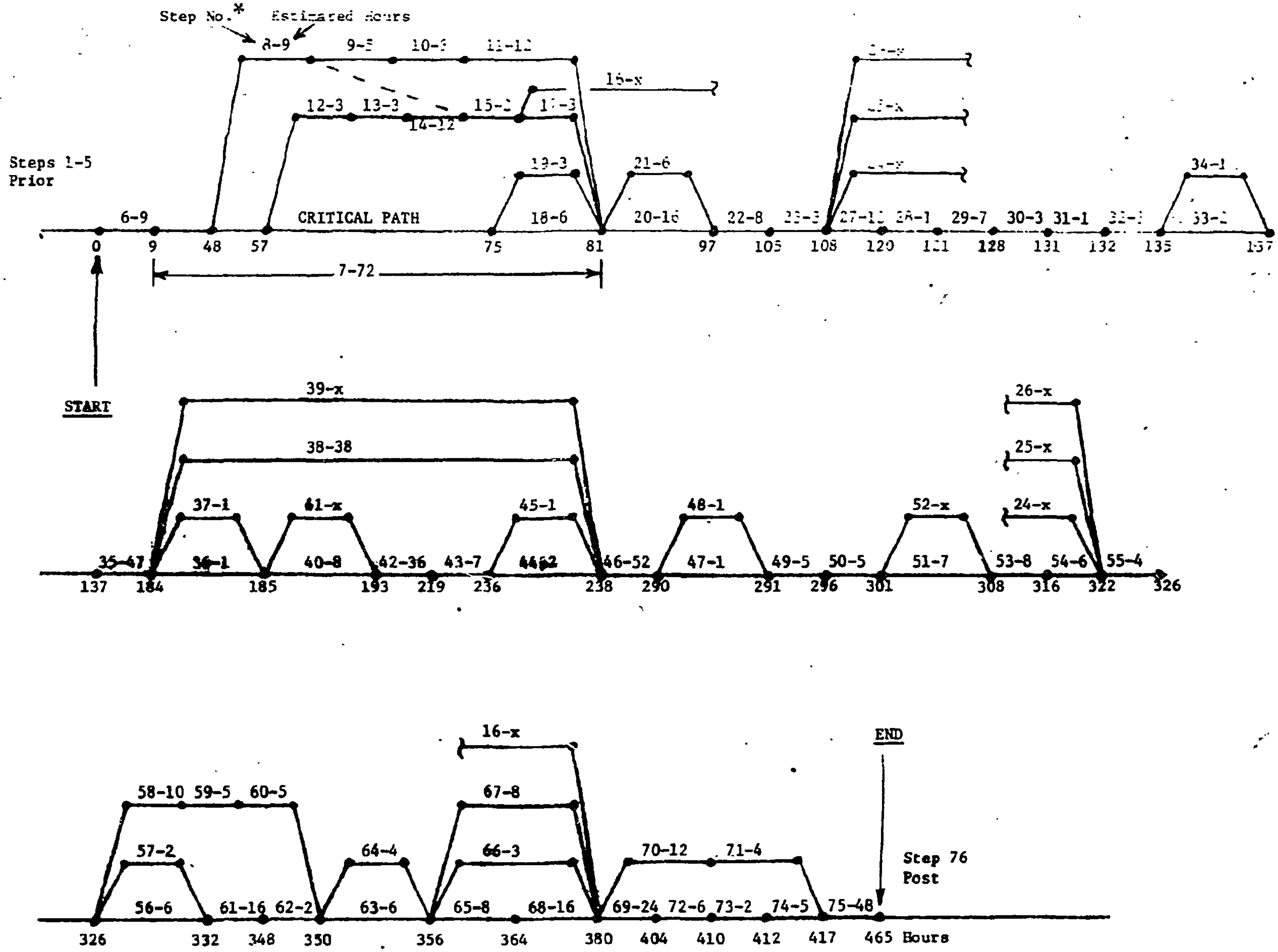


Figure A-27. Critical Path Schedule — Wet Refueling, Land-Based Plant



*See Table 3-3 for a description of each step for land-based wet refueling.

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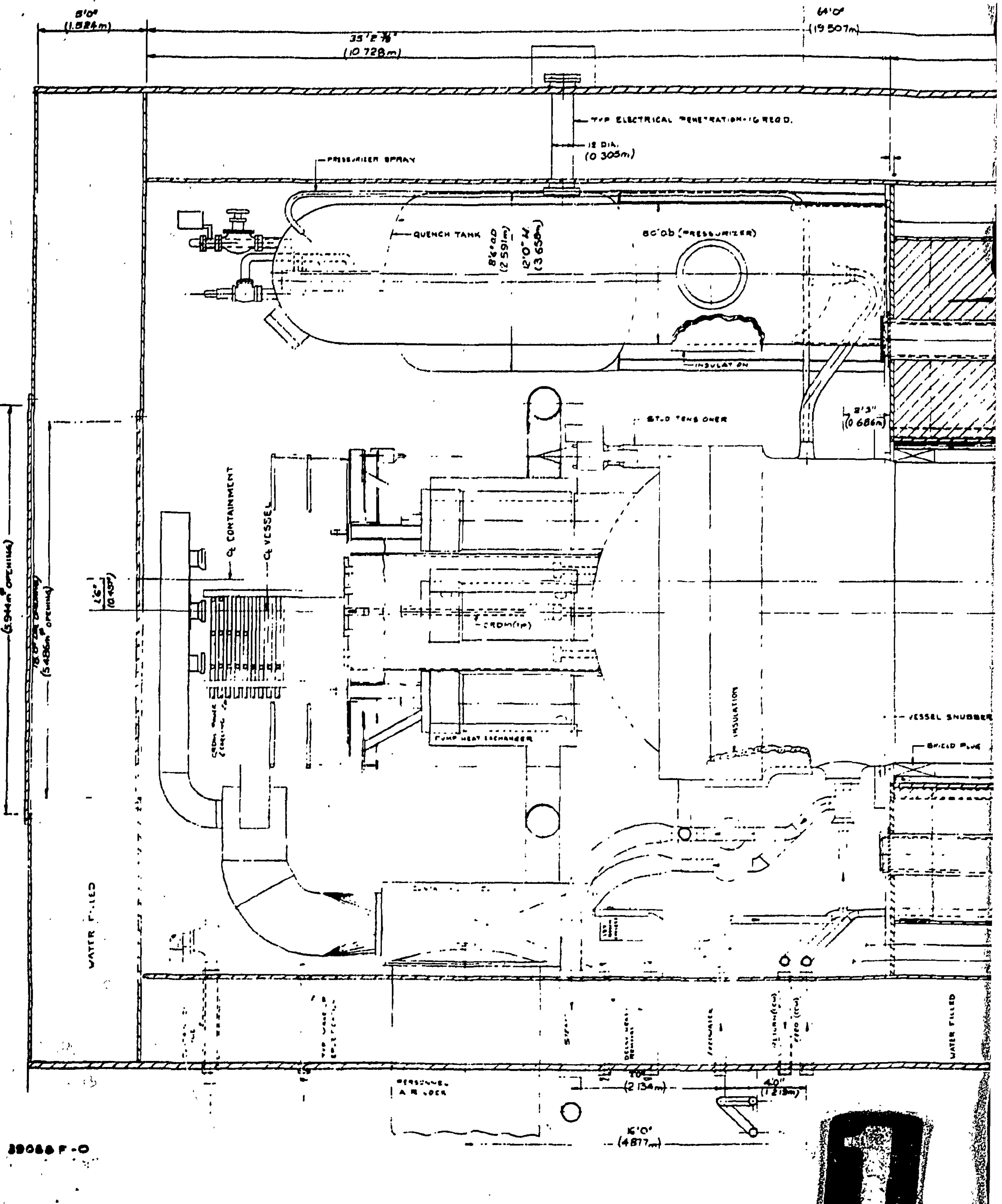
5'0"
(1.524m)
35'2 7/8"
(10.728m)
64'0"
(19.507m)
TYP ELECTRICAL PENETRATION-16 REQ'D.
12 DIA.
(0.305m)
PRESSURIZER SPRAY
QUENCH TANK
8'6" OD
(2.591m)
12'0" W
(3.658m)
INSULATION
STUD TENSIONER
2'3"
(0.686m)
℄ CONTAINMENT
℄ VESSEL
1'6"
(0.457m)
19'6" DIA. OPENING
(5.944m DIA. OPENING)
18'0" DIA. OPENING
(5.486m DIA. OPENING)
CRDM(TYP)
PUMP HEAT EXCHANGER
INSULATION
VESSEL SNUBBER
SHIELD PLUG
WATER FILLED
STEAM
FEEDWATER
WATER FILLED
PERSONNEL AIR LOCK
7'0"
(2.134m)
4'0"
(1.219m)
16'0"
(4.877m)

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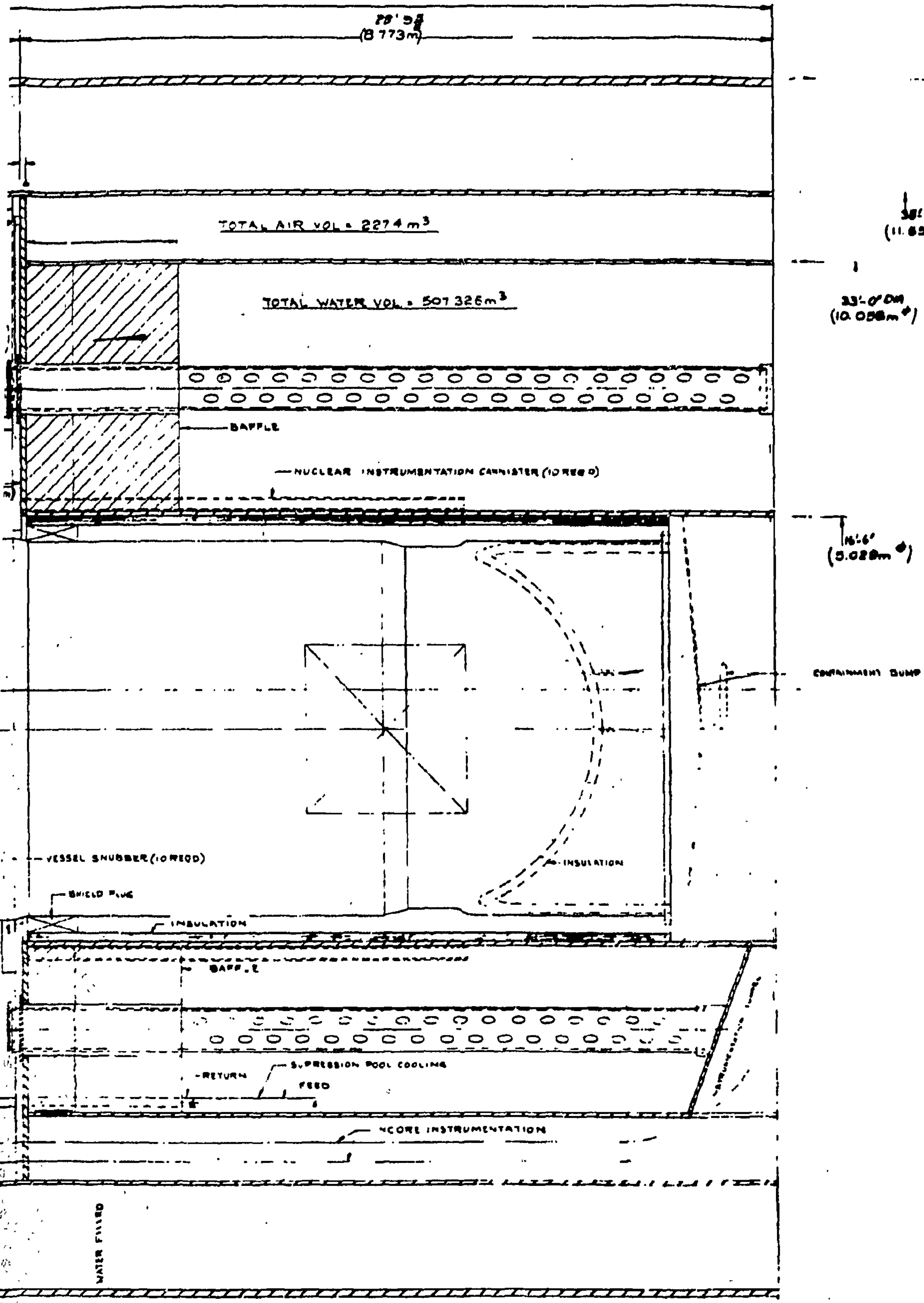
Figure A-28. Containment



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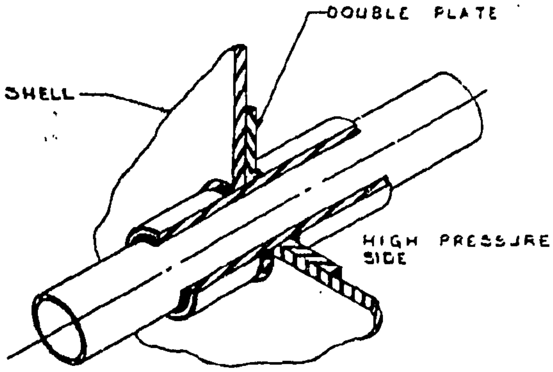
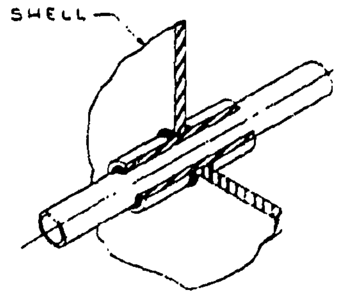
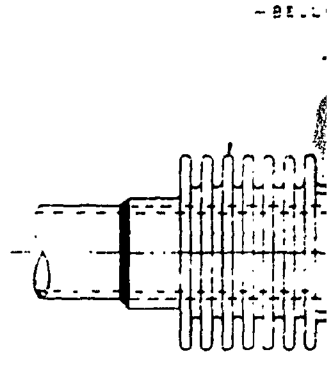


LARGE (2" UP)

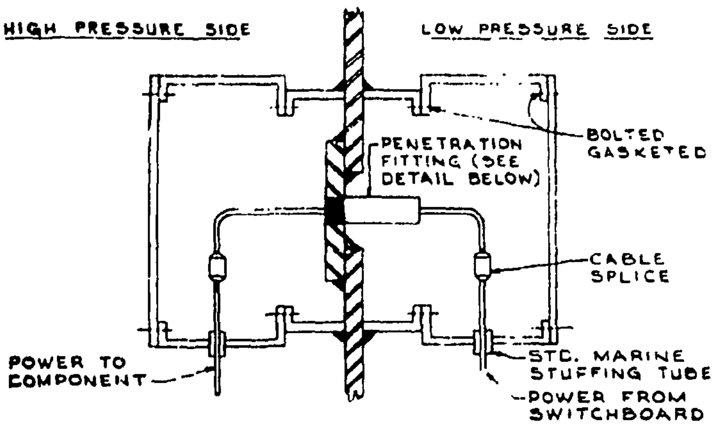


SMALL (UP TO 2")

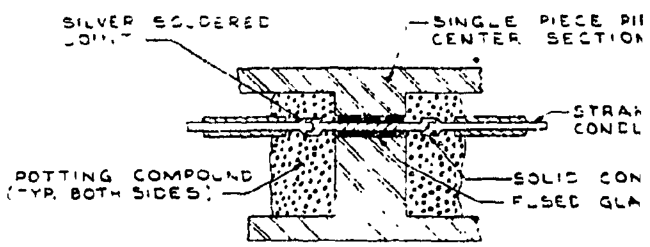


LARGE (2

TYPICAL PIPE PENETRATIONS (COLD)



CONTAINMENT ELECTRICAL
PENETRATION



DETAIL 'A'

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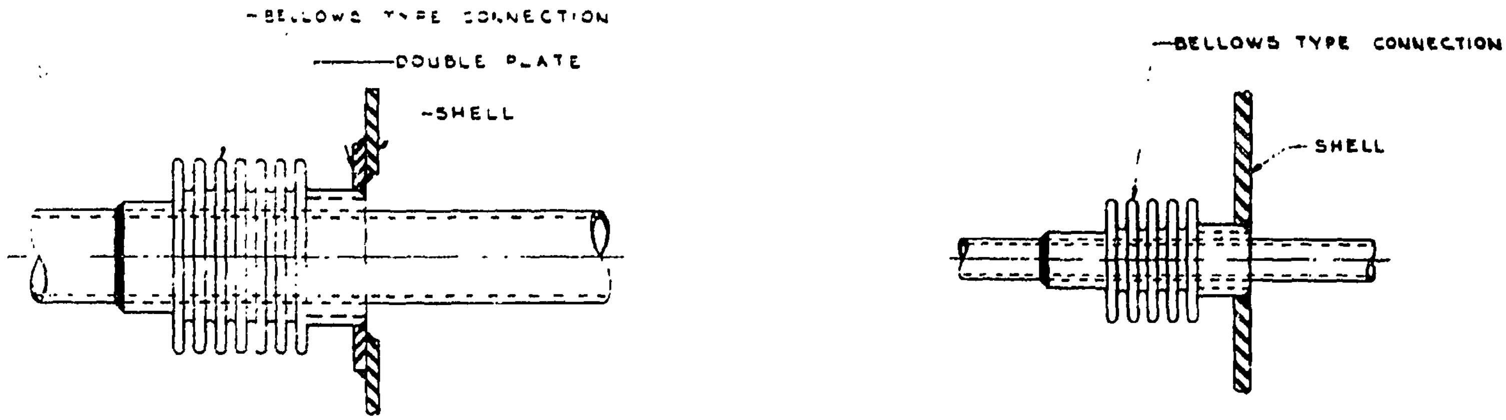
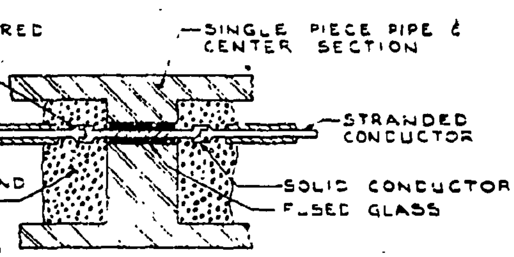
Figure A-29. Typical Electrical and Piping Penetrations



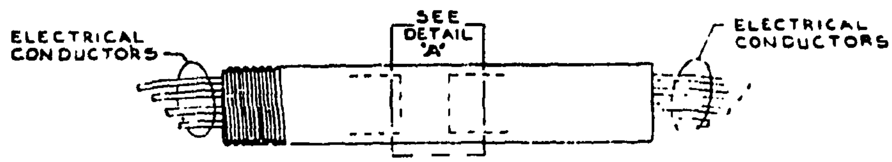
LARGE (2" UP)

SMALL (UP TO 2")

TYPICAL PIPE PENETRATIONS (HOT)



DETAIL 'A'



TYPICAL PENETRATION FITTING

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The PE-CNSG plant construction schedule is shown in Figure B-1. Reactor vessel delivery is scheduled 24 months after the start of work under a LWA. The LWA was assumed to be available six months before the receipt of a construction permit.

Certain buildings are critical to the overall success of the project schedule. The control building must be completed early to preclude delays in construction and to house equipment that is dependent on a temperature- and humidity-controlled atmosphere. The administration building should be completed in sufficient time to provide operating and maintenance people with a headquarters and classroom space to train for fuel loading; on a unit of this size, the administration building should be ready for occupancy nine months before the fuel is loaded.

The schedule for the reactor service building shows the situ times for certain operations and does not include off-site or on-site fabrication times; for example:

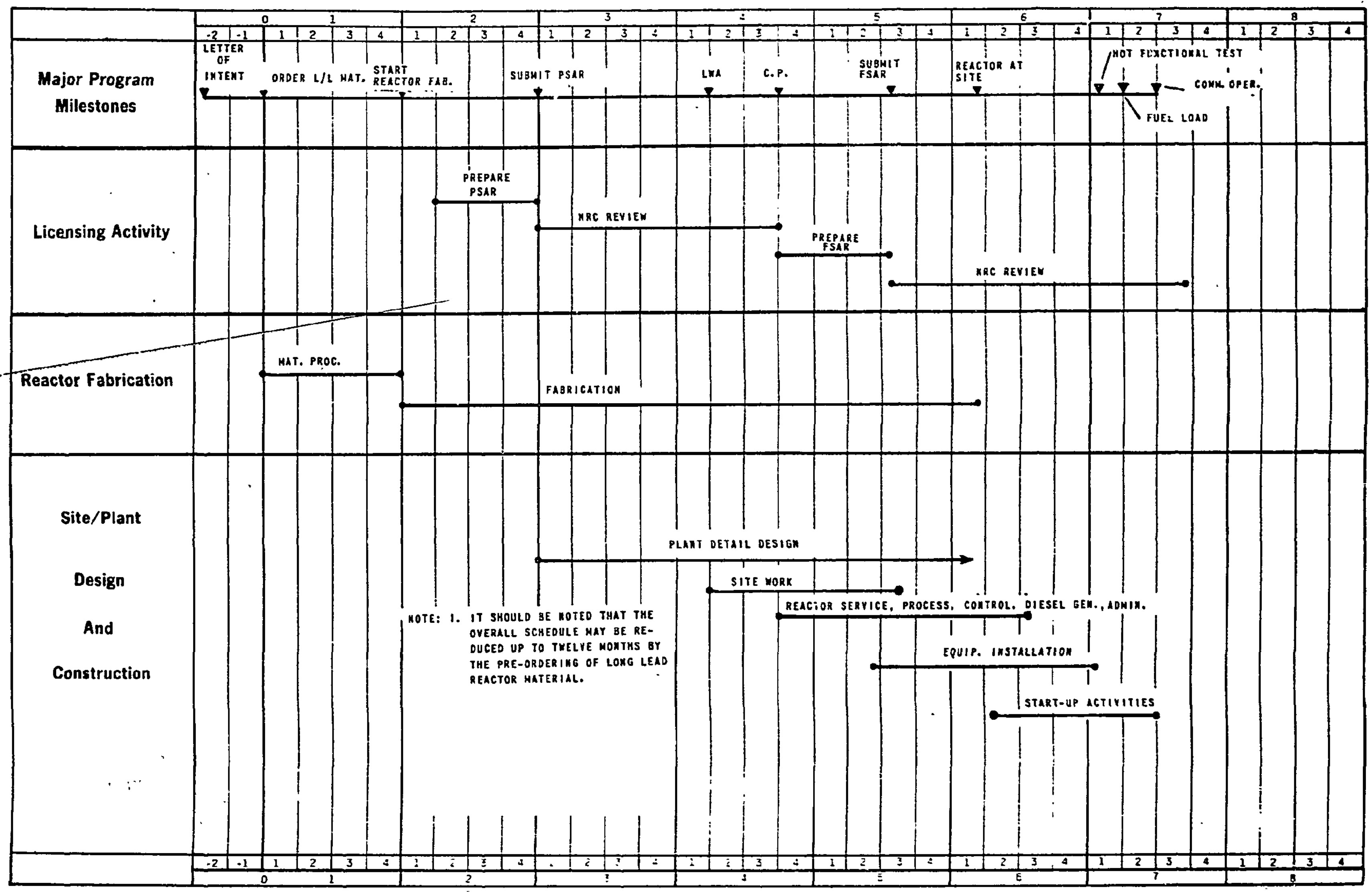
1. The baseplate is a single assembly including a section of shell already welded to the plate, the inner and outer suppression tank guard walls, and the machined base for the reactor vessel. The stress relieving of this assembly will be completed prior to shipment.
2. The reminder of the containment vessel (minus head) will be set in one piece to match the base previously set.
3. The internal containment work will be the installation of components fabricated and/or assembled elsewhere.
4. The reactor vessel is one piece (minus head).
5. The reactor vessel internals are prefit to eliminate major field assembly problems.

The preliminary schedule is based on limited information and quantities; some buildings have been scheduled on the basis of volumes. More detailed and concise information would be required to permit the development of a comprehensive schedule.

Based on the information made available for this study, a 36-month schedule from LWA to fuel load is attainable. Assuming that the planned prefabrication of components can be demonstrated and deliveries of shop-fabricated equipment can be improved, the schedule could be further compressed for subsequent units.

The only potential delays for the PE-CNSG due to the remoteness of the Duval site are field installation of steam generators and field erection of the containment base plate rather than one-piece shipment. It is expected that these potential delays can be overcome and the schedule has not been modified in this study to reflect the potential delays.

Figure B-1. Duval Corporation Nuclear Plant Nth-of-a-Kind Construction Schedule



APPENDIX C
References

1 A Small Pressurized Water Reactor for Process Energy — Plant Costs and Design Studies, BAW-1428, Babcock & Wilcox, Lynchburg, Virginia, June 1976 (ORNL-Sub-4390-2).

2 Guide for Economic Evaluation of Nuclear Reactor Plant Designs, NUS-531, NUS Corporation, Rockville, Maryland, January 1969.

3 "Subsoil Investigations and Consultation on Foundations — Water Reservoir and Site Grading for Proposed Duval Sulphur Plant West of Orla," Woodward-Clyde Associates for Stearns-Roger Corporation, Culberson County, Texas, September 30, 1968.

4 "Additional Consultation Ground Water Storage Reservoir Duval Sulphur Plants," Woodward-Clyde Associates for Stearns-Roger Corporation, Orla, Texas, March 17, 1969.

5 Reactor Site Criteria, 10 CFR 100.

6 R. A. Hedrich, J. J. Cudlin, and R. C. Foltz, CRAFT2 — Fortran Program for Digital Simulation of a Multinode Reactor Plant During Loss of Coolant, BAW-10092, Rev 2, Babcock & Wilcox, Lynchburg, Virginia, April 1975.

7 Assumptions Used for Evaluating the Potential Radiological Consequences of a Loss of Coolant Accident for Pressurized Water Reactors, USNRC Regulatory Guide 1.4, Rev 2, June 1974.

8 R. E. Clark to D. C. Schluderberg, Memorandum, "Product Development — Interest During Plant Construction," Babcock & Wilcox, Lynchburg, Virginia, October 27, 1976.

9 "A Coal-Fired Steam Generating Plant for the Radford Army Ammunitions Plan," United Engineers and Constructors, Inc., Philadelphia, Pennsylvania, September 1976.

[10] B. W. Podhurst and J. A. Garner, CYCO/FUPAC — Nuclear Fuel Cycle Economics Code, BAW-415, Rev 1, Babcock & Wilcox, Lynchburg, Virginia, March 1974.

[11] T. R. Stauffer, H. L. Wycoff, and R. S. Palmer, "An Assessment of Economic Incentives for the Liquid Metal Fast Breeder Reactor," No. 75-WA/ME-5, ASME Annual Winter Meeting (1975).

[12] Licensing of Production and Utilization Facilities — General Design Criteria for Nuclear Power Plants, 10 CFR 50 (Appendix A).