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MASTER

HIGH TEMPERATURE GAS-COOLED REACTOR

GAS TURBINE APPLICATION STUDY

GAS-COOLED REACTOR ASSOCIATES
LA JOLLA, CALIFORNIA

DECEMBER 1980

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Gas-Cooled Reactor Associates
3344 N. Torrey Pines Court, Suite 300
La Jolla, California 92037

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Gas-Cooled Reactor Associates
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EXECUTIVE SUMMARY

1.0 INTRODUCTION

The high-temperature capability of the High-Temperature Gas-Cooled Reactor (HTGR) is a distinguishing characteristic which has long been recognized as significant both within the U.S. and within foreign nuclear energy programs. This high-temperature capability of the HTGR concept leads to increased efficiency in conventional applications and, in addition, makes possible a number of unique applications in both electrical generation and industrial process heat.

In particular, coupling the HTGR nuclear heat source to the Brayton (gas turbine) Cycle offers significant potential benefits to operating utilities. Three major fundamental advantages over the more conventional Rankine, vapor-turbine power cycle have been cited: (1) heat is rejected from the cycle at a relatively high temperature (approximately 150°C or about 300°F), making it more economically attractive to employ dry or peak-shaved, wet/dry cooling or, alternatively, to achieve higher efficiency by the addition of a bottoming power cycle when wet cooling is available; (2) the gas turbine possesses inherent capability to achieve relatively larger increases in power output and efficiency with increasing temperature than the Rankine Cycle, thereby better utilizing the higher temperature capabilities of high-temperature gas-cooled reactors; and (3) the low compression-expansion ratio helium turbomachine and modularized components arranged within a pressurized closed-cycle system result in a compact, integrated power conversion system with potential for reduced cost and high reliability.

While the unique capabilities of the HTGR in general and the gas turbine derivative in particular have engendered broad interest, this interest has tended to be diverse rather than focused and has not led to a fruitful National HTGR Program. This diversity in part reflects the broad application potential of the HTGR, which has led to differences among participating organizations with regard to the preferred program development path and program priorities. With the difficulties facing the continuation of the HTGR Program, it was evident that a focusing effort was required to reconcile the diversity of opinion and to obtain the concurrence and support of all HTGR Program participants with regard to program direction. GCRA, therefore, initiated the HTGR Lead Project Identification Plan in December 1979.

Central to the Plan was the investigation of HTGR Lead Project options which could provide a basis for a strong National HTGR Program. Working with the U.S. Department of Energy (DOE), General Atomic Company (GA), General Electric Company (GE), and Oak Ridge National Laboratory (ORNL), GCRA identified four HTGR Lead Project options for consideration. These four options--the Gas Turbine (HTGR-GT), the Reformer (HTGR-R), the Steam Cycle/Cogeneration (HTGR-SC/C), and the Nuclear Heat Source Demonstration Reactor (NHSDR)--were selected by the participants to encompass all potentially viable HTGR Lead Proj-

ects. The balance of the FY 1980 HTGR Program activities was adjusted to support these four Lead Project options, leading toward their evaluation, prioritization and, thus, sequencing within the HTGR Program. The ultimate result of these activities was envisioned to be an ordered HTGR Program, supported by all participants, which would logically evolve to a Lead Project commitment through a private sector initiative and subsequently provide for the follow-on development of other viable options.

This HTGR-GT Application Study documents the effort to evaluate the appropriateness of the HTGR-GT as an HTGR Lead Project. The scope of this effort included evaluation of the HTGR-GT technology, evaluation of potential HTGR-GT markets, assessment of the economics of commercial HTGR-GT plants, and evaluation of the program and expenditures necessary to establish HTGR-GT technology through the completion of the Lead Project.

2.0 SUMMARY OF APPLICATION STUDY RESULTS

The major finding of this study is that an HTGR-GT plant is not a suitable Lead Project candidate. While a substantial market exists for a reactor system with its characteristics, a program oriented to an HTGR-GT Lead Project cannot be recommended for the following reasons:

- Based upon the specific design and economic characteristics evaluated, no compelling economic advantage relative to existing and potential competition was identified as a result of this study. In particular, no substantial improvements in economics, performance or market application were identified over the HTGR-SC system.
- The projected costs of the design, development, and demonstration programs are large relative to the limited incentives of the HTGR-GT system. In addition, the design, development, and demonstration programs contain a significant element of cost and schedular risk associated with the resolution of key technical issues.
- The protracted schedule identified as necessary for the HTGR-GT lead plant would extend the date for introduction of commercial gas turbine plants and follow-on HTGR plants to 2010 and beyond.

It is recognized that the HTGR-GT plant has developmental potential beyond that considered in the scope of this study. Specifically, designs for higher temperature regimes and for the utilization of high quality reject heat via bottoming cycles or other means may lead to electricity generation efficiencies of 46% or higher. Associated with these projected gains, however, will be increased technical risk which must be addressed through development or experience. On balance, it is concluded that such systems are more prudently evolved as follow-ons to less demanding concepts. Accordingly, and given the evolving economic and institutional framework that exists in the energy supply indus-

tries, it is recommended that the HTGR-GT be viewed as a longer-term goal of the HTGR Program with reduced priority in the near term relative to predecessor options.

3.0 SUMMARY TECHNICAL DESCRIPTION

The components and systems of the Gas Turbine HTGR described below are shown in Figs. 1 and 2. The reactor core is cooled with pressurized helium, moderated and reflected with graphite, and fueled with a mixture of uranium and thorium. It is constructed of prismatic hexagonal graphite blocks with vertical holes for coolant channels, fuel rods, and control rods. Both the core and the two power conversion loops (PCL) are integrated in the multicavity prestressed concrete reactor vessel (PCRVR). The turbomachines are located in horizontal cavities in a chordal arrangement at an elevation below the core cavity. The other major PCL components--the recuperator and pre-cooler--are located in vertical cavities around the central core cavity. In addition to the PCL equipment, three core auxiliary cooling system (CACS) loops are also provided for safety-related core cooling capability. The CACS loops, the PCL equipment, and the core are connected by a series of ducts internal to the PCRVR. The internal surfaces of the PCRVR cavities and ducts are lined with an impermeable steel membrane and covered with a thermal barrier to limit system heat losses and at the same time maintain liner and concrete temperatures within design limits.

The 850°C (1562°F) core outlet gas drives the turbine end of the non-intercooled turbomachine which, in turn, powers the compressor and a 400 MWe, 60 Hz generator. The HTGR-GT utilizes a recuperator to increase system efficiency and to reduce heat rejection through the pre-cooler. The pre-cooler is a helium-to-water heat exchanger which rejects cycle waste heat to the plant cooling system, or potentially a bottoming power cycle. Depending on the particular site conditions, the cooling system may utilize all dry or a combination of dry and wet cooling towers to reject heat to the atmosphere.

The PCRVR and ancillary systems are enclosed within a secondary containment building. This containment, together with the PCRVR, incorporates safety features that limit the release of fission products in the event of a leak in the primary system boundary. Certain nuclear heat source related systems, such as fuel handling and helium purification, and most balance-of-plant systems and equipment are located outside the secondary containment in separate structures. Among the plant structures envisioned are the reactor service building, the controls, auxiliaries and diesel building, and the fuel storage building.

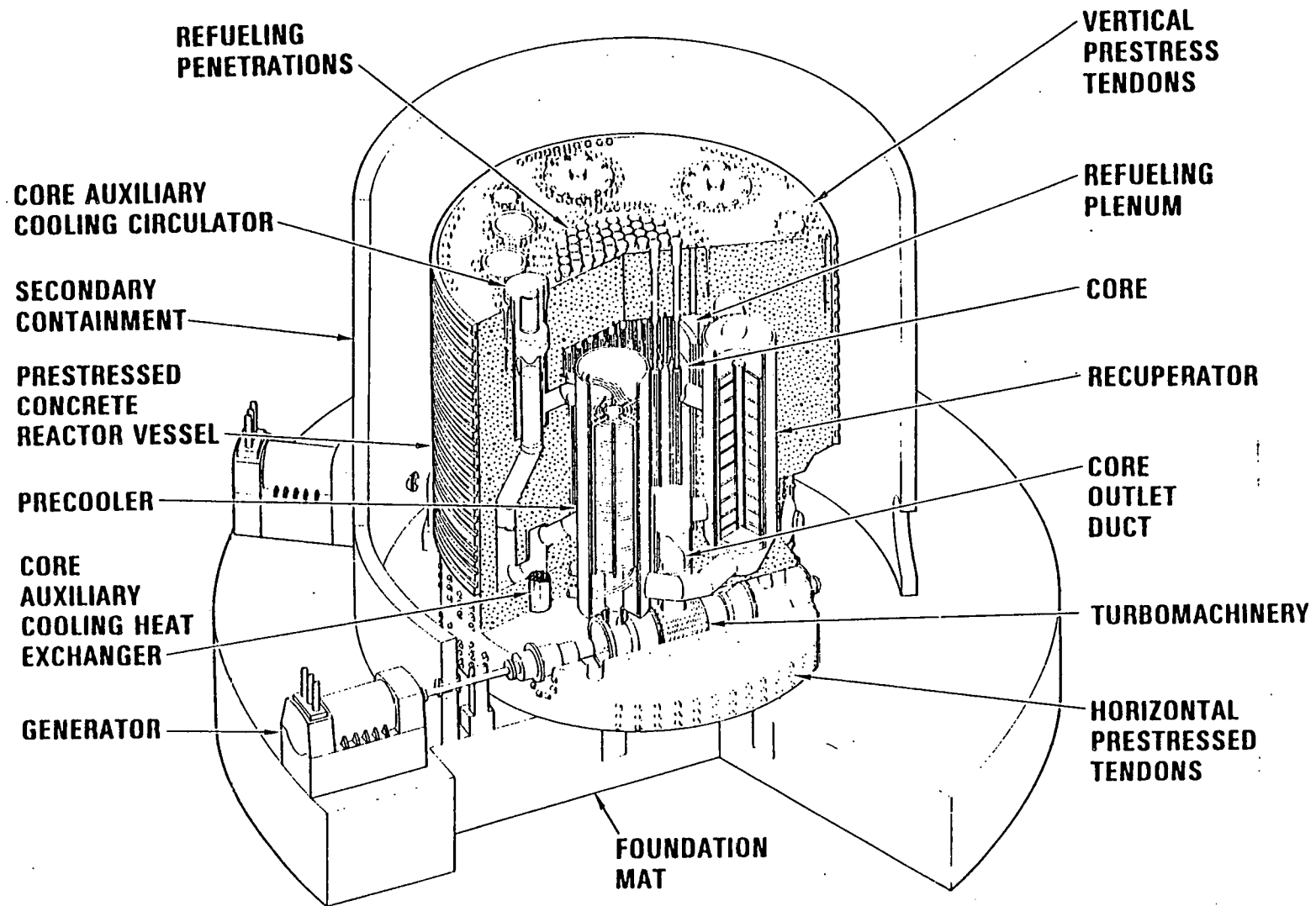


Figure 1 Two-loop 2000-MW(t) HTGR-GT power plant

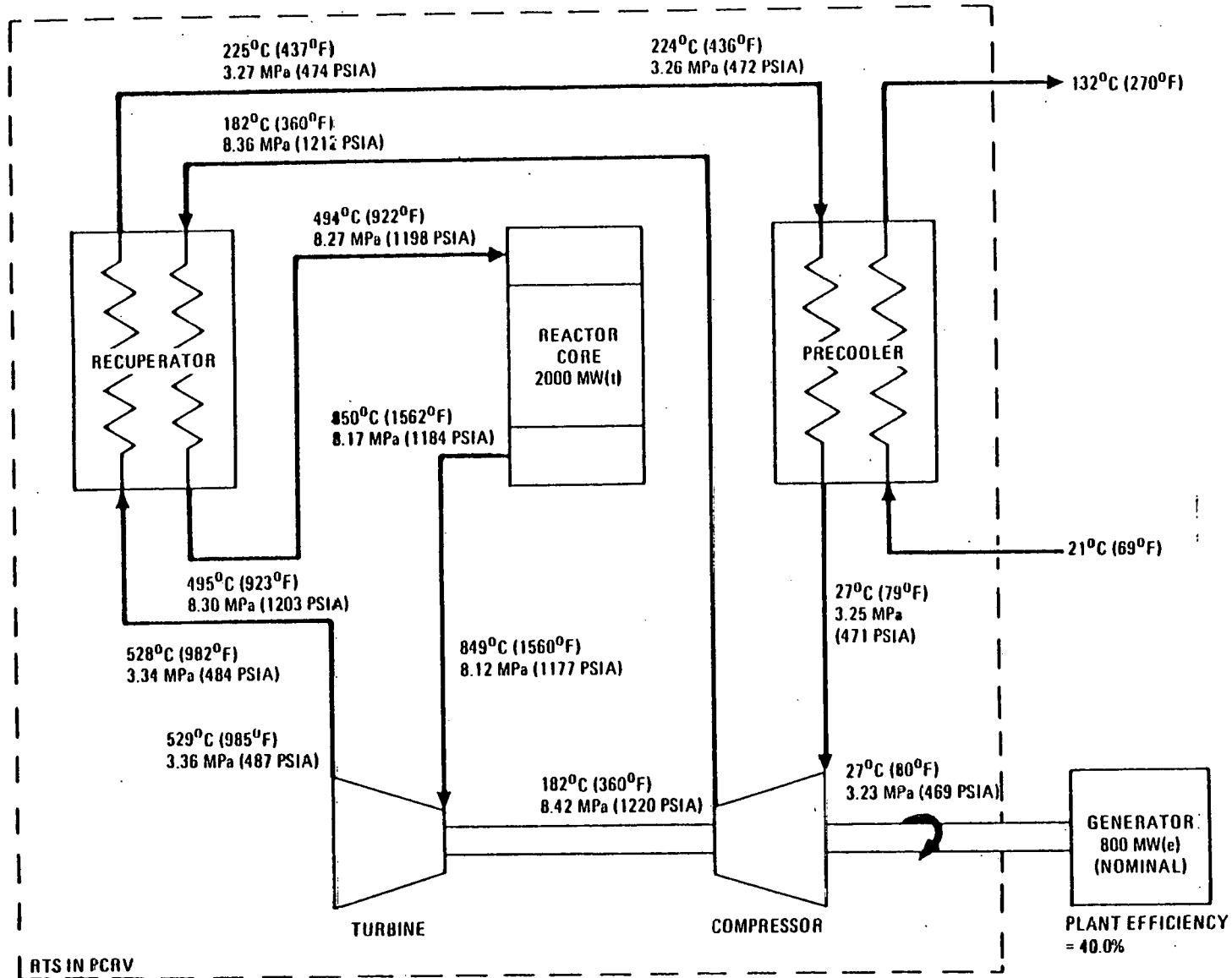


Figure 2 HTGR-GT cycle diagram

The major parameters of this design are as follows:

Nominal reactor power, MW(t)	2000
Nominal electric output (net), Mw(e)	800
Nominal net station efficiency, %	40.0
Power conversion loops, qty	2
Plant layout	Single unit
Availability factor, %	90
Capacity factor, %	80
Type of cooling	Peak-shaved wet/dry
Fuel cycle	LEU/Th throw-away
Core power density (KW/liter)	6.7
Fuel lifetime, years	4
Refueling cycle time, years	1

4.0 SUMMARY OF EVALUATIONS

During the course of this assessment, the incentives and issues concerning the HTGR-GT plant have been evaluated. The observations and conclusions drawn by program participants from this work may be characterized in terms of (1) the market forces influencing the deployment of this system; (2) the technical effort projected as necessary to establish this concept as a viable industrial power plant; and (3) costs projected to support the development program and Lead Project. These are discussed in the following sections.

4.1 Market Incentives

For several years, characteristics of the HTGR-GT plant have lead HTGR designers and interested utilities to anticipate a favorable market entry. Expected attributes of the HTGR-GT are:

- A high reject heat temperature which favors optimal dry or wet/dry cooling and the siting of plants in areas of limited water availability for such purposes.
- The elimination of several costly systems within the balance of plant which was expected to result in an overall simplification of the plant with an attendant high degree of plant availability relative to competing systems.
- A reactor system with response characteristics which potentially provide for increased investor protection, reductions in the level of personnel exposure to radioactive environments, and reduced reliance on precipitous operator action to reactor excursions and changes in operating modes.
- A significant improvement relative to competing systems when the combined effects of plant economics and performance are considered.

In confronting the first of these considerations, GCRA and Hanford Engineering Development Laboratories (HEDL) conducted a joint study to quantify the projected market. In the time interval of interest for commercializing the HTGR-GT (year 2000 to 2020), it was projected as a result of the study that 850 GWe of generating capacity will be constructed. Of this, the nuclear portion of the market was projected to be approximately 430 GWe. This study also examined the size of market anticipated in areas of restricted water usage. As a result, over one-fourth of the expected market for new and replacement plants was projected to face significant limitations on the availability of water for the purposes of cooling power plants. Thus, the initial perceptions of interested utilities relative to the size and importance of this market were confirmed.

The issue of plant availability proved to be somewhat more obscure in that claims of performance superior to existing systems are contingent upon two key aspects of the development program. These are:

1. The reliability to be assigned to the future performance of developmental components which rely on significant extensions of existing industrial practice, and
2. Operating and maintenance plans which require sufficient advancement of the design to accurately assess the procedures and facilities necessary to support the direct cycle gas turbine and to identify the need for spare units.

While the observation that the mature gas turbine plant should attain high levels of availability still appears generally accurate, justification of levels higher than those assigned to competing systems for the purposes of economic comparisons was not achieved in the course of this study.

A number of advantages accrue to the gas turbine which are shared in common with other HTGR concepts. Ongoing activities within the technology development programs continue to support the expectation that inherent features of the HTGR (ceramic-based core, high thermal inertia, etc.) will permit operator response times measured in terms of hours for operational events. Analyses and tests also show very low activity levels reaching operating personnel and/or the public as a result of normal operation, maintenance, and postulated accidents. These characteristics, which are confirmed by the operating experience of Fort St. Vrain, are expected to result in increased siting flexibility, reduced difficulty in licensing, improved plant operability, and reduced maintenance costs. While these advantages lend further credence to HTGR technology, no basis for assigning an economic advantage to the HTGR-GT relative to other HTGR concepts was found as a result of this study.

As noted, designers and users had expected a more favorable economic prognosis for the HTGR-GT. It was anticipated that an overall simplification of the plant (conceptually, the elimination of steam genera-

tors, condensate and feedwater trains, piping, and the turbine building) would result in significant capital cost improvements relative to steam cycle plants. Simply stated this study has shown that these savings were less than the costs associated with locating the gas turbine power conversion loop inside the PCRV which resulted in nearly doubling the primary coolant boundary (liner and thermal barrier exposed to primary helium) with consequent increases in PCRV and containment size as compared to HTGR-SC plants. In addition, provisions for maintaining the gas turbine include a building in the balance of plant which costs about the same as a conventional steam turbine building.

It was also expected that the costs of dry-cooling competing systems would provide significant margin for market penetration by the HTGR-GT in that sector. The results to date indicate that while the associated cooling cost advantage of the HTGR-GT is real, the magnitude does not offset the higher cost of the base plant. Consequently, when comparing evaluated power generation costs for plants of optimal configurations, the HTGR-GT does not offer any significant gains over the HTGR-Steam Cycle (SC), even with dry cooling considerations taken into account. In summary, therefore, economic incentives for the HTGR-GT are inadequate to support a recommendation for its early deployment.

4.2 Design and Development Requirements

The development of the HTGR-GT represents a significant extension of the technology required for HTGR-SC/C plants. Schedules showing design and development activities through to completion of the Lead Project were developed in the course of this study. Assuming an aggressive program, development of the HTGR-GT may be viewed in three phases:

Phase I:	Concept Development	FY 1981-1987
Phase II:	Design Development	FY 1988-1995
Phase III:	Construction and Startup	FY 1996-2003

During Phase I, a period of intensive research and development is required to characterize the dynamic effects of the gas turbine on other elements of the reactor within the primary circuit. A key activity during this period is the performance of tests to establish the failure modes and behavior of turbine rotors which affect design basis events for the gas turbine plant. In parallel activities, development work and tests on turbine bearings and seals will be conducted. Other technology development activities will emphasize materials development and the design of reactor internals for this environment. These activities will culminate in the submission of a Preliminary Safety Information Document to the NRC in 1987.

During Phase II and concurrent with early licensing review, the plant will progress through PSAR submittal in 1993 to the receipt of a

Construction Permit in 1995. A major element of the development program will be conducted in this period. A Helium Component Test Facility (HCTF) will be constructed and a non-nuclear demonstration of prototypic components will be performed. Such tests are considered necessary to refine and confirm methods used to predict the effects of coolant flow, acoustics, and other systems interactions and to exercise a prototype gas turbine through a representative range of operating conditions. Completion of these activities is scheduled to mark the onset of Lead Project construction.

The Construction and Startup period, Phase III, is scheduled with somewhat longer intervals than those for conventional plants to account for the added construction complexity of the nuclear island caused by the direct cycle gas turbine within the PCR. V.

Design and development costs for the HTGR-GT Program are projected at slightly over one billion dollars (1980s) and would be expended over a time interval to 2003. Peak cash outlays for design and development occur in support of the Helium Component Test Facility in the early 1990s at an annual rate of about \$165M per year. Therefore, a large, long-term program will be required to develop the HTGR-GT to commercial fruition. In addition, the GT Program involves significant uncertainty and related risk in that the development program is assumed to be successful in many critical areas such as materials, licensing, and component development.

4.3 Conclusions

The apparent lack of near-term incentive, prolonged schedule, and high cost for deployment combine to support a recommendation that the HTGR-GT not be pursued as a Lead Project. Further, the nature of the technical issues confronted in this plant suggests that a prudent development path would cast the direct cycle gas turbine concept in a second generation of gas-cooled reactors. As a follow-on system, features that limit the HTGR-GT would evolve through systems with less complex loading conditions and temperature regimes than those encountered in the HTGR-GT.

With the projected constraints on funding of gas-cooled reactor technology, it is recommended that minimal effort in the near term be expended to advance the HTGR-GT. The following areas should be emphasized in future work:

- Future activities should emphasize the resolution of priority technical issues that are common to the HTGR-GT and applications with more immediate Project potential.
- Cogeneration and process heat studies should characterize the role of the HTGR-GT in expected future markets where waste heat can be effectively utilized.
- Studies should continue to investigate areas of potential cost reductions.

1.0 INTRODUCTION

The High-Temperature Gas-Cooled-Reactor - Gas Turbine (HTGR-GT) concept has been recognized for many years as offering significant potential benefits to operating utilities. In particular, the closed Brayton-Cycle Gas Turbine power plant offers three major fundamental advantages over the more conventional Rankine, vapor-turbine power cycle: (1) heat is rejected from the cycle at a relatively high temperature (approximately 150°C or about 300°F), making it more economically attractive to employ dry or peak-shaved, wet/dry cooling or, alternatively, to achieve higher efficiency by addition of a bottoming power cycle when wet cooling is available; (2) the gas turbine possesses inherent capability to achieve relatively larger increases in power output and efficiency with increasing temperature than the Rankine Cycle, thereby better utilizing the higher temperature capabilities of high-temperature gas-cooled reactors; and (3) the low compression-expansion ratio helium turbomachine and modularized components arranged within a pressurized closed-cycle system result in a compact, integrated power conversion system with potential for reduced cost and high reliability.

1.1 Purpose

The HTGR-GT Application Study evaluates the HTGR-GT as a potential Lead Project candidate. The particulars of this study, which were outlined in the HTGR Lead Project Identification Plan (Ref. 1), were developed from the following points:

- The HTGR-GT is a long-term objective of the HTGR Program; however, suitability as a Lead Project requires assessment.
- If suitability is not established, a comprehensive treatment of issues and disincentives is needed.
- If suitability is ascertained, extensive discussion is needed on benefits, deployment scenarios, and the relationship with follow-on applications.

The HTGR-GT Application Study will also define design and development programs associated with a Lead Project position for the HTGR-GT as well as anticipated schedule and cost information.

1.2 Scope

Fundamental emphasis is placed on examining the candidacy of the HTGR-GT as a Lead Project and, in other scenarios, as a longer term follow-on objective. A position is developed relating benefits and incentives with a market assessment, an evaluation of potential commercial applications, and intrinsic developmental issues. This position, along with a deployment strategy and a sufficient technical basis, is the framework for an implementation plan.

A comprehensive technical description for the Gas Turbine is provided to form the basis for risk assessment and developmental uncertainty. In addition, certain advanced Gas Turbine concepts are described to illustrate alternatives for increasing efficiency.

SECTION 1 REFERENCES

1. HTGR Lead Project Identification Plan, Gas-Cooled Reactor Associates, February 1980.

2.0 OVERALL PROGRAM APPROACH

The high-temperature capabilities of the HTGR have long been recognized as significant both in the U.S. and within foreign nuclear energy programs. With its ceramic-based fuel, graphite core, and helium coolant, the HTGR is routinely capable of operation in temperature ranges well above those possible in contemporary nuclear systems. This high-temperature capability of the HTGR leads to increased efficiency in conventional applications and also makes possible a number of unique applications such as coupling with the Brayton (Gas Turbine) Cycle as described herein and generation of high-temperature process heat for industrial and synfuels applications.

Reflecting its broad potential in concepts and applications, the HTGR has interested various organizations and individuals in both private industry and government for many years. However, these interests, while supportive of the technology, have tended to be diverse rather than focused and have not led to a fruitful National HTGR Program. The long-term evolutionary potential of the HTGR has consequently led to differences among individuals and organizations with regard to the preferred program development path, program priorities, and demonstration schedules. With the difficulties facing the continuation of the HTGR Program in general, it was evident that a focusing element was required to reconcile the diversity of opinion and to obtain the concurrence and support of all HTGR Program participants. GCRA, therefore, initiated the HTGR Lead Project Identification Plan in December 1979.

Central to the Plan was the investigation of HTGR Lead Project options from which support for a strong National HTGR Program might grow. Working with the U.S. Department of Energy (DOE), General Atomic Company (GA), General Electric Company (GE), and Oak Ridge National Laboratory (ORNL), GCRA identified four HTGR Lead Project options for consideration. These four options--the Gas Turbine (HTGR-GT), the Reformer (HTGR-R), the Steam Cycle/Cogeneration (HTGR-SC/C), and the Nuclear Heat Source Demonstration Reactor (NHSDR)--were selected by the participants to encompass all potentially viable near-term HTGR projects. The balance of the GFY 1980 HTGR Program activities was directed to the development of these four Lead Project options, leading toward their evaluation, prioritization and, thus, sequencing within the reference HTGR Program. The HTGR design and development program in following years would be based upon the results of this effort. The ultimate result of these activities was envisioned to be an ordered HTGR Program, supported by all participants, which would logically evolve to a Lead Project commitment through a private sector initiative and subsequently provide for the follow-on development of other viable options.

2.1 Objectives

The long-term objectives of the HTGR Program relate to securing the unique advantages of HTGR technology for the nation and for energy

users. As presently envisioned, these objectives are embodied in the orderly development of three HTGR options: the HTGR-SC/C, the HTGR-GT, and the HTGR-R. The sequence of and timing for development are principal issues addressed within the HTGR Lead Project Identification Plan effort.

Incentives for the HTGR-GT are summarized in Section 1.0 and detailed in the body of this report. The HTGR-SC/C and HTGR-R are addressed in the following paragraphs. These options have received increased attention as national concerns for displacing oil and natural gas usage have emerged. Through such applications, the HTGR will eventually enable production of cogenerated steam, synthesis gas or hydrogen through reforming, direct steam carbon gasification, and thermochemical water splitting. These applications have tremendous potential for specific displacement of oil and natural gas use for electric power needs and/or industrial process heat supply and are the key to large-scale replacement of fossil energy by nuclear power. The current role of nuclear power is limited to baseload electrical production, which comprises less than 20% of the total U.S. energy consumption. The process heat and load-following electricity markets can easily double or triple the percentage of U.S. energy production supplied by nuclear power. The high-temperature capability of the HTGR permits consideration of the nuclear option in markets which have been historically fueled by fossil sources and can broaden the role of nuclear power in meeting world energy requirements.

The Steam Cycle/Cogeneration concept represents the HTGR option nearest commercialization. While there was a commercial offering by GA in the early 1970s, significant changes to the design have occurred in the process of improving plant reliability and/or cost and have not been subjected to recent licensing scrutiny. As an all-electric application, the HTGR-SC/C can only be envisioned as an alternative to the light water reactor (LWR) or coal for baseload production. The higher efficiency of the HTGR-SC/C, however, resulting from the 1005°F steam conditions does lead to improved uranium utilization. Application to cogeneration, however, offers the added potential for penetration of the industrial process heat market. In generating high-quality, high-temperature steam (1005°F, 2500 psia), the HTGR-SC/C can supply power for generation of electricity and can deliver steam to a process user up to 1005°F with significant economic margins versus transported coal.

Even greater potential for the HTGR may be realized through high-temperature process heat applications. In these concepts, the HTGR delivers high-temperature helium to a process heat exchanger (reformer in the case of the HTGR-R) and to a bottoming steam generator. The steam generator provides steam for electrical production for plant needs and for export. The process heat exchanger utilizes the high-temperature heat to drive chemical reactions to produce synthesis gas (carbon monoxide and hydrogen) or hydrogen. The applications envisioned include remote energy distribution, intermediate and peaking

electricity, synthetic fuel production, and feedstock production for methanol, ammonia, fertilizer, steel, or petrochemical industries.

2.2 Program Scenario Based Upon HTGR-GT Lead Project

The HTGR-GT Lead Project provides the most direct path to deployment of commercial HTGR Gas Turbine plants in the U.S. The long-term Program scenario based on the HTGR-GT Lead Project is shown in Fig. 2.2-1. In this scenario, the HTGR-SC/C system is bypassed and the HTGR-R is projected as a follow-on system.

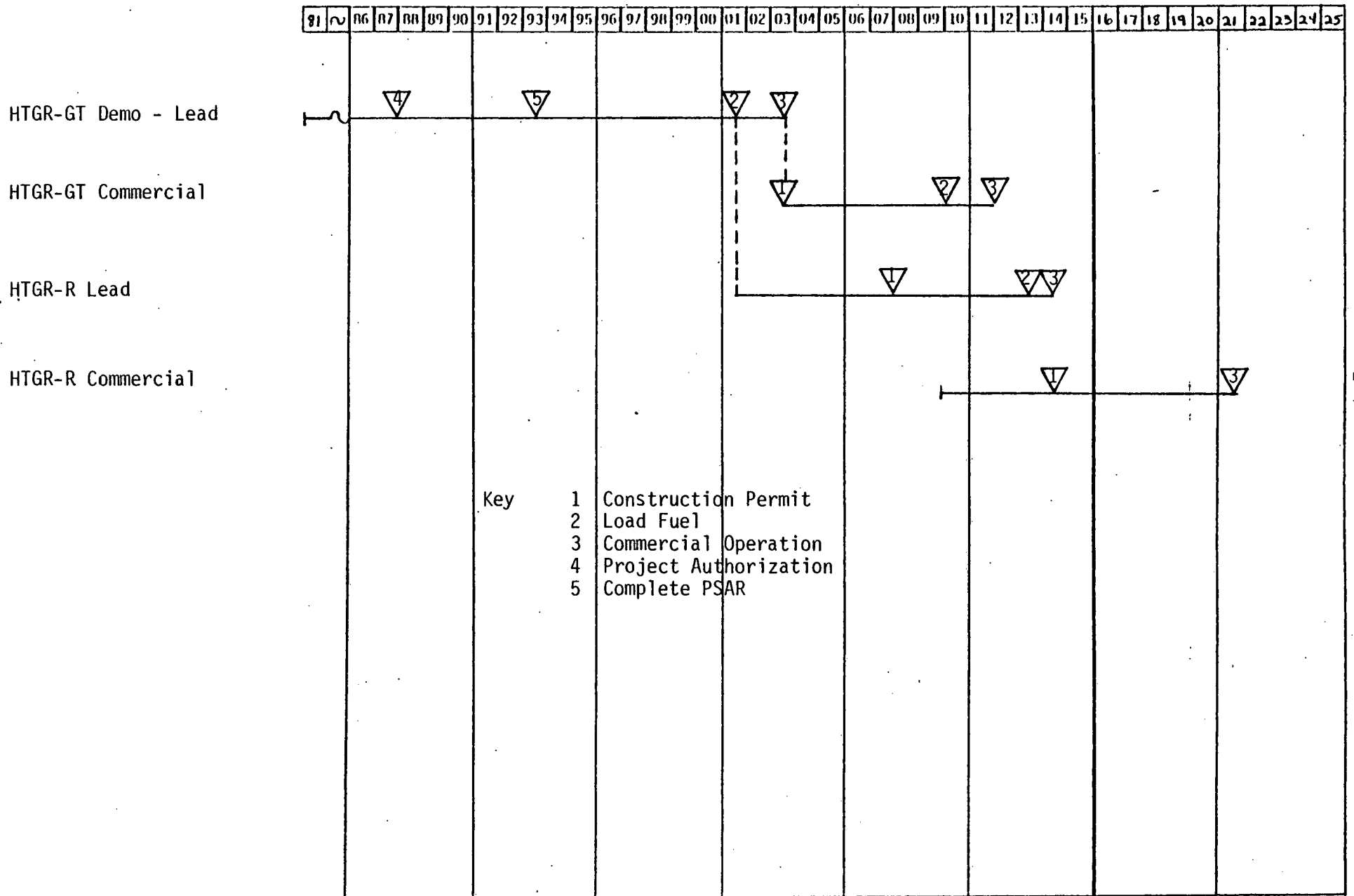
The HTGR-GT Lead Project has been configured to confirm the currently envisioned commercial plant component and system technologies wherever possible. Under the assumption that a commercial market does evolve for an 850°C core outlet temperature HTGR-GT, the commercial plant would likely replicate the identified demonstration plant. If the commercial market cannot be justified until the HTGR-GT design achieves a 950°C core outlet temperature, the identified demonstration plant provides a large step towards commercialization. A direct step to 950°C would require significant advances in materials technology and could require many additional years to resolve.

Depending on the level of design confidence and scope of the development program, the HTGR-GT demonstration plant could lead to an initial commercial plant operation within a minimum of nine years. This ambitious schedule results from a commercial plant construction permit immediately after commercial operation of the demonstration plant and from an unperturbed licensing and construction schedule. While a certain amount of risk is involved in achieving this schedule, it is important in maintaining the supplier industry capability established for the demonstration plant. An intermediate or lead commercial plant with some amount of government financial support may be required to help offset this risk while maintaining the industrial capability and establishing the commercial market.

Since many of the HTGR-R development requirements are enveloped by the HTGR-GT, considerable progress would be achieved toward the HTGR-R through the HTGR-GT Lead Project. The unique HTGR-R requirements associated with the intermediate heat exchangers, the process heat exchangers, and other secondary systems could be pursued in parallel with the HTGR-GT Lead Project.

The emphasis of this report dwells on the HTGR-GT Lead Project and the resulting commercial gas turbine systems.

Figure 2.2-1 HTGR-GT Program Schedule



3.0 COMMERCIAL PLANT

3.1 Technical Description of the Commercial HTGR-GT

3.1.1 Reactor Turbine System

The reactor turbine system (RTS) consists of power conversion loops (PCLs), the reactor core, and auxiliary cooling loops. The two PCLs are rated at 400 MW(e) each and consist of a turbomachine, a recuperator, a precooler, and control valves. Figure 3.1.1-1 shows the cycle diagram for the RTS. The core auxiliary cooling system (CACS) consists of three separate and independent cooling loops comprised of auxiliary circulators, core auxiliary heat exchangers (CAHEs), and ultimate heat sinks to the atmosphere. The system is capable of removing the core residual and decay heat for cooldown following loss of helium circulation in the RTS with the reactor in a shutdown condition and with the primary coolant system either pressurized or depressurized.

The secondary coolant systems are significantly different from those of the HTGR-SC/C, the LWR, or fossil steam plants because the power turbine is located in the primary loop. The circulating water system (CWS) is a pressurized, nonnuclear-safety, closed-loop system (one per PCL), which transfers heat from the precoolers and rejects it to the environment through cooling towers. During all modes of normal plant operation and during extended plant shutdown, the CWS provides the normal means of plant heat rejection.

3.1.2 Prestressed Concrete Reactor Vessel

The PCRV (Fig. 3.1.2-1) is a multicavity pressure vessel that, together with liners and penetrations, functions as the primary containment for the reactor core, PCLs, and CACS, which together form the RTS. The PCRV also provides biological shielding around the core and provides the necessary structural support for the RTS. The diameter of the PCRV is 32.6 m (107 ft) and its height is 35.4 m (116 ft). The PCRV is constructed of high-strength concrete reinforced with conventional reinforcing bars and prestressed by two post-tensioning systems, the linear prestressing system and the circumferential prestressing system. All cavities and penetrations are lined with welded steel liners, which act as impermeable gas-tight membranes to contain the primary coolant. The liners are anchored by studs welded to the liners and embedded in the concrete. To protect the concrete from the high internal temperatures, the liners of the vessel cavities are covered and insulated with a thermal barrier. Any heat passing through the thermal barrier is removed by the liner cooling system via coolant flowing through tubes attached to the concrete side of the liner.

3.1.3 Turbomachinery/Generator

Conceptual design of the turbomachinery has resulted in a simple, rugged arrangement consisting of a single-shaft, direct-driven generator

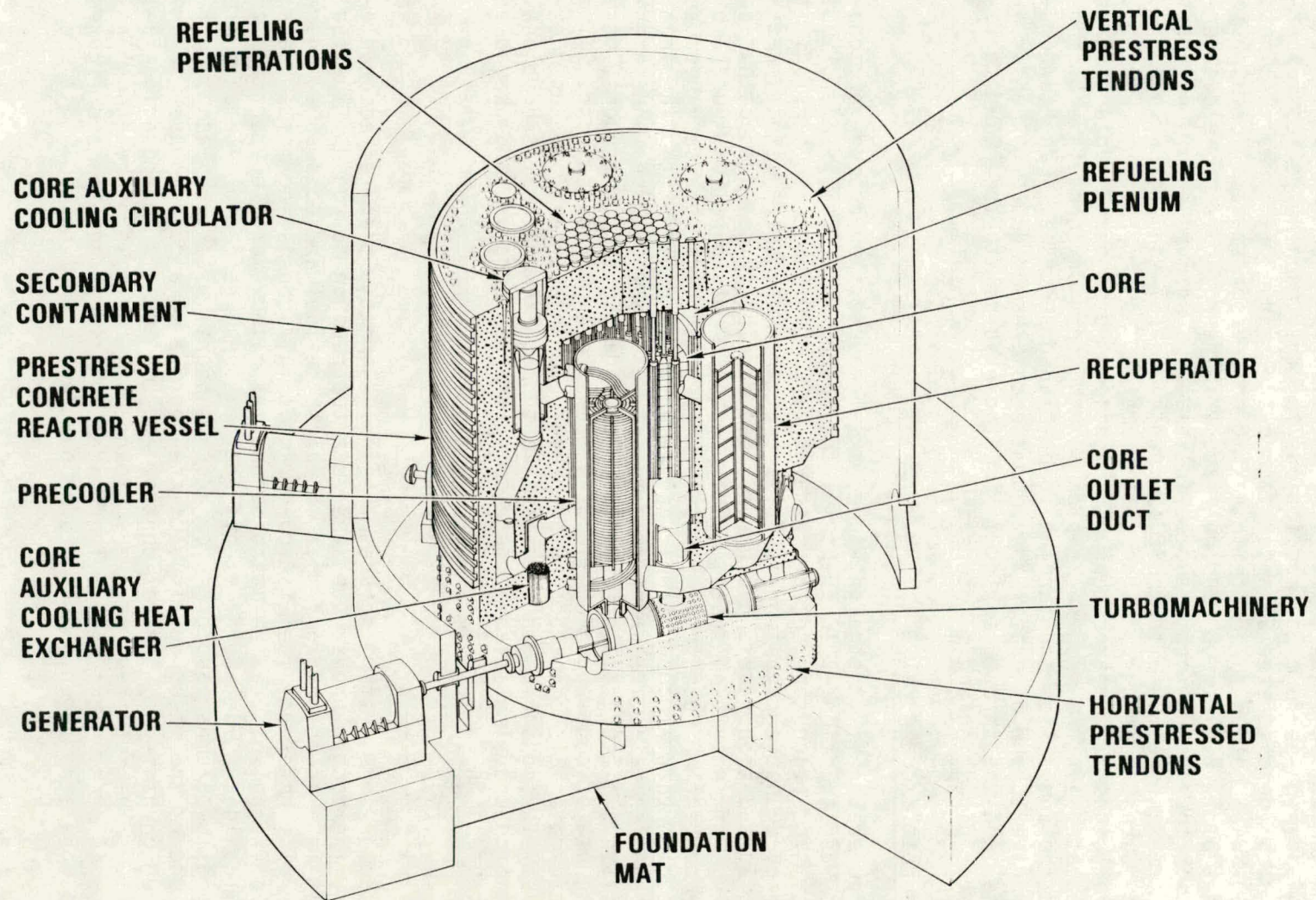


Figure 3.1.2-1 Two-loop 2000-MW(t) HTGR-GT power plant

design. A simplified cross section of the 400-MW(e) 60-Hz machine, along with additional information, is provided in Section 5.1.1. The design and high performance predictions for this machine reflect experience with demonstrated advanced-technology industrial gas turbines. The 400-MW(e) helium turbomachine has 18 compressor stages (for a pressure ratio of 2.6) and 8 turbine stages. The rotor is of welded construction that has a long, successful history in Europe for both gas and steam turbines. With the 60.8 metric ton (67 ton) rotor supported on two journal bearings, the overall diameter is 4 m (13 ft), and the machine weighs 277 metric tons (305 tons). Rotor burst protection is incorporated in the machine design in the form of containment rings around the compressor and turbine rotor-bladed sections. The turbomachine drive to the generator is from the compressor end of the turbomachine, and the thrust bearing is located external to the PCR to facilitate inspection and maintenance.

3.1.4 Heat Exchangers

The recuperator is a straight-tube, axial counterflow design of monolithic construction. Inherent in the design are "fail safe" tube sheets; since each tube sheet is stayed by 53,000 tubes in tension, it is believed that a catastrophic failure at the tube sheet is not credible. In addition, this design provides for inservice inspection and plugging of individual tubes. The precooler has a helical bundle of cross-counterflow geometry. Large surface areas are necessary because of the high thermal conductance requirements associated with the large heat transfer rates. However, the modest metal temperatures and internal pressure differentials, compared with modern steam generators, permit the use of code-approved lower-grade alloys of reduced cost. The ferritic materials selected for both heat exchangers have been used extensively in industrial and nuclear plant heat exchangers. Additional information and simplified isometrics of the recuperator and precooler are provided in Section 5.1.1.

3.1.5 Reactor Core

The reactor core system includes the fuel elements, the reflector elements, the control rods, and the startup neutron sources. The fuel element is a graphite block that contains the fuel and acts as a moderator. The reference fuel is low-enriched fuel consisting of a coated fissile kernel of uranium carbide and a coated fertile particle of thorium oxide. The fuel elements and hexagonal reflector elements are arranged in columns supported on core support blocks. Each support block under the major portion of the active core corresponds to one fuel region having a central control column and six surrounding fuel columns. The fuel regions are surrounded by two rows of hexagonal reflector columns. Section 5.1.1 provides additional details and figures of the core arrangement and core support system.

3.1.6 Plant Arrangement

A simplified plant arrangement (Fig. 3.1.6-1) utilizes horizontal electric generators and turbomachines with access and removal through grade-level penetrations in the containment. The generators are located outside the containment and are connected to the turbomachine via a shaft/bearing/seal through the containment building. Greater details of the plant arrangement are provided in Section 5.1.2.

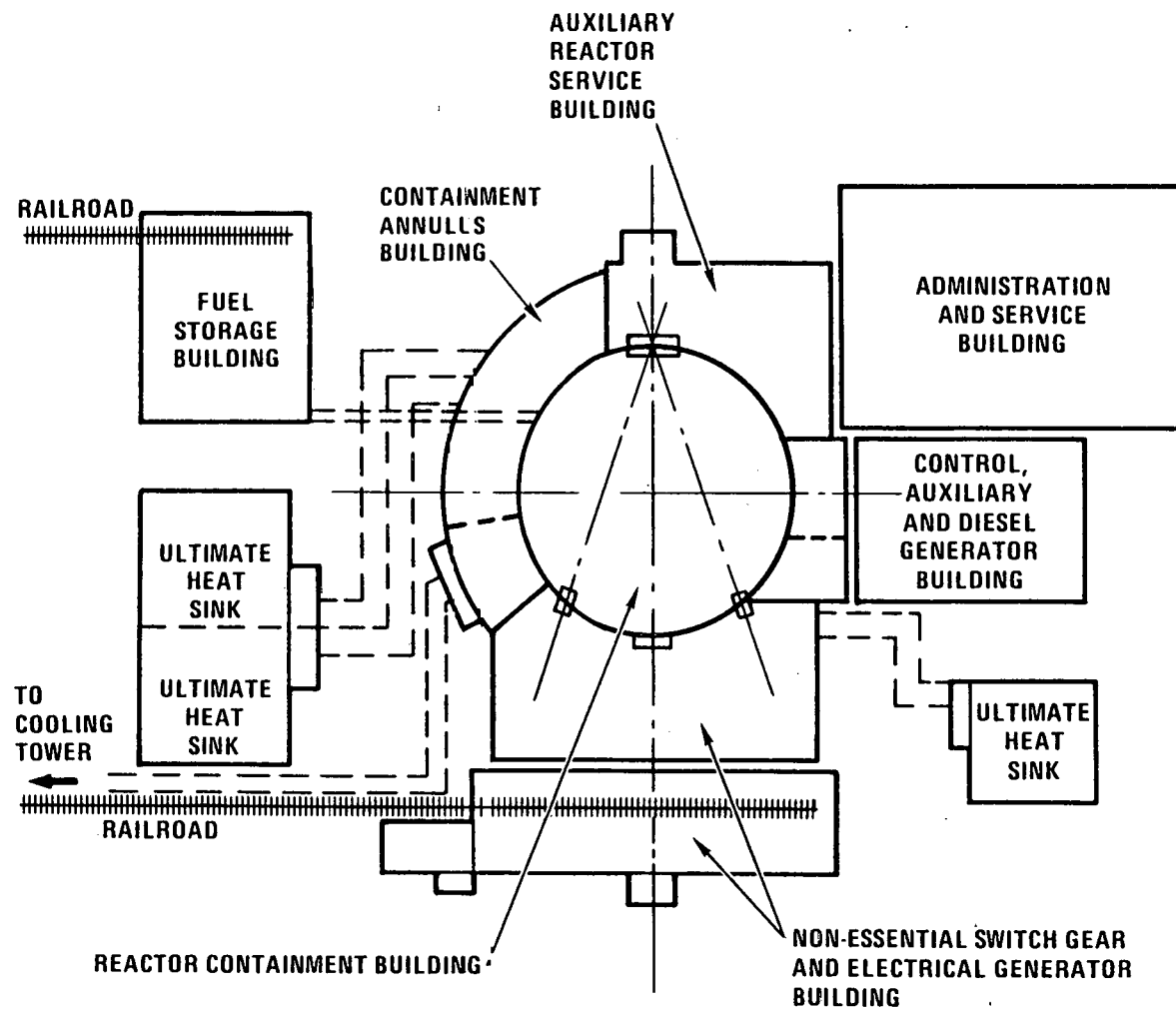


Figure 3.1.6-1 Plant arrangement

3.2 Potential of Higher Temperatures

The singularly most important parameter affecting the performance of the gas turbine is the turbine inlet temperature. A value of 850°C (1562°F) was selected for the plant design so that technology consistent with the existing HTGR core and internals design bases could be utilized. At this temperature, the turbine could be designed with uncooled blades using existing nickel-base alloys.

The HTGR has the capability of providing a much higher core outlet (turbine inlet) temperature than previously discussed. Temperatures on the order of 950°C (1742°F) are being investigated in current nuclear process heat studies. There are two acceptable approaches to increasing the reactor outlet temperature: (1) the most readily predictable approach is by means of modifications to existing fuel block cooling arrangements, application of existing fuel particle types, and changing the fuel loading arrangement; and (2) using present fuel block cooling and loading arrangements, obtain increased core coolant outlet temperatures by modifying the fuel particle and particle coating chemistry to make fuel particles capable of withstanding higher fuel temperatures without unduly increasing fission product release.

With the foregoing modifications, it is postulated that the reactor outlet temperature could be increased to 982°C (1800°F); in fact, the reactor outlet temperature is not limited by the reactor but rather by the capabilities of the thermal barrier system upstream of the gas turbine. The nuclear gas turbine can be designed to accommodate a gas turbine inlet temperature of 982°C (1800°F) by the use of advanced refractory turbine blade materials, such as TZM, or the use of present-day materials designed with blade cooling. With a turbine inlet temperature increase, the power rating from a given turbomachine frame size will be significantly higher than the current reference design. The plant efficiency has been estimated to increase to about 46% with a turbine inlet temperature of 950°C (1742°F) for the direct-cycle non-intercooled plant.

3.3 Advanced Applications

The heat rejection characteristics of the HTGR-GT plant are conducive to cogeneration, which in the broadest sense covers all possible uses of the waste heat reject energy. In place of dry cooling towers, the two possible modes of heat utilization are the bottoming cycle and hot water/steam heat production. The bottoming rankine cycle can utilize the waste heat from the PCL to produce additional electric output to achieve higher overall efficiency of the combined cycle.

3.3.1 Flashed Steam Bottoming Cycle

The waste heat rejected from the PCL precooler cooling waste systems is rejected at a relatively high temperature, which facilitates the recovery of the energy as additional electric power by flashing the water to steam which, in turn, drives a low-pressure, condensing steam turbine. The condensed steam is mixed with the unflashed precooler water and returned to the precooler through an intermediate heat exchanger (IHX) which dissipates the remaining waste heat to a wet cooling tower. The optimized temperature of the precooler outlet and the number of flashing steam stages and pressures are shown on the cycle diagram in Fig. 3.3.1-1. A computer program, CODER, has been used in preliminary efforts to optimize the combined cycle. Initial studies have indicated that the economics can be improved by decreasing the precooler outlet temperature to produce more power in the primary circuit.

Precooler outlet water at 4.83 MPa (700 psia) and 252°C (485°F) is flashed at 0.83 MPa (120 psia) and 0.14 MPa (20 psia) to drive two separate condensing steam turbine generators, which produce 128 MW(e) and 42 MW(e), respectively, resulting in a combined plant output of 915 MW(e) and an efficiency of 42.2%. This represents an overall efficiency increase of 2.8% due to the bottoming cycle addition.

A dry cooling tower is not used in this cycle since the water from the low-pressure flash tank is at 109.9°C (228°F), which does not warrant the use of a dry cooling tower in series with the IHX. The heat rejected in the IHX and the two condensers is rejected, in turn, to a common wet cooling tower. Both the IHX and wet cooling tower are components of the basic wet/dry cooled cycle without a bottoming cycle.

The high-pressure 0.83-MPa (120-psia) flash tank is a relatively simple carbon steel vessel, 24,400 mm (80 ft) long and 4570 mm (15 ft) in diameter. The rough configuration is shown in Fig. 3.3.1-2. Three low-pressure flash tanks are required to keep the vessel size practical. Each tank has the same overall dimensions as the unit shown in Fig. 3.3.1-2. Changes have been made in the shell thickness [31.8 to 15.9 mm (1-1/4 to 5/8 in.)] and in the size and arrangement of connections. A conceptual plot plan of this design is provided in Fig. 3.3.1-3.

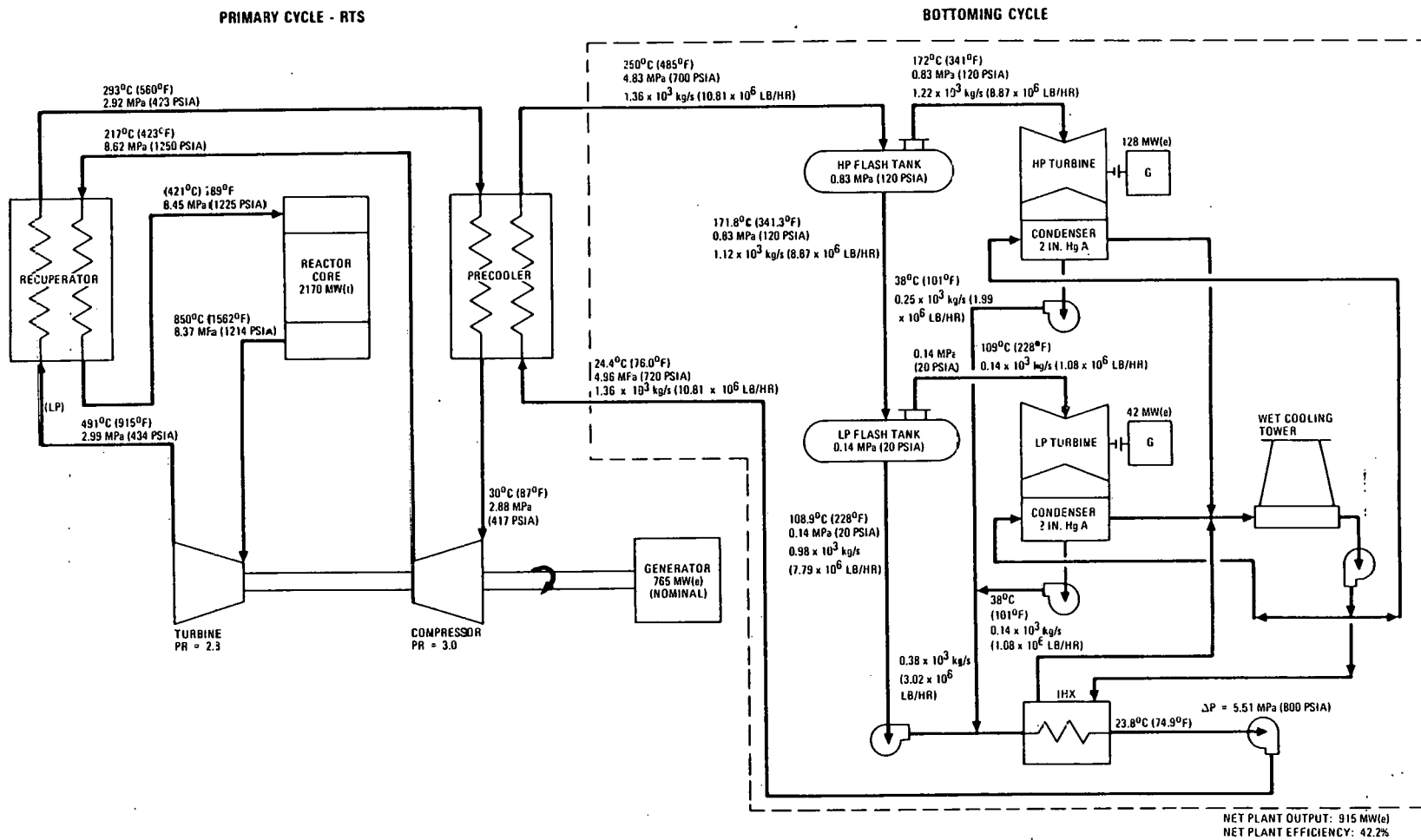


Figure 3.3.1-1 HTGR-GT cycle diagram with flash steam bottoming cycle

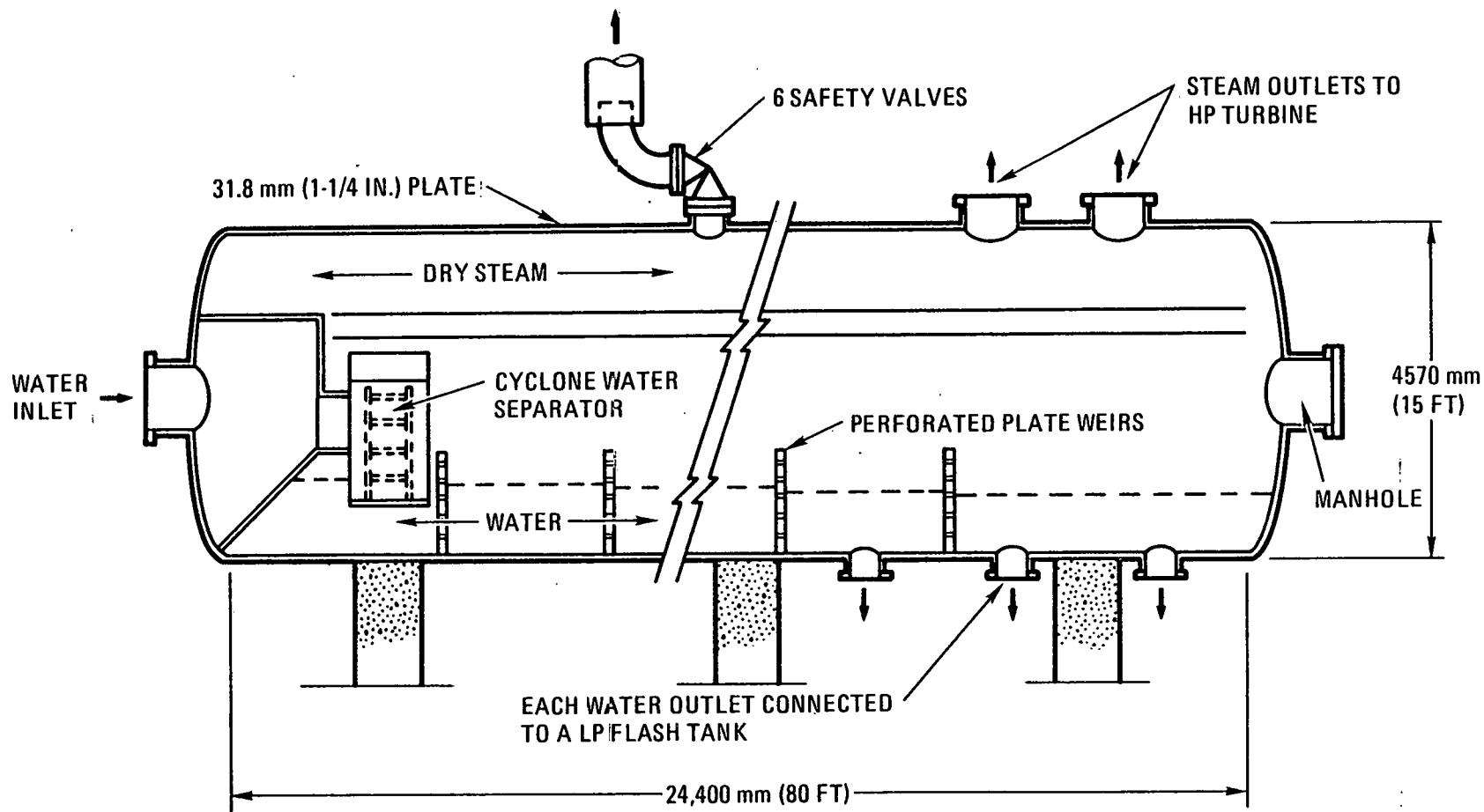


Figure 3.3.1-2 High-pressure flash tank for HTGR-GT flash steam bottoming cycle

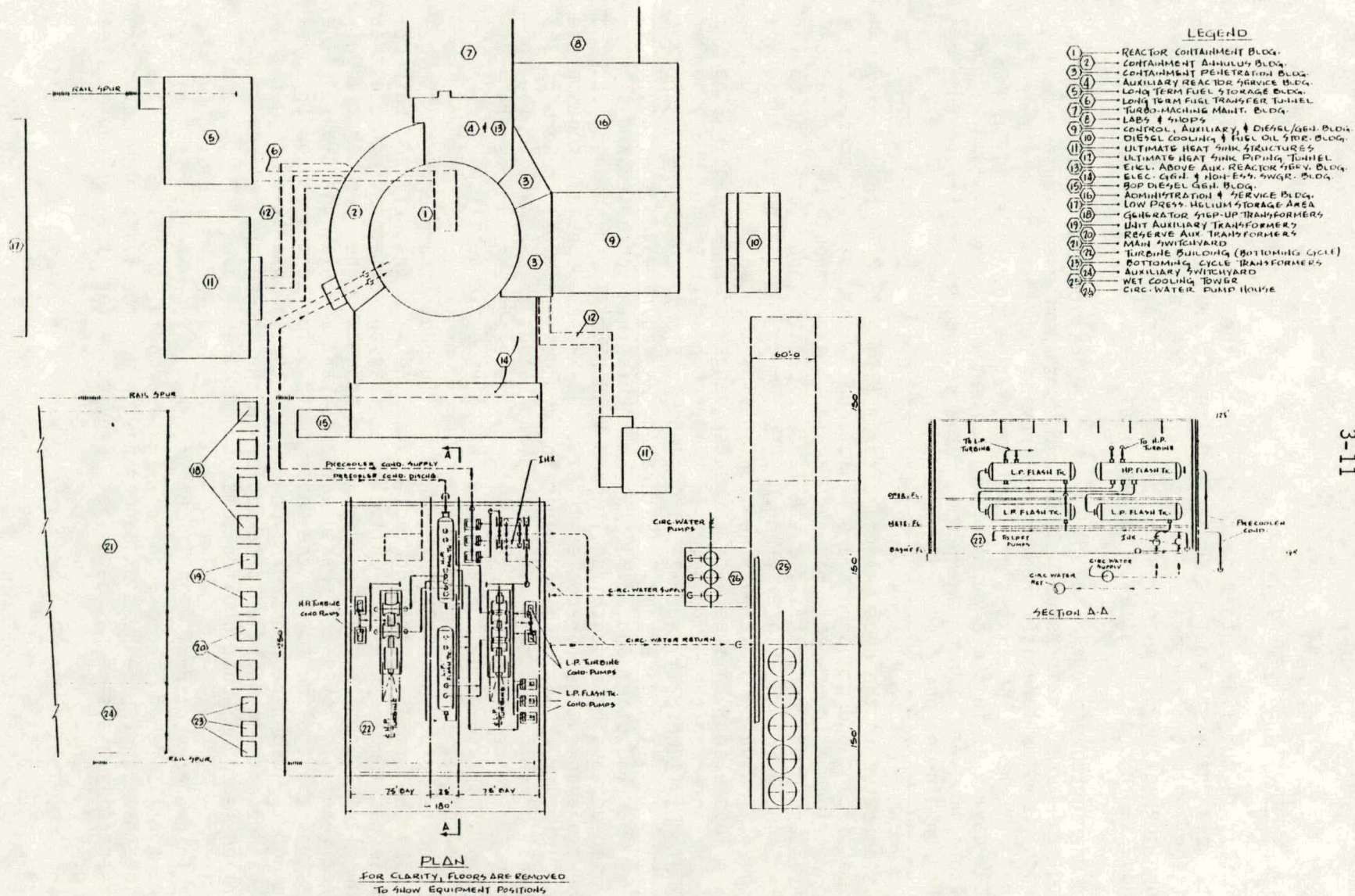


Figure 3.3.1-3 Flash steam bottoming cycle plot plan
(See 11" x 17" drawing in Appendix C)

3.3.2 Ammonia Bottoming Cycle

Similar to the flash steam bottoming cycle, the ammonia bottoming cycle utilizes the waste heat rejected from the PCL precooler to produce additional electrical power. Utilization of the waste heat power results in higher plant efficiency with only moderate changes in the RTS and no increase in the core size or nuclear fuel cost.

In the bottoming cycle, the precooler (which formerly used water to cool helium to compressor inlet conditions) now uses liquid ammonia, which is simultaneously heated to provide superheated NH_3 vapor for the turbine. The precooler, or NH_3 evaporator, is still within a PCRV cavity as in the dry-cooled (or conventional direct-cycle) HTGR-GT plant; however, other components in the secondary circuit are outside the containment building.

As indicated in the heat balance diagram (Fig. 3.3.2-1) the primary helium circuit and components are functionally the same as those for a conventional direct-cycle gas turbine facility, although heat exchanger sizes are different; i.e., a smaller recuperator and larger precooler are needed in the binary plant. Also, most of the temperatures are higher, including those for the coolant in the two precoolers of the twin loop plant; this permits a greater degree of superheat in the NH_3 vapor entering the turbine. In the secondary system, NH_3 vapor at 15.87 MPa (2300 psia) and 251.7°C (485°F) flows to a single turbine, which drives the generator. Saturated NH_3 vapor leaves the turbine at 1.24 MPa (180 psia) and 32.8°C (91°F) and is then condensed and slightly cooled; a single centrifugal pump feeds the liquid ammonia back to the two evaporators (precoolers) at 17.93 MPa (2600 psia) to complete the circuit. The turbine generator produces a gross power of 363 MW(e), contributing to a total net plant output of 998 MW(e). The combined plant efficiency is thus 46%. This represents an overall efficiency increase of 6.3% due to the bottoming cycle addition.

A computer program, CODER, has been used in preliminary efforts to optimize the ammonia bottoming cycle, starting with the reference direct-cycle gas turbine plant. Initial optimization attempts indicate that the plant economics can be improved by adjusting the recuperative effectiveness, compressor pressure ratio, and ammonia conditions. The cost of the cooling tower, the major contributor to the secondary system costs, might be reduced by condensing the ammonia at a higher temperature and including an aftercooler on the condenser. Also, an NH_3 -driven feedpump would provide some economic benefit.

As in the conventional HTGR-GT plant, the primary helium is still on the shell side of the precooler; liquid ammonia, therefore, flows inside the tubes. The heat exchanger design and the fluid flow paths are essentially the same as described previously for the precooler (see Fig. 5.1.1-2).

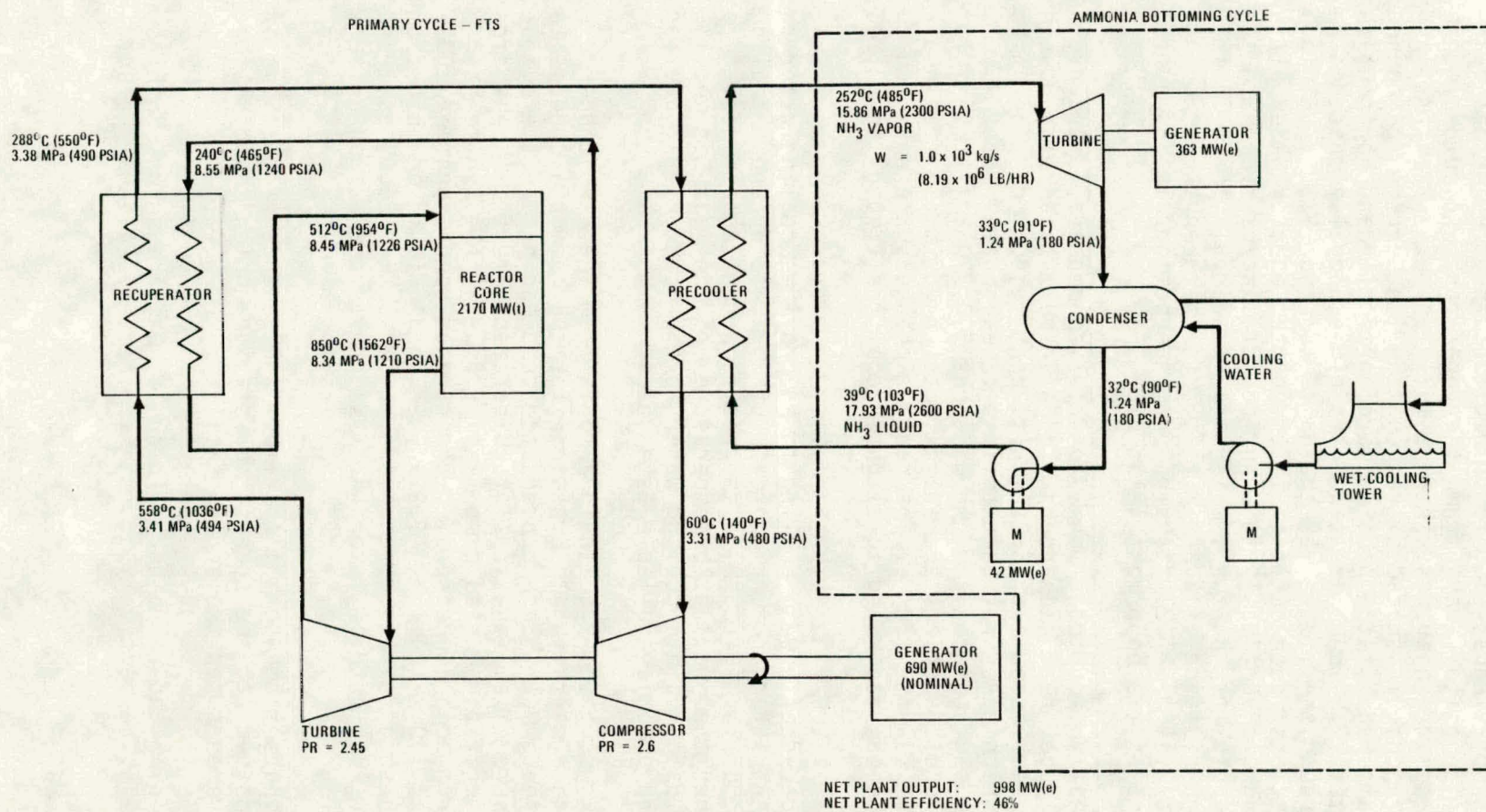


Figure 3.3.2-1 HTGR-GT cycle diagram with ammonia bottoming cycle

The turbine (Fig. 3.3.2-2) is a 375-MW(e) double (split) flow, seven-axial-stage unit rotating at 3600 rpm. It contains two journal bearings on the end opposite (away from) the generator. Liquid ammonia cools and lubricates the bearings, which simplifies seal problems, and the bearing and journal surfaces are coated with chrome oxide and ground. The turbine and stator housings are vertically and horizontally split, respectively. There are six inlet pipes, equally spaced circumferentially around the middle of the unit, and two 1.37-m (54-in.) diameter vapor outlets; each outlet, at the bottom ends of the split-flow turbine, connects vertically to a horizontal condenser beneath the turbine. The length of the turbine is more than 6.81 m (22.33 ft), with a centerline distance of 5.44 m (17.83 ft) between the two outlets; the unit is 3.41 m (11.21 ft) high from the bottom of its outlet flanges to the top of its uppermost inlet pipe.

The two horizontally mounted ammonia-to-water condensers are conventional shell and tube heat exchangers (TEMA J-shell type with double nozzle entries), each capable of 562.5-MW(t) heat rejection. The units (Fig. 3.3.2-3), which have not been optimized, each have about 45,580 m² (490,600 ft²) surface area and weigh about 500,770 kg (1.10 x 10⁶ lb); their tube bundle sections are approximately 4.57 m (15 ft) in diameter by 27.43 m (90 ft) long. The heat exchangers are designed to fully condense the NH₃ vapor and to slightly cool the liquid to 32.2°C (90°F).

A mechanical forced-draft (wet) cooling tower provides cooling water to the condensers. At a temperature approach of 2.8°C (5°F) and a cooling temperature range of 11.1°C (20°F), the heated water can be cooled to 18.9°C (66°F) when the wet bulb temperature of the air is 16.1°C (61°F). Two pumps circulate cooling water to the condensers, one for each unit; a third pump is available as a spare.

Two receiver tanks collect the liquid ammonia from the two condensers and two single-stage centrifugal pumps transfer the ammonia condensate to the ammonia feedpump. Each condensate pump can handle the output from both condensers (i.e., the total turbine discharge flow). A single three-stage centrifugal feed pump (Fig. 3.3.2-4) is sized to handle the condensate from both condensers. Under rated conditions the 3600-rpm unit transfers 1.0 x 10³ kg/s (8.2 x 10⁶ lbm/h) liquid ammonia from the receiving tanks to the two precoolers at 17.93 MPa (2600 psi) and 39.4°C (103°F). An electric motor (42 MW) drives the ammonia feedpump.

The generator outputs from both the primary and secondary portions of the plant are combined at the transformer station, which supplies the electrical requirements for the entire facility.

Preliminary plot plan and turbine building layouts are shown in Figs. 3.3.2-5 and 3.3.2-6, respectively, for this direct-cycle binary plant.

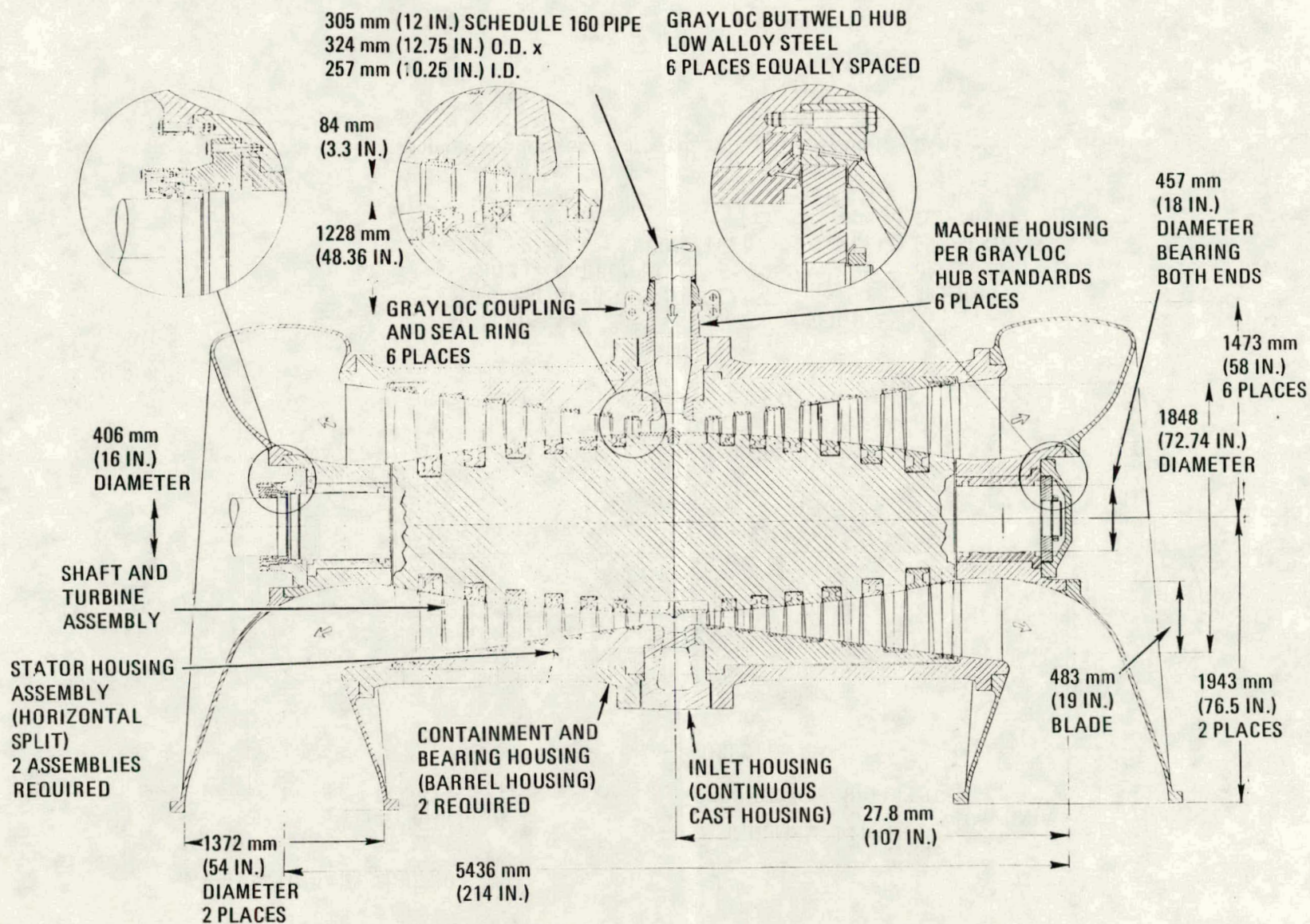
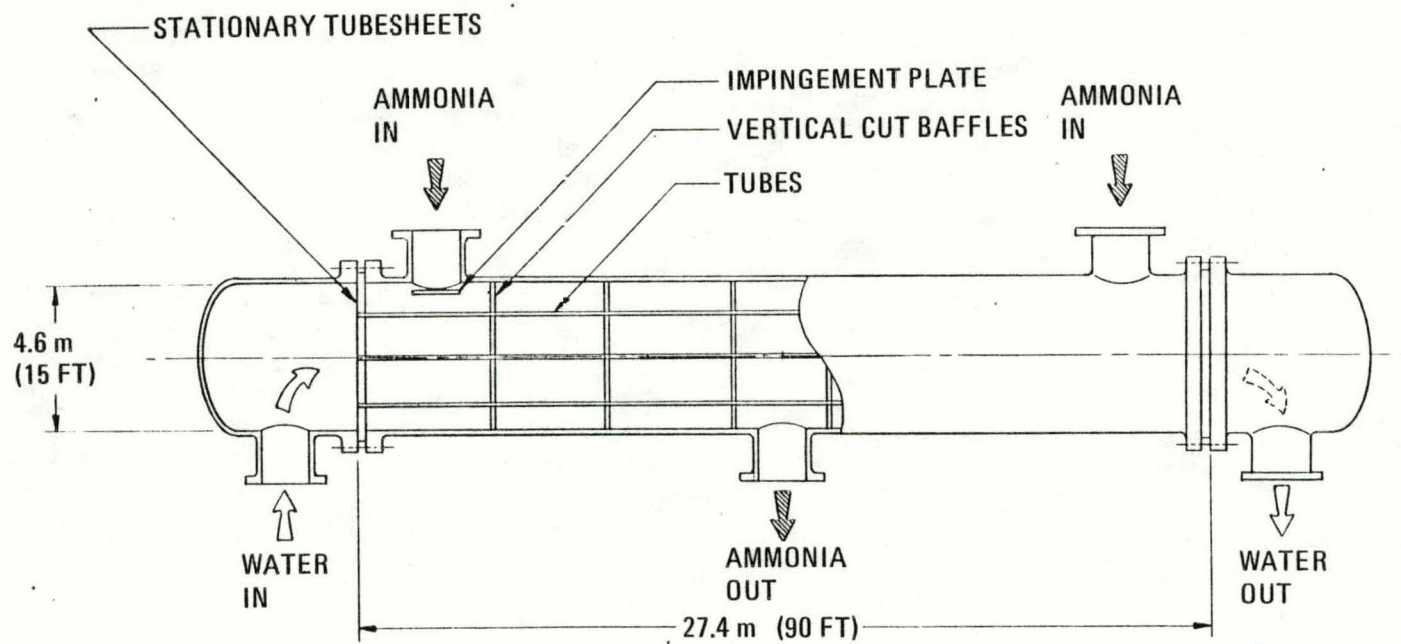


Figure 3.3.2-2 Ammonia turbine, 375 MW(e)



MATERIAL:
 TUBES: STAINLESS STEEL
 SHELL: CARBON STEEL
 INTERNALS: CARBON STEEL

WEIGHT:
 TUBES: 445,000 KG
 (981,000 LB)
 SHELL: 55,800 KG
 (123,000 LB)

Figure 3.3.2-3 Ammonia condenser for direct-cycle HTGR-GT binary cycle plant

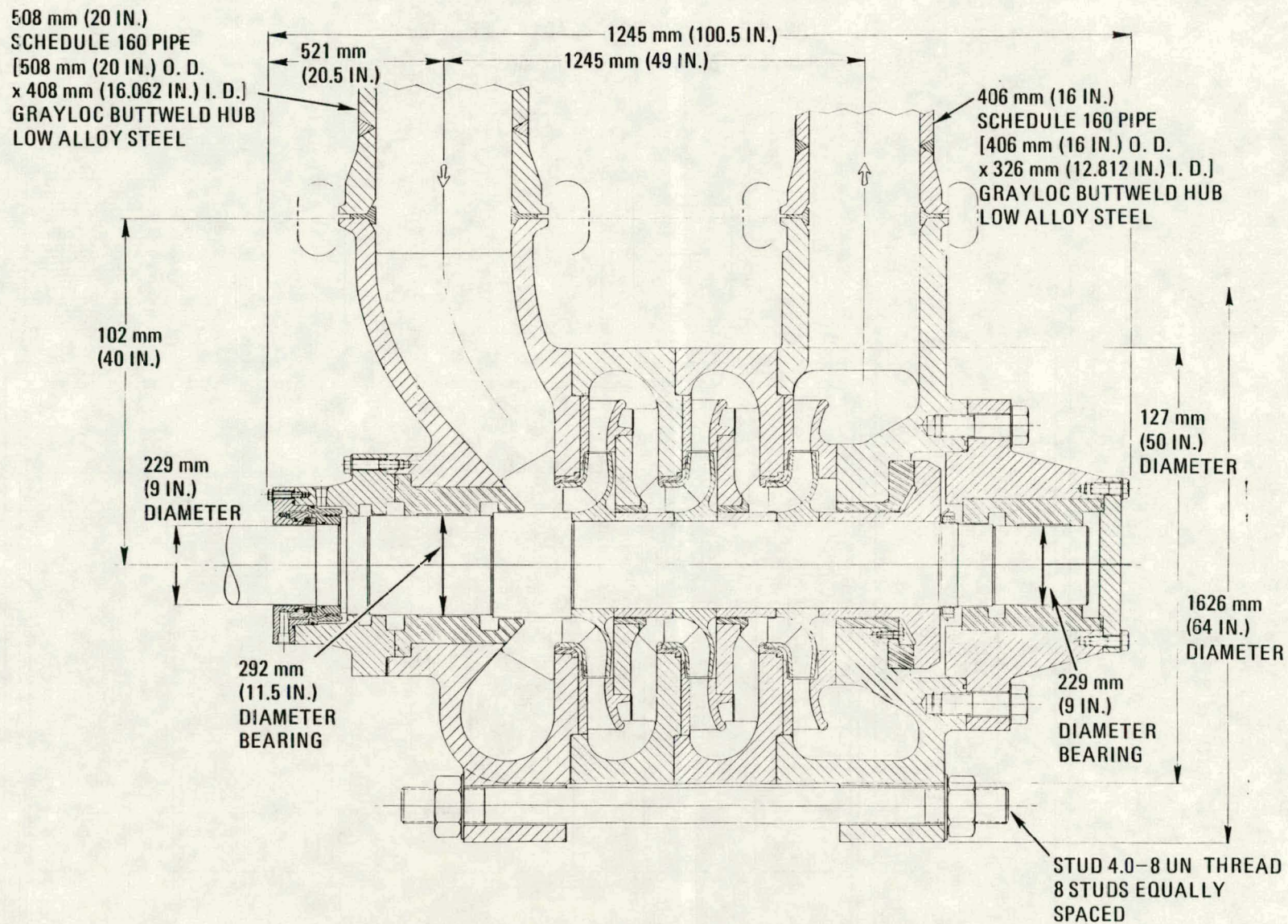


Figure 3.3.2-4 Feed pump for ammonia evaporator

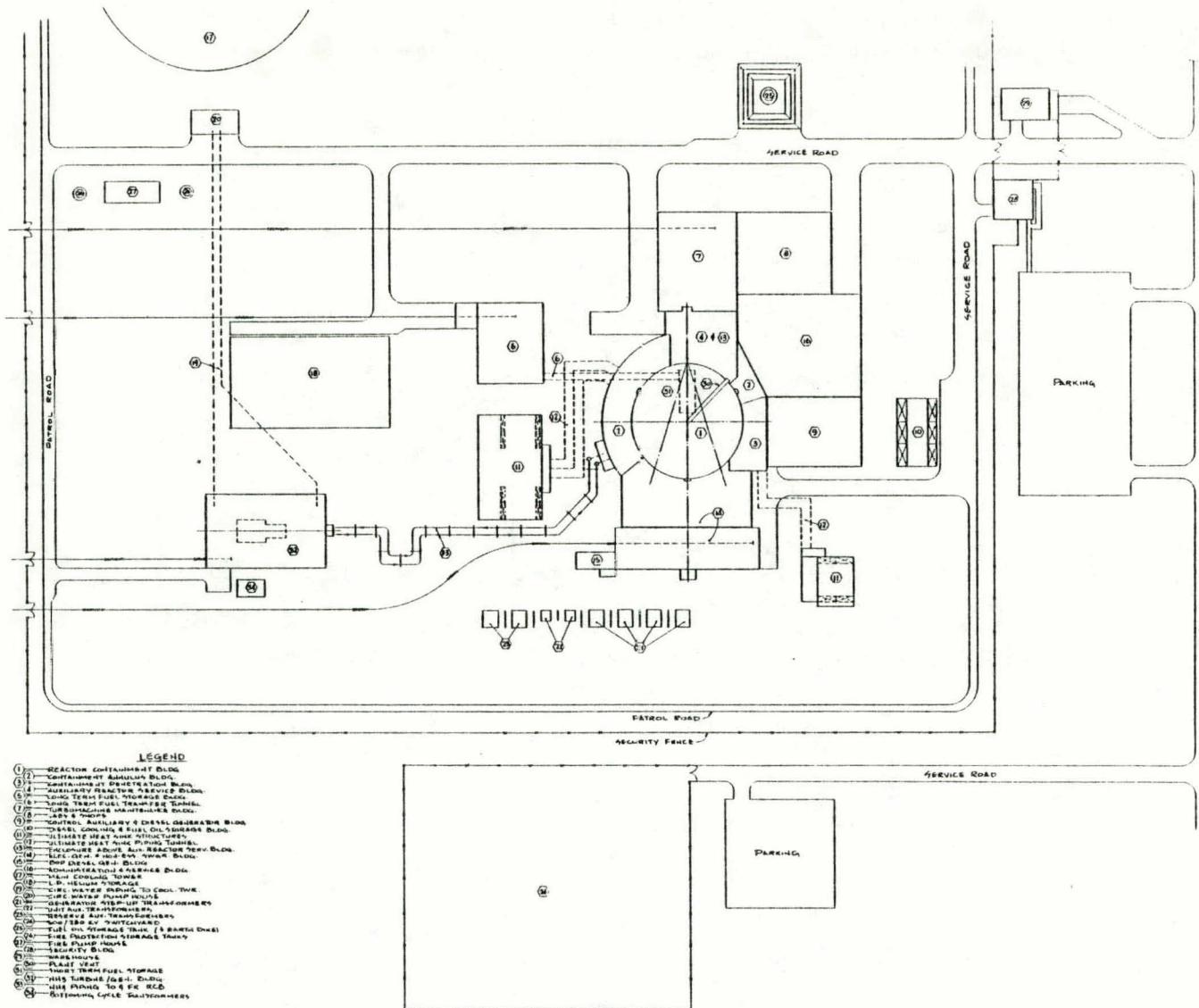


Figure 3.3.2-5 Ammonia bottoming cycle plot plan
(See 11" x 17" drawing in Appendix C)

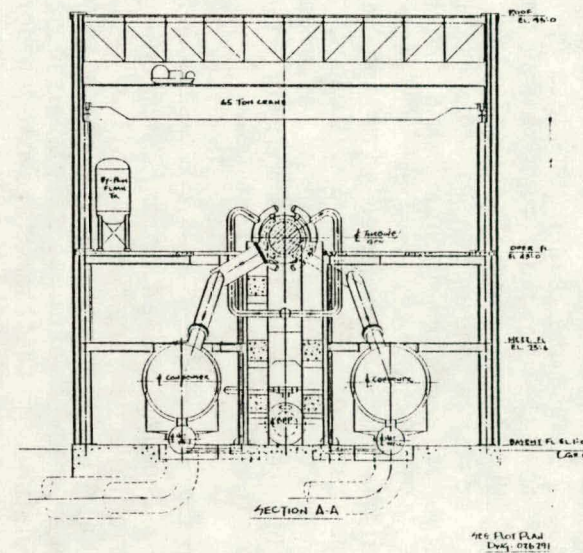
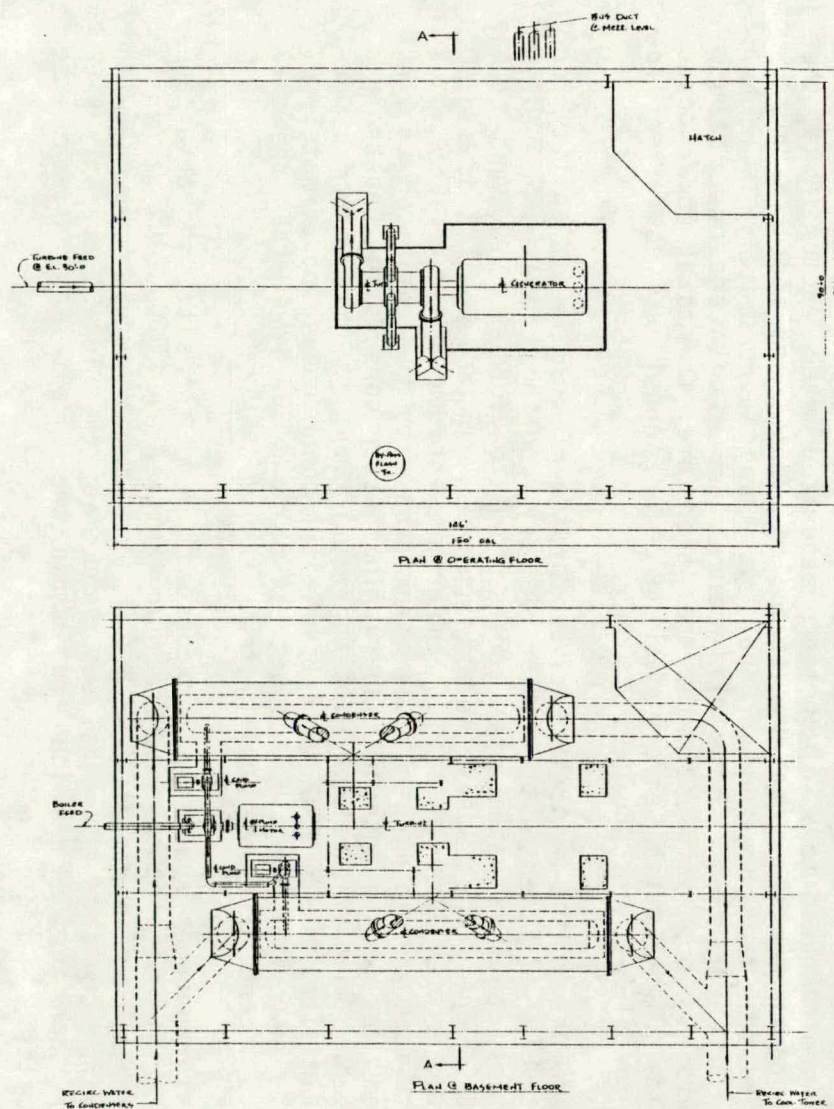


Figure 3.3.2-6 Ammonia bottoming cycle turbine building layout
(See 11" x 17" drawing in Appendix C)

3.4 Economic Assessment

Table 3.4-1 presents summary-level cost data for the reference HTGR-GT plant. These costs are based on data developed by General Atomic Company (GA) for the reactor turbine system and United Engineers & Constructors (UE&C) for the balance of plant. A more detailed breakdown of base costs is presented in Appendix D. Costs are projected as representative of a mature industry and are referred to as equilibrium plant costs. 1995 costs assume an inflation rate of 6% and IDC of 10%. Operating costs associated with the capital investment assume a fixed charge rate of 18% with a .7 capacity factor. Operating and maintenance costs are based upon an assumed parity with LWRs. Fuel costs have been estimated by GA similarly assuming a .7 capacity factor, leveled over a 30-yr plant lifetime.

In Table 3.4-2, cost data for the reference HTGR-GT plant utilizing both wet/dry cooling and an ammonia bottoming cycle (binary cycle) for heat rejection are compared with similarly sized coal, LWR, and HTGR-SC plants. The HTGR-GT binary cycle costs were also supplied by GA and have a higher degree of uncertainty associated with them due to the conceptual nature of the ammonia cycle components. The cooling system adder costs were based on the previous analyses. The total estimated power costs present a valid comparative analysis of the plants.

Based on the information presented in Table 3.4-2, it can be seen that when compared to a wet-cooled pressurized water reactor (PWR), the total power costs of the dry/wet-cooled HTGR-GT are approximately 5% higher with the once-through fuel cycle and approximately 3% higher with full recycle. These differences are within the uncertainty bandwidth of the cost estimates and, therefore, the total power costs for the HTGR-GT with wet/dry cooling can be considered equivalent to both the HTGR-SC and the LWR.

Because one of the major incentives for the HTGR-GT is its ability for relatively efficient utilization of dry cooling, Table 3.4-2 also provides cooling system cost adders for the HTGR-GT, LWR, and coal plants. These cost adders were calculated from the results of a UE&C cooling system study (Ref. 1) and represent the added capital plus capitalized operating penalties associated with the optimized cooling systems for each plant under the various water consumption constraints imposed. When a total dry cooling requirement is imposed for a particular site, the HTGR-GT will have total power costs approximately 6% to 9% lower than the LWR and between 25% and 30% lower than the coal plant, depending on the fuel cycle.

As water becomes available for consumption at the site, the advantage of the HTGR-GT becomes smaller. For example, when 500×10^6 gal/yr is available, or approximately 15% of what an LWR would consume with total wet cooling, the total power costs of the HTGR-GT are only 4-6% lower than the LWR, which are within the uncertainty range of the estimates.

TABLE 3.4-1

HTGR-GT EQUILIBRIUM PLANT COSTS

PLANT PARAMETERS

Power Rating - MW(t)	2000
Net Electrical Output - MW(e)	800
Efficiency (%)	40
Cooling	Wet/Dry

PLANT COSTS X 10⁶

RTS Direct Costs ('80 \$)	180
RTS Indirect Costs ('80 \$)	<u>15</u>
Total RTS ('80 \$)	195
BOP Direct Costs ('80 \$)	330
BOP Indirect Costs ('80 \$)	210
Contingency ('80 \$)	<u>35</u>
Total BOP ('80 \$)	575
Total Plant Base Costs ('80 \$)	770
Escalation (for '95 Startup)	750
Interest During Construction ('95 Startup)	<u>500</u>
Total Capital ('95 Startup)	2020

OPERATING COSTS (MILLS/KW-HR - '95 STARTUP)

Capital	74
O&M	14
Fuel LEU/Th (20%) - Once-Through	<u>40</u>
Total	128
Fuel HEU/Th (93%) - Recycle	<u>29</u>
Total	117

TABLE 3.4-2
HTGR-GT COST COMPARISONS

	<u>HTGR-GT Dry/Wet- Cooled</u>	<u>HTGR-GT Binary</u>	<u>PWR Wet- Cooled</u>	<u>Coal Wet- Cooled</u>	<u>HTGR-SC Wet- Cooled</u>
<u>Plant Parameters</u>					
Power Rating - MW(t)	2000	2170	2400	2200	2240
Net Electrical Output - MW(e)	800	998	800	800	858
Efficiency	40	46	33	36	38
<u>Plant Costs (x 10⁶)</u>					
Direct Costs ('80 \$)	560	628	509	412	552
Indirect Costs ('80 \$)	210	234	227	85	211
Total Base Costs ('80 \$)	770	862	736	497	763
\$/KW(e) ('80 \$)	965	864	920	621	889
Escalation	750	839	736	562	753
Interest During Construction	500	559	442	212	480
Total Investment	2020	2260	1914	1271	1996
\$/KWe ('95 \$)	2525	2264	2392	1588	2326
<u>Power Costs (Mills/KW-hr) ('95 \$)</u>					
Capital	74	66	70	47	68
O&M	14	15	14	16	14
Fuel LEU/Th (20%) - Once-Through	40	35	38	96	37
Total	128	116	122	159	119
Fuel HEU/Th (93%) - Recycle	29	25	30		27
Total	117	106	114		109
<u>Cooling System Cost Adders*</u> <u>(Mills/KW-hr) ('95)</u>					
100% Dry Cooling	2	N/A	17	14	14
Dry/Wet - 115 x 10 ⁶ gal/yr	0	N/A	11	7	7
Dry/Wet - 500 x 10 ⁶ gal/yr	0	N/A	9	5	5
100% Wet Cooling	2	0	0	0	0

*Reflects cooling system capital plus operational penalties for cooling system performance at typical arid site: Modesto, California.

It is also interesting to compare the dry-cooling performance of the HTGR-GT against that of the HTGR-SC. Using the cooling system cost adders for the HTGR-SC shown in Table 3.4-2 for a total dry-cooling scenario, the advantage of the HTGR-GT ranges from 3% to 6%, depending on the fuel cycle scenario. With a 500×10^6 gal/yr water consumption constraint, the HTGR-GT is at a disadvantage of 2-3%. This differential is well within the cost estimate uncertainty range; therefore, it appears that the HTGR-GT has limited economic advantage with dry or wet/dry cooling over the LWR and essentially none over the HTGR-SC.

Another incentive for the HTGR-GT is its ability to operate at higher efficiencies. Table 3.4-2 includes the estimate of power costs for a binary cycle HTGR-GT, i.e., one that uses an ammonia bottoming cycle for utilization of waste heat. This system has marked effects on HTGR-GT economics as the capital costs in terms of \$/KW(e) drop by approximately \$100/KW(e) and the total plant efficiency increases to 47%, resulting from a net plant output of 998 MW(e). When compared with the dry/wet-cooled HTGR-GT, the binary version has power costs 10% lower. However, when compared to the wet-cooled HTGR-SC and the LWR, it has total power costs of only 5-7% lower, which are within the uncertainty bandwidth of the cost estimates.

If the HTGR-GT plant reject heat is utilized for district heating the cycle efficiency approaches 60%. The low grade heat can be economically transported 10-15 miles. This combined with higher cycle efficiency makes the HTGR-GT combined with district heating an attractive energy system for future population centers.

The above comparisons lead to the following conclusions:

- The HTGR-GT binary cycle should be utilized instead of the dry/wet-cooled HTGR-GT where sufficient water exists for wet cooling.
- For sites where dry or dry/wet cooling is required, the HTGR-GT provides minimal economic advantage over the HTGR-SC and the LWR. If neither of the latter options were to be available, the HTGR-GT provides an economic competitor to a coal-fired steam plant.
- The binary cycle HTGR-GT provides minimal economic incentive over the HTGR-SC and LWR.
- The HTGR-GT combined with district heating represents a high potential future market for the HTGR-GT.

The above conclusions have been drawn from the latest available cost estimates. While it is recognized that significant uncertainty exists in the LWR cost estimates due to unresolved cost trends resulting from Three Mile Island, the uncertainty in the HTGR cost estimates is considerably higher due to the conceptual nature of the present HTGR designs. For this reason, the HTGR cost estimates are considered to have a higher probability of future significant cost increases.

SECTION 3 REFERENCES

1. 2000 MWt HTGR Power Cycle Cooling System Alternatives, United Engineers & Constructors Inc. for Gas-Cooled Reactor Associates, June 1980.

4.0 DEPLOYMENT INCENTIVES - HTGR-GT COMMERCIAL PLANT

4.1 Market Assessment

The marketability and hence the market potential of the HTGR-GT can be assessed relative to the factors that influence members of the electric utility industry in the selection of power plants. While this market forecast is subjective because of the various factors that are examined, it does present a reasonable evaluation of the HTGR from the utility/owner perspective and gives a realistic view of the potential HTGR-GT market.

Forecasted Energy Demand

The projected market size for nuclear power plants between 2000 and 2020 is expected to be approximately 430 GW(e) (Ref. 4). Of this 430 GW(e), approximately 150-190 GW(e) will require dry or dry/wet cooling. This presents a sizable potential market for the HTGR-GT.

It is beyond the scope of this document to predict a market penetration rate for the HTGR-GT; however, because the lead plant will not be completed prior to 2003, it is doubtful that the HTGR-GT will be able to capture a significant share of the projected market prior to 2020. The status of competing technologies after 2020 cannot be predicted; therefore, the market penetration rate for the HTGR-GT after 2020 cannot be predicted at this time. Overall, Forecasted Energy Demand is considered to be a positive market factor for the HTGR-GT.

Siting Flexibility

The HTGR-GT offers increased advantages with regard to siting flexibility. In addition to the radiological advantages inherent to the HTGR, the ability for relatively efficient use of conventional dry-cooling technology gives the HTGR-GT an operational advantage over the current coal and LWR technologies for siting in areas where acute water shortages will occur. For this reason, this market factor is considered to be positive for the HTGR-GT.

Technical Development Status

Relative to the HTGR-SC, the HTGR-GT is several years behind in development. A large amount of work remains to be done to obtain and qualify materials which are suitable for long-term operation in the higher temperature environment of the HTGR-GT. Much work also must be done to qualify and test the turbomachinery, particularly to document the failure modes of the machine. The third area of technology advancement required over that of the HTGR-SC is in the area of fuels design. Particle coatings must be qualified for the 850°C core outlet temperatures to prevent unacceptable fission product release.

In the marketplace for the total 430 GW(e) of nuclear capacity between 2000 and 2020, the HTGR-GT will probably be competing against LWR

technology, which will have a very large advantage in reactor years of operating experience. For the estimated 150-190 GW(e) of dry or dry/wet nuclear capacity to be added in 2000-2020, the HTGR-GT will have to compete against the advanced dry-cooling technologies, which will also probably be available by this time period. As a result, the HTGR-GT may be competing against technologies for which water consumption constraints do not create operational or economic problems. This market factor must be considered as negative for the HTGR-GT.

Regulation and Licensing

Several major issues exist which could cause licensing delays in the lead plant. The costs to resolve these open licensing issues, however, cannot be estimated until a full-scale regulatory review is performed and further analyses and testing have been performed. The GCRA pre-review licensing program is designed to minimize the impact of such a review by providing continuous regulatory feedback into the design process.

Because the regulatory authorities have not had the opportunity to conduct an in-depth review of the current HTGR-GT design, design criteria and regulatory guides have not been generated which pertain to the unique aspects of the HTGR-GT. Based on the proliferation of regulations as the result of LWR operating experience, it is expected that the HTGR-GT will also cause new regulations to be written for its design and operation. The magnitude of the regulations should be considerably less than has been experienced with the LWR due to the generic nature of many of the existing regulations. Overall, after the lead plant has been constructed and operated, this market factor will have a neutral effect on the HTGR's marketability.

Commercial Status

The earliest commercial availability for the HTGR-GT is projected to be late in the first decade of the 21st century. Assuming that the present commercialization effort succeeds in building a lead plant in the 2000-2010 time frame, the industrial manufacturing base for the HTGR components will still need to be established. The manufacturing facilities that will be required to support a commercial HTGR-SC venture are also applicable for the HTGR-GT. The facilities for the manufacture of the turbomachinery which presently exist will require some retooling.

The commitment of the system supplier to the HTGR-GT will greatly affect the utility industry's perception of the HTGR's commercial status. The prospective owners will require contractual assurances as to the availability of field and home office technical assistance and support during the life of the plant as well as the availability of fuel and spare parts. Since these components are developmental, the commitment and/or ability to fulfill these requirements are not readily apparent at the present time, and are viewed as a risk.

Plant Capabilities

The operational capabilities of the HTGR-GT are still being investigated. Dynamic analyses are being performed to determine the controllability of and the interaction between the dual power conversion loops. The reactor core will exhibit the inherent characteristics of the HTGR, in particular, slow core heatup during operational transients. Overall, while the HTGR-GT possesses the capabilities inherent to HTGR technology, the unique operational aspects of the Gas Turbine have not been fully defined and evaluated. However, early studies have shown a unique capability to drop and resume power generation without protracted time intervals. In addition, the potential exists for high efficiency load following. For this reason, this is considered a positive market factor for the HTGR-GT at this time.

Economics

The comparative economics of the HTGR-GT and the plants with which it will compete were presented in Section 3.4. Based on these latest cost estimates, it appears that the HTGR-GT has essentially cost parity with the LWR and the HTGR-SC for both the dry- and wet/dry-cooling scenarios. Further, it appears that the high efficiency binary cycle HTGR-GT essentially has cost parity with the wet-cooled LWR and HTGR-SC. In addition, the uncertainties in the HTGR-GT cost estimates are rather high due to the conceptual stages of the design and, therefore, have a high probability of becoming larger. For these reasons, Economics is considered to be a neutral market factor for the HTGR-GT with a probability of becoming negative.

Capital Risk

The events at Three Mile Island II indicate the susceptibility of the large capital investments in an LWR plant to operational transients. The inherent design of the HTGR enhances protection of the investment in the plant by allowing longer operator response times to plant operational transients. While this advantage is manifested in the HTGR-SC, the ability of the HTGR-GT to provide the same degree of capital protection is not assured and has not been completely assessed. Because of economic considerations, the turbomachines have been located within the PCRV. A postulated shaft or disc failure (a highly unlikely event) could create high-energy missiles within the PCRV as well as cause a collapse of the pressure differentials across the turbine and compressor sections of the machine, thereby possibly causing internal PCRV damage. The risks of these types of transients are still being investigated; therefore, their consequences have not been fully quantified. For these reasons, this market factor can only be considered as neutral for the HTGR-GT at the present time.

Safety

The accident initiation and progression analysis (AIPA) results previously presented were based on the HTGR-SC design. A complete proba-

bilistic risk analysis has not yet been completed for the HTGR-GT. As a result of the additional accident sequences that are possible for the HTGR-GT, significant variations may occur between the AIPA results for the HTGR-GT and SC. Therefore, safety must be considered as a neutral market factor at the present time for the HTGR-GT.

Personnel Radiation Exposure

Section 4.2 examines the projected and the experienced HTGR-SC personnel exposure data. The major maintenance procedure which would cause the results of that analysis to differ for the HTGR-GT is the turbomachine removal, repair, and replacement. It is not possible to accurately quantify expected exposures from that activity at this time, but because of the level of fission product retention that is expected due to HTGR fuel design, this market factor is still considered to be positive for the HTGR-GT.

Fuel Cycle Flexibility

The performance of the HTGR relative to this nuclear market factor is examined in Section 4.2. While this factor is not currently a concern for utilities because of the present Administration's commitment to the once-through fuel cycle, it is expected that fuel cycle flexibility will become a major advantage to the HTGR in the early 21st century. For these reasons, this is considered a positive nuclear market factor for the HTGR-GT.

Advanced Applications

The advanced applications of the HTGR-GT serve to provide an incentive for the HTGR commercialization effort by exposing the unique markets which the HTGR may ultimately be able to serve. These advanced applications of the HTGR-GT, namely cogeneration and higher efficiencies through a bottoming cycle or higher temperatures, do provide an added incentive for government participation in the Program. It does appear, however, that more advanced technologies may satisfy these perceived future demands before the HTGR-GT can enter the market and that the economic incentive for higher HTGR-GT efficiencies may be limited.

Summary

While a sizable market is projected to exist for electricity production in the 2000-2020 time frame, a significant fraction of which will require dry or dry/wet cooling, the economic incentives for the development of the HTGR-GT to satisfy these markets do not appear to be sufficiently large at the present time to warrant the expenditure of significant funds for commercialization of the HTGR-GT. The economic data indicate that existing LWRs can satisfy the dry cooling market and that the HTGR-SC can outperform the LWR in these markets. In areas where water availability is of no concern, the HTGR-GT binary cycle plant has projected power costs essentially equivalent to the HTGR-SC

and LWR. While these conclusions are based on currently available cost estimates which have large uncertainties associated with them, the relative uncertainties of the HTGR-GT cost estimates are greater than for the LWR and coal cost estimates due to the relative immaturity of the HTGR-GT designs. It is expected that the HTGR cost estimates are susceptible to further large cost increases.

Because the economics are essentially at parity with the LWR, the HTGR-GT's technical development status will become the governing market factor in the reference time period. This factor will be viewed as negative because of the many years of operational experience that the LWR will possess which will give it a significant advantage in the dry-cooling market. Relative to the coal-fired unit, the economic advantage of the HTGR-GT can also be achieved by the HTGR-SC with much lower expenditures of development funds; therefore, the HTGR-SC becomes the HTGR-GT's chief competitor if HTGR technology achieves commercial status.

Based on the above, it is concluded that future work must produce significant cost and performance improvements in the HTGR-GT in order to justify the expenditure of the research, design, and development funding necessary to bring the HTGR-GT into the commercial marketplace.

4.2 HTGR Benefits

This section will assess the various characteristics of the HTGR which affect its ability to be operated safely as well as its perceived licensability. These characteristics will be assessed relative to the latest engineering information that is available to GCRA as well as the latest regulatory criteria.

4.2.1 Safety and Licensing

Proponents of the HTGR have long cited its inherent safety characteristics as a major advantage in licensing considerations. The evaluation of these characteristics by regulatory authorities has been limited to the review of the applications for construction permits for the Summit and Fulton HTGR generating stations. These applications were withdrawn prior to granting of the construction permits, but Summit had received a Limited Work Authorization.

These inherent safety characteristics of the HTGR are given in Table 4.2.1-1, which was taken from Ref. 1. The HTGR core is constructed exclusively of ceramic materials, primarily graphite, which maintain their integrity at very high temperatures, well above normal operating conditions. The core is designed with a low power density and strong negative temperature coefficient of reactivity, thereby creating relatively slow reactor temperature and power transients. In the event of loss of core cooling, the graphite acts as a heat sink. Interruptions of core cooling of approximately 30 minutes can be tolerated without any damage to primary system components. Approximately three hours is available to restore cooling before fuel damage or radioactivity release occurs.

Another inherent characteristic of the HTGR of importance to safety is the use of helium as the primary coolant. Helium cannot react with the core or reactor internals because it is chemically inert and remains in the gaseous phase. Because heat can be removed from the reactor core with any gas, even at low pressure, it is not necessary to maintain a full inventory of coolant in the reactor vessel during cooldown.

A passive safety feature of the HTGR is the prestressed concrete reactor vessel (PCRVR), which was introduced in gas reactors in Britain because of its safety characteristics. It is a structurally redundant concrete monolith which encloses the entire primary system. The strength and redundancy of the PCRVR are provided by a large number of steel tendons that run axially through and circumferentially around the vessel. The concrete acts as a neutron shield and is under compression; therefore, cracks are not subject to propagation.

In order to quantify the relative worth of these inherent safety characteristics, General Atomic Company (GA) performed the accident initiation and progression analysis (AIPA) study using probabilistic risk assessment methodology. It studied a wide spectrum of accident

TABLE 4.2.1-1

KEY INHERENT AND PASSIVE SAFETY CHARACTERISTICS OF THE HTGR

Inherent or Passive Feature	Relevant Properties	Safety Significance
Reactor core	High heat capacity	Slow transient response
	Low power density	More time for remedial measures, both within and external to plant
	Strong negative temperature coefficient	Fast-acting shutdown system not required
	Graphite cannot melt but may locally sublime	Structural integrity of core maintained for weeks following loss of cooling
	Coated particle ceramic fuel	Slow controlled release of volatile nuclides under no-cooling conditions
Helium coolant	Single-phase gas	No boiling, bubbles, liquid level, or pump cavitation problem; no added coolant inventory needed for core cooling, only forced circulation
	Neutronically transparent	No reactivity effects, no coolant activation
	Chemically inert	No chemical fuel/helium interactions
	Low stored energy	Reduced containment damage potential
PCRV	Multiplicity of tendons	Failure of individual structural members inconsequential
	Tendons shielded by concrete	Neutron embrittlement and subsequent fracture eliminated
	Concrete under compression	Tiny cracks self-sealing, do not propagate
	Massive robust structure	Effective retention of radioactivity; retains great fraction of heat escaping core
Containment	Not unique to HTGRs but particularly effective because of above-listed inherent features	

sequences which might result in release of radioactivity from a large HTGR Steam Cycle plant. A summary table of results of the AIPA is presented in Table 4.2.1-2 from Ref. 1. The AIPA study received peer review from several offices of the Nuclear Regulatory Commission (NRC), Brookhaven National Laboratory, Oak Ridge National Laboratory, Aerojet Nuclear Company, KFA in Julich, Federal Republic of Germany, and the Safety and Reliability Directorate of the United Kingdom Atomic Energy Authority. Generally, the comments did not change or contest the major conclusions of the study. Work is continuing to study new initiating events, fission product transport assumptions under accident conditions, and in other areas where the uncertainty bounds which were originally used are considered to require further refinement.

As part of the Nonproliferation Alternative Systems Assessment Program (NASAP) study, NRC submitted to the Department of Energy a list of 29 questions and comments on 8 topics concerning the safety and licensing documentation for the proposed large Steam Cycle HTGR design. These questions and the GA responses are presented in Ref. 14. The major topics and their responses are reviewed below:

- Use of Graphite as a Structural Material - The design criteria for graphite structures have not yet been completed or approved. A joint subcommittee of the American Concrete Institute and American Society of Mechanical Engineers (ASME) has been formed to generate a code section for graphite. Many of the items before the subcommittee require experimental verification which will be obtained from the ongoing base technology program. Tentative adoption of the code is at least a year away.

Graphite corrosion is another significant area of graphite research. Oxidation of the graphite occurs at high temperatures in the presence of water vapor. Experimental work to date indicates that oxidation under HTGR operating conditions causes a surface-predominated attack which can be allowed for in the structural analysis and design. GA's position is to design the graphite components so that the minimum safety factors required by the proposed design criteria will be available at the end of plant life. Design oxidation rates and Design Basis Events for water ingress into the PCRV have not yet been determined or approved.

- Core Seismic Response - NRC questions in this area centered on the seismic design criteria and the seismic analysis methods to be used. Several computer codes have been written which utilize test data for values used in the models. Large array tests have been performed to verify the codes and to give information on the characteristics of the core for design purposes. There are no major open licensing issues in this area.
- Fuel Transient Response - A large data base of information was compiled on highly enriched uranium (HEU-93%) fuel in a 750°C helium environment. The reference fuel for the Lead Project is LEU (20% enriched). As a result, much experimentation remains to

TABLE 4.2.1-2

RISK ASSESSMENT RESULTS FOR HTGR FROM AIPA STUDY

Accident Consequences ^(a)	Accident Frequency (reactor-year ⁻¹)	
	10 ⁻⁶	10 ⁻⁷
Early fatalities	<1	<1
Early illnesses	<1	<1
Property damage, \$ million	<1	2
Relocation area, sq miles	0	0
Decontamination area, sq miles	0	0.2
Latent cancer fatalities ^(b)	1	8
Thyroid nodules ^(c)	10	100
Genetic effects	<1	1

(a) Representative U.S. site.

(b) Beir Commission recommendations used.

(c) Sum of benign and cancerous nodules.

Source: Ref. 1

be done on LEU fuel particles and their properties, including fission product retention. As higher temperature applications are pursued, i.e., 850°C core outlet temperatures for the Gas Turbine and Reformer variants, new data will need to be generated for these fuels in order to meet the NRC licensing criteria.

- In-Service Inspection and Testing (ISI) - Section XI, Division 2 of the ASME Boiler and Pressure Vessel Code contains the proposed guidelines for ISI of HTGR components. The categories of affected components include those required for (a) shutdown heat removal, (b) control of nuclear reactivity, (c) detection or control of chemical ingress, and (d) controlled primary coolant depressurization. An open question in this area is the requirement for the possible ISI of the PCRV liner. GA's current position is that a thermally insulated liner will not require ISI. This remains to be confirmed by the NRC. NRC did, however, require ISI of the core support structure for the Fulton HTGR.
- Primary System Integrity - Corrosion effects within the primary loop will be insignificant because of the inert helium environment except in two potential areas: metal carburization in the top head and oxidation in the lower graphite core support blocks due to impurities in the helium. The carburization problem increases with temperature and is, therefore, of more concern in the 850°C core outlet applications of the HTGR. Research in these areas is continuing to establish appropriate design criteria.

The design bases for the design of the PCRV closures are not yet approved by the NRC. Most closures are designed and fabricated to ASME Code Section III, Division 1. In previous licensing efforts, these closures have utilized flow restrictors to limit the free flow area to 100 in² or less in the event of a failure. LWRs are not required to assume failure of Class-1 pressure vessels; therefore, the GA position is that the assumption of such failures for the HTGR is excessive and should not be considered as a Design Basis Accident, provided the penetrations and closures are not operated at temperatures above those at which ASME Section III applies. Steel closures whose temperatures exceed those allowed by the low-temperature provisions of the Code may be used at steam pipe penetrations. These are designed to meet the rules of high-temperature code cases and utilize flow restrictors. Prestressed concrete closures used for large heat exchanger cavities are designed and constructed to ASME Code Section III, Division 2. Due to their redundant prestressing elements, GA considers their gross failure to be incredible and, therefore, precludes rapid depressurization due to their failure.

Another item concerning primary system integrity is acoustic excitation within the primary system. For the Steam Cycle and Reformer variants, the primary source of this acoustic excitation is the main circulators. In the Gas Turbine plant, the turbo-machinery will generate much larger noise levels which may affect

the liner insulation. Studies are continuing in this area to model the acoustic propagations as well as to determine the long-term effects on reactor internals. This is a major area of uncertainty for the HTGR-Gas Turbine (GT).

- Emergency Core Cooling Provisions - During the Fulton and Summit Licensing process, the NRC treated the core auxiliary cooling system circulators and shutoff valves as prototypical items which deserved special testing programs. GA still must develop core auxiliary cooling system testing criteria for preoperational design verification and on-line testing. Also, a computer program must be developed for assessing the stability margin of the core auxiliary heat exchanger. While these are still open licensing issues, they are not expected to impact overall plant licensability.
- Anticipated Transient Without Scram (ATWS) - The subject of ATWS remains an unresolved licensing and design issue. There have been some preliminary studies of HTGR-Steam Cycle ATWS to support earlier licensing efforts, but they were directed toward NRC interpretation of LWR ATWS requirements. Work remains to be done to resolve the ATWS issue for the HTGR on the basis of its inherent safety features. This issue is not expected to impact overall plant licensability.

The safety and licensing issues discussed above are applicable to the generic HTGR design and were developed from reviews of the Steam Cycle concept. The direct cycle or Gas Turbine version of the HTGR has additional major safety and licensing issues which result from the following major design differences:

Relative to the Steam Cycle HTGR, the Gas Turbine has:

- An increase in core outlet temperature from 700°C to 850°C.
- 50% higher operating pressures and differential pressure across the core.
- Potentially higher fission product release due to higher temperatures.
- 50% higher primary coolant flow rates.
- The turbomachinery located within the PCRV.
- A rotating shaft penetrating the PCRV.
- Precoolers and recuperators which require ASME qualification.
- The possibility of turbomachine lubricant leakage into the primary system.
- The possibility of overstressing the thermal barrier due to the effects of rapid internal pressure transients.

The safety significance of several of these items is discussed in more detail below:

- Shaft Seal Failure - The turbomachine/generator shaft penetrates the primary coolant system boundary. Failure of the seal can cause rapid depressurization of the PCRV. The present design also calls for location of the generator outside the containment building, thereby adding an additional rotating shaft seal in the containment wall. A postulated failure which could cause loss of integrity of both the PCRV and containment rotating seals would be unacceptable and remains a major licensing concern.
- Internal Pressure Equilibration Accidents - During normal operation of the HTGR-GT, large pressure differentials exist across the turbine and compressor sections of the turbomachine (~685 and 743 psia respectively). A failure of the turbomachine which would cause the collapse of these differentials is called an Internal Pressure Equilibration Accident. Designers have recently completed an in-depth study of the licensing effects of such an accident and have concluded, based on existing turbine and turbomachine failure rate data, that a defensible failure rate for the turbomachine is 10^{-4} per machine-year. Based on this probability, the catastrophic failure of the turbomachine must be assumed as a Design Basis Event. The study also concluded that the consequences of such an accident strongly depend on the assumed rate of pressure differential collapse. A conservative assumption that the turbine will completely deblade in one revolution at 42% overspeed, or .012 seconds, results in a depressurization rate in the core outlet plenum of 6310 psi/sec over .01 seconds (Ref. 15). This speed assumes failure at the free-free critical speed. The stresses imposed on the reactor internals by this accident must be combined with the stresses imposed by the safe-shutdown earthquake because of its classification as a Design Basis Event. The results of preliminary analyses show that redesign of the core support posts, permanent side reflector posts, core auxiliary cooling system components and the core support structure will be required if the above design assumptions are utilized. A testing program is required to reduce the conservatism in the above assumptions and establish a more appropriate and defensible design basis.
- Turbomachine Missiles - In addition to causing rapid pressure transients, turbomachine failures can create high energy missiles. The design of missile shields to contain blade and rotor fragments is under way; however, these designs will be affected by the results of the turbomachine failure testing.
- Core Auxiliary Cooling System Design - The turbomachine forms the principal resistance to reverse flow through each power conversion loop. The design basis for the core auxiliary cooling system flow requirements assumes a single turbomachine failure which reduces the loop flow resistance. This design basis and the

amount of resistance that can be assumed after a turbomachine failure must be analyzed and accepted.

- Overpressure Protection - Unlike the HTGR-Steam Cycle (SC), there is no identified source for overpressurization of the PCRV in the HTGR-GT. It is proposed that overpressure protection as required by the ASME Code be provided by internal pressure relief by using safety-grade valves in each power conversion loop to regulate pressure from high- to low-pressure portions of the loop. It is not anticipated that this issue will impact the overall plant licensability.

4.2.2 Water Utilization

One of the major claims of HTGR proponents has been the assertion that the HTGR can be cooled efficiently while consuming less water than current LWR or fossil-fired plants. This claim has been presented as a major incentive for the development of the HTGR-GT. Several studies in the past have shown that, indeed, the HTGR-SC with wet cooling consumes 15% to 25% less water than a wet-cooled LWR under similar site conditions due to its higher thermodynamic efficiency and lower reject heat load (Ref. 2). A study performed in 1975 by General Electric Company for the Electric Power Research Institute (Ref. 3) attempted to quantify the cooling performance of the HTGR-GT relative to other nuclear systems. It also projected areas of the country where some form of dry cooling would be needed by the year 2000. The results of this study were documented in the GCRA HTGR Market Assessment - Interim Report. In order to better quantify the size of the potential dry-cooling market in the 2000-2020 time frame, which is where the HTGR-GT is targeted for commercialization, GCRA enlisted the Hanford Engineering Development Laboratory (HEDL) because of their Water Use Information System and its capabilities. The results of the HEDL study (Ref. 4) are presented in this section.

The study utilized the projections of the Water Resource Council Second National Water Assessment and the HEDL data base to identify aggregated subareas (ASA) where surface water deficiencies would exist by the year 2020. These areas are shown in Fig. 4.2.2-1. The study then utilized the econometric model to derive a projected electric energy demand by National Electric Reliability Council (NERC) region in the years 2000, 2010, and 2020. This case was taken as the median, and high and low cases were also generated. They are shown in Table 4.2.2-1. Two separate techniques were used to determine where the capacity would be located to serve the projected demands. The first technique located the new capacity in existing power generation areas (PGA). This is consistent with the theory that many future plants will be located on existing power plant sites. The second technique used a population-weighting factor which located the new capacity in proportion to projected population location. Projected growth rates for cogeneration, and retirements of existing plants were calculated to project the total capacity produced by central power plants that would need to be constructed during the period in each ASA. Overlay techniques were used to determine the fuel mix of the capacity in each

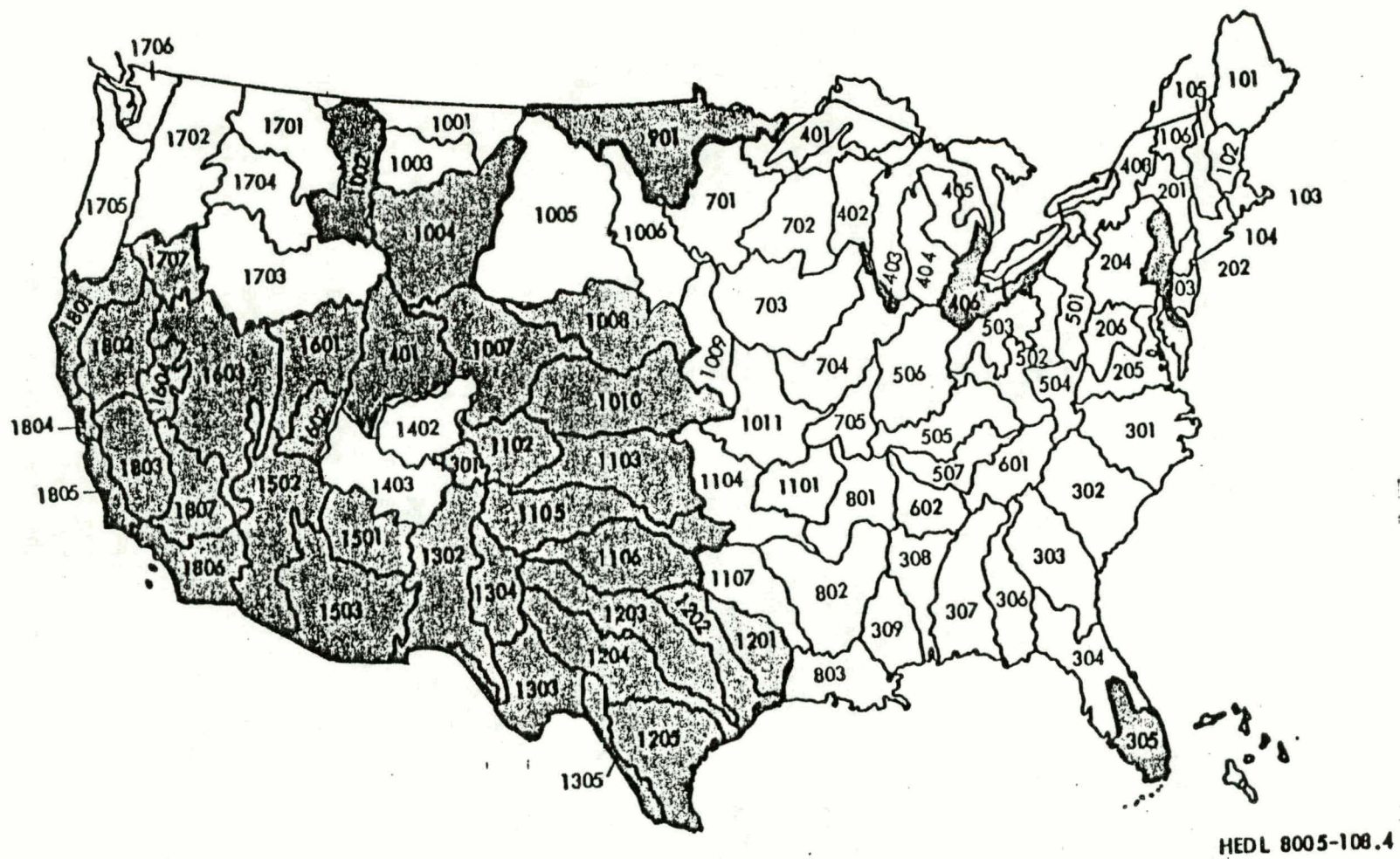


Figure 4.2.2-1 Location of Surface Water Deficient ASAs Forecasted for the year 2000

TABLE 4.2.2-1

Forecast of Installed Capacity by NERC Region (GW)High-Case Scenario

<u>Region</u>	<u>2000</u>	<u>2010</u>	<u>2020</u>
ECAR	242	339	436
ERCOT	94	131	169
MAAC	83	116	150
MAIN	117	164	211
MARCA	60	84	108
NPCC	95	171	306
SERC	296	414	533
SWPP	156	218	281
WSCC	226	317	407
Total	1370	1954	2466

Low-Case Scenario

<u>Region</u>	<u>2000</u>	<u>2010</u>	<u>2020</u>
ECAR	211	246	276
ERCOT	82	96	107
MAAC	73	85	95
MAIN	102	119	133
MARCA	53	62	69
NPCC	83	97	109
SERC	259	302	339
SWPP	137	160	179
WSCC	198	231	259
Total	1200	1398	1566

ASA. Coal, geothermal, solar, and biomass projections were made. Nuclear projections were based on a total installed nuclear capacity of 250 GW(e) in 2000 and 600 GW(e) in 2020. The end result is a projection of the number of GW(e) of new electrical capacity that will require either totally dry or dry/wet cooling in each ASA for three growth scenarios. These requirements are further defined by projections for fossil, nuclear, and other heat sources. Fig. 4.2.2-2 summarizes these findings using the PGA disaggregation technique. Fig. 4.2.2-3 presents the results using the population-weighting technique. It can be seen that even in the low-growth scenario in Fig. 4.2.2-2, some form of dry cooling will be required for 170 GW(e) of capacity. For the median-growth case, 257 GW(e) will require some form of dry cooling. Fig. 4.2.2-4 shows the location of capacity additions requiring some form of dry cooling using the PGA technique for the base case. The values shown are the number of GW(e) requiring some form of dry cooling. Fig. 4.2.2-5 shows the same information using the population-weighting technique.

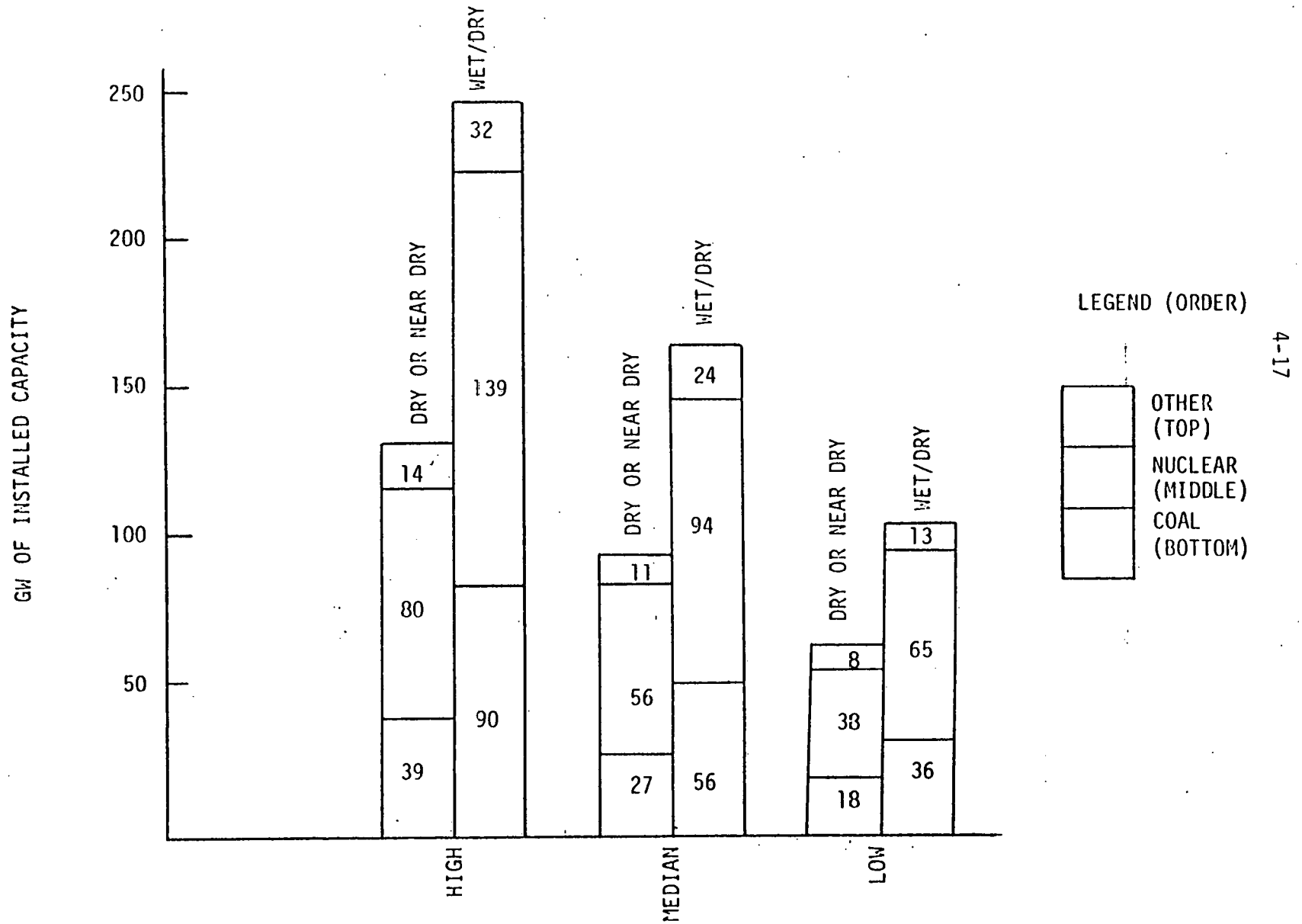
Based on the above data, there appears to be a substantial market for dry cooling technology after the year 2000. In order to evaluate the market potential of the HTGR-GT based on these findings, GCRA contracted with United Engineers & Constructors Inc. (UE&C) to perform an economic evaluation comparing the HTGR-GT to a comparably sized LWR and coal plant, all at a common site with cooling systems optimized for given water consumption constraints. The parameters of these plants are given in Table 4.2.2-2. Water consumption was constrained in these analyses because the HEDL study indicates that water shortages will be manifested by constraints on water consumption and not by higher water prices (Ref. 4). All analyses were performed for a site at Modesto, California, because it exhibits characteristics which are representative of many of the areas where dry cooling will be required; however, cooling system performance and evaluated costs are extremely site-dependent. The results obtained at the Modesto site, while valid only at that particular site, are considered to be representative of the results that would be obtained elsewhere in the dry cooling market areas.

UE&C designed an optimized cooling system for each plant with a given water constraint. Six cost penalty categories were then evaluated against the system performance with site-specific temperature variations. These penalties were evaluated on an annual basis, capitalized over the plant lifetime, and added to the capital cost of the cooling system. The sum of the capital cost and the capitalized penalty costs is called the total evaluated cooling system cost.

The results of the analyses are shown graphically in Figs. 4.2.2-6, 7, and 8 for the fossil plant, the LWR, and the HTGR-GT respectively and are based on the economic assumptions shown in Table 4.2.2-3. For totally dry-cooled plants, the total evaluated cooling system costs of the fossil plant are 127% higher than for the HTGR-GT, while the LWR costs are 193% higher. These higher costs are due primarily to their high operating penalties associated with the Rankine Cycle plants with dry cooling during periods of high ambient temperature. The implica-

Figure 4.2.2-2

Advanced Cooling Requirements 2000-2020 (Disaggregation Based Upon Power Generation Area)



Advanced Cooling Requirements 2000-2020 (Disaggregation Based Upon Population)

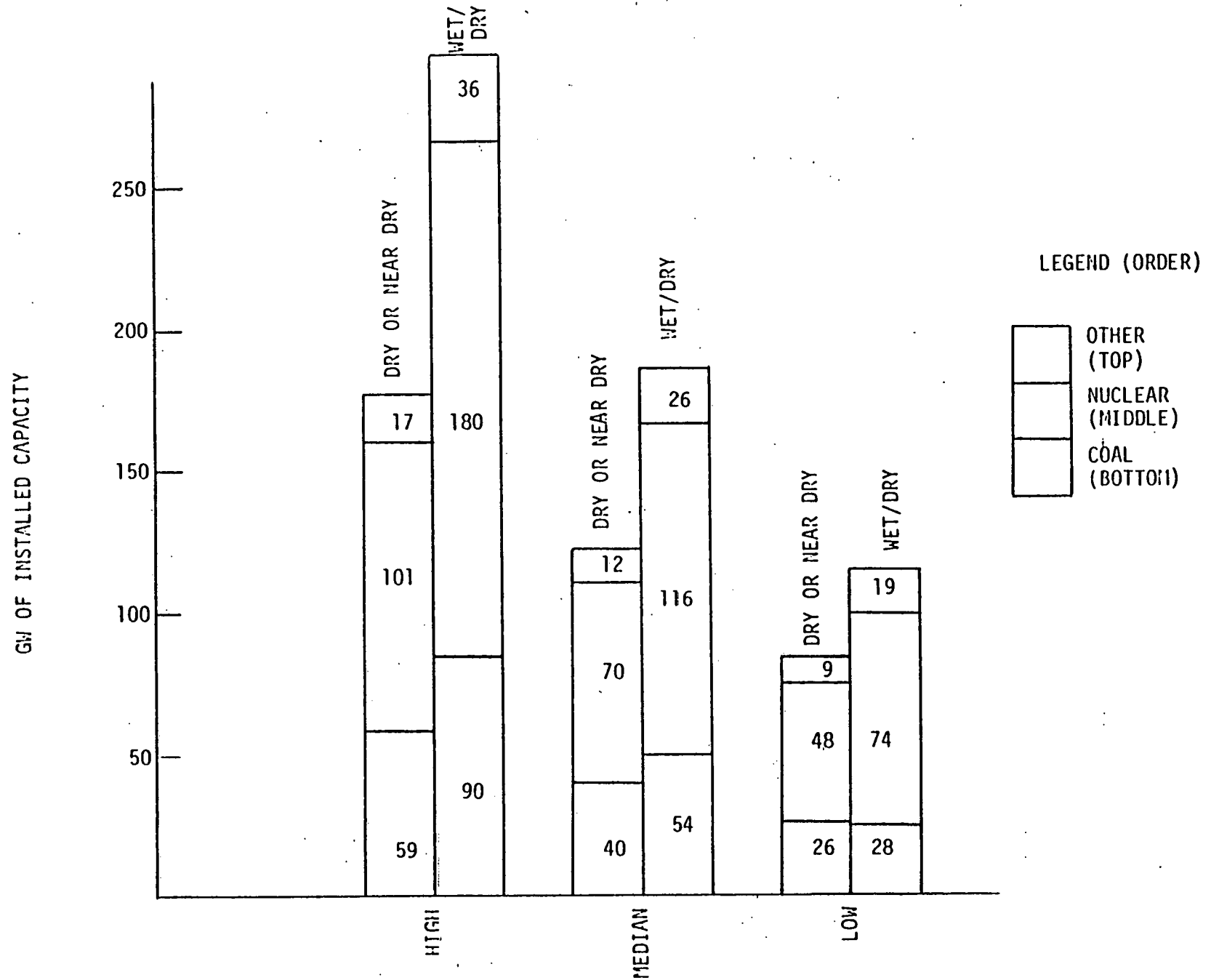
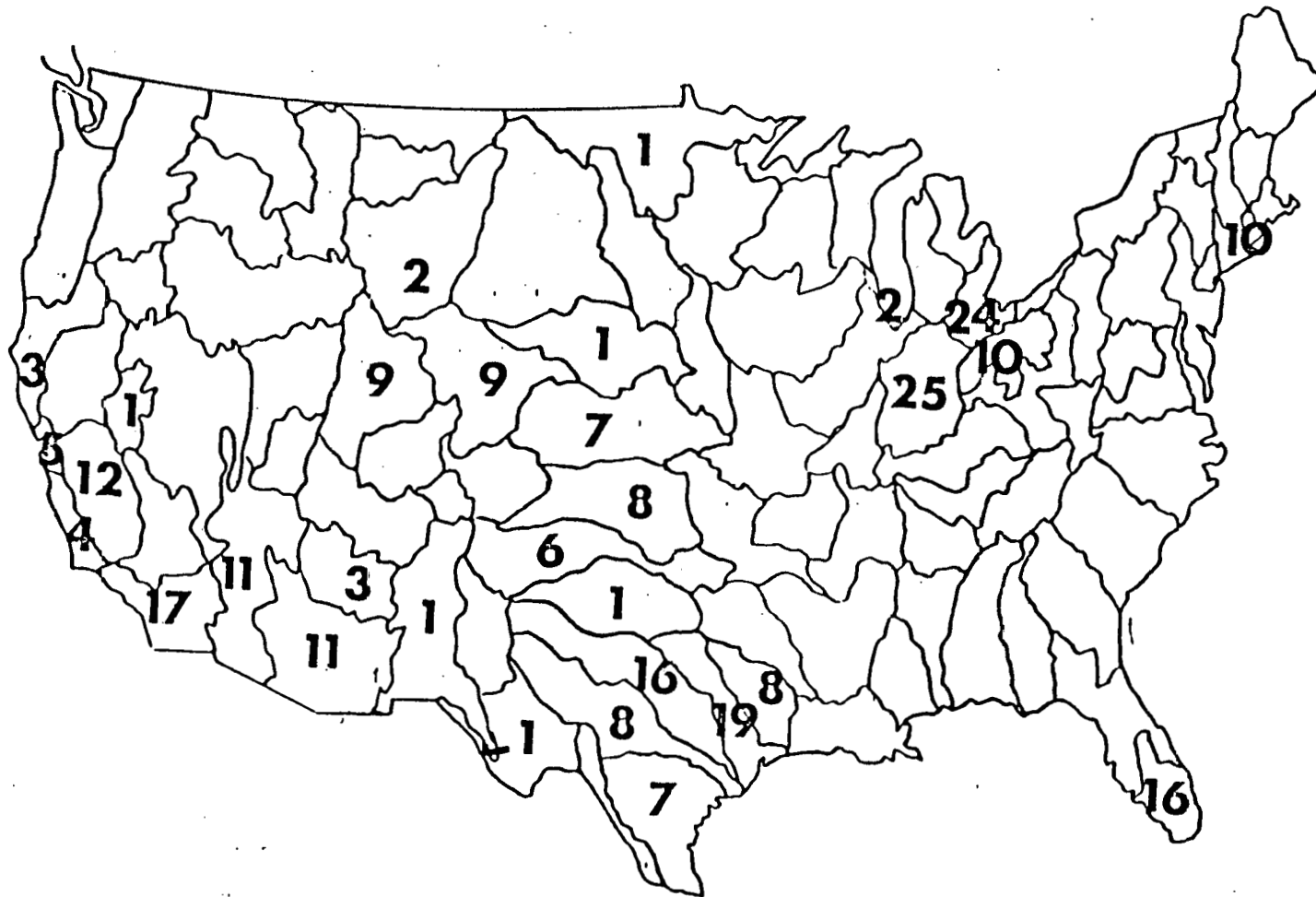


Figure 4.2.2-4

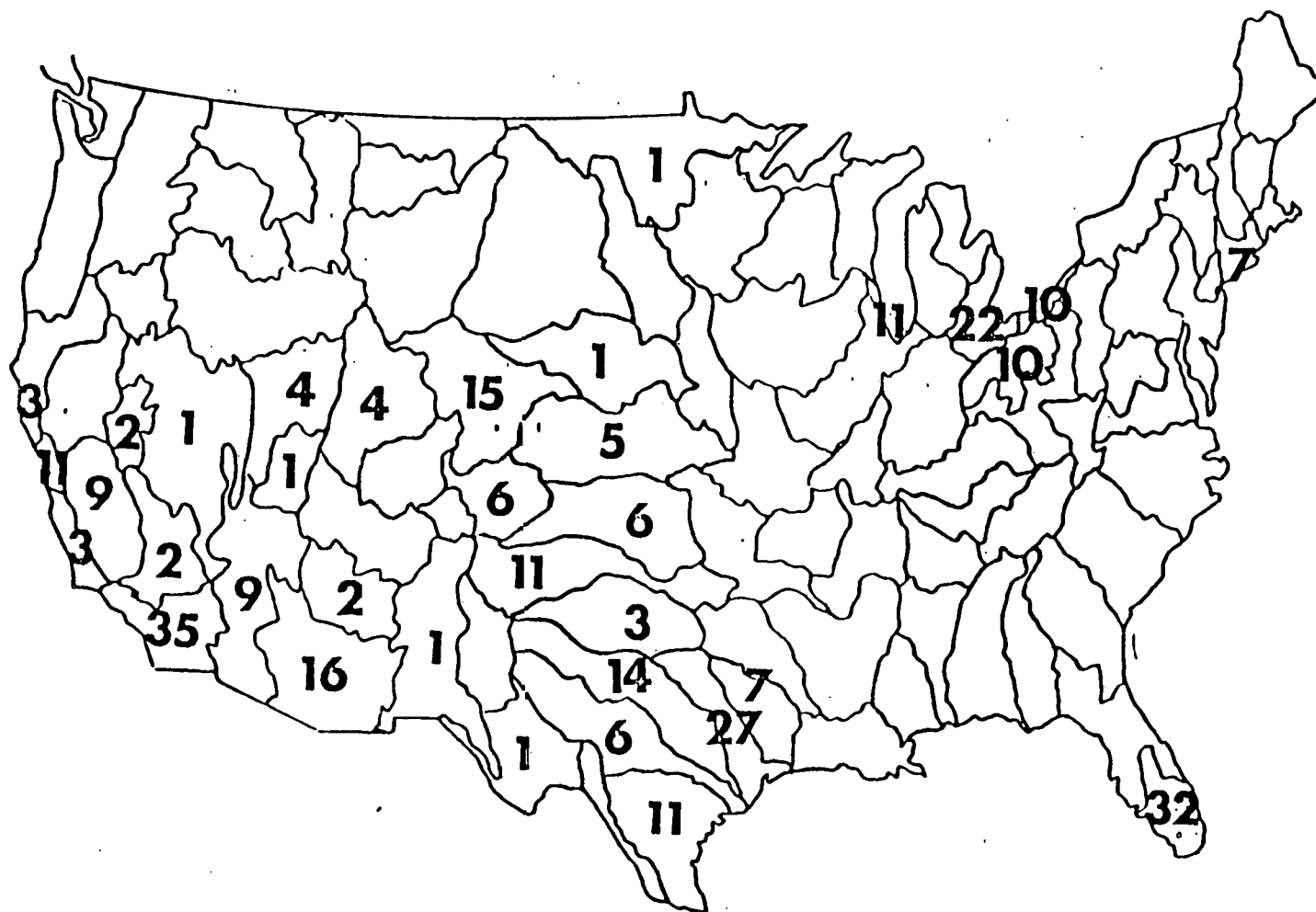
Areal Distribution of Forecasted Need for Advanced Cooling:
Median Growth Scenario, Power Generation (GW)



Note: Numbers indicate GWe of capacity requiring dry or dry/wet cooling

Figure 4.2.2-5

Areal Distribution of Forecasted Need for Advanced Cooling:
Median Growth Scenario Population (GW)



Note: Numbers indicate GWe of capacity requiring dry or dry/wet cooling

TABLE 4.2.2-2
 PLANT PARAMETERS FOR COOLING SYSTEM
 ECONOMIC ANALYSIS

	<u>LWR</u>	<u>Coal</u>	<u>HTGR-GT</u>
Rated Power - MW(t)	2896	2022	2000
Gross Station Heat Rate - BTU/kw-hr	9760	8025	8448
Gross Electrical Output - MW(e)	830	858	808
Net Thermal Efficiency	33.4	39.6	40.0
Net Electrical Output - MW(e)	800	800	800

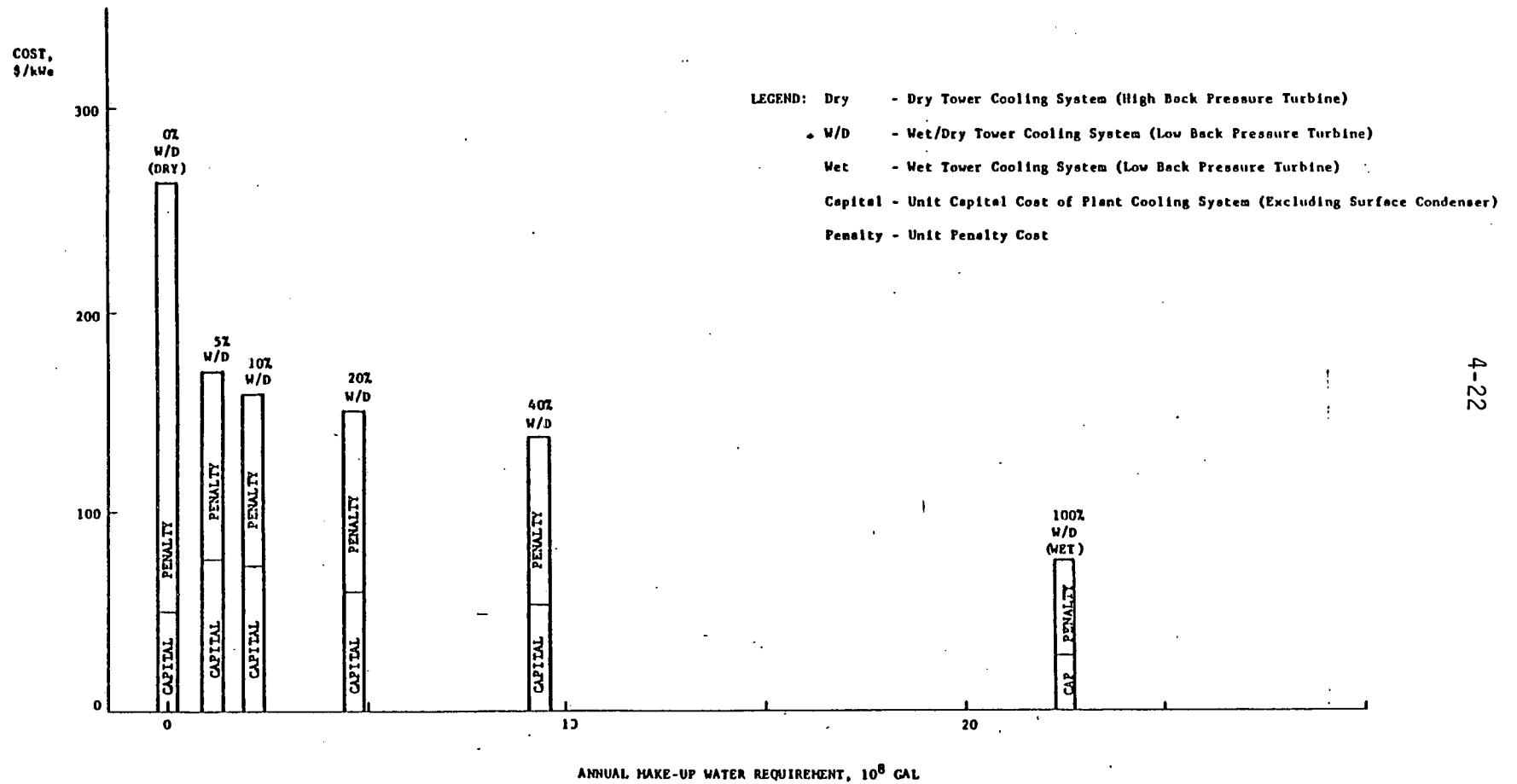


Figure 4.2.2-6 Cost Characteristics as a Function of Make-Up Water Requirements for Cooling Systems Designed for an 855-MW(e) Fossil Plant at Modesto, California

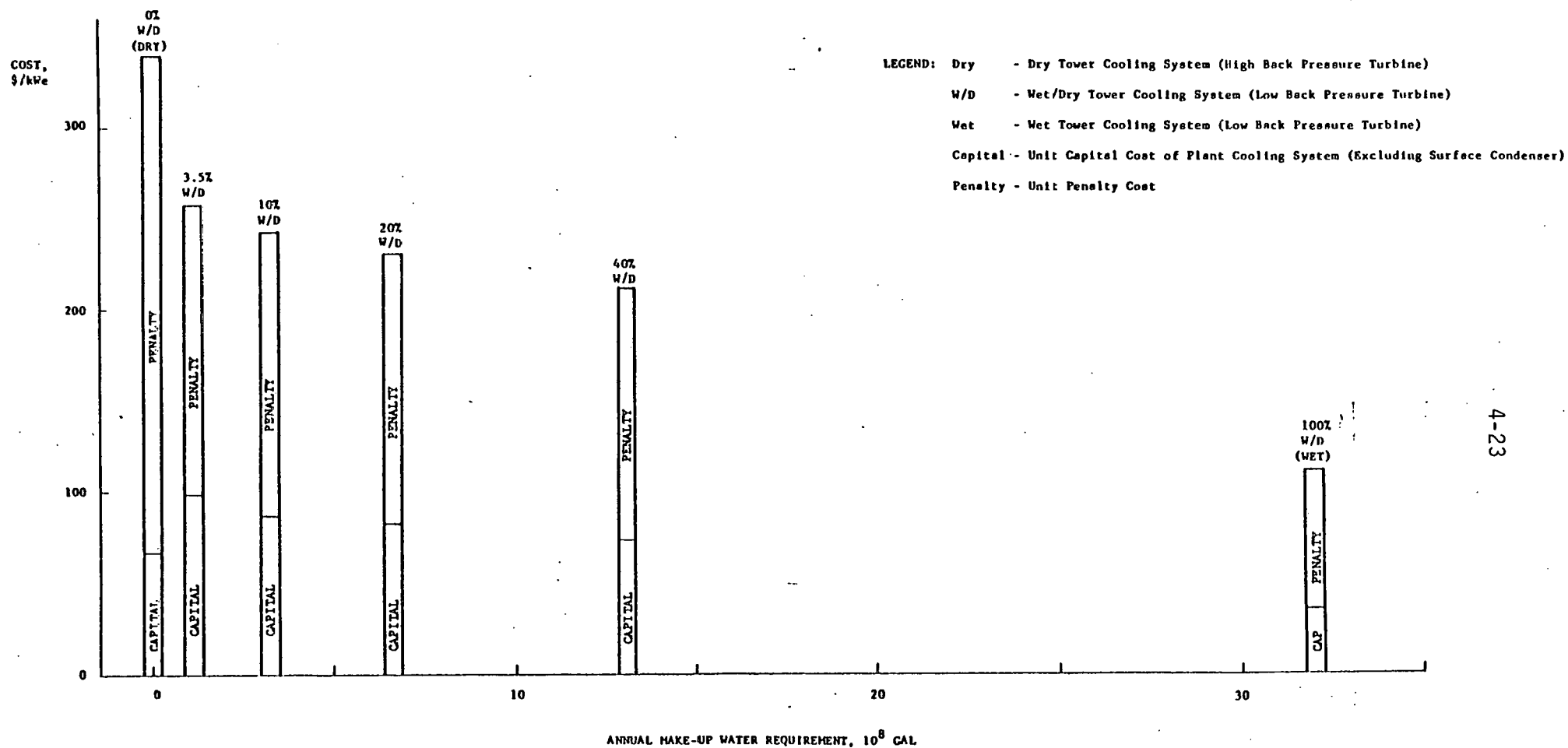


Figure 4.2.2-7 Cost Characteristics as a Function of Make-Up Water Requirement for Cooling Systems Designed for an 838-MW(e) PWR Plant at Modesto, California

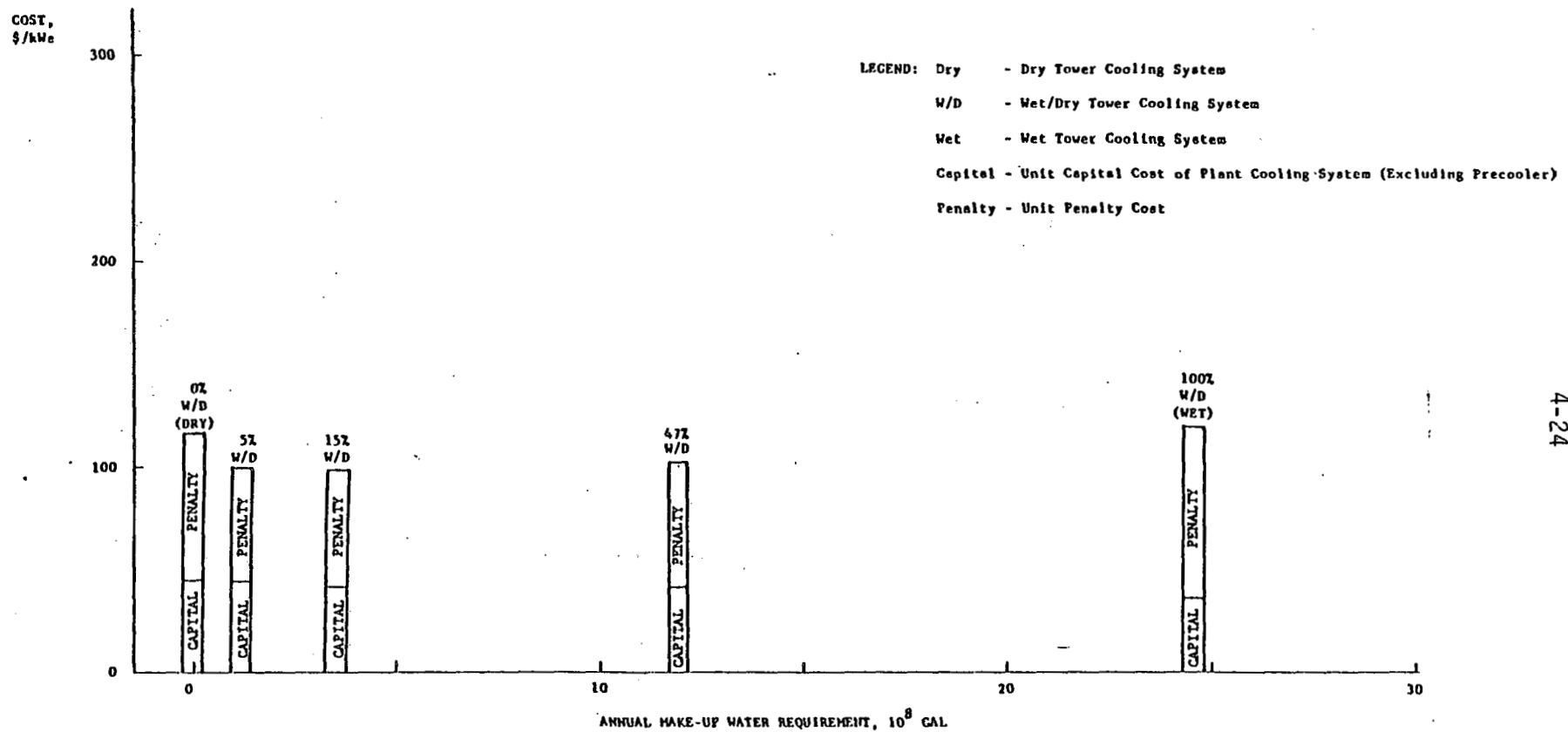


Figure 4.2.2-8 Cost Characteristics as a Function of Make-Up Water Requirement for Cooling Systems Designed for an 808-MW(e) HTGR-GT Plant at Modesto, California

TABLE 4-2-2-3
ECONOMIC FACTORS FOR COOLING SYSTEM ANALYSIS

Pricing Date	January 1980
Average Plant Capacity Factor	70%
Annual Fixed Charge Rate	18%
Plant Book Life	30 years
Capacity Penalty Charge Rate [\$/KW(e)]	
Base	621
Low	311
High	1242
Replacement Energy Cost (Levelized mills/kw-hr)	
Base	40.8
Low	20.4
High	122.5
Water Cost (Levelized \$/1000 gal)	
Base	1.0
Low	0.5
High	2.0

tion is that if any water is available for consumption, a peak-shaved, dry/wet system should be used to decrease the penalties. Fig. 4.2.2-9 compares the relative evaluated costs of the HTGR-GT against the LWR and the coal plant as a function of water available for annual consumption. The dashed lines indicate extrapolations from the specific optimization points which are marked by "+." Optimizations were not performed in the dashed-line region because dryness ratios lower than 50/50 do not conserve sufficient water to warrant their use. From Fig. 4.2.2-9, the following conclusions can be drawn:

- All three plant types can be dry- or dry/wet-cooled. The Rankine Cycle plants must utilize a high back pressure turbine with total dry cooling in order to obtain the performance shown.
- The HTGR-GT can utilize dry or dry/wet cooling more efficiently, hence more economically, than the LWR or fossil-fired plants.
- The most economical cooling system for the HTGR-GT is a wet/dry system, while all-wet cooling is preferred for the LWR and fossil plants.
- The magnitude of the HTGR-GT advantage is a function of water availability. Generally, the less water that is available for consumption, the larger the advantage of the HTGR.
- The total evaluated cooling system costs of the HTGR-GT with wet/dry systems are relatively insensitive to cooling water availability when compared to the fossil plant and the LWR sensitivities to water availability.

In order to check the sensitivities of the above results, UE&C performed low- and high-case studies for each type of plant using the factors shown in Table 4.2.2-3. The results are summarized in Figs. 4.2.2-10, 11, and 12 for the fossil, LWR, and HTGR-GT plants respectively. By comparing these figures, it can be seen that the HTGR-GT is relatively insensitive to variations in the three sensitivity variables. Table 4.2.2-4 summarizes these sensitivities for the fossil and LWR plants relative to the HTGR-GT with water consumption constrained to 115×10^6 gal/yr, which corresponds to a 95% dry/wet system for the HTGR-GT.

The effects of varying individual parameters were also examined for each plant in Ref. 5. The following conclusions can be drawn from the sensitivity analysis:

- The HTGR-GT maintains its advantage over the LWR and the fossil plant over a wide range of economic variables.
- As replacement power costs increase, the HTGR-GT becomes more economic with 100% dry cooling than with wet/dry cooling. This increases the HTGR-GT advantage with all-dry cooling over its competitors.

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ECONOMIC FACTORS

Pricing Date	January, 1980
Average Plant Capacity Factor	70%
Annual Fixed Charge Rate	18%
Plant Life	30 Years
Capacity Penalty Charge Rate	\$621/kW
Replacement Energy Cost (Levelized)	40.8 Mills/kWh
Water Cost (Levelized)	\$1.0/1000 Gal

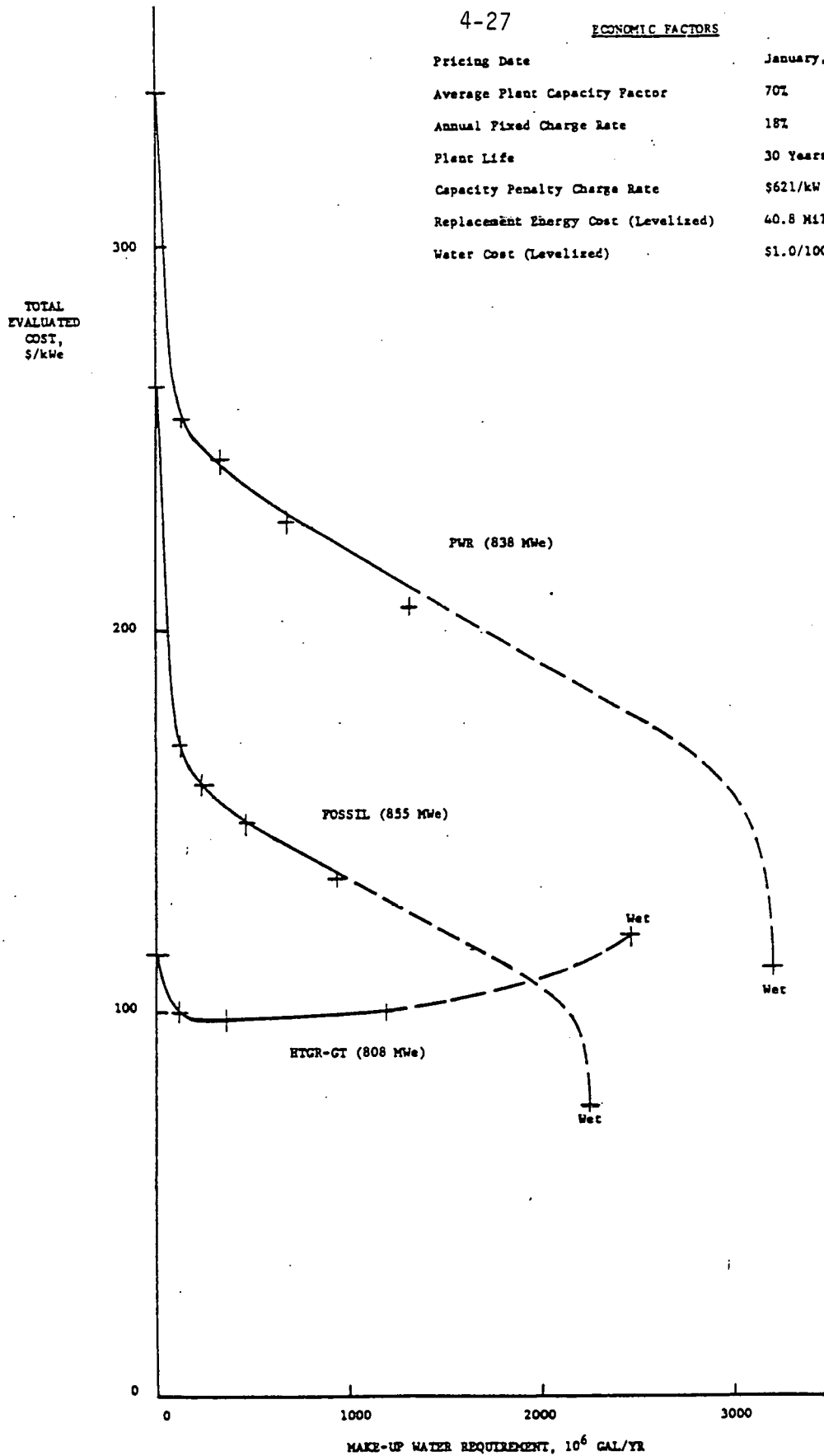


Figure 4.2.2-9 Comparison of Total Evaluated Cost Characteristics of 800-MW(e) (Nominal) Power Plants at Modesto, California

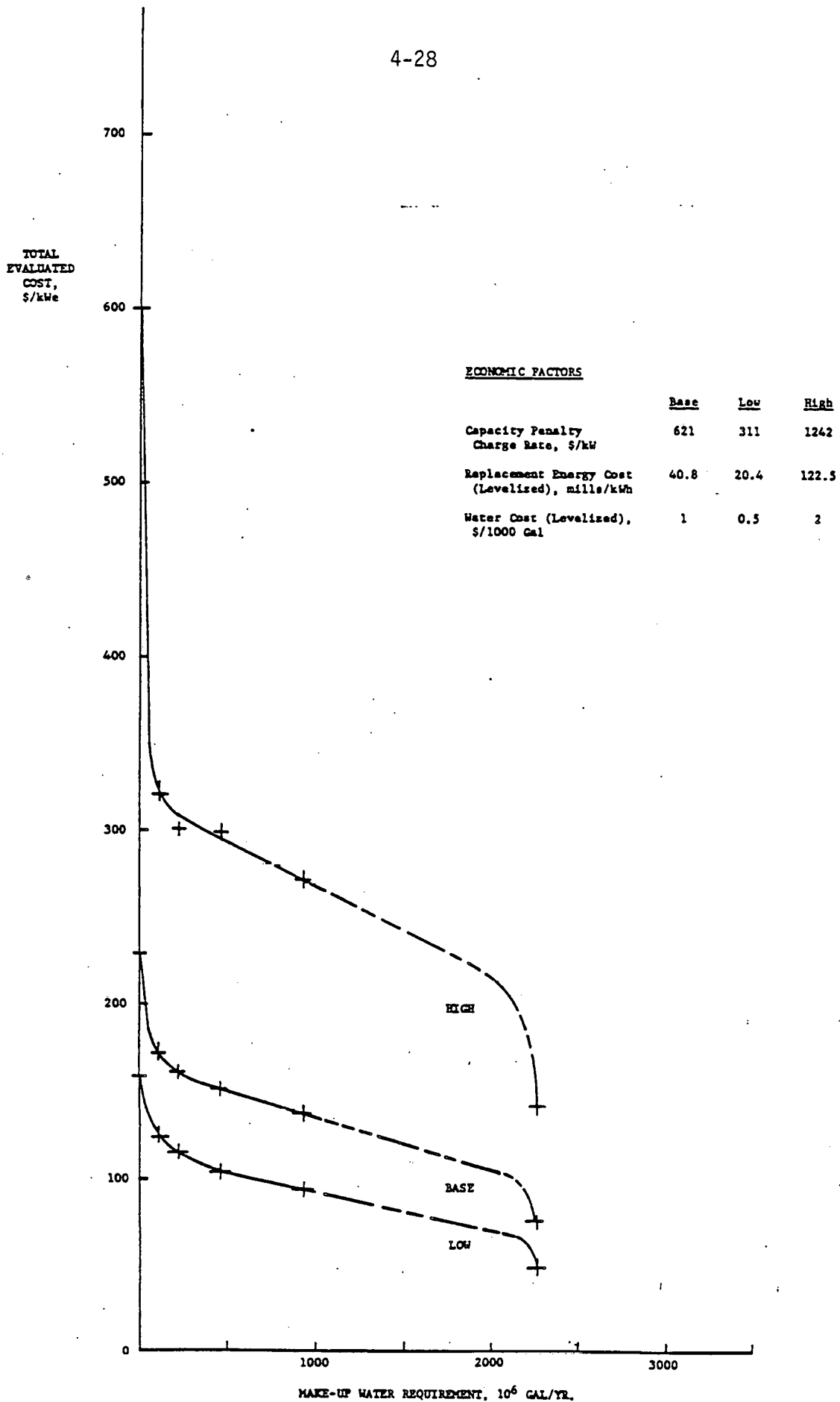


Figure 4.2.2-10 Effects of Composite Changes of Economic Factors on the Total Evaluated Costs of Alternate Cooling Systems for an 855 Fossil Plant

TOTAL
EVALUATED
COST,
\$/kWe

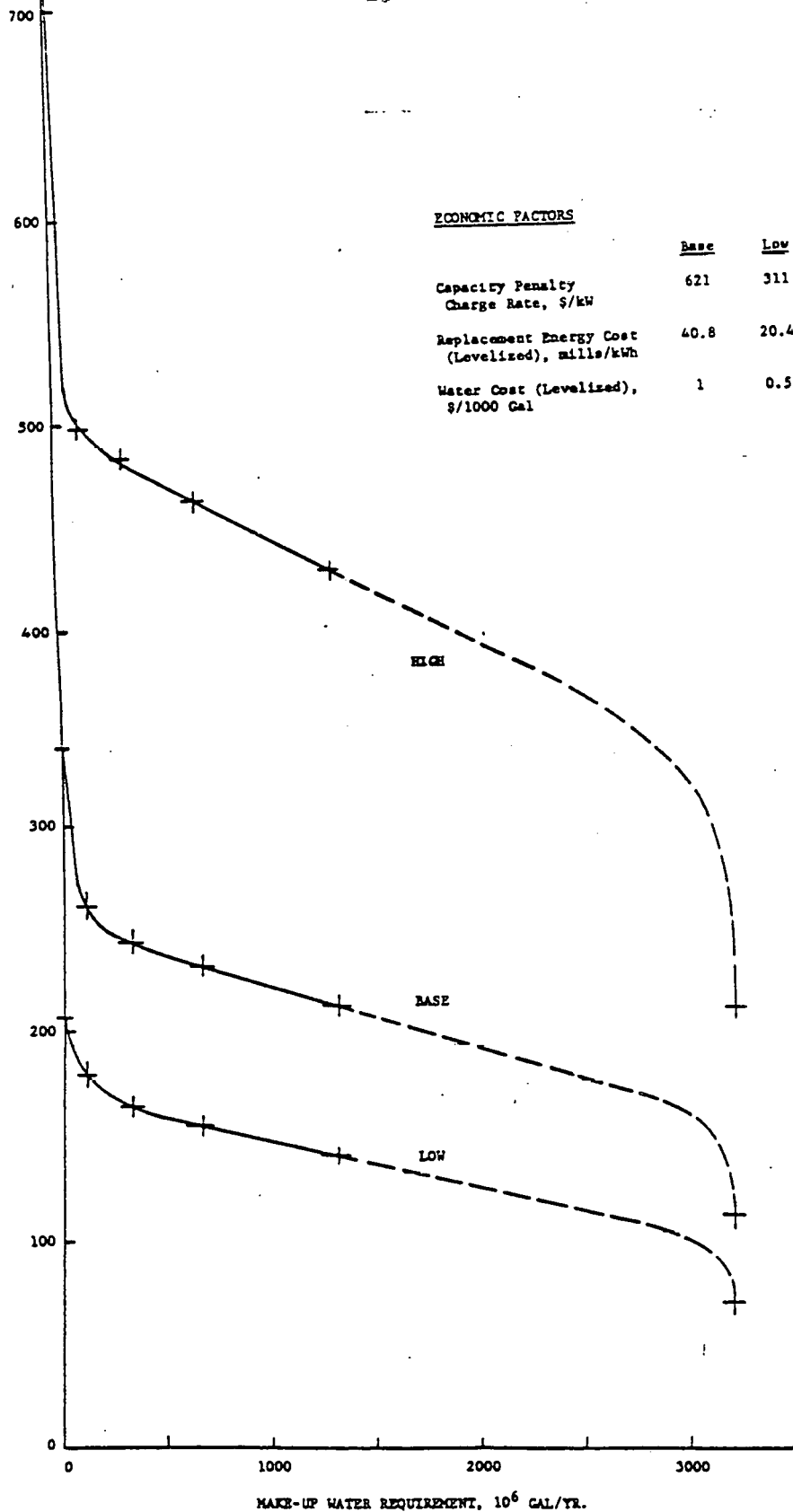


Figure 4.2.2-11 Effects of Composite Changes of Economic Factors on the Total Evaluated Costs of Alternate Cooling Systems for an 838 MW(e) PWR Plant

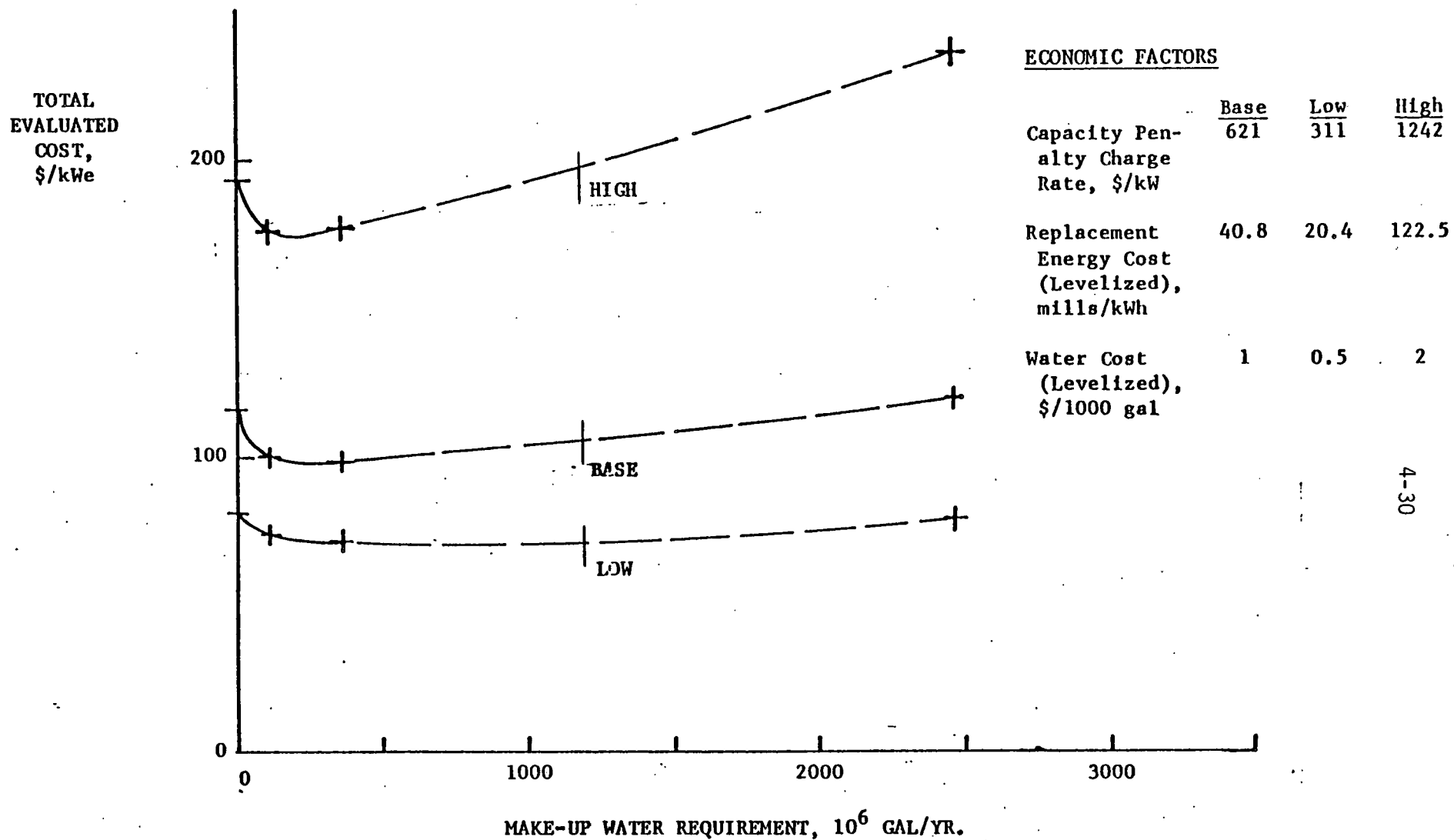


Figure 4.2.2-12 Effects of Composite Changes of Economic Factors on the Total Evaluated Costs of Alternate Cooling Systems for an 808 HTGR-GT Plant

TABLE 4.2.2-4

SENSITIVITY ANALYSIS SUMMARY
TOTAL EVALUATED COOLING SYSTEM COST DIFFERENTIALS

Water Constraint: 115 x 10⁶gal/yr

<u>Sensitivity Case</u>	<u>Fossil</u>	<u>LWR</u>	<u>HTGR-GT</u>
Base	1.81 (\$65.2M)	2.68 (\$135.2M)	1.00 (Base)
Low	1.30 (\$24.3M)	1.85 (\$68.3M)	0.73 (-\$21.8M)
High	3.37 (\$192M)	5.14 (\$335M)	1.76 (\$61.4M)

It should be noted that work is progressing on the development of a cooling system which uses ammonia as the working fluid to reject heat from the steam condenser to the atmosphere through dry-cooling towers. This effort, which is documented in Ref. 6, indicated that substantial economic savings are possible when the system is used to dry-cool or wet/dry-cool Rankine Cycle power plants. For example, high back pressure turbines will not be required for dry-cooling applications with the ammonia system. Fig. 4.2.2-13 from Ref. 6 is based on a plant located at Philadelphia, Pennsylvania, and shows substantial savings available over the conventional dry-cooled systems examined in this section. Further work on this system must be done to verify these projections, but successful commercialization of this ammonia phase-change system will significantly affect the relative results of the analyses of this section.

4.2.3 Other Siting Considerations

The HTGR offers advantages over its competition for siting in areas where water is not readily available for consumption. This section will examine the other factors which will affect the siting of the HTGR, namely its radiological and seismic characteristics, and the institutional barriers of nuclear siting as they apply to the HTGR.

4.2.3.1 Radiological

This section will present a comparative analysis which attempts to identify the envelope of site characteristics that would meet the radiological impact objectives necessary to the siting of an HTGR. For this analysis, Appendix I of 10CFR50, which specifies the radiological impact objectives for the LWR, was assumed as the applicable regulatory guideline for the HTGR. This analysis applies specifically to the large [3000 MW(t)] HTGR Steam Cycle design, but the results are comparable to those that would be obtained for the Gas Turbine and Reformer systems.

Appendix I of 10CFR50 specifies that each reactor shall be designed such that the maximum calculated radiation dose to an individual in the unrestricted area around the plant shall not exceed the following values per reactor per year:

Liquid Effluent

- Whole body dose 3 mrem
- Dose to any organ 10 mrem

Gaseous Effluent

- Gamma air dose 10 mrad
- Beta air dose 20 mrad
- Skin dose 15 mrem
- Whole body dose 5 mrem

Airborne Radioactive Iodine and Particulates

- Dose to any organ 15 mrem

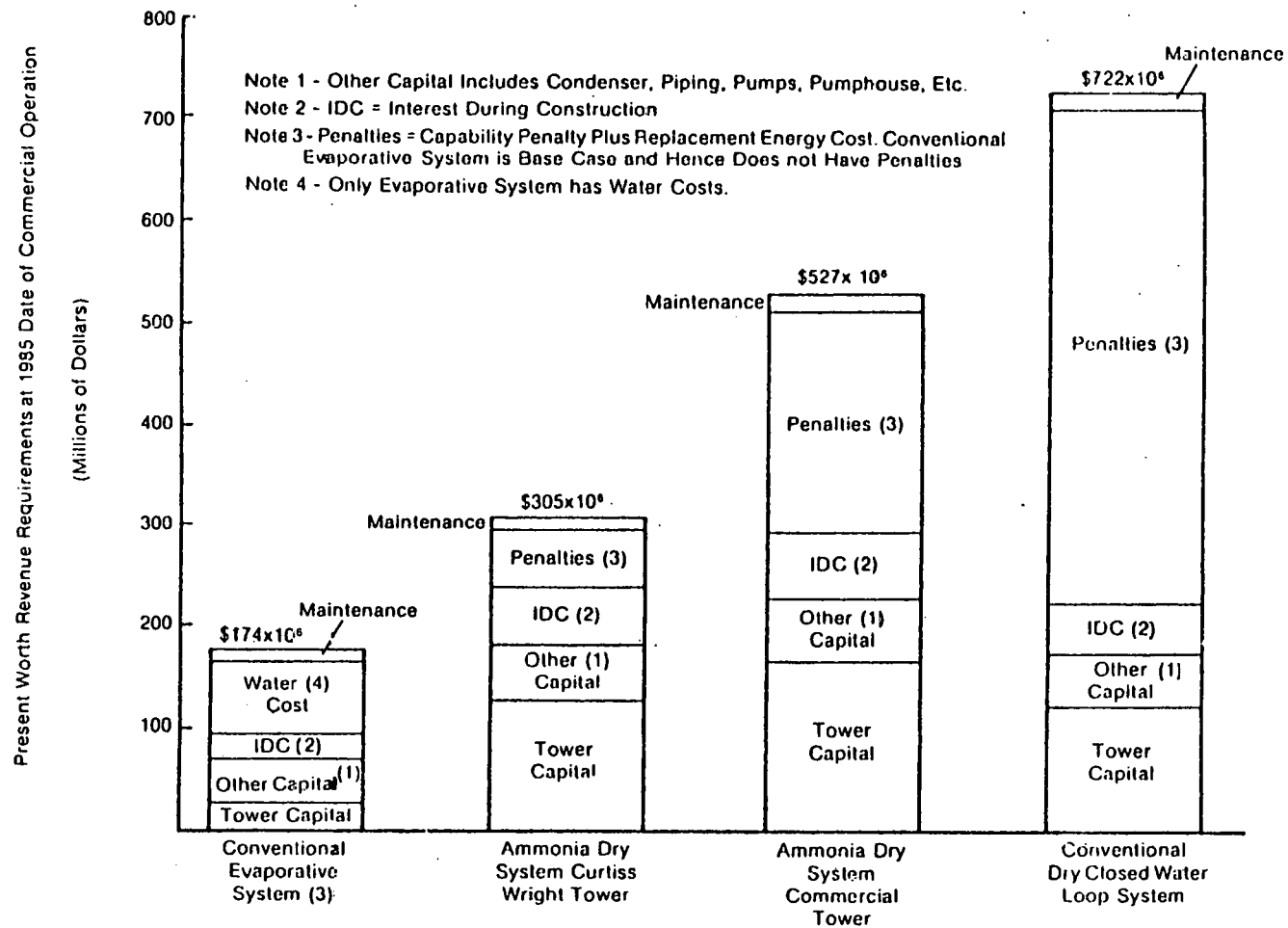


Figure 4.2.2-13 Comparison of Cooling System Present Worth Revenue Requirements for an 1140-MW(e) Plant

Other regulations such as 10CFR40 and 10CFR20 also affect the radiological design of nuclear power plants, but generally compliance with 10CFR50 almost assures compliance with all others.

The two groups of factors which determine if a particular plant will meet the above guidelines at a particular site are (1) Plant Performance Characteristics and (2) Site-Dependent Characteristics. The designer must take the site-dependent characteristics into account in order to design the radwaste systems such that the plant performance characteristics will result in releases that comply with the above criteria. One of the most important site-dependent characteristics that affect gaseous releases of radioactivity is the atmosphere's ability to disperse and dilute the releases. A quantitative measure of dilution is the annual average atmospheric dispersion factor X/Q where:

- X is the concentration of the diluted radioactive gaseous releases at a given point of interest in units of mass/volume.
- Q is the strength of radiological releases at the source in units of mass/time.

The factor X/Q is estimated by atmospheric diffusion models and will vary from approximately 2×10^{-6} to 2×10^{-5} sec/m³ at 500 m for ground releases, and 1×10^{-9} to 2×10^{-7} sec/m³ for stack releases at 500 m.

The other site-dependent characteristic influencing radiological dose is the pathway through which the radionuclides are transported to man. This analysis will only consider the air pathways as the current reference design for all HTGR systems is based on zero liquid releases. Generally, the air pathways considered include:

- Air submersion.
- External exposure to deposited materials.
- Inhalation and transportation.
- Ingestion of food crops.
- Ingestion of animal products.

Doses are calculated for the whole body and its significant organs. A limiting pathway, i.e., the worst case, is then selected and analyzed.

Ref. 2 compared a large HTGR Steam Cycle plant to both BWRs and PWRs. Table 4.2.3-1, which is taken from Ref. 2, shows the strength of various radionuclides released from various plants according to their Safety Analysis Reports. The Fulton Station HTGR, which was used in the study, did release some liquid effluents. These will be eliminated in future HTGRs. Because the plants listed are of differing sizes,

TABLE 4.2.3-1

ESTIMATE OF ANNUAL RELEASE OF RADIOACTIVITY IN EFFLUENTS
(AS EXTRACTED FROM FINAL ENVIRONMENTAL STATEMENT)

Plant Name	Vendor & Reactor Type	Location	Effluents to Atmosphere (Ci/yr)							Liquid Effluents (Ci/yr)							Net M(e) Unit
			I-131	Kr-85	Kr-88	Xe-133	Xe-135	Xe-138	Co-58	Co-60	Sr-89	Sr-90	I-131	Cs-134	Cs-137	H-3	
Fulton	GA-HTGR	Fulton, Pa.	7.9 x 10 ⁻⁷	(a)	14	9	7	2	*	*	.0001	.0009	*	.018	.036	21	1160
Grand Gulf 1&2	GE-BWR	Port Gibson, Miss.	.037	880	55	3400	420	240	.11	.013	.11	.006	.16	.91	.58	20	1250
Skagit 1&2	GE-BWR	Sedro Nooley, Wash.	.028	840	49	3254	418	230	No Planned Releases of Radioactive Liquids								1269
Clinton 1&2	GE-BWR	Clinton, Ill.	.27	660	47	3200	420	230	.0091	.0011	.0053	.00027	.0098	.0081	.0041	20	950
River Bend 1&2	GE-BWR	St. Francisville, La.	.08	664	11	2357	367	57	.013	.0015	.0098	.0005	.025	.0097	.0065	10	910
Nine Mile Point 2	GE-BWR	Scriba, N.Y.	.58	760	450	18,400	250	280	.015	.002	.0082	.00051	.042	.0064	.0042	20	1100
Susquehanna 1&2	GE-BWR	Berwick, Pa.	.018	740	72	1400	260	77	.040	*	.028	.0014	.057	.023	.017	20	1052
Byron 1&2	N-PWR	Byron, Ill.	.05	988	24	2357	40	6	.00028	.00004	.00001	*	.062	.0059	.0046	350	1120
Braidwood 1&2	N-PWR	Braidwood, Ill.	.05	988	24	2357	40	6	.00028	.00004	.00001	*	.062	.0059	.0046	350	1120
Shearon Harris 1&2	N-PWR	Newhill, N.C.	.043	795	28	2500	46	6	.00021	.00003	*	*	.049	.0058	.0045	350	900
South Texas 1&2	N-PWR	Palacios, Tex.	.087	1050	32	340	26	8	.0045	.0088	*	*	.075	.017	.025	350	1250
Comanche Peak 1&2	N-PWR	Glen Rose, Tex.	.044	970	22	280	26	6	.0021	.00028	.00008	*	.18	.015	.0097	350	1150
Catawba 1&2	N-PWR	Lake Wylie, S.C.	.094	970	20	290	24	6	.0018	.00023	.00008	*	.20	.017	.12	350	1180
Pebble Springs 1&2	B&W-PWR	Arlington, Ore.	.049	1030	34	3440	53	8	.001	.00013	*	*	.045	.0043	.0012	160	1260
North Anna 3&4	B&W-PWR	Mineral, Va.	.018	738	16	931	27	4	.00037	.000012	.000014	*	.75	.0048	.0034	1000	938
Surry 3&4	B&W-PWR	Gravel Neck, Va.	.05	738	16	912	27	4	.00028	.00004	*	*	.023	.013	.0085	685	900
NTPSS 4	B&W-PWR	Richland, Wa.	.032	1000	23	840	35	5	.00016	.00002	*	*	.005	.024	.015	350	1250
Greenwood 2&3	B&W-PWR	St. Clair Co., Mich.	.019	1000	24	2800	41	6	.00012	.00002	*	*	.011	.0027	.0018	350	1208
Pilgrim 2	CE-PWR	Plymouth, Mass.	.16	1000	32	2970	54	8	.052	.015	.0021	.00006	1.3	.10	.15	350	1180
Waterford 3	CE-PWR	Taft, La.	.11	988	26	3500	43	6	.0025	.000082	.00013	.000005	.29	1.6	1.3	1000	1165
San Onofre 2&3	CE-PWR	San Clemente, Ca.	.29	932	24	2473	38	6	.00096	.0003	.00037	.000013	2.2	.42	.36	1000	1140
NTPSS 3&5	CE-PWR	Satsop, Wa.	.017	1100	28	290	31	8	.0046	.00037	.00017	*	.016	.019	.012	350	1240

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(a) HTGR plant design has the option of either recycling all of the Kr-85 through the helium purification system or releasing Kr-85 intermittently to the atmosphere under favorable meteorological conditions. If recycled, there will be no release of Kr-85 to the atmosphere. If released, the rate of release is estimated to be 4185 Ci/yr.

* Too low to be included.

Table 4.2.3-2 normalizes the releases to 1160 MW(e), which was the size of the Fulton unit, for direct comparison purposes. It can be seen from this table that the HTGR generally enjoys a two-orders-of-magnitude reduction in the amount of radionuclides released in gaseous form.

Ref. 2 then calculated the dose rates for each limiting pathway and determined the estimated annual dose to an individual and his significant organs for these pathways as a function of the X/Q at the location where a person may live. Fig. 4.2.3-1 shows the whole body dose (mrem/yr) from gaseous effluents to the atmosphere. Fig. 4.2.3-2 shows the adult skin dose and Fig. 4.2.3-3 shows the thyroid dose to an infant drinking milk. It can be seen from these figures that the HTGR can be sited in areas with relatively unfavorable atmospheric dispersion factors; therefore, it does not require tall stacks for gaseous releases.

General conclusions that can be drawn on the above information and from Ref. 2 analyses are:

- The HTGR and the LWR can be located over a relatively wide range of site conditions and still meet the radiological impact objectives specified in Appendix I of 10CFR50.
- For gaseous releases, the LWR usually requires about two orders of magnitude more atmospheric dilution than the HTGR. Because of the additional dilution required, the LWR will require additional radwaste equipment in the plant design.
- For the HTGR, the limiting radiological impact for gaseous release is the whole body dose from immersion in a gaseous cloud. For the LWR, it is the thyroid dose from the iodine-milk-infant pathway.
- From the standpoint of radiological impact, the HTGR in general has greater flexibility in siting than the LWR.

4.2.3.2 Geological

According to the requirements set forth in the GCRA Functional Specification for the HTGR-SC, the design maximum horizontal ground accelerations are .15g OBE, .3g SSE for a hard rock site, and .2g OBE, .4g SSE for a firm soil site. For the HTGR-GT, the selected values are .15g OBE and .3g SSE for all types of soil. As a reference, a major LWR supplier uses .25g as its reference SSE acceleration. To quantify the areas of the U.S. market which would be satisfied by these geological design criteria, GCRA commissioned studies by Dames & Moore and URS/John Blume & Associates.

The purpose of the Dames & Moore report (Ref. 7) was to make a preliminary assessment of the areas within the contiguous U.S. having sufficiently suitable subsurface conditions to permit economical design and

TABLE 4.2.3-2

ESTIMATE OF ANNUAL RELEASE OF RADIOACTIVITY IN EFFLUENTS
(NORMALIZED TO 1160 MWe)

Plant Name	Vendor & Reactor Type	Effluents to Atmosphere (Ci/yr)						Liquid Effluents (Ci/yr)						Normalizing		
		I-131	Kr-85	Kr-88	Xe-133	Xe-135	Xe-138	Co-58	Co-60	Sr-89	Sr-90	I-131	Cs-134	Cs-137	H-3	Factor
Fulton	GA-HTR	7.9 x 10 ⁻⁷	(a)	14	9	7	2	*	*	.0001	.0009	*	.018	.036	21	1.00
Grand Gulf 142	GE-BNR	.034	820	51	3200	390	220	.10	.012	.10	.006	.15	.84	.54	19	0.928
Skagit 142	GE-BNR	.031	920	53	3500	460	250	No planned releases of liquids containing radioactivity**								1.09
Clinton 142	GE-BNR	.33**	810	57	3900	510	280	.0111	.0013	.0065	.00033	.012	.0099	.0050	24	1.22
River Bend 142	GE-BNR	.10	820	14	2900	450	70	.015	.0018	.012	.00062	.031	.012	.0077	25	1.23
Nine Mile Point 2	GE-BNR	.61**	800	470	19300**	310	290	.016	.0021	.0086	.00054	.044	.0067	.0044	21	1.05
Susquehanna 142	GE-BNR	.020	840	500	1500	310	85	.017	*	.025	.0015	.063	.025	.018	22	1.10
"Representative"	GE-BNR	.046	840	190	3000	410	200	.032	.0034	.030	.0018	.060	.18	.12	22	
Byron 142	N-PWR	.05	1000	25	2500	42	6	.00029	.00004	.00001	*	.065	.0062	.0048	360	1.04
Braidwood 142	N-PWR	.05	1000	25	2500	42	6	.00029	.00004	.00001	*	.065	.0062	.0048	360	1.04
Shearon Harris 142	N-PWR	.055	1000	110	3200	59	8	.00027	.00004	*	*	.063	.0075	.0055	450	1.29
South Texas 142	N-PWR	.081	970	30	320	32	7	.0042	.0082	*	*	.070	.016	.023	330	0.928
Comanche Peak 142	N-PWR	.044	970	22	280	26	6	.0021	.00028	.00008	*	.18	.015	.0097	350	1.00
Catawba 142	N-PWR	.092	950	20	290	24	6	.0018	.00023	.00008	*	.20	.17	.12	340	0.983
"Representative"	N-PWR	.062	980	39	1500	38	7	.0015	.00015	.00003	*	.11	.037	.028	370	
Pebble Springs 142	B&W-PWR	.045	950	31	3200	53	7	.0009	.00012	*	*	.041	.0040	.0011	150	0.921
North Anna 344	B&W-PWR	.022	920	20	1200	33	5	.00046	.000015	.000015	*	.93	.0060	.0042	1200	1.24
Surry 344	B&W-PWR	.06	950	21	1200	35	5	.00036	.0005	*	*	.029	.017	.011	450	1.29
NPPSS 4	B&W-PWR	.03	930	21	780	32	5	.00015	.000019	*	*	.0046	.022	.014	330	0.928
Greenwood 243	B&W-PWR	.018	960	23	2700	39	6	.00012	.00002	*	*	.011	.0026	.0017	340	0.96
"Representative"	B&W-PWR	.035	940	23	1800	38	5	.00040	.00016	*	*	.20	.010	.006	490	
Pilgrim 2	CE-PWR	.16	980	31	2900	53	8	.050	.015	.0021	.000059	1.3	.098	.15	340	0.983
Waterford 3	CE-PWR	.11	980	26	3500	43	6	.0025	.000082	.00013	.000005	.29	1.6	1.3	1000	0.996
San Onofre 243	CE-PWR	.30	550	24	2500	39	6	.00098	.00031	.00038	.000013	2.2	.43	.37	1020	1.02
NPPSS 345	CE-PWR	.016	1030	26	270	29	8	.0043	.00081	.00016	*	.015	.018	.011	330	0.935
"Representative"	CE-PWR	.15	990	27	2300	41	7	.014	.00041	.00069	.000019	.95	.54	.46	670	

(a) Plant design calls for recycling all of the Kr-85 to the helium coolant. If it is released, the total estimated Kr-85 would be 4185 Ci/yr.

*Too low to be included

**Not included in average

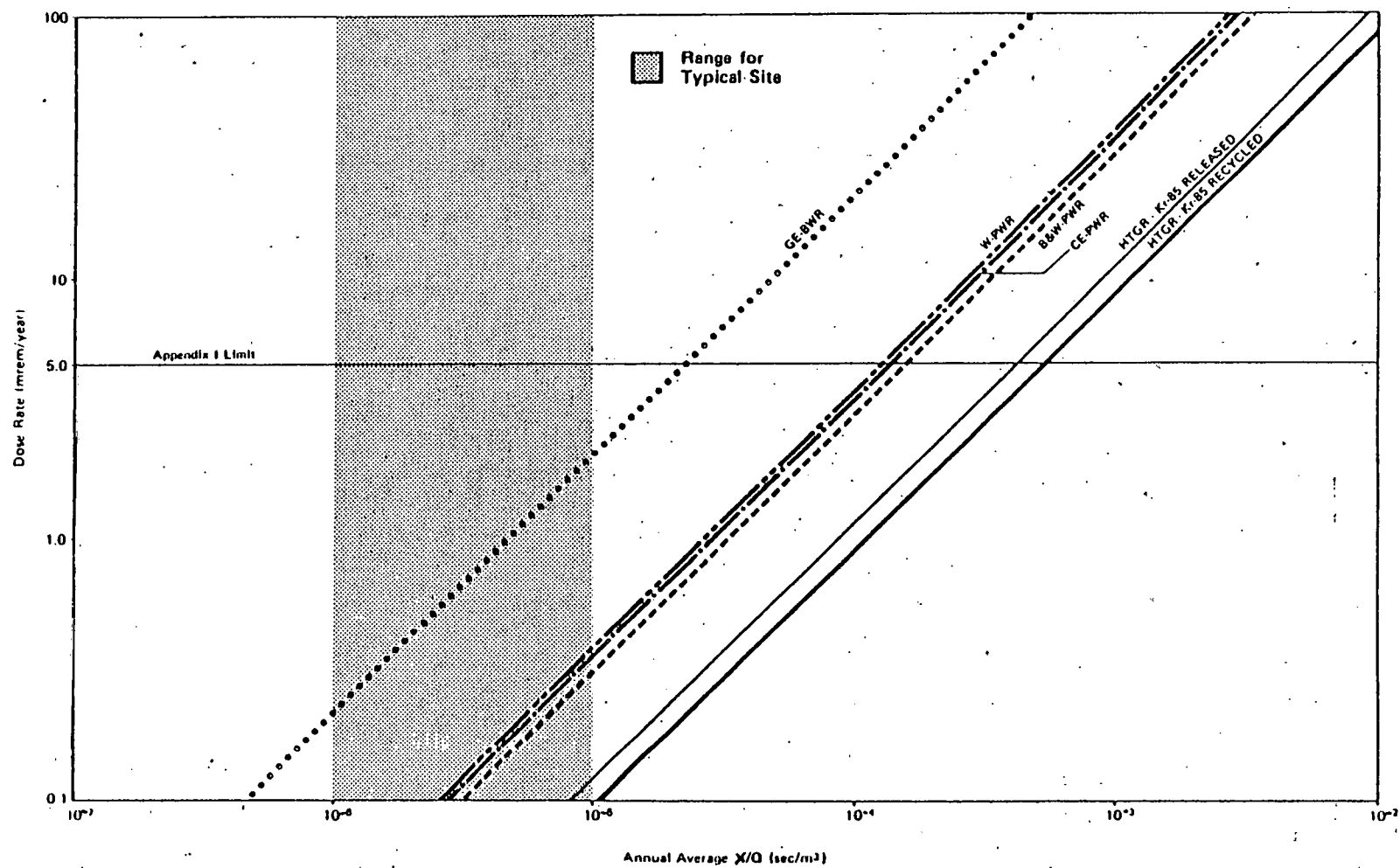


Figure 4.2.3-1 Whole Body Dose from Gaseous Effluents to the Atmosphere

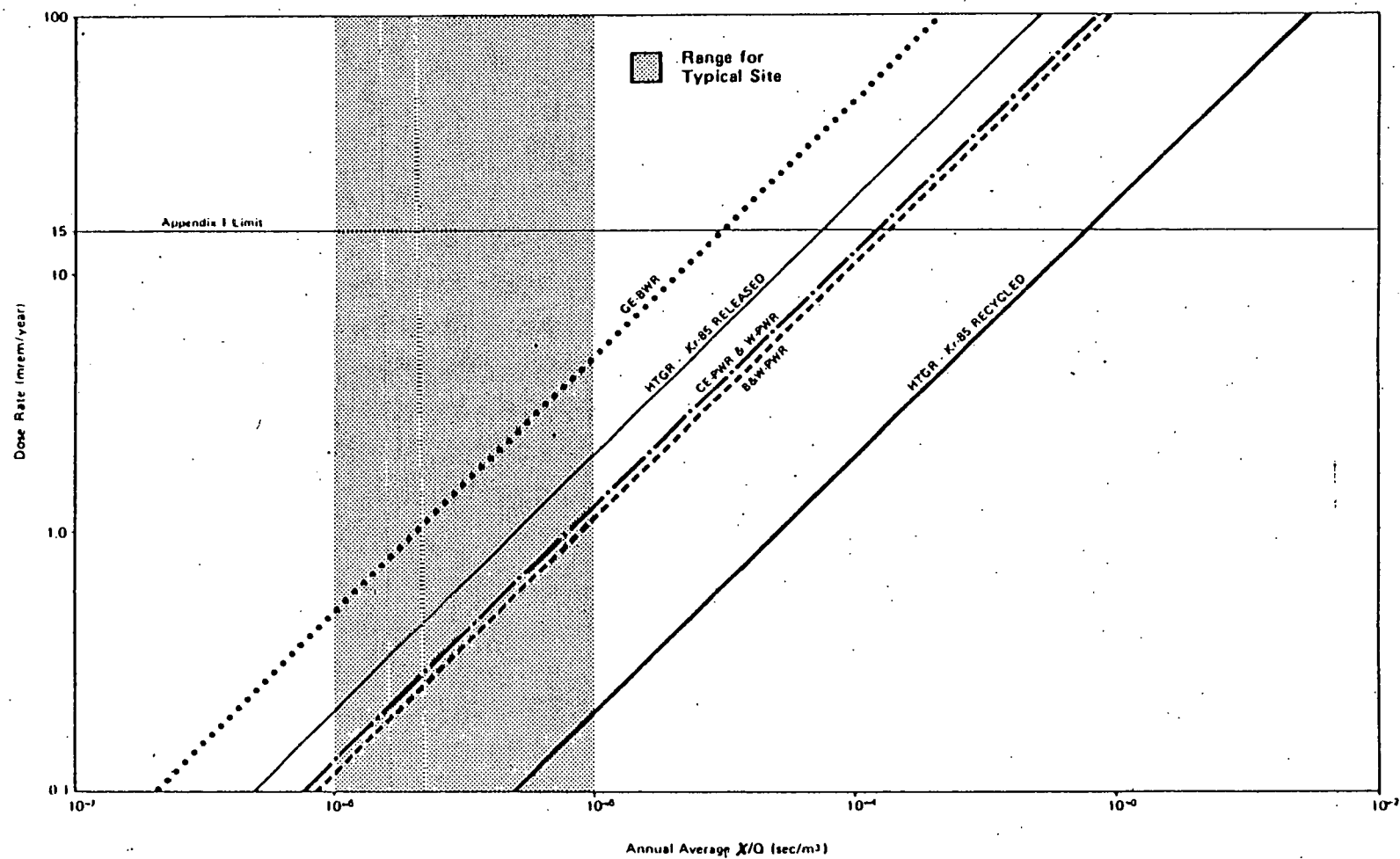


Figure 4.2.3-2 Skin Dose from Gaseous Effluents to the Atmosphere

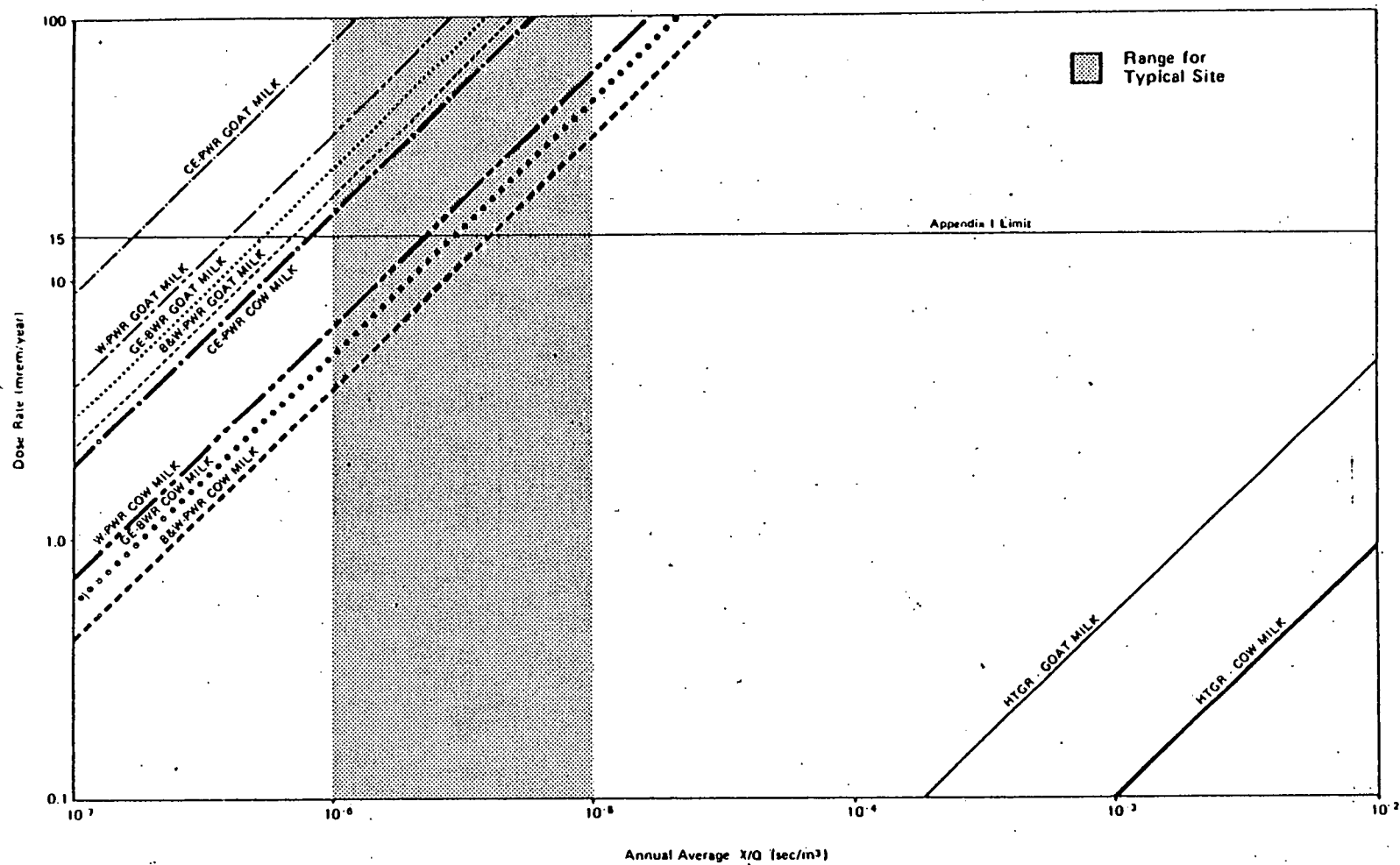


Figure 4.2.3-3 Thyroid Dose Rate to an Infant from Ingesting Milk Containing I-131

construction of HTGR foundations. Maps were prepared showing zones of geologically similar conditions. Within these zones, approximate percentages of potential sites were determined having four different soil bearing pressure categories at an assumed foundation depth of forty feet. The GCRA reference static allowable bearing capacity is 10 kips/ft² at an assumed depth of forty feet. This value is important in reference plant design because as the allowable bearing capacity decreases, the size of the foundations must increase to distribute the structure weight over a larger area and, therefore, the design is necessarily more expensive.

The four foundation condition categories were defined as:

Category A: Where bearing pressure of 20 to 30 kips per square foot (ksf) or more may be appropriate.

Category B: Where bearing pressures on the order of 10 ksf may be appropriate.

Category C: Where site improvement methods will render sites suitable for designs using bearing pressures in the range of 10 to 20 ksf.

Category D: Where prospects for upgrading to conditions commensurate with design for 10 ksf are technically or economically unfeasible, i.e., costs of site improvements will be greater than \$50 million.

The results of the study are summarized in Table 4.2.3-3. The physiographic provinces are shown in Fig. 4.2.3-4. These results show that for the reference design static bearing capacity of 10 ksf, the HTGR will be siteable in most areas and will not be unduly restricted.

The URS/Blume study (Ref. 8) attempted to define safe-shutdown-earthquake (SSE) design ground accelerations by regions of the contiguous U.S. The purpose of this study was to determine the impact of the reference design seismic levels specified by GCRA on the potential HTGR market size.

There are two basic steps in specifying the SSE for a site. The first is to determine the maximum earthquake potential of capable faults and seismic sources in the area. The second is to ascertain the dependence of the amplitude of ground motion on earthquake size and distance. The potential earthquake that produces the strongest motion at the site in the critical frequency band can then be identified as the SSE. By utilizing data from existing nuclear plants in the various regions and by analyzing these data with state-of-the-art methodology, the report produced a map, which is shown in Fig. 4.2.3-5, of contours of the maximum expected horizontal ground accelerations for nuclear sites. Based on this effort, it appears that the seismic design values for the reference HTGR plants will not unduly restrict the siting capability of any of the HTGR systems.

TABLE 4.2.3-3

SUMMARY OF RESULTS

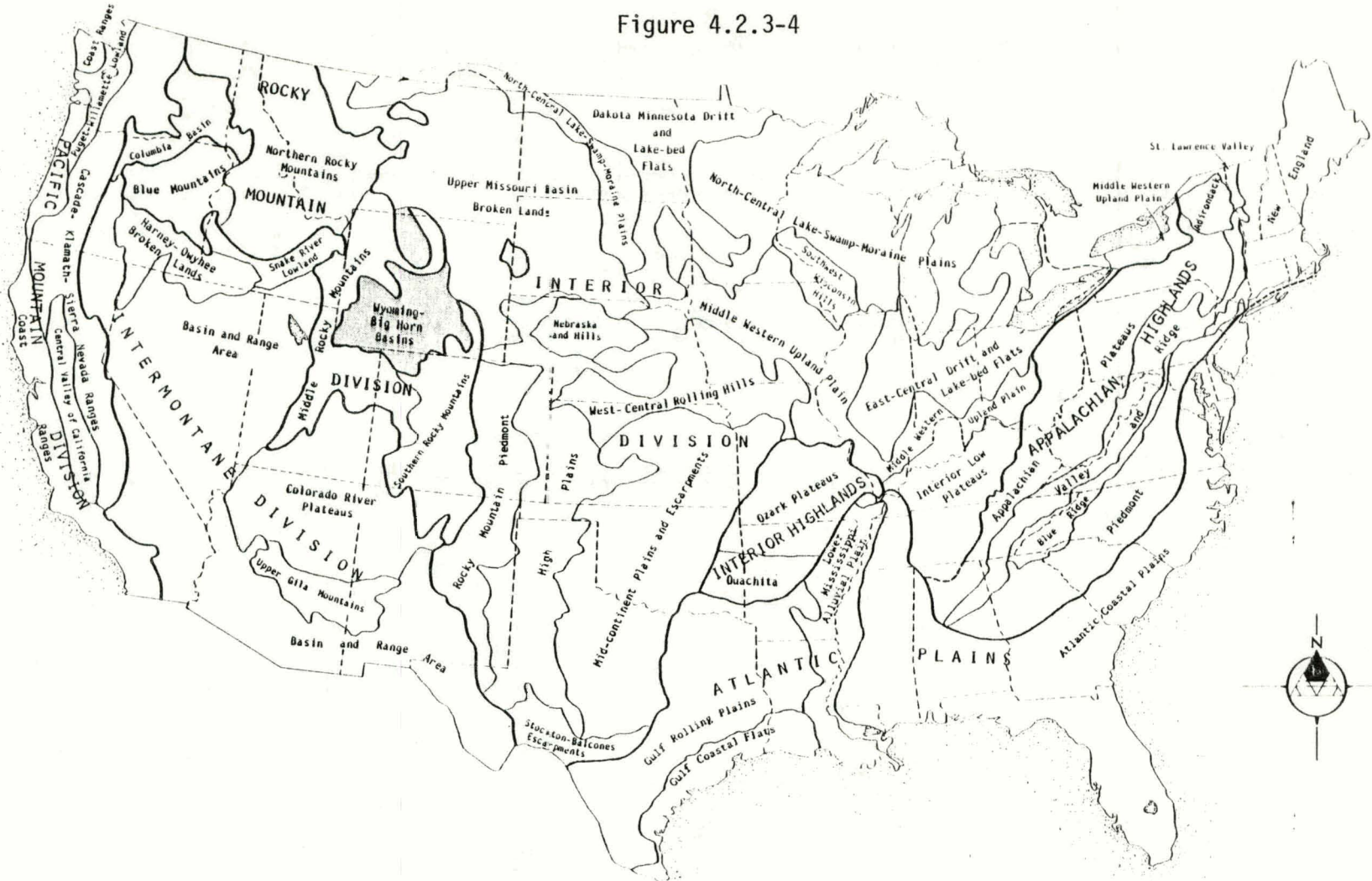
<u>Physiographic Province</u>	<u>Estimated Percentage of Area in Different Categories</u>			
	<u>A</u>	<u>B</u>	<u>C</u>	<u>D</u>
<u>Pacific Mountain Division</u>				
Puget-Willamette Lowland				
Washington	50	20	20	10
Oregon	<5	10	65	20
Cascade, Klamath, Sierra				
Nevada Ranges	90	5	5	<1
Central Valley of California	10	30	55	<5
Coast Ranges	75	10	10	5
<u>Rocky Mountain Division</u>				
Northern Rocky Mountains	90	5	5	<1
Blue Mountains	80	10	10	<1
Middle Rocky Mountains	90	5	5	<1
Southern Rocky Mountains	80	10	10	<1
Wyoming Basin	80	10	10	<1
<u>Intermontane Division</u>				
Columbia Basin	80	5	10	5
Harney-Owyhee Broken Lands	80	10	10	<1
Snake River Lowland	50	25	25	<1
Basin and Range				
Arizona	5	85	5	5
California	80	5	10	<5
Nevada	60	10	15	15
Utah	20	10	20	50
Colorado River Plateau	80	10	10	<1
Upper Gila Mountains	80	10	10	<1
<u>Interior Division</u>				
Central Lowlands				
Dakota-Minnesota Drift				
and Lake Bed Flats	10	20	60	10
North-central Lake-Swamp				
Moraine Plains (east)	<1	25	70	5
(west)	10	25	45	20
Southwest Wisconsin Hills	70	20	10	<1
Middle Western Upland Plain	30	40	30	<1
Mid-Continent Plains and				
Escarpments	40	35	20	<5
East-central Drift and				
Lake Bed Flats	10	50	40	<1
Interior Low Plateaus	40	30	20	10

TABLE 4.2.3-3
(Continued)

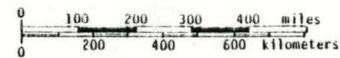
SUMMARY OF RESULTS

<u>Physiographic Province</u>	<u>Estimated Percentage of Area in Different Categories</u>			
	<u>A</u>	<u>B</u>	<u>C</u>	<u>D</u>
<u>Interior Division (continued)</u>				
<u>Great Plains</u>				
Upper Missouri Basin				
Broken Lands	70	15	15	<1
Nebraska Sand Hills	0	20	50	30
West-central Rolling Hills	20	30	45	<5
High Plains	20	55	20	<5
Rocky Mountain Piedmont	80	10	10	<1
Stockton-Balcones Escarpments	25	45	25	<5
<u>Appalachian Highlands</u>				
Piedmont	30	40	25	5
Blue Ridge	40	30	25	5
Valley and Ridge	55	20	20	<5
Appalachian Plateaus	40	40	20	<1
Adirondack	80	10	10	<1
New England	70	15	15	<1
St. Lawrence Valley	25	30	35	10
<u>Interior Highlands</u>				
Ozark Plateaus	60	20	20	<1
Ouchita Plateaus	40	30	20	20
<u>Atlantic Plains</u>				
Atlantic Coastal Plain	5	20	40	35
Gulf Rolling Plains	10	60	30	<1
Gulf Coastal Flats	0	40	55	5
Lower Mississippi Alluvial Plain	0	5	50	45

Figure 4.2.3-4



REFERENCES: U.S.G.S. THE NATIONAL ATLAS OF THE UNITED STATES OF AMERICA, 1970, SHEETS 60 & 61.

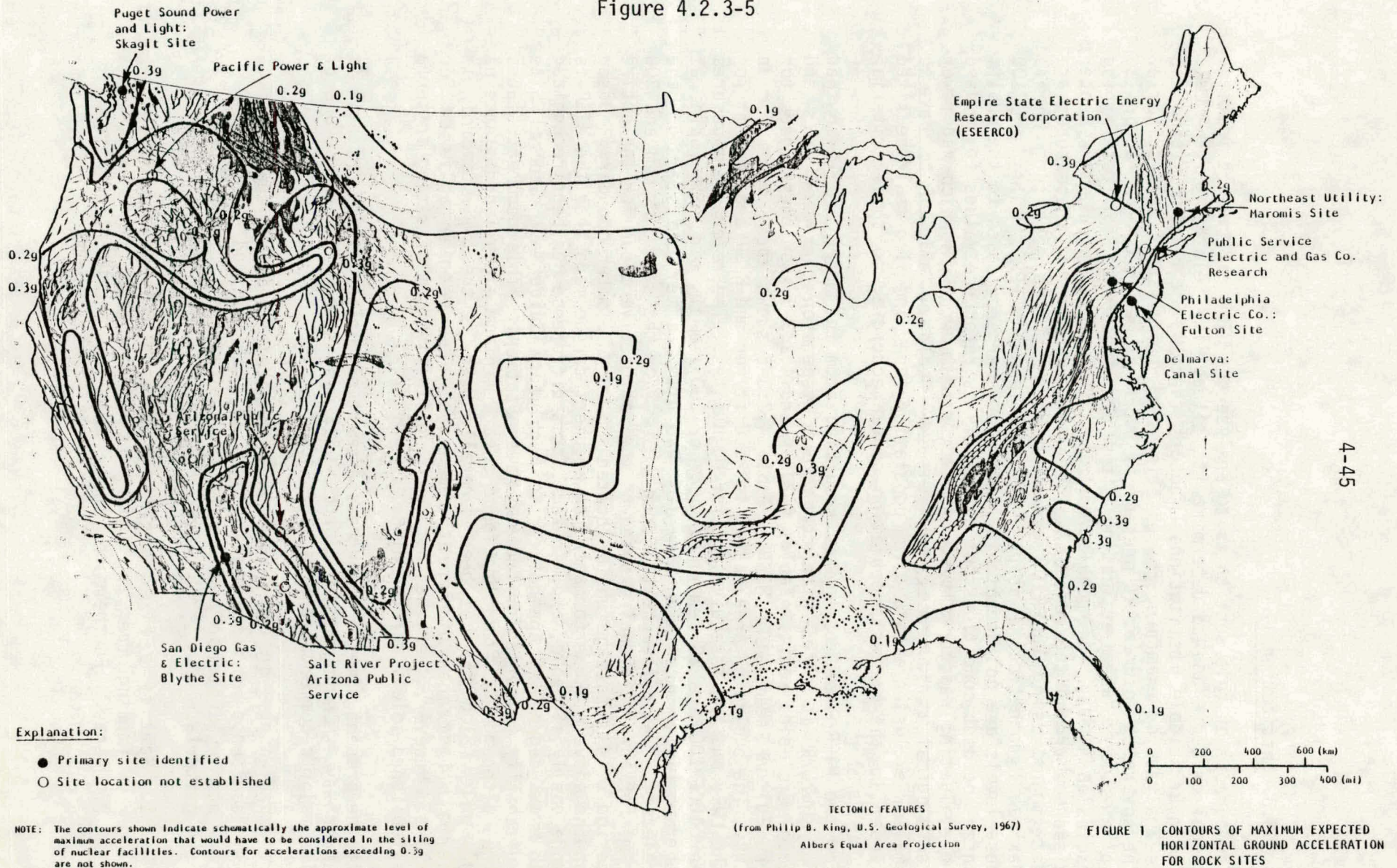


PHYSIOGRAPHIC PROVINCES OF THE UNITED STATES

DAMES & MOORE

PLATE 1

Figure 4.2.3-5



4.2.3.3 Institutional

The purpose of this section is to analyze the HTGR relative to institutional criteria and to determine its siteability relative to the LWR. The quantifiable institutional factors that affect reactor siting can be analyzed by examining the existing siting regulations and criteria. The governing document at present is 10CFR100, which covers factors to be considered when evaluating sites and procedures for determining exclusion areas, low population zones (LPZ), and population center distances around the reactor. The areas and sizes of these zones are determined using 10CFR100 guideline dose limits resulting from a Maximum Hypothetical Fission Product Release (MHFPR). The Part 100 dose levels are not acceptable limits for emergency doses to the public under accident conditions but serve only as reference values to be used in evaluating reactor sites. A reactor with an MHFPR which had fewer radiological consequences than the LWR would be allowed to have a smaller exclusion area and low population zone, and also could be sited closer to the nearest population center according to these criteria.

For the HTGR, the dose rates resulting from the MHFPR are lower than for an LWR at sites with comparable dispersion characteristics and equal exclusion radii. Tables 4.2.3-4 and 4.2.3-5 show these doses for the Fulton Steam Cycle HTGR. Table 4.2.3-6 shows these doses for an LWR. The exclusion area radius is 2500 ft for the HTGR and 3400 ft for the LWR, and X/Q is equivalent for both sites. The reference exposures per 10CFR100 are 25 rem whole body or 300 rem thyroid from iodine at the exclusion area boundary for two hours after the start of the release. The reference LPZ dose limits are the same (25 rem and 300 rem) for any individual exposed to the radioactive cloud for the entire period of its passage while located at the LPZ outer boundary. Based on the above HTGR performance, an argument can be made that the HTGR is more easily siteable, and, in fact, this difference has been recognized to some degree in Appendix A to Regulatory Guide 4.7, General Site Suitability Criteria for Nuclear Power Stations. This document states that although NRC staff has found that a minimum exclusion distance of 0.4 mile (2100 ft) has been generally used as the LWR standard, in certain instances different dimensions have been established for HTGRs. However, there has not been a definitive set of siting criteria established solely for the HTGR.

At the present time, a major reassessment of nuclear siting criteria is under way. In August 1979, NUREG-0625 was issued (Ref. 12). One of its major recommendations is as follows:

Revise Part 100 to change the way protection is provided for accidents by incorporating a fixed exclusion and protection action distance and population density and distribution criteria.

1. Specify a fixed minimum exclusion distance based on limiting the individual risk from design basis accidents. Furthermore, the regulations should clarify the

TABLE 4.2.3-4

HTGR

LOW POPULATION ZONE 30 DAY DOSES FOR THE MHFPR

Dose Type	30 Day Dose at the LPZ Boundary (Rem)
Thyroid	150.0
Whole Body	3.2
Bone	1.1
Lung	1.0

TABLE 4.2.3-5

HTGR

EXCLUSION ZONE BOUNDARY DOSES FOR THE MHFPR (2500')

Dose Type	0-0.5 hr Dose (Rem)	0.5-1.0 hr Dose (Rem)	1.0-1.5 hr Dose (Rem)	1.5-2.0 hr Dose (Rem)	Total 0-2 hr Dose (Rem)
Thyroid	0.022	0.075	0.347	1.37	1.81
Whole Body	0.002	0.003	0.015	0.042	0.062
Bone	0.009	0.009	0.011	0.017	0.046
Lung	0.002	0.003	0.009	0.031	0.045

Source: Ref. 16

TABLE 4.2.3-6

EXCLUSION ZONE BOUNDARY DOSES FOR THE LWR MHFPR (3400')

<u>Dose Type</u>	<u>Total 2-Hour Dose (Rem)</u>
Thyroid	12.5
Whole Body	.4

required control by the utility over activities taking place in land and water portions of the exclusion area.

2. Specify a fixed minimum emergency planning distance of 10 miles. The physical characteristics of the emergency planning zone should provide reasonable assurance that evacuation of persons, including transients, would be feasible if needed to mitigate the consequences of accidents.
3. Incorporate specific population density and distribution limits outside the exclusion area that are dependent on the average population of the region.
4. Remove the requirement to calculate radiation doses as a means of establishing minimum exclusion distances and low population zones.

At the time of this writing, a rulemaking process is under way to implement the intent of the above recommendation. The current NRC position in this area was announced in October 1979 when an NRC-EPA task force recommendation was approved requiring two emergency planning zones (EPZ) around each nuclear plant. Within each EPZ, local authorities must be able to take remedial action in case of a radiological threat. Within the inner 10 mile EPZ, the local governments should be able to deal with airborne radiation exposure. Within the outer 50 mile EPZ, plans should be in effect for the impounding of food that may be contaminated. If the recommendations of NUREG-0625 are adopted, the radiological advantages that the HTGR enjoys over the LWR may not be recognized as having merit in the site selection process.

At the present time, GA is providing information to the NRC for the rulemaking process in order to obtain recognition of the HTGR's inherent design differences.

4.2.4 Operation and Maintenance

Although the charges arising from operation and maintenance are a relatively small part of the total cost of power from a nuclear plant, the length of the outages caused by poor operability and maintainability can be the source of major financial losses due to large replacement power costs. For this reason, utilities constantly strive to minimize plant down time by the institution of design and procedural measures. This section of the document will examine the unique aspects of the HTGR that are pertinent to its perceived ability to provide reliable power with high availability, thus minimizing replacement power costs.

4.2.4.1 Fort St. Vrain

One of the ways of examining the operability of future HTGR Steam Cycle plants is by examining the operational record of the prototype Fort St.

Vrain reactor. Several studies have done this in the past, one of the more detailed being included in Section 6 of the RAMCO commercialization study (Ref. 13). This study chronicled the plant performance from September 1970 through September 1977. A review of the data indicates that the vast majority of the problems encountered have been non-HTGR-related, such as cable separation, feedwater chemistry and fire protection. While these types of problems are plant specific, two generic problem areas have been identified which are having a direct effect on plant availability, namely helium impurity and core power oscillations.

- Helium Impurity - The HTGR primary coolant is pure helium. In order to limit impurities such as oxygen and moisture in the coolant, helium purification systems are provided; however, these systems at Fort St. Vrain were not designed to remove the quantities of moisture and oxygen that have been introduced into the PCRV. Oxygen and moisture have a tendency to be absorbed by the graphite in the core and cannot be off-gassed until the core temperature is elevated, thus allowing some graphite corrosion to take place. Technical specifications limit the amounts of impurities allowed in the helium; therefore, high impurity levels restrain plant power ascension. This has been one of the major reasons for Fort St. Vrain's slow return to power after outages.

Large water ingresses have been experienced, primarily due to leaks from the helium circulator auxiliaries. The effect of water in the PCRV on the life of the graphite is currently under study to determine graphite oxidation rates and their long-term effects on plant life. Future HTGRs will have motor-driven circulators instead of the steam- and pelton-wheel-driven circulators used at Fort St. Vrain. These new circulators will have water-lubricated bearings but with a redesigned buffer helium system so that they should experience a greatly reduced frequency of leakage. Through proper design attention to possible sources of water ingress and to graphite component design margins, the effects of water ingress on plant availability should be minimized.

The effect of moisture on fuel performance is a related problem. The fuel particles at Fort St. Vrain have four layers of coatings surrounding the fuel kernel. These coatings act like a pressure vessel to retain fission products within the particle. At local high temperatures during normal plant operation, some particles may experience failure of the outer coating. When the failed particles come in contact with moisture at elevated temperatures, hydrolysis of the fuel occurs, releasing all of the particle's fission products into the primary coolant. The amount of moisture needed for hydrolysis is extremely small as the moisture from humid air that enters the primary system during fueling operations is sufficient to cause hydrolysis. New multiple layer coatings have been developed which should reduce this problem for future HTGRs; however, cost/benefit analyses show

that it is not cost effective to try to design fuel particles that will experience a near-zero failure rate. Fission products that are released over fuel life can generally be tolerated at the levels predicted with the new coatings as discussed in the Maintenance section. It is interesting to note that even though Fort St. Vrain has experienced large water ingresses in its operation to date, primary coolant activity is 30% of that predicted and a factor of 60 below the technical specification limit. This indicates the degree of conservatism that has been incorporated into the fuel design.

- Core Temperature Fluctuations - In late 1977, while approaching 60% power, temperature fluctuations were observed in the Fort St. Vrain primary coolant circuit at individual core region outlets and at the steam generator module inlets. An intensive investigation into the nature and causes of the fluctuations was initiated. These fluctuations have been observed at power levels between 30% and 70%, with the temperature swings generally staying within technical specification limits. The period of the temperature fluctuations is irregular, ranging from 5 to 20 min; however, the power changes observed on the nuclear channels are very rapid and do not show the normal temperature feedback effects. The temperature changes are much faster than could be achieved with region power changes, and the total core coolant flow and average power remained constant during the fluctuations.

The most probable explanation of the temperature and neutron flux fluctuations is small movements of internal reactor components with resulting changes in coolant flow through the gaps between the components. This motion is most likely induced by pressure differences in the gaps and thermal gradients in the core components. Through data collection and experimentation, a core pressure drop vs. core flow rate threshold has been defined. Based on this threshold, operation of Fort St. Vrain has been limited to 70% power.

Region constraint devices were added in early 1980 that are intended to limit fuel block movement by tying the top block in each column to the adjacent blocks. Two series of tests, at power levels up to 70% of full power, conducted since these devices were installed have shown no temperature fluctuations even well into operating regions where fluctuations had been previously observed.

One additional restraint to sustained full-power operation of Fort St. Vrain is the warranty on the existing fuel. The warranty for fuel segments 2 through 9 is limited to reactor operation up to 590 MW(t) until December 31, 1984, which corresponds to a reactor power level of 70%. A special allowance of no more than 500 hr per cycle at higher power levels is covered by the fuel warranty. This limited performance warranty is not based on a technical limitation of the fuel, only a contractual one. The fuel manufacturing technology that is in place will allow full fuel performance warranties; therefore, it is not expected that future plant performance will be limited by this type of problem.

One of the outstanding operational characteristics of the HTGR that has been confirmed by Fort St. Vrain experience is its ability to withstand severe transient conditions without adverse effects. The Fort St. Vrain reactor has been subjected to complete loss of forced circulation several times during its life, the longest duration being approximately 15 minutes, without any damage or increased primary circuit activity. This "forgiving" characteristic of the plant is perhaps the single most important operational feature of the HTGR and provides an effective line of defense against human error which otherwise may have significant consequences with regard to plant protection.

4.2.4.2 HTGR-GT Operations

Because the HTGR-GT is a direct, closed Brayton Cycle plant, its operation will be unique relative to other types of large generating stations. As a result, specialized operator training will be required to operate and maintain the plant.

Three control concepts have been considered for the HTGR-GT, namely:

- Bypass flow from the high-pressure to the low-pressure sections of the power conversion loop.
- Turbine inlet temperature variation.
- Helium inventory variation.

The bypass flow control is the only concept with sufficiently rapid response to limit turbomachine overspeed following a 100% load rejection. This concept has, however, the disadvantage of low efficiency at partial load as shown in Fig. 4.2.4-1. For this reason, it is desirable to combine bypass flow control with turbine inlet temperature control and helium inventory control. This combination is currently the reference control concept and allows 10% step-load pickup with a two-hour, steady-state period between steps. The optimization of partial-load efficiency requires a larger helium inventory than exists in the reference design and represents an added cost, the economics of which depend on whether or not the plant would be utilized in a load-following mode. It can be seen from Fig. 4.2.4-1 that inventory control does have a significant effect on plant performance for partial-load operations. However, rapid load pickup does not appear feasible on economic grounds with inventory control due to the amount of helium which must be injected into the PCR and the rate at which this injection must take place. For these reasons, a utility which may wish to operate the HTGR-GT in a load-following mode must be cognizant of its performance limitations at partial load and analyze the cost effectiveness of the helium inventory system for its specific needs.

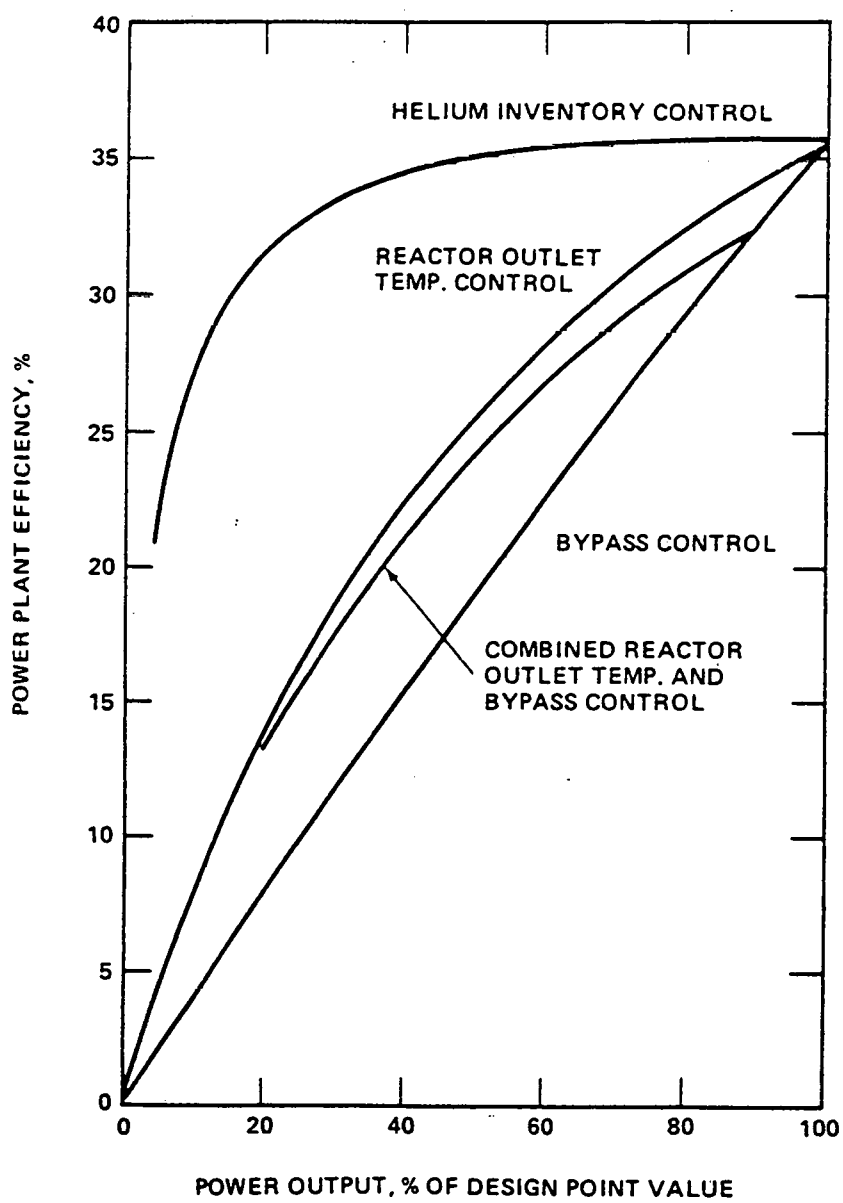


Figure 4.2.4-1 HTGR-GT Part-Load Efficiency for the Various Control Modes

In order to start up the HTGR-GT, the turbomachines must be powered by an external source to attain a rotational speed at which the compressor develops a sufficient pressure ratio to allow the turbine to produce the power needed to drive and accelerate the compressor. This occurs at one-third to one-half synchronous speed and requires approximately 5000 hp per machine. This power will be supplied by motoring the generator with a 5-MW, static-frequency inverter for each machine. During startup, helium inventory is reduced to 40-50 percent of full-load inventory in order to minimize power requirements. The turbomachine will operate in this motoring mode for several hours to allow for thermal conditioning of the system.

In summary, the HTGR-GT has unique operational characteristics which have been identified but not yet fully analyzed due to the preliminary nature of the reference design. As work continues on this concept, a better understanding of its complexities will be gained.

4.2.4.3 Maintenance

As in the previous section, the generic maintenance aspects of the HTGR will be examined by reviewing the experiences of HTGR Steam Cycle plants.

The Steam Cycle HTGR system has many of the same components and numbers of components as conventional fossil and LWR generating plants. Therefore, a significant difference in maintenance arising from component design is not anticipated. However, the dose rates and aggregated radiation exposures due to maintenance activities for the HTGRs are expected to be less than those experienced in LWRs. Since federal law regulates the maximum exposure which can be accumulated by maintenance personnel in a given time period, this characteristic reduces personnel requirements and hence maintenance costs. Experience in operating the early HTGRs supports this expectation, which is the result of the primary circuit activity being relatively low when compared to LWRs.

The Peach Bottom 1 HTGR, operated by Philadelphia Electric Company, generated a total of 1200 GW(e) hours of net power from March 1966 to October 1974. Yearly and cumulative exposure data are listed in Table 4.2.4-1, which was taken from Ref. 9. Because Peach Bottom was a 40-MWe prototype reactor, it can be compared with early, low-power LWRs. Exposure data for Big Rock, Humbolt and Lacrosse are presented in Fig. 4.2.4-2, where they are compared against the Peach Bottom data. The man-rem exposure rate at Peach Bottom can be seen to be appreciably less than the LWRs.

TABLE 4.2.4-1
PEACH BOTTOM HTGR OPERATING EXPERIENCE

Year of Operation	Man-Rem Exposure		Net Power Generation [GW(e)y]		Cumulative Occupational Exposure [man-rem/GW(e)y]
	By Year	Cumulative	By Year	Cumulative	
1967	~3	~3	0.017	0.017	176
1968	~3	~6	0.015	0.032	188
1969	~3	~9	0.0157	0.048	188
1970	~3	~12	0.0163	0.068	176
1971	~4	~16	0.024	0.088	182
1972	~3	~19	0.012	0.102	186
1973	~3	22	0.021	0.1205	183
1974	NA	NA	0.0183	0.140	NA

Source: Ref. 9

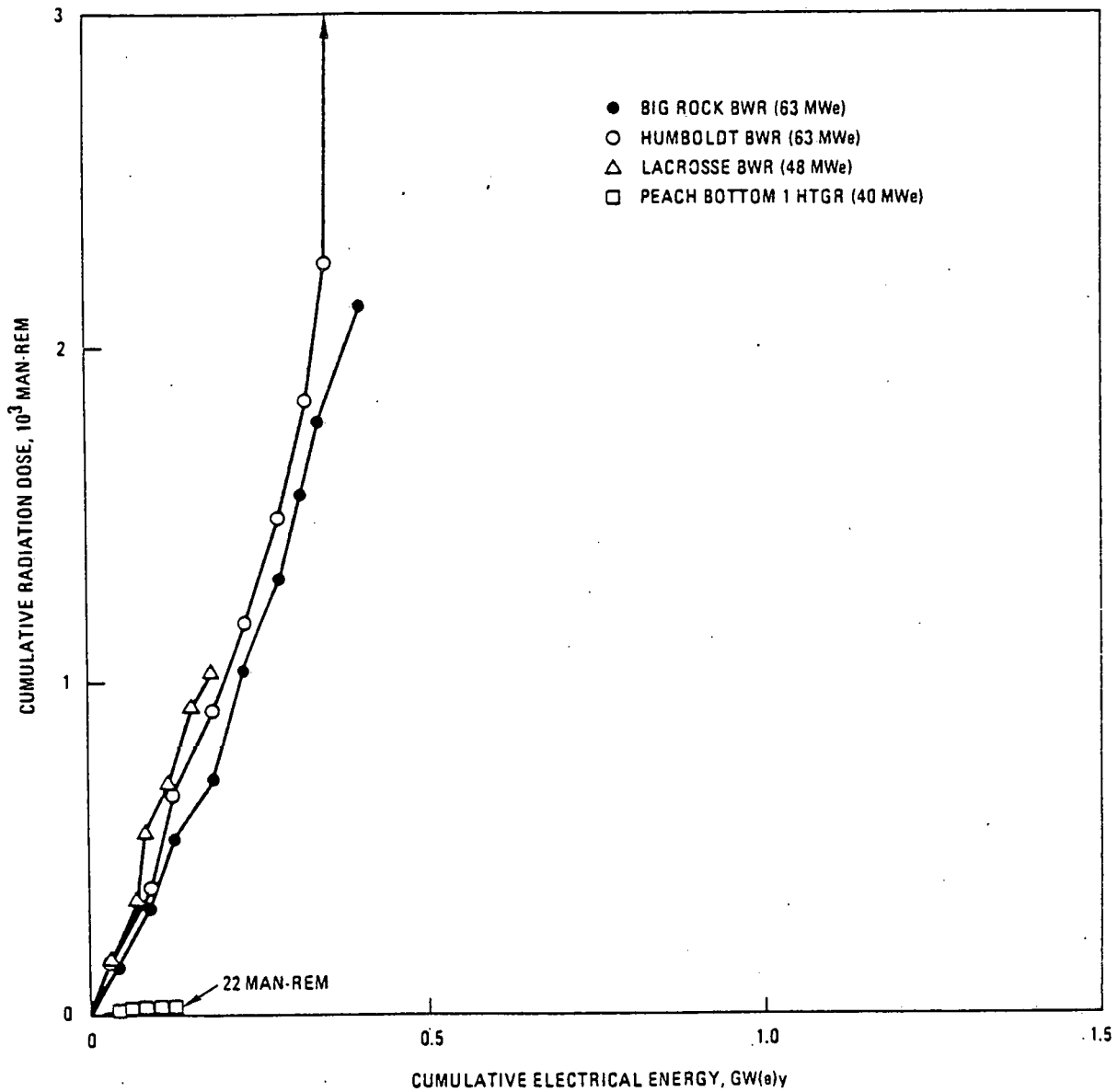


Figure 4.2.4-2 Cumulative Occupational Exposures for Early, Low-Power Nuclear Plants

Source: Ref. 9

At the time of its first refueling in February 1979, Fort St. Vrain had generated 953 GW(e) hours of net power. Personnel exposure data collected indicate that Fort St. Vrain has exposure characteristics similar to those shown by Peach Bottom 1. Table 4.2.4-2 shows the Fort St. Vrain man-rem exposure data for the years 1977 and 1978, which are then compared to similarly sized plants in Fig. 4.2.4-3. Although it is still relatively early in its life, it can be seen that Fort St. Vrain exposures are below all LWR exposures with the exception of the San Onofre PWR, which had relatively equivalent exposures. It is interesting to note that during the Fort St. Vrain refueling outage, exposure to personnel amounted to 0.27 man-rem. Of this total, 0.013 man-rem was due to replacement of one of the main helium circulators, 0.037 man-rem was due to work performed in the hot service facility, and the remaining 0.22 man-rem was due to handling spent fuel elements and control rod drive units (Ref. 10).

For the reference 900-MWe HTGR-SC plant, the refueling operation is expected to result in 5.5 man-rem of exposure, which is consistent with the Fort St. Vrain data when it is extrapolated for reactor size and the time delay that occurred at Fort St. Vrain between shutdown and the start of refueling operations. This compares to an average actual LWR refueling exposure of 39 man-rem in 1976 according to NUREG 0323. Table 4.2.4-3 shows the expected and the design basis exposures for the reference steam cycle HTGR. These data are then compared to LWR experience in Fig. 4.2.4-3. It can be concluded that total HTGR exposure rates should be significantly lower than those of the LWR.

It should be noted that the actual HTGR exposure rates, while they have been at or below predicted values to date, are dependent on fuel design. The Peach Bottom and Fort St. Vrain reactors both utilize highly enriched uranium (HEU-93%) fuel. The reference HTGR fuel is currently LEU-20% enriched uranium, which will result in differences in the generated fission products. For example, Ag-110 m, which is not a major product in HEU fuels, is produced from the more abundant PU-239 in the LEU fuels. With a half life of approximately 250 days, it could prove to be a significant factor to be considered in future HTGR maintenance work and is currently being investigated along with improved particle coatings to retain higher percentages of the fission products.

The economic value of the reduced exposures of the HTGR is difficult to quantify at this time. GCRA performed a survey of its member nuclear utilities in an attempt to determine the worth of this feature to the utility industry. In response to one of the questions, several utilities felt that lower personnel exposure levels would be an advantage for the HTGR as long as it could achieve a 20% to 75% decrease from the current LWR levels of about 500 man-rem/plant year. Other utilities felt that the lower exposures would not prove to be a significant advantage for the HTGR for two reasons: (1) the claims of the lower

TABLE 4.2.4-2
FSV MAN-REM EXPERIENCE

Personnel	Exposure	Averaged Man-Rem	Net Power Generation [GW(e)y]	Rate of Accumulation [man-rem/GW(e)y]
<u>1977</u>				
946	None	0		
55	<100 mrem	2.75		
1	100-250 mrem	0.175		
		2.9	0.0256	113
<u>1978</u>				
896	None	0		
34	<100 mrem	1.7		
0	100-250 mrem	0		
		1.7	0.0695	24
Cumulative		4.6	0.0951	48

Source: Ref. 9

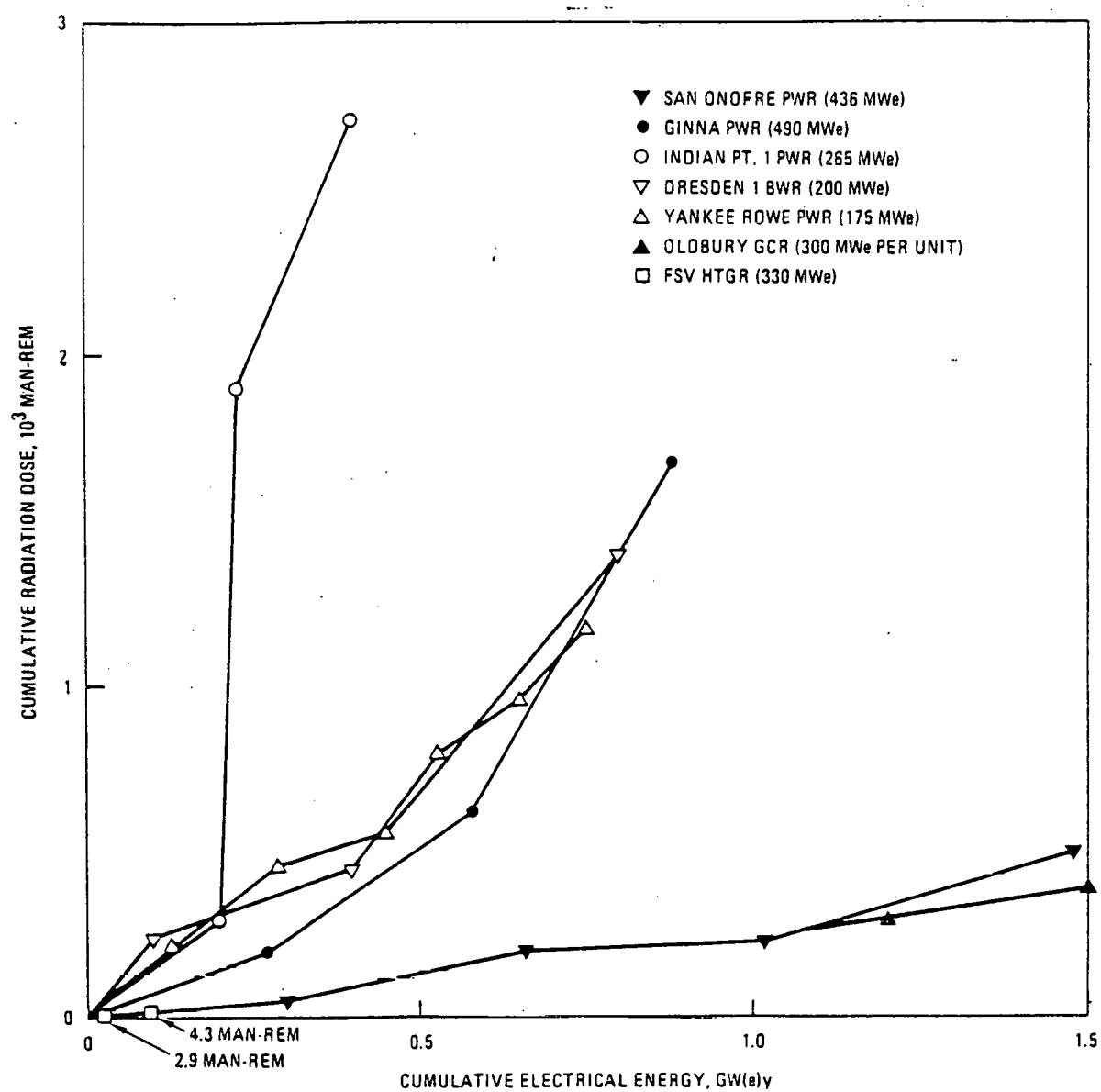


Figure 4.2.4-3 Cumulative Occupational Exposures for Medium-Power Nuclear Plants

Source: Ref. 9

TABLE 4.2.4-3
MAN-REM PREDICTIONS FOR HTGR-SC

Type of Operation	Annual Man-Rem Exposure for 900 MW(e) Unit	
	Expected	Design Basis
Refueling	5.5	20
Reactor Operation and Surveillance	7.0 [*]	20
NSS Maintenance and ISI	10.1	20
BOP Maintenance	25.0 ^{**}	50
Special Maintenance	$\frac{3.2}{50.8}$ ^{***}	$\frac{20}{130}$
Rate of Accumulation [900 MW(e), 80% load factor]	$\frac{50.8}{0.9 \times 0.8} = 70 \frac{\text{man-rem}}{\text{GW(e) y}}$	$\frac{130}{0.9 \times 0.8} = 180 \frac{\text{man-rem}}{\text{GW(e) y}}$

* From low-level noble gas activity in containment building.

** Assumed; no information is available from an architect-engineer.

*** Tube plugging every year @ 1.0 man-rem; steam-generator removal every 10 years @ 1.65 man-rem; circulator removal every 2 years @ 1.0 man-rem.

Source: Ref. 9

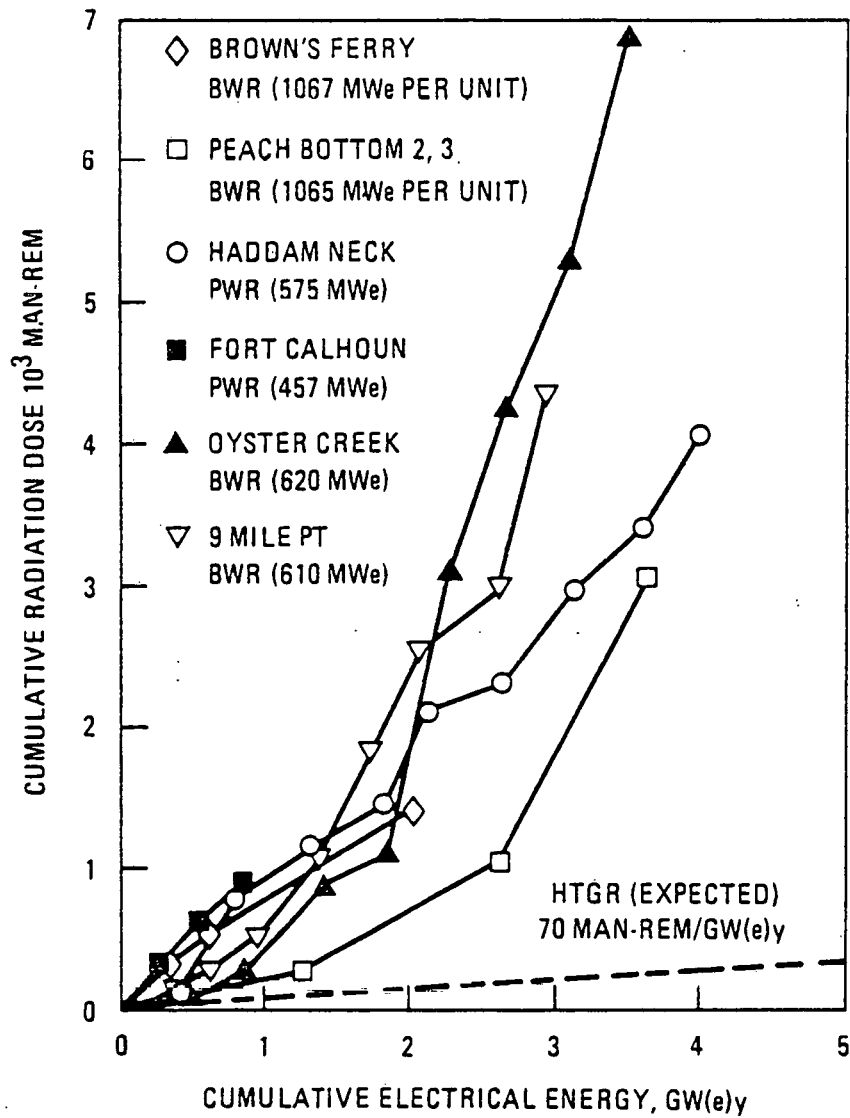


Figure 4.2.4-4 Cumulative Occupational Exposures for Large Nuclear Plants (Part 2)

Source: Ref. 9

exposure levels cannot be given much credibility until they are proven in a commercial size plant having significant operating history and (2) LWR exposures will probably decrease in plants built after the year 2000, thereby decreasing the relatively large perceived HTGR advantage. In trying to quantify the value of a man-rem of exposure, it was quite evident that this number is very site specific. Values ranged from \$600 to \$20,000, with the weighted average around \$1.5K per man-rem, which is close to the 10CFR50, Appendix I guideline of \$1K. It was also noted that for specialty skilled workers, a man-rem can be as high as \$15K to \$20K. The consensus of opinion of the survey was that it is important for the HTGR to retain its potential for lower personnel exposure and that the best means of accomplishing this would be through fuel performance.

A major maintenance activity for any nuclear plant is refueling. For the HTGR, refueling activities and equipment have received a great deal of attention and analysis. A new rapid refueling scheme has recently been developed and is being incorporated into the reference HTGR designs. It is expected that with properly operating equipment, the refueling activities for the reference 2240-MW(t) HTGR can take place in 11 days. A complete description of HTGR refueling activities at Fort St. Vrain is contained in Ref. 10.

4.2.4.4 HTGR-GT Maintenance

Maintenance for the Gas Turbine HTGR has been the subject of an extensive study by GA. The critical path item for maintenance activities is the removal and installation of a turbomachine. With the present design, a period of 21 days is required for removal and installation of a spare machine. It is expected that each plant will have one spare turbomachine which will be available for immediate replacement. After a machine is removed, it is placed in a shielded cask and transported to a maintenance facility where it will be decontaminated to allow hands-on maintenance. It has not been determined as of yet how the turbomachine will be decontaminated and what effect this decontamination will have on machine life. These are major open issues for the HTGR-GT. Preliminary assumptions predict that the exposure for turbomachine removal will be 2.1 man-rem (Ref. 11). Total yearly exposure for reactor turbine system maintenance is estimated to be 22 man-rem.

4.2.5 Fuel Cycle

Two basic fuel cycles have generally been considered for thermal-spectrum reactors: the low-enrichment uranium (LEU) cycle and the thorium cycle. While the LEU cycle has traditionally appeared more attractive for LWR plants, the thorium cycle generally looks advantageous for the HTGR. A number of variants on each of these cycles is possible depending upon whether fuel recycle is utilized and upon the makeup fuel to be used with recycle. The various fuel cycles and reactor systems were the subject of study by the Department of Energy

(DOE) through the Non-Proliferation Alternative Systems Assessment Program (NASAP).

It is becoming increasingly apparent that the selection of a national fuel cycle strategy is, and will continue to be, surrounded by confusion and uncertainty. Probably most apparent is the current uncertainty in policy directions as a result of nuclear weapons proliferation concerns. The economics of fuel recycle must also be regarded as an uncertainty until commercial experience is available with spent-fuel reprocessing, bred-fuel refabrication, nuclear waste processing and waste storage. The commercialization process itself poses a serious uncertainty on recycle implementation, largely due to the uncertainties in acceptable technology directions and the economic incentives for those directions. As a result of these uncertainties, the interest of utilities might be best served by the support of reactor and fuel cycle technologies having sufficient flexibility to accommodate any of the possible directions that might evolve. Not only should a reactor have sufficient fuel cycle flexibility to accommodate any of the several possible preferred directions, but it also should allow an evolution to more advanced technologies as policy definition, technology development and commercialization favor the appropriate evolutionary steps. With the HTGR, it is feasible to deploy an HTGR industry on the basis of a once-through fuel cycle strategy and subsequently adopt a recycle fuel management plan if and when it becomes desirable with no significant change to the reactor. This flexibility of the HTGR would assure that a utility could progress along an evolutionary fuel cycle path with no inconvenience to the potential user. See Appendix A.3 for a description of the planned Spent Fuel Treatment Program.

4.2.5.1 Resource Utilization

The HTGR offers considerable potential for improvements in U_3O_8 utilization efficiency over the LWR, independent of which policy direction might be pursued by this or future administrations. Both plant thermal efficiency and reactor conversion ratios are important factors in the U_3O_8 utilization. Table 4.2.5-1 summarizes U_3O_8 requirements for several fuel cycle alternatives, for both LWR and HTGR plants. The table shows inventory requirements as well as annual makeup requirements. The load factor chosen here is slightly higher than the 65% generally assumed in previous national cost-benefit studies, but somewhat lower than the 75% now being used in NASAP studies. An enrichmant tails assay of 0.1% has been selected (rather than 0.2% now used by the DOE), since a lower assay is expected after the turn of the century as a result of improved enrichment technologies.

Present data indicate that the 20% LEU/Th once-through cycle allows a 30-year U_3O_8 commitment for the HTGR which is only 75% of the standard LWR once-through, i.e., a U_3O_8 commitment improvement of 34% over the LWR. The improvement is still about 20% relative to the LWR with an extended fuel burnup lifetime.

For the 20% LEU/Th recycle mode of fuel management, the HTGR offers a reduction in the 30-year U_3O_8 commitment of almost 50% over that of

TABLE 4.2.5-1

U_3O_8 REQUIREMENTS AND Pu_f DISCHARGE
FOR ALTERNATIVE FUEL CYCLES IN LWR AND HTGR PLANTS*
(LOAD FACTOR = 70%; ENRICHMENT TAILS = 0.1%)

REACTOR	FUEL CYCLE	INVENTORY, ST U_3O_8 /GWe	ANNUAL MAKEUP ST U_3O_8 /GWe-yr	30-YR TOTAL ST U_3O_8 /GWe	Pu PRODUCTION kg/GWe-yr
LWR	3.2% LEU; O.T. (Once-Through)	566.	155.	5061	152.
LWR	4.4% LEU; O.T.	734.	131.	4533	110.
LWR	3.2% LEU; U RECYC	559.	120.	4039	152.
LWR	20% LEU/TH; RECYC	655.	93.	3352	57.
LWR	20% LEU/TH; RECYC	590.	77.	2823	6.
HTGR	20% LEU/TH; O.T.	435.	114.	3741	31.
HTGR	20% LEU/TH; RECYC	400.	79.	2691	31.
HTGR	93% HEU/TH; RECYC	500.	43.	1747	3.
HTGR	93% HEU/TH; RECYC (Heavy Load)	750.	29.	1591	3.

*LWR thermal efficiency assumed at 33.4%;
HTGR thermal efficiency assumed at 39.6%.

the LWR once-through mode, or a commitment improvement of 86%. Other comparisons are equally as impressive. For example, the commitment improvement of the HTGR HEU/Th fuel with recycle over that of the LWR once-through cycle is a factor 2.9, or 190%. As previously indicated, the HTGR offers significant improvements in resource utilization for all comparable cases.

4.2.5.2 Fuel Cycle Economics

On the basis of most recent nuclear growth projections, it is unlikely that the cumulative U_3O_8 requirements will present a significant problem in the next 50 years, but large annual uranium requirements could have an impact on U_3O_8 prices and, therefore, on reactor economics. There are two factors that are particularly important relative to fuel cycle economics:

1. Fuel costs, i.e., the costs associated with fuel consumption and fuel inventory working capital.
2. Handling costs, i.e., the costs associated with fabrication, shipping, reprocessing, refabrication, etc.

The first of these factors is generally recognized as being important, though its importance frequently tends to be overstated. The second factor is not so generally appreciated, perhaps due to unfamiliarity with recycle operations.

- Fuel Costs - At the outset, it is important to note that fuel costs, and particularly those associated with U_3O_8 costs, are not as significant relative to generating costs as are fuel costs in fossil-fueled plants. Specifically, while coal resource and freight costs typically contribute 50% to the cost of generation in coal-fired plants, U_3O_8 supply costs contribute only 5% to 15%, depending on the particular fuel cycle. The more important question, then, is the effect of possible increases in U_3O_8 prices on energy generating costs.

The cost penalty of increased U_3O_8 prices on generating costs is influenced by two factors:

- The cost increase of the enriched uranium (per unit of U-235 contained) resulting from U_3O_8 price increases.
- The cost increase of energy generation resulting from the fuel cost (U-235) increase.

The first factor arises simply because the U-235 cost depends both on the U_3O_8 price and the separative work price; the U_3O_8 price contributes typically about one-half of the U-235 cost. The second factor reflects the fact that resource-efficient reactors use less fuel; therefore, fuel cost increases contribute less to generating costs in the more efficient reactors.

Some of the nuclear fuel cost penalty arising from potential U_3O_8 price increases in the next few decades might also be largely offset by separative work price decreases as Advanced Isotope Separation Technology is introduced. The basic conclusion, then, is that the potential effect of large U_3O_8 price escalations can be minimized by resource-efficient reactors. In addition, the introduction of resource-efficient reactors can help to stabilize the price of U_3O_8 through the lowering of U_3O_8 demands. The HTGR is well suited for the role of this resource-efficient reactor as shown by Table 4.2.5-1.

- Fuel Handling Costs - Typically, the fuel handling cost for the LWR once-through fuel cycle contributes about 15% to the total fuel cycle cost; the fuel cost contributes the other 85%. With recycle, the handling cost fraction increases to approximately 30% to 40%, both because the costs of reprocessing and refabrication are significant and because the fuel cost component is reduced by the more efficient uranium utilization. For the HTGR, fuel handling costs for the once-through fuel cycle contribute about 25% to the total fuel cycle costs while with uranium recycle, handling costs are approximately 50% of total fuel cycle costs. The handling cost component tends to be different for different fuel cycles because of differences in handling difficulties (mostly refabrication) and because of the fraction of recycled fuel that is involved.
- Total Fuel Cycle Costs - Table 4.2.5-2 presents the relative total fuel cycle costs for the HTGR and the LWR for various fuel cycle scenarios. These costs are based on recent comprehensive studies performed by GA using the economic assumptions presented in Table 4.2.5-3. They represent the calculated fuel cost differentials measured in mills/kwhr. It should be noted that a 0.2% enrichment tails was used for the fuel cycle cost calculations. This value was selected in order to maintain a consistent reference cost data base with the government's recently released NASAP studies (Ref. 16).

Based on the data presented in Table 4.2.5-2, it can be seen that for the once-through fuel cycle for the Steam Cycle HTGR, the fuel cycle costs are basically the same as for present LWRs and their once-through fuel cycle. The Steam Cycle fuel cost advantage would be maximized if it were to utilize HEU fuel with full recycle. For the HTGR-GT and HTGR-R, these reactors will have a fuel cycle cost disadvantage relative to the LWR unless HEU and full recycle are allowed. This is basically due to the more conservative fuel design that is required at the higher coolant temperatures (850°C core outlet) of these reactors.

4.2.5.3 Advanced Converter Reactors and Symbiotic Systems

Both the LWR and the HTGR have potential for reactor and fuel cycle improvements. These two systems plus the light water breeder reactor

TABLE 4.2.5-2

RELATIVE FUEL CYCLE COSTS FOR
ALTERNATIVE FUEL CYCLES

<u>Reactor</u>	<u>Fuel Cycle</u>	<u>Relative Fuel Cycle Cost</u>
LWR	LEU; O.T.	1.00
LWR	U+Pu Recycle	0.80
LWR	U Credit	0.91
HTGR-SC	LEU/Th(20%); O.T.	0.99
HTGR-SC	HEU/Th(93%); U Recycle	0.73
HTGR-GT	LEU/Th(20%); O.T.	1.10
HTGR-GT	HEU/Th(93%); U Recycle	0.78
HTGR-R	LEU/Th(20%); O.T.	1.11
HTGR-R	HEU/Th(93%); U Recycle	0.81

TABLE 4.2.5-3
FUEL CYCLE COST ECONOMIC ASSUMPTIONS

Capacity Factor	70%
Tails Assay	0.2%
Startup Date	6/95
Plant Efficiency	39.5% HTGR and 33.0% LWR
Inflation Rate	6%
Working Capital Rate*	15.6%
Discount Rate*	10.2%
Levelizing Period	30 yrs.
<u>Fuel Costs ('80 \$M)</u>	
Conversion (\$/kg)	5.0
Enrichment (\$/kg-SWU)	100.0
U ₃ O ₈ (\$/lb)	Reference - 1990 - 45.0 2010 - 75.0 2020 and onward - 120.0
U233/U235 Parity Ratio	1.15
Pu/U235 Parity Ratio	0.60

*The discount rate is the weighted cost of capital. The working capital rate is the interest rate on monies used to purchase fuel and fuel handling services and includes the weighted cost of capital and the federal tax on the equity capital.

(LWBR) are the candidates with the greatest potential as advanced converter reactor (ACR) concepts; however, the HTGR appears to offer the best possibility for an economically attractive, resource-efficient reactor.

Although traditional thinking some five to ten years ago envisioned the complete replacement of thermal-spectrum reactors by fast breeder reactors (FBR) in the long-range future, it is now becoming apparent that the optimum nuclear system will consist of a symbiotic combination of ACRs and FBRs. Several factors contributing to this realization are:

- The nuclear growth projections now indicate that severe resource strains will not be imposed on the mining and milling industry for some 30 to 50 years, particularly if more resource-efficient reactors and fuel cycles are introduced.
- The cost penalty associated with increased U_3O_8 prices will not be substantial if resource-efficient reactors and fuel cycles are introduced.
- The capital cost and operating cost of the liquid metal fast breeder reactor (LMFBR) now appear to be such that very high U_3O_8 prices would be required to justify the LMFBR (without improvements or modified fuel cycles).

A strategy creating a symbiotic relationship with the coupling of four HTGRs to one fast breeder reactor is one with much potential and many long-range benefits. In order to implement such a strategy, it would be necessary to create the marketplace for U-233 utilization prior to FBR deployment rather than subsequent to it. In this sense, the thorium cycle could actually be used to expedite the introduction of the FBR, with the ACR becoming the nuclear energy "work horse" of the future.

4.2.5.4 HTGR Flexibility

Two basic fuel cycles were examined in the NASAP studies for thermal spectrum reactors:

LEU (with $\leq 20\%$ uranium enrichment) Cycle
HEU (with $> 20\%$ uranium enrichment) Cycle

An LEU cycle with 20% uranium enrichment has received considerable attention, particularly in the NASAP studies because:

- The enrichment of the initial feed material is below that of weapons-grade U-235.
- The plutonium bred into the cycle is largely consumed so that the discharge plutonium content is substantially reduced over that of the LEU cycle.
- The U-233 is (or can be) "denatured" with U-238.

While the primary NASAP attention for near-term utilization has centered on the once-through fuel cycle using LEU fuel, it is expected that greater economic pressure for recycle will develop as the price of U_3O_8 increases. The NASAP studies indicate that one desirable possibility for subsequent recycle in thermal-spectrum reactors would involve the use of the thorium cycle with the recycle of either de-natured U-233 or gamma-active U-233.

Not only should a reactor have sufficient fuel cycle flexibility to accommodate any of the several possible preferred directions, but it should also allow an evolution to more advanced technology possibilities as policy definition, technology development, and commercialization favor the appropriate evolutionary steps. It is quite practical to deploy an HTGR industry on the basis of a once-through fuel cycle strategy and subsequently adopt a recycle fuel management plan if and when it becomes desirable with no significant change to the reactor. In contrast, the development of an advanced LWR involving movable fuel control (as in the LWBR) or spectral-shift control would require major changes in the reactor design. In addition, the introduction of breeder reactors would require the deployment of an entire recycle industry before the breeder reactors could be used. The flexibility of the HTGR, however, would allow utility users to progress along an evolutionary fuel cycle path with no inconvenience during successive steps.

In the near term, it is expected that the LEU/Th (20%) once-through fuel cycle with fuel storage would represent the optimum direction for the HTGR in terms of national policies, although the economic incentives for utilizing this cycle are meager. At some appropriate future date, the U-233 stored in the spent fuel could be separated and recycled in the same reactor. Finally, when U-233 becomes available from an external source such as an FBR, the same HTGR plant could then utilize the U-233 as a makeup fuel and the plant would perform as a near-breeder reactor, i.e., with a conversion ratio of approximately 0.9.

Hence, the flexibility of the HTGR allows it to accommodate any policy or economically attractive technology direction.

4.2.5.5 Summary

Future directions for nuclear fuel cycles are being complicated by uncertainties arising from national policies, economic factors and industry commercialization problems. While long-range development should favor the recycle of fuel in resource-efficient reactors, it is desirable for utilities to have access to reactors that operate economically on a once-through fuel cycle in the near term but that can accommodate the more efficient fuel cycles as policies and facilities allow these improvements. The HTGR has the unique flexibility to adapt to these changing conditions with no redesign of the reactor itself. Furthermore, the efficiencies of the alternative cycles for the HTGR are such that improved resource utilization will occur.

When compared to LWR fuel costs, the economics of the HTGR fuel cycles lead to the following conclusions:

- The HTGR fuel cycle cost advantage is appreciable only when HEU and recycle are utilized. It is important that this option be maintained as the HTGR fuel cycle goal.
- The standard 33,000-MWD/T LWR once-through and the LEU/Th (20%) HTGR Steam Cycle once-through fuel costs are the same. Extending the LWR burnup to 50,650 MWD/T leads to a 7% reduction in the LWR once-through costs.
- For a recycle LEU/Th (20%) cycle, the HTGR and LWR costs are within 2%. The previously calculated HTGR advantage of 8%-10% has diminished due to the \$23,800/block refab cost, which is twice the HEU/Th refab cost due to much lower recycle block throughputs for the LEU cycle.

Future work which may affect the above conclusions must be performed to resolve the uncertainties that exist regarding HTGR waste treatment, particularly for C-14, and HTGR fuel block shipping and packaging costs. It is clear, however, that one of the primary incentives for the HTGR, namely the flexibility of its fuel cycle, will remain.

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5.0 LEAD PROJECT AND PROGRAM

5.1 Lead Project Technical Definition

5.1.1. Reactor Turbine System

This section presents a brief description of the reactor turbine system (RTS) for a direct-cycle, HTGR-GT 2000-MW(t), two-loop nonintercooled lead plant (Fig. 5.1.1-1).

The RTS will operate with a core outlet temperature of 850°C (1562°F) and a primary coolant compressor discharge pressure of 8.41 MPa (1220 psia). At nominal (100%) power, the primary helium flow at the compressor discharge of each loop will be 563 kg/s (4.471×10^6 lb/hr); this will produce a nominal net output of 800 MW(e) for a plant efficiency of 40%. A plant meeting these criteria provided the basis for the component designs that have been developed at GA. A more optimized design, with a 2170-MW(t) core, has recently been defined, and this has become the basis for follow-on gas turbine design efforts. Only the 2000-MW(t) plant variant is described in this section. Section 5.2 gives further details on the plant performance of both plants. The performance of a gas turbine plant can be further improved by the addition of a binary or bottoming cycle, e.g., using ammonia or flashing steam; these cycles are described in the general discussions of the commercial gas turbine plant in Section 3.

The HTGR nuclear power plant contains a graphite-moderated, helium-cooled reactor core located within a PCRV. The scope of supply of the RTS extends to the cooling water nozzles at the inlet and outlet of the precooler, to the cooling water nozzles at the inlet and outlet of the core auxiliary heat exchanger (CAHE), and to similar nozzles for the PCRV liner cooling water system. The RTS comprises the following five major systems:

1. PCRV.
2. Reactor internals.
3. Reactor core.
4. Primary coolant.
5. Core auxiliary cooling system (CACS).

The design and development program required to support the Lead Project is presented in detail in Appendix A of this report.

5.1.1.1 PCRV System

The PCRV system comprises the PCRV structure, the cavity liners, penetrations, closures, and the thermal barrier. The functions of this system are to provide the primary coolant pressure boundary, to house the major RTS components, and to provide a biological shield around the reactor core.

The prestressed concrete (PCRV) structure is a multicavity vessel characterized by a central core cavity and peripheral cavities that house

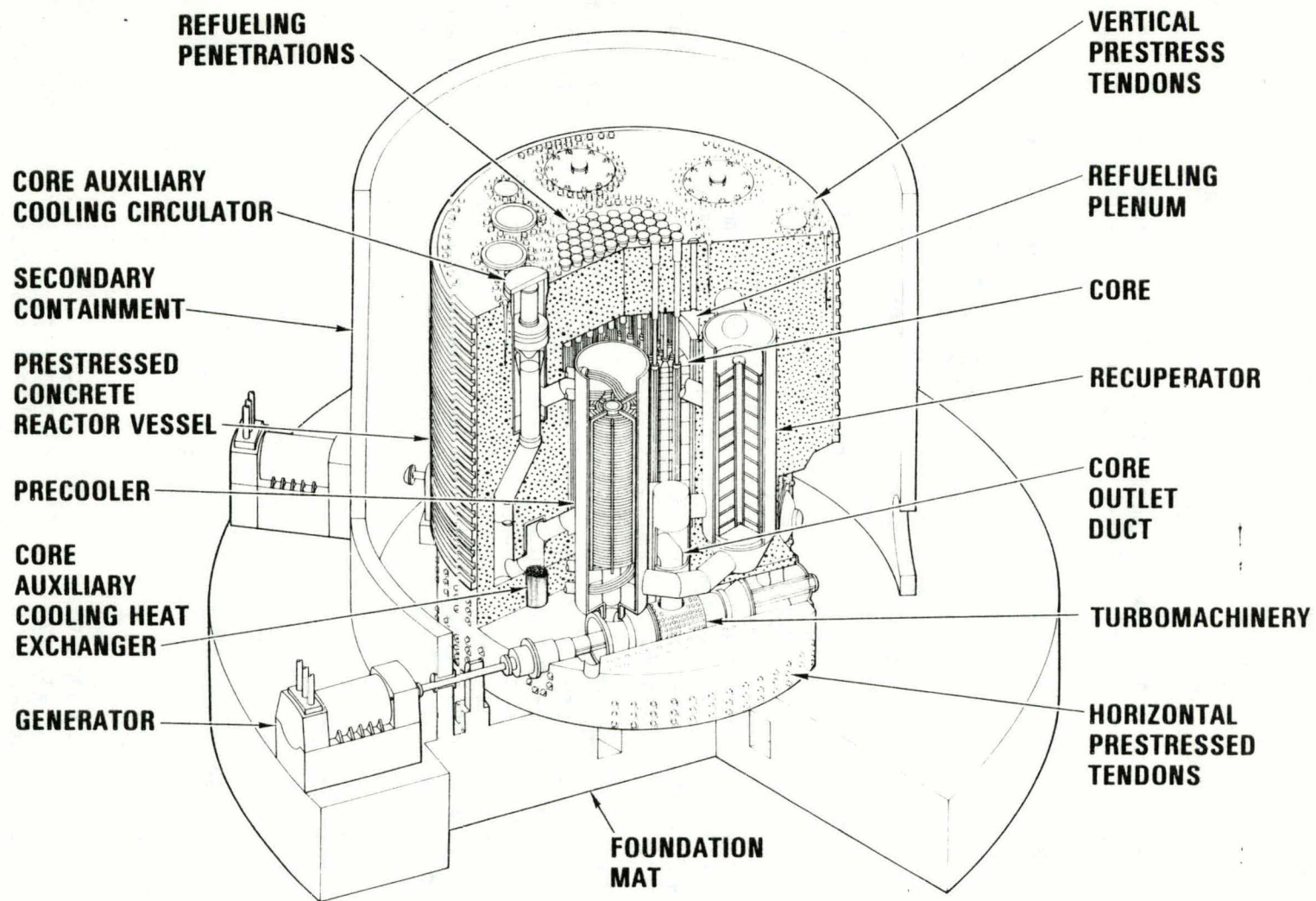


Figure 5.1.1-1 Two-loop 2000-MW(t) HTGR-GT power plant

the main heat exchangers and the CACS components. The turbomachine cavities are located horizontally in the bottom head of the PCRV. The vessel is prestressed circumferentially by wound strand cables and vertically by linear strand tendons. The bottom head region of the PCRV uses horizontal linear strand tendons instead of circumferential prestressing. These two prestressing systems provide sufficient precompression in the concrete to resist the primary and secondary loads during the life of the vessel. The principal design parameters of the PCRV are given in Table 5.1.1-1.

The steel liners and the penetration closures form the continuous, gas-tight, primary coolant pressure boundary of the PCRV. The penetrations, closures, and PCRV concrete act collectively to resist primary coolant pressures. The liner and penetration anchors transmit loads to the concrete from the internal equipment supports and the steel closures, respectively; the concrete, in turn, transmits these loads to the vessel support structure. Circulating water, in tubes attached to the liners, cools the liners at their interface with the concrete, removing the heat that permeates the thermal barrier and that is generated in the steel and concrete by ionizing radiation.

The thermal barrier minimizes heat losses from the primary coolant and maintains liner and concrete temperatures within acceptable limits. Various temperature zones of the PCRV use different thermal barrier types. Typically, the thermal barrier consists of layers of fibrous insulation held against the liner by metal cover plates. Attachment fixtures, designed to minimize thermal conduction to the liner, anchor these plates to the liner.

An internal valve system, which controls the helium bypass between the high- and low-pressure zones of the primary coolant system, provides pressure relief to the PCRV. Table 5.1.1-1 includes the design allowable maximum pressures.

PCRV structural instrumentation provides data on PCRV response to temperature and pressure loads to verify the adequacy of the design and the quality of materials and construction.

5.1.1.2 Reactor Internals System

The reactor internals system comprises six major components: core support floor structure, core lateral restraint, permanent side reflector, core peripheral seal, upper plenum structure of the in-vessel refueling system, and core outlet duct.

The major function of the core support floor is to provide vertical support for the reactor core (Fig. 5.1.1-2). The core support floor consists of graphite core support blocks resting on graphite posts, which in turn are supported on graphite seats atop ceramic bases; the open space between the graphite posts forms the lower plenum for the core outlet gas. Each core support block under the active core contains gas flow passages and a plenum, which allows mixing of the

TABLE 5.1.1-1
PCRV MAJOR PARAMETERS
DIRECT-CYCLE, HTGR-GT 2000-MW(t) TWO-LOOP PLANT

Type	Multicavity PCRV
Overall dimensions	
Diameter, m (ft)	32.61 (107.00)
Height, m (ft)	35.36 (116.00)
Core offset from PCRV, m (ft)	1.07 (3.50)
Core cavity (a)	
Quantity	1
Diameter, m (ft)	11.33 (37.17)
Height, including in-vessel refueling plenum, m (ft)	18.92 (62.08)
Recuperator and precoolers cavity (a)	
Quantity	2
Diameter at mid-height, m (ft)	5.74 (18.83)
Height, m (ft)	25.55 (83.83)
Auxiliary loop cavity (a)	
Quantity	3
Circulator cavity diameter	
At mid-height, m (ft)	2.74 (9.00)
At top, m (ft)	3.05 (7.67)
CAHE bundle cavity diameter	
At mid-height, m (ft)	2.34 (7.67)
At bottom, m (ft)	3.05 (10.00)
Turbomachine cavity (a)	
Quantity	2
Diameter, m (ft)	4.72 (15.50)
Length, (turbine pullout to generator pullout end), m (ft)	20.02 (65.67)
Maximum cavity pressure	
High pressure, MPa (psig)	8.17 (1185)(b)
Low pressure, MPa (psig)	3.27 (474)
Equilibrium pressure, MPa (psig)	6.21 (900)
PCRV support	Integrated with containment

(a) Diameter dimensions are to the insides of the cavity liners.

(b) Maximum cavity pressure resulting from heat balance is 8.69 MPa (1260 psig), which may result in possible increase in PCRV dimensions.

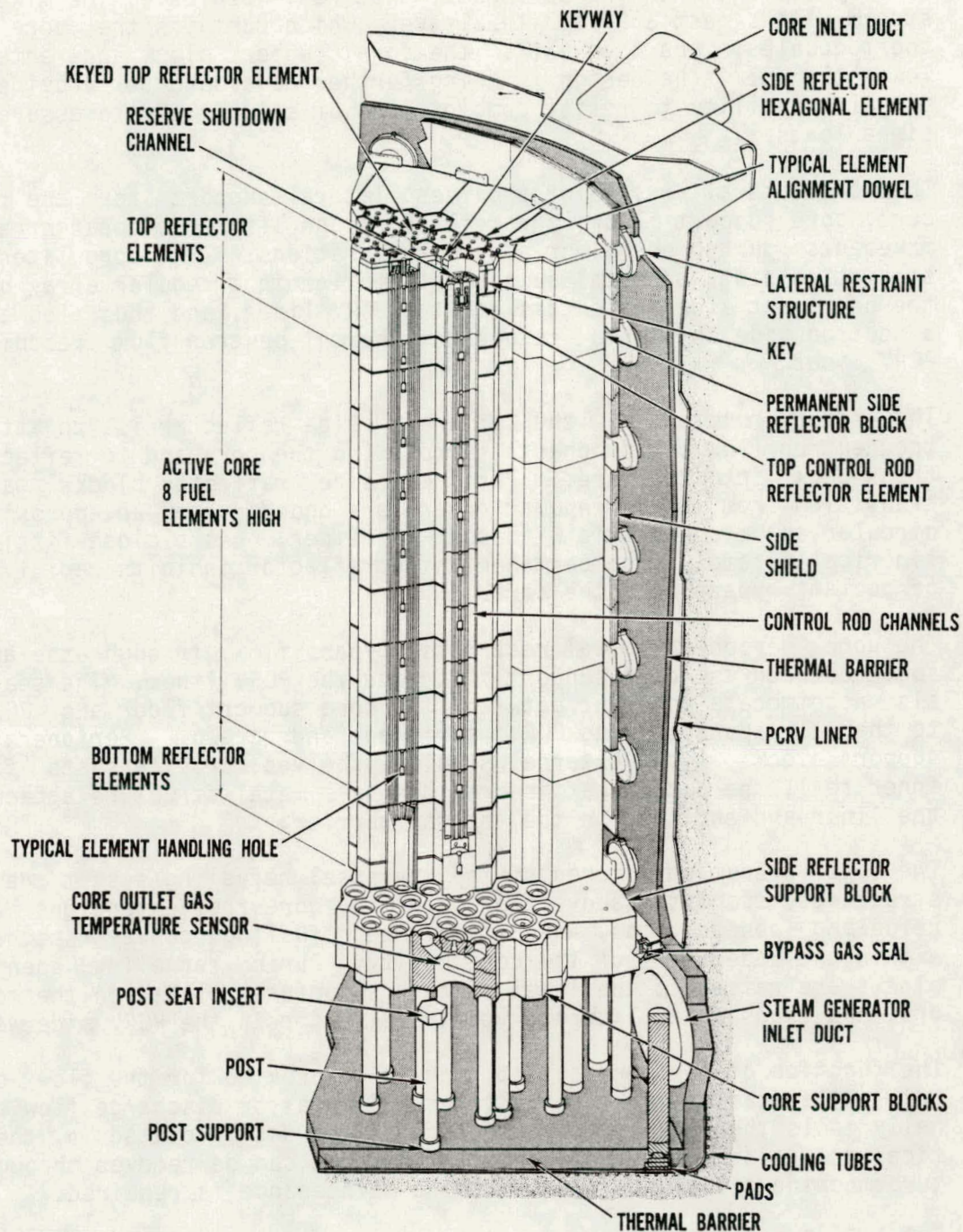


Figure 5.1.1-2 HTGR reactor core and support arrangement

primary coolant from the columns of that fuel region. The mixed gas stream flows past a graphite sleeve, which contains the core outlet thermocouples, and then exits the core support block and enters the lower plenum. The design is being further developed to provide additional capability to resist combinations of seismic and pressure transient loads.

The core lateral restraint provides lateral support for the reactor core, core support floor, and reflectors and also accommodates relative movements during normal or abnormal operations. The core lateral restraint consists of metal support assemblies in a regular array between the permanent side reflectors and the PCRV liner, and thus also acts as a neutron side shield to reduce the thermal neutron flux reaching the PCRV liner.

The primary function of the permanent side reflector is to attenuate the neutron flux to components surrounding the core and to reflect neutrons back into the core. Permanent side reflector blocks make the transition from the irregular outer core boundary to an approximately circular shape concentric with the PCRV liner. As a close-fitting cylindrical assembly, the permanent side reflector inhibits radial inflow of coolant bypass into the core.

The core peripheral seal restricts bypass flow through the annular space between the core support floor and the PCRV liner. The seal must also accommodate movement between the core support floor and PCRV due to thermal expansion and PCRV shrinkage and creep. Peripheral core support blocks, with external sloping shelves, provide seats for the inner seal; the outer seat is provided by a metal structure attached to the liner and enclosed in the thermal barrier.

The upper plenum structures of the in-vessel refueling system are steel structures, supported above the top of the core from extensions of the refueling penetrations, which carry the refueling conveyor mechanisms within the upper part of the core cavity. During refueling, spent fuel blocks are raised to the level of this structure, placed on the conveyors, and transported laterally to the elevator in the PCRV side wall.

The function of the core outlet duct is to provide for the flow of hot core outlet gas to the turbine inlet. Compressor discharge flow externally cools the free-standing section of the duct, located in the vertical compressor discharge cavity. The duct can be removed through the turbomachine cavity when turbomachine maintenance is required.

5.1.1.3 Reactor Core System

The function of the reactor core system is to provide nuclear-generated heat for the HTGR-GT plant. This system consists of the fuel elements, reflector elements, top layer/plenum elements, and the startup neutron sources.

The hexagonal fuel elements and reflector elements form columns supported by the core support blocks. Each core support block under the

active core supports one refueling region, which has a central control column and up to six surrounding fuel columns. Each refueling region has its own flow orifice valve. Two rows of hexagonal reflector columns, which are surrounded by the permanent side reflector, enclose the fuel regions.

The reactor coolant enters the reactor core system after passing through the region flow control equipment and then flows downward through the upper reflector, active core, and bottom reflector while absorbing the nuclear-generated heat. The coolant exits the system to the core support region where it collects, mixes, and finally discharges to the lower core cavity plenum.

Table 5.1.1-2 summarizes the major reactor core system design parameters, and Fig. 5.1.1-3 shows the core layout for the 2000-MW(t) HTGR-GT plant.

The fuel cycle uses a patch loading scheme, which refuels a certain fraction of the core, known as a "fuel segment," on an annual basis. Thus, on the 4-yr cycle, reloading of about one-fourth of the fuel regions occurs each year, and each region would remain in the core for 4 yr. The refueling scheme is chosen so that the regions to be refueled are symmetrically distributed throughout the core. Figure 5.1.1-3 illustrates the region distribution for a 4-yr fuel cycle. The letters A, B, C, and D for each of the 61 fuel regions identify the fuel segment distribution within the core for the 4-yr cycle.

The fuel elements (Fig. 5.1.1-4) are graphite blocks with the dual function of containing the fuel and acting as a moderator. The fuel elements have drilled coolant passages and parallel fuel channels, which contain the fuel rods. Individual fuel rods contain fuel particles distributed in a graphite matrix. The central fuel column of a region includes the control rod and reserve shutdown holes.

The reflector elements are graphite hexagonal right prisms that have coolant holes, control rod and reserve shutdown holes, and shielding material as required but do not contain fuel.

The top layer/plenum elements include the alloy steel hexagonal components that provide the flow plenums for distributing the flow from the region flow control valves to the individual columns, to the lateral restraint during refueling, and to the flow control valve/lower guide tube assembly support.

The startup neutron sources, which consist of Cf-252 in a suitable container, are inserted into the fuel elements to provide a source of neutrons of sufficient strength to ensure a safe, controlled approach to reactor criticality.

The HTGR core can accommodate many different cycles including those with fully enriched uranium and thorium (HEU/Th) and those with low

TABLE 5.1.1-2
BASIC CORE PARAMETERS FOR HTGR-GT

Nominal core power, MW(t)	2000
Nominal core power density, W/cm ³	6.6
Core layout	See Fig. 5.1.1-3
Number of fuel blocks per column	8
Number of fuel columns	415
Number of seven-column regions with variable orifices	55
Number of five-column regions with variable orifices	6
Number of control rod pairs	61
Number of power rods	61
Core volume, cm ³	3.014 x 10 ⁸
Fuel cycle	
Initial fuel cycle	LEU/Th
Refueling cycle	4-yr cycle 25% reloaded each year or "thick buffer"
Fissile material/particle	UC ₂ /TRISO
Fertile material/particle	ThO ₂ /TRISO
Fuel enrichment, % U-235	19.9

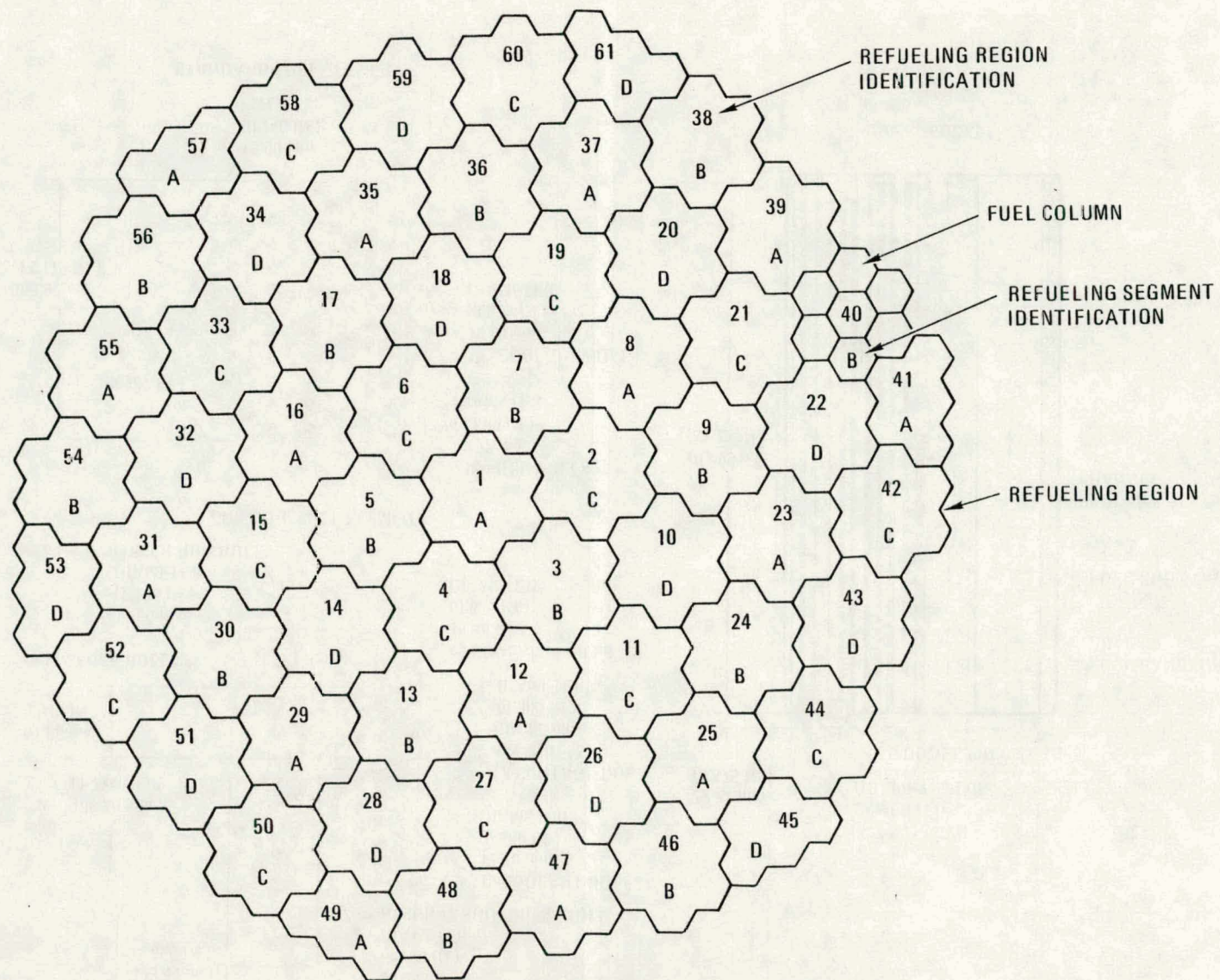


Figure 5.1.1-3 HTGR-GT core layout

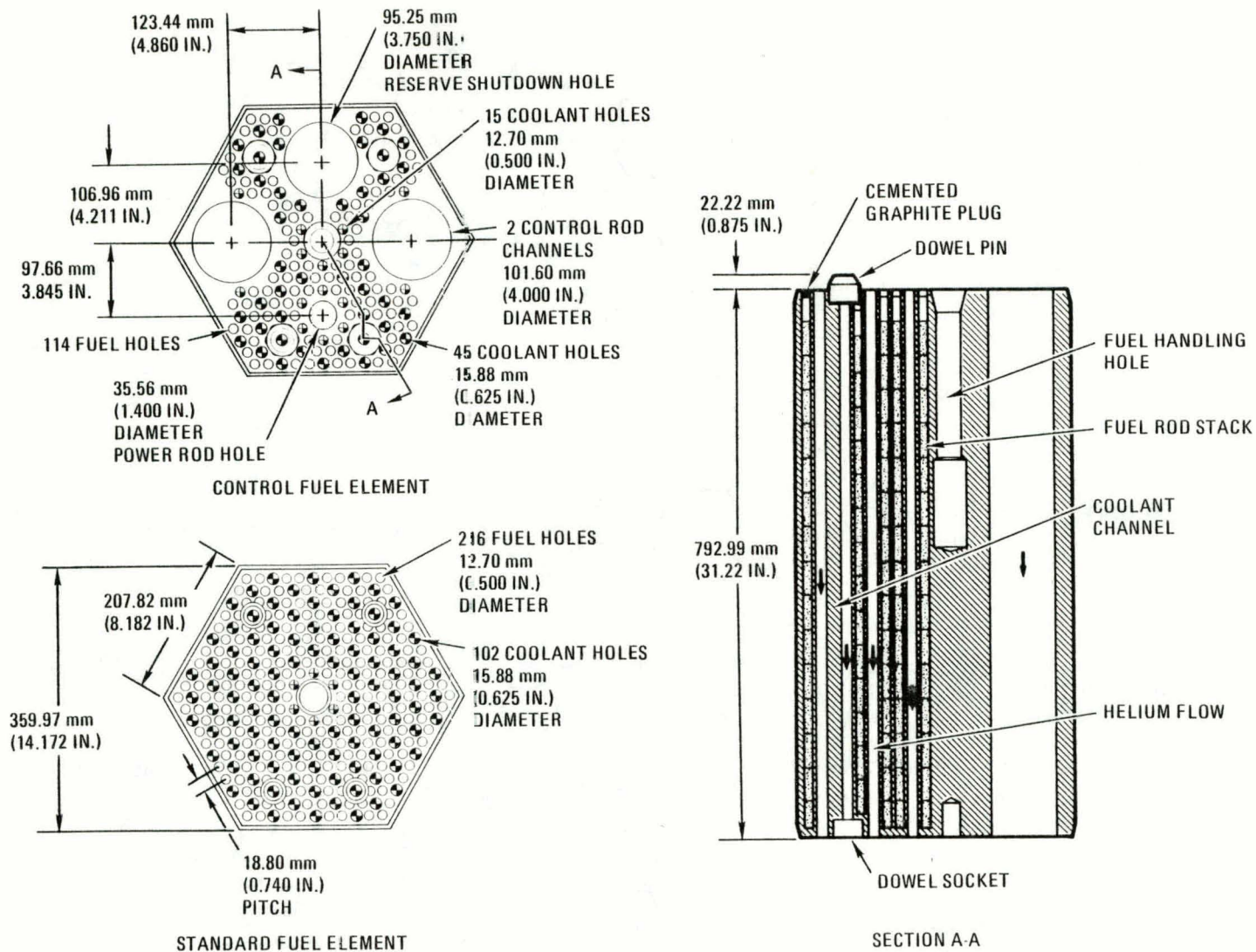


Figure 5.1.1-4 HTGR fuel element

enriched uranium (less than 20% U-235) with and without thorium (LEU/Th and LEU cycles).

For the reference equilibrium cycle with HEU, the fissile kernel is uranium carbide surrounded by a buffer layer of low-density pyrolytic carbon; a thinner layer of high-density pyrolytic carbon; a layer of silicon carbide, which improves containment of fission products; and an outer layer of high-density pyrolytic carbon, which adds strength to the coating. This coating system is referred to as a TRISO coating. The fertile kernel is thorium oxide surrounded by a TRISO coating.

In LEU cycles the fuel design would be essentially unchanged, except the fuel rod diameters might be adjusted to accommodate the different fuel loadings. The fuel particle designs would be similar to those used for the reference cycle.

5.1.1.4 Primary Coolant System

The primary coolant system comprises those subsystems and components required for the transfer of energy from the reactor core to the electric generator and secondary coolant systems. The major system components involved in this function are the turbomachine, recuperator, pre-cooler, and control valves, which constitute a single power conversion loop.

Sensors measure the helium temperature at the exit of each core support block and at the turbine inlet. Adjustment of the flow orifice valve or control rod configuration, respectively, controls these temperatures. The orifice valve for manual adjustment of region flow is at the plenum element at the entrance to the core. The average turbine inlet temperature automatically regulates the control rods.

The system uses an inventory of helium to transfer energy from the reactor core to the gas turbine unit, where expansion of the helium drives the compressor and electric generator. After leaving the turbine, low-pressure helium passes through the recuperator and gives up heat to the high-pressure helium flowing countercurrently from the compressor discharge. Leaving the recuperator, the cooled low-pressure helium flows to the pre-cooler and transfers additional heat to the secondary coolant loop (circulating water system). The cold, low-pressure helium then passes to the compressor, which pumps it through the recuperator and the reactor core to complete the circuit.

The 400-MW(e) single shaft, direct drive helium turbomachine has 18 compressor stages (with a pressure ratio of 2.6) and eight turbine stages (Fig. 5.1.1-5). Two journal bearings support the rotor (of welded construction). The overall length of the machine is 11.3 m (37 ft) and its diameter is 4.12 m (13.5 ft); the weight of the machine is 277,000 kg (305 tons).

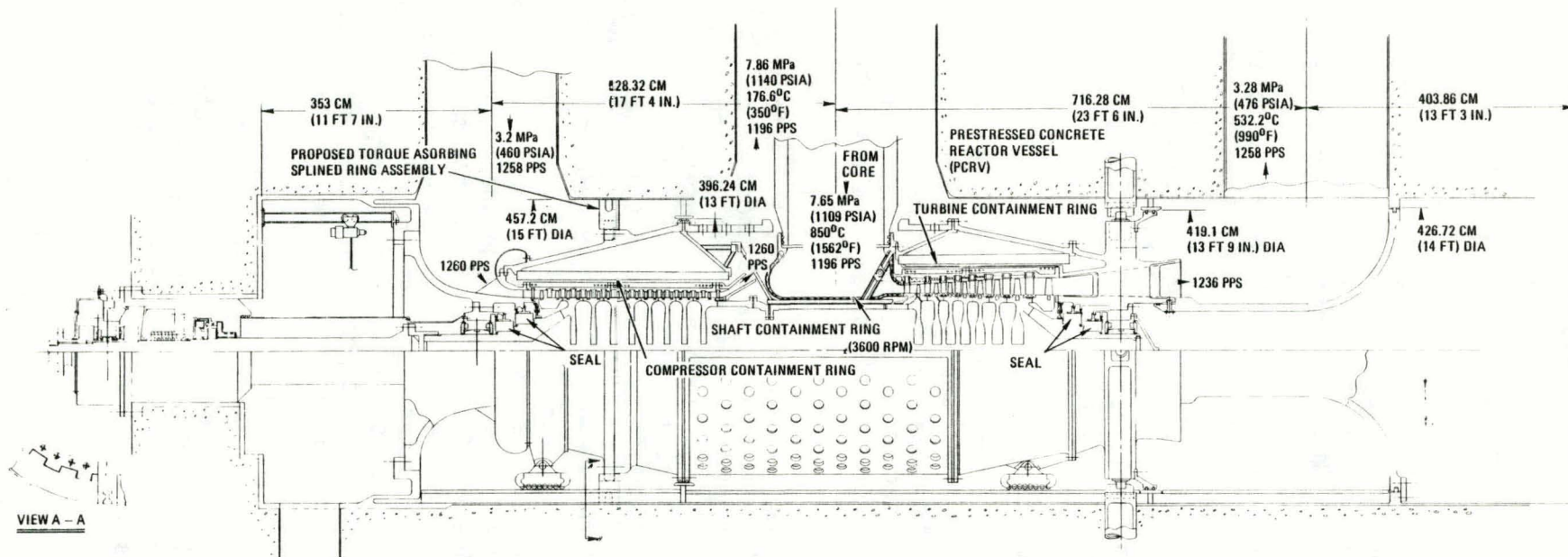


Figure 5.1.1-5 HTGR-GT gas turbine

The machine design incorporates rotor burst protection by means of containment rings around the compressor and turbine rotor-bladed sections. Man-access cavities in the PCRV provide for inspection and limited maintenance work on the journal bearings. Shielding isolates the spaces in which the bearings are located from the helium primary coolant; purged gas from the purification system provides an acceptable radiological environment for man access. The generator drive is at the compressor end of the turbomachine, and the thrust bearing and No. 3 journal bearing are external to the primary coolant system (in the turbomachine cavity plug) to facilitate inspection and maintenance.

The turbomachinery couples to an all water-cooled generator, which is located outside the secondary containment boundary. The shaft penetration of the containment building contains a bearing and seal system to eliminate any gas leakage from within the containment building.

The recuperator, located entirely in the PCRV, is a tubular, straight-tube, gas-to-gas counterflow heat exchanger that recovers heat from the turbine discharge helium by preheating the compressor discharge helium before it enters the core. The "stayed tubesheet" design (Fig. 5.1.1-6) contains two flat tubesheets, which are connected by about 53,000 straight tubes. An outer shroud extends the full length of the tubes, is attached to the lower tubesheet, and is perforated top and bottom for shell-side gas flow. A pipe in the middle of the tube bundle provides inspection access, and an "egg-crate" design supports the tubes laterally. Low-pressure helium from the turbine enters the upper plenum, flows down through the tubes, and leaves the lower cavity through a duct to the precooler. High-pressure gas from the compressor enters the bundle at the lower end, turns 90° and flows upward parallel to the tubes, turns 90° at the top, and exits to the core. The tubes, therefore, are externally pressurized and loaded in tension, thus "staying the tubesheets." Table 5.1.1-3 provides pertinent data on recuperator geometry and parameters.

The precooler, which is contained entirely within the PCRV, is a helically coiled, finned tube, gas-to-water cross counterflow heat exchanger. The precooler removes heat from the hot gas that leaves the recuperator after having been discharged from the turbine (Fig. 5.1.1-7). It comprises a helical bundle supported by eight support plates, which extend the length of the bundle. The support plates are welded to an inner and outer shroud and a cavity-liner flange supports the entire assembly. Lead-in and lead-out tubes connect the water inlet/outlet penetrations to the helical bundle. The (circulating) cooling water inlets are at the top of the module and the exits are at the bottom. Hot recuperator discharge gas enters the bottom of the precooler, flows upward over the outside of the tubes, turns 180°, flows downward through the inner shroud duct, and exits at the bottom into the compressor. The two separate cooling water lines can be isolated and drained independently of each other, permitting continued reactor

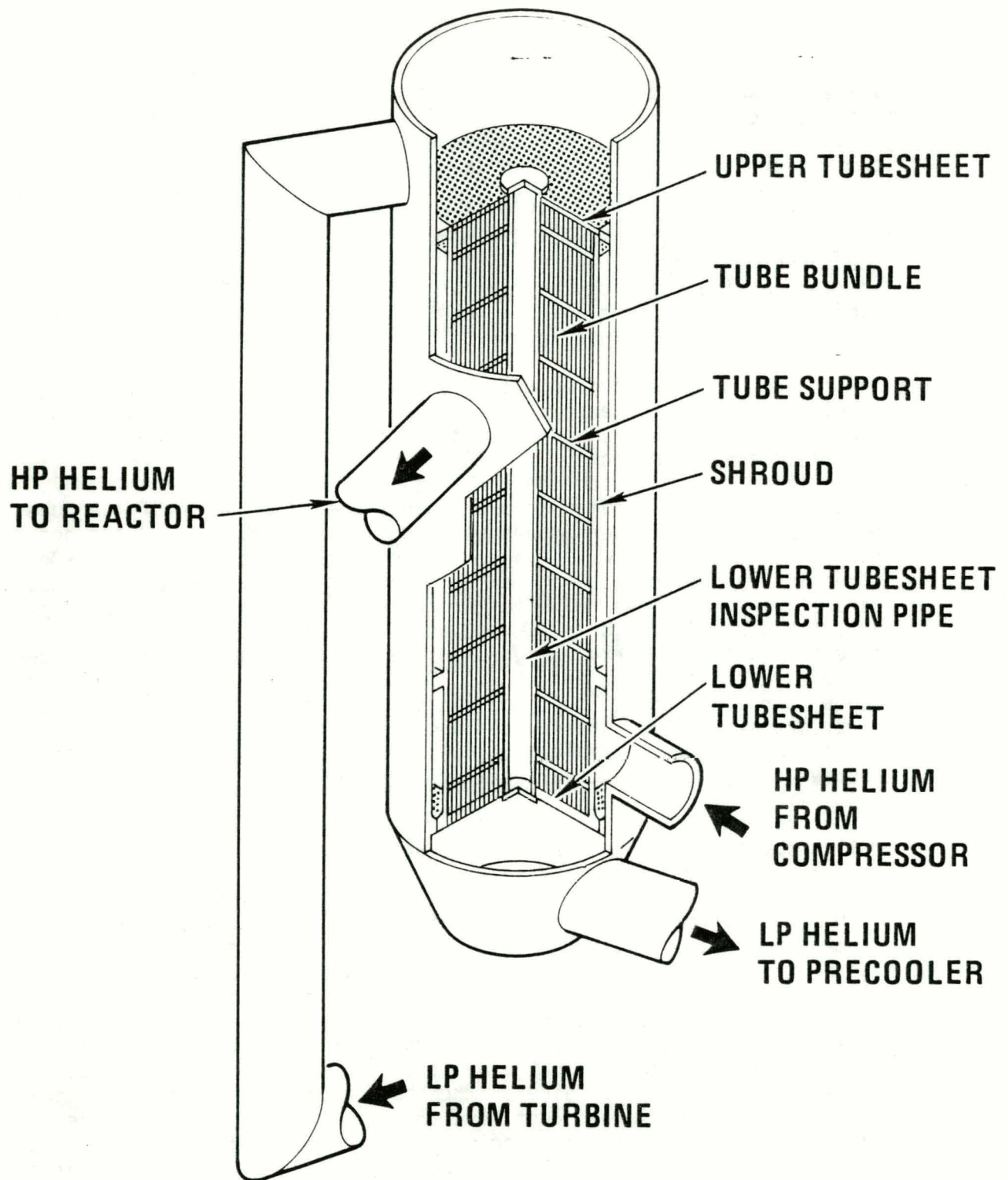


Figure 5.1.1-6 Stayed tubesheet recuperator

TABLE 5.1.1-3
 RECUPERATOR GEOMETRY AND PARAMETERS
 HTGR-GT 2000-MW(t) TWO-LOOP PLANT

Inner shroud o.d., m (ft)	0.66 (2.17)	
Outer shroud i.d., m (ft)	5.03 (16.5)	
Bundle height, m (ft)	15.54 (51)	
Number of tubes	52,930	
Pitch-to-diameter ratio	1.25	
Tube diameter/wall thickness, mm (in.)	16.51/1.07 (0.65/0.042)	
Tube material	316 SS	
Low-pressure helium inventory, kg (lbm)	567 (1250)	
High-pressure helium inventory, kg (lbm)	978 (2160)	
Low-pressure helium pressure drop, kPa (psi)		
Inlet	2.28	(0.33)
Bundle	22.75	(3.30)
Exit	<u>1.65</u>	<u>(0.24)</u>
Total	26.68	(3.87) [0.8%]
High-pressure helium pressure drop, kPa (psi)		
Inlet	22.47	(3.26)
Bundle	23.78	(3.45)
Exit	<u>38.05</u>	<u>(5.52)</u>
Total	84.30	(12.23) [1%]
Low-pressure helium		
Flow rate, kg/s (lbm/hr)	558	(4.43 x 10 ⁶)
Inlet temperature, °C (°F)	527.8	(982)
Exit temperature, °C (°F)	225.0	(437)
Pressure, MPa (psia)	3.34	(484)
High-pressure helium		
Flow rate, kg/s (lbm/hr)	540	(4.29 x 10 ⁶)
Inlet temperature, °C (°F)	182.2	(360)
Exit temperature, °C (°F)	494.4	(922)
Pressure, MPa (psia)	8.34	(1210)

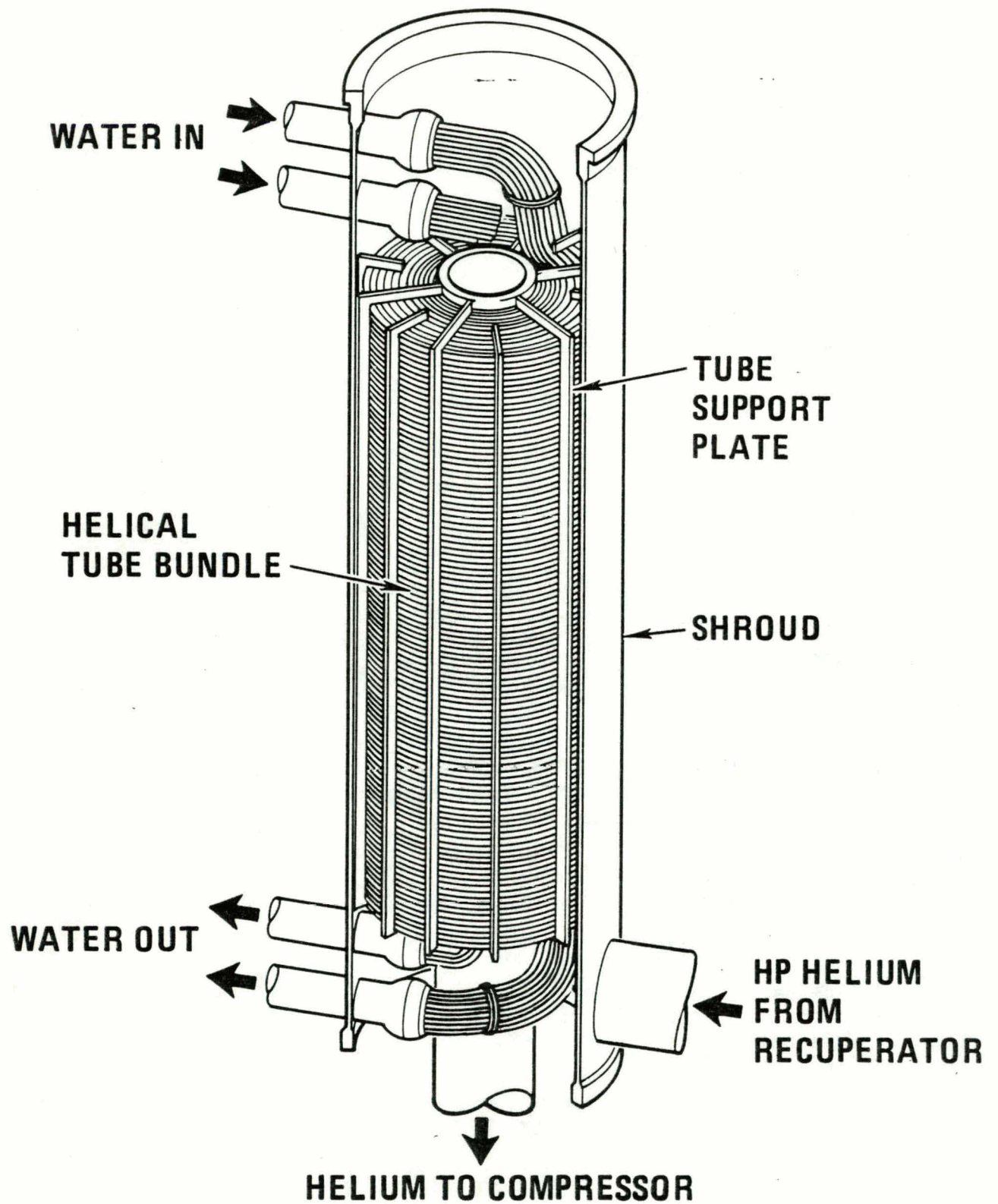


Figure 5.1.1-7 Precooler mechanical arrangement

operation even after a failure in one of the circuits; this circuit prevents compressor overheating under backflow conditions. Table 5.1.1-4 gives pertinent geometry and parameter data on the precooler.

The turbine control valve system has four valves arranged in a split-flow, bypass configuration in each of the power conversion loops (Fig. 5.1.1-8). Trim, safety, and primary bypass valves control core-turbine bypass flow between the core inlet and the turbine outlet in each loop; the attemperation valve controls flow between the compressor exit and the turbine exit. The trim valve makes fine adjustments of turbine speed and load and is of particular use when synchronizing with the grid. The primary bypass valve operates in two modes: (1) it can be modulated by the plant control system for plant load control, or (2) it can be operated as a safety bypass valve in an open/close mode as part of the safety bypass valve system that is included in the plant protection system. The safety valve is used primarily for turbine overspeed/overpressure protection; it is actuated by the plant protection system and operates only in an open/close mode. The attemperation valve mixes cold compressor discharge helium with warm exhaust helium, thereby minimizing thermal shock to the power conversion loop components, specifically to the recuperator, during transients. A commercial off-the-shelf valve, such as that shown in Fig. 5.1.1-9, has been considered for these control operations.

Use of the power conversion loops is the preferred means for effecting shutdown cooling and afterheat removal of the reactor core (hereafter referred to simply as shutdown cooling). The primary helium inventory control system, which is used for normal plant startup, can also be used for these shutdown purposes. Programmed reduction of the primary helium inventory prolongs coastdown of the turbomachines until pressures and temperatures in the primary coolant system are low enough to permit motorized operation of the turbomachines to maintain continued forced circulation of helium through the core. Off-site power is required to motorize the generators and to maintain circulation of water through the precoolers and cooling towers. This means of shutdown cooling is not available following a sudden loss of plant load or other events involving an immediate trip of the turbomachine.

5.1.1.5 Core Auxiliary Cooling System

The function of the CACS, an engineered safety feature, is to provide an independent means of cooling the reactor core if the primary cooling system should become inoperable. It consists of three parallel independent cooling loops, each comprising a heat exchanger, a circulator with its drive motor, a loop shutoff valve, and the motor controls. The components of a single loop, with the exception of the motor and controls, are located in a PCRV cavity peripheral to the central core cavity. Table 5.1.1-5 summarizes the design details of the CACS.

TABLE 5.1.1-4
PRECOOLER GEOMETRY AND PARAMETERS
HTGR-GT 2000-MW(t) TWO-LOOP PLANT

Inner shroud o.d., m (ft)	1.98 (6.5)
Outer shroud i.d., m (ft)	5.29 (17.35)
Bundle height, m (ft)	12.29 (40.33)
Active tube length, m (ft)	124.05 (407)
Number of tubes/tube cylinders/tube rows	788/31/277
Transverse/longitudinal pitch, m (in.)	0.05/0.04 (2.1/1.75)
Fin tip diameter/root diameter/wall thickness, mm (in.)	31.75/28.58/2.29 (1.25/1.125/0.09)
Fin thickness/fin height, mm (in.)	0.38/1.59 (0.015/0.0625)
Number of fins per mm (in.)	0.75 (19)
Tube material	1/2 Cr - 1/2 Mo
Water inventory, kg (lbm)	43,545 (96,000)
Helium inventory, kg (lbm)	1613 (3555)
Water pressure drop, kPa (psi)	688.5 (100)
Helium pressure drop, kPa (psi)	
Inlet	9.72 (1.41)
Bundle	25.79 (3.74)
Exit	3.10 (0.45)
Total	38.61 (5.60) [1.2%]
Helium	
Flow rate, kg/s (lbm/hr)	559 (4.44 x 10 ⁶)
Inlet temperature, °C (°F)	224.4 (436)
Exit temperature, °C (°F)	26.1 (79)
Pressure, MPa (psia)	3.25 (472)
Water	
Flow rate, kg/s (lbm/hr)	1230 (9.765 x 10 ⁶)
Inlet temperature, °C (°F)	20.6 (69)
Exit temperature, °C (°F)	132.2 (270)
Pressure, MPa (psia)	2.07 (300)
Exit subcooling, °C (°F)	71.1 (160)

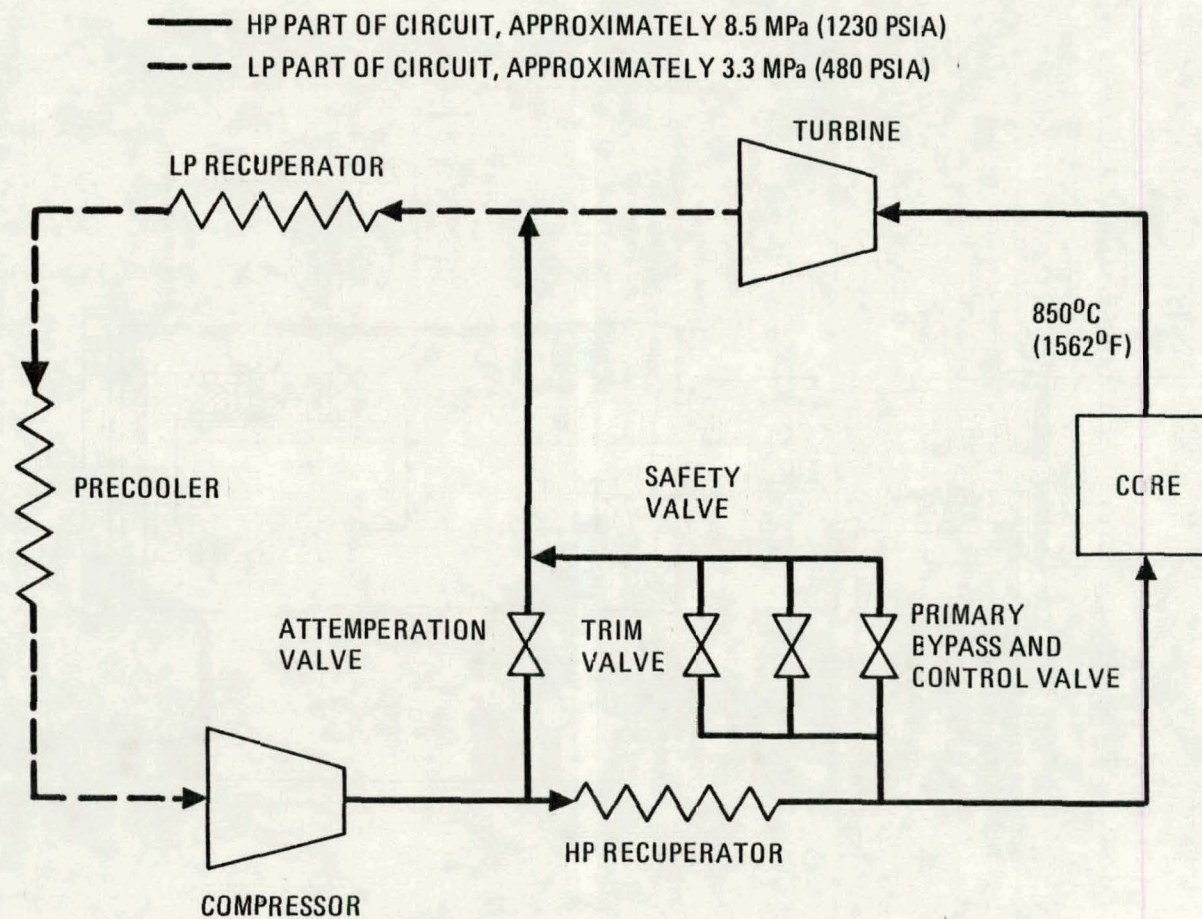


Figure 5.1.1-8 Simplified turbine control valve diagram for HTGR-GT power plant

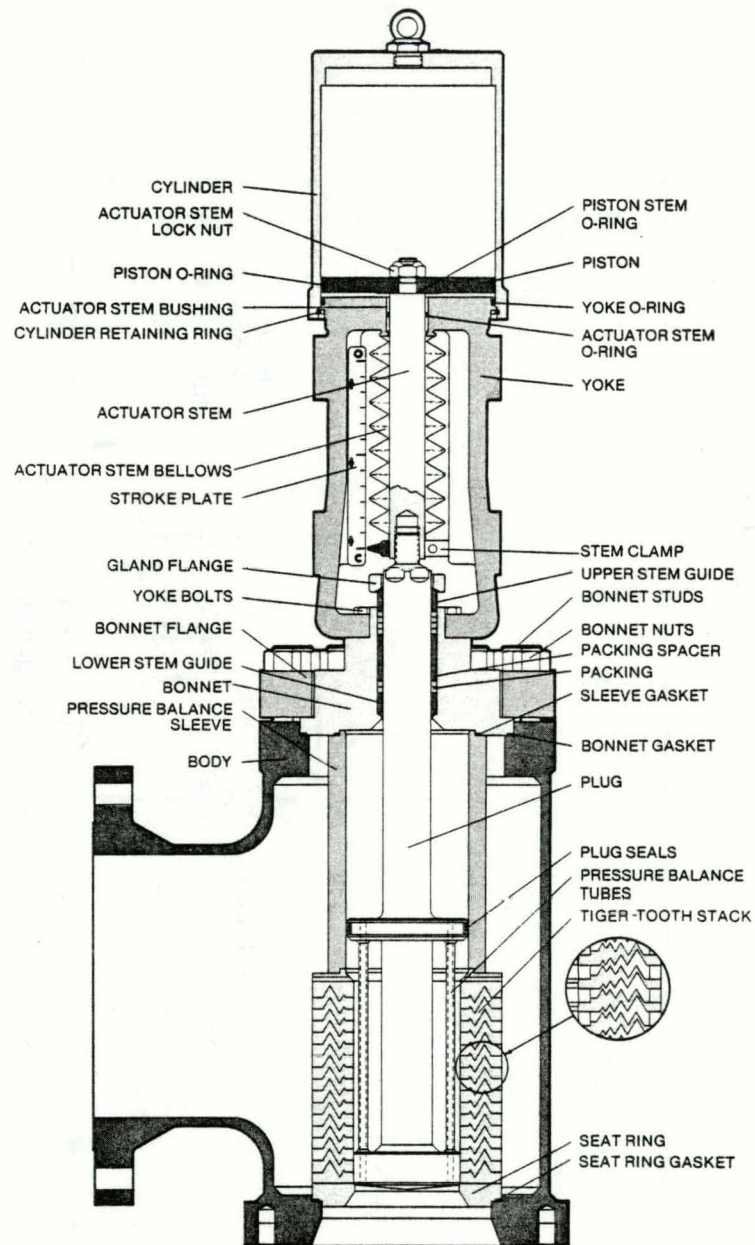


Figure 5.1.1-9 Typical bypass control valve for HTGR-GT plant

TABLE 5.1.1-5
 CACS DESIGN DATA FOR THE DIRECT CYCLE HTGR-GT PLANT

Plant size, MW(t)	2000
Number of CACS loops	3
Number of loops required	
PCRV pressurized	2
PCRV depressurized	2
Heat duty per loop, MW(t)	
PCRV pressurized	48.4
PCRV depressurized	17.1
CAHE data (per loop)	
Tube surface area, m ² (ft ²)	259 (2792)
Tube bundle length, m (ft)	2.84 (9.3)
Tube bundle diameter, m (ft)	1.53 (5.0)
Auxiliary circulator data	
Type of compressor	Axial
Pressure ratio	1.048
Design speed, rpm	3600
Design maximum power, kW (hp)	410 (548)

The CACS has the capability of maintaining the temperatures of all PCRV components within safe limits with the system either pressurized or depressurized. Forced circulation of the primary coolant transfers the reactor core heat through the CAHE to the core auxiliary cooling water system (CACWS). Air-blast heat exchangers cool this water and ultimately reject the heat to the atmosphere.

Each CACS loop is capable of removing 50% of the residual and decay heat from the core following a reactor trip from 102% normal rated power when the PCRV is pressurized. Heat removal of the depressurized PCRV requires use of two of the three loops.

The CAHE is a straight-bayonet-tube heat exchanger, as shown in Fig. 5.1.1-10. Hot gas from the lower cross duct enters the top of the tube bundle and flows downward, parallel to the tubes. At the bottom of the bundle the gas turns 90° and exits radially into the return duct to the auxiliary circulator through the annulus between the CAHE shroud and liner thermal barrier. The tube assemblies consist of two concentric tubes, sealed at the top end. Water enters the CAHE through a penetration in the bottom of the PCRV and flows upward counterflow to the helium through the annular space between the tubes, gaining heat through the outer tube wall. It then reverses its direction and flows downward in the central tube, leaving the CAHE through the same penetration in the bottom head of the PCRV.

The auxiliary circulator equipment includes the compressor, the drive motor, the motor controls, and the shutoff valve (Fig. 5.1.1-11). The variable-speed induction motors have static inverters for speed control through frequency variation. The shutoff valve is of the butterfly type and is self-closing to restrict reverse flow through the system when the main loops are operating. It is pressure-balanced to open automatically when the auxiliary circulators are operating.

The CACWS and the air-blast heat exchangers are external to the PCRV and are within the BOP scope of supply.

5.1.1.6 Other Systems

Several other service and control systems, in addition to those described above, support the reactor turbine system.

The neutron and flow control system regulates reactor power to meet the demands of the plant control system, the plant protection system, or the plant operator. It also regulates the helium flow distribution through various regions of the core by adjusting the region inlet orifices.

The rotating machinery service system is comprised of a turbomachinery turning gear system, a turbomachinery lubricating and buffer system, a main shaft penetration seal oil system, a generator and PCRV

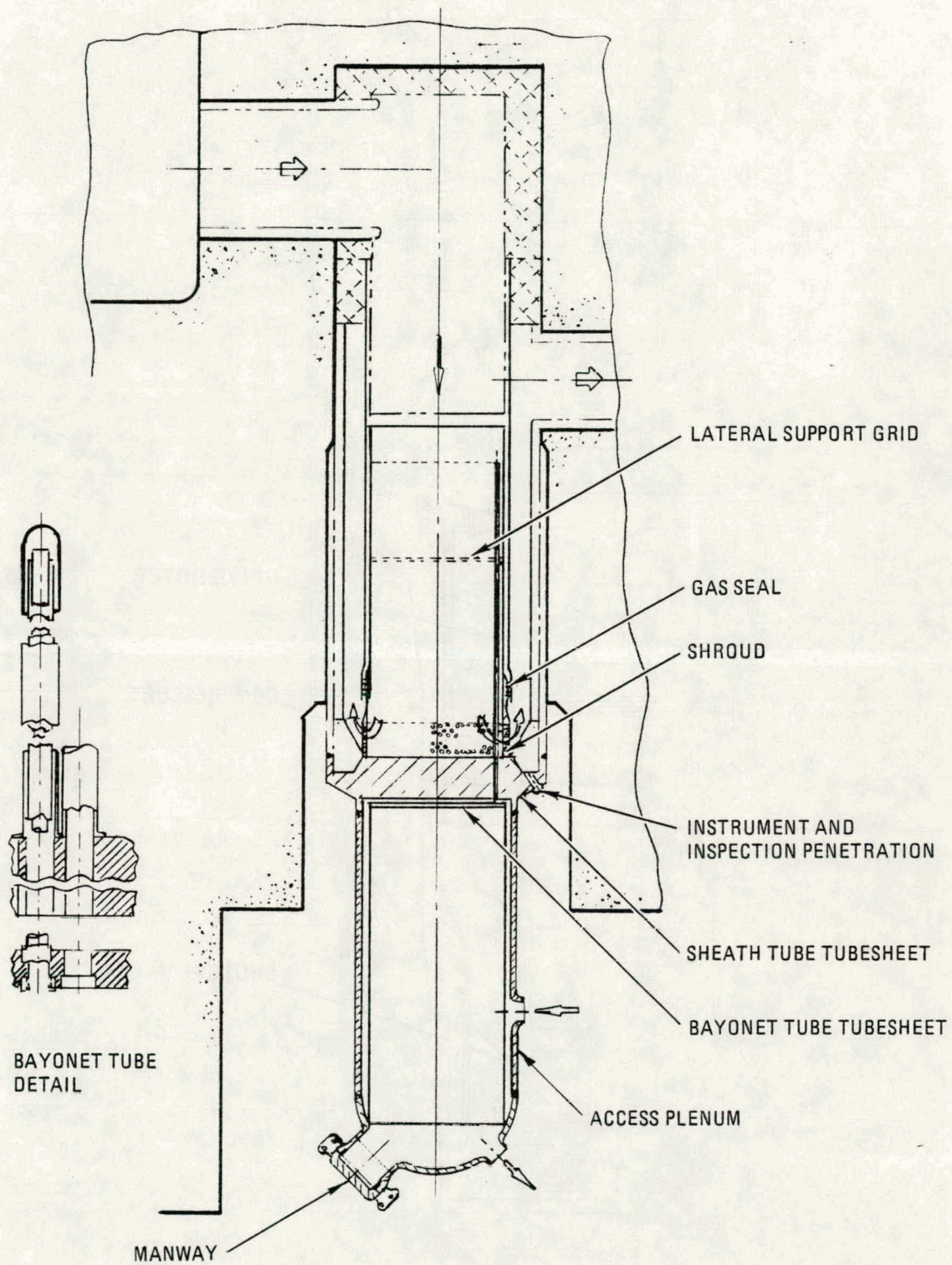


Figure 5.1.1-10 Direct-cycle 2000-MW(t) HTGR-GT bayonet tube CAHE

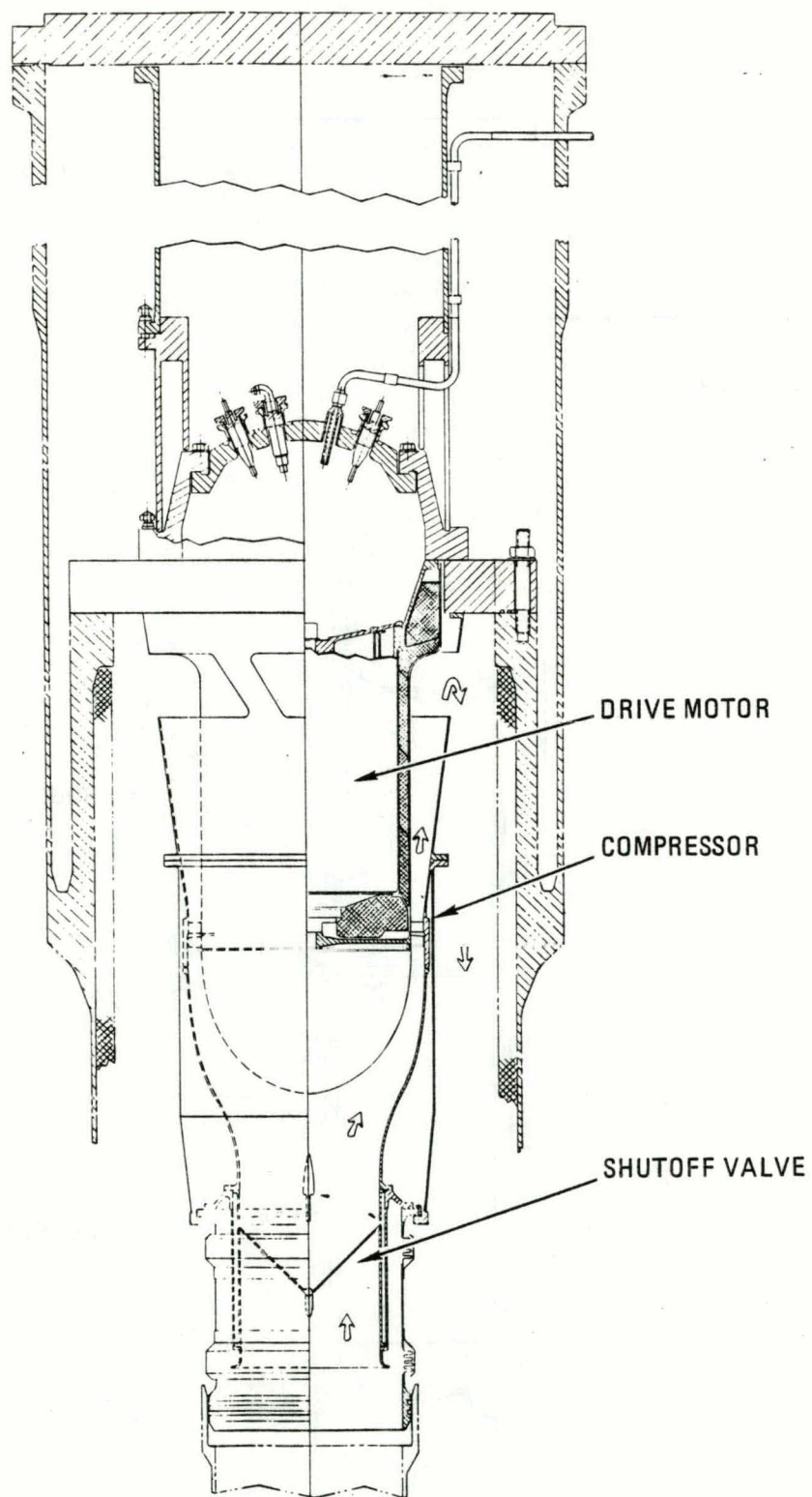


Figure 5.1.1-11 Cross section through auxiliary circulator

penetration bearing lubrication system, and an auxiliary circulator service system. These systems provide oil lubrication to bearings and seals, buffer helium to minimize oil ingress to the primary coolant system (or vice versa), and purified cooling water to the generator (i.e., rotor, stator, and air-gap cooling passages). These systems also supply mechanical support equipment where needed.

The helium service system removes helium from the primary system and processes it to remove particulates, chemical impurities, and radioactive contaminants. The clean helium is then used as a purge and seal gas throughout the plant; in the latter case, the helium pressurizes the seal interspaces of selected penetration closures to prevent leakage of primary coolant and to permit continuous monitoring of the integrity of these seals.

The plant protection system initiates actions to prevent unacceptable releases of radioactivity that could constitute a hazard to the health and safety of the public. It monitors several plant operating parameters and takes appropriate action when any are outside of specified safe limits. The system initiates actions to protect fission product release barriers and to limit the release of radioactivity if any barrier failures occur.

The plant control system provides safe plant operation and high plant availability. It regulates reactor power and controls the pressure and temperature of the primary coolant helium produced by the reactor turbine system based on appropriate interfaces with the balance of the plant.

The service systems that are not directly involved with reactor turbine system power production are the fuel handling system, the fuel shipping system, the reactor service system, and the analytical instrumentation system.

Appendix C provides more detailed plan and elevation drawings for the reactor turbine system structures and further assembly drawings of components.

5.1.2 Balance of Plant

This section presents the plant description for the 2000 MW(t) HTGR-GT. A general description of the site arrangement is presented below. Summary functional descriptions of each building are presented along with a summary of approximate building dimensions in Table 5.1.2-1. The general arrangement drawings of major site structures, the heat balance, and the electrical single line diagrams are provided in Appendix C.

5.1.2.1 Site Arrangements

The plot plan for the 2000 MW(t) HTGR-GT is shown on drawing SK-107. The illustrated arrangement is predicated upon a prestressed concrete reactor containment building which houses a prestressed concrete reactor vessel (PCRV) with turbomachinery loops canted at an angle of 18°. The electrical generators are located in the electrical generator building external to the reactor containment building and are connected to the turbomachines by shafts which penetrate the containment boundary. Locating the generators inside the reactor containment building was examined and found to be unattractive due to the increased size and cost of the reactor containment building.

The containment building, containment annulus and penetration buildings, auxiliary reactor service building, and electrical generator building are located on a common mat. This arrangement permits effective utilization of the oversized containment mat required to meet soil bearing requirements and achieves a high degree of coupling between major structures so as to minimize piping and cable runs and the travel distance for refueling equipment. Minor increases in size or changes in configuration of the major structures can be accommodated without invalidating the basic arrangement. The arrangement also incorporates the in-core refueling concept, which eliminates many of the height and arrangement constraints on the reactor containment building and the auxiliary reactor service building imposed by the rapid refueling system utilized in previous designs.

Three other major structures are located near the reactor containment building but on separate mats. The control auxiliary and diesel generator building abuts directly to the containment complex to minimize cable runs. The fuel storage building is sized for 1.3 core storage and connects to the reactor containment building by an underground tunnel as required by the in-core refueling concept. The turbomachine maintenance facility is located adjacent to the auxiliary reactor service building and is sized to provide for repair/refurbishment of the turbomachinery based upon decontamination to support "hands-on" operations.

A dimensional summary of each building and summary descriptions of the major structures are provided in Table 5.1.2-1.

TABLE 5.1.2-1

2000 MW(t)/800 MW(e) HTGR-GT DIRECT CYCLE PLANT

STRUCTURE/SYSTEM FUNCTIONAL DESCRIPTION

	<u>STRUCTURE</u>	<u>SEISMIC CATEGORY</u>	<u>NUMBER OF STORIES</u>	<u>HEIGHT FEET</u>	<u>WIDTH FEET</u>	<u>LENGTH FEET</u>	<u>VOLUME 10³FT³</u>
212	Reactor Contain- ment Building	Cat. I	NA	269'	150'I.D. 159'O.D.	NA	4312
213	Turbomachine Maintenance Building	Non-Cat. I	1	59'	108'	132'	841
214	Security Building	Non-Cat. I	1	14'	69'	72'	70
215	Auxiliary Reactor Service Building	Cat. I	6	137'	72'	96'	962
217	Fuel Storage Building	Cat. I	1	78'	100'	110'	858
218A	Control Aux. & Diesel Gen. Building	Cat. I	6	126'	93'	128'	1500
218B	Administration & Services Building	Non-Cat. I	2	26'	160'	240'	1000
218C	Labs & Shops	Non-Cat. I	1	26'	100'	140'	364
218D	Fire Pump House	Non-Cat. I	1	26'	30'	75'	27
218E	L.P. Helium Storage Area	Non-Cat. I	NA	NA	180'	233'	NA
218F	Electrical Gen- erator Building	Non-Cat. I	2	90'	123'	210'	2325
218H	Diesel Cooling & Fuel Oil Storage	Cat. I	2	73'	54'	94'	371
218I	Warehouse	Cat. I	1	20'	50'	70'	70

TABLE 5.1.2-1
(Continued)

2000 MW(t)/800 MW(e) HTGR-GT DIRECT CYCLE PLANT

STRUCTURE/SYSTEM FUNCTIONAL DESCRIPTION

STRUCTURE	SEISMIC CATEGORY	NUMBER OF STORIES	HEIGHT FEET	WIDTH FEET	LENGTH FEET	VOLUME 10 ³ FT ³
218J Containment Annulus Building	Cat. I	2	52'	240'O.D. 159'I.D.	NA	440
218K Containment Penetration Building	Cat. I	5	109'	37'	93'	429
218S Holding Pond & Control House	Non-Cat. I	NA 1	8' 10'	80' 8'	80' 10'	51 1
218T Ultimate Heat Sink Train A & B Train C	Cat. I	2 1	72' 34'	89' 62'	146' 73'	936 154
218U Control Room Emergency Intake Structures (2)	Cat. I	1	10'	10'	10'	1

5.1.2.2 Structures and Improvements

- Reactor Containment Building - The reactor containment building houses the PCRV and other nuclear steam supply system (NSSS) components and is designed to protect them against normal, abnormal, and environmental conditions and against tornado-borne missiles. The reactor containment building is also designed to limit fission product releases during normal conditions and during accident conditions, which include the Design Basis Depressurization Accident and the postulated Maximum Hypothetical Fission Product Release. The reactor containment building is a Seismic-Category-I, prestressed-concrete structure composed of a foundation mat, cylindrical shell, and hemispherical dome with a design pressure of approximately 70 psig. The reactor containment building is completely lined on the interior with steel plate to provide a pressure-tight boundary, and the liner on the bottom is protected by a concrete slab which serves as the reactor containment building floor. There are three hatches: a 28' I.D. equipment hatch at the refueling floor level, a 27' I.D. turbomachine removal hatch to accommodate grade-level removal, and an 8' I.D. personnel hatch.

The PCRV is supported on a 12'6"-high, reinforced-concrete ring wall and pedestal bearing on the reactor containment building mat. The PCRV is offset from the reactor containment building centerline by 5.5'; the annular space between the PCRV and reactor containment building shell is 27' at the turbomachine removal hatch and 16' on the opposite side of the PCRV. Within the annular space between the PCRV and the reactor containment building walls is a steel structure extending up to the refueling floor for support of major equipment, piping, electrical trays, HVAC equipment, access platforms, and stairs.

Above the refueling floor near the reactor containment building springline is located a polar crane for handling of refueling equipment. A temporary fuel storage facility is located in the containment mat consistent with the in-core refueling system requirements.

- Turbomachine Maintenance Facility (Drawing SK-97) - The turbomachine maintenance facility design is based upon the premise that the turbomachine can be transferred into the facility without significant shielding. Once in the facility, surface contamination is removed in the decontamination pit, thereby reducing the activity level to a point where "hands-on" maintenance can be performed without undue exposure to personnel. The facility also includes areas dedicated to spare parts storage, tool and fixture storage, plug removal device parking, and rotor balancing. An observation room is provided from which transfer and crane operations are controlled.

- Security Building - The security building is a masonry building which provides a controlled means of access into and out of the plant area.
- Auxiliary Reactor Service Building - The auxiliary reactor service building houses the facilities, systems, and components necessary for fuel handling, control rod drive storage, equipment decontamination and inspection, radioactive waste management, helium purification, and other auxiliary equipment associated with operation and maintenance of the reactor. The fuel handling machinery, including tracks and support structure for the fuel transport system, is located on the top floor.

The auxiliary reactor service building is a Category-I, five-story, reinforced-concrete structure that is located adjacent to the reactor containment building. All walls are concrete, and the upper floors are concrete slabs supported on structural steel framing. The roof of the auxiliary reactor service building, above the refueling floor level, is enclosed by a non-Category-I, steel-framed structure having insulated metal siding, and a roof composed of a metal deck, insulation, and built-up roofing. The enclosure supports an overhead traveling crane for use in the refueling operation. The grade floor of the auxiliary reactor service building acts as a transition area for removal of a turbomachine from the containment to the maintenance facility.

- Fuel Storage Building - The fuel storage building houses all equipment related to new and spent fuel shipping, receiving, and storage and is sized to provide storage for 1.3 cores of spent fuel. The facility is capable of handling either truck or rail shipping of spent fuel and can handle either type of shipping cask.

The fuel storage building is a Category-I, concrete structure that is situated near the auxiliary reactor service building. Fuel is moved from the temporary fuel storage facility in the reactor containment building to the fuel storage building by way of a tunnel. The fuel is sealed in storage containers in a facility located in the fuel storage building and stored in walls in the fuel storage pool, also located in this building. A rail/truck shipping bay is located at one end of the building to accommodate long rail cars.

- Control Auxiliary and Diesel Generator Building - The control auxiliary and diesel generator building houses the control, monitoring, and indicating equipment required for plant operation during normal, abnormal, and accident conditions. This includes the main control room, cable spreading areas, switchgear area, and

diesel generator. The control auxiliary and diesel generator building is a Category-I, six-story, concrete structure with a mat foundation, concrete exterior and interior walls, and intermediate floor slabs and roof slab supported on structural steel framing.

- Administration and Services Building - The administration and services building is a framed-structural-steel building which houses general offices. Also included are a health physics complex and a checkpoint to control entry to nuclear island structures.
- Labs and Shops - The labs and shops building is a framed-structural-steel building. Both clean and controlled areas are provided.
- Fire Pump House - The fire pump house is a reinforced-concrete building which houses pumps and associated equipment and controls for the plant protection system.
- L. P. Helium Storage Area - The L. P. helium storage area is a covered tank farm which provides makeup and storage capacity for plant helium inventory.
- Electrical Generator Building - The electrical generator building is a two-story, metal structure supported on an extension of the containment mat in the area of the generator pedestals. The adjacent portion of the structure is supported on a separate mat with a truck/rail way separating the two sections. An overhead traveling crane is also provided.
- Diesel Cooling and Fuel Oil Storage Building - The diesel cooling and fuel oil storage building is a reinforced-concrete building which houses the dry cooling towers and seven-day storage fuel oil tanks for the control auxiliary and diesel generator building diesels.
- Warehouse - The warehouse is a framed-structural-steel building which houses a temporary storage and search area for incoming materials.
- Containment Annulus Building - The containment annulus building houses both loops of the reactor cooling water system, two of the auxiliary cooling water system loops, two of the auxiliary circulator motor cooling water system loops, and associated safety-related HVAC equipment, electrical, and piping penetration areas. The non-safety-related HVAC equipment for the containment annulus building is located on the roof.

The containment annulus building is a Category-I, annular, two-story, reinforced-concrete structure that surrounds the reactor containment building in the area between the electrical generator building and the auxiliary reactor service building and between the auxiliary reactor service building and the containment penetration building. The containment annulus building utilizes the projected portion of the reactor containment building base mat as a common foundation. The annular space in both stories is partitioned by various radially oriented concrete walls to provide functional separation of the areas. A non-Category-I, steel-framed structure encloses the entire roof area of the containment annulus building. The enclosure has insulated metal siding and a roof composed of a metal deck, insulation, and built-up roofing.

- Containment Penetration Building - The containment penetration building houses the containment electrical penetration areas, instrument areas, the remote safe-shutdown room, the data acquisition system room, including switchgear and ventilation areas, portions of the helium purification system, and the third train of the core auxiliary cooling water system and auxiliary circulator motor cooling water system. The containment penetration building is a Category-I, five-story, reinforced-concrete structure that links the reactor containment building and the control auxiliary and diesel generator building and utilizes the projected portion of the reactor containment building base mat as a common foundation.
- Holding Pond and Control House - The holding pond is an open-top, reinforced-concrete basin, and the control house is a steel-framed building with insulated metal siding and roof and a floor on grade. These structures permit collection and treatment of non-radioactive, contaminated effluent prior to discharge.
- Ultimate Heat Sink Structures - The train A and train B ultimate heat sink structure provides cooling for two of the three trains of the core auxiliary cooling water system and for both trains of the nuclear service water system. The structure houses the dry towers for core auxiliary cooling water system cooling and the wet towers, including pumps, associated electrical equipment, and water basins, for nuclear service water cooling. The entire below-grade portion of the structure is a basin that serves both trains and provides 30 days of storage. The roof of the basin serves as the floor of the superstructure and supports all equipment.

The superstructure is a reinforced-concrete enclosure around each train's wet and dry tower.

The train C ultimate heat sink structure provides cooling for the third train of the core auxiliary cooling water system and houses a dry tower and associated equipment. This is a single-story, reinforced-concrete structure supported on a foundation mat which serves as the first floor and which is located at grade level.

Underground, Category-I, rectangular, reinforced-concrete piping and electrical tunnels connect the ultimate heat sink structures with the core auxiliary cooling water system/reactor plant cooling water system areas in the containment annulus building and the containment penetration building.

- Control Room Air Intake Structures - The control room air intake structure is a reinforced-concrete intake that provides clean air from a remote, uncontaminated source to the control room in the control auxiliary and diesel generator building during an emergency. Two intakes are provided, located about 180° from each other on opposite sides of the control auxiliary and diesel generator building.

5.1.2.3 Balance of Reactor Plant

The design of the reactor plant is based on the GA two-loop, 2000 MW(t) HTGR Gas Turbine Plant design. The major systems which comprise the balance of reactor plant are discussed below.

- Safeguards Cooling System - The safeguards cooling system consists of the core auxiliary cooling system, which is part of the reactor coolant system, the core auxiliary cooling water system, and the auxiliary circulator motor cooling water system. The core auxiliary cooling system is a Safety-Class-2 system which provides an independent means of cooling the reactor core with the primary system pressurized or depressurized. It includes the auxiliary circulators and their drive motors, motor controls, diffusers and valves, the core auxiliary heat exchangers, and the control instrumentation and hardware.
 - Core Auxiliary Cooling Water System - This system is a Safety-Class-3 system which circulates cooling water through the core auxiliary heat exchangers to remove stored and decay heat from the primary coolant and to reject this heat to the atmosphere.

- Auxiliary Circulator Motor Cooling Water System - This system is a Safety-Class-3 system which provides cooling water to the auxiliary circulator motors during periods of full core auxiliary cooling system operation. The safeguards cooling system incorporates sufficient redundancy and capacity to ensure adequate core cooling when one of the cooling trains is lost under worst-case (depressurized) conditions.
- Radioactive Waste Process System - The radioactive waste process system consists of the liquid, gaseous, and solid waste management systems. None of the three systems are safety-related. The liquid waste management system includes tanks for collection of liquid effluent and utilizes filtration, demineralization, evaporation, and reverse osmosis singularly or in combination for processing. The gaseous waste management system has the capability to selectively release or retain gaseous effluent for a suitable period prior to a controlled release. The solid waste management system is capable of low-level compacted waste drum storage and complete remote handling of high-level solidified waste.
- Fuel Handling and Storage Facility - This structure utilizes the in-vessel refueling equipment supplied by GA. Seismic-Category-I, long-term and temporary storage facilities provide water-cooled storage. The long-term storage facility, located in the fuel storage building, provides storage for 1.3 cores. The temporary facility, located in the reactor containment building mat, provides storage for one refueling. Fuel pool cooling is provided by two Safety-Class-3 cooling trains, each equipped with one 100% pump and one 100% heat exchanger.
- Helium Storage System - The helium storage system, which is not safety-related, provides storage capacity for the entire primary coolant inventory plus two months' makeup requirements and provides PCRV depressurization and pressurization capabilities.
- Helium Purification System - The helium purification system is a Safety-Class-3 system which is designed to purify helium from the primary coolant system, remove fission products and chemical impurities, and return the purified helium to the primary loop. The liquid nitrogen system, which is not safety-related, supplies refrigeration to the helium purification system low-temperature absorbers.
- Nuclear Service Water System - The nuclear service water system is a Safety-Class-3 system which provides cooling for the reactor plant cooling water system, the fuel handling and storage cooling water system, and the reactor plant auxiliaries.

- Reactor Plant Cooling Water System - This system has an essential subsystem consisting of two 100% redundant trains which provide cooling water to the PCRV cooling coils, the moisture monitoring equipment, and the auxiliary circulator motor cooling water system. The reactor plant cooling water system also has a single-train, non-essential subsystem which is not safety-related and which provides cooling water to non-safety-related equipment and to a separate non-essential cooling coil in each auxiliary circulator motor to remove parasitic heat losses when the circulator is not operating.
- Reactor Plant Instrumentation and Control System - This system is designed to ensure that the unit can be safely and efficiently operated and that in the event of an abnormal or accident condition, it can be shut down and maintained in a safe-shutdown condition. The system consists of automatic and manually initiated protection systems for safety under accident conditions, safety-related display systems required during normal, upset, emergency, and faulted conditions, a computer-based data acquisition and display system, and regulating systems used for normal operation of the unit.

The instruments and controls are located in the main control room, which provides remote operation of the unit. In the event that access to the main control room is lost, equipment is provided outside the main control room to shut the reactor down and maintain it in a safe-shutdown condition.

5.1.2.4 Electric Plant Equipment

The electric plant equipment transfers the power generated in the plant to the high voltage switchyard through the generator stepup transformers, controls and meters the electric energy, and protects the power-carrying components. It is the source of power for the plant auxiliaries, the plant control, protection, and surveillance systems, and the engineered features equipment during normal operation and abnormal accident conditions and during plant shutdown and refueling.

The electric plant design reflects all the applicable regulatory and technical requirements. Physical and electrical separation of equipment and systems is provided to assure the availability of the minimum required safety features equipment to mitigate the consequence of any Design Basis Event. Physical separation of equipment and circuits is achieved in such a way that the single-failure criterion is met.

- Main Generation - Two generator circuit breakers are provided to facilitate rapid disconnection of either generator from the offsite power system, allowing the auxiliary power system to be fed through three single-phase generator stepup and two-unit auxiliary transformers. Two 5-MW static freeway converters are provided to motorize the generators during plant startup. There are two 5-KV, non-Class-1E, metal-clad switchgear buses and three 5-KV, Class-1E switchgear buses to provide the power sources for the plant auxiliary loads. All engineered safety features equipment is automatically sequenced into the Class-1E buses being fed from the diesel generators in the event of a loss of offsite power supplies.
- Electric Systems - Non-Class-1E and Class-1E, 460-volt motor control centers are provided for power distribution to motors up to 100 hp lighting loads and other miscellaneous loads such as motor-operated valves, resistance heaters, heat tracing, and space heaters. There are two unit auxiliary transformers feeding into the two 4.16 non-Class-1E buses and three 4.16-KV, Class-1E buses, and two reserve auxiliary transformers. Each transformer is sized to carry with margin the plant auxiliary loads under heavily loaded conditions. Transformer impedances are selected to limit the available short-circuit currents on the switchgear buses without adversely affecting the acceptable voltage regulators during extreme plant operating conditions. Appropriate protections are provided for the transformers. In addition, a balance-of-plant diesel generator (optional) is provided to supply the two 4.16-KV, non-Class-1E buses in the event that all offsite power is lost. Unit substations are provided to furnish power sources to the low-voltage (460V), Class-1E and non-Class-1E distribution system. Motors rated 101 hp through 200 hp are connected to the unit substations. Unit substation transformer impedances are based on matching the available fault-current-withstand-capability of the switchgear with appropriate voltage regulation consideration. The unit substations for the cooling towers are fed from a loop feeder. The d-c system comprises the plant non-Class-1E and Class-1E batteries and battery chargers. Each Class-1E d-c bus is supplied from a Class-1E battery and two Class-1E battery chargers. During normal operation, d-c power is supplied from the battery charger. During emergency operation, d-c power is supplied from the batteries in the absence of any d-c source to the battery chargers. During stepup and shutdown, d-c power is supplied from whichever source is available. Non-Class-1E, 125/250V d-c buses are fed from two non-Class-1E batteries and two non-Class-1E chargers.

- Emergency Systems - Three independent diesel generators are provided to furnish the onsite d-c power sources to the Class-1E, 4.16-KV buses. Diesel generators are properly sized such that any two units have the capability of operating all protection systems and the engineered safety features to mitigate the consequence of a Design Basis Depressurization Accident concurrent with a loss of offsite power. Class-1E and non-Class-1E, solid-state inverters are provided to serve as an uninterruptible power source for miscellaneous vital and non-vital d-c plant loads. Switchboards, protective equipment, appropriate electrical structures, and wires/cables for various plant and process equipment would be procured and installed as per established guidelines and procedures.

5.1.2.5 Miscellaneous Plant Equipment

The miscellaneous plant equipment provides miscellaneous water, compressed air, auxiliary steam, and general maintenance and service equipment for the overall plant. It includes crane systems, compressed air systems, service water system, fire protection system, potable water system, auxiliary steam system, communications system, fire detection system, security system, laboratory equipment, office furnishings, and environmental monitoring equipment.

5.1.2.6 Main Condenser Heat Rejection System

The waste heat rejection system for the HTGR-GT provides cooling for the main thermal cycle and all plant service water during normal plant operation. This system includes the main cooling tower, precooler cooling water system piping, pumps, and structures; the makeup and blowdown system piping, pumps, and structures; and associated instrumentation, controls, and chemical feed systems. Cooling is accomplished using a wet/dry cooling system with the dry tower sized to reject approximately 85% of the waste heat at the system design point. Cooling water passes through the dry tower, which is part of the closed-loop precooler cooling water system. The wet cooling is achieved through a shell and tube heat exchanger connected to a mechanical draft, wet tower. Meteorological conditions typical of Modesto, California, were assumed for purposes of sizing these towers.

The 100% mixed flow vertical pumps are assumed for the makeup system. The pumps are located in the intake structure adjacent to the river. Two traveling screens are assumed, each suitable for 100% of the flow requirements with an approach velocity of 1/2 foot per second. Servicing the traveling screens are two 100%-capacity screen wash pumps. The screens are protected by a bar rack and trash rake.

5.2 Lead Plant Performance

This section summarizes those design and operational aspects of the plant that are the major contributors to the plant performance evaluation. Information on a two-loop 2000-MW(t) HTGR-GT plant is presented as well as performance parameters for a recently optimized 2170-MW(t) plant, which will be the base for future work.

5.2.1 Design Performance Parameters

Two designs have been studied for the direct-cycle HTGR-GT plant. An earlier version with a 2000-MW(t) core and dry tower cooling provided the basis for most of the component designs that have been developed. A more recent design with a 2170-MW(t) core and wet/dry tower cooling will be emphasized in further on going studies. Tables 5.2.1-1 and 5.2.1-2 summarize the major parameters that characterize these two plants. Figures 5.2.1-1 and 5.2.1-2 are simplified cycle diagrams, with pertinent operating conditions and generator output for optimized (capital cost) parameters, for design reactor core power levels of 2000 and 2170 MW(t), respectively.

The principal differences between the two plants result from the plant with the 2170-MW(t) core having a redesigned precooler and a higher precooler water temperature combined with a slightly higher primary system operating pressure, for a nominal plant efficiency of 39.7%. The optimization studies showed the 2170-MW(t) design to have slightly lower efficiency with lower overall capital costs (\$/kW).

5.2.2 Control and Dynamics

Safe plant operation, high plant availability, plant equipment protection, and prevention of any unacceptable releases of radioactivity that would constitute a hazard to the health and safety of the public are provided by the plant control system (PCS) and the plant protection system (PPS).

The PCS is designed to automatically regulate reactor power, control electrical load and turbine speed, control temperature of the helium delivered to the turbines, control thermal transients experienced by the power conversion loops and reactor components, and provide the plant with the capability to perform routine startup/shutdown and automatic load-following cycles. In addition, the PCS provides detection and actions to protect components of the primary and secondary coolant systems from damage. It limits system-induced transients and can also assist the PPS in plant shutdown. Failure of the PCS will not jeopardize the PPS functions or public safety.

The PCS has the capability for continuous plant operation under full automatic control at any point between maximum and minimum load. (The minimum load under full automatic control will not be less than 25% of

TABLE 5.2.1-1
SUMMARY OF PRIMARY COOLANT SYSTEM
EXPECTED PERFORMANCE AT 2000 MW(t)

Number of power conversion loops	2
Reactor power, MW(t)	2000
Nominal electrical output (net), MW(e)	800
Helium flow rate (through compressors), kg/s (lb/hr)	563 (4,471,000)
Helium pressure at high-pressure compressor discharge, MPa (psia)	8.412 (1220)
Total primary circuit pressure loss, kPa (psi)	419.9 (60.9)
Recuperator high-pressure WP, kPa (psi)	64.1 (9.3)
Reactor core, core support block, orifice, and inlet and exit plenums WP, kPa (psi)	100.0 (14.5)
Turbine WP, MPa (psi)	4.766 (691)
Recuperator low-pressure WP, kPa (psi)	72.5 (10.5)
Precooler WP, kPa (psi)	11.0 (1.6)
Core inlet gas temperature, °C (°F)	494 (922)
Core outlet gas temperature, °C (°F)	850 (1562)
Turbine inlet gas temperature, °C (°F)	849 (1560)
Compressor inlet gas temperature, °C (°F)	27 (80.1)
Minimum cycle helium temperature, °C (°F)	26 (79.0)
Recuperator effectiveness	0.905
Turbine isentropic efficiency, %	91.8
Compressor isentropic efficiency, %	89.8
Disk cooling flow, %	2.96
Generator efficiency, %	98.8
Total heat losses from reactor turbine system heat transfer system, MW(t)	24.92
Standby parasitic CACS heat loss, MW(t)	3.4
Loss to liner cooling system, MW(t)	12.33
Miscellaneous heat losses, MW(t)	9.19
Power to helium compressors (total), MW(t)	929
Core power density, W/cm ³	6.6
Total compressor pressure ratio	2.6
Dry bulb/wet bulb ambient air temperature, °C (°F)	23/16.1 (73/61)
Precooler inlet water temperature, °C (°F)	24 (76)
Precooler outlet temperature, °C (°F)	142 (288)

TABLE 5.2.1-2
SUMMARY OF PRIMARY COOLANT SYSTEM
EXPECTED PERFORMANCE AT 2170 MW(t)

Number of power conversion loops	2
Reactor power , MW(t)	2170
Nominal electrical output (net, MW(e)	860.4
Helium flow rate (through compressors, per loop), kg/s (lb/hr)	625.8 (4,970,000)
Helium pressure at high-pressure compressor discharge, MPa (psia)	8.619 (1250)
Total primary circuit pressure loss, kPa (psi)	446.1 (64.7)
Recuperator high-pressure WP, kPa (psi)	88.1 (12.8)
Reactor core, core support block, orifice, and inlet and exit plenums WP, kPa (psi)	113.9 (16.5)
Turbine WP, MPa (psi)	4.859 (704.5)
Recuperator low-pressure WP, kPa (psi)	36.1 (5.2)
Precooler WP, kPa (psi)	52.2 (7.6)
Core inlet gas temperature, °C (°F)	498 (929)
Core outlet gas temperature, °C (°F)	850 (1562)
Turbine inlet gas temperature, °C (°F)	849 (1560)
Compressor inlet gas temperature, °C (°F)	27 (87.2)
Minimum cycle helium temperature, °C (°F)	30 (86)
Recuperator effectiveness	0.915
Turbine isentropic efficiency, %	91.8
Compressor isentropic efficiency, %	89.8
Disk cooling flow, %	2.96
Generator efficiency, %	98.8
Total heat losses from reactor turbine system heat transfer system, MW(t)	18.63
Standby parasitic CACS heat loss, MW(t)	3.5
Loss to liner cooling system, MW(t)	11.73
Miscellaneous heat losses, MW(t)	3.4
Power to helium compressors (total), MW(t)	1035
Core power density, W/cm ³	7.2
Total compressor pressure ratio	2.6
Dry bulb/wet bulb ambient air temperature, (a) °C (°F)	23/16 (73/61)
Precooler inlet water temperature, °C (°F)	24 (76)
Precooler outlet temperature, °C (°F)	142 (288)

(a) Heat dump to ambient air is 80% through dry tower and 20% through wet tower.

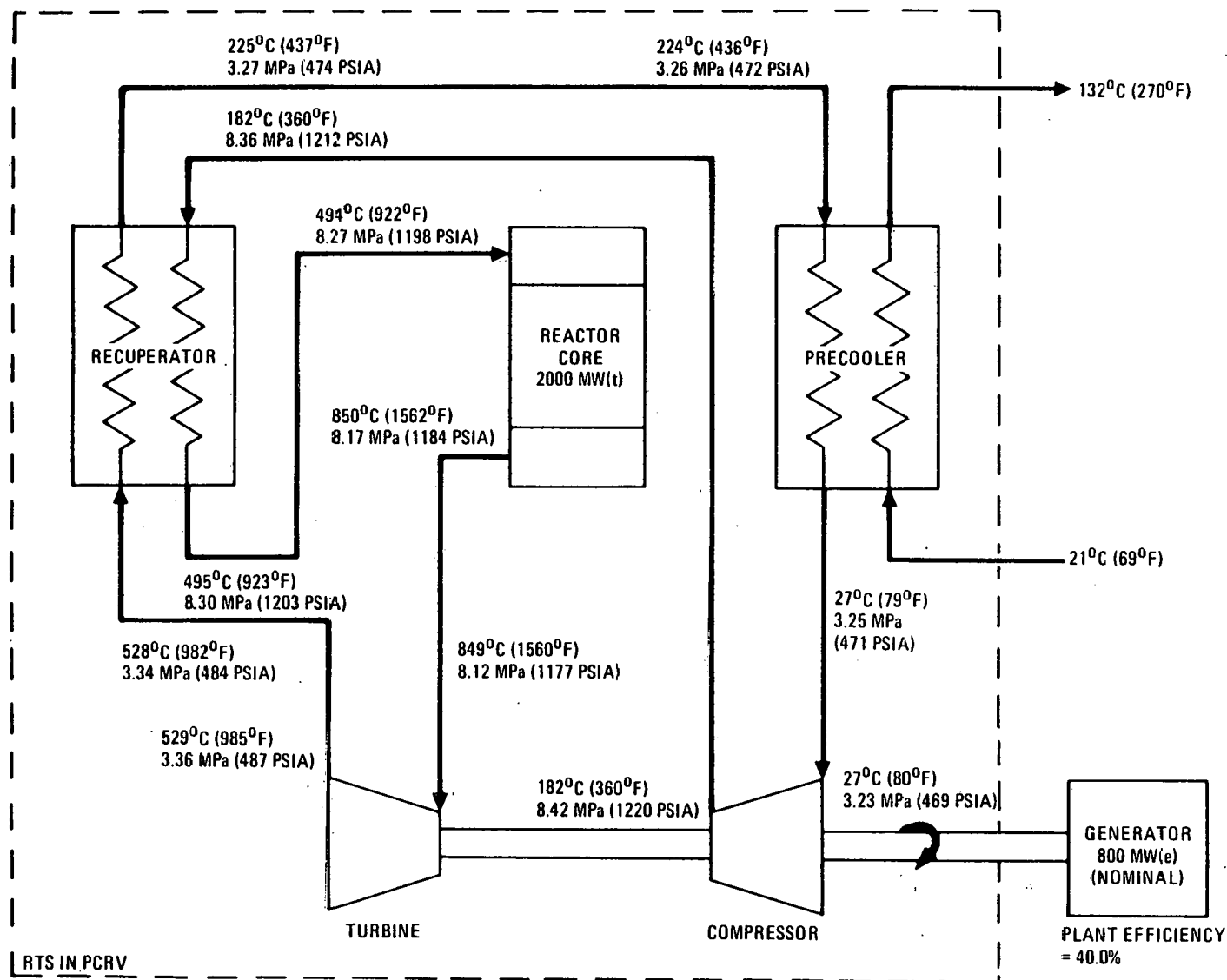


Figure 5.2.1-1 HTGR-GT cycle diagram, 2000-MW(t) rating

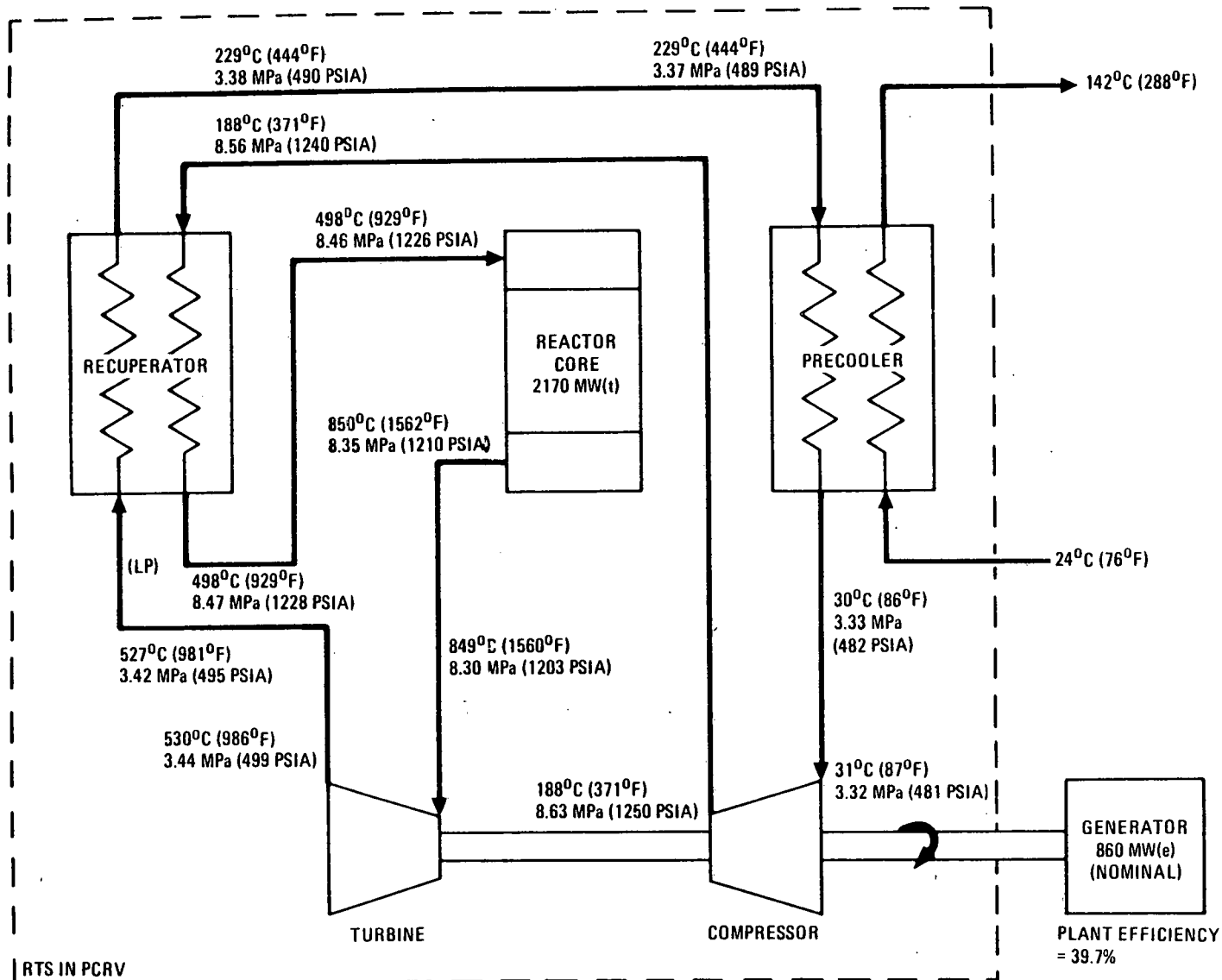
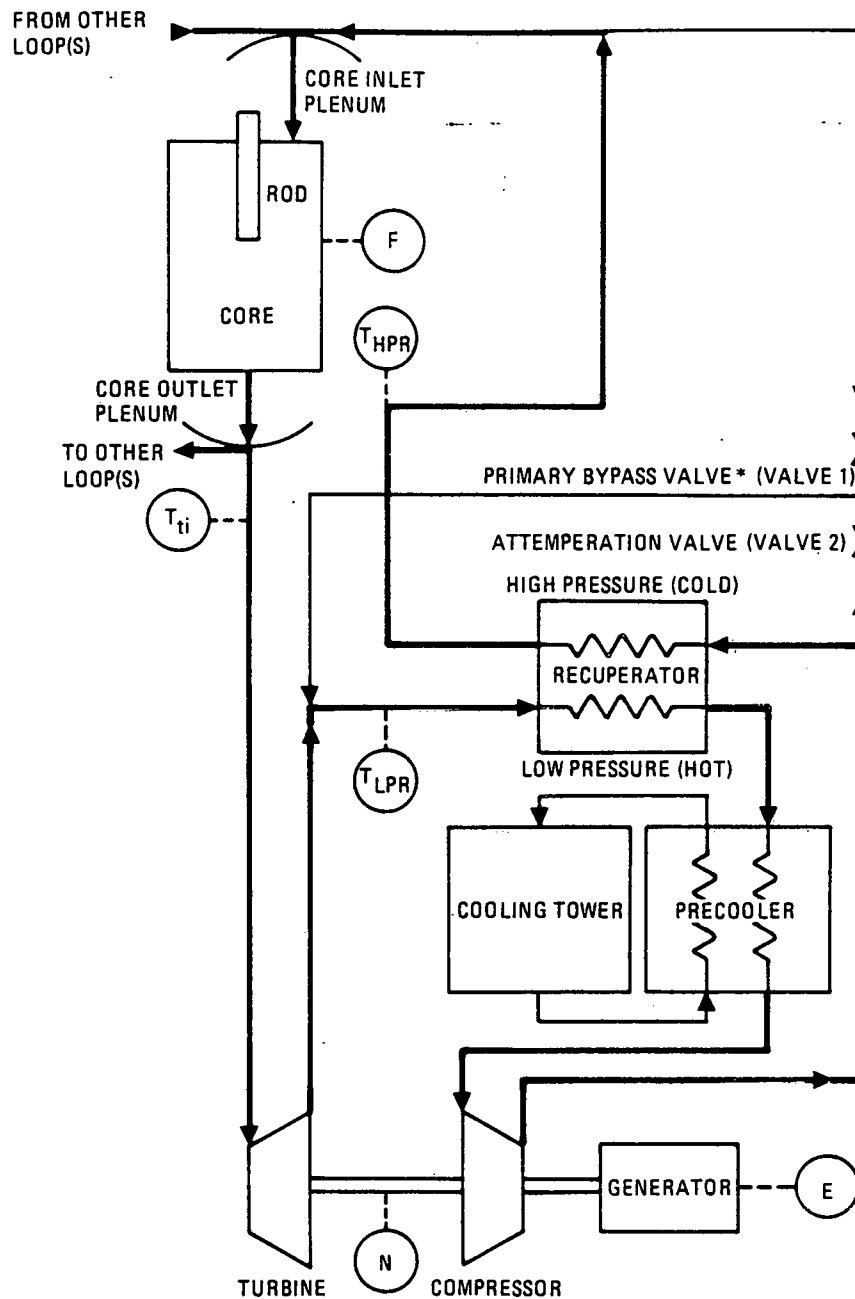


Figure 5.2.1-2 HTGR-GT cycle diagram, 2170-MW(t) rating

rated.) It also has automatic load-following capability (operator-actuated rod reshimming may be required) for rates of electrical load change up to a maximum of 5% of rated power per minute ramp change or 10% of rated power step change. The protective functions of the PCS consist of the main loop trip system, the reactor power setback function, the circulating water protection system, and the turbomachine overspeed protection system.

- Overall Plant Control System - The PCS manipulates six plant variables (as shown in the process flow diagram in Fig. 5.2.2-1) by five closed-loop controllers (Fig. 5.2.2-2). Tables 5.2.2-1 and 5.2.2-2 summarize the PCS control and protective functions.
 - Turbine Inlet Temperature Control - The average inlet temperature to the turbines of the operating power conversion loops is controlled by adjusting reactor power. This is accomplished by the turbine inlet temperature controller (Fig. 5.2.2-2), which provides a command signal to the reactor neutron flux controller. The reactor flux controller adjusts the position of the control rods to vary reactor power, which alters the amount of heat available to the helium. The neutron flux controller (Fig. 5.2.2-2) consists of an on-off controller with dead band and hysteresis. The temperature loop is composed of a proportional-plus-integral controller with limited output. The limits for this loop are chosen to prevent control-system-induced power excursions from causing any unintentional reactor trip.
 - Load/Speed Control - The load/speed controller (Fig. 5.2.2-2) is designed to maintain the power delivered to the generator through the turbomachine shaft equivalent to the demanded electrical power throughout the normal load range. To accomplish this function, the load/speed controller modulates turbine power using the primary bypass valve to maintain turbomachine speed and power at desired values. Actuation of the primary bypass valve causes partial diversion of helium flow from the core inlet plenum to the inlet of the low-pressure side of the recuperator, thus reducing turbine drive power by reduction of the turbine pressure ratio and turbine flow. The controller will use load and speed errors to generate the required valve command signals. The controller uses proportional-plus-integral action to regulate load and speed. The integral action of the controller moves the primary bypass valve such that load and speed are brought to demanded values. A term proportional to speed error is summed with the integrated error signal to provide load damping and rapid system response.



SYMBOLS

N - SPEED
 E - ELECTRICAL POWER
 T_{HPR} - TEMPERATURE-HIGH
 PRESSURE RECUPER-
 ATOR OUTLET

T_{LPR} - TEMPERATURE-LOW
 PRESSURE RECUPER-
 ATOR INLET

T_{ti} - TEMPERATURE-TURBINE
 INLET

F - NEUTRON FLUX

*THE SAFETY VALVE AND THE TRIM VALVE ARE AT THIS
 SCHEMATIC POSITION ALSO.

Figure 5.2.2-1 HTGR-GT plant process flow diagram showing plant control inputs

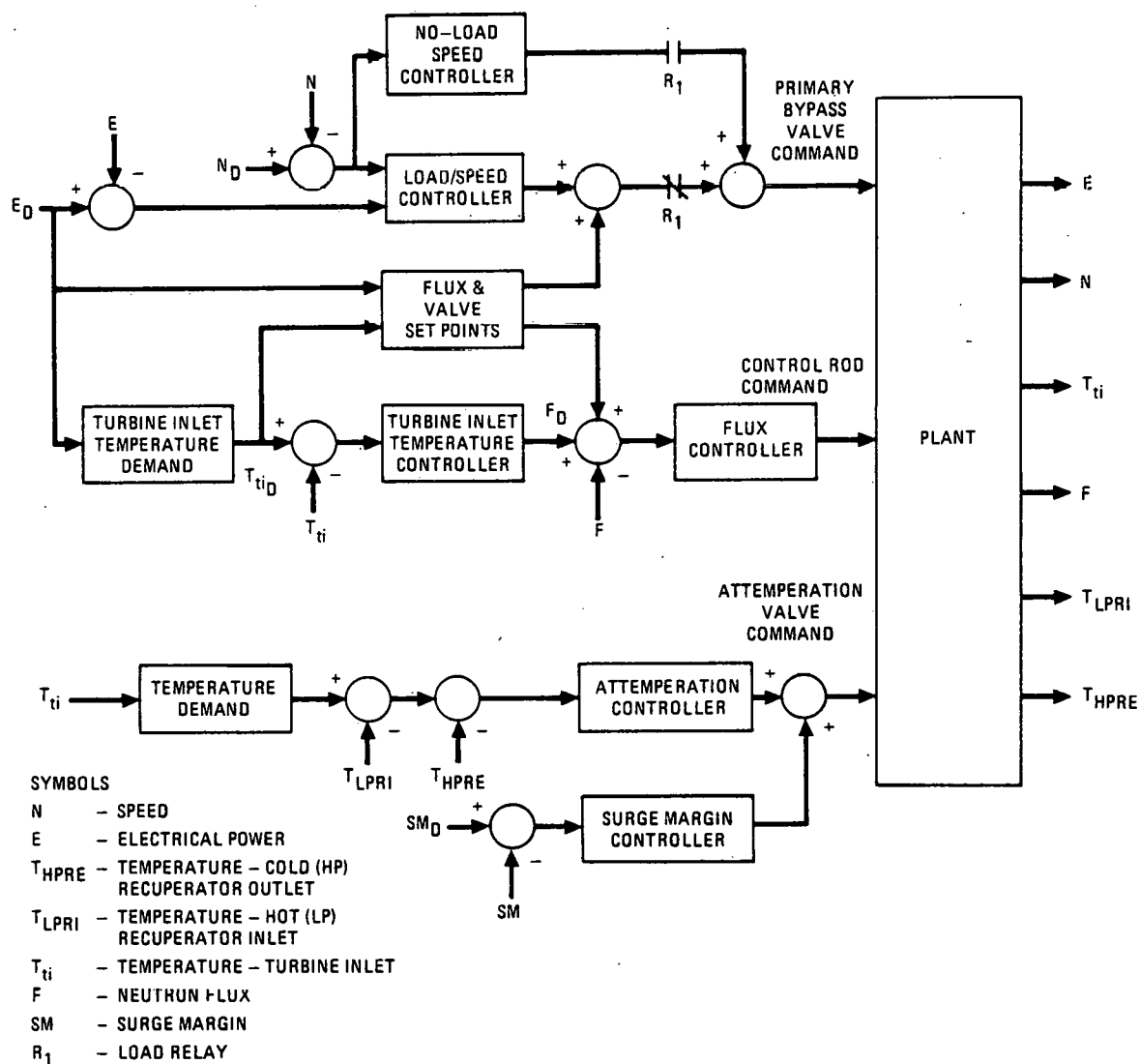


Figure 5.2.2-2 HTGR-GT plant control system

TABLE 5.2.2-1
SUMMARY OF PCS CONTROL FUNCTIONS FOR HTGR-GT 2000-MW(t) PLANT

Protective Function	Initiating Condition	Purpose	System Action/ Interfaces
Load/speed control	Normal operating mode	Regulate electric power generation and control turbomachine speed	Control speed and load with trim and primary bypass valves
Turbine inlet temperature control	Normal operating mode	Regulate reactor power to control turbine inlet temperature	Control reactor power using control rods
Attemperation control	Normal operating mode	Control thermal transients experienced by power conversion loop and reactor components	Control thermal transients experienced by power conversion loop and reactor components by actuation of the attemperation valve
No-load/speed	1. Receipt of PCS over-speed protection signal 2. Startup 3. Shutdown 4. Synchronization	Control turbine speed when operating under no-load conditions	Control turbomachine speed with primary bypass valve
Surge margin control	Surge margin below set point	Protect compressor	Open attemperation valve
Plant or loop startup	Operator command	Startup	To be determined
Plant or loop shutdown	Operator command	Shutdown	To be determined

TABLE 5.2.2-2
SUMMARY OF PCS PROTECTIVE FUNCTIONS FOR HTGR-GT 2000-MW(t) PLANT

Protective Function	Initiating Condition	Purpose	System Action/ Interfaces
Main loop trip (automatic loop shutdown)	Total or partial loss of turbomachine or generator lubricant system or lubricant design limits exceed	Prevent damage to turbomachine/generator bearings	1. Initiation of main loop shutdown through PPS main loop shutdown system 2. Initiate reduction of PCS reactor power through setback and turbine inlet temperature controller 3. Initiate turbo-machine barring
Main loop trip (automatic loop shutdown)	Turbomachine bearing seal failure or leakage	Prevent damage to primary seal	
Main loop trip (automatic loop shutdown)	Speed difference between generator and its associated turbomachine (>10 rpm)	Detect turbomachine-generator coupling failure	
Main loop trip (automatic loop shutdown)	Penetration shaft seal failure	Detect damage or leakage from shaft seal	
Main loop trip (automatic loop shutdown)	PCS or PPS precoolers isolation and dump	Prevent damage to power conversion loop components due to loss of precoolers cooling	

TABLE 5.2.2-2 (Continued)

Protective Function	Initiating Condition	Purpose	System Action/ Interfaces
Main loop trip (automatic loop shutdown)	Excessive vibration of turbomachine or generator	Prevent damage to bear- ings or seals	
Circulating water system protection	Mismatch of water flow to and from the precooler(s)	Prevent loss of water from the cooling water system	1. Close precooler isolation valves 2. Initiate shutdown of PCS affected loop through main loop trip
Circulating water system protection	Water flow to the pre- cooler(s) below design limit	Prevent boiling water in the precooler(s)	1. Initiate shutdown of PCS affected loop through main loop trip ^(a)
Reactor power setback	High reactor power-to- flow ratio (1.25)	Avoid causing a PPS power-to-flow trip	1. Insert rod pairs in preselected group to reduce reactor power
Reactor power setback	High average turbine inlet temperature (1600°F)	Avoid causing a PPS turbine inlet temp- erature trip	
Reactor power setback	Two or more electrical breakers tripped	Avoid causing a PPS power-to-flow trip	

TABLE 5.2.2-2 (Continued)

Protective Function	Initiating Condition	Purpose	System Action/ Interfaces
Reactor power setback	PCS or PPS main loop trip signal	Avoid causing a PPS power-to-flow trip	1. Initiate opening of the primary bypass valve through the PCS no-load/speed controller
Reactor power setback	Decrease of water flow to and from a precoolers below design limits	Prevent boiling of water in precoolers(s)	
Overspeed protection	Generator breaker trip	Prevent turbomachine overspeed	
Overspeed protection	Generator output power and demanded power differ by preset value	Prevent turbomachine overspeed	

(a) Check valves close on loss of flow. Isolation valves left open to avoid excessive pressure.

- No-Load/Speed Control* - Direct control of turbomachine speed is required for plant or loop startup, controlled manual or automatic shutdown, synchronization, hot standby, and overspeed protection. — For these instances the turbomachine speed will be controlled by the no-load/speed controller (Fig. 5.2.2-2). This controller commands actuation of the primary bypass valve to maintain speed at a demanded value. This demanded value may be a fixed set point, i.e., loss of load with return to idle, or it may be a programmed ramp profile for purposes such as plant startup or shutdown. The controller consists of a proportional-plus-integral network with limiters to prevent integrator saturation during large transients.
- Attemperation Control - Thermal transients experienced by the power conversion loops and reactor components are controlled throughout the normal load range by the attemperation controller (Fig. 5.2.2-2). The controller manipulates the sum of the low-pressure recuperator inlet and high-pressure recuperator exit temperatures to a demanded value that will be a programmed function of average turbine inlet temperature. Control is accomplished by actuation of the attemperation valve, which diverts helium flow from the compressor exit to the turbine exit. The temperature demand signal is designed to hold the attemperation valve closed under normal operating conditions. The command signal is nominally limited to 1°F/s to control the rate of temperature change experienced by the components. The remainder of the loop consists of a proportional-plus-integral controller with limiters to prevent integrator saturation.
- Surge Margin Control - The surge margin controller (Fig. 5.2.2-2) acts to maintain the compressor surge margin. This controller comes into action through actuation of the attemperation valve whenever the approximate surge margin is not above a set point. Opening of the valve reduces the compressor pressure ratio and, as a result, increases compressor surge margin. The measurement of surge margin for the control is not feasible in terms of direct measurement. However, direct measurements of the compressor inlet pressure, pressure rise, and inlet density are used for control purposes.
- Load-Following Capability - The HTGR-GT plant is designed to be capable of continuous operation under manual and automatic control at any point between design power and minimum load. The plant is also designed to follow load within this operating range at rates

*This function is expected to be implemented as a subfunction of the load/speed control.

of load change up to those shown in Table 5.2.2-3. The weekly load-following cycle used as a design basis is shown in Fig. 5.2.2-3. The reactor turbine system will be capable of accomplishing routine startup and shutdown operations in the approximate times shown in Table 5.2.2-4.

- Response to Critical Transients - The plant transients were analyzed to identify limiting design requirements for plant components, to assess the adequacy of the plant transient performance, and as a basis to establish the PPS requirements. This work is not complete and represents only the analysis accomplished to date on the HTGR-GT 2000-MW(t) plant. Table 5.2.2-5 shows a brief comparison of the expected performance of the 2000-MW(t) plant transients. The transients are described in more detail in Appendix B.

5.2.3 Inservice Inspection (ISI)

The basis for development of an ISI program for the plant is proposed in the ASME Boiler and Pressure Vessel Code for Gas-Cooled Systems, Section XI, Division 2.

The impact of ISI and testing on plant availability can be minimized by scheduling ISI operations to coincide with refueling and other planned outages. While some ISI operations can be conducted during normal plant operations, the bulk can be performed only during shutdowns. Although many possibilities exist for diversity in the owner's program plan due to the flexibilities permitted in the application of the code, the basic concern is the extent to which outages resulting from ISI operations can be expected to reduce plant availability.

A program of inspection and testing for NSS components, NSS/BOP interface components, and components of BOP systems will follow the prescribed typical work area logic shown by Fig. 5.2.3-1.

The plant concept under discussion comprises elements generic to the HTGR and unique to the GT. Previous studies have identified plant performance aspects of ISI and testing for generic elements. Quantitative differences in the selection of elements, such as core size and number of loops, will impact proportionally on ISI scope and duration. The following major components are unique to this concept and cycle and are located inside the containment within the PCR/V: turbomachine, hot duct, recuperator, precoolers, and helium control valves.

Inspections that directly impact on the refueling duration are those which are conducted from the top head area and which interrupt refueling sequences. Components of this category include reactor internals, specific top head penetrations, and top head portions of piping systems extending from the PCR/V. Inspection of turbomachine components requiring the use of the containment crane will also impact on refueling.

TABLE 5.2.2-3
DESIGN RATES OF ELECTRICAL LOAD CHANGE

Maximum rate of load change (for changes >10%)	5% of rated load per minute between 25% and 100% of rated load
Maximum step load change (no less than ()h ^(a) between load changes	10% of rated load in 10 s

^(a)To be determined.

TABLE 5.2.2-4
APPROXIMATE HTGR-GT STARTUP/SHUTDOWN TIMES

	Hours
Generator reload following generator trip	0.1 ^(a)
Loop restart following loop trip	(To be determined)
Normal shutdown from full load to shutdown mode	10 ^(b)
Normal startup from shutdown mode, hot turbine (Xe decayed)	10 ^(b)
Normal startup from shutdown mode, cold turbine	18
Normal shutdown from full load to refueling mode (atmospheric primary system pressure)	24
Normal startup from refueling mode, cold turbine (atmospheric primary system pressure)	22

^(a)Depends on automatic control selection, electrical system, and grid characteristics. The time required could be shorter or longer.

^(b)Times required for these modes can be varied according to utility requirements.

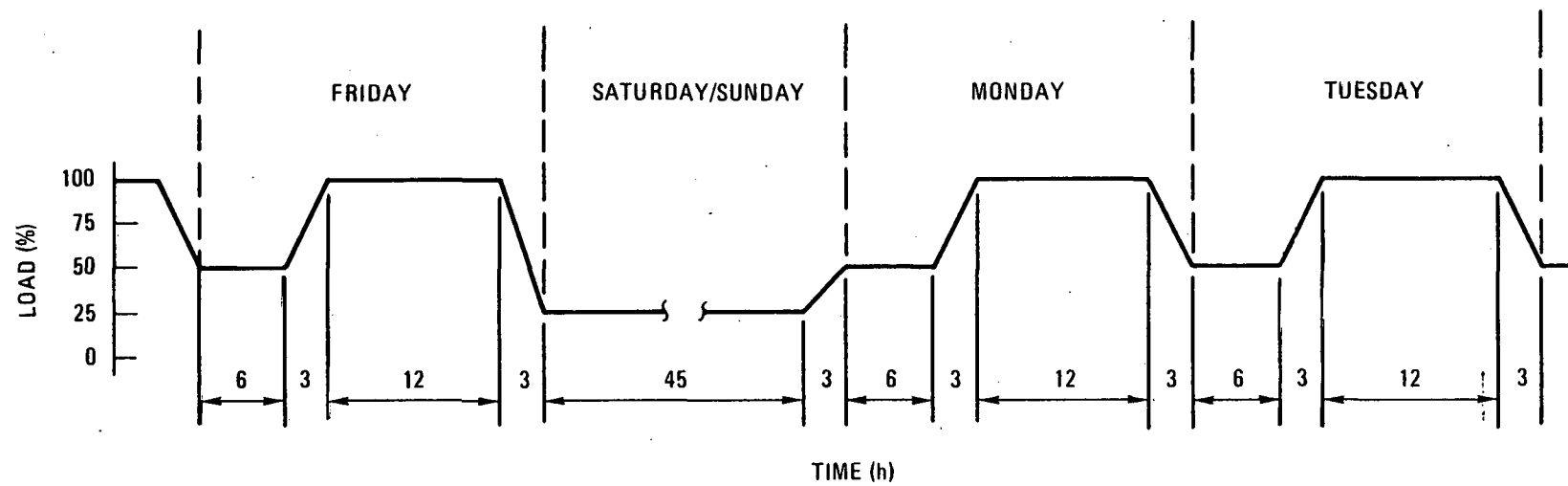
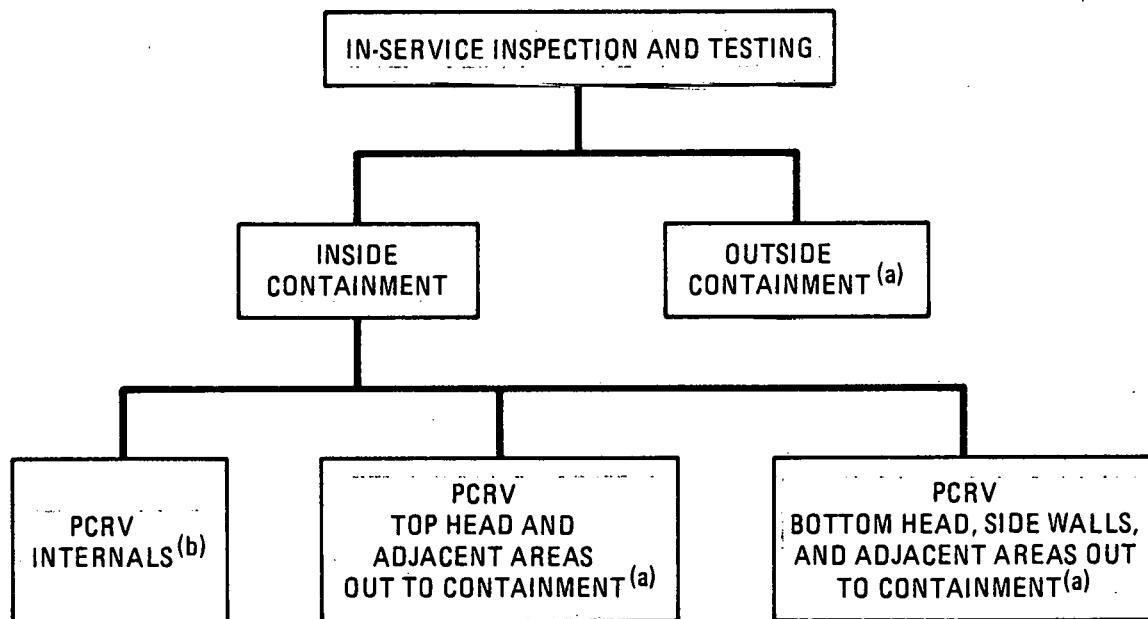


Figure 5.2.2-3 HTGR-GT plant load-following cycle

TABLE 5.2.2-5
CRITICAL TRANSIENT PERFORMANCE COMPARISON FOR HTGR-GT 2000-MW(t) PLANT

Transient Description	Max. Core Inlet Temp. [°C (°F)]	Max. Core Outlet Temp. [°C (°F)]	Max. Power-to-Flow Ratio	Max. Press. at LPR (a) Inlet [kPa (psia)]	Max. Temp. at LPR Inlet [°C (°F)]	Max. Press. at Comp. Inlet [kPa (psia)]	Max. Rate of Press. Increase [kPa/s (psi/s)]	Max. Flow Rate at LPR Inlet [10 ⁶ kg/s (10 ⁶ lbm/h)]	Max. Turbine Speed (rpm)	Max. Rate of Press. Decrease [kPa/s (psi/s)]	Max. Precooler Inlet Temp. [°C (°F)]	Max. Precooler Outlet Temp. [°C (°F)]	Max. Recup. Hot End Metal Temp. [°C (°F)]	Max. Cold End Metal Temp. [°C (°F)]
Full load - 100% nominal	498 (929)	850 (1562)	1.00	3234 (469)	534 (994)	3172 (460)	--	1.96 (4.32)	3600	--	224 (436)	27 (80)	514 (957)	208 (406)
Rapid load decrease (5%/min) (100% to 25% range)	498 (929)	850 (1562)	1.15	3310 (480)	534 (994)	3240 (470)	--	2.29 (5.02)	3600	--	224 (436)	29 (85)	514 (957)	208 (406)
10% step load decrease in 10 s	498 (929)	850 (1562)	1.04	3234 (469)	534 (994)	3247 (471)	--	2.10 (4.64)	3600	--	224 (436)	27 (80)	514 (957)	208 (406)
Single-loop load rejection	502 (936)	854 (1570)	1.13	4792 (695)	545 (1013)	4620 (670)	1489 (215)	5.40 (11.90)	3858	530 (77)	238 (460)	42 (107)	519 (967)	217 (424)
Single-loop shutdown with overspeed trip	503 (937)	863 (1586)	1.34	5964 (865)	548 (1018)	5964 (865)	1296 (183)	5.58 (12.30)	4150	572 (83)	239 (463)	32 (89)	529 (985)	218 (425)
Single-loop total loss of precooler flow	508 (946)	862 (1583)	1.19	6805 (987)	705 (1301)	6805 (987)	951 (138)	4.17 (9.20)	3615	414 (60)	309 (589)	158 (316)	588 (1090)	293 (560)
Plant total loss of precooler flow (1029) (1601)	554 (1029)	872 (1601)	1.02	6130 (889)	760 (1400)	6130 (889)	903 (131)	4.17 (9.20)	3605	758 (110)	310 (590)	165 (330)	604 (1120)	288 (550)
Slow rod withdrawal at design	512 (954)	904 (1660)	12.9	3468 (503)	537 (999)	3399 (493)	--	2.26 (4.99)	3600	--	245 (473)	39 (87)	529 (968)	218 (425)
Single-loop turbomachine shaft break	503 (938)	864 (1588)	1.32	6433 (933)	541 (1005)	6433 (933)	1262 (183)	5.72 (12.60)	4200	434 (63)	242 (467)	31 (88)	524 (975)	220 (428)

(a) Low-pressure recuperator.



NOTES: (a) DURING REFUELING SHUTDOWN AND NORMAL OPERATIONS.
(b) DURING REFUELING SHUTDOWN ONLY.

Figure 5.2.3-1 Typical work areas

All other components inside the containment external to the PCRV, including portions of piping systems extending to containment boundaries, can be inspected without impact on refueling time frames. This category includes components that can be inspected and tested during normal operations.

Inspections and tests conducted on components located outside the containment do not impact on refueling; in many cases they can be conducted during normal plant operation and are of little consequence to plant availability.

Although no specific studies relating to ISI for this plant concept have been conducted, it is estimated that tasks to be accomplished during annual shutdown will not exceed 14 days (average for 10-yr inspection interval). Four of the 14 days will impact directly on refueling.

5.2.4 Maintenance

The achievement of satisfactory plant performance is highly dependent upon a well planned and executed maintenance program. The key reactor turbine system maintenance tasks that are included within the overall plant maintenance activity include refueling, ISI (Section 5.2.3), and component and system servicing.

The major portion of the scheduled maintenance work is conducted during the annual refueling outage, with some ISI and component/system servicing performed during plant operation. Figure 5.2.4-1 illustrates those activities that are conducted concurrent with the refueling operation and those that are performed sequentially and thus contribute to increased plant outage. It can be seen that the turbomachinery and generator overhaul is conducted during the scheduled reactor turbine system outage. Current plans for the initial plant call for the removal and servicing of a turbomachine unit at 6-yr intervals. To minimize the outage, the unit replaced would be a spare machine.

The critical path for the annual RTS scheduled maintenance outage encompasses refueling, high-temperature filter and adsorber handling, and ISI tasks. The most time-consuming operation within this sequence of events is the refueling activity. The highly efficient fuel handling components are serviced during plant operation immediately preceding plant shutdown for the annual refueling outage. With the principal exception of the control rod drive mechanism, the remaining RTS components are maintained in parallel with the refueling effort. The maintenance tasks for the control rod drive mechanisms are performed at the reactor equipment service facility after the reactor has returned to powered operation. Scheduling of control rod drive servicing fairly soon after the annual refueling will provide an early indication of any problems which might be developing with the drives.

The maintenance studies conducted for the HTGR-GT were related principally to the turbocompressors, precoolers, recuperators, hot ducts, and

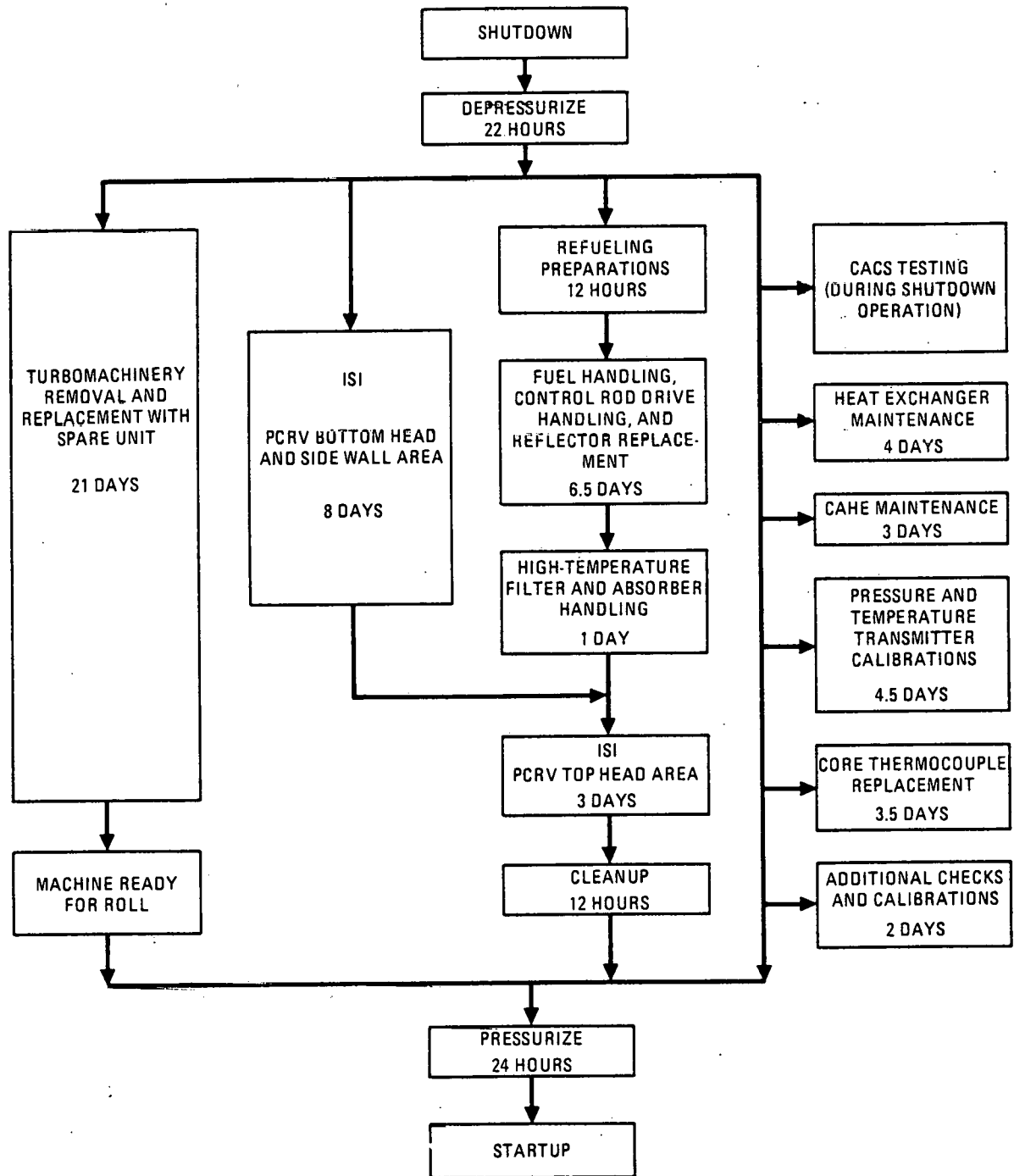


Figure 5.2.4-1 HTGR-GT plant typical refueling outage

helium control valves. These components had been identified by a utility advisory committee as being of particular concern from the maintenance and servicing aspect.

During the scheduled maintenance program, the pacing item is turbomachinery removal and replacement. Based upon the use of a spare turbo-machine, the estimated outage time is 21 days. The refueling and other scheduled maintenance tasks are conducted within this time frame.

The major unscheduled maintenance tasks that would promote a plant shutdown for restoration or replacement of RTS components include tasks involving the precoolers, recuperators, CAHEs, and auxiliary circulators. Other than tube plugging of the heat exchangers, no other major servicing of these components can be performed in-situ. The removal of the circulators has been provided for, i.e., cask and handling capabilities. Temporary structures, handling capabilities, etc., would have to be provided in the unlikely event of heat exchanger removal and replacement.

5.2.5 Reliability and Availability Goals

Availability is defined as the percent of time, averaged over plant lifetime, that a reactor is operating, or capable of operating, at design conditions. Experience with LWR electric generating stations shows achieved availabilities on the order of 70%. Considerable effort is being made in various plant improvement programs to increase this value for operating LWRs. In addition, an extensive availability assurance plan was adopted in the design phase for the Sundesert plant, which set a goal of 90% availability. The drive toward achieving higher availability is expected to continue into the 1990s, and HTGR power plant programs must be competitive. The availability goal for the mature (>5-yr operation) HTGR-GT plant, therefore, has been set at 90%, and a program to achieve this goal will be devised and implemented.

In order to guide the design effort toward achieving a target availability, the total permissible annual average downtime is divided and allocated as goals for both scheduled and unscheduled categories. Scheduled downtime is, of course, the time allocated for operations such as refueling and planned maintenance. Unscheduled downtime is associated with shutdowns forced by equipment failure or operator error and is estimated from projected reliability and time required for repair. These plant level allocations are further divided into major portions such as RTS and BOP and then further divided into systems, subsystems, and components. At all design phases, a tradeoff of equipment cost versus availability can be considered in terms of the cost of downtime, and the allocations adjusted as necessary for feasibility and/or cost considerations.

To achieve an average availability of 90% over the plant lifetime, the total average annual downtime cannot exceed 876 h. For initial allocations, it is often assumed that scheduled and unscheduled downtimes contribute equally at 438 h/yr. Case I in Table 5.2.5-1 shows an allocation based on this assumption. The allocated times of scheduled items other than turbomachine service are based on estimates for previous HTGR-GT and HTGR-SC concept studies. This allocation would force the turbomachine overhaul or service to be accomplished, on the average, within 438 h/yr. As noted in the prediction column of the table, however, the estimated time for turbomachine service (every 6 yr for each of the two machines for an average interval of 3 yr) is 504 h plus 45 h for shutdown and startup of the reactor. This yields an annual average of 480 h of scheduled outage, which is the best estimate at this time. Hence, Case II in Table 5.2.5-1 provides an allocation that assumes the average scheduled downtime is 480 h/yr.

The allocation unscheduled downtime for Case II cannot exceed the difference between 876 h and the scheduled downtime, yielding 396 h/yr. Based on previous HTGR-GT and HTGR-SC analyses, allocation of 195 h for systems common to the GT and SC concepts is adopted for preliminary allocation versus the 280 h predicted. This leaves 201 h/yr for the unscheduled downtime associated with components that are exclusive to the HTGR-GT. The total predicted downtime for these components, scaled from previous analyses of a three-loop HTGR-GT, is 275 h/yr (last column of Table 5.2.5-1).

In summary, to achieve a 90% availability goal, the presently predicted total unscheduled downtime of 550 h needs to be reduced to 400 h, given that the predicted scheduled downtime and unscheduled downtime for the GT/SC common components can be realized. The above preliminary predictions and goal allocations should not be viewed as an accurate assessment of the availability factor for the HTGR-GT but as indications of the feasibility of achieving the high availability goal and the magnitude of the improvements needed. Moreover, achievement of an availability goal of 90% or greater will require a disciplined engineering program that considers the effects of every facet of design, procurement, installation, and operation on the availability objectives.

5.2.6 Operator Exposure

Radiation exposures to plant personnel include the exposures arising from reactor operation and surveillance, refueling, waste processing, ISI, and special (or unplanned) maintenance. Occupational exposures for the HTGR-GT plant are yet to be assessed. It is projected that the man-rem exposure for the HTGR-GT plant will be approximately 25% greater than that for a comparably sized HTGR-SC plant, as discussed below.

Two major factors contribute to the increase in the occupational exposure for the HTGR-GT plant. One is the maintenance associated with the primary system, which is more complex in the HTGR-GT. The other is the additional access requirements to the containment building during

TABLE 5.2.5-1
AVAILABILITY GOALS AND PREDICTIONS FOR TWO-LOOP DIRECT CYCLE HTGR-GT

	Preliminary Goals [h/yr (%)]		Preliminary Predictions [h/yr (%)]
	Equal Scheduled and Unscheduled (Case I)	Dominant Scheduled (Case II)	
Scheduled downtime			
Shutdown and startup	45	45	45
Refueling	330	330	330
Inservice inspection	60	60	60
Filter absorber replacement	10	10	10
Turbomachine (3-yr interval)	<438(a)	504 (max)(b)	504
Annual average	438(a) (5)	480 (5.5)	480 (5.5)
Unscheduled downtime			
Requirements common with other concepts(c)	195	195	280
Requirements exclusive to HTGR-GT(d)	243	201	275
Total unscheduled	438 (5)	396 (4.5)	555 (6.3)
Total scheduled and unscheduled	876 (10)	876 (10)	1035 (11.8)
Availability (%)	90	90	88

(a) Not presently considered achievable.

(b) Maximum 3-yr interval.

(c) For example, reactor, helium pump, CACS, circulator, etc.

(d) Turbomachine, precooler, recuperator, and power conversion loop valves.

reactor operation because of more auxiliary equipment items in the HTGR-GT containment.

It has been established as a design basis for the HTGR-GT that the rate of occupational dose accumulation be limited to 200 man-rem/GW(e)y with the breakdown by the type of operation as shown in Table 5.2.6-1; this is based on the information presented in Ref. 1 and includes adjustments to account for the in-vessel refueling scheme. The expected rate of accumulation will be less than 100 man-rem/GW(e)y.

Although the exposure for the HTGR-GT is greater than that for the HTGR-SC, it is still considerably lower than the actual LWR experience of 1570 man-rem/GW(e)y. Actual man-rem exposures at the Peach Bottom HTGR and at the FSV HTGR have been exceptionally low. The annual collective dose has never exceeded 10 man-rem. Furthermore, the operating experience at the FSV plant confirms that the rate of dose accumulation for the HTGR plant is less than 100 man-rem/GW(e)y.

5.2.7 Radioactive Effluents and Waste

During normal operation of the HTGR-GT, radioactive material will be produced by fission and by neutron activation of constituents of the primary helium coolant. Most of the fission products will remain within the coated fuel particles; however, small quantities may escape through the pyrolytic carbon coatings into the graphite of the fuel elements and finally diffuse into the primary helium coolant. It is expected that reactor core components may be contaminated with graphite dust and lightly adherent films of plateout activity. Additionally, the primary coolant and its attendant fission products may leak at a very low rate from the operating reactor to the containment and secondary systems and subsequently to the environment. The HTGR-GT plant will have installed waste treatment systems designed to collect and process the gaseous, liquid, and solid waste produced by coolant purification processes, decontamination procedures, and various system leakages that may occur during plant operations.

Preliminary estimates of the quantities of radioactive waste generated during the operation of the HTGR-GT plant have been calculated. The following discussion briefly summarizes the information on HTGR-GT radioactive wastes and compares estimates of HTGR-GT waste quantities to those for LWR plants of similar thermal power rating.

5.2.7.1 Liquids

During operation of the HTGR-GT plant, a number of radioactive liquid wastes are generated, collected, and subsequently processed by the liquid waste processing system. The expected sources of the liquid radioactive waste are as follows:

1. Decontamination system fluids.

TABLE 5.2.6-1
MAN-REM DESIGN BASIS FOR 2000-MW(t) HTGR-GT PLANT

Type of Operation	Design Basis (a) (man-rem/y-unit)
Refueling	10
Reactor operation and surveillance (b)	25
Reactor turbine system maintenance and ISI	22
Balance of plant maintenance	50
Special maintenance	<u>25</u>
Total	135
Rate of accumulation, (b) man-rem/GW(e)y	~200

(a) 40% thermal efficiency and 80% load factor assumed.

(b) Mainly from containment access during operation.

2. Water drained from the helium regeneration cooler and from the radioactive gas recovery system (if installed).
3. Low-level laundry and contaminated shower liquid waste.
4. Water condensate that may result from infrequent operational occurrences of a precooler dump.

Most high specific activity liquids will not be processed for reuse but will be solidified or otherwise fixed and treated as solid waste. Other low specific activity fluids (e.g., laundry and contaminated shower water) will be sampled, analyzed, and processed if necessary. (Presumably, processed liquids could be recycled into plant systems.)

Normally, no water from the liquid waste system would be released to the environment. However, if the processed water is not recycled as makeup water to various plant systems, and if its concentration satisfies discharge limits, the liquid may be discharged to either receiving water bodies or sanitary sewage systems.

Estimates of the quantity and activity levels present in liquid waste effluent for the HTGR-GT are summarized in Table 5.2.7-1. Also provided in Table 5.2.7-1 are similar estimates of the liquid waste effluents for LWR (both BWR and PWR) plants. It is apparent that anticipated HTGR radioactive liquid waste discharges to the environment are but small fractions of those for similarly sized LWR plants in terms of both activity level and quantity.

5.2.7.2 Gases

In the normal conduct of plant operations, small quantities of radioactive material will be released to the atmosphere in gaseous effluents. Radioactive gaseous sources include the following:

1. PCRV leakage of primary coolant to the containment building.
2. Helium purification system regeneration off-gas.
3. Radioactive gas recovery system off-gas (if this option is employed).
4. Radioactive analytical instrument sampling effluent.
5. Fuel handling operations (auxiliary service cask off-gas, fuel handling machine off-gas during refueling, and fuel shipping cask off-gas).
6. Liquid and solid radioactive waste processing system off-gas.

TABLE 5.2.7-1
ANNUAL RADIOACTIVE WASTE GENERATION - NORMALIZED TO A POWER OF 800 MW(e)

Reactor/Data Source	Annual Release of Radionuclides (Ci)				Discharge to Waste Storage Facilities, Solid Wastes	
	Liquid Wastes		Airborne Wastes		Volume (m ³)	Total Activity (Ci)
	Mixed Fission Products (No Tritium)	Tritium	Noble Gases	Iodine and Particulate		
HTGR-GT						
Continuous purge option (a)	0.004(b)	0(c)	170(d,e,f)	0.012	58	13,000(g)
Intermittent purge option(h)	0.004(b)	0(c)	1.6(d,e,f)	0.005	58	13,000(g)
BWR						
Operating plant average - 1976(i)	4.8	32	240,000	0.8	1,600	5,500
Improved treatment capability(j)	0.32	16	52,000	0.32	630	1,200
Most advanced treatment capability(k)	0.32	16	3,300	0.006	~630+	~1,270
PWR						
Operating plant average - 1976(i)	5.5	790	16,000	0.16	790	320
Improved treatment capability(j)	0.024	280	3,200	0.040	160	4,800
Most advanced treatment capability(k)	0.002	280	1,070	0.0045	~160+	~4,800+

TABLE 5.2.7-1 (Continued)

- (a) The continuous purge option employed assumes once through containment ventilation at a rate of 0.5 volume/hour. Effluent is filtered at an efficiency of 99.97% for particulate and 99% for Halogens. No entrapment of noble gases is possible.
- (b) Essentially only low specific activity fluids (e.g., laundry and contaminated shower water) would be available for release after sampling and processing. Discharge would be to either cooling tower blowdown (if wet cooling tower option selected) or to receiving water bodies, streams, lakes, etc. or perhaps sanitary sewers if MPC levels are satisfied. [Ref. Fulton PSAR Table 11.2.2-1.]
- (c) Waste containing tritium in significant concentrations occurs only in high specific activity liquids which are subsequently solidified and processed as solid waste. No release of H-3 contaminated liquid to the environment in liquid waste discharge is anticipated.
- (d) Airborne tritium release of 0.09 Ci/year included in total.
- (e) Discharge of noble gases from the gas waste system not anticipated. Recycle of Kr-85 with the eventual licensed disposal at plant decommissioning is planned.
- (f) Airborne release indicated is primarily due to containment building ventilation and leakage. Reactor service building ventilation release of Noble Gases are expected to be less than 0.2 Ci/year and are primarily the result of gas waste compression and gas recovery system expected leakage [Ref. Delmarva PSAR, Table 11.3.6-2].
- (g) Approximately 48 m³ as low level waste (620 Ci), 3 m³ as titanium sponge waste (8300 Ci) and 10 m³ as replaceable reflector block waste (4100 Ci). [Ref. Fulton SER, NUREG-75/033.]
- (h) Intermittent purge for HTGR-R anticipated to be 2 complete containment purges per year. The containment atmosphere engineered clean-up system is actuated 24 hours prior to containment ventilation to the atmosphere. Ventilation of the containment atmosphere is assumed to be filtered during discharge to the environment, effluent is filtered at an efficiency of 99.97% for particulates, 99% for Halogens and 0% for noble gases.
- (i) Information reported in NUREG-0367, "Radioactive Materials Released from Nuclear Power Plants (1976)," T. R. Decker 3/78 was normalized to form a "typical" 800 MW(e) PWR or BWR plant for the year 1976. For the year 1976, NUREG-0367 reports a total BWR thermal capacity of 26.3 GW(t) and PWR thermal capacity of 41.9 GW(t).
- (j) Ref. WASH-1258, "Numerical Guides for Design Objectives and Limiting Conditions for Operation to Meet the Criterion 'As Low As Practicable' for Radioactive Material in Light-water-cooled Nuclear Power Reactor Effluents," Volume 1, July 1973. NOTE: Treatment Systems BWR-3 and PWR-5 selected as representative of improved treatment capability.
- (k) Most advanced treatment capability of WASH-1258 selected as follows: Airborne treatment system BWR-7 and PWR-8, and liquid treatment system BWR-3 and PWR-6.

The reactor containment structure will be purged with air on a once-through basis to maintain airborne radioactive material at a level below allowable limits for access to the containment. This ventilation air, which contains PCRV leakage, will be exhausted through prefilter, HEPA, and activated charcoal filters to the atmosphere. For the HTGR-GT, the remaining environmental discharge of gaseous wastes is through reactor service building leakage and normal operation and discharge from the gas waste system.

Table 5.2.7-1 summarizes the expected release of gaseous wastes to the environment for the 800-MW(e) HTGR-GT operating on a once-through air purge basis. Additionally, the table displays expected gaseous release estimates for an HTGR-GT closed containment/ intermittent purge option. As would be expected, the intermittent purge option greatly reduces the release of gaseous waste, as containment ventilation to the environment is precluded between purges, thus allowing radioactive decay of the contained noble gases.

The LWR gaseous waste (scaled to power level) is also included in Table 5.2.7-1 for comparison. Two estimates of LWR gaseous release are given: (1) actual measured releases for existing plants averaged over 1976, and (2) estimates of BWR and PWR releases with moderate and extensive liquid and gaseous waste treatment systems installed.

It is apparent from Table 5.2.7-1 that HTGR-GT gaseous waste releases to the environment are but small fractions of measured LWR discharges in 1976. Extensive gaseous waste stream treatment would reduce LWR releases; however, it is felt that HTGR-GT releases would continue to be less than LWR discharges.

5.2.7.3 Solid Wastes

Solid radioactive waste will be generated during plant operation and will require disposal. Sources of solid waste for the HTGR-GT are as follows:

1. Reactor core components such as replaceable reflector blocks, in-core instrumentation, control rods, and drive mechanisms.

*Intermittent purge for the HTGR-GT is anticipated to be two containment purges per year. The containment atmosphere cleanup system is assured to be activated 24 h prior to containment ventilation to the atmosphere. Ventilation of containment atmosphere contents is assumed to be filtered during discharge to the environment.

2. Spent resins resulting from demineralizer use and CO₂ absorber present in the helium purification system.
3. Low specific activity material resulting from plant operation such as paper, plastic film, tape, protective clothing, small tools, air filter elements, and miscellaneous electronic equipment from contaminated areas.
4. Spent tritium absorption medium - titanium sponge from the helium purification system.
5. Spent high-temperature filter absorber material.
6. Solidified liquid waste.
7. Turbomachinery blades (assuming replacement rather than decontamination).

Solid wastes are processed and packaged on-site and shipped off-site to a licensed burial site in accordance with NRC and Department of Transportation regulations. Gaseous and liquid wastes potentially generated in the operation of the solid waste processing and packaging system will be collected and processed by their respective waste systems.

Estimates of the annual quantities of solid waste anticipated for HTGR-GT operations are summarized in Table 5.2.7-1. This table also summarizes (1) reported LWR solid radioactive waste generation for 1976, and (2) estimates of LWR solid wastes generated by radioactive waste treatment systems of moderate capability.* The quantity of solid radioactive waste anticipated for HTGR-GT operation is less than that produced in LWR operations; however, the total activity** present in HTGR-GT solid waste is slightly higher than LWR estimates.

* See WASH-1258 for additional information on projected solid radioactive waste processing system characteristics.

**The higher specific activity of some HTGR solid waste, notably the replaced reflector blocks and the titanium sponge tritium content, make HTGR total activity in solid waste somewhat larger than that for LWRs.

5.3 Safety/Licensing

5.3.1 Inherent and Passive Safety Features

The HTGR design has inherent and passive features that make the risk to the public from accidental releases of radioactivity extremely low. These features require less reliance on complex active systems and also help to maintain the integrity of the reactor core and to retain the radionuclide inventory. An additional desirable characteristic is that consequences of accidents develop rather slowly, thus allowing time for deliberate and planned actions by the operators. The key safety features are described briefly below.

- Graphite-Moderated Ceramic Core - The HTGR core is constructed exclusively of ceramic materials, primarily graphite, which maintain their integrity at very high temperatures - well above normal operating conditions. The massive core, together with a low power density and strong negative temperature coefficient of radioactivity, ensures that reactor temperatures and power transients will proceed slowly. In the event of loss of core cooling capability, the graphite acts as a crucible by maintaining its structural form for weeks with only some local sublimation and with no possibility of meltdown. Volatile radionuclides are released gradually over a period of days, while the nonvolatile nuclides are retained. Because meltdown is not possible, the integrity of the PCRV and containment is maintained as a retardant to fission product transport. Interruptions in core cooling of more than 30 min can be tolerated without any damage to primary coolant system components. Because of the slow rate of temperature rise and the retention characteristics of the fuel at elevated temperatures, at least 3 h would always be available to restore cooling system operation after interruptions in core cooling before any fuel damage or radioactivity release could occur.
- Helium Primary Coolant - The primary coolant of the HTGR is helium. Because helium is chemically inert, has a low stored energy content, and remains in the gaseous phase under all conceivable operating conditions, it cannot chemically react with the core, vessel, or containment structures in a destructive way and results only in mild pressure transients in the containment subsequent to a blowdown. Since heat can be removed from the reactor core with any mixture of gases present even at atmospheric pressure, it is not necessary to maintain an inventory of coolant in the reactor vessel to prevent overheating of the fuel. Hence, coolant injection systems are not needed. The single-phase gas coolant permits the design of cooling systems that minimize the dependence on operator actions to bring the reactor to a safe cooldown, a feature that has great benefit during an accident.
- Prestressed Concrete Reactor Vessel - A special passive feature of the HTGR is the use of a PCRV, which was introduced in gas-cooled reactors in Britain because of its safety characteristics.

The PCRV is a massive, structurally redundant monolith that encloses the entire primary coolant system boundary with the exception of relatively small process lines. During postulated core heatup accidents, the PCRV retards the transport of fission products to the containment and helps protect the containment from the degrading effects of the core heatup environment. The strength of the PCRV is provided, in part, by a large number of steel tendons, which run axially through and circumferentially around the vessel. The concrete shields the tendons from neutron irradiation, thereby preventing embrittlement, and since the concrete is under compression, tiny cracks tend to be self-sealing and are not subject to propagation.

5.3.2 Design Safety Features

In addition to the inherent safety features of the HTGR, a number of safety systems are incorporated in the plant design to further reduce the risk from transients and accidents. The principal safety systems are described briefly below.

- Core Auxiliary Cooling System - The CACS consists of three independent cooling loops, which circulate and cool primary system helium to remove reactor decay heat during reactor shutdown when the main loops are unavailable. The CACS is also used if the main cooling loops are out of service for maintenance. Each independent core auxiliary cooling loop includes a heat exchanger, an auxiliary circulator, and a helium shutoff valve, all located within an independent PCRV cavity. The cooling water system, which supplies water to the heat exchangers inside the PCRV, is external to the PCRV and transfers heat to the ultimate heat sink. Each of these cooling loops is capable of removing 50% of the core residual and decay heat from full-power steady-state operation with the PCRV either pressurized or depressurized (with air ingress).
- Containment Building - The containment structure is the final barrier to the release of radioactive material. Although it is of conventional design, the containment structure of the HTGR is particularly effective because of the unique HTGR features described above. The passive features of the core and PCRV and the choice of coolant ensure that rapid releases of energy will not occur and that the low leakage characteristics of the containment will be maintained. As a result, the consequences of even severe HTGR accidents are inherently low. The containment structure also protects the reactor from external events such as explosions on transportation routes or at nearby industrial facilities and from natural events such as tornadoes.
- Containment Atmosphere Cleanup System - The purpose of the containment atmosphere cleanup system is to minimize the availability of fission products that might leak from the containment. This

system is capable of removing and retaining radioactive particulates and halogens that would be present in the containment atmosphere as a result of accidental release of fission products from the reactor coolant system.

- Containment Isolation System - The containment isolation system assures that a protective barrier exists for each process line that penetrates the containment structure. The system of isolation valves and associated controls is designed to automatically limit the release of radioactivity to the environment as a result of accidents.
- Plant Protective System - The plant protective system consists of sensors, electronic logic, and actuated devices. The system functions to prevent a release of radioactivity by initiating action to protect the integrity of (1) the fuel particle coatings, (2) the primary coolant system boundary, and (3) the containment. The plant protective system initiates safety functions such as reactor trip, CACS startup, containment isolation, and precooling isolation and dump.
- Reserve Shutdown System - The HTGR is provided with a shutdown system independent of and diverse from the normal control rod system. Neutron-absorbing material, in the form of pellets, is stored in a hopper in each refueling penetration. If required, this material can be released by the operator to fall into a channel in each region of the core. In the absence of control rod action at any time in core life, the reserve shutdown system by itself has sufficient negative reactivity to shut the reactor down.
- Liner Cooling System - The PCRV liner cooling system consists of two independent water cooling loops attached to the PCRV liner and penetrations and, for the most part, embedded in the concrete. During normal operation, the liner cooling system maintains the temperature of the liner and concrete within specified limits. In the unlikely event of complete loss of forced circulation cooling in the HTGR-GT, the system can remove sufficient heat to delay and perhaps prevent PCRV concrete decomposition, thereby maintaining the structural integrity of the PCRV and containment building.
- Gas Turbine Safety Features - The direct-cycle gas turbine includes safety features designed to maintain a low probability of serious accidents or to mitigate their consequences should they occur. The main features are:
 1. Overspeed control system, which opens bypass valves to prevent turbomachine overspeed.
 2. Pressure relief system, which prevents PCRV overpressurization by discharging gas from the high-pressure region of the PCRV to the low-pressure region of the PCRV.

3. Precooler leak detection, pressure relief, and isolation and dump systems, which prevent the release of radioactivity to the environment and the ingress of significant quantities of water to the precooler cavity.
4. Turbomachine missile barriers, which are designed to contain high-energy segments of turbine or compressor disks.
5. Extensive diagnostic instrumentation, which continuously monitors machine performance and provides advanced warning of incipient failures of various types.

5.3.3 Safety/Licensing Issues

Although analyses and risk assessment studies have shown the risk associated with an HTGR to be very low, there are some development-related issues and questions from past reviews that have not been entirely resolved with the licensing authorities. In addition, there are design features of the direct-cycle plant that have not been subjected to regulatory review. It is believed, however, that all issues that have been raised in the past can be resolved by relatively straightforward engineering. The major licensing issue for the direct cycle is expected to be the turbomachine, its location within the primary coolant system, and the potential for severe accidents arising from turbomachine structural failures (see Section 5.5.6).

- Issues from Previous NRC Reviews - An application has not been made for review of a plant of the design described herein. However, the direct-cycle HTGR-GT is a variation of a design that has undergone three NRC reviews at the PSAR stage: the 3000-MW(t) Fulton Generating Station (Philadelphia Electric), the 2000-MW(t) Summit Power Station (Delmarva Power and Light), and GA's standard NSSS design for a 3000-MW(t) plant. Reviews of both the Summit and Fulton PSARs were carried to completion of NRC Safety Evaluation Reports (SERs). The NRC review of the standard plant PSAR (GASSAR-6) was not completed, but a partial preliminary safety evaluation was prepared.

As a result of these reviews, a number of safety or licensing issues have been raised by the NRC, and GA has responded or made plans to respond to these issues in several ways. The most significant of these issues are discussed briefly below.

- Design Criteria for Graphite Structures - Previous criteria for allowable stresses and the treatment of secondary stresses have been criticized as being nonconservative. Continuing effort has been devoted to development of stress criteria and their experimental verification.
- Core Seismic Response - Code development and correlation of code calculations with results from a variety of verification tests have been largely completed. Correlations generally gave satisfactory results, and confidence in the ability to predict core seismic response is improved.
- Inservice Inspection and Testing - Inservice inspection and testing of safety-related equipment are important functions in any nuclear plant. The HTGR, having some unusual features, will receive close attention in this regard. Since the earlier NRC reviews, Section XI, Division 2 of the ASME Boiler and Pressure Vessel Code, "Rules for Inspection and Testing of Components of Gas-cooled Plants," has been developed and should provide a basis for resolving inservice inspection issues.

- Preoperational Vibration Testing of Reactor Internals - This item is an issue only to the extent that the procedures and extent of testing may be debated. The proposed HTGR-GT design may motorize the turbomachine with the electric generator to produce gas flow through the primary loops but not at full helium inventory or temperature.
- Anticipated Transients Without Scram - This perennial issue for LWRs is slowly being resolved; at such time that the NRC makes a final determination on possible new requirements for LWRs, the potential issue for the HTGR can be better defined. It is believed, however, that the HTGR design with the completely independent reserve shutdown system and large negative temperature coefficient of reactivity makes anticipated transients without scram an insignificant safety event.
- Confirmation of the Containment Design Basis - Questions have been raised concerning containment mixing models for a depressurization accident, back-pressure for CACS operation in a depressurized mode, depressurization blowdown areas, gas flammability, and containment leak rates. While it has always been felt that these questions were satisfactorily addressed, efforts have continued since the earlier reviews to more fully define and resolve significant problems, and the analytical tools (e.g., containment atmosphere response code) are now considerably advanced.
- Long-Term Behavior of Metallic Components of Primary Coolant System - One of the main advantages of the HTGR - the high temperature produced - requires using some metals to near their structural limits. While it is intended that adequate conservatism will be provided by the design, long-term behavior (e.g., creep properties) is not always well known. Therefore there is a need for long-duration, high-temperature testing of a variety of materials; these tests are ongoing so that maximum advantage can be taken of HTGR capability.
- Thermal-Hydraulic Phenomena During Safe Shutdown Cooling - It has been recognized that the HTGR can potentially produce streaks of high-temperature gas in the lower plenum, and during an accident in which there is low circulation or loss of forced circulation, reverse flow of hot gas into the upper plenum may occur. Also under low-flow conditions, laminar rather than turbulent flow may exist. The complex flow conditions are difficult to model; however, code development (e.g, RECA) is continuing and plans have been developed for mixing tests.
- Low-Probability Accident Definition - Low-probability accidents are of continuing interest for all types of reactors. Applicants for construction permits are required to analyze a

variety of transients and accidents of severity up to and including the so-called design basis accidents. While design basis events are expected to be of very low frequency, there is a spectrum of even lower frequency events that is studied because of its potentially severe consequences. For several years GA has had the methodology to treat these events (AIPA), and it continues to be used on a limited basis. This methodology is consistent with recommendations for greater use of risk assessment from investigations of the Three Mile Island accident.

- Issues Specific to Direct Cycle - The direct-cycle HTGR-GT concept brings large rotating machinery in close proximity to the primary coolant system boundary. It also requires that the turbomachine shaft cross that boundary with a rotating seal at the interface. Potential failure modes of this machinery with their consequential effects on plant systems introduce licensing issues that have not been considered in previous evaluations of the HTGR-SC plant.

Several reviews of turbomachine failure data have produced estimates of the rate of catastrophic turbomachine failure with a range of 10^{-3} to 10^{-5} /machine-yr. This range is consistent with the 10^{-4} /yr rate for steam turbine failure used by the NRC staff. While it is believed that careful design, quality assurance procedures, and diagnostic instrumentation will ensure a rate for major failures of below 10^{-4} /turbine-yr, the consequences of these events can be severe if they are not adequately accounted for in the design of other systems.

The most severe accident appears to be the rapid deblading or rotor failure that results in pressure equilibration within the PCR. (Failures in the recuperator or ducts can also lead to rapid pressure equilibration, but these passive systems are more amenable to preventative or mitigating measures.) The accident will produce direct mechanical damage, perhaps including a failure in the shaft seal. The predominant effect, however, is the sudden load applied to the reactor core and core support, thermal barrier, CAHs, and other PCR internals due to the pressure transient. The major safety/licensing issue is therefore to ensure by design that given this low probability accident, the reactor can be shut down and adequately cooled and that containment integrity will be maintained. This ensures that offsite dose consequences will be within acceptable levels. This design issue is also discussed in Section 5.5.6.

5.3.4 Site Boundary Dose - Gaseous Releases

The radiological impact of HTGR-GT and LWR effluents has been estimated for this study in terms of boundary dose levels for the corresponding airborne effluents. Table 5.3.4-1 provides the estimated site boundary airborne effluent dose levels for normal and off-normal conditions.

TABLE 5.3.5-1
ESTIMATED SITE DOSE LEVELS FOR NORMAL OPERATION, DESIGN BASIS ACCIDENTS, AND THE SITING EVENT,
AIRBORNE EFFLUENTS ONLY

Normal Operations - Steady-State Releases, Annual Basis [Normalized to 2170 MW(t)]						
Dose Category	10CFR50 Appendix I Limits	Annual Site Boundary Dose ^(a) (mrem)				
		HTGR-GT ^(b)		BWR ^(c)		PWR ^(c)
		Continuous Containment Ventilation	Intermittent Purge	Estimated Most Advanced Treatment Capability	Estimated Most Advanced Treatment Capability	
Whole body gamma (external exposure)	5	0.6	0.0014	0.65	0.02	
Thyroid inhalation - adult	15	6 x 10 ⁻⁶	1.4 x 10 ⁻⁶	0.09	0.05	
Thyroid - infant (grass/cow/milk/ infant pathway)	15	0.002 ^(d)	0.0004 ^(d)	7.6 ^(e)	7.0 ^(e)	
Severe Accident Conditions - Design Basis/Siting Events (Normalized to 2170 MW(t))						
Representative Accident	10CFR100 Limits (rem)		0 to 2 h Exclusion Area Boundary Dose (rem)		0 to 30 Day Low Population Zone Dose (rem)	
	Whole Body Gamma	Inhalation Thyroid	Whole Body Gamma	Inhalation Thyroid	Whole Body Gamma	Inhalation Thyroid
HTGR-GT						
Design basis depressurization ^(f) accident (DBDA)	25	300	0.001	0.03	0.0003	0.003
Maximum hypothetical ^(g) fission product release (MHFPR)	25	300	0.001	0.03	0.3	17
PWR						
Loss of coolant accident ^(h) (LOCA)	25	300	1.8	96	1.2	83

TABLE 5.3.5-1 (Continued)

(a) For the HTGR exclusion area boundary (EAB) distance at 425 m and low population zone (LPZ) distance at 1600 m.

(b) Footnotes (a) and (h) of Table 5.2.7-1 apply. Also, an annual average X/Q of 2.0×10^{-5} s/m³ was used.

(c) See Table 5.2.7-1 for definition of most advanced treatment capability air-borne waste release of noble gas and iodine/particulates.

(d) An adult thyroid inhalation dose to child milk pathway thyroid dose conversion factor of 320 was used.

(e) An adult inhalation to child pathway conversion factor of 85 x BWR and 145 x PWR was employed [developed from BWR-3 and PWR-5 (WASH-1258) gas treatment capability categories, child milk thyroid dose/adult inhalation thyroid dose].

(f) Analysis conditions assumed: lined containment with leak rate of 0.1%/day first 24 hr, 0.05%/day times >24 hr; Gail loop lift-off fractions; reference site (lead plant) meteorology.

(g) Analysis conditions of footnote (f) apply: standard Site GASSAR Fuel Source through 100 in.² hole; DBDA with Gail loop lift-off; instantaneous PCRV transport rate, 1 vol/hr recirculation; no containment deposition.

(h) The Sundesert Nuclear Power Plant was selected as representative of PWR LOCA events. (See Sundesert Nuclear Power Plant PSAR, Table 15.0-8).

For normal operations (annual release) the HTGR-GT limiting airborne radiological impact is whole body exposure from immersion in the gaseous airborne wastes, primarily noble gases. Nevertheless, the HTGR-GT pathway exposure ranges from 8 to 3600 times below the 10CFR50 Appendix I "as low as reasonably achievable" limit of 5 mrem whole body exposure. Additionally, the projected HTGR-GT gaseous dose levels are significantly below LWR and PWR comparable doses for the HTGR-GT intermittent purge design.

For the LWR, the normal operation (annual release) airborne limiting impact is the thyroid exposure from the iodine-cow-milk-infant pathway. The HTGR-GT infant thyroid dose is negligible in comparison to LWR values.

For severe accidents such as the HTGR-GT DBDA, the HTGR-GT maximum hypothetical fission product release (MHFPR), and the PWR loss of coolant accident (LOCA), estimates of the off-site dose levels have been made and are included in Table 5.3.5-1. Both the HTGR-GT and the representative PWR conform to 10CFR100 dose limits; however, the HTGR-GT exhibits significantly more dose margin than does the LWR.

The HTGR-GT doses reported in this section are scaled by power level from HTGR-SC doses.

5.4 STATUS OF DESIGN AND DEVELOPMENT

5.4.1 U.S. Program

Design studies and R&D work on the HTGR-GT plant have been in progress at GA since late 1979 under several complementary programs supported by DOE, GCRA, GA, and participating industrial firms. These studies include:

1. Thermodynamic cycle options.
2. Turbomachinery design.
3. Safety/licensing evaluation.
4. System dynamics analysis.
5. Heat exchanger design.
6. Control system design.
7. Capital cost incentives and tradeoffs.
8. Development program requirements.
9. Waste heat utilization.

Several significant assessments of the technology and the economics of the HTGR-GT have been conducted and documented by U.S. organizations other than GA, beginning in 1975 and extending into 1978. In the fall of 1978, the HTGR-GT program became the lead HTGR development program. The definitive goal set for the program was to establish a strong U.S. HTGR-GT program through international cooperation with Germany and Switzerland. The program was defined in several phases, with a plant startup date scheduled in 1995.

The FY 1979 program objectives were:

1. Evaluate the gas turbine technical concept.
2. Establish the preferred gas turbine plant design features as the basis for further development.
3. Establish the short-term (to FY 1982) and long-term (to FY 1995) program plan.

4. Establish the basis for international cooperation.
5. Establish the development requirements for the program.

Until June of 1979 the major engineering effort related to the HTGR-GT was the three-loop nonintercooled 1200-MW(e) reference design plant. This was to be the basis for comparison with alternate design studies being conducted in the same time frame. These studies included:

1. Parameter studies to select major plant parameters, i.e., reactor outlet temperature, pressure ratio, etc.
2. Intercooled versus nonintercooled cycle studies.
3. Plant major features study, i.e., loop number and loop rating.
4. Maintenance and contamination studies as required to determine fuel type and to address the impact of remote maintenance on component designs.
5. Warm versus conventional liner studies as required to address the feasibility of the high-temperature reactor with helium turbine warm liner concept.

In June 1979, agreement was reached between GA and GCRA with the concurrence of DOE, to shift the reference design from the three-loop 1200-MW(e) plant to a two-loop 800-MW(e) plant.

In the last half of 1979 and the first quarter of 1980, the design and development effort at GA was concentrated in the following areas:

1. Continuation of the conceptual design of the two-loop 800-MW(e) plant.
2. Studies and evaluations to identify critical technical issues that could impact implementation of the program outlined in the fall of 1978.

In the course of conceptual design and design studies, several specific issues arise in the HTGR-GT lead project that are imposed by the direct coupling of the turbomachine with the primary system. Included are the following:

1. With the turbomachine located inside the PCRV, unique licensing issues arise due to severe transients resulting from possible failures such as turbine deblading, compressor deblading, and rotor failures. The present licensing requirements indicate that the required design basis accident for the gas turbine is an earthquake of safe-shutdown magnitude and simultaneous turbomachine rotor failure. A recent turbomachine failure analysis has determined that the gas turbine machine has an expected major

failure probability of 10^{-4} to 10^{-5} failures per reactor year based on an evaluation of the turbomachine experience data. An analysis of all major core internal components has determined that all components except the core supports can withstand the design basis accident and meet the requirement that the probability of exceeding 10CFR100 site boundary dose limits is less than 10^{-7} per reactor year. The rapid deblading associated with a rotor failure imposes such a severe depressurization rate that the loads [in combination with the safe shutdown earthquake (SSE) loads] on the core supports exceed the design rating. It is, therefore, required that the entire core support undergo extensive redesign to meet the new depressurization rate simultaneously with an SSE.

2. The impact on the thermal barrier is considerable due to the large mass flow and potential for acoustically induced vibrations generated by the turbomachine compressor. These excitations also combine with high-frequency high-intensity noise from the gas turbine and introduce random vibrations in ducts and internal structures. Design issues include:
 - a. Possible deterioration of resilient blanket insulation under sustained vibratory excitation.
 - b. Fatigue design of hot ducts and cover plates operating well within the creep range for metallic components.
3. Under steady-state conditions, coolant will infiltrate the interspace occupied by the blanket insulation and pressure will equilibrate across the thermal barrier cover plates. Design of the thermal barrier must accommodate variations in pressure such that damage or dislocation does not occur as a result of pressure imbalances across cover plates resulting from rapid depressurization during a turbine deblading.
4. Extensive efforts have been expended to establish a maintenance approach for the turbomachinery in the direct-cycle plant. The installation, operation, and removal of the turbomachine have been studied in sufficient detail to establish the feasibility of reasonably being able to monitor and maintain a machine that is integrated into the PCR. However, the design of the turbomachine, removal equipment, and maintenance facilities is highly dependent on the degree to which the machine must be decontaminated in order to perform maintenance, inspections, or repairs. This is an issue since the ability to reasonably decontaminate the machine and the cost of improved fuel performance both impact the extent to which remote capability must be provided to maintain the equipment.

Due to the licensing uncertainties and the component redesign required to accommodate the rapid pressure transients associated with turbomachine rotor failure coupled with SSE, the HTGR-GT plant is not deemed

to be a near-term lead project (1995). The projection of design and development requirements results in the first gas turbine plant in approximately year 2003. The longer time allows for redesign and component testing required to overcome problems identified in a recent turbomachine failure analysis.

Deployment of the HTGR-GT is related to the degree of economic gain for dry or wet/dry cooling applications, or conversely, higher efficiency through bottoming cycles while further offering the potential for improved reliability and operability. Additional benefits would accrue from HTGR characteristics involving inherent safety, reduced exposure of operating personnel, and fuel cycle features.

5.4.2 International Program

The HHT project began in 1972 under the joint sponsorship of FRG and Switzerland. The first HHT plant was designed for an electric power rating of 1080 MW. To reduce the risks of the first HHT system and to prove its marketability, it was necessary to subject its most important components to largely original operating conditions. For experimental studies, several minor experimental units were already available within the scope of the HHT project. The Oberhausen helium turbine power plant (EVO) and the Julich high-temperature test facility (HHV) are intended to serve as large-scale pilot plants for the HHT project. These systems will enable conclusive studies on loop elements for the first HHT system under maximum proximity to real scale and operating conditions.

In the mid-1970s, both parties (U.S. and HHT Project) in cooperation performed plant configuration and economic studies with the objective of defining a common plant design based on existing HTGR and gas turbine technology and, at the same time, fulfilling specific requirements in the U.S. and in Europe. The helium-cooled reactor development program in Europe and in the U.S. would provide the technology base for development of the direct-cycle HTGR. Effective international cooperation was achieved. Common design concepts were being evolved, although differing geographical and governmental requirements such as electrical grid frequency, licensing requirements, and climate exerted some divergent influences. The design illustrated the compact primary system components, simplification, provisions for access, flexibility for siting, and adaptability to both dry cooling and wet cooling with higher efficiency and other forms of waste heat utilization, which were objectives throughout. The needs for R&D were identified and basic R&D work was under way in most areas.

From the mid-1970s to the late 1970s, various technical and economic parameters were evaluated by both the German group and the U.S. group in cooperation. The HHT project finally settled on a single primary loop, single turbomachine/generator set, 675-MW(e) plant with a pebble bed core. The design differs from that of the U.S. two-loop 800-MW(e)

plant, but there are still sufficient commonalities to warrant technical cooperation. At this time, both EVO and HHV facilities are completed and conducting essential DV&S work for both the HHT program and the U.S. HTGR-GT program.

For 1980, the HHT project design and development activities are concentrated on the following objectives:

1. Answer the German Utilities criteria questions as completely as possible by the end of March 1980, so that the German Utilities Group, on their part, can present a summary evaluation of the following HHT demonstration plant items:
 - a. Pressure/volume control system.
 - b. Cooling concept for reactor internals.
 - c. Pressure equilibration accidents (i.e., turbomachine failures).
 - d. Decay heat removal concept.
 - e. Hot start, xenon override.
 - f. Fuel element handling.
 - g. Open questions on reserve shutdown system (KLAK).
 - h. Large and small inspections of the gas turbine.
 - i. Decommissioning concept for the LEU cycle.
 - j. Design of the ceramic and metal components.
 - k. Design of the reactor pressure vessel including liner.
 - l. Construction of the vessel closures.
2. Continue the HHT demonstration plant design, which will be presented by the middle of 1980 by the HHT partners in a conclusive report along with an evaluation based on the prepared documents. The report will include the following key points in addition to the utility question indicated above:
 - a. Turbine blade and rotor failures.
 - b. Operation efficiency.
 - c. Wear reflector.
 - d. Warm liner.

- e. Recuperator.
 - f. Pebble bed core versus block core evaluation.
3. The work in the second half of 1980 will be concentrated on specific pebble bed and specific HTR questions. In addition, HHT specific key point questions should be reworked independent of the results from the evaluation provided for the HHT Project.

A decision was recently made to continue the HHT Project on a reduced scale compared to the most recent funding and to put primary emphasis on an HTR process steam generator to be applied to high-pressure Lurgi gasification of coal. The HHT Project will concentrate on the solution of the priority technical issues outlined above.

5.5 Priority Technical Issues

The status of design work on the HTGR-GT concept and the state of supporting methods, materials, engineering and fabrication technologies are such that several technical issues exist that will require an extended effort to characterize their influence on the commercial viability of this system. These issues are beyond the state of current industrial precedent and will require resolution in the course of a Lead Project. Their identification provides an indication of the nature of the technology development program required to support a Lead Project, which is presented in its entirety in Appendix A of this report. These issues also provide an indication of where funds should be expended to minimize technical risk associated with decisions on commitment to the next stage of concept development. Currently, six priority technical issues have been identified and defined sufficiently to warrant programmatic emphasis on their timely resolution:

1. Rapid PCRV pressure equilibration due to hypothetical turbine accidents.
2. Core support graphite stress and oxidation.
3. Fuel element graphite stress analysis uncertainty.
4. High noise levels from the direct cycle gas turbine.
5. Effects of high operating temperature on primary system components.
6. Core region temperature fluctuations.

The status and future program needed to resolve each issue are discussed in sections 5.5.1 through 5.5.6.

As the definition of the HTGR-GT proceeds, additional focus to technology development activities will be achieved. It is likely that additional technical issues which identify significant risk to the progression of the project will emerge as more work is completed. Items such as the development of rotating shaft seals at primary and secondary boundaries; oil ingress to the primary circuit; characterization of pressure and temperature transients and acoustic loadings; and the inspectability of primary components are being evaluated. As such issues and their relevance to programmatic decisions are understood, programs for their resolution will be conducted on the appropriate schedule.

5.5.1 Rapid PCRV Pressure Equilibration Due to Hypothetical Turbine Accidents

- Issue - Turbine deblading and rotor failures result in rapid pressure changes in the core outlet plenum that severely impact the core support structure and the structural components of the thermal barrier in the region of the outlet plenum. The load on the structural components of the thermal barrier results from a pressure differential (ΔP) across the metallic cover plate containing the fibrous insulation. The predicted ΔP depends on the venting characteristics of the fibrous insulation as well as the cover plate itself. Pressure loads combined with an SSE require

substantial redesign of the class C thermal barrier and the core support structure. Uncertainties in the predicted depressurization rate and the pressure-loads due to turbomachine deblading or rotor failure need to be resolved in the course of redesigning these components.

- Program for Resolution - Resolution of uncertainties relative to predicted loads requires testing of the affected components under rapid depressurization conditions. The core support structure must be redesigned to accommodate the pressure loads during a depressurization combined with the SSE loads in accordance with regulatory procedures applied to LWR plants. To reduce the uncertainties of the depressurization and pressure load predictions, turbomachine rotor failure tests must be performed to determine the actual modes of failure and the flow resistances through the turbomachine.

5.5.2 Core Support Graphite Stresses and Oxidation

- Issue - Graphite is a material having excellent high-temperature mechanical and physical properties. However, oxygen-carrying species in the primary coolant must be kept low to limit corrosive degradation of the structural capacity of graphite components.

Designers and the NRC recognize that structural criteria and analytical methods must be developed to adequately account for these effects in the design of graphite core supports. It is necessary to establish a three-dimensional failure theory for graphite, quantify strain rate effects on stress-strain behavior and strength, characterize the effects of oxidation and irradiation on material behavior and strength, investigate time-dependent stress effects on strength, and develop analytical tools to predict the effects of oxidation.

Also, uncertainty in the prediction of seismic loads further complicates the design of the core support, and further code development is required to reduce this uncertainty.

- Program for Resolution - The core support must be designed to make it tolerant to the effects of corrosion. Specifically, the long-life components such as the post, the post seats, and the upper half of the core support block must be made from a low-oxidation-rate, high-strength graphite. Graphites with these properties must be selected and characterized.

Design criteria are being developed by the Joint ACI/ASME committee for incorporation into the ASME Boiler and Pressure Vessel Code, Section III, Division 2. The first draft was submitted to the Main Committee in September 1980. This activity must be pursued until an agreed upon industrial code is established.

Oxidation profile prediction methods must be developed for cylindrical geometries, which can account for rate changes as a function of position, porosity changes, prior oxidation, and temperature-moisture-impurity history. Development of a general finite element approach for arbitrary geometries is also required to evaluate the complex geometries of graphite parts in the core support floor.

Criteria and methods development must be confirmed with an experimental program that is designed to obtain data on graphite oxidation rates and profiles, the effects of properties caused by oxidation, failure theories (both triaxial and fracture mechanics), fatigue data and cumulative damage data, material constitutive behavior studies, and structural model tests for methods verification.

The information outlined above is needed prior to the start of detailed design. However, some of the fundamental tests must be completed earlier to provide a sound basis for formulating preliminary structural criteria so that the design effort can proceed smoothly.

5.5.3 Fuel Element Graphite Stress Analysis Uncertainty

- Issue - Stresses in the fuel element blocks are complex and the structural design criteria are not yet established. Designers believe the analytical techniques currently used to estimate the combined effects of seismic loads and irradiation-induced stresses are adequately conservative. However, the structural design criteria for permanent graphite structures proposed by the NRC's consultant, the Franklin Institute Research Laboratories (FIRL), are more conservative and are considered excessively stringent for application to replaceable fuel and reflector elements which have lives of about four years. If the more conservative criteria recommended by FIRL is imposed by the NRC, adequacy of the present design would become an issue requiring resolution in the licensing process.
- Program for Resolution - Further theoretical and experimental work is required to improve material models and analytical methods. The program to develop improved analytical material models is designed to reconcile discrepancies in the stress calculation models, such as the strain-gradient effect. The capability to calculate and combine dynamic seismic stresses with thermal and irradiation stresses while accounting for the effects of fatigue and changes in materials properties must be developed and experimentally verified.

Design modifications may be necessary to reduce thermal and irradiation-induced stresses. Reductions in calculated mechanical (seismic) loads may be achieved by accounting for plant embedment, specific soil conditions, and damping mechanisms in the core.

5.5.4 Effects of High Noise Levels

- Issue - The location of the turbine and compressor inside the PCRV introduces a high level of acoustic loading to reactor internals. The presently estimated sound pressure levels emitted from the turbomachine may limit the design life of the thermal barrier components, flow guides, and duct assemblies and could result in the requirement for recurring replacement of the affected components during the life of the plant.
- Program for Resolution - A program to develop analytical methods to describe the nature of sound pressure waves impinging upon internal components is required. In addition, the definition of material behavior to high frequency loadings at elevated temperature and the development of design criteria must be undertaken. Behavior of materials and components must be verified under prototypic conditions in an integrated test. A parallel program to reduce noise emission at the turbine must be conducted to provide for an optimal resolution. A Helium Component Test Facility (HCTF) is required to demonstrate that this issue and other conditions arising from complex systems interactions are properly addressed.

5.5.5 Effects of High Operating Temperatures

5.5.5.1 Core Restraint and Peripheral Seal

- Issue - Increased helium temperatures (relative to the design for the HTGR-SC) have required re-evaluation of the metallic components of the core lateral restraint and the core peripheral seal. The existing designs for these components have not been qualified for service above the HTGR-SC conditions. The nickel alloy materials used in the different parts of the spring packs and peripheral seal were near their limit for the HTGR-SC conditions. At higher temperatures, embrittlement due to carburization and aging will require the qualification of a different material and/or design solutions.
- Program for Resolution - A materials properties development program is required to obtain creep rupture, tensile strength, and stress relaxation data at elevated temperatures. Creep rupture and stress relaxation testing of Inconel Alloy 718 will be necessary. Parallel programs to identify alternatives to the existing designs will be evaluated.

5.5.5.2 Thermal Barrier and Hot Ducts

- Issue - The temperature level of the primary coolant in the core outlet region controls the selection of materials for the thermal barrier. The 850°C mixed mean core outlet temperature requires the use of superalloy castings for the structural elements of the thermal barrier assembly (cover plates and attachment fixtures) because the wrought alloys used to date do not retain sufficient strength and carburization resistance at this temperature. However, a comprehensive development program will be required to demonstrate that such cast elements can be made to the desired quality and property criteria to completely satisfy the intended nuclear application. Fabrication development is necessary to demonstrate that the desired structural configurations can be made sufficiently defect-free.
- Program for Resolution - Material evaluation studies have identified Inconel 713LC as the casting alloy candidate. Handbook property data for this alloy indicate acceptable strength and carburization resistance compatible with an 850°C mixed core outlet temperature. Designers must conduct initial discussions with foundries to ascertain the castability of desired component sizes and geometries. A comprehensive data base for the most promising thermal barrier casting materials will be established. A multiyear program will be required to provide basic materials design data in HTGR environments and to subsequently develop suitable design criteria.

5.5.6 Core Region Temperature Fluctuations

- Issue - Fluctuations in the region outlet temperatures have been experienced in the FSV core during reactor operation at approximately 70% of the full power level. The fluctuations, as registered by the region outlet thermometers, are characterized by rapid temperature changes separated by "hold times" of 5 to 20 min. The temperature changes are on the order of 38°C (100°F), and sometimes greater. It is possible that such thermal cycles could cause fatigue damage to the core or steam generators. For this reason, the NRC has not allowed sustained operation of the FSV reactor in a fluctuating mode.

Since the cause of the temperature fluctuations is not entirely understood, design modifications necessary to assure that future cores do not experience this phenomenon are not completely defined. The large HTGR has smaller gaps and less bypass flow than the FSV reactor and thus should be more stable. However, whether the differences are sufficient to preclude fluctuations is not known.

- Program for Resolution - The current hypothesis is that the fluctuations are a thermal-hydraulic phenomenon, where fuel regions are moved by the combined effect of transverse pressure forces and thermal distortions. This conjecture is supported by all the circumstantial evidence but has not yet been proven. Region constraint devices have been installed in the top of the FSV core to prevent fluctuations, and successful operation at power levels up to 70% suggests this approach will resolve the problem for the FSV core configuration. Model tests have shown these devices to work. However, due to other plant operational requirements, testing of the devices may not be resumed until early 1981 to demonstrate the solution at FSV.

The program for resolution of this issue for the HTGR-GT will entail the development of analytical methods to predict core movements and the use of small scale physical models to verify their accuracy. The program must provide for the progressive development of one-, two-, and three-dimensional analytical and physical models of a several year period. Both the codes and models will be used to verify design solutions. This effort is to be coordinated with the activities under way at FSV to resolve the core fluctuation problems.

SECTION 5 REFERENCES

1. Personnel Radiation Exposure in HTGR and LWR Plants, report by General Atomic Company, GA-A15568, September 1979.

6.0 TECHNICAL BASIS FOR FOLLOW-ON/ADVANCED SYSTEMS

6.1 Gas Turbine Follow-On/Advanced Systems

Based on the 850°C core outlet temperature used as the design basis for the lead plant, a series of commercial plants of similar design is expected to be produced without further reactor plant development.

The next step would be to produce combined cycle gas turbine plants in order to utilize the waste heat. Section 3.3 describes in detail such combined cycles using flashing steam or ammonia bottoming cycles. The lead plant or existing commercial units could easily be back-fitted to demonstrate these bottoming cycles. The bottoming cycle equipment will not require large development programs as the equipment used is readily available, with the possible exception of the ammonia turbine for the size necessary.

The development of other applications for the waste heat usage such as district heating, desalination, and agricultural uses will not require additional reactor plant development but development of only the specific application process itself.

Because the gas turbine (Brayton) cycle effectively utilizes higher turbine inlet temperatures (i.e., core outlet temperature) to gain cycle efficiency, the next logical step following the development of the gas turbine plants for the 850°C core outlet temperature would be to develop all components necessary for higher core outlet temperatures. The development would include fuel particles, fuel blocks, core internals, thermal barrier under the core, hot cross ducts, and the turbomachine.

Major facility requirements relate to the high-temperature aspects and are anticipated to include:

1. High-temperature (950°C) helium component test facility (HCTF) to test the hot duct and turbomachine. (This would require modifying the proposed HCTF for component tests at 850°C.)
2. Expanded metallurgical test facilities for new alloys and nonmetallic materials.

Subsequent to material development and component testing, a demonstration plant could be constructed for final testing of the core at the higher operating conditions.

6.2 Other Follow-On Applications

The overall program scenario of the HTGR-GT as the Lead Project with the HTGR-SC or HTGR-R as follow-on projects was developed in Section 2. This section outlines the additional design and development steps for each of the follow-on/advanced options.

Approximately one third of the HTGR-GT design and development programs including the HCTF are classified as generic and pertain to the other HTGR options. Therefore, it will be primarily in the following nongeneric areas where the additional steps will be needed:

1. Systems design and DV&S.
2. Licensing.
3. Safety and reliability.
4. PCRV component design for the specific applications.
5. Heat exchanger design and DV&S.
6. Helium circulators/service systems design and DV&S (where applicable).
7. NSS instrumentation, plant protection system, and plant control system.

Table 6.2-1 indicates the application of the HTGR-GT lead project design and development contribution to the other follow-on applications. In general, the main design differences are associated with the method of power conversion for the gas turbine and the steam cycle applications or reformer options. The gas turbine does not require steam generators, circulators, or intermediate heat exchangers and consequently these components would require development and testing for the steam cycle or reformer options. Conversely, these follow-on options cannot use the technology developed for the gas turbine power conversion loops consisting of the turbomachine, recuperator, precooler, and bypass control valves. The follow-on options could use the HCTF to flow test components at design temperature.

The generic program will supply approximately 60% of the steam cycle or reformer option development, leaving the balance to be completed by the specific follow-on plant application.

TABLE 6.2-1
APPLICATION OF HTGR-GT LEAD PROJECT
DESIGN AND DEVELOPMENT TO FOLLOW-ON PLANTS

Program Area	Follow-on Plant ^(a)	
	HTGR-SC	HTGR-R
Fuel (MEU)	A	A
Core	A	A
Reactor internals	A	A
Neutron and region flow control	A	A
Thermal barrier	A	A
PCRV, liner	A	A
Fuel handling	A	A
Reactor service system	A	A
CACS	A	A
Plant control/protection/instrumentation	B	B
Turbomachine	NA	NA
Recuperator	NA	NA
Precooler	NA	NA
Bypass control valves	NA	NA

(a) A = lead program provides development for follow-on programs.

B = lead program provides minor advancement to follow-on programs.

NA = not applicable.

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7.0 LEAD PROJECT/PROGRAM SCHEDULE

The HTGR-GT requires a significant extension of the state-of-the-art of nuclear power plant technology. The temperature regimes and the complex systems interactions encountered in the nuclear heat source will require the development of new regulatory criteria and advanced industrial capability. A comprehensive program of basic technology development and component demonstrations is planned to satisfy the needs of both sectors. The project and program described for the HTGR-GT in the preceeding sections reflects the current perception of the effort necessary to develop and demonstrate this reactor technology. Schedules, as well as costs, are subject to the developmental nature and attendant uncertainties of the HTGR-GT Lead Project.

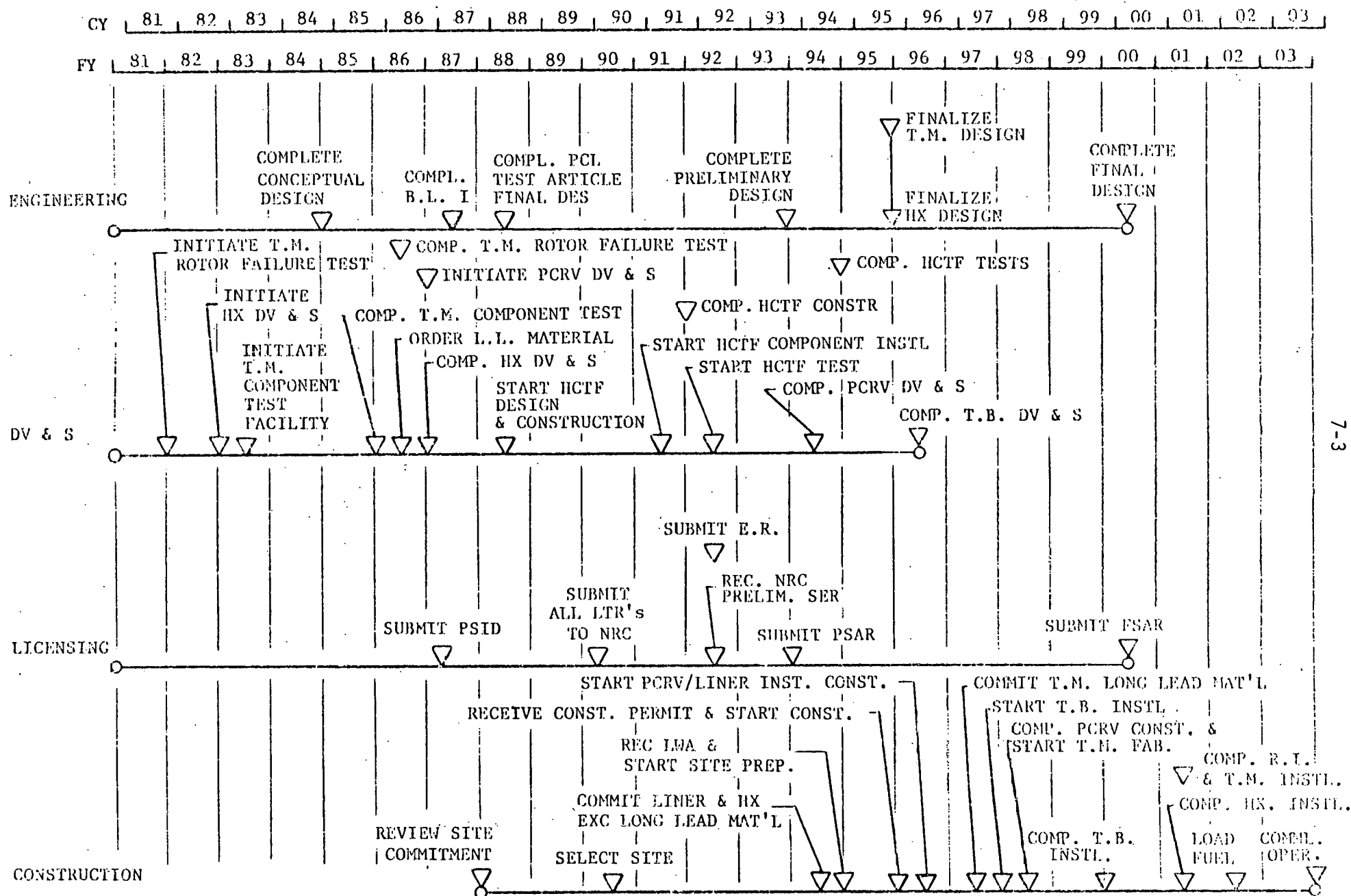
The schedule presented in Figure 7-1 is viewed as a "reasonable target" to design, license and build an HTGR-GT plant. Assuming an aggressive program, the development may be viewed in three phases:

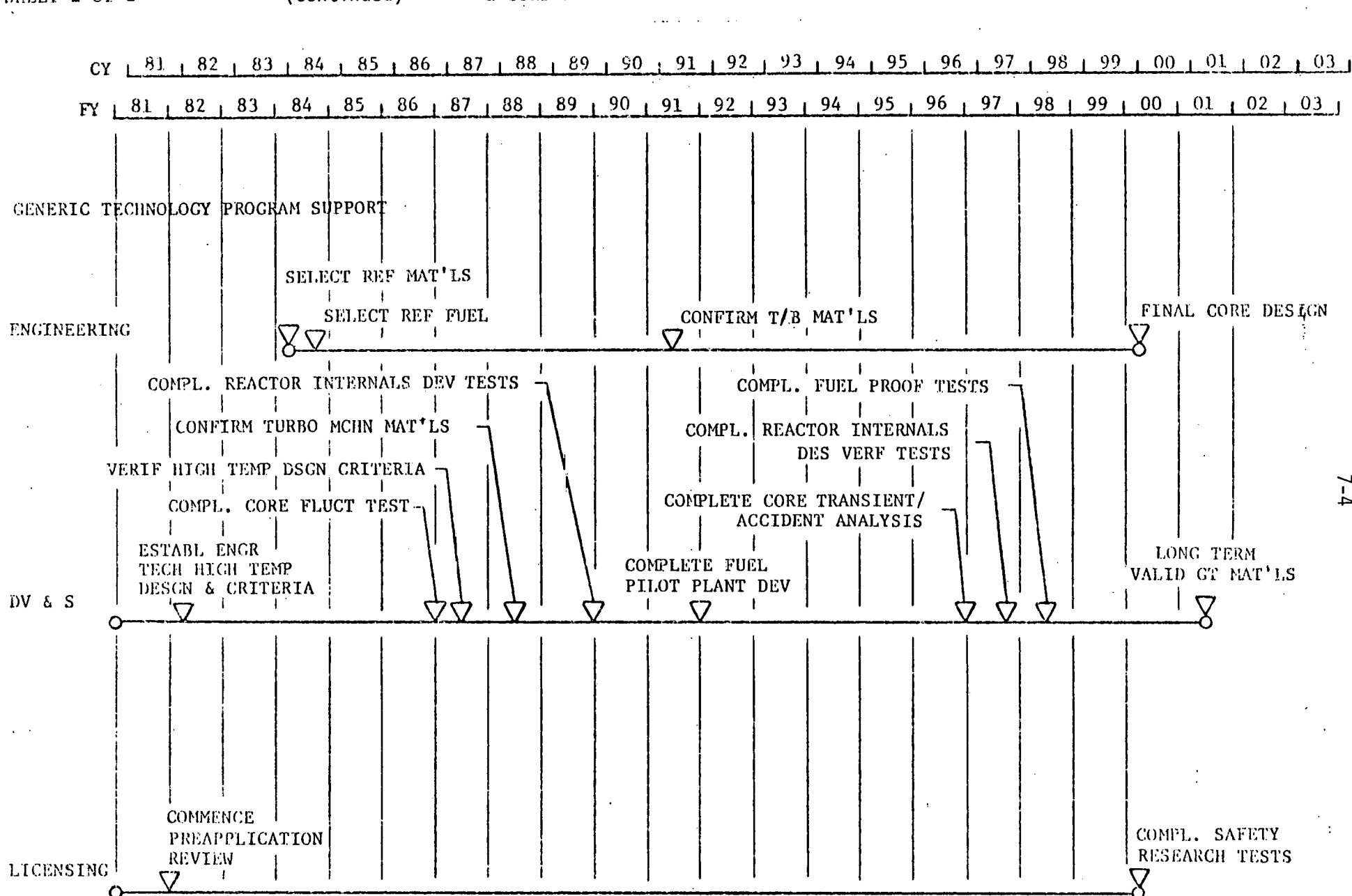
Phase I:	Concept Development	FY 1981-1987
Phase II:	Design Development	FY 1988-1995
Phase III:	Construction and Startup	FY 1996-2003

During Phase I, a period of intensive research and development is required to characterize the dynamic effects of the gas turbine on other elements of the reactor within the primary circuit. A key activity during this period is the performance of tests to establish the behavior and failure modes of turbine rotors which affect design basis events for the gas turbine plant. In parallel activities, development work and tests on turbine bearings and seals will be conducted. Other technology development activities will emphasize materials development and the design of reactor internals for this environment. These activities will culminate in the submission of a Preliminary Safety Information Document (PSID) to the NRC in 1987.

During Phase II and concurrent with early licensing review, the plant will progress through PSAR submittal in 1993 to the receipt of a Construction Permit in mid-1995. A major element of the development program will be conducted in this period. A Helium Component Test Facility (HCTF) will be constructed and a non-nuclear demonstration of prototypic components will be performed. Such tests are considered necessary to refine and confirm methods used to predict the effects of coolant flow, acoustics, and other systems interactions and to exercise a prototype gas turbine through a representative range of operating conditions. It is projected that the early discourse initiated with the NRC via the PSID will facilitate the formal review of the PSAR which is estimated to require 24 months. Completion of these activities will mark the onset of Lead Project construction.

The Construction and Startup period, Phase III, is scheduled with somewhat longer intervals than those for conventional plants to account for the added construction complexity of the nuclear island caused by the direct cycle gas turbine within the PCR. The construction period specified on the schedule is 78 months, which was extrapolated from LWR experience and previous studies on HTGR-SC plants. The period between completion of fuel loading and full power commercial operation is estimated at 18 months. With these assumptions, commercial operation is projected to occur in 2003.





8.0 LEAD PROJECT COST EVALUATION

Projections of costs associated with major elements of the HTGR-GT Lead Project are presented in the following paragraphs. These elements are:

- Technology Development Costs.
- Reactor Turbine System (RTS) Design and Development Costs.
- Lead Plant Costs.

The technical description of the plant is given in Section 5, and the project schedule is presented in Section 7 of this report. Component and technology development programs with their schedules are provided in Appendix A.

Table 8-1 gives further definition of the cost elements indicated above. For the HTGR-GT, the emphasis within Technology Development is on fuels and materials, with nearly three-fourths of the projected \$230M directed toward their qualification. Work on fuels emphasizes the development of an MEU fuel for the temperature regimes of the HTGR-GT. Materials development is also oriented to address the high-temperature requirements of this plant.

RTS Development costs are estimated at \$800M, with heavy emphasis assigned to the development of the turbomachinery. The \$220M entry for turbomachinery includes the effort projected to experimentally establish rotor failure modes and to produce the final design. The \$340M entry for a Helium Component Test Facility (HCTF) provides for an integrated systems test of turbomachinery and other components in the power conversion loop. The HCTF is described in Appendix A.

To determine the Lead Plant costs shown on Table 8-1, a subjective assessment was made to determine the degree of uncertainty that may be encountered in designing and constructing the first HTGR-GT plant. Starting with the Commercial Plant costs presented in Section 3, the following factors were applied to account for first-of-a-kind (FOAK) type engineering and design effects and inexperience with construction techniques unique to the HTGR-GT.

<u>Account</u>	<u>Factor</u>
Structures and Improvements	1.10
Reactor Plant	1.50
Electric Plant	1.10
Miscellaneous Plant	1.05
Heat Rejection System	1.05
Construction Services	1.25
Engineering Services	1.50

TABLE 8-1

HTGR-GT LEAD PROJECT COSTS ('80 \$M)

Technology Development

Fuel	80
Materials	90
Plant Technology	30
Technology Transfer	10
Capital Equipment	20
Total	230

RTS Development

Systems and Design Support	38
Safety-Licensing/GES	25
Reactor Vessel	45
Reactor Internals	20
Reactor Core and Flow Control	30
Turbomachinery	220
Heat Exchangers	35
CACS	15
Miscellaneous Reactor Services	15
Fuel Handling	8
Control/Electrical	9
Helium Component Test Facility	340
Total	800

Lead Plant

Structures and Improvements	149
Reactor Plant Equipment (Includes RTS)	399
Electric Plant Equipment	56
Miscellaneous Plant Equipment	13
Heat Rejection System	29
Total Directs	646
Indirects	296
Contingency	48
Total Lead Plant ⁽¹⁾	990
Project Development ⁽²⁾	20
Total	1010

(1) Excludes owner's costs, interest during construction, and escalation.

(2) Front-end design/tradeoff effort required to define Project.

The RTS cost for the Lead Plant reflects similar judgments by General Atomic Company to account for soft tooling and non-recurring engineering and licensing costs.

Table 8-2 presents a comparison of the Lead and Commercial Plant costs for the HTGR-GT. The FOAK uncertainties produce an estimated cost differential of \$220M. In the interest of presenting major cost elements in a comparative format, other costs including owner's costs, interest during construction, and escalation have not been accounted. However, it should be noted that the greatest risk in FOAK plants lies in unplanned schedule extensions. Prior experience with demonstration plants has shown that schedule risks are particularly evident in the licensing, construction, and start-up phases. Obviously, capital-intensive projects are particularly vulnerable to the effects of interest during construction and escalation, and uncertainties in project schedules may result in exposure to substantially greater economic risk than the \$220M ascribed to FOAK costs.

In Table 8-3, costs of major elements are distributed in accordance with the Project phases described in Section 7. Estimated expenditures through the Concept Development Phase total \$230M and reflect the effort to characterize turbine rotor accident characteristics and develop appropriate bases for reactor design. Peak annual funding requirements arise in the Design Development Phase at \$165M per year corresponding to work on the HCTF. Funding requirements for this interval are \$780M. Expenditures during the Construction and Start-Up Phase are estimated at \$1010M.

Table 8-3 also projects a plausible degree of utility participation. Utility group participation at a modest level may be anticipated throughout the course of the Program. Lead utility financial involvement is not projected until the onset of Phase II with the offering of a site. Total in-progress financial support by the Lead Utility has been projected at a level that is equivalent to the commercial value of the plant. It is noted that this level of financial support by the Lead Utility is projected on the basis of the government's basically underwriting the Gas Turbine Lead Project for risk protection against schedule, technology readiness, plant performance, etc.

It should also be noted that other elements of a total HTGR Program are not included in Table 8-3. Further work on follow-on process heat and advanced gas turbine systems would be required, as would continued low-level effort toward fuel recycle.

TABLE 8-2
 HTGR-GT LEAD AND COMMERCIAL PLANT BASE COSTS
 ('80 \$M)

Power Rating [MW(t)]	2000	
Net Electrical Output [MW(e)]	800	
Efficiency (%)	40	
	<u>Commercial</u>	<u>Lead</u>
	<u>Plant</u>	<u>Plant</u>
RTS Direct Costs	180	240
RTS Indirect Costs	<u>15</u>	<u>65</u>
Total RTS	195	305
BOP Direct Costs	330	341
BOP Indirect Costs	210	296
Contingency	<u>35</u>	<u>48</u>
Total BOP	575	685
Total Plant Base Costs	770	990

TABLE 8-3

HTGR-GT LEAD PROJECT COST PROJECTIONS ('80 \$M)

COST ELEMENTS	CONCEPT DEVELOPMENT							DESIGN DEVELOPMENT	CONSTR. & START-UP	TOTALS
	81	82	83	84	85	86	87	88-95	96-2003	
<u>Technology Development</u>										
DOE	15.3	15.7	16.0	16.0	16.0	16.0	15.0	90	30	230
<u>RTS Component Development</u>										
DOE	10.0	10.0	12.0	14.0	16.0	18.0	20.0	630	70	800
<u>Lead Plant</u>										
DOE	1.0	1.0	1.0	1.5	1.5	2.0	2.0	25	85	120
GCRA/EPRI	1.0	1.0	1.0	1.5	1.5	2.0	2.0	25	85	120
Lead Utility	-	-	-	-	-	-	-	20	750	770
Subtotal	2.0	2.0	2.0	3.0	3.0	4.0	4.0	70	920	1010
<u>Total Lead Project</u>										
DOE	26.0	26.0	28.0	30.5	32.5	35.0	36.0	745	185	1150
Utility	1.3	1.7	2.0	2.5	2.5	3.0	3.0	45	835	890
Subtotal	27.3	27.7	30.0	33.0	35.0	38.0	39.0	790	1020	2040

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9.0 RECOMMENDED APPROACH TO HTGR-GT DEVELOPMENT

Major considerations on the appropriate course for continued HTGR-GT development may be drawn from the preceding sections of this report.

- In Sections 3 and 4, no compelling economic or other advantages relative to competing systems were identified. It was noted, however, that markets may emerge if future institutional factors encourage the utilization of the high-quality reject heat of the HTGR-GT via district heating or similar applications. Thus, the development of HTGR-GT remains a worthy long-term Program objective.
- In Sections 5 and 7, it was noted that technical development of the HTGR-GT would require a period of initial work to establish design bases for loading conditions imposed by the presence of the direct cycle gas turbine within the primary system. It was further projected that an integrated test of components would be required to verify that interactions were accounted for in design. The time required for these activities would extend the date for plant operation to 2003.
- In Section 8, projections indicate that development costs and Lead Plant costs will both be in the vicinity of \$1000M. In addition, peak funding requirements are expected to be in the vicinity of \$165M per year to support component development.

The apparent lack of near-term incentive, prolonged schedule, and high cost for deployment combine to support a recommendation that the HTGR-GT not be pursued as a Lead Project. Further, the nature of the technical issues confronted in this plant suggests that a prudent development path would cast the direct cycle gas turbine concept in a second generation of gas-cooled reactors. As a follow-on system, features that limit the HTGR-GT would evolve through systems with less complex loading conditions and temperature regimes than those encountered in the HTGR-GT.

With the projected constraints on funding of gas-cooled reactor technology, it is recommended that minimal effort in the near term be expended to advance the HTGR-GT. The following areas should be emphasized in future work:

- Future activities should emphasize the resolution of priority technical issues that are common to the HTGR-GT and applications with more immediate Project potential.
- Cogeneration and process heat studies should characterize the role of the HTGR-GT in expected future markets where waste heat can be effectively utilized.
- Studies should continue to investigate areas of potential cost reductions.

APPENDIX A

PROJECT DESIGN AND DEVELOPMENT PROGRAM

A.0 PROJECT DESIGN AND DEVELOPMENT PROGRAM

This appendix describes the design and development activities required to support the HTGR-GT program. Included are a specific program (Section A.1) devoted to tasks related only to the HTGR-GT plant and a generic technology program (Section A.2) that is associated with development requirements common to most HTGRs, such as those pertaining to the core, nuclear fuel, fuel handling equipment, reactor internal components, and CACS components.

A.1 HTGR-GT Specific Program

A variety of routine conventional design/development activities are required to proceed from the current preconceptual design status of the direct-cycle HTGR-GT plant through conceptual, preliminary, and final design and eventually to construction, licensing, testing, and startup of the plant. By and large, these include standard, but highly necessary, design approaches used for any construction project. For a nuclear reactor, which must be built with maximum concern for the health and safety of the public, designs and analyses are carried out leading to the generation of applicable word documents, drawings, computer programs, and design reports, not only to define the specific component or system but also to substantiate, validate, and provide traceability of its design. In many cases, commercially available components are used "as-is" (i.e., "off-the-shelf") or modified as required, and often special equipment or a system is developed through subcontract by suppliers knowledgeable in the field. In addition, DV&S tests are carried out to optimize and proof test the designs, and these results are incorporated into final designs. Also, ongoing programs must be maintained to review, update, and approve these designs and to document changes to assure the quality and reliability of the end product for its intended application.

In the course of these efforts, the following activities are normally completed:

1. Preparation of word documents, such as technical descriptions, system descriptions, plant and component specifications, test plans, test specifications, test procedures, and operating and service manuals.
2. Preparation of sketches, drawings, layouts, process flow diagrams, piping and instrumentation diagrams, instrumentation and wiring schematics, and detailed component engineering drawings.
3. Preparation of other special reports, such as general, design (stress or overpressure), design calculations, general test data, test status, design status, test evaluation, quality assurance data, safety analysis, and LTRs.

4. Safety, reliability, and availability analyses to support the design effort, identify and assess specific risks to the plant, develop and confirm analytical models and data (including experimental studies) used for these safety assessments, and provide input to the PSAR and FSAR.
5. Licensing support and guidance to interpret regulatory requirements and nuclear safety criteria, establish the strategies for compliance with these regulations and standards, review plant designs for adherence to these regulatory needs, and provide necessary documentation to state and federal agencies.
6. Systems engineering analyses to optimize plant parameters based on cost and performance goals; establish parameter bases for system/component designs so that the plant can meet its specified performance criteria; define the primary system and CACS requirements including their interfaces with associated systems; define the transient design bases by performing dynamic simulations for alternate operating modes; and develop requirements for plant operating control and protection systems. Further efforts include identifying and resolving technical, safety, and design issues; assuring design control of top level design documents; and establishing and maintaining requirements for RTS/BOP interfaces, plant layout, ISI, and maintenance.
7. Technical support of project management, licensing, experimental engineering, procurement, and cost development.
8. Liaison with architect-engineers, suppliers, utility customers, testing subcontractors, and regulatory agencies.
9. Project management and support to provide overall project management of design and development of the HTGR-GT lead plant program. This includes coordination of the program technical aspects; design reviews; technical status reporting; planning criteria definitions; design basis definition; coordination of program needs to support licensing issues; development, updating, and issuance of overall program plans and schedules including detailed system and component construction schedules; and development, maintenance, and control of the lead project plant engineering data base, document configuration system, RTS equipment list, etc.
10. Quality Assurance will be provided to assure identification, implementation, and documentation of technical Quality Assurance requirements, design reviews, supplier evaluations, planning and inspection, and Quality Assurance audits and corrective actions. Other Quality Assurance functions include documentation to support licensing; implementation of regulatory guides, codes, and standards; and interfacing with DOE, NRC, and other agencies to ensure acceptability and qualification of the Quality Assurance program.

11. Development and evaluation of all cost data in support of the preliminary and detailed cost and risk evaluations for the lead plant.

A.1.1 Functional Work Areas (RTS Design)

The developmental design requirements for the HTGR-GT specific program are carried out by many functional work areas (organizations), as follows:

1. Licensing.
2. Safety, reliability, and availability.
3. Systems engineering.
4. Turbomachinery and generator.
5. Rotating machinery auxiliary systems.
6. Reactor vessel components (including PCRV liners, penetrations, closures, and thermal barrier).
7. Reactor internals (hot duct only).
8. Heat exchangers.
9. Plant protection.
10. Plant control.
11. Plant data acquisition and processing system (DAP).
12. Analytical instrumentation.
13. Reactor turbine system support (including structural mechanics and seismic designs, shielding design and analysis, miscellaneous control and auxiliaries, and materials engineering design).
14. Project management and support.

A.1.2 Special Development Activities

Four of the large development areas in the HTGR-GT specific program that do not fall into the routine design requirement categories or entail extensive development with subcontractors are the PCRV model test, precooler and recuperator heat exchangers, turbomachine, and Helium Component Test Facility (HCTF). The design/development and testing associated with these are described below.

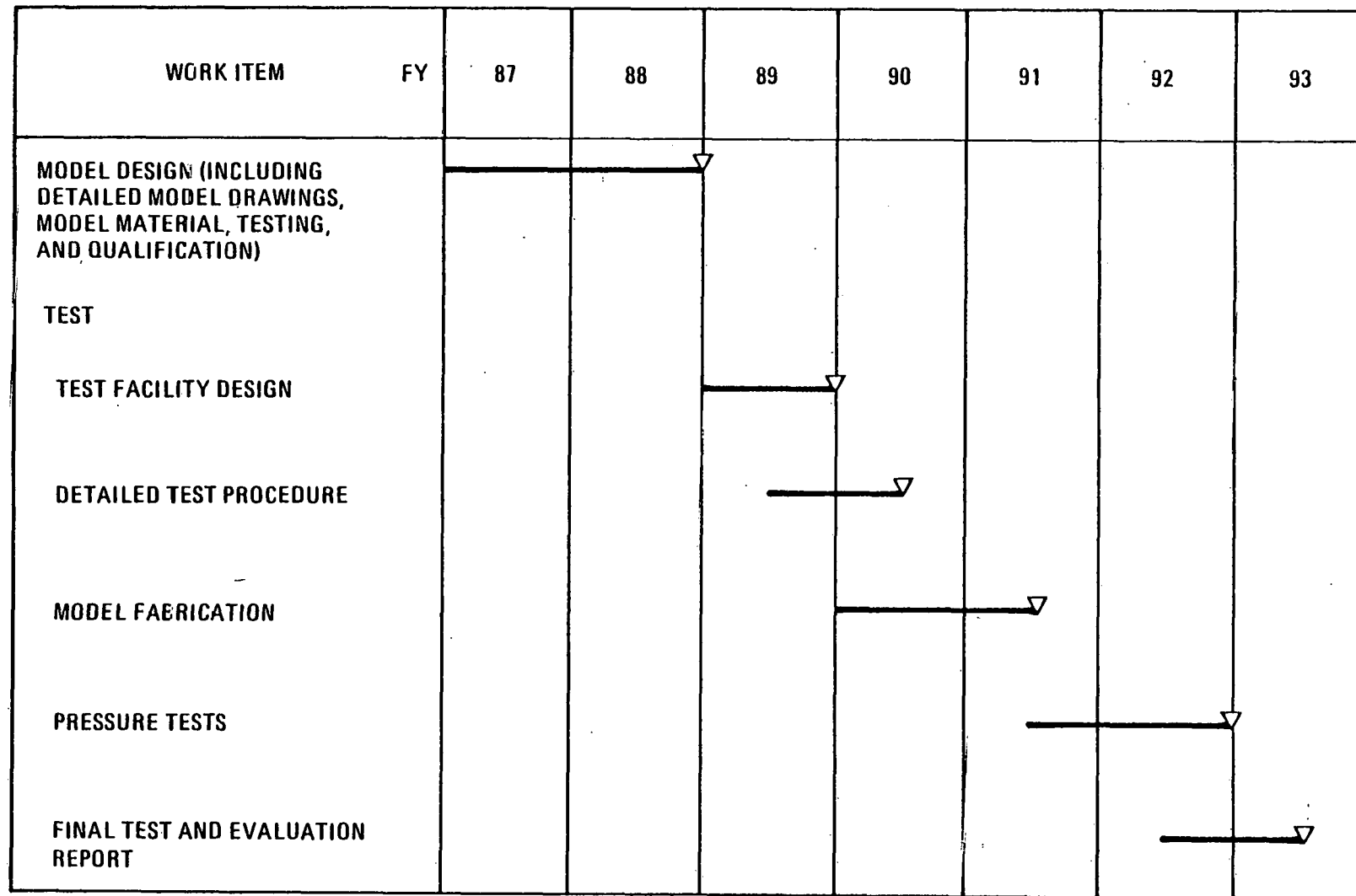
• PCRV Model Test

- Scope - The DV&S program encompasses the testing of a scale model of the PCRV. The model will be constructed of pre-stressed concrete with representative cavities, penetrations, and closures and will embody the structural characteristics of a PCRV with horizontal turbomachine cavities and an asymmetric arrangement of major components. The PCRV model scale will be selected consistent with recommendations of the ASME Boiler and Pressure Vessel Code.

The scope of this task includes model design; material testing and qualification; preparation of model drawings, specifications, and test procedures; and finally model testing. It also encompasses liaison activity with the subcontractor

through the design and testing phases. The model test program will be designed to demonstrate the overall structural response of the asymmetric PCRV to overpressure, which meets the ultimate structural capacity criteria of twice the maximum cavity pressure (MCP), and to identify the most probable ultimate failure mode.

- Major Activities (Fig. A.1.2-1) - The principal effort involves the development of PCRV model design, drawings, test specifications, and procedures and supporting structural analyses for correlation with model test data. Detailed model drawings will be generated to provide sufficient information for material procurement and qualification, test facility design, model fabrication, and construction subcontracts and to establish instrumentation and logging system requirements. Special attention will be directed to the design of model penetrations and closure devices in such a manner that the overall behavior of the prototype under the overpressure condition is accurately represented. Model test data collected during pressure tests will be evaluated and correlated with analytical predictions for structural performance assessments. Test data and evaluations will be documented in compliance with ASME Code and Regulatory Guide requirements.
- Status - Previous experiences in design and testing of single cavity and multicavity PCRV models in support of FSV and LHTGR designs provide the necessary expertise for the HTGR-GT PCRV model program. The PCRV model program definition completed for the reference 900-MW(e) HTGR-SC plant will serve as a good indication of the scope of work involved. Because of the horizontal turbomachine cavities incorporated in the direct-cycle HTGR-GT PCRV configuration and their associated linear tendon prestressing schemes, additional efforts in model design and fabrication are anticipated.
- Recuperator and Precooler Design and Development
 - Scope - The scope of work consists of establishing and executing a program that provides for the design, development, and manufacture of recuperator and precooler heat exchangers and associated auxiliary equipment for an HTGR-GT plant. The program includes accomplishment of all the activities required to meet this objective beginning with the conceptual design of the components and extending through technical support during manufacture, shipment, installation, startup, and plant acceptance by the owner. The program provides for subcontracting the recuperator and precooler heat exchanger final design and manufacture to a heat exchanger supplier and therefore includes preparation of a bid package for potential suppliers' quotation. Also included in the program are technology exchange, technical review, and coordination between the selected supplier(s) and the RTS supplier.



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Figure A.1.2-1 Schedule for PCRv model test program

- Major Activities (Fig. A.1.2-2) - The emphasis in the early part of the program will be on the engineering design and development work required for formulation of viable designs of the components. Tests will be performed and heat exchanger methods (codes) will be developed such that basic technology data common to HTGR heat exchanger designs will be made available for application to specific designs. This initial program phase will include conceptual sizing, performance and thermal analyses, mechanical design, and high-temperature structural analysis associated with steady-state and transient operation. Systems and physical interfaces will be addressed, as well as cost and preparation of the design specification for the heat exchanger supplier bid package(s).

Tests to be performed will include flow distribution, high-temperature materials design data, heat transfer, fretting and wear, vibrations, seismic, and acoustics; methods associated high-temperature heat exchanger design, sizing, performance, and structural analysis will be developed.

The effort in succeeding phases of the program will be on the completion of the detailed design of the components, manufacturing-related support, transportation, installation, and other site support. Typical documentation produced will be stress reports; heat transfer fluid flow and performance reports; mechanical design reports; maintenance, installation, and removal/replacement reports; material service reports; and systems descriptions, design specifications, and inputs to SARs.

- Status - Reference recuperator and precooler configurations and associated critical design areas have been identified. Preliminary steady-state thermal sizing of the concepts has been completed, and this together with some mechanical design and structural analysis has, in the case of the precooler, demonstrated feasibility and established the current envelope. The current recuperator concept is relatively new and still requires design, analysis, and tests to confirm feasibility in the areas of flow distribution, tube-to-tube thermal mismatch, and hot seal design.

A draft development plan has been prepared, which summarizes all the test and methods development programs. This plan provides the essential information for the individual programs that are required to obtain basic technology data or to verify the designs. The bulk of this test and methods work has not been started.

Integrated structural and heat transfer analysis remains to be done, as well as the evaluation of structural interactions among heat exchanger components.

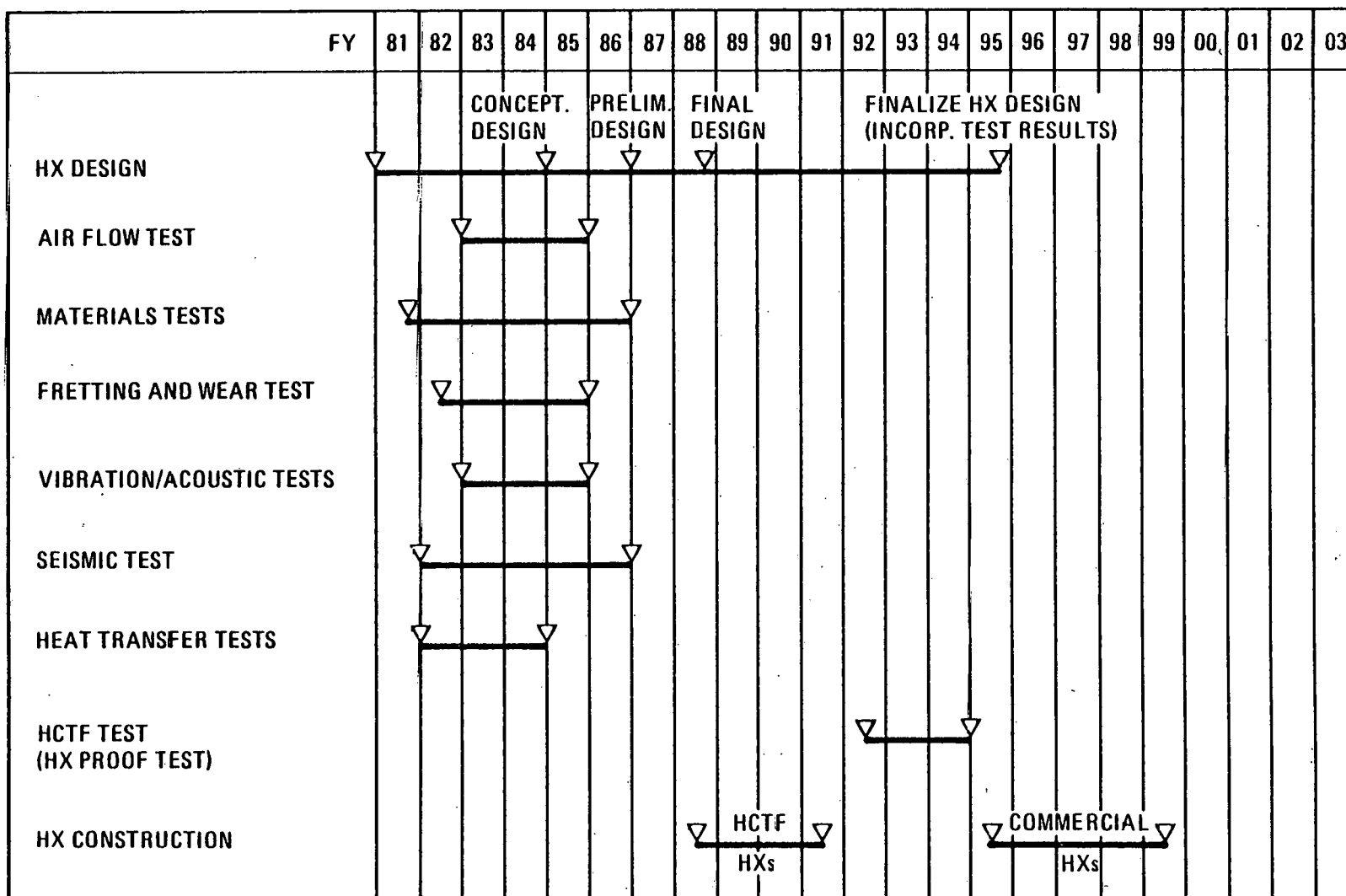


Figure A.1.2-2 Schedule for recuperator and precooler heat exchanger program

Detailed mechanical design is still required, particularly in the areas of tube supports, expansion loops, main heat exchanger supports, and seals. In addition, maintenance, ISI, material surveillance, shipping/handling, and installation criteria have yet to be assessed.

- Turbomachinery Design and Development

- Scope - The development of the turbomachinery by the turbomachinery subcontractor represents a major design and DV&S program. It does not, however, present any unique problems requiring development of new technology since steam and open-cycle gas turbines of a comparable physical size have been manufactured and operated commercially. In addition, open-cycle gas turbines in commercial operation have been operating at temperatures considerably above those found in the HTGR-GT application. The operation of a gas turbine at a rating in excess of 160,00 hp per stage (1.3 million hp total) is, however, unique. In addition, the requirement to establish the sequence and timing of a possible rotor failure is unique.

Design, DV&S, and supporting test activities are required to develop the turbomachinery and its support auxiliary systems. The major areas addressed are:

1. Major test facility/rig design and fabrication.
2. Turbomachine mechanical design and test.
3. Turbomachine aerodynamic design and test.
4. Seal and buffer system design and test.
5. Primary and secondary seal system design and test.

The culmination of the turbomachinery development testing will be in the HCTF, where it will be run at full power and under transient conditions.

- Major Activities (Fig. A.1.2-3) - In support of the design tasks associated with the turbomachinery, a series of extensive component tests are planned. For these tests, the turbomachinery subcontractor would design and build a test facility that would provide a large-volume, temperature-controlled helium stream. The facility would provide flows to 264 kg/s (120 lb/s) or 1/10 rated power conversion loop flow. This flow would be sufficient to test 1/10-scale compressors and turbines. These rigs would be run in time sequence to provide economy in the helium supply system. Performance mapping of these components requires relatively short duration tests that lend themselves to sequencing.

Bearing and seal tests would be both performance and life characteristics tests. Because of the small quantities of helium required for buffering and for simulating main loop

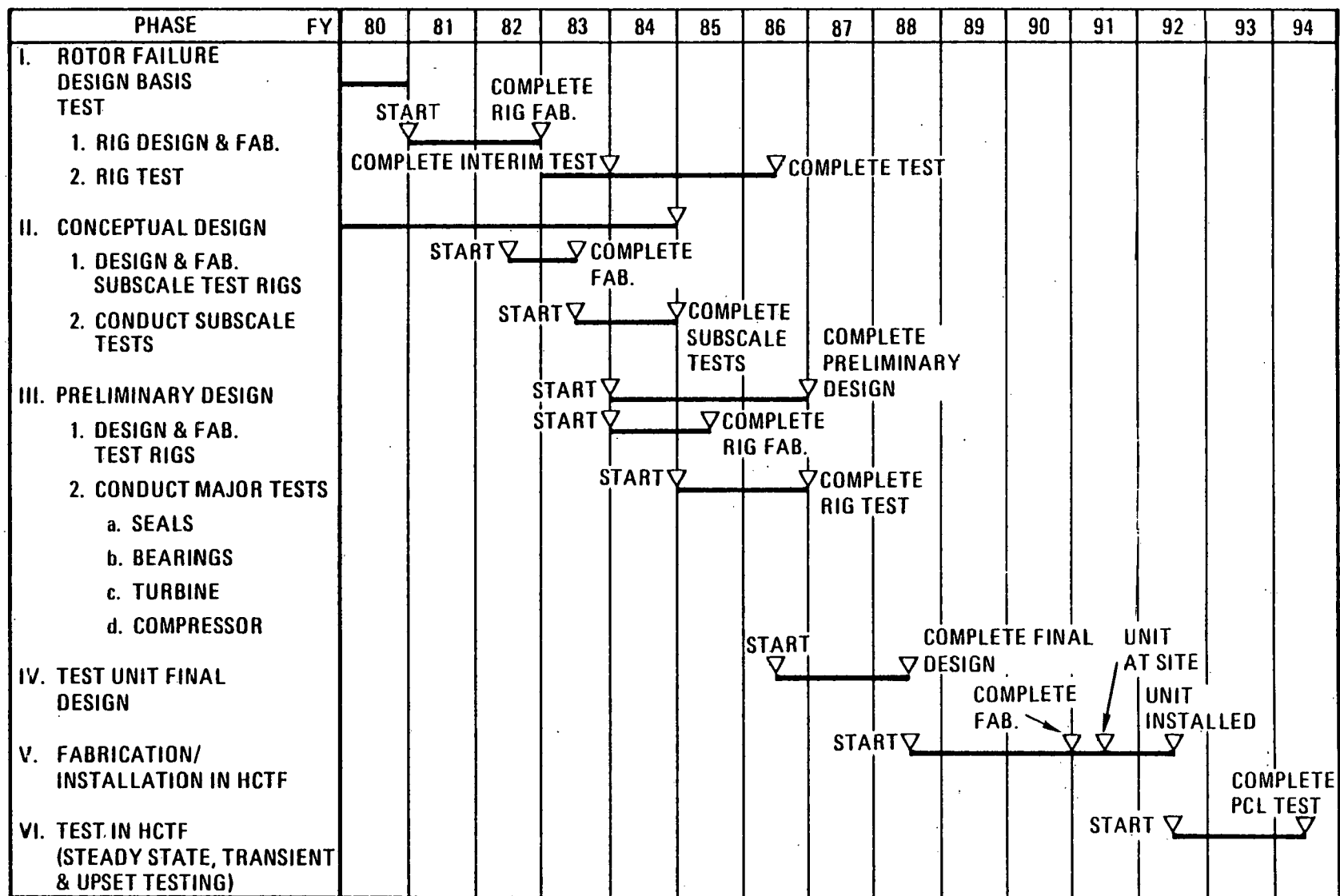


Figure A.1.2-3 Turbomachinery development schedule

pressure variations (a few pounds per second), these tests would be run in parallel with the compressor and turbine tests.

The tests to be included are:

1. Rotor failure mechanism test.
2. Bearing and seal tests.
3. Bearing oil system test.
4. Bearing seal and buffer system test.
5. Primary and secondary shaft seal system test.
6. Compressor performance test.
7. Turbine performance test.
8. Materials compatibility test.
9. Containment ring test.
10. Flow distribution test.

Descriptions of some of these tests are given below:

1. Rotor Failure Mechanism Test - The present basis for a turbine rotor failure sequence is the postulated shearing of the shaft between the turbine and compressor followed by loss of flow path resistance and geometry due to deblading of the turbine and/or compressor in one revolution. This assumption results in high-pressure loads across major components due to rapid pressure equilibration of the PCRV core cavity and the other cavities.

A credible design basis rotor failure must be defined to permit proper component redesign; it is planned to accomplish this by carrying out mechanical and aerodynamic tests. The rotor failure mode for the worst credible failure will be established first by mechanical testing and modeling of centrifugal and thermal stresses as realistically as possible. In the second phase, the depressurization rate in the ducts and plenums will be determined in aerodynamic tests based on the worst credible mechanical failure identified in the first test series; the ducts and plenums would be included in the low-temperature, low-pressure closed-loop model that will be used. The resulting depressurization transient will form the basis for the design of primary system components.

2. Bearing and Seal Testing - The bearing and seal configurations in the HTGR turbomachine represent designs that have been successfully operated in other applications. In the HTGR application, there is a stringent oil leakage limit of $0.27 \text{ m}^3/\text{yr}$ ($1 \text{ ft}^3/\text{yr}$) due to the potentially harmful impact on core and heat exchanger operation. In addition, the bearing diameters are large and the DN factor (surface speed) is high. The generator

drive shaft seal is a safety class component because it seals a rotating penetration through the PCRVR and prevents contamination of the secondary containment building.

For these reasons, extensive testing is planned to optimize bearing and seal configurations and flows and to provide verification of the integrity of the designs. The rigs will be operated at design conditions and extreme transients and off-design conditions to assure that adequate margins are available. After the completion of tests that verify the designs meet all performance parameters, endurance testing will be done to identify and eliminate any random or wearout failure modes.

The two main shaft bearings are tilted pad journal bearings. Since the main support bearings are identical to each other, one journal bearing rig is planned. This rig will also include the double buffered, labyrinth seal arrangement and the lubricating and helium cleanup system.

The thrust bearing is a double-acting tilted pad design. It is designed to handle the rotor thrust plus loads associated with an earthquake shock up to 6.7 on the Richter scale. It is located at the compressor end of the turbomachine near the PCRVR shaft penetration. Its lubricating system is located in the secondary containment building to minimize oil ingress problems.

The PCRVR penetration seal is a multiple piston ring floating oil lubricated seal. It incorporates a hydraulically operated shutdown face seal and is a safety-class component.

It is planned that the bearing and seal rigs would share common drives for further economy. Although the thrust bearing rig does not require a helium supply, it would be run along with the other bearing and seal rigs in order to share common resources of equipment, instrumentation, techniques, and engineering coverage.

3. Compressor and Turbine Performance Tests - The turbomachinery computer model gas dynamic analysis for compressor and turbine design accurately describes the required gas path configuration. Although experience on numerous power plants designed by use of this computer program has shown excellent correlation between calculated and actual results, compressor and turbine performance tests are included in the HTGR-GT program. Compressor and turbine efficiency and compressor surge margin are critical to the success of the installation.

Also, although design with helium as a working fluid poses no serious concerns, actual experience is very limited and design assumptions should be proven.

Gas dynamic design considerations readily yield to scaling effects. For economic and practical considerations, 1/10-scale helium compressor and turbine rigs would be built and run to confirm design efficiencies and compressor surge margin.

Compressor and turbine performance mapping tests will be run so that pressure ratios versus corrected flow can be plotted for various rotor speeds. By measuring these parameters in test rigs, as opposed to engine operation, performance limits of the normal operating line can be evaluated.

4. Rotor Stress Tests - Stresses in rotors do not scale well. Some information on stress concentration effects is obtainable by scaled rig tests, but the expense is not justified. Therefore, a spin test program on critical full-scale disks and rotor segments will be run to verify the design margins. The disks and rotor sections will be run in a vacuum to minimize drive costs and to allow sufficient overstress conditions.

Containment of rotor failures is a plant safety consideration. Both compressor and turbine sections are shrouded by heavy-wall containment rings designed to prevent blade, disk, or rotor segment failures from making excursions to the cavity liner or other parts of the installation. Containment rings will be tested as part of the blade, disk, and rotor segment spin tests to failure. These tests are normally conducted in pits with the shaft in a vertical orientation. Data from these tests will be critical for licensing considerations.

A rig is planned for balancing complete rotors. It will simulate the dynamic environment including the three main shaft bearings and their support cases. The rotor will be shrouded so that a vacuum can be drawn to reduce drive motor capacity requirements to a practical level. Vibration tests will be run after balancing to verify design analysis of critical speed mode and vibration energy levels.

- Status - The reference design of the turbomachinery and its associated auxiliary lubrication oil and seal/buffer systems has been identified. Preliminary steady-state and transient

performance has also been developed together with mechanical design and aerodynamic analysis to establish design feasibility.

Critical developmental areas have been identified. These include flow distribution, establishment of a design basis rotor failure, bearing and seal development, and containment ring design.

A preliminary development plan has been prepared by the turbomachinery subcontractor defining all major design and development tasks. The test rigs associated with the turbomachinery are extensive, with an estimated cost of \$50 million. This is independent of the HCTF, which will provide for the final full-power integrated test of the turbomachinery in the power conversion loop.

The design, as presently envisioned, benefits greatly from designs and technology developed for existing commercial steam and open-cycle gas turbines and the closed-cycle gas turbine experience of approximately one million hours of operation.

- Helium Component Test Facility

- Scope - The HCTF is a major fossil-fired test facility capable of full-power testing of the integrated gas turbine power conversion loop (Figs. A.1.2-4 and A.1.2-5). The major components to be tested in this facility include:

1. Turbomachinery.
2. Turbomachinery auxiliary systems.
3. Generator.
4. Generator auxiliary systems.
5. Control valves.
6. Control valve auxiliary systems.
7. Plant control system.
8. Plant protective system.
9. Hot duct.
10. Thermal barrier system.

Some of the main facility systems/components to be designed to support these test components include:

1. Helium heater.
2. Power conversion loop pressure vessel.
3. Main and auxiliary heat rejection systems.
4. Electrical distribution system.

The major tests to be conducted in the HCTF include startup, shutdown, transient and upset response, and performance testing.

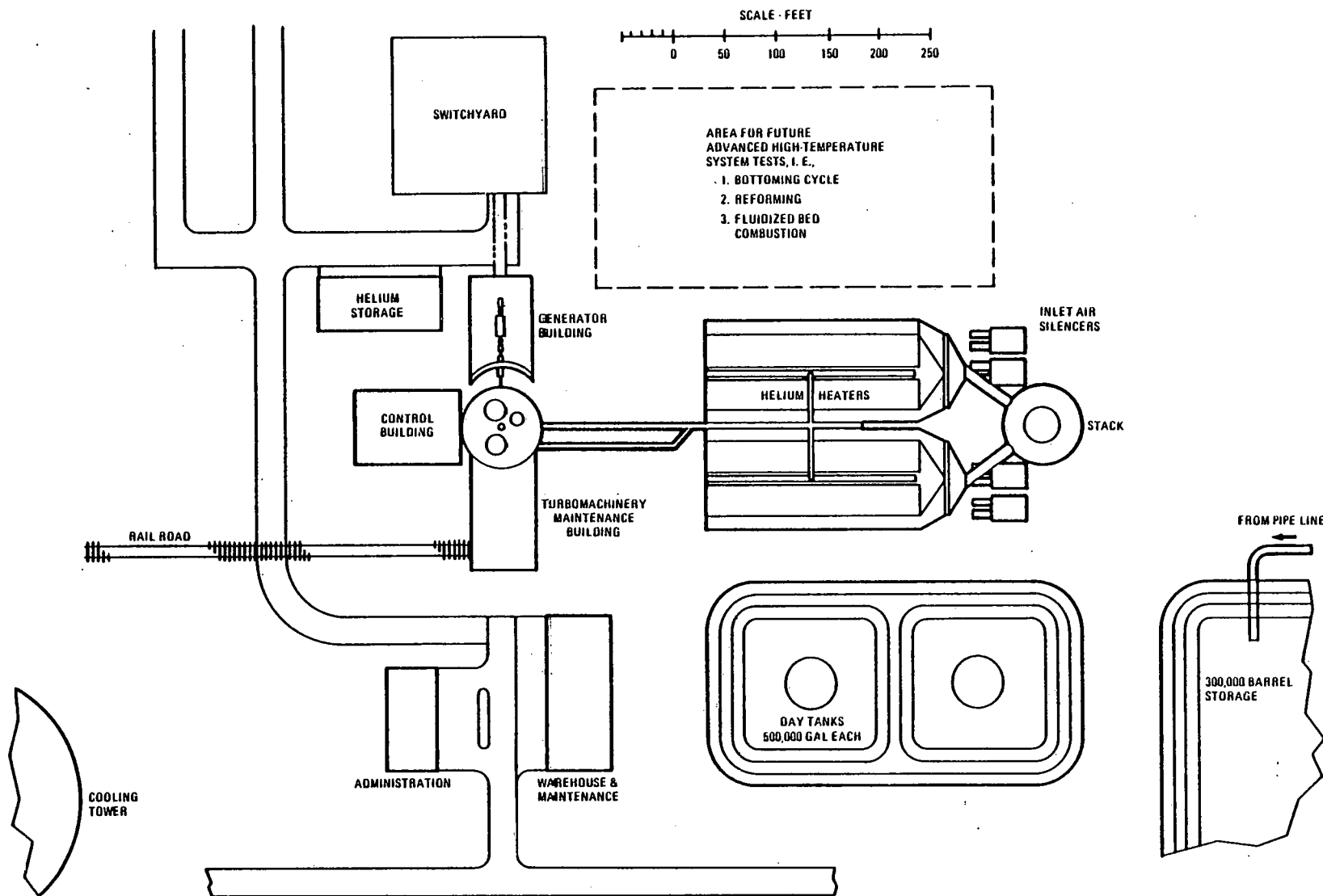


Figure A.1.2-4 HCTF plant layout

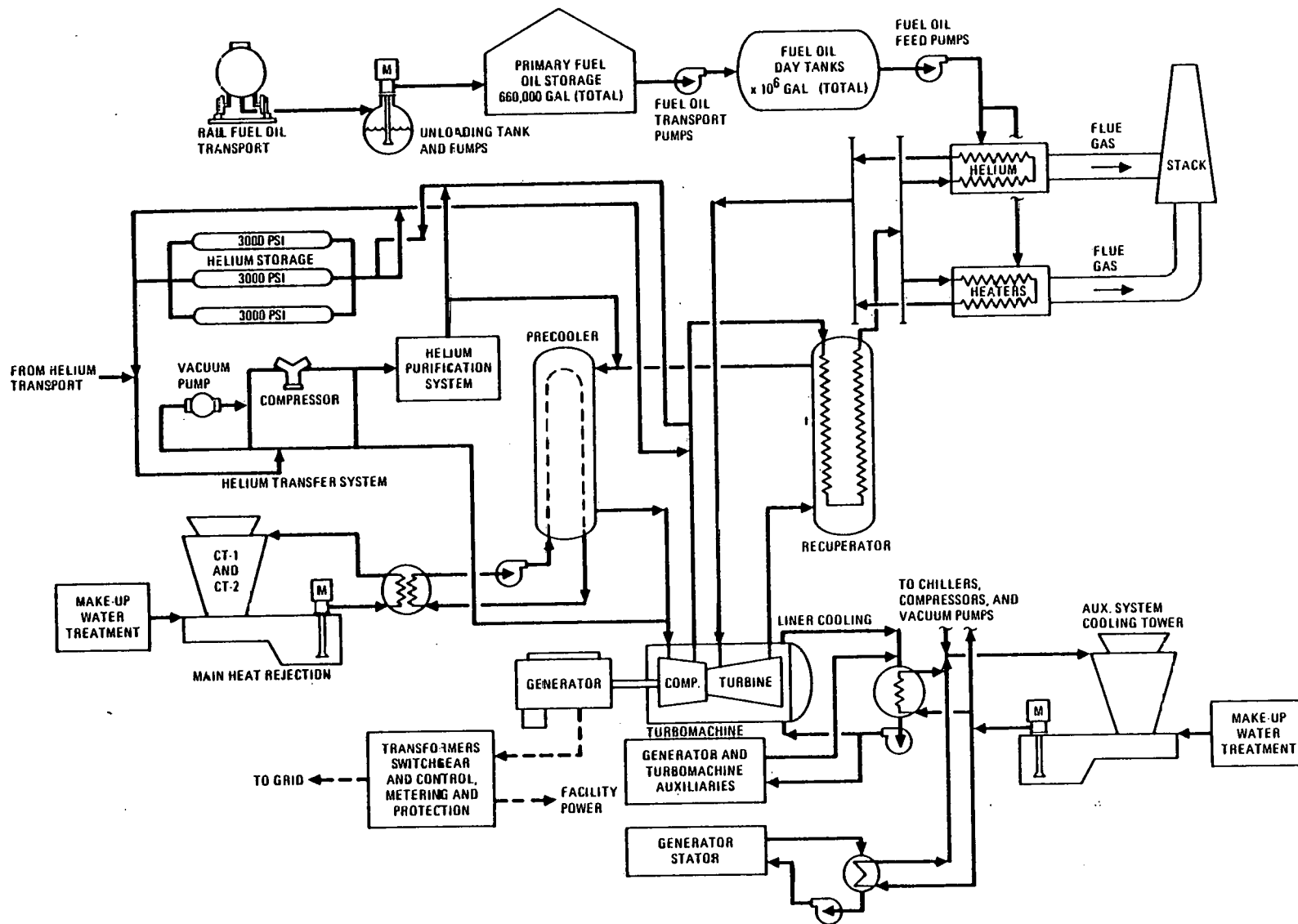


Figure A.1.2-5 HCTF flow diagram

- Major Activities (Fig. A.1.2-6) - The activities are divided into four phases:

Phase I. Establish facility requirements to most efficiently demonstrate the power conversion loop under all postulated operating conditions short of deliberate failure.

Phase II. Design the HCTF. The major task associated with this effort is the development of the high-temperature fossil-fired helium heater.

Phase III. Construction of the HCTF and fabrication of the test components.

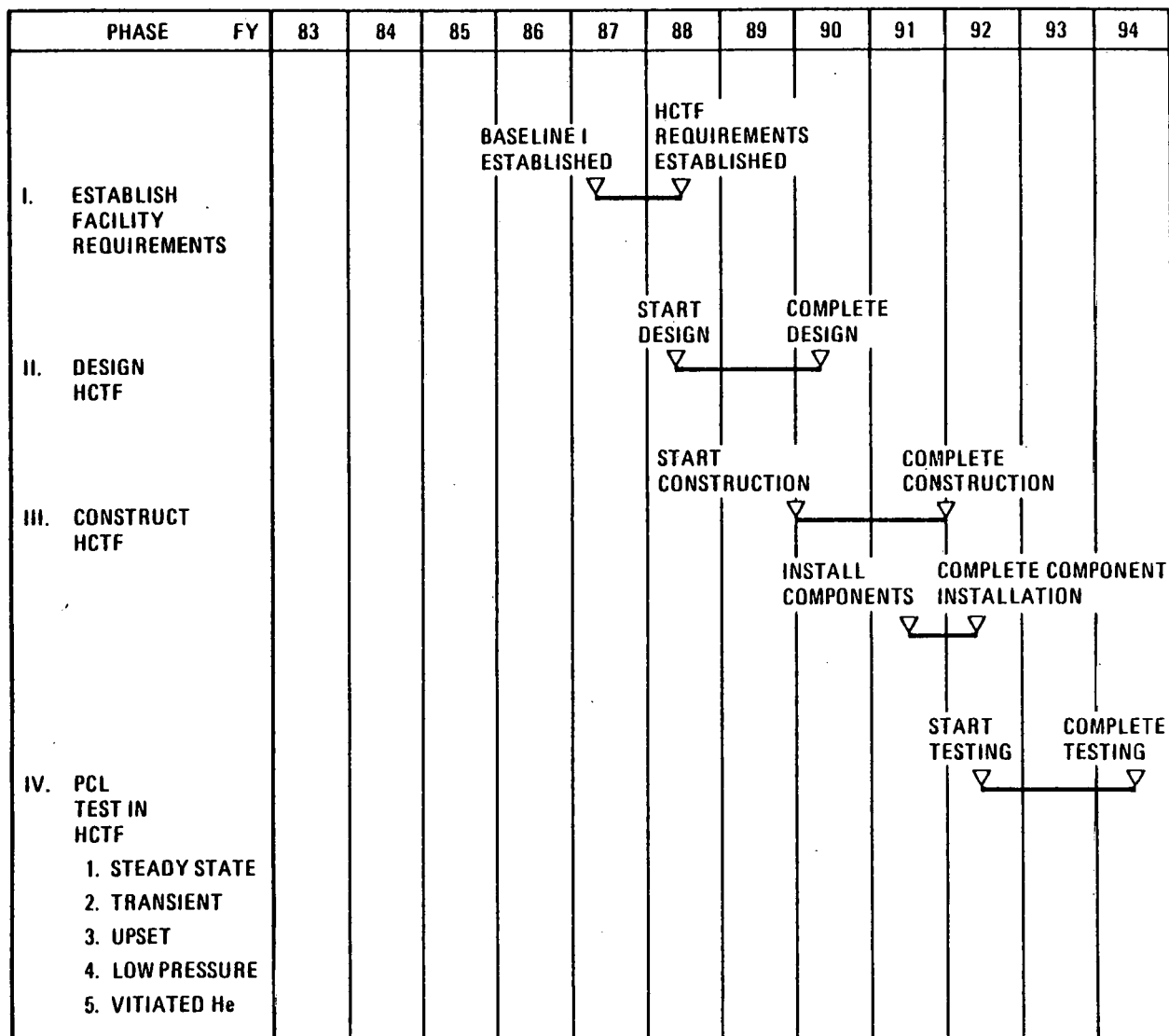
Phase IV. Test of the integrated power conversion loop at full power. The following tests are planned:

1. Startup, shutdown, loading, and unloading.
2. Drop load.
3. Transient response/performance.
4. Simulated reactor startup.
5. Simulated reactor shutdown.
6. Simulated reactor scram.
7. Simulated loss of precoolers water.
8. Operation at low pressure.
9. Operation in vitiated helium.
10. Operation with plugged precoolers tubes.
11. Operation in decay heat removal mode.
12. Generator short-circuit tests.
13. Depressurization tests.
14. Maintenance demonstration.
15. Endurance tests.
16. Simulated single-loop operation.

- Status - In 1975, R. M. Parsons was contracted by the National Aeronautics and Space Administration to develop the preconceptual design* of the HCTF for the DOE (then ERDA). The design was based upon a 370-MW(e) turbomachine, which is about 15% less in rating than the present 440-MW(e) reference design. The facility was also based on a 816°C (1500°F) turbine inlet temperature, rather than the 849°C (1560°F) present reference. Also, the power conversion loop heat exchangers were of a different configuration.

As a result of these differences, a significant amount of work would be required by the RTS supplier and an architect/engineer to update the facility interface requirements and preliminary design.

*Helium Component Test Facility Preliminary Engineering Report, Ralph M. Parsons Company, Pasadena, California, Contract NAS3-19579, June 9, 1975.



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Figure A.1.2-6 Helium Component Test Facility development schedule

A.2 HTGR Generic Technology Program

The HTGR Generic Technology Program develops the base technology and performs design and development common to the HTGR-SC/C, HTGR-GT, and/or HTGR-R plants on a schedule consistent with the specific applications. The overall schedule is shown in Fig. A.2-1. For each of the work areas addressed in this schedule, a description of the development tasks follows together with a more detailed schedule of the work in that area.

A.2.1 Fuel and Process Development

- Scope - Fuel development tasks include out-of-pile thermal stability studies, fuel performance model development and verification, fuel product specifications, and accelerated and real-time irradiation tests and evaluation.

Fuel process development tasks are directed toward establishing and demonstrating corresponding fresh fuel manufacturing processes. The work includes process engineering and equipment development, pilot scale-up and demonstration, test fuel production, and preparation of fresh fuel manufacturing process and equipment specifications.

- Objectives - The primary objective of the fuel development program is to provide the technical basis for selection of a reference generic low-enriched uranium/thorium (LEU/Th) fuel design in the 1984 time frame and to develop a data base for this fuel, which is required to establish fuel product specifications and to support core design and licensing data needs.

The objective of the fuel process development program is to develop and demonstrate fresh LEU fuel manufacturing processes that are scalable to commercial use while providing fuel that fully satisfies HTGR mechanical, thermal, and fission product retention specifications. Process and equipment development work will support reference fuel selection and confirmation decisions by providing manufacturability information, economic assessments, and test fuel product for fuel candidates under consideration. Product fuel from pilot-scale equipment will be manufactured for irradiation tests to relate process parameters to fuel performance.

- Status - The highly enriched uranium/thorium (HEU/Th) fuel cycle was well developed and utilized in the Peach Bottom and FSV HTGRs. Fuel design for the lead plant was advanced to the point that fuel specifications, design data, and mechanistic performance models for the HEU/Th fuel system were issued prior to 1977. In early 1977, the HTGR fuel development effort was redirected toward LEU/Th fuel systems in accordance with the national recognition of the risks associated with highly enriched nuclear materials diversion and weapons proliferation. While much of the data developed for

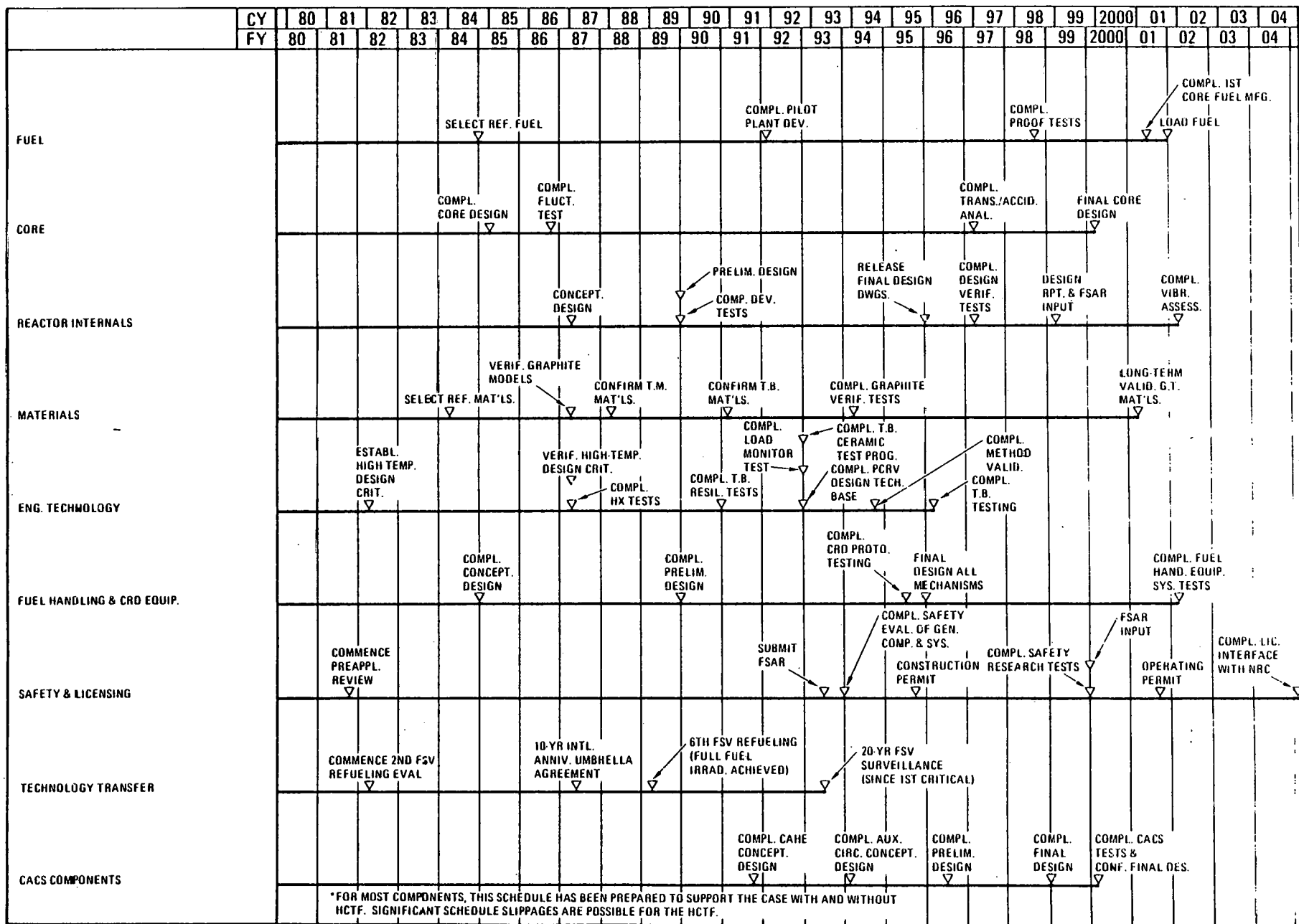


Figure A.2-1 HTGR Generic Technology Program lead plant schedule:* HTGR-GT

HEU/Th fuel is applicable to LEU/Th fuel, additional irradiation, performance, and design data are required to complete the development and licensing of the LEU/Th fuel system.

Irradiated candidate LEU fuel samples are being heated isothermally and in a thermal gradient at temperatures representative of normal and simulated accident conditions. Fuel performance data obtained from these tests will be used to support the choice of a reference fissile fuel in 1984 and the development of LEU fuel performance models.

All candidate fuel types are being evaluated in a series of accelerated irradiation tests: HRB-14, HRB-15B, GF-6 and GF-7 (undergoing postirradiation examination), R2-K13, HRB-15A, HT-35 (under irradiation), and HRB-16 (in the planning stage). Results from these tests will provide the basis for selection of the reference LEU/Th fuel system and development of the fuel specifications for follow-on qualification tests.

Eight fuel test elements (FTEs 1-8) containing some LEU/Th fuel candidates were fabricated and inserted in the FSV HTGR during the first reload in the spring of 1979. The first comprehensive post-irradiation examination is scheduled for FTE-2, beginning June 1983.

Gel-supported precipitate (GSP) LEU fissile kernel process studies are proceeding for the candidate UC_2 , UCO , and UO_2 kernel candidates. Kernel product for each of the kernel candidates has been prepared for inclusion in the irradiation experiments.

Modification of a production-scale coater has been completed with installation of a $ZrCl_4$ powder feeder to deposit ZrC getter coatings. ZrC deposition process studies are in progress. Design of a 240-mm-diameter coater for low-defect PyC and SiC coating development is proceeding.

An assessment of alternative fuel rod heat treatment processes is nearing completion based on FSV production experience and information developed from in-block carbonization experience with the FTEs prepared for insertion into FSV.

A detailed evaluation of the precision of existing fuel quality control test techniques has been completed. Reduced defects and increased quality control precision will improve product yield while retaining low core fission product release.

- Planned Program (Fig. A.2.1-1) - Fuel performance models that describe the kinetics of fuel particle failure and fission product release for normal and hypothetical accident conditions are needed to support core design, reactor safety evaluations, and licensing efforts. Out-of-pile thermal annealing experiments on irradiated

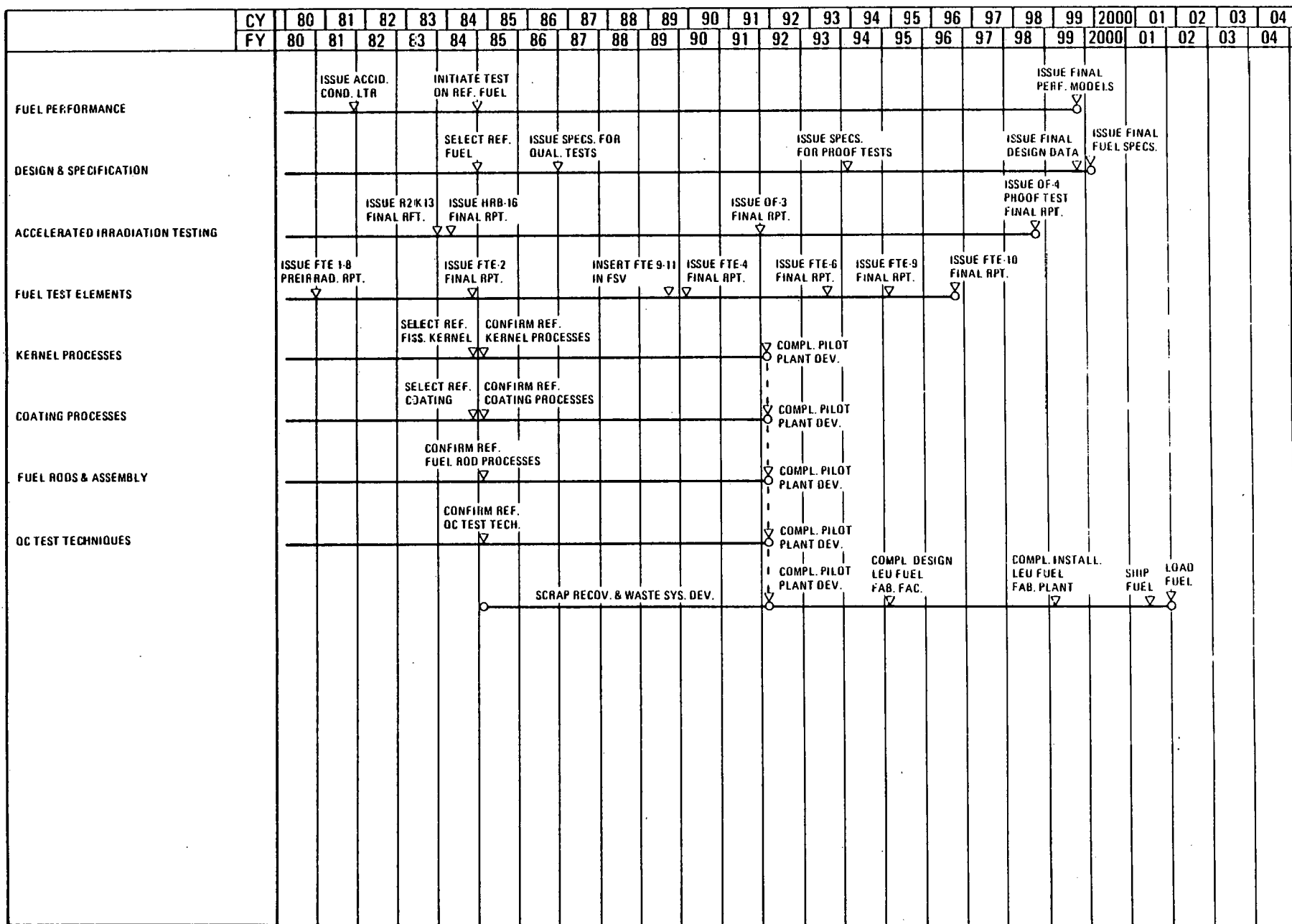


Figure A.2.1-1 HTGR Generic Technology Program detailed fuel milestone schedule: HTGR-GT

fuel under isothermal, thermal gradient, and simulated unrestricted core heatup conditions will be performed to generate the data required to develop and verify these performance models. Initial tests will be performed on candidate fuels to support the reference fuel selection in 1984. Follow-on tests will support core design and licensing of the LEU reference fuel.

Fuel product specifications and design data consistent with fuel manufacturing capability, core design, and fuel cycle requirements will be developed. A preliminary specification will be issued for the series of reference fuel qualification tests based on the results of these tests, the specification will be updated for the proof test (OF-4), and a final fuel product specification will be issued prior to the start of fuel manufacture for the lead plant. Support documents containing the technical justification for the fuel product specifications will also be written.

A series of accelerated capsule experiments is under way to evaluate the irradiation performance of candidate LEU/Th fuels. Following selection of the reference fuel system in 1984, a series of qualification irradiation tests will be performed to generate data required to finalize fuel product specifications and to support core design and licensing. A final integral proof test (OF-4) will then be conducted to demonstrate acceptable irradiation performance on a statistical basis of fuel manufactured in production equipment.

Full-scale integral fuel elements will be tested in FSV to demonstrate fuel performance and to verify design methods under actual HTGR operating conditions. Postirradiation examinations will be performed on the eight fuel test elements (FTEs 1-8) inserted into FSV in the spring of 1979, and three additional elements (FTEs 9-11) containing LEU/Th reference fuel will be fabricated and tested. Since these tests contain large, statistically significant quantities of fuel irradiated in an operating HTGR, they are particularly important to the development of a strong reactor design and licensing data base.

Fuel kernel process and manufacturing scale-up information is needed to support selection and confirmation of the reference fuel. Thereafter, efforts will focus on scale-up process development and pilot demonstration of processes for the reference kernel.

Kernel preparation by the GSP process offers potentially high product yield of uniformly sized spherical particles by methods that are readily scaled up to large-capacity production. Earlier dry-mix processes used for the Peach Bottom and FSV (HEU,Th)C₂ fuel kernels produce less uniform product size distributions than GSP processes. Moreover, the dry-mix process is not applicable for UCO and UC₂/ThO₂. Uniform LEU fuel kernel sizes are needed to

optimize coating designs for the higher local metal loadings required for LEU cores.

Development work is required in broth preparation, gelation/precipitation, drying, and calcining/reduction process steps for each of the kernel candidates to establish process conditions and equipment requirements. Following confirmation of the fuel decision, pilot scale-up and demonstration will provide the basis for manufacturing process and equipment specifications and production capability for proof-test fuel product.

Fuel particle coating specifications have become more stringent for the LHTGRs and for advanced applications. In addition, higher plutonium production and increased fission yields of certain fission products such as silver and palladium increase coating performance requirements for LEU fuels. Also, the use of LEU, particularly for higher-temperature plants, requires higher local fuel metal loadings. These requirements combine to demand fewer defective coatings and more uniform coating thickness control.

Coating process development requires the use of a production-scale coater. Scaling of process parameters from small to larger coaters has not proven feasible in the past. Early in the program, it is necessary to design and construct a 240-mm-diameter pilot coater unit to be used for development as well as subsequent pilot operations.

Required work includes coating process development for TRISO and ZrC "getter" type coatings, pilot operations, and development of manufacturing process and equipment specifications.

In the fuel rod formation process, coated fissile and fertile fuel particles and graphite shim material are metered and blended into single fuel rod size charges, which are then injected with binder matrix in an injection mold. The pitch-bonded rods are then heat treated to carbonize the matrix material. A packed Al_2O_3 bed heat treatment process is used for FSV fuel rod carbonization. An improved process for carbonization of the rods within the fuel element [cure-in-place (CIP) process] has been demonstrated for FSV fuel test elements, but production scale equipment and process control require further development.

Fuel rod manufacturing process and matrix improvements are necessary to demonstrate a low level of fuel particle coating defects and fuel contamination in fired fuel rods containing LEU fuel particles. An area of particular emphasis will be scale-up development and demonstration of a production-scale furnace for CIP fuel rod heat treatment.

To obtain the required quality confidence levels and high process yields of fuel materials with low heavy metal contamination and

very low particle coating defect levels, improved quality control (QC) test techniques are required. Test procedures and high-accuracy equipment systems capable of routinely handling the necessary sample populations will be developed and qualified.

UF₆-to-UNH conversion process development is required to provide fissile kernel broth feed. LEU feed as UNH will not be available. In addition, scrap and waste recovery processes and equipment development are required to support the manufacturing processes. This work will be performed during the pilot scale-up and demonstration phase.

Pilot demonstrations of the key fuel processes and equipment will be completed prior to installation of the LEU fuel manufacturing facility. Pilot equipment will be scaled to provide quantitative demonstration of critical process elements with full-scale features or units as required to demonstrate the process. Major pilot units will include a UF₆-to-UNH process line; fissile and fertile GSP kernel process lines; full-scale ZrC, SiC, and PyC coaters; updated molds for process studies using the existing HEU production fuel rod metering, blending, and forming press system; a fuel rod/element carbonization/heat-treatment furnace; QC test equipment/systems; and scrap, waste recovery, and special nuclear material (SNM) safeguards pilot systems.

A proof-test fuel unit will be fabricated at the completion of pilot development, as shown by the detailed fuel development schedule. Irradiation results will be available for input to the FSAR and prior to final fuel process specification issue.

Following pilot demonstration, fuel manufacturing facility design, construction, and shakedown will be completed on a schedule allowing 27 months for manufacture and shipment of the first core fuel.

A.2.2 Reactor Core

A.2.2.1 Fuel Cycle

- Scope - This task includes all work necessary to select the HTGR fuel cycle. Analysis of fuel mass flow requirements, approach-to-equilibrium cycles, and fuel cycle economics are included. Also included is the fuel design/fuel cycle/core design integration work, which assures that the various design efforts are properly coordinated.
- Objectives - This task is designed to provide the basic HTGR fuel cycle requirements for use by core and fuel design groups. It also will define a fuel cycle which is competitive economically. Fuel cycle analyses will be done on a schedule consistent with the licensing and construction schedule and to provide cost data to support economic evaluations.

- Status - Conceptual fuel cycles for steam cycle applications have been devised for both HEU/Th and LEU/Th systems, and there appears to be no difficulty with the designs, although use of LEU/Th does result in a less economic, more resource intensive cycle. Fuel cycles for higher-temperature applications with LEU/Th systems are not well in hand because the core power distribution tends to shift with burnup, resulting in higher peak fuel temperatures than desired for acceptable fuel performance, and zoning LEU/Th fuel is more difficult than zoning HEU/Th fuel. Core physics design efforts to devise an axial fuel loading scheme consistent with other fuel cycle requirements are presently under way. Also under consideration are alternative fuel cycle schemes which potentially can ease the zoning task.
- Planned Program (Fig. A.2.2-1) - The program provides for definition and refinement of LEU/Th fuel cycle requirements in support of the core and fuel design schedules. Mass flows, burnups, and equilibrium and approach-to-equilibrium cycles will be completed to support preliminary core design and confirmation of the reference fuel choice.

The preliminary fuel cycle will be designed to have acceptable economics and stable axial power shapes. Work on axial power shape stability will be closely coordinated with the core designers to resolve the question satisfactorily by FY 1982. Various zoning patterns to allow mixed HEU/LEU cores and transition cycles from LEU to HEU cores will also be designed as part of preliminary fuel cycle definition. Fuel cycle impacts on recycle plant design and on capsule irradiation tests will be studied as part of the preliminary fuel cycle design task, and updating of fuel cycle cost calculations to support cost estimates will be performed.

Details of the final fuel cycle design, including final mass flows and fuel cycle costs, will be completed prior to beginning final core design. Detailed fuel cycles for HEU-233 cores will be designed to support follow-on reactors and potential lead plant change-over to HEU-233. Long-term strategies involving symbiotic systems with breeders will also be provided.

A.2.2.2 Reactor Core Design

- Scope - This task includes all core design effort in four major areas: core physics, core thermal and hydraulic performance, fuel and replaceable reflector block design and stress analysis, and fuel performance. It also covers design of control rods, neutron sources, reserve shutdown material, and reactor plenum elements. To support the design effort, this task includes the test programs for design and methods verification.
- Objectives - This task is structured to provide the design and resolution of major technical issues for the reactor core and core

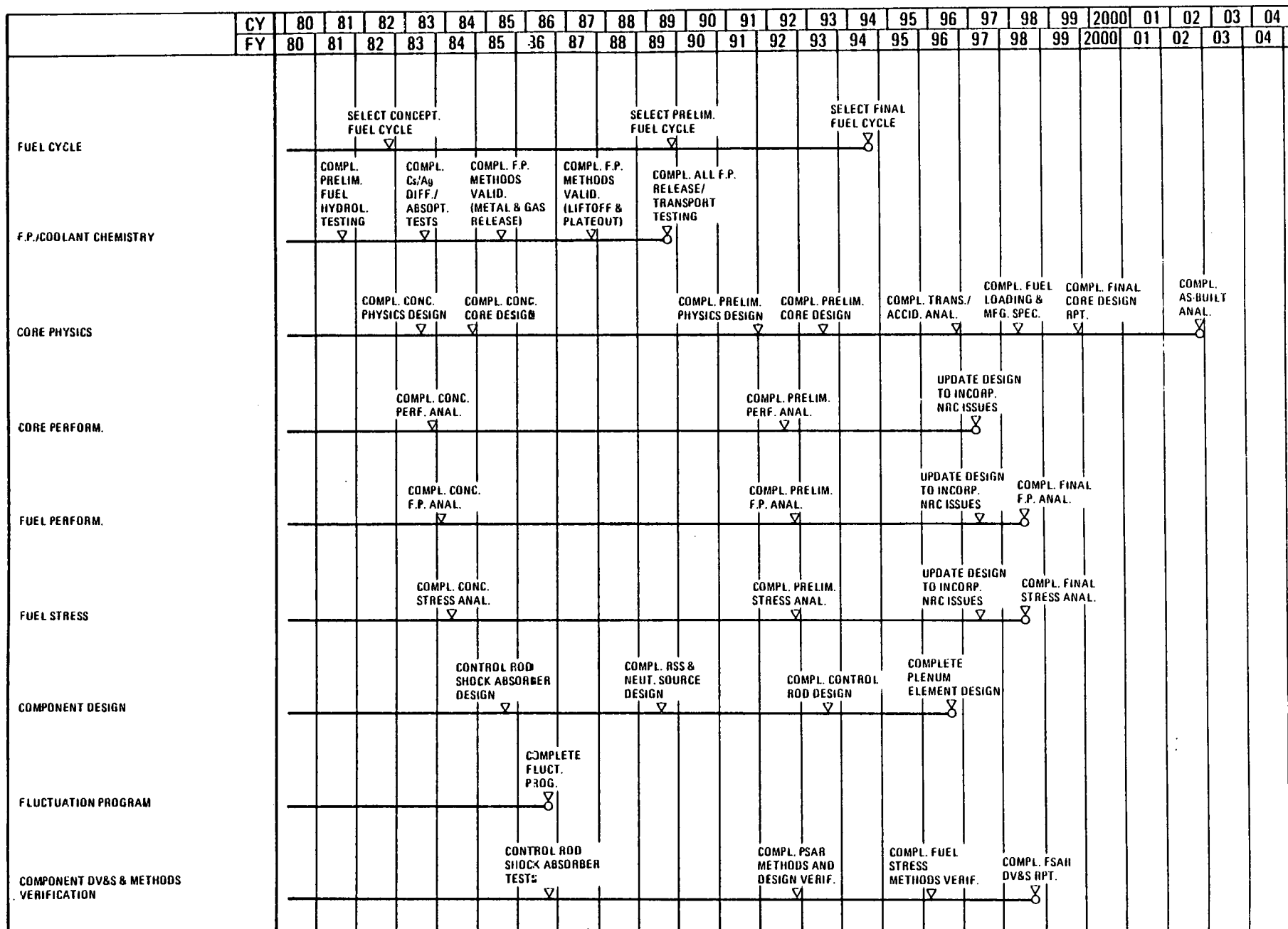


Figure A.2.2-1 HTGR Generic Technology Program detailed core milestone schedule: HTGR-GT

components on a schedule consistent with the plant licensing and construction program.

- Status - Preliminary core design studies for the 2240-MW(t) steam cycle plant have been completed for LEU/Th and HEU/Th designs. These studies show that both LEU and HEU designs are feasible for steam cycle application. For higher-temperature applications, such as the 850°C (1562°F) outlet temperature HTGR-R plant, a steeper axial power profile is desired to keep fuel temperatures as low as possible. This leads to a more difficult axial zoning problem with LEU fuel. Work on 850°C cores is progressing toward producing an acceptable axial zone loading design, but the design has not been fully resolved.

The program to demonstrate structural adequacy of fuel blocks under combined thermal, irradiation, and dynamic loads is ongoing, supported by graphite development and structural mechanics base technology work. Work to date has been directed toward defining the accuracy limits of available methods, benchmark analysis of "simple" geometries, and analysis of test results from the French RWG experiments on irradiated specimens.

Work on optimization of standard fuel block configurations to reduce thermal stresses has shown that peak stresses can be reduced by modification of fuel and coolant hole patterns near peak stress areas. Work is continuing on control block optimization, and preliminary indications are that proper choice of hole patterns can substantially reduce thermal stresses.

Fuel performance analysis has been performed on a number of core designs, and the results to date show that the steam cycle designs with LEU/Th fuel are below regulatory limits for fission product release and plateout. Analysis of higher-temperature cores using existing fuel models show them exceeding circulating activity for the steam conditions by about 40% and metal plateout criteria by about a factor of 2. It is expected that improvements in power distributions, fuel product improvements, and better fuel behavior models will show the calculated fission product release and distribution to be within criteria limits established for all design options.

The issue of possible core outlet temperature fluctuations in the LHTGRs, similar to those observed at FSV, is being addressed in several ways. Analysis of model tests made in FY 1979 have been performed, and an attempt to produce a computer model to predict the flow behavior observed in the model has achieved some success. A long-range plan of design, testing, and analysis has been prepared, and work is under way to resolve the issue of fluctuations in the LHTGR core.

Design of core components other than fuel and reflector blocks (control rods, reserve shutdown material, neutron sources, and

plenum elements) is reasonably well defined for steam cycle application, although changes to the plenum element and design of the power rods remain to be completed. Design of core components for higher-temperature application has not been done in any detail, and work needs to be done to ensure the higher temperatures can be accommodated, either by existing designs or by new designs using different materials.

- Planned Program (Fig. A.2.2-1) - The program proceeds as a series of design iterations which provide the core system and core component information required at each stage of licensing and manufacture. The design effort is supported by the DV&S program, which provides component test data and methods verification.

The first stage of the program is the screening and optimization process by which a conceptual core design is chosen. Core outlines, fuel block designs, loadings, power profiles, and other parameters are varied, and the combination with the best potential performance is chosen for continued design. A core system description is written to give a set of design parameters upon which other plant design work can be based and which provides a reference for the core design work which supports the PSAR.

A number of licensing topical reports (LTRs) will be submitted, detailing the physics, thermal, and fuel design bases to be used in the more detailed design work. This design effort will also support the choice and confirmation of the reference fuel. Existing experimental data will be reviewed to determine the need (or confirm the lack of need) for a test program to verify physics methods for use in LEU/Th fueled core design. The possibility exists that data from European experiments (HITREX, KAHTER, PROTEUS, etc.) can be obtained through international cooperation. This could eliminate the need for a U.S. experimental program.

The second stage is the preliminary design effort, which provides the core system information to be used in the PSAR. The core technical specifications to be submitted as part of the PSAR will be prepared. Transient analysis, fission product release and plateout analysis, reactivity analysis, and stress analysis will be performed.

Results will be provided from methods verification programs and fuel element tests, such as stress tests on irradiated fuel elements from FSV. The core fluctuation program is scheduled to be completed in advance of PSAR submittal, and the results will be factored into the preliminary design, anticipating that core fluctuations will be of particular interest in the licensing review.

Other component design and DV&S efforts are not extensive in this phase of the program, because there are no known major design issues with the components. The one possible exception is control

rods in high-temperature design applications, and early investigation of control rod designs for high-temperature application is planned. Definition of the test programs specifically for fuel element design and methods verification (as opposed to base technology work on graphite and structural mechanics) is scheduled for completion in FY 1981, as is the companion plan for thermal/hydraulic DV&S. Definition of other component test programs and test facility requirements will be completed prior to PSAR submittal.

The first part of final design work will provide information necessary to select the final fuel cycle and fuel loading specification and will establish the core parameters upon which the FSAR will be resolved, and any design changes required will be incorporated. The accident and transient analysis will be reviewed to determine which events lead to the most severe stress conditions, and a detailed stress analysis will be done using updated material models and methods.

The rest of the final design work provides the complete design and DV&S package for the FSAR. The detailed physics, performance, and stress analysis reports will be completed. Final fuel loading will be confirmed. Reports from the methods and design verification tests will be provided. Detailed component designs and proof-test results will be completed. An independent design analysis will be done and all issues raised by it resolved. Startup procedures will be prepared, as will the fuel surveillance plan.

Following submittal of the FSAR, the as-built analysis will be performed, and support of loading and startup will begin.

A.2.2.3 Fission Products and Coolant Chemistry

- Scope - This task includes all the experimental work necessary to describe the mechanisms for release of fission products from fuel particles and to describe the interactions of fission products and other primary coolant impurities with fuel and other reactor plant components. It also includes the analytical work needed to develop models to predict behavior of fuel, fission products, and coolant impurities and to verify the adequacy of the models.
- Objectives - The primary objective of this task is to develop verified predictive models for fission product release, fission product and carbon transport and plateout, and fuel/coolant impurity interactions. The models will be used in design computer codes to compute circulating activity, plateout activity on reactor components, and the effects of coolant impurity interactions on the integrity of fuel and other reactor components.
- Status - The fission product retention characteristics of HEU-235 carbide TRISO coated fissile particles developed for FSV are reasonably well known from an extensive irradiation test program.

Thorium oxide BISO and TRISO coated fertile particles are also reasonably well understood. However, the fission product yield differs for low-enriched fuels. Therefore, additional work is required to get the same information about candidate LEU fuels: UC₂, UCO, and zirconium-buffered UO₂.

Transport of fission products through graphite has been studied extensively for some isotopes, for example, but again additional work needs to be done because of the change to LEU fuel. In particular, actinide and silver transport, which was not previously significant, now requires more attention.

Plateout and liftoff of fission products are being addressed from two points of view. Laboratory studies of the sorption and desorption of fission products on graphite and metals have been started to obtain an understanding of the interactions between fission products and reactor materials. Fission product liftoff has been studied in the GAIL loop and the French CPL tests, but uncertainties in the results require further testing. In addition, the question of formation and distribution of carbon dust has not been systematically studied, although some Peach Bottom surveillance work did briefly examine carbon deposition. Fission product plateout methods validation using Dragon and IDYLLE-03 experimental data has been started, and performance of the FSV initial core is being monitored. The original plan was to use the CEA COMEDIE loop for a series of integral tests designed to provide a complete plateout and liftoff methods validation. A DOE decision not to allow funding of the COMEDIE test program has been a major setback. Either an alternate program or a reversal of the DOE position will be necessary to provide adequate methods validation.

Carbide fuel particles with failed coatings have the potential for hydrolysis in the presence of moisture in the primary coolant. Some recent studies indicate that the release of gaseous fission products from hydrolyzed particles may be larger than the value presently assumed. Since release rate can have a direct effect on circulating activity, additional work is required to verify the higher release rates.

- Planned Program (Fig. A.2.2-1) - Fission product release from LEU fuels will be studied using particles from capsule irradiation tests performed under the fuel development program. The program is scheduled to update the fission product design data prior to the start of the fission product release calculations done for each phase of reactor core design. The first such updating is scheduled for the end of FY 1982 to provide the most recent data for use in the preliminary (pre-PSAR) core design. The second updating is scheduled for the beginning of final design, and a proof test and final analysis report is scheduled for use in FSAR preparation.

Tests to characterize cesium and silver migration in graphite and SiC will be performed and the results included in the updating of the fission product design data. In-pile and out-of-pile experiments on irradiated and unirradiated graphites and SiC are planned. Work on actinide transport and tritium/graphite interactions will also be performed in conjunction with ORNL.

The methods validation program will make maximum use of data available from other international HTR programs and from FSV. Data from IDYLLE (CEA), SMOC (FRG), DRAGON (U.K.), and SAPHIR (CEA) will be analyzed, to the extent that international co-operation permits access to the data, along with FSV information. A resolution of the CEA COMEDIE loop question will be made in FY 1981 either by establishing a replacement program or by obtaining DOE concurrence to proceed with COMEDIE. The methods validation program is structured to have a significant amount of validation work completed prior to commencement of preliminary design. This will give results for the PSAR which have high confidence limits. Most of the methods validation program is scheduled for completion prior to the beginning of final design.

Tests to confirm fission product release characteristics of hydrolyzed fuel are scheduled to commence in FY 1981 and be completed prior to PSAR submittal. Tests will be done in the TRIGA reactor on failed and unfailed hydrolyzed and unhydrolyzed fuel particles.

A program is planned to study the potential for carbon dust formation, the behavior of dust in the reactor (e.g., plateout on metals, collection in stagnant areas, etc.), and the role of carbon dust in the transport of fission products by sorption of fission products on dust particles. This program will be scheduled to define the scope of any potential problem and a plan to resolve it prior to PSAR submittal and to achieve resolution prior to FSAR submittal.

A.2.3 Reactor Internals

- Scope - This task involves the design of the generic reactor internals components consisting of the core support floor, permanent side reflector, core peripheral seal, core lateral restraint and side shield, and the permanent upper plenum bridge structure associated with the in-vessel refueling system. The end products of this task are design drawings and specifications sufficient to procure, fabricate, and install the reactor internals components and final design reports as required by regulatory code and licensing requirements.
- Objectives - The objectives of this task are to use the basic technology developed in the generic graphite, materials, and plant technology tasks to produce component designs that will meet all

the functional, structural, and safety requirements, and to verify these designs analytically and experimentally as required to satisfy the NHS supplier, the customer, and regulatory agencies that the reactor internals components will perform their design purpose satisfactorily.

- Status - Although the reactor internals, particularly the core support floor (CSF) and permanent side reflector (PSR), are outgrowths of and generally similar to the comparable components in FSV, there are significant differences. In FSV the entire core, CSF, and lateral restraint structure (core barrel) are mounted on an intermediate concrete floor within the PCR. The PSR is keyed to the core barrel, which is in turn radially keyed to the PCR cavity wall to prevent lateral motion of the entire core assembly.

In the LHTGR, the core and CSF are supported directly through the PCR bottom head, so there is no need for a core barrel. Instead, there is a simple seal structure around the periphery at the CSF. Also, more stringent seismic requirements resulted in the design of a spring-type core lateral restraint system, which extends from the PCR liner and interfaces with the PSR to hold the core in its correct position during normal operation and cushions the core assembly during seismic events.

In addition to the design changes caused by the overall reactor arrangement and seismic requirements, numerous detailed design improvements have been effected to enhance structural capability, facilitate in-service inspection, increase control instrumentation accuracy, and minimize hot streaking.

The conceptual design of CSF and core restraint structures for the HTGR-SC is complete, and extensive component and parametric testing have been completed at GA and CEA.

With the adoption of the in-vessel refueling scheme, the upper plenum bridge-like structure required by that scheme was assigned to the reactor internals system. This structure is defined only by basic outline drawings and rudimentary design requirements.

- Planned Program (Fig. A.2.3-1) - The most important design problems to be worked out for the reactor internals include the seismic load capability of the CSF, the effect of temperature on the core lateral restraint and core peripheral seal structures, the stability of the PSR, and development of the design definition for the upper plenum refueling bridge.

In addition, a modified CSF design is required for the HTGR-GT to accommodate the combined maximum turbine depressurization accident

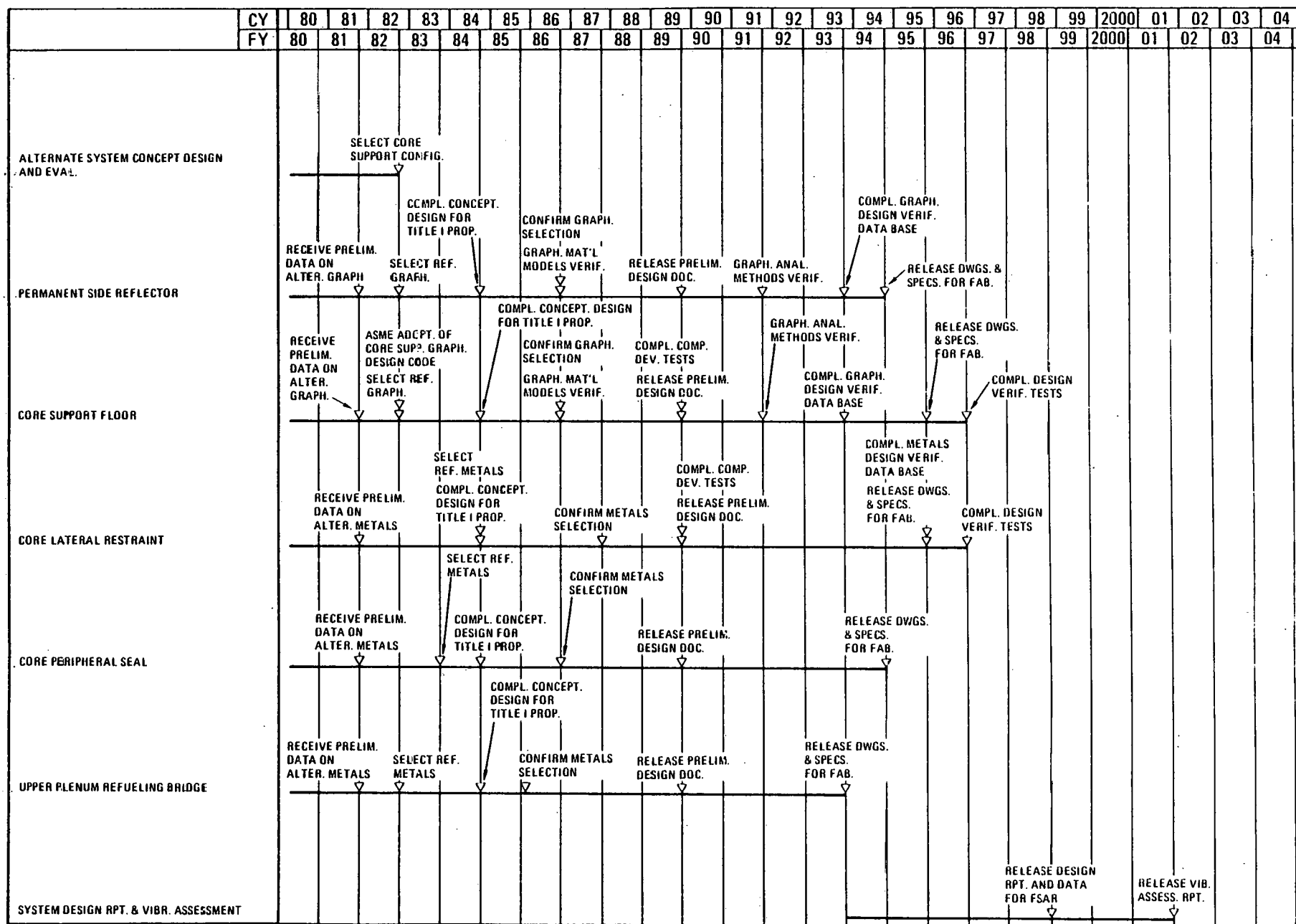


Figure A.2.3-1 HTGR Generic Technology Program detailed reactor internals milestone schedule: HTGR-GT

and safe shutdown earthquake. A basic concept has been tentatively identified, but further design and analysis will be required to confirm that or any other alternative design. The potential benefits of the alternative high-strength CSF for the HTGR-GT may make it worthwhile to develop that concept sufficiently to permit evaluation of its applicability to other plants as well. However, in parallel, the outstanding issues in the current CSF are planned to be resolved as discussed below.

A design issue for the core support has been the high stresses in the graphite due to a combination of normal dead weight and pressure loads, thermal stresses, and seismic loads. Through division of the CSF into multiple, stronger pieces and minor design changes within the CSF, the combination of normal loads and thermal stresses has been reduced to the level of a significant but not unmanageable design problem. However, superposition of the seismic loads with the others is still a design issue. Resolution of this problem requires not only more design and analysis, but also better definition of the seismic loads using the new seismic analysis codes and the better materials data available from the Graphite Materials Program. A secondary part of this design effort will be to consider the effects of oxidation on the subsequent shape of coolant flow passages and the influence on coolant mixing and core outlet temperature measurement accuracy. Ultimately, proof tests will be required to confirm CSF component designs.

When the springs for the core lateral restraint were first designed in 1974, it was known that they would be operating near the limit of the useful temperature range for Inconel 718, particularly those located down near the CSF level. Service temperatures for these springs must be confirmed, and determination of the effects of long-term exposure will be needed. The High Temperature Materials Program has incorporated an effort to identify and provide preliminary data on several candidate materials for the springs. The component designers will evaluate these candidates and select up to three for final screening tests. Tests will also be conducted to confirm the feasibility of fabricating springs from these materials and to determine spring relaxation properties. Although the initial intent is to narrow the choice to one material, cost considerations may dictate the use of two materials for springs at different temperature levels. In either case, the material(s) selected will be fully characterized and spring samples will be tested individually and in assemblies.

There is a concern that the pressure differential acting on the outer periphery of the PSR could cause buckling instability of the cylinder formed by these columns of stacked graphite blocks. If so, the outer core regions could be deformed to the detriment of neutronic and coolant flow control. Additional design and analysis will be required to resolve this issue, and a model test is planned for design development and verification. However, before

this work can be done, the question of availability of material for the PSR blocks must be answered. A joint effort is planned between the Graphite Materials Program and Reactor Internals Design to evaluate the alternatives.

The effect of oxidation on the load-bearing surfaces of the core support posts and seats requires resolution. Analysis has indicated that the graphite of the post and seat hemispherical surfaces would deplete the coolant of oxygen before it reached the critical contact area. Although this should also be true for normal operating conditions, it is not yet clear that seismic motions would not move the contact area onto oxidized material. A kinematic study is required to define the size of the load-carrying area. Oxidation characteristics of 2020 graphite, determined by the Graphite Materials Program, will be used to calculate whether oxidation extends into the potential contact area and to what extent. The effects of oxidation on strength, also provided by the Graphite Materials Program, will permit an assessment of the effect of any predicted oxidation of the contact surfaces.

For the traditional reactor internals components, there is an ongoing responsibility to maintain and enhance the replaceability of the components, especially for higher plant operating temperatures. Normally, this will be a part of the regular design effort and not a special task. Design features to satisfy in-service inspection (ISI) requirements will also be incorporated as a normal design activity.

It will be necessary, at a relatively early date, to bring the design definition of the upper plenum refueling bridge to a level at least comparable to that of other components. This structure interfaces with the core, refueling system, control rod drives, and PCRV liner and thermal barrier. The horizontal structure to support the in-vessel refueling equipment and fuel elements must be supported so as to withstand normal dead weight and operating loads plus seismic loads, without causing any distortion of the refueling penetration extensions due to those loads or relative thermal expansion.

A vibration assessment of all the reactor internals components in compliance with Regulatory Guide 1.20 will be conducted. This will include analysis and testing of selected components such as the springs, side shield plates, and upper plenum bridge and will conclude with an evaluation of data to be taken during flow testing prior to reactor startup.

In addition to resolution of the foregoing problems, design drawings, specifications, and reports will be provided as required for the completion of conceptual design and preliminary design and for the final design for fabrication. Input information will be provided for the PSAR and FSAR, and the final design report and installation specification will be provided prior to installation of the components in the reactor.

A.2.4 Materials

A.2.4.1 Graphite Material Development

- Scope - The scope of this task includes the identification or development, if required, of commercial graphites and the procurement and evaluation of production logs for qualification as HTGR components. Key reactor components manufactured in graphite are the fuel and replaceable reflector blocks, the core support blocks, posts, and seats, the PSR blocks, and the triangular core peripheral seal logs. The graphite experimental program includes the following: characterization of the reactor component graphites for properties and chemical impurity content; determination of irradiation behavior, including dimensional and property changes; evaluation of graphite fatigue behavior and behavior under complex loads; assessment of coolant impurity effects on strength and safety margins; determination of irradiation-induced dimensional and property changes; establishment of a statistically significant design data base for the selected graphites; development and verification of material behavior models for reactor service conditions; and verification of analytical methods. The work is organized and identified by related component.
- Objectives - The objectives of this task are to identify and qualify commercial graphites and boronated graphite control material capable of meeting the long-term requirements of the HTGR industry and to develop the support technology essential to the safe, reliable use of these materials in HTGRs.
- Status - H-451 graphite has been especially developed as a high-purity graphite for HTGR fuel elements. H-451 graphite has been licensed for use in FSV, although the NRC has expressed an intent to require more thorough understanding of the material for any future HTGR. A requirement to demonstrate satisfactory calculated stresses leads to a need not only to understand the material behavior better, but possibly also to achieve a higher-strength grade of H-451 graphite.

For the core support posts, peripheral seal logs, upper core support blocks, and bottom-most replaceable reflectors, a high-strength, low-oxidation-rate commercial-grade 2020 graphite from the Stackpole Carbon Company has been selected. The remainder of the core support block is currently designed from PGX from Union Carbide Corporation, and HLM graphite from Great Lakes Carbon Company has been the reference material for the PSR blocks. For both of these last two materials, the required size of graphite logs from which to machine the finished part was a strong factor in their selection. Work is in progress with Union Carbide Corporation to develop an improved, purified grade designated TS-1621 in an attempt to improve on the oxidation rate of PGX and secondarily to increase the strength of the bottom core support block. In May

1980, Great Lakes Carbon notified GA that they will no longer produce the blocks of HLM in the sizes needed for the HTGR PSR. The reason cited was an unsatisfactory yield of acceptable logs in those large sizes. Assuming this situation continues, it may be necessary to find an alternative material or to redesign using smaller blocks.

The design data base for H-451 graphite is almost fully established, requiring only updating for the effects of changes in raw materials in the manufacturing of graphite and final confirmation of the effects of irradiation. The design data base is not well established for either 2020 or PGX graphite, although much information has been obtained on these materials, and the least information is available on HLM graphite. In the past, the emphasis has been on selection and design data base testing. However, in the last 3 yr, there has been a growing recognition that well-established analytical methods and material models for metals, even those for typical brittle materials, do not accurately predict the behavior of graphite. Consequently, there has been a shift toward more fundamental tests, very carefully planned and instrumented, to develop more accurate material behavior models for graphite in order to aid in the development of analytical techniques which yield results representative of the observed response of the material to the test loads. The results of these fundamental tests may modify the data requirements for the design verification data base.

A series of tests has been run for PGX and 2020 graphites to evaluate the effects of thermally induced stresses. These tests showed that actual fracture of the material occurred at thermally induced stresses equal to or only slightly above the failure stress for uniaxial tension. Therefore, the existing design criteria had to be changed, in agreement with the recommendation of Franklin Institute Research Laboratory (FIRL), to treat thermal and direct load stresses equally. The thermal stress test program was concluded with demonstration of the feasibility of performing thermal fatigue tests. Since the thermal fatigue characteristics were very similar to those observed in direct-load fatigue tests, it may not be necessary to conduct extensive thermal fatigue characterizations of the graphites.

PGX graphite has also been tested to determine the applicability of fracture mechanics techniques, with positive results. It remains to be proven that fracture mechanics are applicable to fine-grained graphites like 2020.

Carefully accelerated oxidation of PGX and 2020 graphites in a reducing atmosphere of helium, hydrogen, and water vapor (representative of reactor conditions) has confirmed that the graphite has a surface-oriented oxidation profile. Thus, it has been possible to provide a corrosion allowance on the wetted surfaces of the core support blocks and posts and rely on full strength in the bulk of

these thick structural members. Other researchers, intent upon oxidizing graphite as rapidly as possible for subsequent tests of oxidized material, have often used a highly oxidizing atmosphere and have sometimes observed greater oxidation in the center of their specimens than at the surface. This has been explained as an artifact of the highly oxidizing atmosphere, produced by changes in the impurities of the helium as it penetrates the graphite, which activate the catalytic impurities in the graphite near the center but not at the outside. Since the reactor coolant is maintained in a reducing state, these other experiments are of no concern for the HTGR.

- Planned Program (Fig. A.2.4-1) - The graphite development work is divided into four main groups as discussed below. This work is further broken down on the basis of which component material (e.g., fuel block, core support post, etc.) is being tested and on the basis of which organization has the lead for a particular test.
 - Material Modeling - This group of tasks has the objective of theoretically and experimentally developing analytical models of graphite which will accurately represent the response of the material to directly imposed loads, thermally induced stresses, irradiation, and oxidation. Experiments will proceed in parallel with theoretical development, beginning with very simple, fundamental tests and progressing to more complex situations that are more representative of reactor service conditions. The material models that will be developed and the accompanying analytical technology are essential to the establishment and regulatory acceptance of graphite structural design criteria, particularly for the fuel block design.

The enumeration and description of the material modeling tests are too lengthy to present here. However, these tests can be generally described as follows:

1. Tests for the effects of test atmosphere (e.g., humidity), hydrostatic gas pressure, and rapid depressurization on the properties of the graphite and a determination of how these factors may influence the material testing and/or component designs.
2. Tests to develop the constitutive equations for stress-strain behavior of coarse- and fine-grained graphites, beginning with uniaxial monotonic stress-strain tests and progressing in an orderly fashion to multiaxial stress-strain tests with an arbitrary load history. This is one of the most critical test series for the development and establishment of graphite structural design criteria.

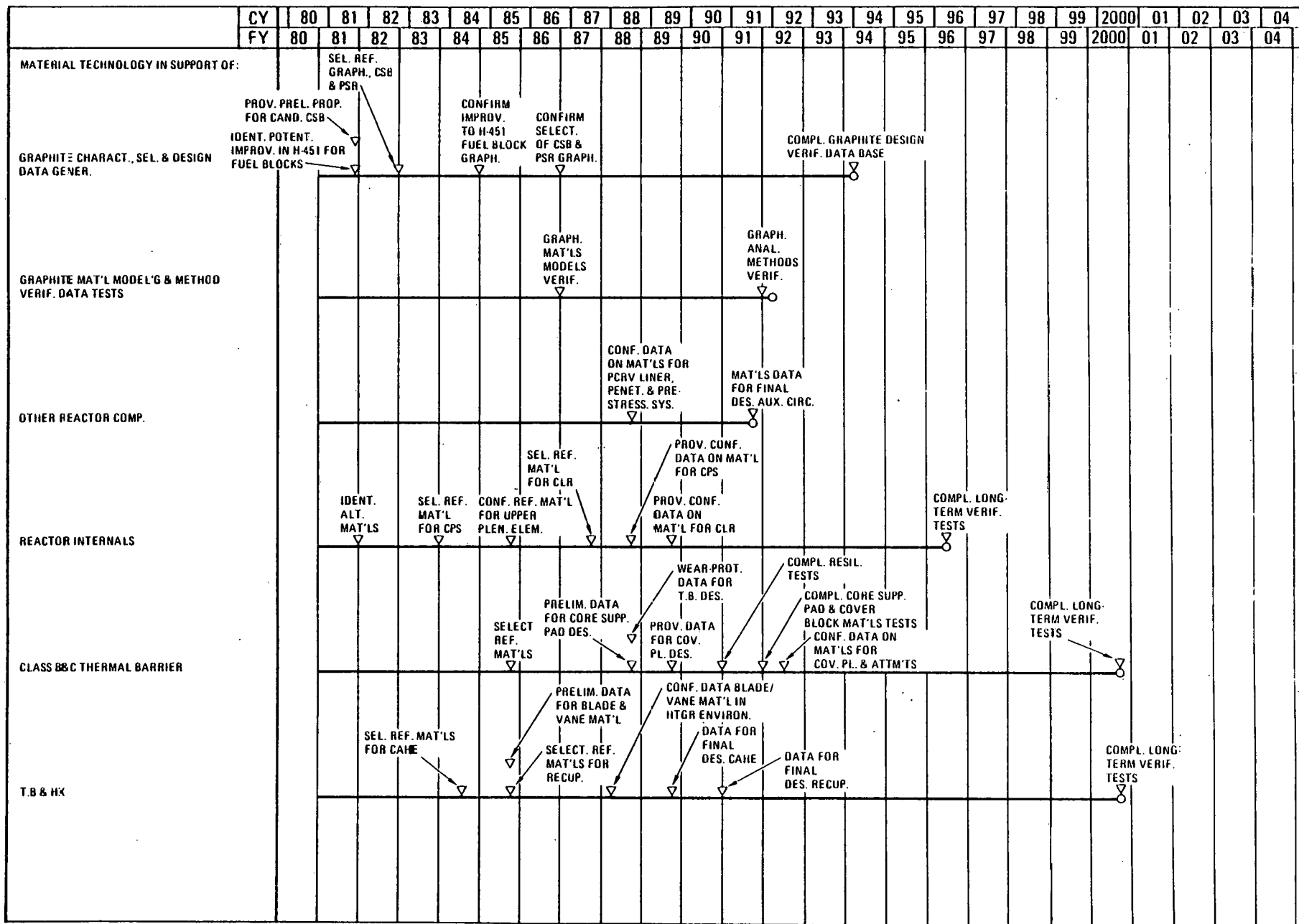


Figure A.2.4-1 HTGR Generic Technology Program detailed materials milestone schedule: HTGR-GT

3. Tests to determine the effects of high strain rate loading, including impact loads (for use in seismic analyses).
 4. Tests to determine the effects of a strain gradient and to develop a failure theory which explains the observed higher apparent strength of graphite in bending than in simple tension.
 5. Tests to develop a reliable statistical failure model based on the modified Weibull theory using four parameters. This will satisfy NRC concerns over why such an approach has not been adopted.
 6. Fatigue and cumulative damage rule tests, which will provide the basis for analysis of multiple shutdown and startup and power change thermal stresses combined with seismic loadings.
 7. Tests to determine the effects of uniform oxidation on several material properties which affect the strain, stress, and strength of the graphite under reactor conditions. These properties will permit an analytical division of the oxidized outer material into successive layers with changing properties for more accurate prediction of structural strength and behavior.
 8. Tests to determine the effects of irradiation and of irradiation at different temperatures on material properties, including fatigue strength. These data will permit the prediction not only of irradiation-induced stresses but also of the graphite response to those plus other loadings.
 9. Tests to determine the effects of reactor conditions on fracture mechanics properties of graphite and to extend the graphite fracture mechanics technology to include three-dimensional stress fields representative of service loadings on the fuel blocks, core support blocks, and support posts.
- Methods Verification. This task will consist of a series of tests designed to verify that the material models and analytical techniques developed properly predict the graphite behavior under various individual and combined loading conditions. This information will be essential in supporting the design criteria and the design information to be included in both the PSAR and FSAR. Here, too, the tests are grouped in related series for discussion:
1. Tests of simple, two-dimensional structural shapes (e.g., beams and bars) to verify the response to increasingly

complex loadings and load histories, ultimately including the effects of irradiation-induced stresses.

2. Impact tests beginning with simple bars and progressing to simple models representative of reactor components, again ultimately including irradiation-induced stresses.
 3. Fatigue tests of specimens progressing from simple shapes to models representative of reactor components to verify fatigue analysis techniques and the cumulative damage rules.
 4. Static loading and thermal stress tests of specimens having an oxidation profile to ensure that the analyses correctly predict the effects of these combinations.
- Reference Grade Selection - The work in this area will identify candidate alternative graphites and provide preliminary mechanical, physical, and oxidation properties for components for which a reference graphite has not been confirmed or becomes unavailable or where design analysis indicates that material improvements are needed. At present, there are three component materials which need work:
1. The PGX graphite used for the lower core support block has a very high oxidation rate and low strength. Work is in progress and will continue to develop a domestic replacement for PGX having better oxidation characteristics and higher strength, as well as to identify possible back-up grades of foreign manufacture in cooperation with the FRG.
 2. More rigorous regulatory requirements to objectively demonstrate acceptable calculated stresses in fuel blocks lead to a requirement for a multi-faceted design approach: establishment of more definitive design criteria, improvement of analytical technology, and development of higher strength in the material as well as detail design changes in the components. A study is planned for determining the technical and economic feasibility of strengthening H-451 graphite without invalidating the existing data base. To a somewhat lesser degree, the same type of activities are required for the core support and PSR, which are less complex structures than the fuel blocks.
 3. The recent decision of Great Lakes Carbon Company not to market the large blocks of HLM graphite needed for HTGR PSRs makes it necessary to evaluate alternative materials and to evaluate the feasibility of redesigning to use smaller blocks in the PSR. A study will be conducted

jointly with the design organization to identify and evaluate solutions to this problem. The preference will be to utilize, if at all possible, an established material in common with another reactor component.

- Data Base Development - This work includes extensive testing on many logs of each reference graphite to establish statistically significant design values for the chemical, physical, mechanical, and irradiation properties required for HTGR component design and verification. These graphite design data are needed by the component designers and will be referenced in the PSAR, design reports, and the FSAR. Typical data to be determined include stress-strain curves in tension and compression, minimum ultimate strengths, chemical compositions, oxidation and irradiation properties, thermal expansivity and conductivity, fatigue properties, fracture toughness data, etc.

A.2.4.2 Structural Materials Technology

- Scope - Materials evaluation and development tasks will include all those property tests and characterizations necessary to select and qualify structural materials to meet the design requirements of the HTGR systems and components. The major emphasis in this work will be to address the issues of retention of adequate strength and toughness by materials to be used in components that will operate at the highest temperatures in the primary coolant circuit, such as the reactor internal structures, the thermal barrier in the core outlet and hot duct areas, the core auxiliary heat exchangers, the gas turbine volute, stator vanes, and turbine blades in the case of the HTGR-GT, the IHX for the HTGR-R, and the steam generator for the HTGR-SC/C. The structural materials tasks include testing and evaluation of all selected design reference metallic materials, except graphite. Included are the fibrous and solid ceramic materials to be used in thermal barrier components.
- Objectives - The overall objectives of the materials program are (1) to provide the materials data and supplementary information required to perform the design tasks, (2) to ensure the use of acceptable materials from the standpoint of performance, costs, reliability, safety, and licensability, and (3) to perform both short-term and long-term tests and surveillance tasks to complete the qualification of materials for service in the specific applications.
- Status - The ongoing generic materials technology studies have made significant progress during the past several years in identifying and outlining solutions to the key materials behavior issues for HTGR systems. Until lately, these studies were concerned principally with materials suitable for service in a steam cycle system, in which the design operating temperatures of the

hottest components are generally in the range of 650° to 750°C (1202° to 1382°F). During the past year, major emphasis has been directed to advanced systems in which component temperatures reach 850°C (1562°F). Some exploratory work has also been initiated toward reaching a target of 950°C (1742°F).

The data needed to support the design of components of the alternative HTGR system have been outlined in matrix format that provides visibility and focus. It serves as a basis for planning and scheduling the performance of the required materials tests. A large amount of materials data has already been developed which is applicable to the HTGR systems up to 850°C (1562°F). However, additional work is required to complete the screening, selection, and qualification of materials for some key components, such as the high-temperature turbine components, IHXs, thermal barrier in the core outlet and hot duct areas, and core restraint mechanisms. For the fairly well-established materials, such as Alloy 800H, 2-1/4 Cr-1 Mo steel, and Hastelloy X, much of the current data represents short- to medium-term tests (e.g., creep rupture and gas corrosion tests for times in the range 10,000 to 20,000 hr). Longer-term confirmatory tests (>50,000 hr) are planned for materials to be used in the primary coolant circuit.

Materials which have been less well characterized, such as Inconel 617, IN713LC, IN100, and MA 754, will be required for some components in HTGR systems having an 850°C (1562°F) core outlet temperature. Such materials will require more intensive testing programs to ensure selection and qualification of the appropriate materials for reliable performance.

In addition to the program at GA, gas-cooled reactor materials testing programs are being sponsored by DOE at the Metals and Ceramics Division of ORNL and at the Energy Systems Programs Department of General Electric Company, Schenectady, New York. Under guidance of DOE and GCRA, a Materials Coordination Committee has functioned during the past year as a means for providing consistency in program scopes and compatibility of the data bases developed at the three laboratories. This committee serves as a forum to review and guide the scope of the materials development work needed to support the reactor program and provides a means to assure that design and schedule requirements are communicated effectively.

- Planned Program (Fig. A.2.4-1) - The planned materials technology work consists of tests and evaluations to provide the data required to support the design of specific components. In particular, attention will be given to materials for those components that are located in the primary coolant circuit and therefore must operate in the hostile environment of very-high-temperature [$\geq 850^{\circ}\text{C}$ (1562°F)], gaseous corrosion due to trace impurities in the coolant helium and extremely long-term (creep) stress conditions. The key high-temperature components that need additional materials test data to support the design are discussed below.

- Class B Thermal Barrier - In simplified terms, this component consists of panels of fibrous insulation which are compressed and retained by cover plates that are attached by mechanical fixtures to the interior surfaces of the PCRV cavities. The principal candidate materials for the fibrous blanket insulation are Saffil (alumina), Kaowool (alumina-silicate), and various grades of graphite felt materials. In order to serve as effective thermal insulators, these blanket materials must maintain their resiliency, and the integrity and strength of the fibers must be retained throughout the design service life (300,000 hr). Hence, tests currently in progress will be continued to determine the long-term resiliency of fiber blanket materials after exposure to the simulated service conditions of high temperature, compression loads, and the helium environment containing trace impurities. In addition, tests must be done on the fibrous blanket materials to determine such properties as thermal conductivity and permeability to helium flow and the effects of thermal cycling, acoustic vibration, neutron irradiation, and the presence of fission products.

For service temperatures up to approximately 750°C (1382°F), the principal candidate materials for cover plates and attachments are Alloy 800H and Hastelloy X. For higher temperatures, the stronger cast nickel-base alloys IN713LC and IN738 are being evaluated. At 850°C (1562°F) and above, temperature-resistant cast alloys have insufficient strength, so testing will be done on candidate composite carbon fiber-carbon matrix materials.

The basic structural properties data that must be provided to support the design of the thermal barrier cover plates and attachment fixtures include elevated-temperature tensile and yield strengths, fracture mechanics data, both low-cycle and high-cycle fatigue test data, creep fatigue test data, and creep-rupture test data. For all the tests, the data are required for the full range of service temperatures.

In addition to the outlined properties data required by the design engineers, the thermal barrier cover plate and fixture materials must be tested to determine their resistance to the effects of long-term exposure to the service environment. In particular, gaseous corrosion effects, such as carburization due to impurities in the primary coolant helium, may embrittle the alloys and cause failure.

Thermal aging effects which are usually detrimental to the properties of the alloys also occur during service. The microstructures of most commercial alloys are in a metastable stage when the alloys are manufactured into usable mill products, e.g., rolled plates. Upon very long exposure to high temperatures, the microstructures may change in a manner that

adversely affects either the strength or fracture resistance. Both Alloy 800H and Hastelloy X undergo some deterioration of properties due to thermal aging. These effects must be satisfactorily evaluated for the candidate alloys by very long-term tests before reliable performance can be predicted.

Similarly, friction and wear tests are required to assure that rubbing action at the attachment fixtures due to thermal expansion will not cause failure.

- Class C Thermal Barrier - Thermal protection at the bottom of the lower plenum of the core cavity includes assemblies of solid ceramic materials, some of which provide the base for the graphite core support posts. Hence, those ceramic blocks must sustain both high compressive loads and high temperatures. Other, non-load-bearing ceramic blocks are used between the support posts to cover panels of fibrous insulation. Candidate support pad ceramics are high-density alumina, fused silica, and silicon nitride. Typical candidate cover blocks are fused silica, silicon oxynitride, and silicon carbide.

Because they serve highly important functions, the ceramic materials that are finally selected for use in the Class C thermal barrier must be very well characterized. Properties must be established by sufficient testing to assure that these materials, which are basically inhomogeneous and brittle, will serve reliably. The properties data required by the design engineers include both time-dependent and time-independent factors. Tests must be done to determine the basic fracture strength, creep strength, fatigue and creep-fatigue behavior, thermal shock properties, modulus of elasticity, Poisson's ratio, and time-dependent fracture mechanics behavior. In addition, tests must be performed to evaluate the effects of long-term exposure to the service environment.

- Turbomachinery - For the case of the HTGR-GT, the materials technology work necessary to support the design of the turbomachine will be done, basically, by the subcontractor. However, some tests are included in this program. These include tests to determine the extent of gas corrosion of the candidate turbine blade and vane alloy, IN100, in the HTGR environment, and the effects that such corrosion will have upon the creep-rupture strength, fatigue strength, and fracture toughness of the alloy.
- Heat Exchangers - Materials technology work will be required to support the design of steam generators, IHXs, CAHEs, and recuperators, as appropriate to the plant design. The principal candidate materials for these components are the low-alloy chrome-molybdenum steels, 12% chrome steels, Alloy 800H, the

austenitic stainless steels, Hastelloy X, and Inconel 617. The test data that are required by the design engineers include friction and wear behavior (support areas of the heat exchanger tubes are subject to rubbing action due to thermal expansion and contraction), fatigue and creep-fatigue data, tensile strength data, thermal aging behavior, and creep-rupture data for these alloys, including the welds. In the longer term, confirmatory test data must be provided (e.g. >50,000 hr) for validation of extrapolations used in the design. Such tests include creep-rupture, fracture mechanics data after long exposure to service environments, etc.

Among the most difficult of the materials issues which must be resolved is the question of potential carburization of the tube materials for the heat exchanger that must operate at the highest temperature: the IHX. Gas-corrosion tests have shown that all of the currently available candidate wrought alloys are susceptible to carburization in the simulated HTGR primary coolant environment at 850°C (1562°F). None of the wrought alloys appear feasible for a 950°C (1742°F) IHX. Several approaches are being pursued to resolve the issue, such as coating, cladding, modifying the coolant, and modifying or developing new alloys. It is planned to continue this work on a high-priority basis. Modification or development of new carburization-resistant alloys would also be an approach to finding an acceptable material for operation of an IHX at 950°C (1742°F).

Since the heat exchangers in the primary coolant circuit will be designed and constructed in accordance with the ASME Code, it will be necessary to perform a significant amount of tensile, creep-rupture, creep-fatigue, and fracture mechanics tests to qualify the materials selected for operation at 850°C (1562°F), and above. The current rules and materials extend only to 815°C (1500°F), and the principal candidate for the IHX, Inconel 617, has not yet been qualified under the Code.

- Reactor Internals - The principal candidate metallic materials for components of the core lateral restraint and peripheral seal mechanisms are Alloy 800H, Inconel 718, Inconel 617, and Hastelloy X. These mechanisms must operate reliably at very high temperatures (up to the core outlet temperature) for the full life of the plant.

At the present time, the design for the core lateral restraint system springs requires materials having strength and stress-relaxation-resistance properties that are not available for service temperatures of >850°C (1562°F). An advanced materials screening program is in progress which is expected to help identify suitable materials for these components.

The tests on the candidate materials which are planned to provide the properties data required by the design engineers include tensile creep-rupture, low-cycle fatigue and creep-fatigue, stress relaxation, static adhesion, and fracture toughness tests.

The planned materials testing activities described above are only those required to support the design of some of the key components that must operate at the highest temperatures in the reactor primary coolant circuit. It is recognized that there are several other high-temperature components for which materials data will be required by the designers. Among them are the control rods, upper plenum elements, primary circuit control valves, circulators, etc. In addition, some materials properties data will be needed for design of lower-temperature structures, such as the PCRV closures, liner steels, tendons, etc. It is not presently expected that these lower-temperature materials issues will be significantly limiting.

A.2.5 Engineering Technology

- Scope - The engineering technology task includes the development of basic design technology, computer methods, and criteria for major NHS components and systems, including the HTGR fuel, core, reactor internals, reactor vessel, heat exchangers, mechanisms, and electrical systems.
- Objectives - The objective of this task is to provide the basic technology necessary to support design, development, and verification of HTGR components and systems with regard to structural, functional, and performance requirements. Particular emphasis is given to the establishment of technology common to HTGR applications.

A.2.5.1 Methods Development

- Status - The majority of the computer programs needed for the design and development of the HTGR in the areas of fuel and core, reactor internals, PCRV, coolant system, and system components have been developed and are partly documented and verified. New code development and code improvements are required primarily for the resolution of current and new design issues. In these areas, significant progress has been made.

Practically all reactor physics and fuel performance codes have been updated for LEU fuel calculations. Flow and natural circulation codes for flow distribution, pressure drop, and hot streak analysis and preliminary acoustic emission methods for component vibration assessment have been completed. Preliminary versions of the heat exchanger sizing and performance codes have been completed. A final user's manual was issued for the helical coil tube

bundle structural analysis code. An improved static structural finite element computer program has recently been completed which features non-linear analysis capability and is primarily designed for fuel element stress analysis. A dynamic version is near completion. The core seismic program model testing and code development tasks have been completed except for final verification of the multicolumn analysis code. Documentation summarizing the total program was issued, and an LTR on the core seismic methods verification is under preparation. An HTGR version of the reactor emergency core cooling code was completed and compiled; however, several GASSAR-ISER (interim safety evaluation report) issues concerning flow and hot streak calculations remain unresolved. An array processor for use with the UNIVAC 1110 was installed, which will result in more efficient use of large structural core seismic and physics codes and reduced running costs.

- Planned Program (Fig. A.2.5-1) - The remaining methods development, code updating, and maintenance and documentation required to support the HTGR design and the validation of these design methods will be completed prior to PSAR submittal. The main activities are described below.

An effort will start in FY 1981 to develop a new three-dimensional power distribution core design code to improve the efficiency and accuracy of spatial flux distributions in the core. Another important task in this area is to improve fission product transport and plateout analysis capability to include treatment of multiple species and in-diffusion in order to better evaluate component maintenance.

An effort to develop turbulent flow code capability for calculating local heat transfer in liner components due to hot streaks in the core outlet flow will continue. Work will also continue on the development of methods for calculating natural convections and temperatures in primary circuit cavities, including a stand-by CAHE, the upper plenum during loss of forced cooling, and in heat exchanger cavities during loop shutdown.

The acoustics analytical and test development program will also be continued to provide more information on noise levels generated from circulators, orifice valves, core support block jets, and turbomachinery and on propagation of the acoustic waves through primary and secondary systems, including the effects on reactor internal and primary loop components.

Of main concern is the development of conceptual/preliminary design methods for heat exchangers, which have not been completed. These include integrated design equations for higher-temperature structures with complex interactions between components, seismic design methods for straight tube and helical tube bundles for parametric

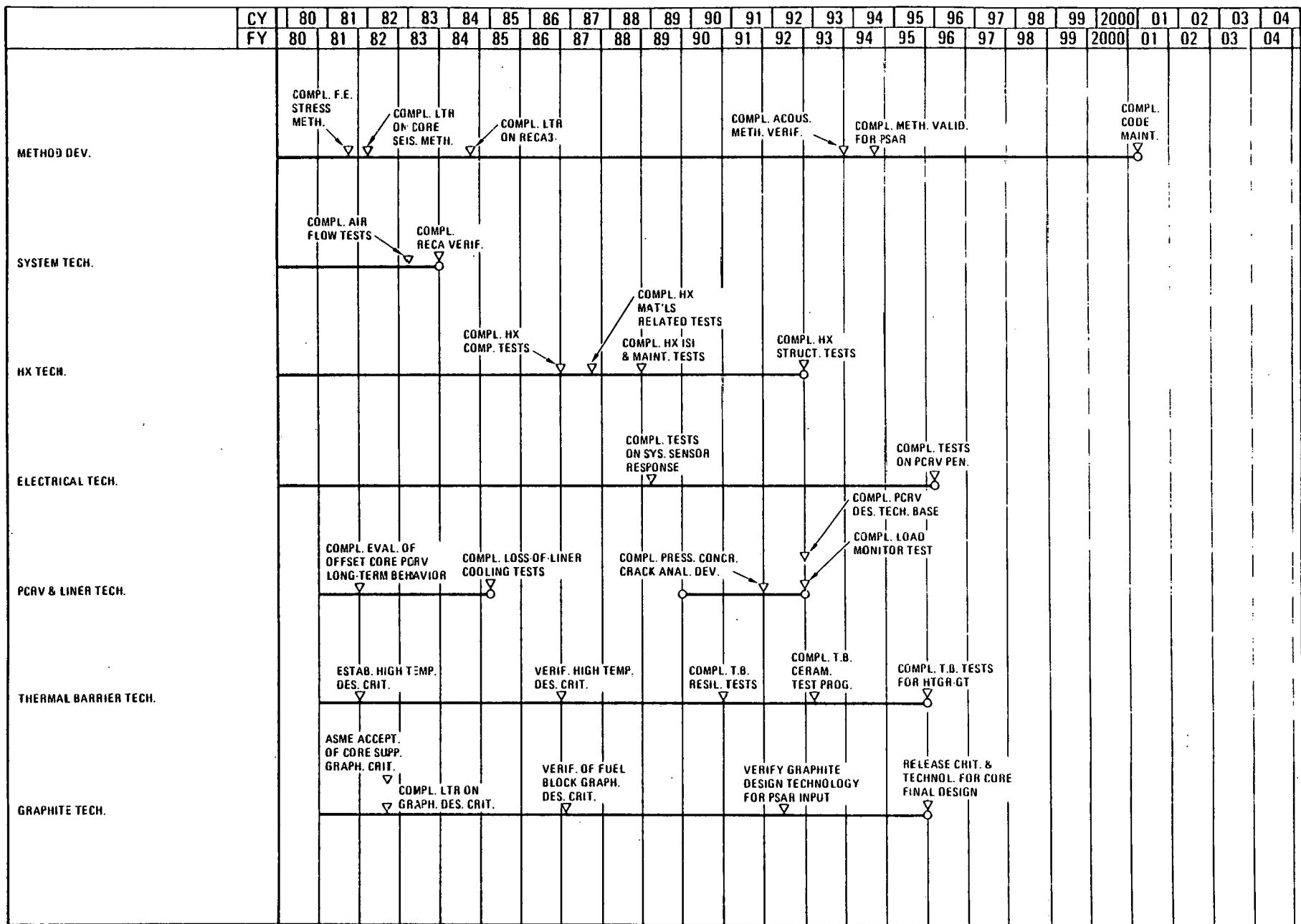


Figure A.2.5-1 HTGR Generic Technology Program detailed engineering technology milestone schedule: HTGR-GT

studies, and a performance analysis computer program for helical coil finned tubes and axial flow heat exchangers.

To complete the core seismic program and present it to the NRC as part of the licensing preapplication review, it is necessary to complete the development and verification of the multicolumn code and to complete the LTR on the verification of the core seismic methods. A computer program to determine multi-building response and interaction to seismic excitation is also planned.

The effort to complete the development of non-linear finite element methods in order to more accurately calculate fuel element stresses will continue. This includes the conversion of a non-linear dynamic finite element code developed at Lawrence Livermore Laboratories.

The development of mesh generation and computer graphics methods plays an important part in aiding the engineering analysis and saves design costs. This activity is planned to continue as well as procurement of hardware for plotting and display.

The reactor emergency core cooling analysis code will be modified in the areas of heat transfer and fluid flow and hot streak modeling as a response to GASSAR-ISER questions in support of CACS licensing activities. Also, a computer program will be developed to predict the probability of successful CACS performance and performance margins. An LTR on reactor emergency core cooling analysis modeling and verification will be written and submitted to the NRC.

Code development in support of resolving other reactor system issues include computer programs to predict the effects of water ingress and oil ingress.

A.2.5.2 Systems Technology

This task consists of upper and lower plenum flow distribution tests to obtain qualitative and quantitative data to support analytical assumptions and modeling for flow, pressure drop, and thermal mixing in the plenum and cross ducts.

- Status - Flow tests with water have been completed on 1/20-scale models of the upper and lower plenums of the 3000-MW(t) HTGR-SC reference design with six steam generator loops and three CAHE loops. Testing was carried out at ambient conditions of pressure and temperature.
- Planned Program (Fig. A.2.5-1) - Similar tests with air on 1/4-scale models are planned to obtain further data for correlation with analytical models.

A.2.5.3 Heat Exchanger Technology

This task includes test programs for obtaining basic data which are generic to all heat exchanger designs. The tests fall in four main categories: materials-related tests, including material creep-fatigue and weld properties; structural tests to determine wear, seismic, damping, and flow-induced vibration characteristics; component tests to obtain flow distribution and pressure drops; and ISI and maintenance tests to demonstrate inspection methods and procedures.

- Status - An extensive DV&S program relating to steam generator design technology was completed for the development of the FSV and LHTGR steam generators carried out at GA and at the facilities of licensees such as CEA in France and Sulzer Brothers in Switzerland, and by associates in previous design endeavors, such as Foster Wheeler Corporation in New Jersey. These test programs focused on developing technology in the areas of thermal sizing, steam generator performance and stability, heat transfer and fluid dynamics, structural integrity, and materials.

More recently, for LHTGR designs, an overall generic test plan for heat exchanger DV&S has been issued. The plan summarizes the individual test programs with respect to technical requirements, schedule, and cost.

According to this plan, an effort has been initiated to advance the following tests: tube fretting and wear, stayed tubesheet air flow, finned tube heat transfer and pressure drop, and tube bundle grid pressure drop tests.

- Planned Program (Fig. A.2.5-1) - With the exception of a few tasks, the overall DV&S program must be completed prior to the end of the preliminary design period so that final design and design analysis are not delayed. Several tests are needed to obtain design information for the conceptual and preliminary design. This includes data on properties of weld materials, creep-fatigue design, fretting wear, and finned tube heat transfer and pressure drop, as well as load path and damping properties determined in seismic tests.

A.2.5.4 Electrical Technology

Two test programs are required to support the final design of electrical systems for control and instrumentation and PCR/V penetration design.

Response tests of control and electrical system sensors to verify sensor time constants are required for the plant control system, plant protection system, data acquisition system, and analytical instrumentation systems, since system performance is largely dependent on these parameters.

Testing and development of penetration configurations, cable routing techniques, and materials are required to design PCRV penetrations for control and power cables running from ambient conditions into reactor interspaces where elevated pressures and temperatures and radiation or other extreme environments are encountered.

A.2.5.5 Mechanical Technology

- Status - The work in this area is principally limited to definition of ISI and maintenance requirements and conceptual design of equipment to perform the necessary ISI and maintenance operations. Another task is to define the solid waste handling requirements, which are expected to be significantly less demanding than for an LWR.

A compilation of ISI requirements has been assembled which is in accordance with the ASME Code, Section XI, and a preliminary assessment of ISI and maintenance equipment requirements has been made.

- Planned Program (Fig. A.2.5-1) - The ISI and maintenance requirements will be updated based on the latest HTGR concepts, and conceptual designs of ISI and maintenance equipment will be generated. This information will be used by HTGR component designers to ensure in their designs that the required ISI and maintenance operations can be performed and by the architect-engineer and customer to provide the necessary facilities and equipment. The solid waste handling requirements specification will be prepared for similar use.

A.2.5.6 PCRV and Liner Technology

- Status - The design technology and criteria and component DV&S for the PCRV and liners of the LHTGR have undergone substantial improvement since FSV, not only from the standpoint of the computer codes mentioned above but also as a result of advances in the design data base. Tensile tests of 2500-kip strand tendons for the linear prestressing system have been completed, as have relaxation tests of prestressing steels. Also, fatigue tests of liner anchor studs and static and fatigue shear tests of cooling tubes welded to the liner have provided basic design and verification data. An interim position has been developed on the treatment of fracture toughness of liners, which will need to be reviewed and possibly revised following completion of fracture toughness testing of liner materials at ORNL.

Linear and biaxial buckling tests of liner material backed by concrete were completed at CEA, providing valuable design data for the PCRV liner, which is held in compression by the inward shrinkage and creep displacement due to prestressing of the PCRV concrete.

A three-dimensional finite element analysis of a representative offset-core PCRV arrangement is under way to assure that the long-term behavior of such a PCRV is acceptable. Initial results are generally as expected, and the analysis is scheduled for completion in FY 1981.

Conceptual design drawings have been completed for a load monitor for the PCRV circumferential prestressing system, and a feasible scheme was designed which would permit removal and replacement of the monitor without removing the prestressing strands.

- Planned Program (Fig. A.2.5-1) - The analysis of the long-term behavior of an offset-core PCRV will be completed in FY 1981. An evaluation will be done of methods of analyzing postulated pressurized cracks in the concrete to develop a standard approach, and analytical models will be developed for analyzing PCRV crack problems. A confirmation test will be performed on the prestressing load monitor design, and if a PCRV model test is required, support will be provided in determining model design requirements and preparation of test plans, specifications and procedures, and test evaluation. Tests are also planned for a 3000-kip tendon needed for more compact PCRV configurations.

For the liner, analysis of the plastic deformation of a typical penetration at the penetration-to-concrete interface will be done in response to previous NRC concerns, and a preliminary fracture toughness analysis will be done for a typical penetration. Additional generic evaluations and brief analyses will be performed leading to updated liner design criteria and fracture toughness criteria, including criteria for concrete closures.

Work is planned to demonstrate that neither access for direct liner ISI nor a liner leak detection/collection system is required for an HTGR plant. The consequences of postulated liner leaks will be determined, followed by development of concepts for optional liner leak detection/collection systems.

A.2.5.7 Thermal Barrier Technology

- Status - Design technology and criteria and component DV&S for the thermal barrier are covered in this area. In addition to the experience gained through FSV, tests and evaluations have advanced the state of the technology for thermal barrier. Long-term (20,000-hr) resiliency and thermal cycling tests of fibrous insulation materials have been completed both at GA and CEA, and emergency and faulted condition tests are currently being performed at CEA. Fatigue tests were done on attachment fixtures, leading to an improved design. Accident condition tests have been done for both Class A and Class B assemblies, and a 0.6-scale hot duct was tested for about 400 hr at around 815°C (1500°F) at CEA. A full-size hot duct was fabricated and assembled at the conclusion of the cooperation agreement between GA and CEA.

Depressurization tests were run for FSV, but additional tests will be required for new materials and to encompass the very high depressurization rates associated with HTGR-GT turbine failures. Preparations are under way at ORNL to test dense ceramic specimens under high-rate depressurization.

Structural tests and evaluations of full-size ceramic support pads for the core support post seats resulted in a simplified and improved design configuration. Also, screening creep tests of small ceramic specimens have narrowed the range of candidate materials.

A general vibration analysis was done for typical thermal barrier assemblies, and acoustic vibration testing was initiated in FY 1980.

Under the GCFR program, tests were done and repeated twice in which the thermal barrier was flooded with water and then was dried out, with minimal degradation of the thermal performance. The HTGR Generic Technology Program provided an assessment of the effect of oil contamination based on data available in the literature and from reports on the Peach Bottom reactor.

Work is in progress to define and describe typical applications of various metallic and ceramic materials and to identify and provide the pertinent design properties for those applications. Closely following this work will be revisions to the high-temperature design criteria for each type of material in thermal barrier applications.

- Planned Program (Fig. A.2.5-1) - Thermal barrier DV&S tests will continue, beginning with the more fundamental tests on simple specimens of individual component materials (e.g., the structural ceramic creep test specimens) and progressing as appropriate through tests of typical subassemblies to tests of arrays of full-size thermal barrier panel assemblies. Structural and thermal tests on candidate ceramic materials for the core support post seat base pads will be completed. Nondestructive examination tests will also be completed to develop a reliable means of acceptance inspection of these components, which are prone to have internal flaws and residual stresses from the fabrication process.

Tests will be conducted to evaluate candidate materials and assemblies for the higher-temperature Class B thermal barrier associated with recent HTGR plants (e.g., carbon-carbon and cast cover plates). In addition, permeation and depressurization tests will be conducted to ensure that the helium will not flow through the thermal barrier and carry excessive heat to the liner, yet will vent sufficiently to avoid functional destruction of the thermal barrier during a depressurization accident.

Vibration testing will complete evaluations of subassemblies of alternative materials and conclude with tests of full-size assemblies of the selected materials and components. These tests are necessary to ensure that the thermal barrier can sustain flow-induced and acoustically induced vibration and maintain its design function for the design life of the plant.

The different classes of thermal barrier will be tested for thermal performance under helium pressure and pressure gradients to ensure proper functional integrity under reactor operating conditions.

Long-term resiliency and thermal cycling tests will be repeated periodically to detect changes in the fibrous insulation resulting from variations in raw materials and processing changes in manufacture.

In parallel with the test program, as new information becomes available, updated versions will be published of design guides to thermal barrier applications of the various ceramic and metallic materials, including the pertinent material properties, and structural design criteria documents for each of these types of materials.

The final tests planned in the program will be design verification of full-size multi-panel arrays of typical thermal barrier assemblies, mock-up studies of the more complex and critical configurations, and heat transfer and accident condition tests of full-scale Class B assemblies.

A.2.5.8 Graphite Technology

- Status - In this area, the material modeling, methods verification, and materials property data base provided by the Graphite Materials Program are combined with design application experience for the graphite fuel blocks and core support to formulate, verify, and promulgate structural design criteria to be applied to the graphite components. Early versions of the design criteria for FSV and the LHTGRs were rather simple and, in recognition of the relative lack of understanding in this area, relied on large safety factors and general statements such as "no loss of safety function." Many felt such an approach was sufficient because of the successful experience with graphite components in British advanced gas-cooled reactors. However, in those CO₂-cooled reactors, the graphite was used as a moderator but not as a structural element, temperature swings were less than in an HTGR, and there was no requirement to design for seismic events.

In the early 1970s, it began to be recognized that design criteria based on rules well established for metals, even those for brittle metals, were unsatisfactory for graphite. The NRC contracted with FIRL to conduct a study of safety aspects of HTGR graphite components. Recommendations made by FIRL required that a much better

understanding of graphite material behavior be developed and that the new knowledge be turned into more specific design criteria for HTGR graphite components. The Graphite Materials Program will provide the necessary knowledge.

A Joint ACI/ASME Code Subcommittee was organized to prepare a Section III, Division 2, Code Subsection CE on graphite core support components. A draft of the subsection was prepared and reviewed by members of the subcommittee, and their comments have been incorporated.

New design criteria for the graphite fuel blocks, which are more complex than core support blocks, are still in the formative stage. There is no industry or regulatory structural standard for either LWR fuel assemblies or HTGR fuel blocks. However, both FIRC and the NRC have shown an intent that specific design criteria must be developed for the HTGR fuel blocks which relate directly to the particular properties and behavior of graphite.

- Planned Program (Fig. A.2.5-1) - A relatively low level of effort will be required in the future on the core support criteria to respond to comments from the ASME Main Code Committee and to incorporate material on the effects of oxidation and irradiation as it is made ready for incorporation into the Code.

A greater level of effort is required for establishment of the design criteria for the fuel block graphite. The more complex configuration of the fuel blocks and the extremes of temperature and irradiation exposure demand a more detailed and exacting treatment of the behavior of material in the various load and environmental conditions. The criteria will be presented directly to the NRC via an LTR on graphite in cooperation with the HTGR safety and licensing efforts.

A.2.6 Fuel Handling, Neutron and Region Flow Control Equipment

- Scope - This task includes the complete design and production of all design drawings and specifications required to procure, fabricate, and install all those items of equipment necessary to raise and lower the control rods and power rods, operate the reserve shutdown system (if required), and control the coolant flow through each region of the core, plus those other items of equipment necessary to accomplish periodic refueling of the HTGR plant, preparation of associated operation and maintenance manuals and design reports, and development of the supporting technical data for the PSAR and FSAR or other licensing documents. The task also includes any required design development and verification testing.
- Objectives - The objectives of this task are to provide the required equipment designs, specifications, and operating manuals

to ensure prompt, accurate operation of the core region neutron control and coolant flow control equipment and to perform the necessary refueling operations safely and in the minimum practical time, to minimize radiation exposures to personnel and equipment, and to minimize the technical and cost impacts on interfacing systems and components.

• Status

- Fuel Handling Equipment - HTGR fuel handling equipment has progressed through an evolutionary process from the design used at FSV, with each change intended to minimize refueling time. However, the most recent change, to an invessel refueling system, was aimed at simplifying refueling operations, increasing flexibility to accommodate different fuel management schemes, reducing net capital cost, and further reducing the already low operator doses for the HTGR, while not increasing refueling time. The basic mechanisms for handling the fuel elements and control rod drives are modifications of FSV designs based on experience. Therefore, the operational portions of the fuel handling machine and auxiliary service cask are well established, and basically only the structural and shielding portions are undergoing change. There are new mechanical devices, however, which operate inside the PCRV and in a new temporary fuel storage vault, that have been defined only in a simple conceptual manner at this time. In addition, there are detail changes relative to FSV which affect the fuel handling equipment. For example:

1. The PCRV head thickness, core height, and refueling penetration configuration require modifications to the fuel handling machine to provide additional radial reach capability.
2. Fuel element modifications which incorporate additional dowels and estimates of increased bowing after irradiation necessitate changes in the fuel handling machine and the fuel transfer equipment.
3. Thicker PCRV top heads (due to higher pressure and larger core cavities) and the increase in core cavity height to accommodate the in-vessel refueling necessitate increased vertical reach for the fuel handling machine and auxiliary service cask.

Testing and operational experience have been gained on the FSV fuel handling system. Additional testing is required in the development of current concepts, and a comprehensive qualification test program is planned for the current fuel handling system because of the differences mentioned above.

- Neutron and Region Flow Control - In preparing the designs for the Fulton and Summit HTGR power plants, certain basic design improvements were incorporated compared with the FSV design: using a torque motor instead of an induction motor, using a different gear train, using grease-lubricated rather than dry-lubricated bearings, changing the orifice valve drive from an Acme screw to a drum and cable mechanism, and changing the reserve shutdown system control gate from a pneumatically powered rupture disc arrangement to an electrically controlled gate. The main change since that time is the addition of drives for the power rods. Very little effort has been expended on neutron and region flow control design since 1978. However, since the functional and performance requirements are essentially generic, much of the mechanism design effort and many of the released drawings for the previous HTGR-SC plant are considered applicable to the latest options. The major areas requiring additional conceptual design work are the power rod drive mechanism, the movable startup detector drive mechanism, and changes in the housing and attachment flanges to effect a minimized standardized length. In addition, operating experience at FSV has demonstrated that heat loads to the refueling penetrations are highly dependent upon relatively small flow paths through structural joints, clearance holes, etc., within the control and orificing assembly. Elimination of these flow paths requires minor design changes.

A conceptual design study has been initiated aimed at substituting a high-temperature fission chamber for the unsatisfactory self-powered neutron detectors in the in-core flux measuring unit (IFMU) design. The fission chamber is physically larger and requires a length of relatively rigid electrical cable, which impacts the lower end of the control rod drive and the reserve shutdown system hopper. Also, changes in the top head thickness of the PCRV and adoption of the in-vessel refueling scheme have increased the overall length of the control rod drive and orifice valve units, which results in a requirement for a much taller auxiliary service cask in the refueling system to handle the control rod drives. This, in turn, requires a greater height above the PCRV in the containment building. A design study conducted during FY 1980 concluded that the length of the control rod drives could be shortened and standardized if the attachment flange could be moved down into the refueling penetration near the bottom of the top head. The details of such an attachment must be worked out, and it must be verified that the penetrations can be suitably modified.

For the control and instrumentation components of the neutron and region flow control system, development of the overall system operating philosophy and conceptual design has been nearly completed. Specifically, the conceptual design of the

following portions of the power rod control system has been completed: switching and control circuitry for manual or automatic operation of the power rods in banks or individually, slack cable and other electronic logic systems, and drive motor electronic logic and power modules. Similar conceptual designs have been completed for the control rod pair control system. For the reserve shutdown system, a manual switch and circuitry design is complete for operation of the fusible links that operate the reserve shutdown system hoppers and for test circuitry to check fusible link circuitry integrity.

The conceptual design of orifice selection and orifice control panels for the region flow control system has been completed, including the panel interfacing logic circuitry.

The neutron measurement system of the startup nuclear detector channels and startup detector motor drive logic, the automatic flux control system that inputs into the plant control system, and the thermocouple arrangement that measures core outlet temperature for monitoring plant performance have also been designed to the conceptual stage. However, tests at CEA have shown that the self-powered neutron detectors (SPNDs) used to measure flux levels in the core are unsatisfactory owing to a very low signal-to-noise ratio. Exploratory tests with promising results were completed in FY 1979 on a fission chamber device made by Toshiba of Japan. Plans are being made to test a high-temperature [800°C (1472°F)] fission chamber which is being prepared by Toshiba.

During the early 1970s, a series of tests was done on the control rod drive system to evaluate the effects of the principal differences from the FSV design mentioned above:

Phase 1 - A general checkout in air for assembly, installation, and operating characteristics in a simulated refueling penetration and core region mock-up.

Phase 2 - Cyclic operating tests of 630,000 jogs and 5400 scrams, performed in simulated reactor conditions of helium and temperature.

Phase 3 - Periodic cyclic operating tests of the grease-lubricated assembly under simulated reactor conditions of helium purity and temperature to evaluate the lubricant.

The first series of tests did not include an orifice valve and drive mechanism and did involve a FSV-type of reserve shutdown system hopper gate. Subsequent changes to the tested design include incorporation of variable orificing, the redesigned reserve shutdown system hopper gate, incorporation of power rods, and several minor configuration and structural changes.

For the controls and instrumentation components, side-entry core outlet temperature thermowell and thermocouple insertion tests have been performed, which showed a need to incorporate larger thermowell tubes. Also, a joint GA/CEA DV&S program on the self-powered neutron detectors for the in-core instrumentation system showed the rhodium-based detectors to be unsatisfactory owing to an inadequate signal-to-noise ratio.

• Planned Program (Fig. A.2.6-1)

- Fuel Handling Equipment - The near-term need in this program is to complete the conceptual design of all those new items of equipment and previous components which have changed as a result of the in-vessel refueling system sufficiently to identify the major parts and materials, permit a conceptual cost estimate, and define the interface requirements. This effort must begin with resolution of the location of the temporary fuel storage vault, either below the containment base mat, which is simplest for fuel handling, or to the side of the containment building, which may be preferable for construction scheduling.

Next, the design will be further developed in cooperation with designers of interfacing systems, shielding requirements will be updated, and data will be provided for the PSAR.

During the preliminary design phase when major design details will be worked out for each component and system operation will be refined, component tests will be performed to develop:

1. Fuel handling instrumentation and controls.
2. Longer vertical reach and horizontal reach capability of the fuel handling machine grapple arm and longer reach for the auxiliary service cask mechanisms.
3. Reliable operation of the plenum (in-vessel) fuel block transporter mechanisms and the elevator/hoist assembly.
4. Reliable operation of the fuel container loading and sealing equipment.

At some time during the preliminary design phase, it is anticipated that a vendor will be selected to complete the design, fabrication, and checkout of the fuel handling system with liaison by GA.

During the final design phase, detail design drawings and specifications will be completed to support fabrication and assembly of the components of the system, operation and maintenance manuals will be prepared, and final design reports

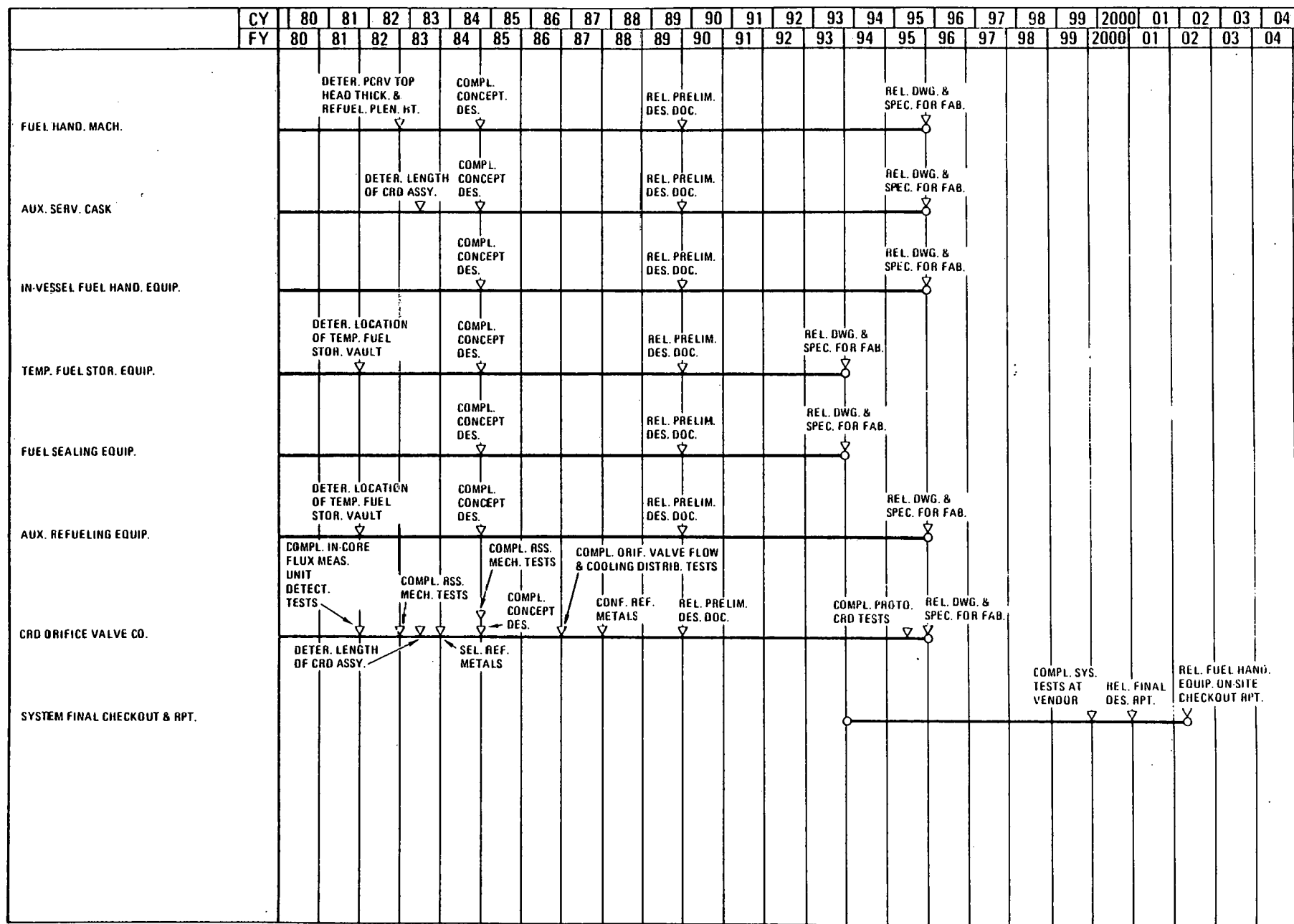


Figure A.2.6-1 HTGR Generic Technology Program detailed fuel handling and CRD equipment milestone schedule: HTGR-GT

will be assembled. Each major operational component will be tested individually and in a full system test before the fuel handling system is delivered to the site. Once installed at the site, the complete fuel handling system will be thoroughly checked out and final revisions will be made to the design reports and operation and maintenance manuals as necessary.

- Planned Program (Fig. A.2.6-1)

- Neutron and Region Flow Control - The most urgent requirement for the conceptual design phase is to confirm the applicability of the Toshiba high-temperature fission chamber to the IFMU and to incorporate that instrument into the control rod drive assembly design, along with necessary changes to the reserve shutdown system hopper valve. In addition, a decision must be made whether to move the attachment flange near the bottom of the refueling penetration and incorporate that into the design.

Owing to the higher core inlet temperature associated with reactor designs other than the HTGR-SC/C, some parts of the control rod drive and orifice valve assemblies will have to be made of different materials which will withstand the higher service temperatures. It is the intent, if at all possible, to select materials which will also suffice for follow-on use in plants having a 950°C (1742°F) core outlet temperature. Selection and incorporation of the new materials will take place as early as possible in the design so that any unsatisfied design data needs will be identified and incorporated into the Materials Program in time to provide data for the final design.

Test plans, specifications, and procedures will be prepared, and tests will be run on the high-temperature Toshiba fission chamber; the orifice valve, for flow control and coolant distribution; the reserve shutdown system hopper, for operation and proper pellet flow; and for alternative means of measuring core outlet temperature which require less room for insertion and are potentially more reliable than the present thermocouples. In connection with the last item, the alternate core support designs being considered to meet the HTGR-GT rapid depressurization transients may require a re-evaluation of the core outlet temperature measurement scheme. Consequently, it may be necessary not only to consider alternative instruments but also different avenues to reach the temperature measurement point. The applicability to all HTGR options would then require review.

It is anticipated that early in the preliminary design phase a vendor will be selected to complete the design, development, testing, fabrication, and checkout of the neutron and region flow control system under the supervision of GA designers and project personnel.

During the preliminary design phase, a series of component development tests will be planned and performed:

1. Installation and moist environment operation tests of the control and orificing assembly to assure trouble-free installation, function, and removal under maximum design tolerances for misalignment due to construction tolerances for the refueling penetrations, control and orificing assembly, and core offset resulting from accumulated gaps between elements. The moist environment will check the cumulative effects of the atmospheric moisture exposure during fabrication, shipment, and installation and the low moisture level in service.
2. Control rod and power rod drive mechanism and controls tests to determine mechanical efficiency, position accuracy, torque, power requirements, operating times, and changes resulting from extensive cycling in earlier tests.
3. Mechanical cycling tests under environmental conditions for the orifice valve to ensure the valves meet the design criteria and perform satisfactorily during the 8-yr service life with minimum unavailability.
4. Functional tests of the reserve shutdown system under simulated reactor helium and temperature, including the effects of vibration.

During the final design phase, detail design drawings and specifications will be completed to support fabrication and assembly of the components of the system, operation and maintenance manuals will be prepared, and final design reports will be assembled. After manufacture, each major component will be functionally checked before delivery to the site, and after all components are delivered to the site and installed, a complete system checkout will be performed. Final design reports and operation and maintenance manuals will be revised, if necessary, following the checkouts.

A.2.7 Safety and Licensing

- Scope - The safety workscope tasks include probabilistic risk assessment (PRA) methods development, accident initiation and progression analysis (AIPA), and application to the HTGR design as well as safety research and computer code development. Generic licensing activities mainly include preapplication review with the NRC on prominent design and safety issues and general support to design organizations.

- Objectives - The main objectives of the safety and licensing tasks are to ensure that the HTGR generic design features meet applicable safety and design criteria. Furthermore, in recognition of the inherent design and safety features of the HTGR, it is sought to amend existing NRC General Design Criteria, Regulatory Guides, and Siting Criteria for HTGR applications.
- Status
 - Safety - Advances in PRA of the HTGR are contained in the AIPA Phase II Status Report issued in FY 1979, which is considered the equivalent of WASH-1400 for LWRs. A large part of the AIPA report is devoted to core heatup studies, considering a broad range of plant accident sequences. The overall probability of core heatup for HTGRs was assessed at about 3×10^{-5} /reactor-year. The AIPA results have largely been confirmed by the German Safety Study (PSH) completed in FY 1980.

Subsequent to the completion of the AIPA report, methods development and risk assessment of accident sequences initiated by major plant fires were completed.

Two safety-related LTRs have been issued for NRC preapplication licensing review. These include interpretation of General Design Criteria for HTGRs and the use of PRA in the selection of design basis accidents. The continuation of the latter task has resulted in several published reports on quantitative safety goals for nuclear reactors, which are concurrent with the national effort under the NRC action plan following the Three Mile Island accident.

Safety research is an ongoing program which provides data for assessment of generic HTGR accident consequences (especially for core heatup scenarios) in three areas: (1) core material redistribution and fission product release, (2) fission product plateout in the PCRV, and (3) containment atmosphere response.

A series of laboratory tests has been documented on the release characteristics of important fission products from fuel particle kernels with failed coatings during core heatup conditions. This work is intended for use in obtaining licensing credit for time-dependent release from failed fuel for the maximum hypothetical fission product release (MHFPR) siting event and in AIPA risk assessment studies.

A computer program has been developed to analyze time-dependent plateout of fission products along specified flow paths in the PCRV before release to the containment during core heatup. PCRV plateout tests have been completed under static conditions and for dynamic conditions in flowing helium for containment atmosphere response code verification.

The analytical program for containment atmosphere response was initiated in 1975 to develop methods of evaluating containment phenomena. The program so far has demonstrated the effect of PCRVR blowdown gas mixing and heat transfer in the containment on the peak containment pressure response to PCRVR depressurization. This program has also focused on the development of analytical models for depressurization jets and their effects on the containment structure.

In addition, chemical composition response of the containment atmosphere during core heatup has been investigated and documented. The current workscope includes fission product plate-out and fallout in the containment, including interactions with aerosol transport, agglomeration, and attachment to containment walls.

Safety-related computer programs have been developed based on AIPA results, recommendations from the NRC, national laboratories, and others. Recent work has concentrated on code development to analyze a DBDA with a steam ingress and an iterative method between codes to more realistically predict the core temperatures and fission product releases during core heatups, such as for the MHFPR.

- Licensing - Currently, generic licensing activity for the LHTGR is confined to the establishment of a preapplication review program for generic HTGR issues.

The proposed program, consisting of ten generic issues for resolution, has been neither accepted nor rejected by the NRC. The NRC has indicated unofficially that part of the topics might be accepted for review, but no commitment has been made. Nevertheless, work has proceeded on a series of LTRs submitted to the NRC in accordance with the procedure for their LTR program. The objective of each proposed LTR is summarized below:

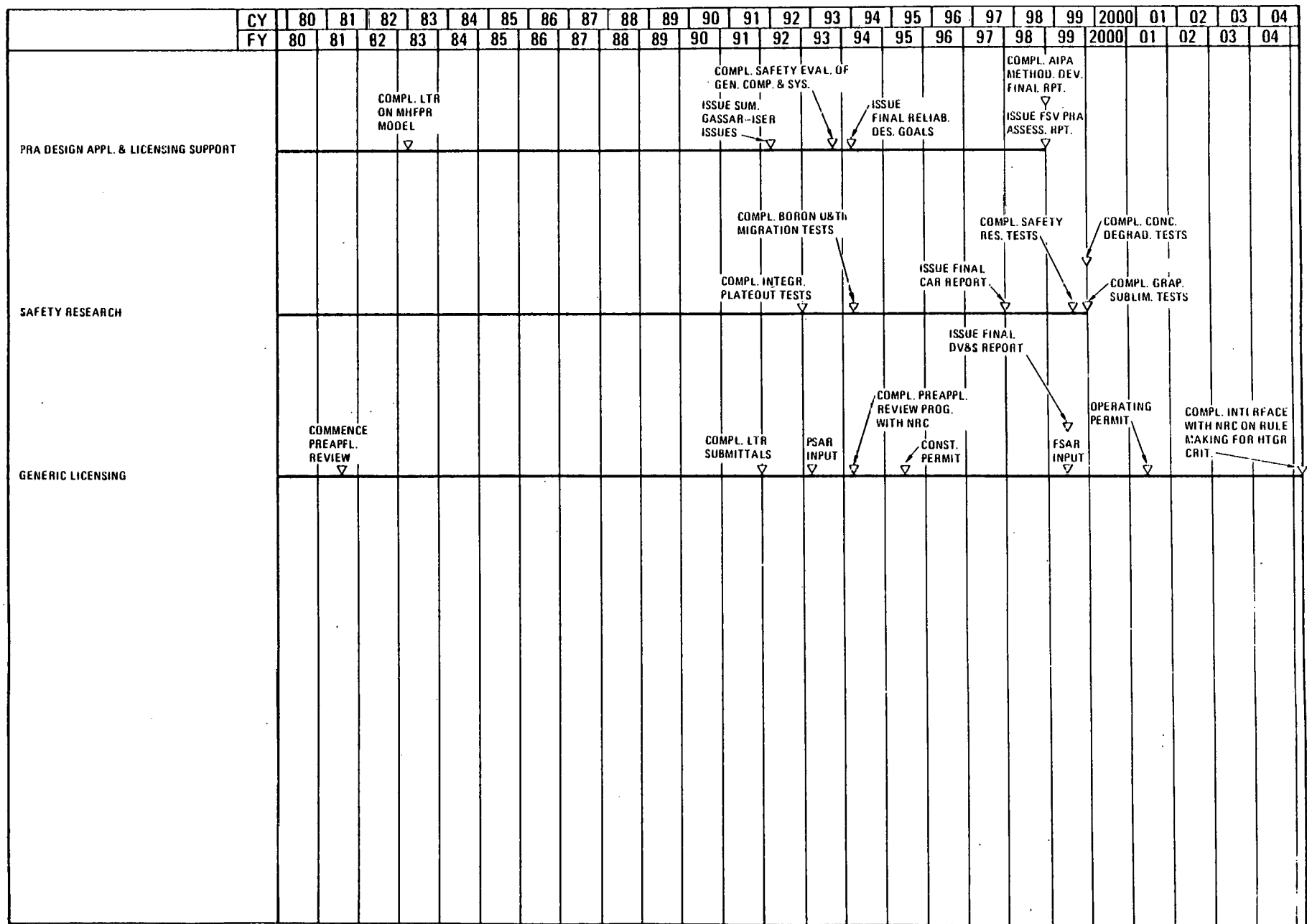
1. Core Seismic Analysis Methods: to obtain endorsement that the described methods and computer codes are acceptable for use in seismic analysis of the HTGR core and core support structure.
2. HTGR Fuel Performance Models for Use in MHFPR Analyses: to obtain endorsement that the performance models described for HTGR fuel are acceptable for use in SAR analyses of the MHFPR siting event.
3. Measurement and Modeling of Fission Product Release from HTGR Fuel Particles under Accident Conditions: to obtain endorsement that the described data and model are acceptable for use in the SAR analyses of the MHFPR.

4. MHFPR Model for the HTGR: to obtain endorsement that the described MHFPR model is acceptable to satisfy 10CFR100 requirements for site analysis.
5. Interpretations of General Design Criteria for HTGRs: to obtain endorsement of a set of modified General Design Criteria intended for application specifically to gas-cooled thermal-reactor nuclear power plants, based on interpretations of the current General Design Criteria as presented in Appendix A to 10CFR50.
6. Interpretation of Reactor Site Criteria for HTGRs: to obtain endorsement of an interpretation of Reactor Site Criteria presented in 10CFR100 intended for application specifically to gas-cooled thermal-reactor nuclear power plants.
7. Application of PRA in the Selection of DBAs: to obtain endorsement that the method is acceptable for use as a supplementary procedure in the selection of DBAs analyzed in Chapter 15 of HTGR SARs and that the PRA methodology of the AIPA study is acceptable as a supplemental method.
8. Selection of DBAs: to obtain endorsement that the proposed list of DBAs will be the list used in Chapter 15 of the HTGR PSAR.
9. Graphite Design Criteria: to obtain endorsement of design criteria for the stress analysis of fuel elements and the core support structure.
10. Positions on NRC Regulatory Guides: to obtain endorsement of exceptions taken or means to comply with Regulatory Guides.

To date, LTRs on the following three areas have been completed and issued: (1) measurement and modeling of fission product release from HTGR fuel particles under accident conditions, (2) interpretations of General Design Criteria for HTGRs, and (3) application of PRA in the selection of DBAs. In addition, LTRs on the following three areas are near completion: (1) core seismic analysis methods, (2) MHFPR model for the HTGR, and (3) positions on NRC Regulatory Guides. The completed LTRs have not yet been formally submitted to the NRC for review.

- Planned Program (Fig. A.2.7-1)

- Safety - Safety analysis and evaluation of prominent design issues are planned to continue in support of plant design



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Figure A.2.7-1 HTGR Generic Technology Program detailed safety and licensing milestone schedule: HTGR-GT

development. These include core support and fuel element graphite burnoff, fission product source terms and circulating activity, primary coolant impurity levels, carbon deposition, and fission product transport issues (e.g., plutonium release, strontium liftoff, and LEU fuel).

The AIPA studies will be expanded to further augment the credibility of safety claims necessary to license the HTGR. Important areas of methods development include means of terminating accident sequences involving core heatup, extrapolating risk assessment results to extremely low levels of accident probability, methods for resolution of GASSAR-ISER issues, and methods for investigating the effect of external events, e.g., flood, earthquakes, acts of terrorism, etc. The majority of these tasks have been planned as a cooperative program with Kernforschungsanlage under the DOE/BMFT auspices. A final AIPA report will be issued prior to FSAR submittal.

Final numerical safety goals will be established for the HTGR design to keep up with regulatory development to incorporate PRA into the design of nuclear plants. Reliability allocations for safety equipment and systems will be determined as well as overall goals expressed as the probability of exceeding a given dose or radioactive release.

A position on the proposed NRC siting criteria for LWRs (NUREG-0625) following the Three Mile Island accident will be completed in cooperation with licensing to seek exemption for the HTGR. The HTGR position will be written to recognize inherent safety features and differing levels of risk from Class 9 accidents for different reactor types. Hopefully, siting advantages over LWRs can be obtained by dealing with the NRC in upcoming rule-making proceedings.

Probabilistic risk assessment will continue for the resolution of issues identified in GASSAR-ISER and for the evaluation of significant FSV recorded system malfunctions. The latter effort enables design changes and corrective operating actions to be incorporated into the LHTGR. Summary assessment reports will be issued prior to PSAR submittal.

Improved availability methods are needed for analysis of passive mechanical components to satisfy an overall plant availability goal of 90%. Analysis to establish availability allocations for the plant availability specification document will then be conducted on generic HTGR components, including core and reactor internals, fuel handling equipment, the CACS, and the helium service system. A program to quantitatively evaluate the operational experience at FSV will also be conducted since it affects HTGR availability.

The safety research program in aid of core heatup consequence assessment will continue in the areas of (1) fission product transport and plateout, (2) PCRV integrity studies, (3) containment atmosphere response, and (4) recriticality effects associated with boron, uranium, and thorium migration. These activities are of fundamental importance to risk assessment and licensing of the HTGR as described below:

1. Further integral fission product plateout tests closely simulating reactor conditions are needed to study molecular iodine formations and verify the plateout code. A new furnace apparatus to accommodate large irradiated fuel bodies will be acquired for these tests.
2. In order to improve accident simulation models of PCRV failure, concrete degradation tests under core heatup temperature conditions will be conducted as well as further model development pertaining to creep, rupture, and eventual melting of thermal barrier and liner components.
3. Further developments in the containment atmosphere response program include improvements to and verification of helium jet models used to analyze jet impingement on the containment structure following PCRV depressurization. Helium discharge jet tests into air will be conducted to provide entrainment coefficients for the analytical models.
4. Boron migration tests are needed to verify the assumption in the risk assessment analysis that the reactor remains subcritical throughout a core heatup event with the insertion of one or both shutdown systems. The tests will investigate such phenomena as slumping and compaction of control rod material and BC₄ balls and boron vapor diffusion and transport.

Updating, verification, and documentation of computer codes used for the safety analysis of the HTGR must be completed prior to PSAR submittal. Specific tasks include amendment of the graphite oxidation model for DBA air/water ingress analysis, incorporation of data in the fission product release model, and modification of the code for core heatup analysis. The task also includes the submittal of amended LTRs on these models.

- Licensing - The main generic licensing near-term activity is to initiate in cooperation with GCRA the preapplication program review with the NRC in FY 1981. This activity also includes coordinating the completion and submittal to the NRC of remaining LTRs under the program and interaction with the NRC to obtain LTR endorsement within the time frame before

PSAR submittal. This program is aimed at resolving important licensing issues prior to initiation of plant licensing procedures and contributing to the shortening of the licensing process.

Continued interaction with the NRC is further required to resolve new issues identified during the design period. Certain issues such as "Interpretation of GDC's for Application to HTGR," if resolved, will require an amendment to the Code of Federal Regulations (10CFR50). In this case, the hearings and rule-making process will most likely continue until the operating license is granted.

To keep the NRC informed about HTGR DV&S program activities and progress, DV&S status reports will be published either as an LTR series or as GA technical reports. The initial report will outline the overall HTGR DV&S program plan, and the final report will be issued prior to FSAR submittal.

A.2.8 Technology Transfer

The main activities in this task are the FSV surveillance program, liaison activities under the Umbrella Agreement, and information exchange.

A.2.8.1 Fort St. Vrain Surveillance

- Scope - A number of surveillance activities have been performed on the FSV reactor since startup commenced. These activities include both plant and fuel surveillance.
- Objective - The objective of this task is to confirm the design basis for the LHTGR using the experience gained during startup and operation of the FSV reactor. Data obtained are used directly to validate computer codes.
- Status - The programs performed to date are described below.
 - PCRV Structural Response - This program generates data on the PCRV structural response, which are then used to validate the two- and three-dimensional design codes. This work has been continuing since the first pressurization of the reactor in 1971.
 - Fission Product and Coolant Chemistry - In this program, fission product data and coolant impurity data are obtained from the operating reactor and used to validate the design codes for the LHTGR.
 - Radiation Monitoring - This program consists of the collection and analysis of radiation data during reactor operation and of maintenance and refueling in order to improve the accuracy of predictions for future HTGRs.

- Steam Generator Performance - During the startup phase of the FSV plant, steam generator performance was closely monitored and compared with the predicted data for both steady-state and transient conditions.
- Non-Destructive Fuel Element Examinations - A robot has been developed which can be used in the hot cell at FSV or in the hot cell at GA. This robot performed a complete dimensional check and gamma-scan of selected irradiated elements removed during the first core refueling. Data from these examinations were used to confirm code predictions of the graphite dimensional changes and the activity levels.
- Destructive Fuel Element Examinations - For these examinations, measurements were made of fuel element fission gas release, and the fuel rods were then removed from the graphite fuel elements. Further examinations are now being made on the rods and the fuel particles in the hot cell at GA.
- Planned Program (Fig. A.2.8-1) - The planned program is based on the current schedule for refueling the plant approximately once every 2 yr. During each operating cycle, it is planned to examine representative fuel elements removed at the previous reloading and also to carry out plant surveillance in several areas. A report will be issued at the end of each operating cycle.
 - PCRV Structural Response - It is planned to continue to take strain and deflection measurements in order to correlate these data with the long-term creep predictions.
 - Fission Product and Coolant Chemistry - Surveillance will continue in order to correlate the data for operation at increased power levels and increased fuel burnup.
 - Radiation Monitoring - Surveillance will continue, again to assess the effects of increased power levels and increased fuel burnup and to provide a basis for projections on advanced HTGR designs.
 - Steam Generator Performance - On a biannual basis, the steam generator performance will be assessed, primarily to check that the performance degradation factors allowed in the design were adequate.
 - Thermal Barrier - On a biannual basis, the thermal barrier performance will be reviewed to determine whether the allowances for in-service degradation are adequate or, alternatively, excessive.
 - Plant Availability - The factors that give rise to the current low availability of the plant are complex and will be analyzed in detail in order to ascertain where design improvements are necessary.

- Fuel Surveillance - It is planned to continue the fuel surveillance program up to the examination of fuel elements from the sixth reload, which will have seen approximately the full design irradiation. The examinations will be the same as those performed on the first fuel removed, i.e., both non-destructive examination using the surveillance robot and destructive examination of the fuel rods and particles removed from the graphite fuel block.

A.2.8.2 Umbrella Agreement Liaison

Currently, activity is restricted to the fuel and graphite subprogram areas, but it is hoped that information exchange under other subprogram areas will be reinitiated in the near future.

A.2.8.3 Information Exchange

Information will be obtained on gas-cooled reactors in other countries which are in operation and under design or construction.

A.2.9 Core Auxiliary Cooling System Components

- Scope - This activity encompasses the design and development of generic CACS components and subsystems within the scope of supply. The major components and subsystems include the auxiliary circulator and motor, CAHE, auxiliary circulator motor controls, and auxiliary circulator service system.
- Objective - Within the framework of current licensing philosophy, it is required to design a CACS that will provide an independent means of cooling the core with the primary system pressurized or depressurized while maintaining the temperature of all components inside the PCRV within safe limits.
- Status
 - Auxiliary Circulator - The design and analysis of the auxiliary circulator were completed for the earlier 3000-MW(t) reference design. Only small changes are expected for an 1170 MW(t) design, with the exception of the design of the auxiliary circulator primary pressure boundary components, which will be subjected to higher pressures.

A DV&S program on the HTGR auxiliary circulator thrust bearing and bearing lubrication and seal systems has been completed. This program consisted of oil flow orifice plate calibration, an impeller performance visualization test, a thrust bearing labyrinth seal resistance test, thrust bearing system operation with heating, an oil evaporation loss test, and a bearing life test.

Test specifications and test rig design for the auxiliary motor cooling test have been completed.

- CAHE - The CAHE conceptual design phase has been initiated. A general arrangement drawing for a new, compact bayonet CAHE with side exhaust was issued. A sizing code for the bayonet CAHE configuration was completed, and the thermal sizing and gas-side pressure drop estimates were completed and documented. Thermal analysis of the CAHE tubesheet was completed, and an updated version of CAHE design issues was documented.

The test plan and test specification for the CAHE development tests have been issued.

- Auxiliary Circulator Motor Controls - The basic design of the auxiliary circulator motor controller for the previous 3000-MW(t) reference plant has been selected for the 1170-MW(t) design. The design of this equipment is essentially complete.
- Auxiliary Circulator Service System - A preliminary design of the auxiliary circulator service system was completed for the previous 3000-MW(t) reference design, and specifications, system descriptions, and the process flow and piping and instrumentation diagrams were issued. It is expected that these deliverables can be reissued with only minor modifications for the 1170-MW(t) design.

• Planned Program (Fig. A.2.9-1)

- Auxiliary Circulator - The tasks required to complete the auxiliary circulator design and development are as follows:
 1. The aerodynamic design for the compressor, including the pressure drop and flow requirement developed for the 3000-MW(t) reference plant, needs to be confirmed.
 2. The design of the pressure boundary components for the auxiliary circulator needs to be updated for higher primary system pressure.
 3. A design is needed for the electric power cable feed-through, since the auxiliary circulator and motor are located inside the PCRV.

The following DV&S program for the auxiliary circulator is required prior to completion of final design:

1. Continue the circulator motor cooling test. This test will establish the heat removal requirement for the stator/rotor heat exchanger and cooling fan assembly.

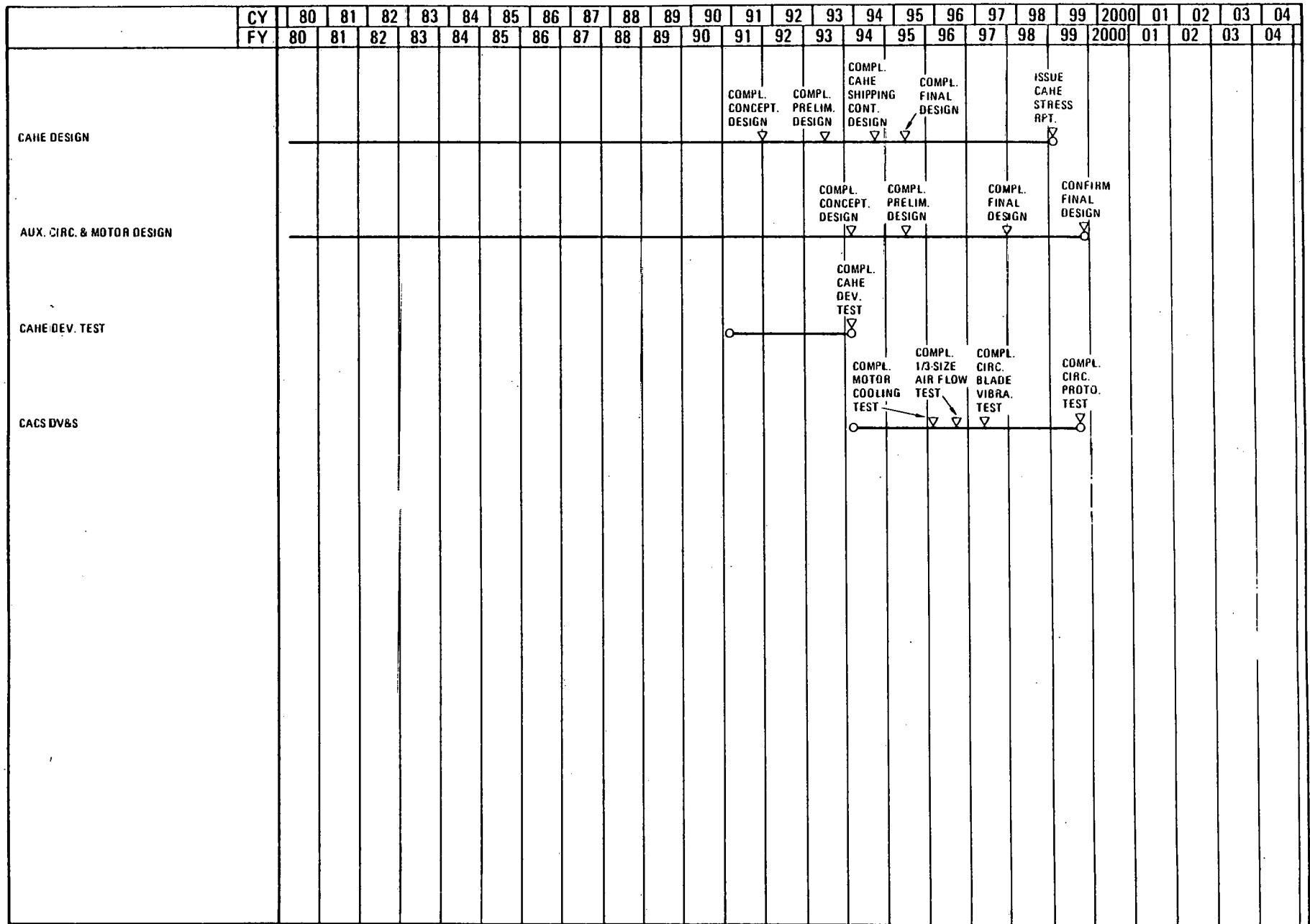


Figure A.2.9-1 HTGR Generic Technology Program detailed CACS components milestone schedule: HTGR-GT

2. Conduct a 1/3-scale air flow test of the compressor at required flow rate and pressure drop, and check further characteristics at opening and closing points of the shutoff valve to confirm the aerodynamic design.
 3. Conduct natural frequency vibration tests on compressor blades to confirm blade design in both compressor stages.
 4. Perform prototype qualification tests including seismic qualification on the auxiliary circulator at pressure and temperature for pressurized, subatmospheric, and transient operating conditions.
- CAHE - The tasks needed to complete the CAHE design and development are as follows:
1. Perform the CAHE thermal and stress analysis prior to final design completion, including detailed seal and seismic support analysis because of the safety significance of these components.
 2. Prepare the shipping and handling specification and design for the CAHE shipping container.
 3. Establish anticipated transients without scram (ATWS) requirements, since they may affect CAHE material selection.
 4. Establish subcooling margins for CACS startup transients.

Core auxiliary heat exchanger DV&S involving large-scale tests with water and air is required to finalize development of the CAHE configuration.

The first phase of the CAHE tests is a 1/4-scale flow visualization test using water at appropriate Reynolds numbers. The purpose is to develop the configuration of the inlet and outlet to the shell side of the CAHE for uniform flow distribution.

The second phase includes a variety of tests with a full-scale test model. The model will be used for both flow distribution and flow-induced vibration testing, as well as maintenance, ISI, and operation testing. Full-scale testing in ambient air will provide accurate flow modeling for heat transfer, pressure drop, flow-induced vibration, and flow distribution.

- Auxiliary Circulator Motor Controls - The tasks needed to complete the design for the core auxiliary circulator motor

control equipment include an effort to establish criteria for motor/controller instrumentation, updating of instrumentation diagrams, and CACS setpoint analysis.

- Auxiliary Circulator Service System - The remaining activities for the auxiliary circulator service system include final design and performance testing of the prototype service system in an auxiliary circulator test loop.

A.3 HTGR Spent Fuel Treatment Program

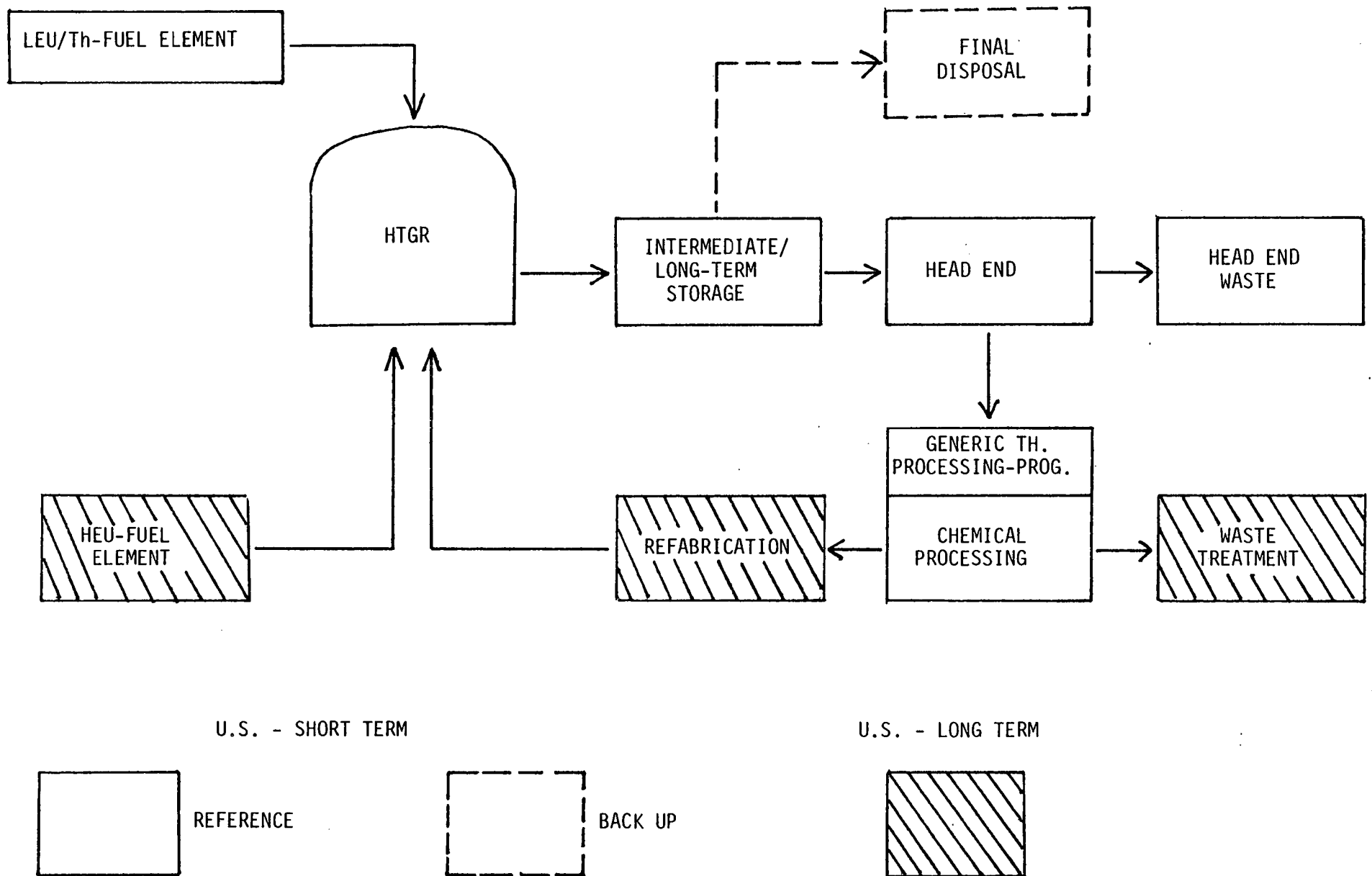
The HTGR Spent Fuel Treatment Program is an important part of the overall development of fuel cycle technology. In the long term, spent fuel treatment is necessary for the HTGR to realize its full economic and resource conservation potential. As such, this program measurably advances the national objectives.

The specific purpose of this program is to advance the technology of HTGR spent fuel treatment to the point where it will be effectively implemented on a commercial scale when national policy objectives support this requirement. To achieve this objective with greater cost effectiveness, the U.S. HTGR Spent Fuel Treatment Program emphasizes international cooperation with the German program.

The reference HTGR fuel cycle strategy is depicted in Fig. A.3-1, which shows that for the near term the LEU/Th fuel cycle will be employed. It is expected that greater economic pressure for recycle will develop within the nuclear industry as the price of U_3O_8 increases; therefore, the long-range fuel cycle strategy for the HTGR is predicated upon the case of the HEU/Th cycle with the recycle of U-233 and/or U-235. It is assumed that some form of HEU fuel will be usable on the following time table, and these dates are associated with the major decision points vis-a-vis the HTGR Spent Fuel Treatment Program:

- Commitment to a Recycle Demonstration Plant with placement of orders for multiple commercial units (circa 2005-2010).
- Introduction of HEU fuel for new and existing plants (circa 2015).
- Full scale operation of Recycle Demonstration Plant, circa 2020 (after approximately ten plants on-line).

It should be noted from Fig. A.3-1 that for both the near-term and long-term strategies, some treatment and ultimate disposal of the spent fuel are needed. Current emphasis of the HTGR Spent Fuel Treatment Program now centers on the near-term goals--those of reducing the spent fuel volume and processing the head-end waste. As the program develops and the need for recycle becomes imminent, the program will shift emphasis toward reprocessing and refabrication of the bred U-233.



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Figure A.3-1 U.S. Reference Fuel Cycle Strategy

- Scope

The HTGR Spent Fuel Treatment Program presently includes the following activities:

- Studies and analyses supporting the development program and facility projects.
- Spent fuel treatment technology development.
- Waste treatment investigations, including off-gas treatment.
- Cold prototype and hot pilot plant projects.

In addition, a comprehensive U.S./FRG cooperative development program has been initiated to obtain cold prototype and hot engineering design information and operating experience, possibly leading to a joint facility for the demonstration of spent fuel treatment technology. This cooperative program is a major part of the U.S. HTGR Spent Fuel Treatment Program plans.

- Description of Work and Status

HTGR spent fuel treatment development in the U.S. has progressed through laboratory development to the installation and operation of engineering-scale equipment. Hot laboratory experiments have been performed in support of this effort to determine the effects of high levels of fission products on these nuclear-chemical processes. A cold engineering-scale pilot plant has also been installed and operated to obtain quantitative information on process details and operating procedures. In the FRG, emphasis is being placed on the operation of a hot pilot plant, utilizing spent fuel from their AVR reactor to demonstrate the viability of the head-end processing of this fuel and to determine the effects of radiation on both process and equipment. The next phase of development includes a fully integrated, international prototypic development program of selected process equipment.

The following work is included in the FY 1981 Program:

- A. Spent Fuel Treatment

1. Studies and Analysis: This work includes the maintenance of HTGR fuel element design data, evaluation of the impact of fuel design changes on spent fuel treatment requirements and costs, maintenance and updating of spent fuel treatment flowsheets and material balances as required, and cold prototype equipment design and evaluation.
2. HTGR Dry Head-End Pilot Plant: This work includes verification testing of the reference unit operations, control and automation studies, and generic technology development in the area of solids handling.

3. Off-Gas Treatment: This work includes functional testing and evaluation of components and integration of the off-gas treatment system with the engineering-scale fluidized-bed (primary) burner and dissolver. Tests will establish process feasibility on an engineering scale, generate scale-up data for individual components, optimize operating parameters, and demonstrate integrated processes.
4. Solvent Extraction Pilot Plant: This work includes pulse column operation and data acquisition to verify computer codes for predicting pulse column performance.
5. Laboratory Studies: This work includes investigations to study scale-up effects and dissolution characteristics of various fuel particles and bench-scale studies to obtain generic thorium processing data.
6. HEF Technical Support: This work includes continuing technical support tasks which address generic design aspects for a spent fuel treatment plant.

B. Refabrication

It is proposed that refabrication development be resumed as a major activity as the national need for recycle becomes more clearly defined.

C. U.S./FRG Cooperative Program

The German government has structured a strong HTR development program which includes spent fuel treatment development. The U.S./FRG Umbrella Agreement, implemented in February 1977, provides for cooperative development of spent fuel treatment technology, and is the most advanced and active part of the international HTR cooperative program.

This program includes joint studies and exchange of technical information and personnel in four major areas: head-end operations, in-plant waste treatment, and fuel shipping and storage. This work is defined by 15 Project Work Statements, which have been organized into a Joint Program Plan.

The objectives of the joint program are the development of reference flowsheets, the design and testing of critical systems and components, and the demonstration of processes and integrated systems performance. Attention will be focused during FY 1981 on continuing cold checkout of the FRG JUPITER pilot plant and on carrying forward the conceptual design of a joint cold prototype facility. In addition, work will start on the feasibility and design requirements for hot demonstration facilities.

The major program activities and milestones for the U.S. and FRG Spent Fuel Treatment Development Programs are shown in Figure A.3-2. Design and engineering activities leading to the construction of the Recycle Demonstration Plant are not shown on this figure but would commence in the mid to late 1990s to be consistent with placing the reprocessing/refabrication demonstration plant in operation by circa 2020.

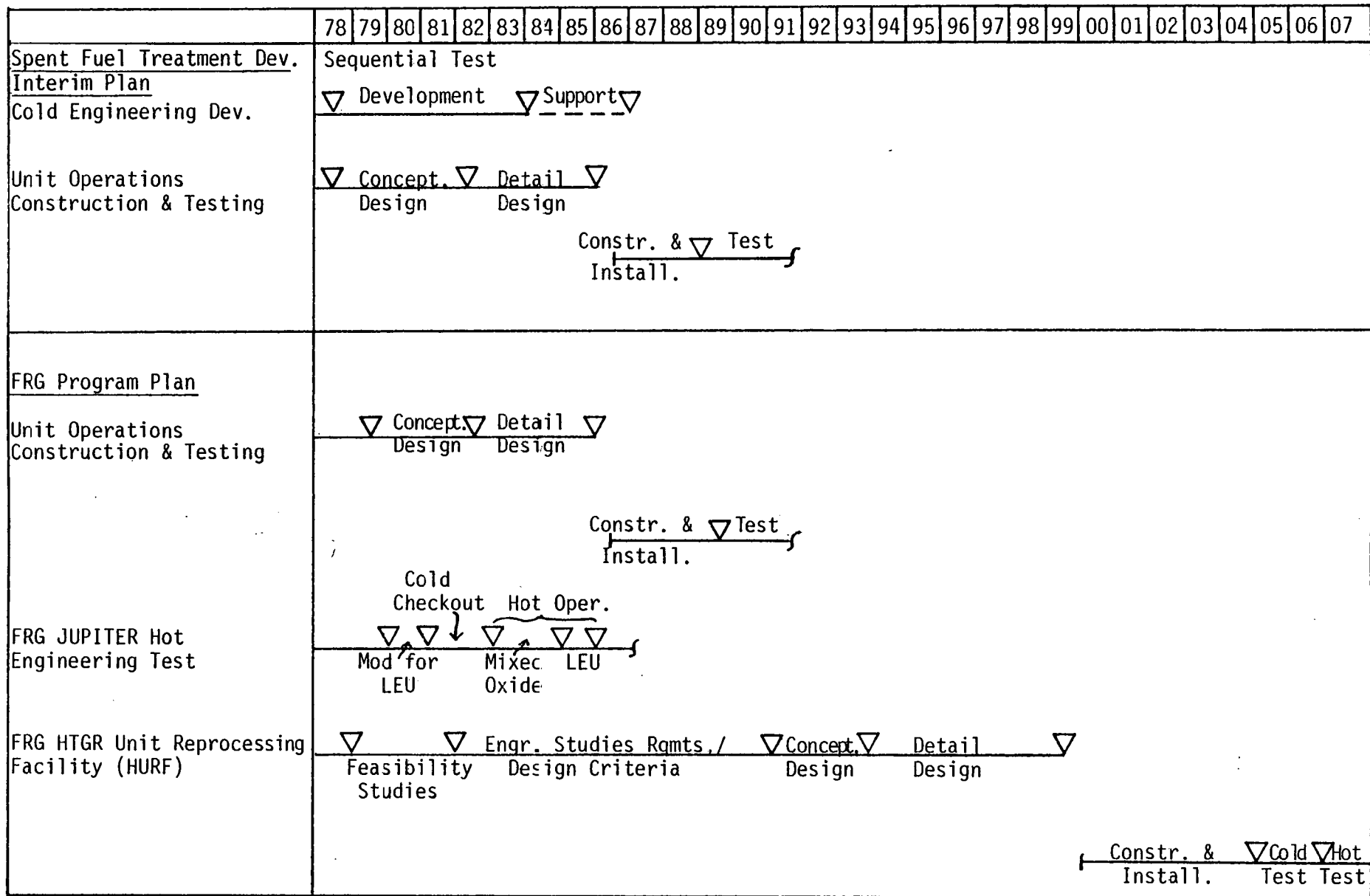


Figure A.3-2 HTGR Spent Fuel Treatment Development Program Schedules

APPENDIX B

DESCRIPTION OF TRANSIENTS

B.0 DESCRIPTION OF TRANSIENTS

B.1 Rapid Load Decrease (5%/min) (100% to 25% Range)

This transient sets the maximum rate of change of temperature throughout the system for normal operation. The sequence of events is as follows. The electric load demand is ramped at 5% full power/min, the load/speed controller opens the primary bypass valve, and the turbine inlet temperature controller reduces the turbine inlet temperature set point, which changes the neutron flux command. The large load demand reduction may require operator manual reshimming of the reactor control rods. The plant moves in this manner until equilibrium is reached at the 25% load level.

B.2 10% Step Load Decrease in 10 s (100% to 35% Range)

This transient shows the adequacy of the load control configuration. The sequence of events is as follows. The electric load demand step change occurs in 10 s, the load/speed controller opens the primary bypass valve, and the turbine inlet temperature controller reduces the turbine inlet temperature set point, which changes the neutron flux command. The quick onset of turbine inlet temperature control and opening of the attemperation valve by the attemperation controller cause a slightly larger decrease in electric power than desired. This effect is rapidly compensated for by reclosure of the primary bypass valve.

B.3 Single-Loop Load Rejection

This transient sets the maximum rate of pressure increase at the low-pressure recuperator inlet. The sequence of events is as follows: A single generator braker trip signals the PCS to switch the affected turbomachine from load/speed regulator control to speed control. The speed controller opens the primary bypass valve, the feedforward signal to the neutron flux controller from the affected loop is decreased to the "no load" level, and at the same time the turbine inlet temperature controller demands a reduction in reactor power. As the shutdown turbine reaches nominal speed and attains "hot standby" conditions, the hot recuperator inlet and cold recuperator outlet temperatures are held constant as a result of the bypass and attemperation flows. Concurrently, the operating power conversion loop decreases its electrical output as the loop pressure decreases due to the relocation of inventory in the system.

B.4 Single-Loop Shutdown With Overspeed Trip

This transient sets the reactor power to helium flow ratio and the PPS set point for main loop shutdown overspeed trip. The sequence of events is as follows. Loss of electrical load to one turbomachine (as described in the previous transient) is followed by failure of the

load/speed controller to detect turbomachine overspeed and command open the primary bypass valve. The PPS main loop shutdown overspeed trip set point is reached. The main loop shutdown signal commands open the primary bypass valve, via safety actuation (full opening in 1 s), which is followed by a demand for a reactor power runback to 50% nominal. As the pressure ratio across the turbine in the shutdown loop collapses, the turbine outlet temperature increases and the attemperation controller opens the attemperation valve; the low ΔP across the loop closes the primary bypass valve and later the attemperation valve. Backflow begins in the high-pressure sides of the recuperator and is quickly established in the entire shutdown loop (at very low flow conditions). In the operating loop, the turbine power decreases due to the relocation of inventory between the loops.

B.5 Single-Loop Total Loss of Precooler Flow

This transient sets the maximum pressure at the low-pressure recuperator inlet and the PPS set point for main loop shutdown on high outlet temperature of the high-pressure recuperator. The sequence of events is as follows. A spurious PPS signal commands closure of the precooler isolation valves but does not open the dump valves or initiate a power conversion loop shutdown. This is followed by failure of the non-safety component protection function of the PCS to detect that the precooler water flow rate is below its minimum design limit. Lack of coolant flow to the precooler results in lower compressor flow, higher inlet gas temperature of the high-pressure recuperator, and reduced turbine flow. Turbine power decreases as the turbine flow is reduced, causing the turbine outlet gas temperature to increase. The attemperation controller opens the attemperation valve further, reducing turbine flow. As the compressor inlet temperature increases and the loop pressure ratio decreases, the effectiveness of the attemperation flow decreases. Turbine load continues to decrease, and reactor power is run back. When 10% minimum generator load is reached, the PCS trips the generator breaker and commands open the primary bypass valve to prevent turbomachine overspeed. The load/speed controller closes the valve shortly thereafter as the turbomachine speed drops below the set point. The PPS main loop shutdown set point is reached by the gas temperature at the high-pressure recuperator outlet. (This set point provides protection of the upper cross ducts, thermal barrier, and the reactor inlet plenum components/materials.) Loop shutdown proceeds as described in the previous transient.

B.6 Plant Total Loss of Precooler Flow

This transient sets the maximum core inlet gas temperature, the maximum rate of pressure decrease at the high-pressure recuperator outlet, and the maximum heat exchanger hot end metal temperatures. The sequence of events is as follows. Shear rupture of the circulating water system pump discharge header is followed by failure of the non-safety component protection function of the PCS to detect that precooler water flow

rates are below their minimum design limit. The transient proceeds as described in Section B.5 but in both loops for this case. The PPS set points for main loop shutdown on high-pressure recuperator gas outlet temperature are reached in both loops almost simultaneously, followed by a PPS reactor trip due to two main loop shutdown signals.

B.7 Slow Rod Withdrawal From Full (Design) Power

This transient sets the maximum average turbine inlet gas temperature and the reactor trip PPS set point on high turbine inlet temperature. The sequence of events is as follows. A degraded flux measurement signal, which withdraws the control rod bank at a slow rate, is followed by failure of the automatic setback rods to insert following a large increase in core reactivity. Core reactivity increases due to the rod bank withdrawal and subsequently decreases due to the negative fuel temperature coefficients. The average core temperatures rise, the core outlet temperature rises, and the PPS reactor trip set point due to high turbine inlet temperature is reached. (This set point is a backup for the PPS power-to-flow trip set point.) As the core outlet temperature rises, the attemperation controllers open the attemperation valves, reducing turbomachine flow and power. The rapid core outlet temperature reduction after the reactor scram causes the attemperation valves to close, which results in an increase in turbomachine power. After the attemperation valves start reclosure, the loops begin normal plant programmed shutdown.

B.8 Single-Loop Turbomachine Shaft Break

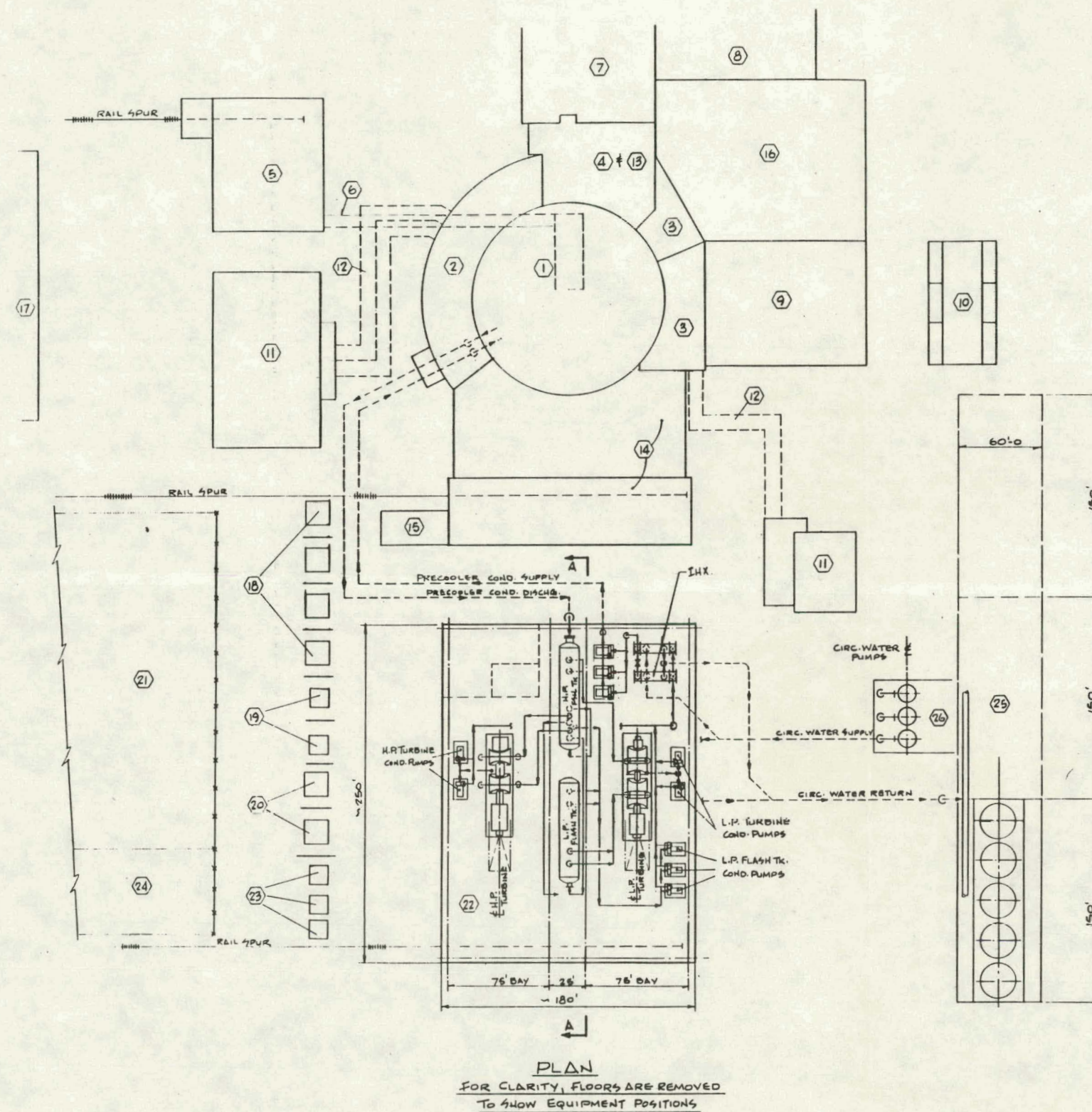
This transient sets the maximum turbomachine overspeed and maximum gas flow rate at the low-pressure recuperator inlet. The sequence of events is as follows. A fault in the electric generator produces an excess torque on the shaft and shears the coupling. "Instantaneous" loss of generator load and inertia accelerate the turbomachine, rapidly reaching the PPS turbomachine overspeed set point. The PCS turbomachine speed controller and the PCS turbomachine/generator speed difference detection controller, which should activate opening of the primary bypass valve immediately after the shaft break, are postulated to be disengaged and/or failed. The primary bypass valves open (safety actuation), reactor power is run back due to the loss of one power conversion loop, and the loop shutdown proceeds as described in previous transients.

APPENDIX C

ENGINEERING DRAWINGS

LIST OF DRAWINGS

Fig. 3.3.1-3	Flash Steam Bottoming Cycle Plot Plan
Fig. 3.3.2-5	Ammonia Bottoming Cycle Plot Plan
Fig. 3.3.2-6	Ammonia Bottoming Cycle Turbine Building Layout
GA 026278	Reactor Core Arrangement 2170 MW(t)
GA 025428	Optimized PCRV General Arrangement
GA 025020	Helium Control Valve Actuating Systems (Single Loop Layout)
GA 025256	Recuperator, Stayed Tubesheet, High Pressure Shell Side
GA 025424	GT-HTGR Precooler, Helical Coil, Side Penetrations
GA 025793	General Assembly Arrangement - Bayonet Tube CAHE - HTGR-GT
--	400-MW(e) Turbomachine
UE&C SK-107	Plot Plan
UE&C SK-87	General Arrangement 150'-0" I.D. Containment Plans at El. 112'-0" and 87'-0"
UE&C SK-88	General Arrangement 150'-0" I.D. Containment Plans at El. 0'-6" and (-) 19'-4"
UE&C SK-89	General Arrangement 150'-0" I.D. Containment Section "A-A"
UE&C SK-91	General Arrangement Auxiliary Reactor Service Building Plan at El. 112'-0" and Section "A-A"
UE&C SK-94	General Arrangement Auxiliary Reactor Service Building Plans at El. 0'-6" and (-) 25'-0"
UE&C SK-97	General Arrangement Hands On Maintenance Facility for Angled Turbines
UE&C SK-98	General Arrangement 150'-0" I.D. Containment Plan at El. 30'-0" and 56'-0", Section "A-A"
UE&C SK-69A&B	Key One Line Diagram Unit Electrical Distribution



- LEGEND**
- 1 REACTOR CONTAINMENT BLDG.
 - 2 CONTAINMENT ANNULUS BLDG.
 - 3 CONTAINMENT PENETRATION BLDG.
 - 4 AUXILIARY REACTOR SERVICE BLDG.
 - 5 LONG TERM FUEL STORAGE BLDG.
 - 6 LONG TERM FUEL TRANSFER TUNNEL
 - 7 TURBO-MACHINE MAINT. BLDG.
 - 8 LABS & SHOPS
 - 9 CONTROL, AUXILIARY, & DIESEL/GEN. BLDG.
 - 10 DIESEL COOLING & FUEL OIL STOR. BLDG.
 - 11 ULTIMATE HEAT SINK STRUCTURES
 - 12 ULTIMATE HEAT SINK PIPING TUNNEL
 - 13 ENCL. ABOVE AUX. REACTOR SERV. BLDG.
 - 14 ELEC. GEN. & NON-ESS. SWGR. BLDG.
 - 15 BOP DIESEL GEN. BLDG.
 - 16 ADMINISTRATION & SERVICE BLDG.
 - 17 LOW PRESS. HELIUM STORAGE AREA
 - 18 GENERATOR STEP-UP TRANSFORMERS
 - 19 UNIT AUXILIARY TRANSFORMERS
 - 20 RESERVE AUX. TRANSFORMERS
 - 21 MAIN SWITCHYARD
 - 22 TURBINE BUILDING (BOTTOMING CYCLE)
 - 23 BOTTOMING CYCLE TRANSFORMERS
 - 24 AUXILIARY SWITCHYARD
 - 25 WET COOLING TOWER
 - 26 CIRC. WATER PUMP HOUSE

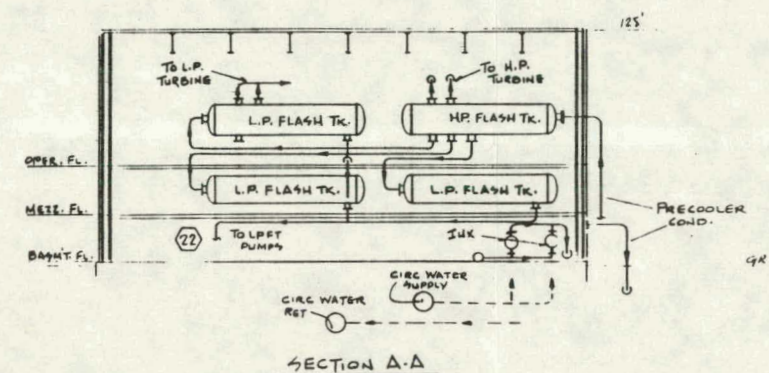


Figure 3.3.1-3 Flash steam bottoming cycle plot plan

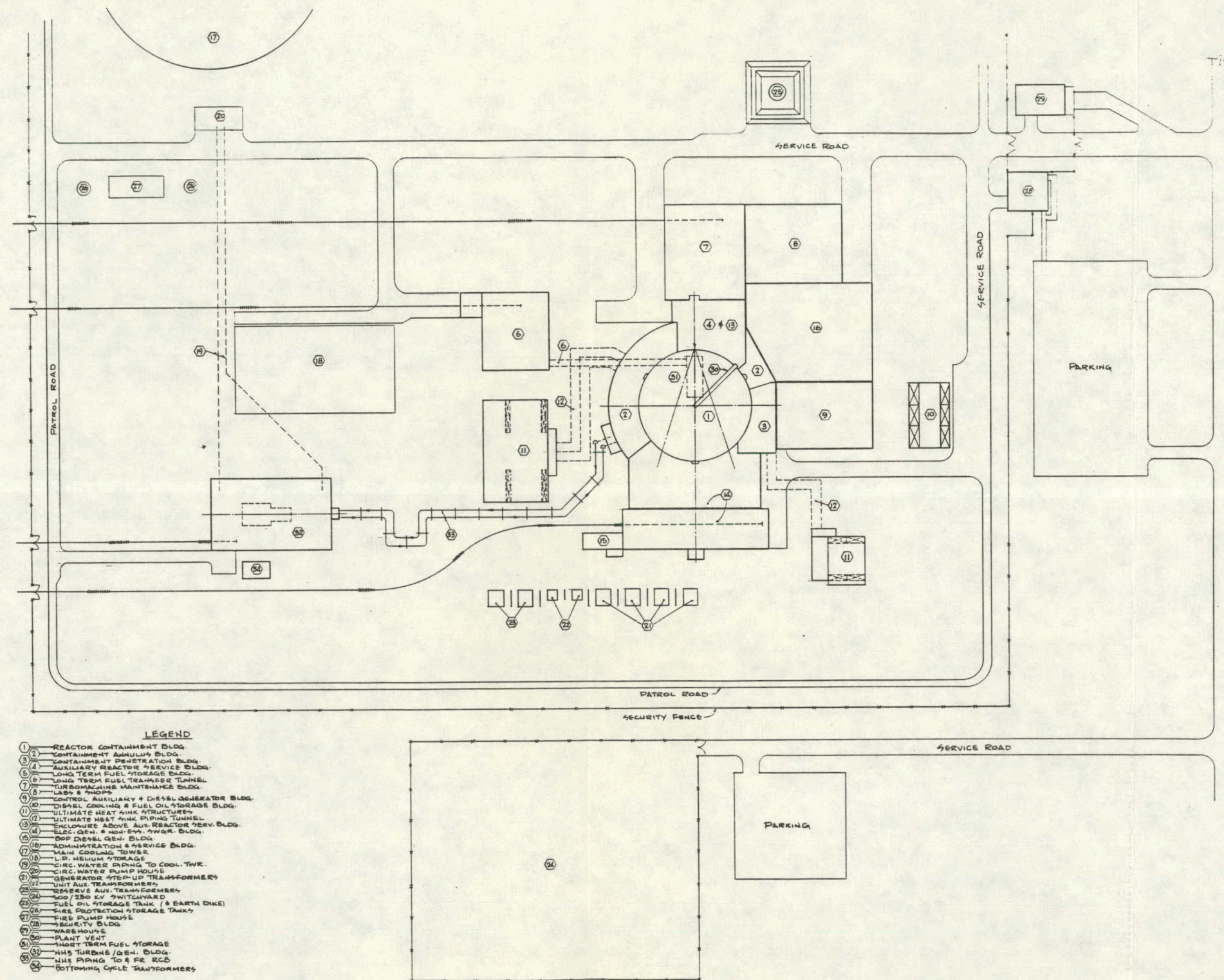
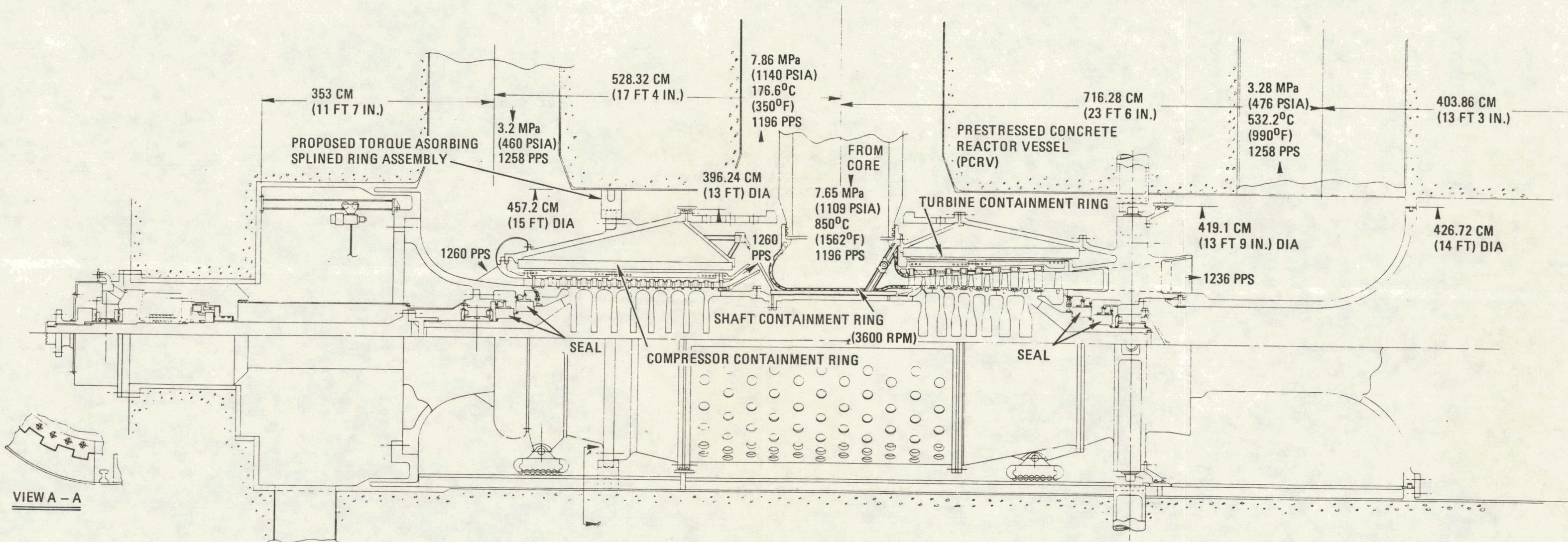
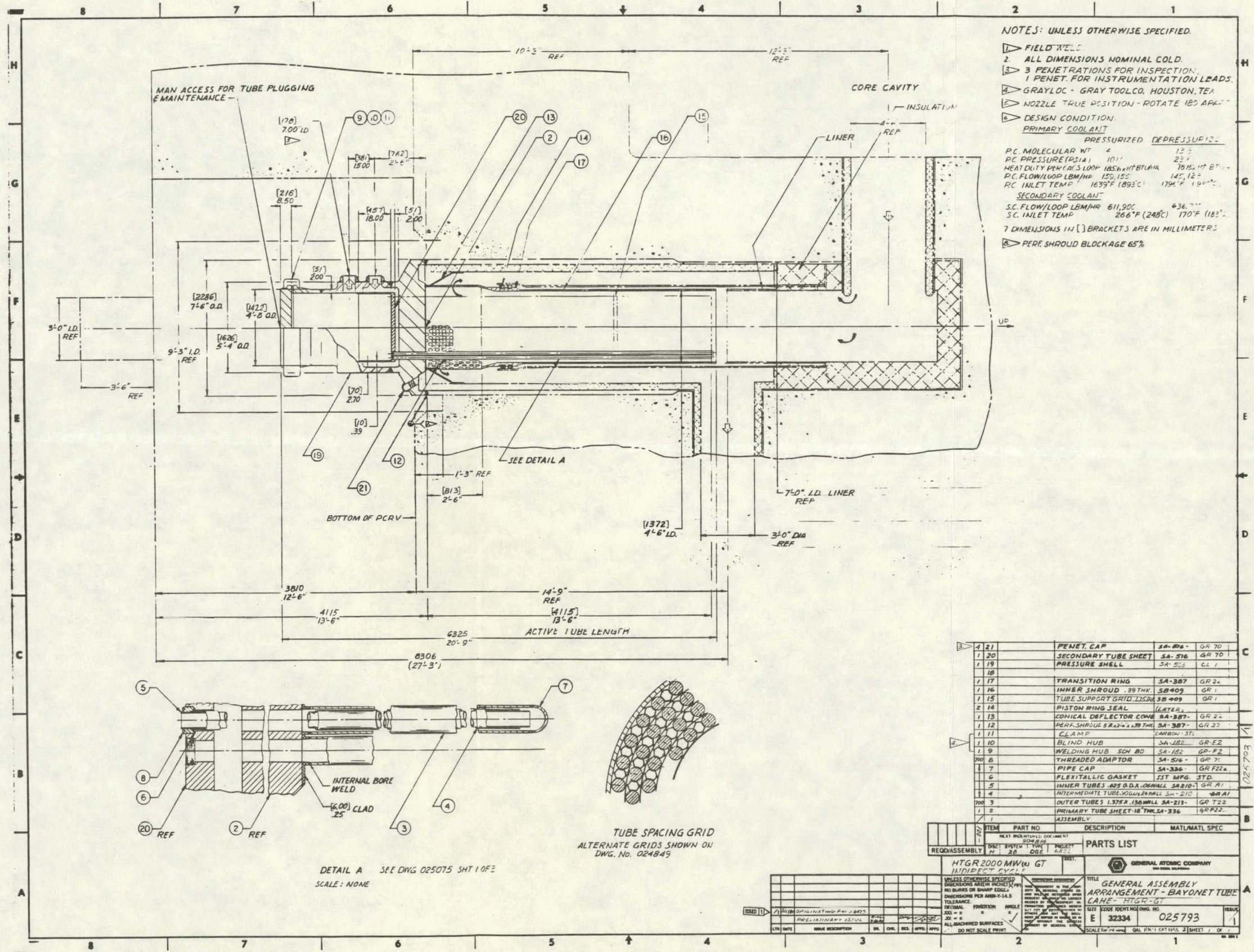
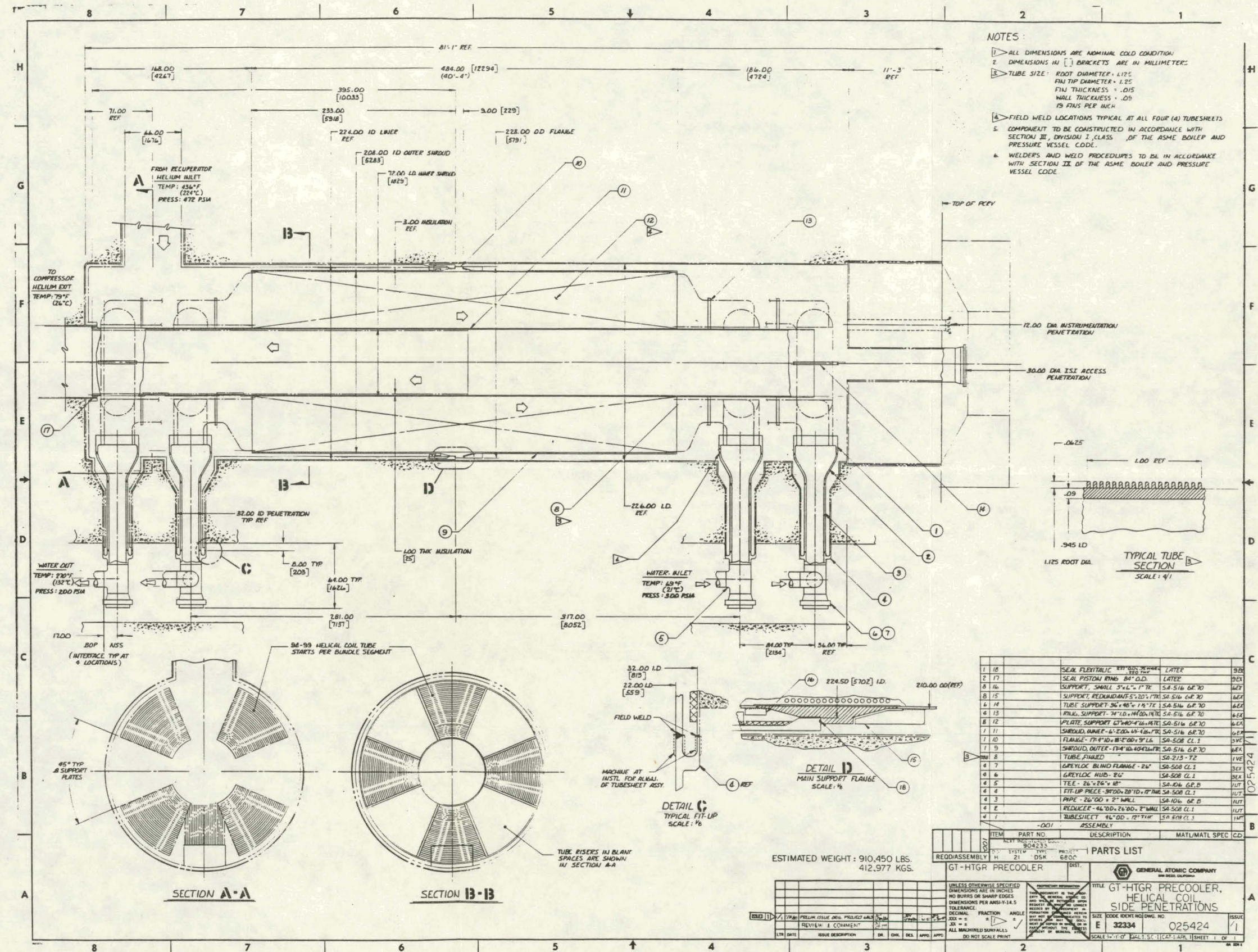


Figure 3.3.2-5 Ammonia bottoming cycle plot plan

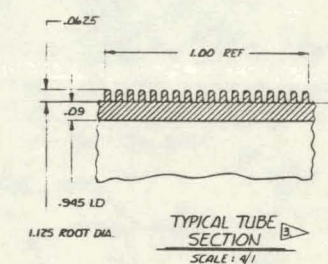


400-MW(e) TURBOMACHINE





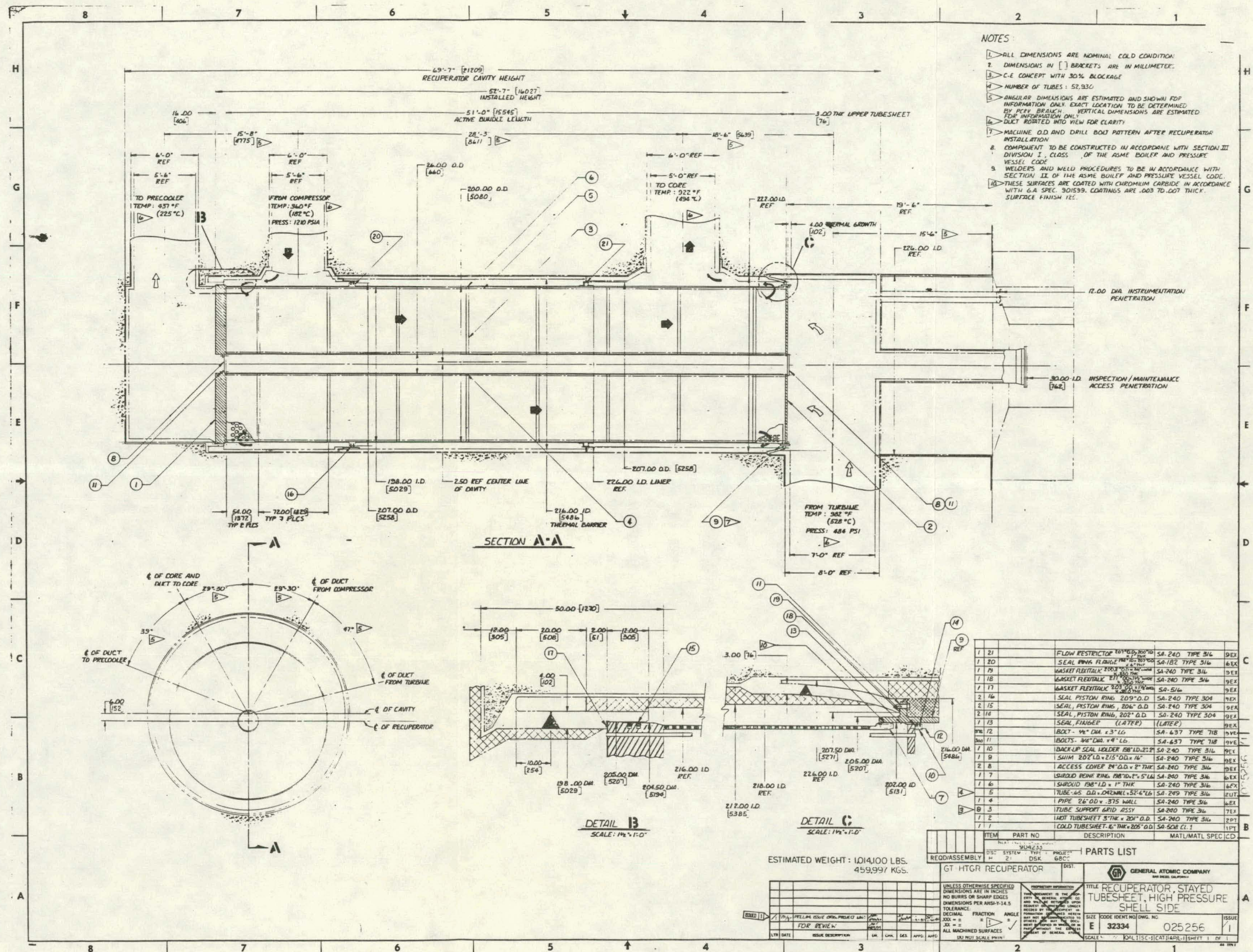
- NOTES:
- 1. ALL DIMENSIONS ARE NOMINAL, COLD CONDITION.
 - 2. DIMENSIONS IN [] BRACKETS ARE IN MILLIMETER.
 - 3. TUBE SIZE: ROOT DIAMETER - 1.125
FIN TIP DIAMETER - 1.25
FIN THICKNESS - .015
WALL THICKNESS - .05
19 FINS PER INCH
 - 4. FIELD WELD LOCATIONS TYPICAL AT ALL FOUR (4) TUBESHEETS.
 - 5. COMPONENT TO BE CONSTRUCTED IN ACCORDANCE WITH SECTION II, DIVISION 1, CLASS 1, OF THE ASME BOILER AND PRESSURE VESSEL CODE.
 - 6. WELDERS AND WELD PROCEDURES TO BE IN ACCORDANCE WITH SECTION II OF THE ASME BOILER AND PRESSURE VESSEL CODE.



ITEM	PART NO.	DESCRIPTION	MATL/MATL SPEC
1	18	SEAL FLEXIBLE	SA-516 GR 70
2	17	SEAL PISTON RING	SA-516 GR 70
3	16	SUPPORT, SMALL	SA-516 GR 70
4	15	SUPPORT, REDUNDANT	SA-516 GR 70
5	14	TUBE SUPPORT	SA-516 GR 70
6	13	ROLL, SUPPORT	SA-516 GR 70
7	12	PLATE, SUPPORT	SA-516 GR 70
8	11	SHROUD, INNER	SA-516 GR 70
9	10	FLANGE, INNER	SA-516 GR 70
10	9	SHROUD, OUTER	SA-516 GR 70
11	8	TUBE, FINED	SA-516 GR 70
12	7	GREYLOC BLIND FLANGE	SA-516 GR 70
13	6	GREYLOC NUT	SA-516 GR 70
14	5	TEE	SA-516 GR 70
15	4	TEE-UP PIECE	SA-516 GR 70
16	3	PIPE	SA-516 GR 70
17	2	REDUCER	SA-516 GR 70
18	1	TUBESHEET	SA-516 GR 70

ESTIMATED WEIGHT: 910,450 LBS.
412,977 KGS.

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DATE	DESCRIPTION	BY	CHK	APP.	1	1/1/77	PRELIM. ISSUE ONLY. PROJECT NO. 100-100000-0000				2	2/1/77	REVIEW: 0 COMMENT				3	3/1/77					4	4/1/77					5	5/1/77					6	6/1/77					7	7/1/77					8	8/1/77					9	9/1/77					10	10/1/77					11	11/1/77					12	12/1/77					13	1/1/78					14	2/1/78					15	3/1/78					16	4/1/78					17	5/1/78					18	6/1/78					19	7/1/78					20	8/1/78					21	9/1/78					22	10/1/78					23	11/1/78					24	12/1/78					25	1/1/79					26	2/1/79					27	3/1/79					28	4/1/79					29	5/1/79					30	6/1/79					31	7/1/79					32	8/1/79					33	9/1/79					34	10/1/79					35	11/1/79					36	12/1/79					37	1/1/80					38	2/1/80					39	3/1/80					40	4/1/80					41	5/1/80					42	6/1/80					43	7/1/80					44	8/1/80					45	9/1/80					46	10/1/80					47	11/1/80					48	12/1/80					49	1/1/81					50	2/1/81					51	3/1/81					52	4/1/81					53	5/1/81					54	6/1/81					55	7/1/81					56	8/1/81					57	9/1/81					58	10/1/81					59	11/1/81					60	12/1/81					61	1/1/82					62	2/1/82					63	3/1/82					64	4/1/82					65	5/1/82					66	6/1/82					67	7/1/82					68	8/1/82					69	9/1/82					70	10/1/82					71	11/1/82					72	12/1/82					73	1/1/83					74	2/1/83					75	3/1/83					76	4/1/83					77	5/1/83					78	6/1/83					79	7/1/83					80	8/1/83					81	9/1/83					82	10/1/83					83	11/1/83					84	12/1/83					85	1/1/84					86	2/1/84					87	3/1/84					88	4/1/84					89	5/1/84					90	6/1/84					91	7/1/84					92	8/1/84					93	9/1/84					94	10/1/84					95	11/1/84					96	12/1/84					97	1/1/85					98	2/1/85					99	3/1/85					100	4/1/85					101	5/1/85					102	6/1/85					103	7/1/85					104	8/1/85					105	9/1/85					106	10/1/85					107	11/1/85					108	12/1/85					109	1/1/86					110	2/1/86					111	3/1/86					112	4/1/86					113	5/1/86					114	6/1/86					115	7/1/86					116	8/1/86					117	9/1/86					118	10/1/86					119	11/1/86					120	12/1/86					121	1/1/87					122	2/1/87					123	3/1/87					124	4/1/87					125	5/1/87					126	6/1/87					127	7/1/87					128	8/1/87					129	9/1/87					130	10/1/87					131	11/1/87					132	12/1/87					133	1/1/88					134	2/1/88					135	3/1/88					136	4/1/88					137	5/1/88					138	6/1/88					139	7/1/88					140	8/1/88					141	9/1/88					142	10/1/88					143	11/1/88					144	12/1/88					145	1/1/89					146	2/1/89					147	3/1/89					148	4/1/89					149	5/1/89					150	6/1/89					151	7/1/89					152	8/1/89					153	9/1/89					154	10/1/89					155	11/1/89					156	12/1/89					157	1/1/90					158	2/1/90					159	3/1/90					160	4/1/90					161	5/1/90					162	6/1/90					163	7/1/90					164	8/1/90					165	9/1/90					166	10/1/90					167	11/1/90					168	12/1/90					169	1/1/91					170	2/1/91					171	3/1/91					172	4/1/91					173	5/1/91					174	6/1/91					175	7/1/91					176	8/1/91					177	9/1/91					178	10/1/91					179	11/1/91					180	12/1/91					181	1/1/92					182	2/1/92					183	3/1/92					184	4/1/92					185	5/1/92					186	6/1/92					187	7/1/92					188	8/1/92					189	9/1/92					190	10/1/92					191	11/1/92					192	12/1/92					193	1/1/93					194	2/1/93					195	3/1/93					196	4/1/93					197	5/1/93					198	6/1/93					199	7/1/93					200	8/1/93					201	9/1/93					202	10/1/93					203	11/1/93					204	12/1/93					205	1/1/94					206	2/1/94					207	3/1/94					208	4/1/94					209	5/1/94					210	6/1/94					211	7/1/94					212	8/1/94					213	9/1/94					214	10/1/94					215	11/1/94					216	12/1/94					217	1/1/95					218	2/1/95					219	3/1/95					220	4/1/95					221	5/1/95					222	6/1/95					223	7/1/95					224	8/1/95					225	9/1/95					226	10/1/95					227	11/1/95					228	12/1/95					229	1/1/96					230	2/1/96					231	3/1/96					232	4/1/96					233	5/1/96					234	6/1/96					235	7/1/96					236	8/1/96					237	9/1/96					238	10/1/96					239	11/1/96					240	12/1/96					241	1/1/97					242	2/1/97					243	3/1/97					244	4/1/97					245	5/1/97					246	6/1/97					247	7/1/97					248	8/1/97					249	9/1/97					250	10/1/97					251	11/1/97					252	12/1/97					253	1/1/98					254	2/1/98					255	3/1/98					256	4/1/98					257	5/1/98					258	6/1/98					259	7/1/98					260	8/1/98					261	9/1/98					262	10/1/98					263	11/1/98					264	12/1/98					265	1/1/99					266	2/1/99					267	3/1/99					268	4/1/99					269	5/1/99					270	6/1/99					271	7/1/99					272	8/1/99					273	9/1/99					274	10/1/99					275	11/1/99					276	12/1/99					277	1/1/00					278	2/1/00					279	3/1/00					280	4/1/00					281	5/1/00					282	6/1/00					283	7/1/00					284	8/1/00					285	9/1/00					286	10/1/00					287	11/1/00					288	12/1/00					289	1/1/01					290	2/1/01					291	3/1/01					292	4/1/01					293	5/1/01					294	6/1/01					295	7/1/01					296	8/1/01					297	9/1/01					298	10/1/01					299	11/1/01					300	12/1/01					301	1/1/02					302	2/1/02					303	3/1/02					304	4/1/02					305	5/1/02					306	6/1/02					307	7/1/02					308	8/1/02					309	9/1/02					310	10/1/02					311	11/1/02					312	12/1/02					313	1/1/03					314	2/1/03					315	3/1/03					316	4/1/03				
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- NOTES:
1. ALL DIMENSIONS ARE NOMINAL, COLD CONDITION.
 2. DIMENSIONS IN [] BRACKETS ARE IN MILLIMETER.
 3. C-E CONCEPT WITH 30% BLOCKAGE.
 4. NUMBER OF TUBES: 52,930.
 5. ANGULAR DIMENSIONS ARE ESTIMATED AND SHOWN FOR INFORMATION ONLY. EXACT LOCATION TO BE DETERMINED BY PERM. BRANCH. VERTICAL DIMENSIONS ARE ESTIMATED.
 6. DUCT ROTATED INTO VIEW FOR CLARITY.
 7. MACHINE O.D. AND DRILL BOLT PATTERN AFTER RECUPERATOR INSTALLATION.
 8. COMPONENT TO BE CONSTRUCTED IN ACCORDANCE WITH SECTION III, DIVISION 1, CLASS, OF THE ASME BOILER AND PRESSURE VESSEL CODE.
 9. WELDERS AND WELD PROCEDURES TO BE IN ACCORDANCE WITH SECTION II OF THE ASME BOILER AND PRESSURE VESSEL CODE.
 10. THESE SURFACES ARE COATED WITH CHROMIUM CARBIDE IN ACCORDANCE WITH 6.A SPEC. 301599. COATINGS ARE .003 TO .007 THICK. SURFACE FINISH 12E.

ITEM	PART NO.	DESCRIPTION	MATL/MATL SPEC
1 21		FLOW RESTRICTOR 10" O.D. x 1.5" I.D. x 1.5" L.	SA-240 TYPE 316
1 20		SEAL RING 10" O.D. x 1.5" I.D. x 1.5" L.	SA-182 TYPE 316
1 19		WASHER 10" O.D. x 1.5" I.D. x 1.5" L.	SA-240 TYPE 316
1 18		WASHER 10" O.D. x 1.5" I.D. x 1.5" L.	SA-240 TYPE 316
1 17		WASHER 10" O.D. x 1.5" I.D. x 1.5" L.	SA-240 TYPE 316
2 16		SEAL PISTON RING 20" O.D. x 1.5" I.D. x 1.5" L.	SA-240 TYPE 304
2 15		SEAL PISTON RING 20" O.D. x 1.5" I.D. x 1.5" L.	SA-240 TYPE 304
2 14		SEAL PISTON RING 20" O.D. x 1.5" I.D. x 1.5" L.	SA-240 TYPE 304
1 13		SEAL, FINGER (LATER)	(LATER)
1 12		BOLT - 1/2" DIA. x 3" LG.	SA-637 TYPE 718
1 11		BOLTS - 3/4" DIA. x 4" LG.	SA-637 TYPE 718
1 10		BACK-UP SEAL HOLDER 18" O.D. x 1.5" I.D. x 1.5" L.	SA-240 TYPE 316
1 9		SHIM 20" O.D. x 1.5" I.D. x 1.5" L.	SA-240 TYPE 316
2 8		ACCESS COVER 18" O.D. x 1.5" I.D. x 1.5" L.	SA-240 TYPE 316
1 7		SHIM 20" O.D. x 1.5" I.D. x 1.5" L.	SA-240 TYPE 316
1 6		SHIM 20" O.D. x 1.5" I.D. x 1.5" L.	SA-240 TYPE 316
1 5		TUBE 1/2" O.D. x 1/2" I.D. x 1.5" L.	SA-240 TYPE 316
1 4		PIPE 26" O.D. x 3/16" WALL	SA-240 TYPE 316
1 3		TUBE SUPPORT GRID ASSY.	SA-240 TYPE 316
1 2		WOT TUBESHEET 5" THK x 20" O.D.	SA-240 TYPE 316
1 1		COLD TUBESHEET 6" THK x 20" O.D.	SA-508 CL. 1

ESTIMATED WEIGHT: 1014,000 LBS.
459,997 KGS.

DATE	FOR REVIEW	ISSUE DESCRIPTION	CHK.	DES.	APP.	APP.

REASSEMBLY		GT HTGR RECUPERATOR		GENERAL ATOMIC COMPANY	
TITLE		RECUPERATOR, STAYED TUBESHEET, HIGH PRESSURE SHELL SIDE		E 32334 025256	
SCALE		1/4" = 1'-0"		1/4" = 1'-0"	
SHEET		1 OF 1		1 OF 1	

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SHT 2
HE. CONTROL VALVE ACTUATING SYS.

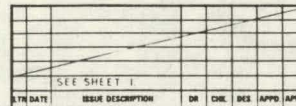
TECHNICAL INFORMATION ACTUATION SYSTEM

EQUIPMENT NUMBER	DESCRIPTION	SUCTION/DISCH. PRESSURE	MAKE (BEST ESTIMATE)	SIZE	VOLUME OR HORSEPOWER	GOVERNING CODE	DESIGN TEMP./PRESSURE
P22101, P22102	HYDR. FLUID PUMPS	13.83/1425 PSIA	CONTINENTAL PVR 15-15B-RMD	24"X14"X12"OA	15 HP / 120V	NONE	120°F / 1500 PSIA
P22201, P22202		13.83 / 1425	PVRD-32A-15-STD	28"X16"X12"OA	25 HP / 120V	NONE	120 / 1500
P22301, P22302		13.83 / 2500	DELAVAL "MO" NO. 64V	76"X24"X25"OA	350 HP / (LATER)	NONE	120 / 3000
P22401, P22402 P22501, P22502		13.83 / 2500	OILCEAN "INDURA" MODEL PVQ-06	15"X10"X22"OA	6 GPM - 13 HP (15 HP MOTOR)	ASME III, CL2	120 / 3000
T22101 T22201 T22301, T22501, T22502	GAS PRESSURIZER	ATMOS.	NONE	20"OD X 69"OA 24"OD X 74"OA 24"X 36"X 60"	10.4 FT ³ 12 FT ³ 30 FT ³	NONE	120 / ATMOS.
T22102		1425 PSIA	NONE	18"OD X 57"	3.66 FT ³	ASME VIII	120 / 1500
T22202		1425 PSIA	NONE	12"OD X 69"	2.67 FT ³	ASME VIII	120 / 1500
T22302		2500 PSIA	NONE	24"OD X 93" (2)	23.3 FT ³	ASME VIII	120 / 3000
T22402, T22502	HYDR. FLUID ACCUMULATOR	2500 PSIA	NONE	24"OD X 93" (2)	23.3 FT ³	ASME III, CL2	120 / 3000
T22103		1425 PSIA	NONE	12"OD X 40"	14.3 GALS.	ASME VIII	120 / 1500
T22203		1425 PSIA	NONE	8"OD X 47"	.83 FT ³	ASME VIII	120 / 1500
T22303		2500 PSIA	NONE	18"OD X 42"	4 FT ³	ASME VIII	120 / 3000
T22403, T22503	FLUID COOLER	2500 PSIA	NONE	18"OD X 42"	4 FT ³	ASME III, CL2	120 / 3000
E22101, E22201		125 PSIG	GRAHAM HELIFLOW 20-20L	20"1/2 X 20 X 22"	DUTY 38,160 BTUH	NONE	120 / 125
E22301		125 PSIG	3/4" U-TUBES	30"OD X 97"	DUTY 445,000 BTUH	NONE	120 / 125
F22101, F22102 F22201, F22202 F22301, F22302 F22401, F22501		ATMOS.	ALL AMF LUNO MICROKLEAN II TYPE G78-Q8	4"OD X 20" 4"OD X 20" 15"OD X 30" 15"OD X 30"	ALL 75 MICRON RATING	NONE	120 / ATMOS.
HV22101 HV22201	SERVO CONTROL VALVE	NONE	PEGASUS MODEL 1282A	7"4"X 5"X 7"4"LL	NONE	ANSI B31.1	120 / 1500
HV22301, HV22401		NONE	PEGASUS MODEL 1820	6"4"X 6"X 15"3/4"LL	NONE	ANSI B31.1	120 / 1500
HV22501		NONE	MOOG MODEL 79-410	10"X 9"1/2"X 20"LL	NONE	ASME III, CL-2	120 / 3000

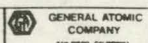
TECHNICAL INFORMATION, VALVES

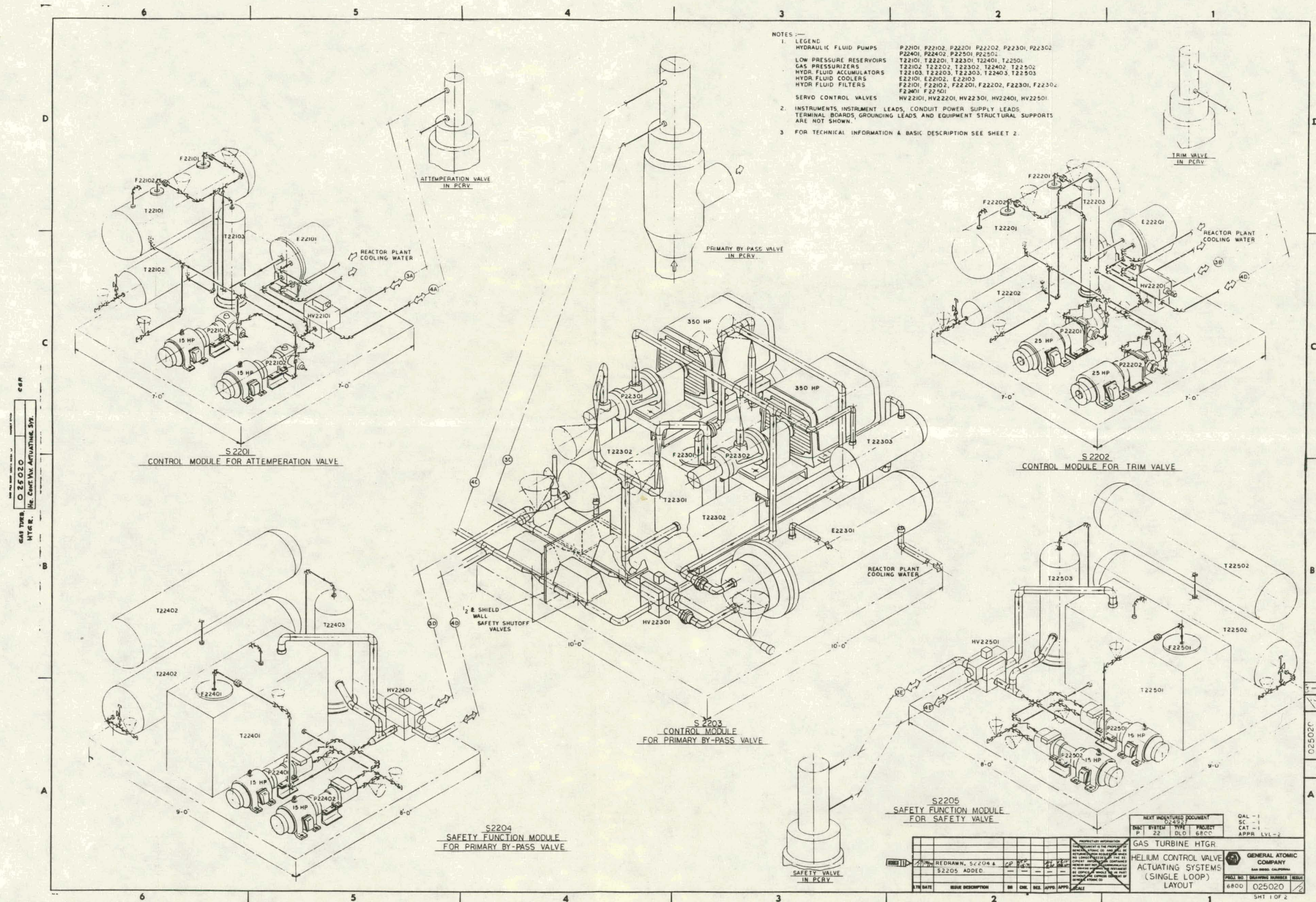
INFORMATION	VALVES			
	PR. BY-PASS	SAFETY	TRIM	ATTENUATION
AMOUNT / LOOP	1	1	1	1
SIZE	26.4 IN.	26.4 IN.	9.145 IN.	16. IN.
FLOW RATE (NOTE 1)	8.3 X 10 ⁶ LB/HR	8.3 X 10 ⁶ LB/HR	0.83 X 10 ⁶ LB/HR	0.9 X 10 ⁶ LB/HR
FLUID	RAD H ₂	RAD H ₂	RAD H ₂	RAD H ₂
TEMP °F	1012	1012	1012	1012
PRESSURE	1128 PSIA	1128 PSIA	1128 PSIA	1150 PSIA
MAX. ΔP	680 PSID	680 PSID	680 PSID	684 PSID
BODY CONFIGURATION	ANGLE	ANGLE	ANGLE	ANGLE
DISC CONFIGURATION	LINEAR	LINEAR	LINEAR	LINEAR
OPEN / SHUT MODE	—	—	—	—
TIME TO OPEN	1 SEC.	1 SEC.	0.2 SEC.	20% IN 0.2 SEC.
TIME TO SHUT	3 SEC.	60 SEC.	CONTROL MODE	CONTROL MODE
CONTROL MODE	—	—	—	—
TIME RESP. (10%)	0.25 SEC.	N/A	0.1 SEC.	0.4 SEC.
MAX. CONT. RANGE	0-100%	N/A	0-100%	0-100%
ACCURACY	0.2 %	N/A	0.04 %	0.2 %
NORMAL SEAT LEAK	1500 LB/HR	1500 LB/HR	15 LB/HR	1500 LB/HR
CODE	ASME III / I.	ASME III / I.	ASME III / I.	ASME III / I.
BODY MATERIAL (NOTE 2)	CARBON STL.	CARBON STL.	CARBON STL.	CARBON STL.

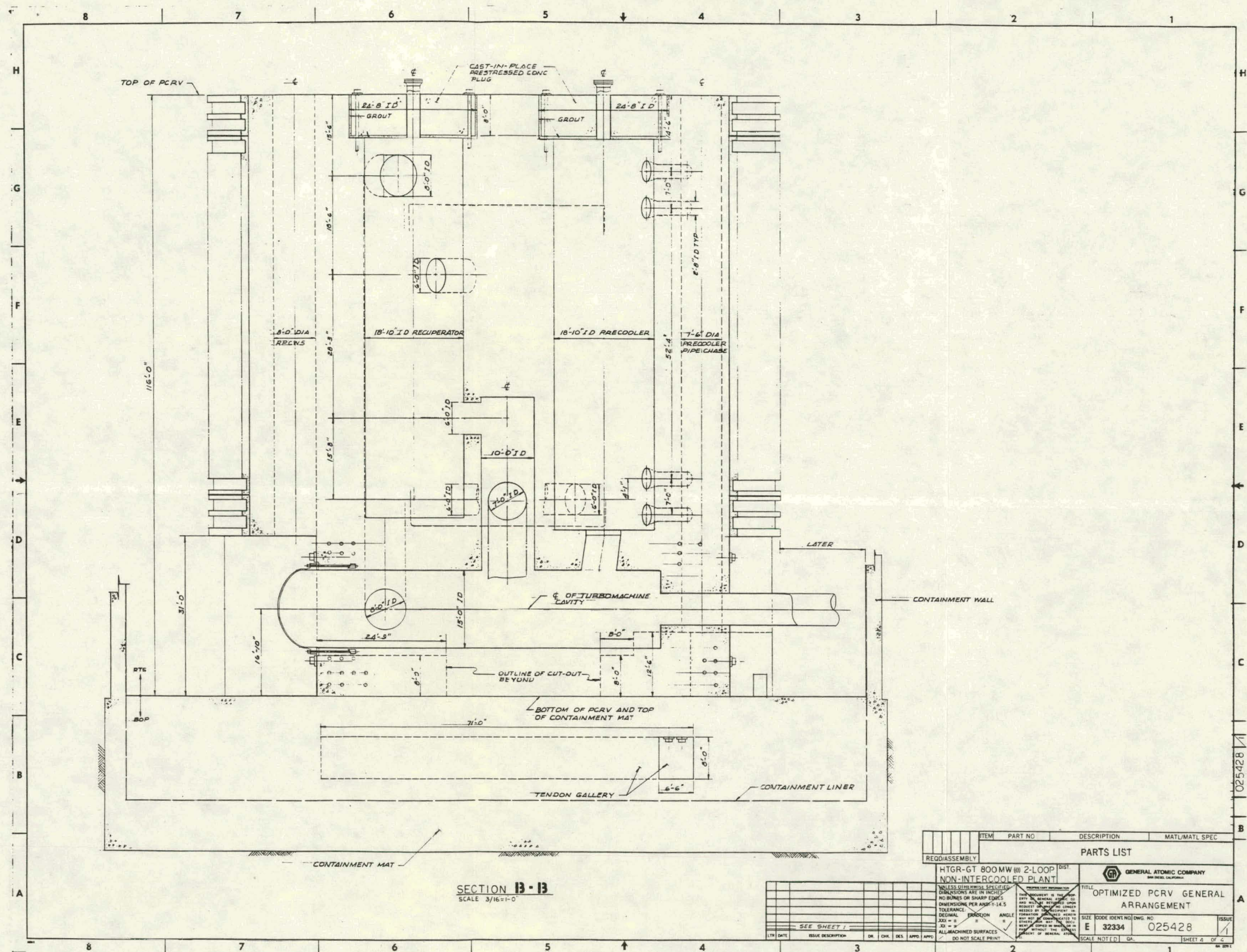
NOTES —
1. BEST ESTIMATE (DURING LOSS OF ELECTRICAL LOAD TRANSIENT).
2. BEST ESTIMATE.

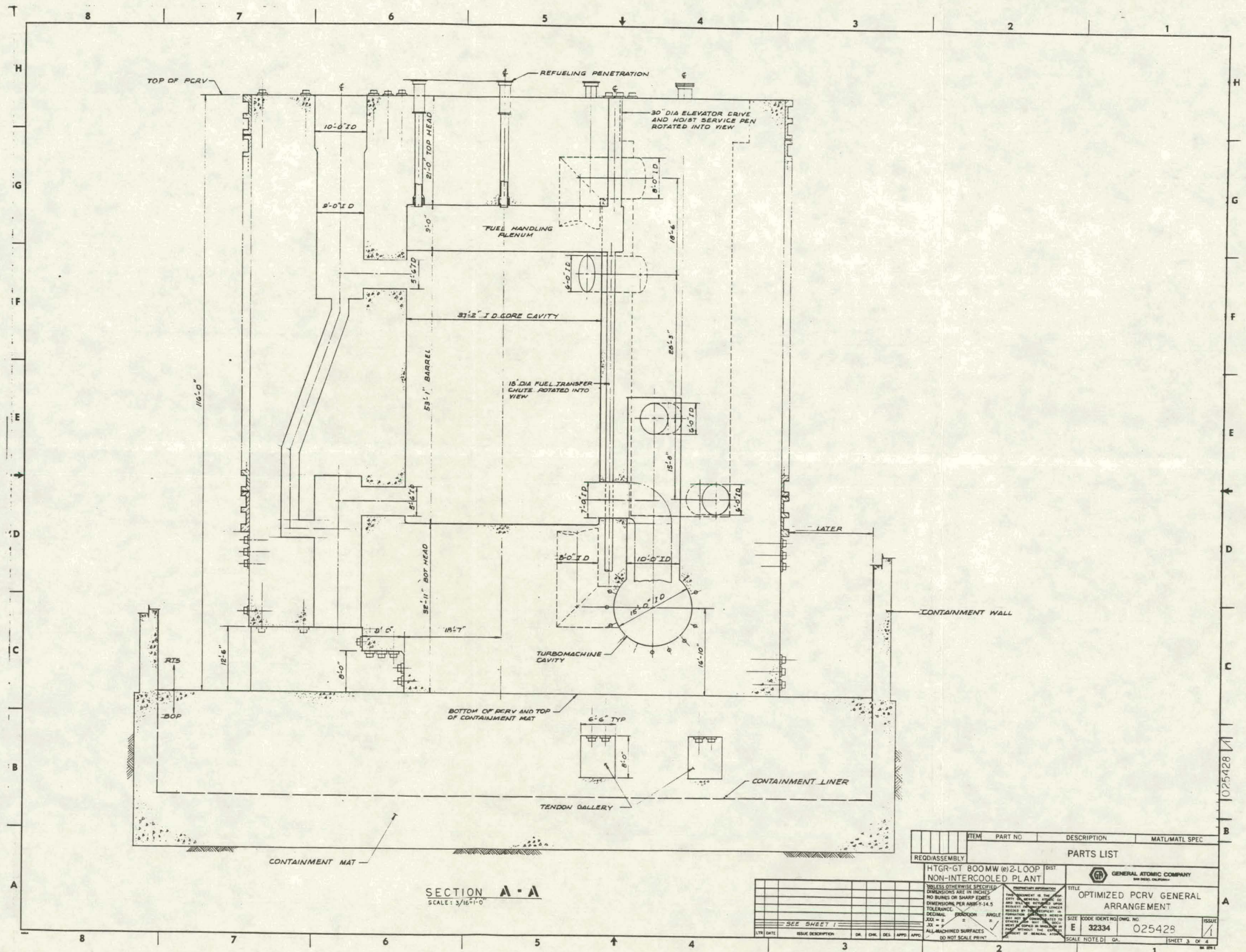


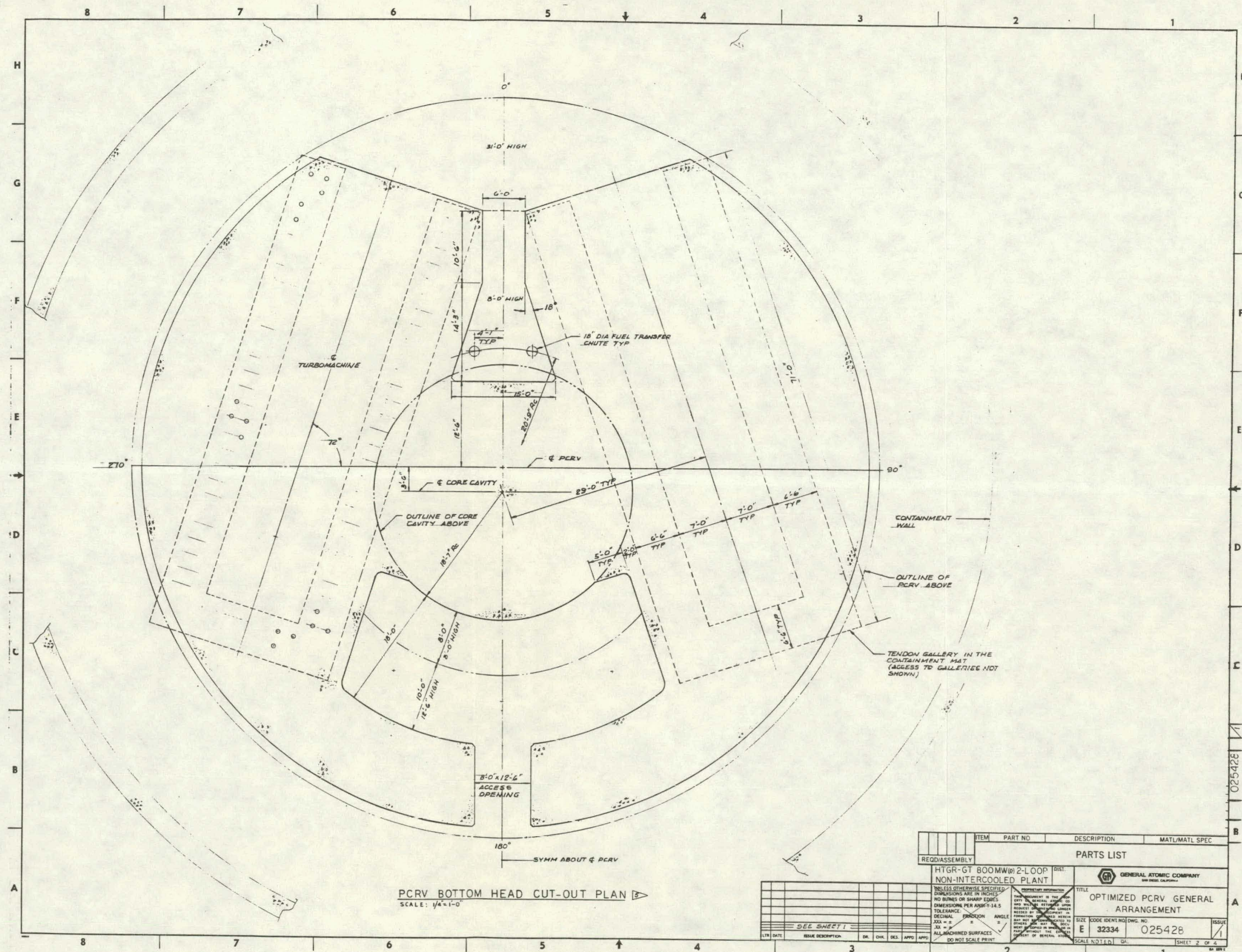
NEXT IDENTIFIED DOCUMENT			
ISSUED BY	DATE	PROJECT	NO.
PI	25	10.01	6875
GAS TURBINE HTGR			
HELIUM CONTROL VALVE			
ACTUATING SYSTEM			
(SINGLE LOOP) LAYOUT			
PROJ. NO. 025020		DRAWING NUMBER 1001	
EACD		1	

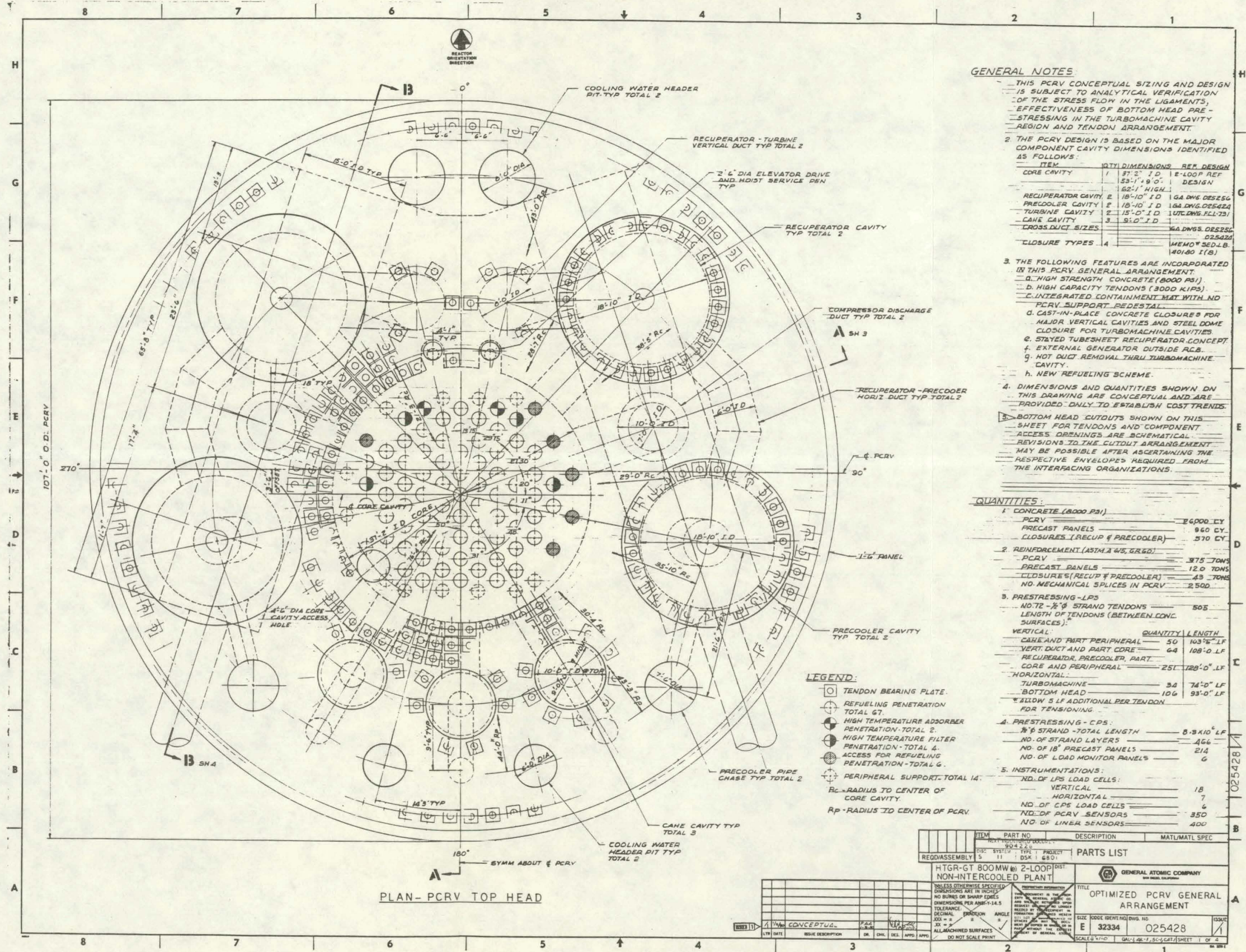


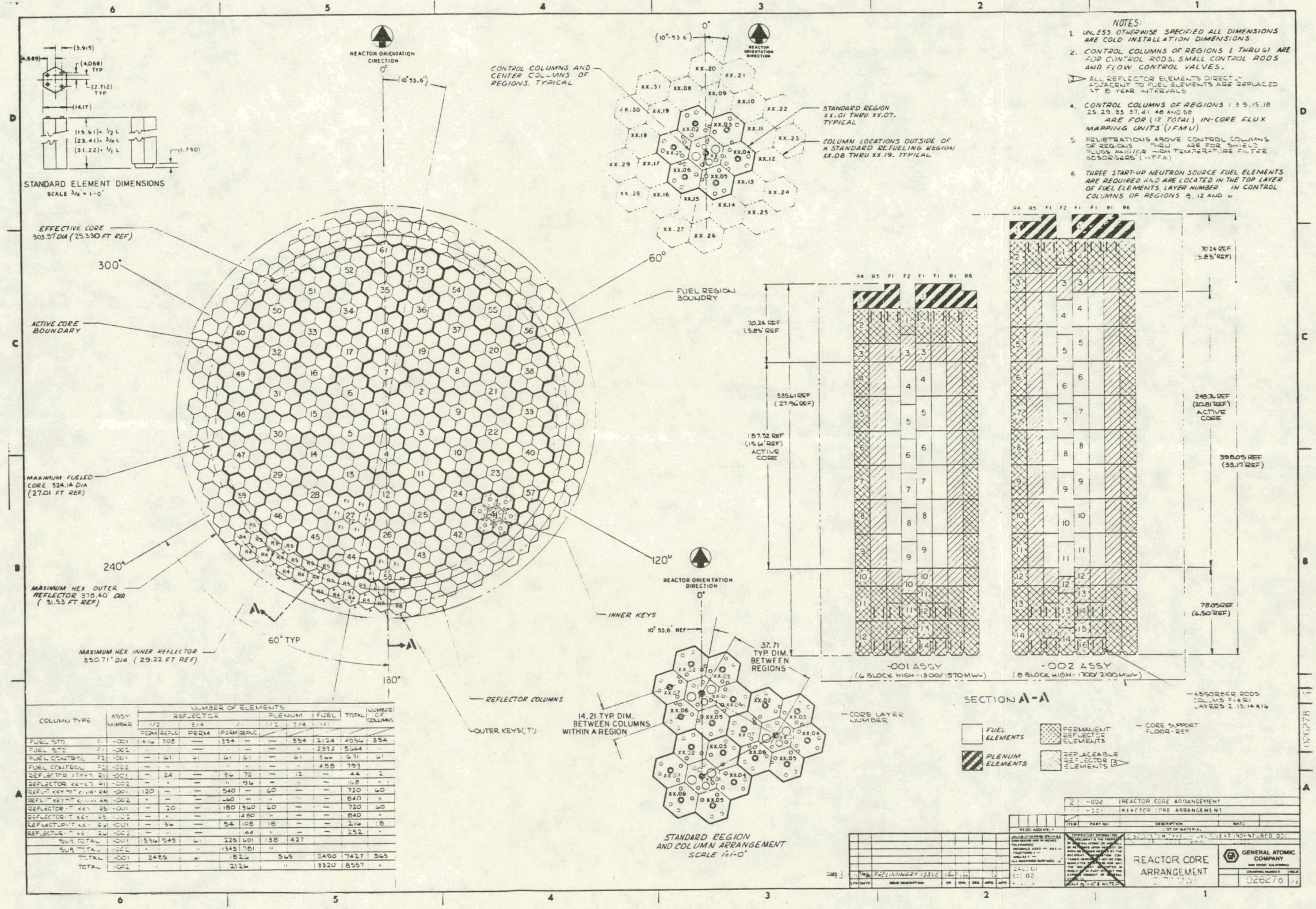








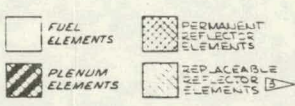




- NOTES:
1. UNLESS OTHERWISE SPECIFIED ALL DIMENSIONS ARE COLD INSTALLATION DIMENSIONS.
 2. CONTROL COLUMNS OF REGIONS 1 THRU 6 ARE FOR CONTROL RODS, SMALL CONTROL RODS AND FLOW CONTROL VALVES.
 3. ALL REFLECTOR ELEMENTS DIRECTLY ADJACENT TO FUEL ELEMENTS ARE REPLACED AT 5 YEAR INTERVALS.
 4. CONTROL COLUMNS OF REGIONS 1, 3, 9, 15, 18, 25, 29, 33, 37, 41, 48 AND 58 ARE FOR (17 TOTAL) IN-CORE FLUX MAPPING UNITS (IFMU).
 5. PENETRATIONS ABOVE CONTROL COLUMNS OF REGIONS 1 THRU 6 ARE FOR SHIELD TUBES AND/OR HIGH TEMPERATURE FILTER ADSORBERS (HTFA).
 6. THREE START-UP NEUTRON SOURCE FUEL ELEMENTS ARE REQUIRED AND ARE LOCATED IN THE TOP LAYER OF FUEL ELEMENTS, LAYER NUMBER 1, IN CONTROL COLUMNS OF REGIONS 8, 12 AND 16.

COLUMN TYPE	ASSY NUMBER	NUMBER OF ELEMENTS						TOTAL	NUMBER OF COLUMNS	
		REFLECTOR			PLENUM					
		1/2			3/4					
		1/2	3/4	1	1/2	3/4	1			
FUEL STD	1-001	416	708	354	—	—	354	2124	4956	354
FUEL STD	2-002	—	—	—	—	—	—	2832	5664	—
FUEL CONTROL	3-001	—	61	61	61	61	61	366	671	61
FUEL CONTROL	4-002	—	—	—	—	—	—	458	793	—
REFLECTOR KEYS (1)	5-001	—	24	—	36	72	—	12	44	2
REFLECTOR KEYS (2)	6-002	—	—	—	—	96	—	—	108	—
REFLECTOR KEYS (3)	7-001	120	—	—	540	—	60	—	720	60
REFLECTOR KEYS (4)	8-002	—	—	—	60	—	—	—	840	—
REFLECTOR KEYS (5)	9-001	—	20	—	180	360	40	—	720	60
REFLECTOR KEYS (6)	10-002	—	—	—	—	180	—	—	840	—
REFLECTOR KEYS (7)	11-001	—	36	—	54	108	18	—	216	18
REFLECTOR KEYS (8)	12-002	—	—	—	—	66	—	—	252	—
SUB TOTAL	1-001	536	949	61	225	601	138	427	—	—
SUB TOTAL	2-002	—	—	—	—	1345	701	—	—	—
TOTAL	1-001	245	—	—	152	—	565	2490	7427	565
TOTAL	2-002	—	—	—	2124	—	—	3320	8557	—

SECTION A-A



ITEM	PART NO.	DESCRIPTION	MAT.
1	001	REACTOR CORE ARRANGEMENT	
2	002	REACTOR CORE ARRANGEMENT	

REACTOR CORE ARRANGEMENT

GENERAL ATOMIC COMPANY

DRAWING NUMBER 02070

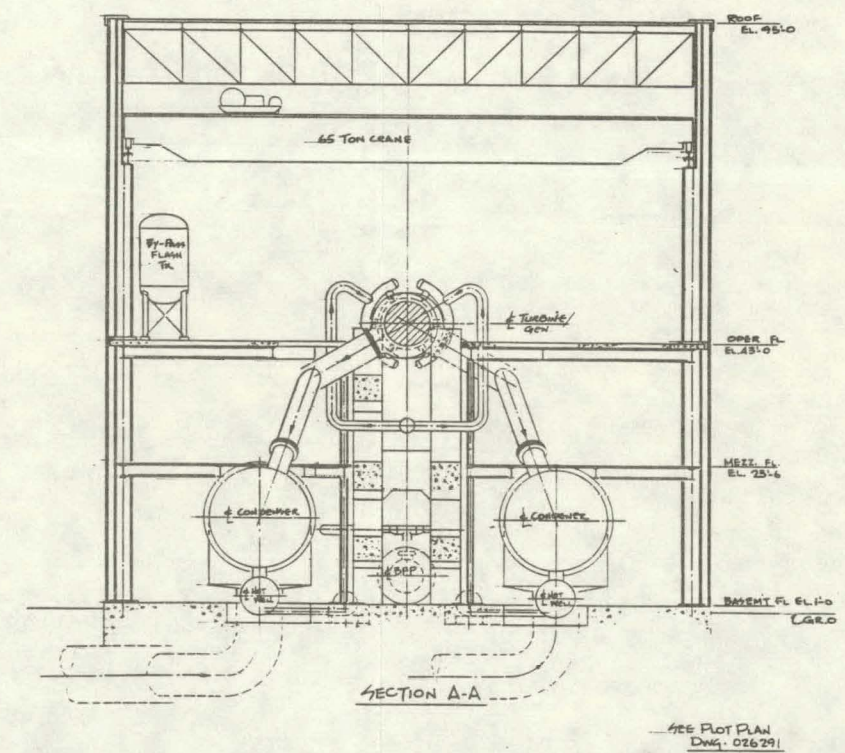
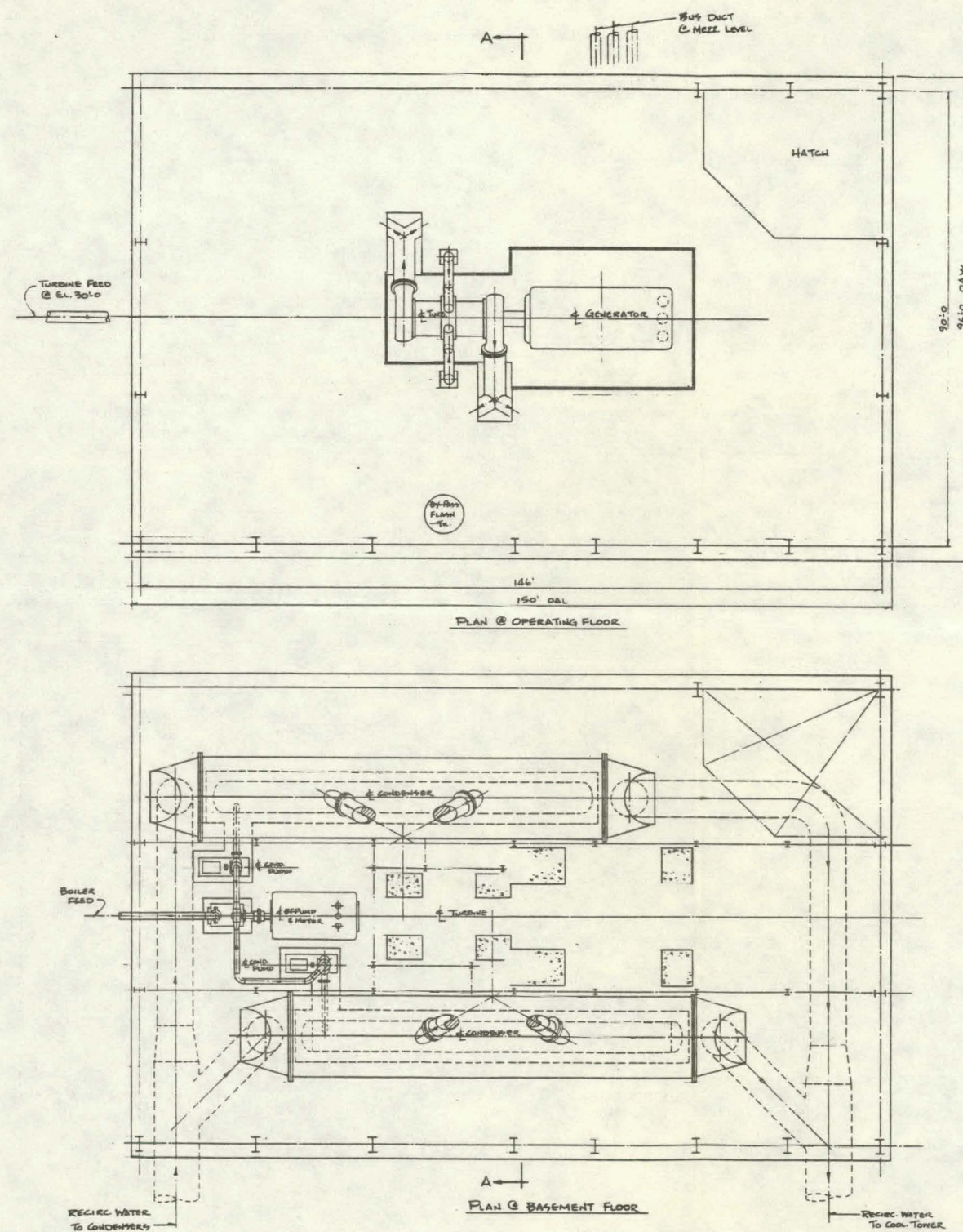
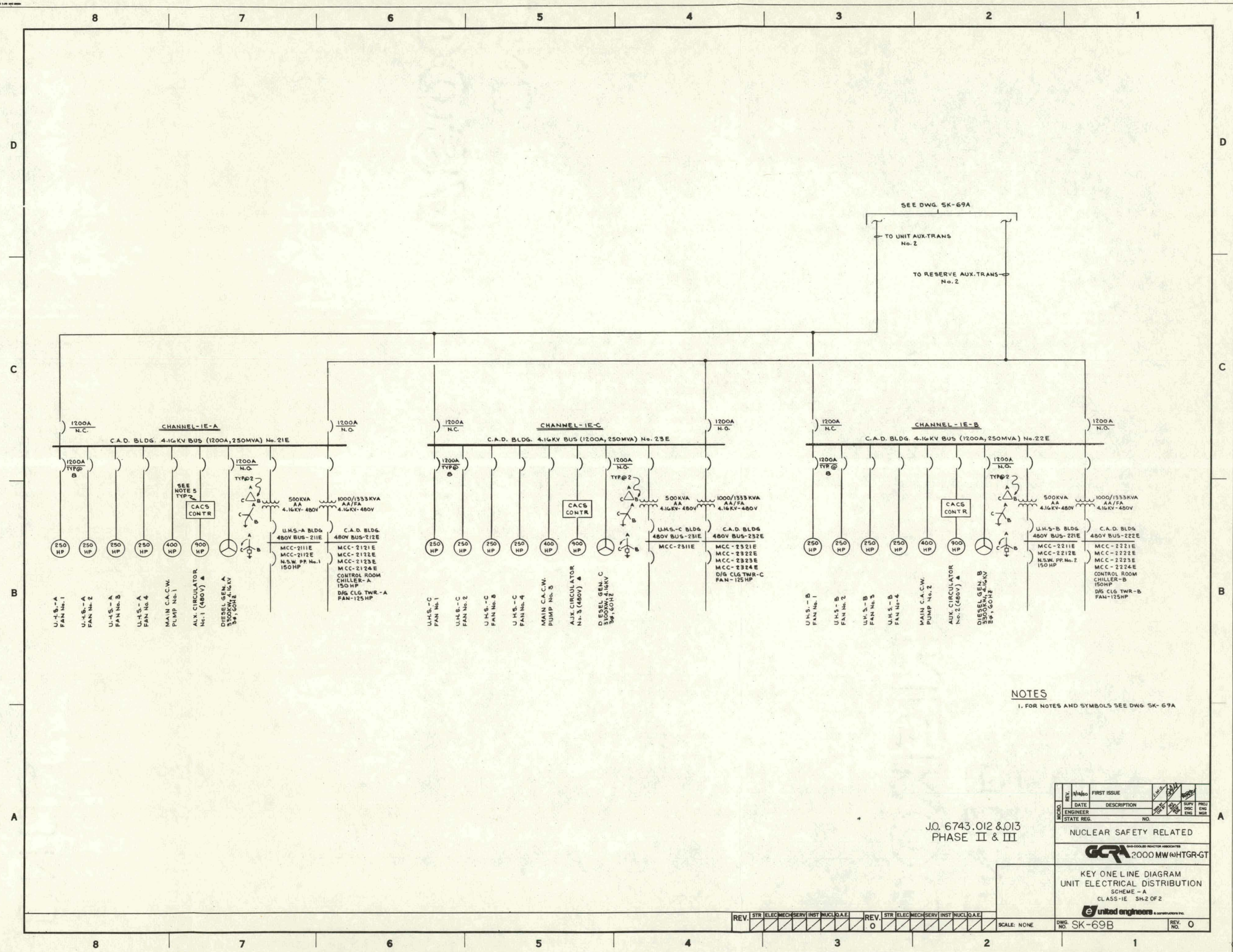


Figure 3.3.2-6 Ammonia bottoming cycle turbine building layout



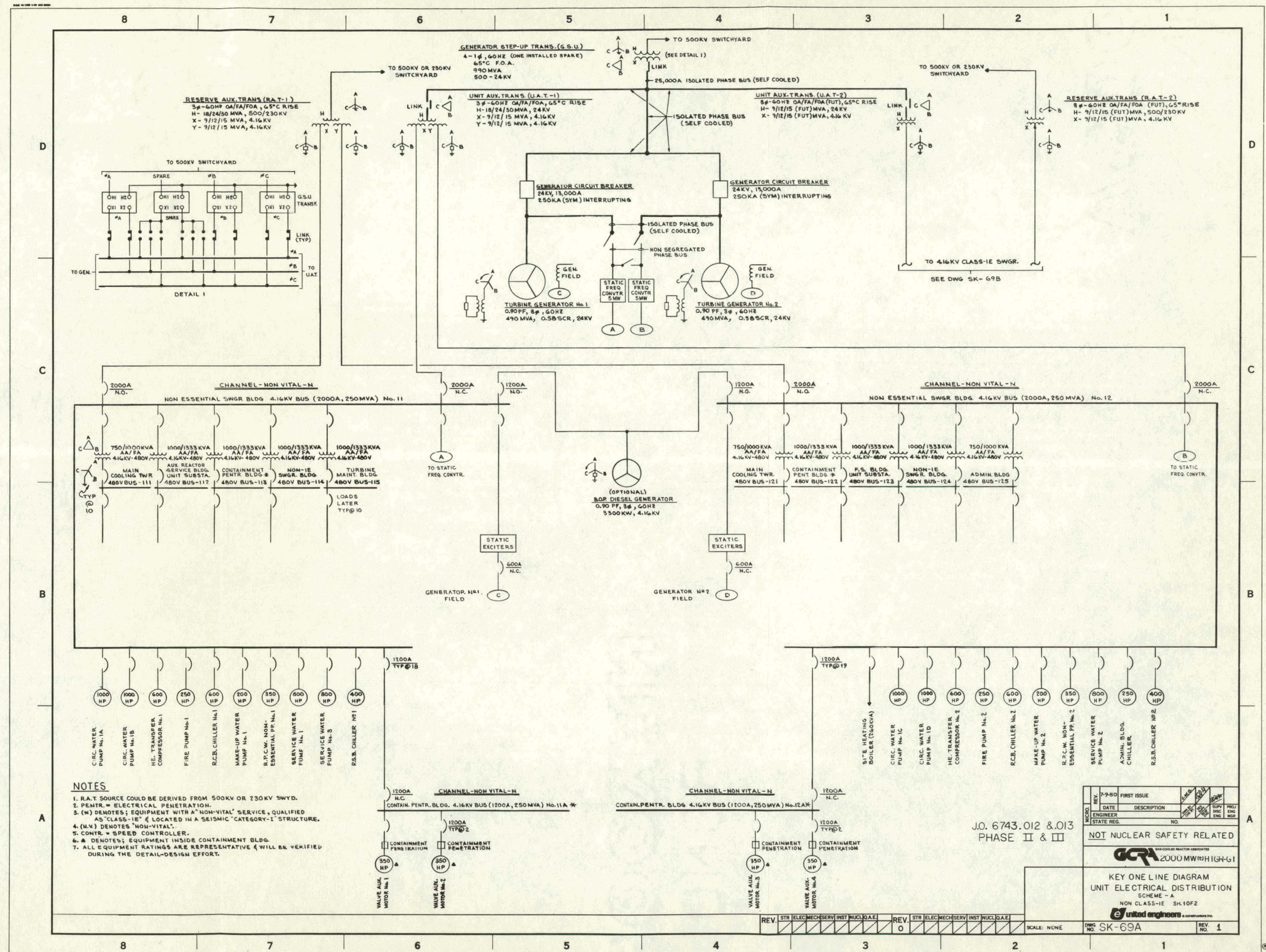
NOTES
1. FOR NOTES AND SYMBOLS SEE DWG SK-69A

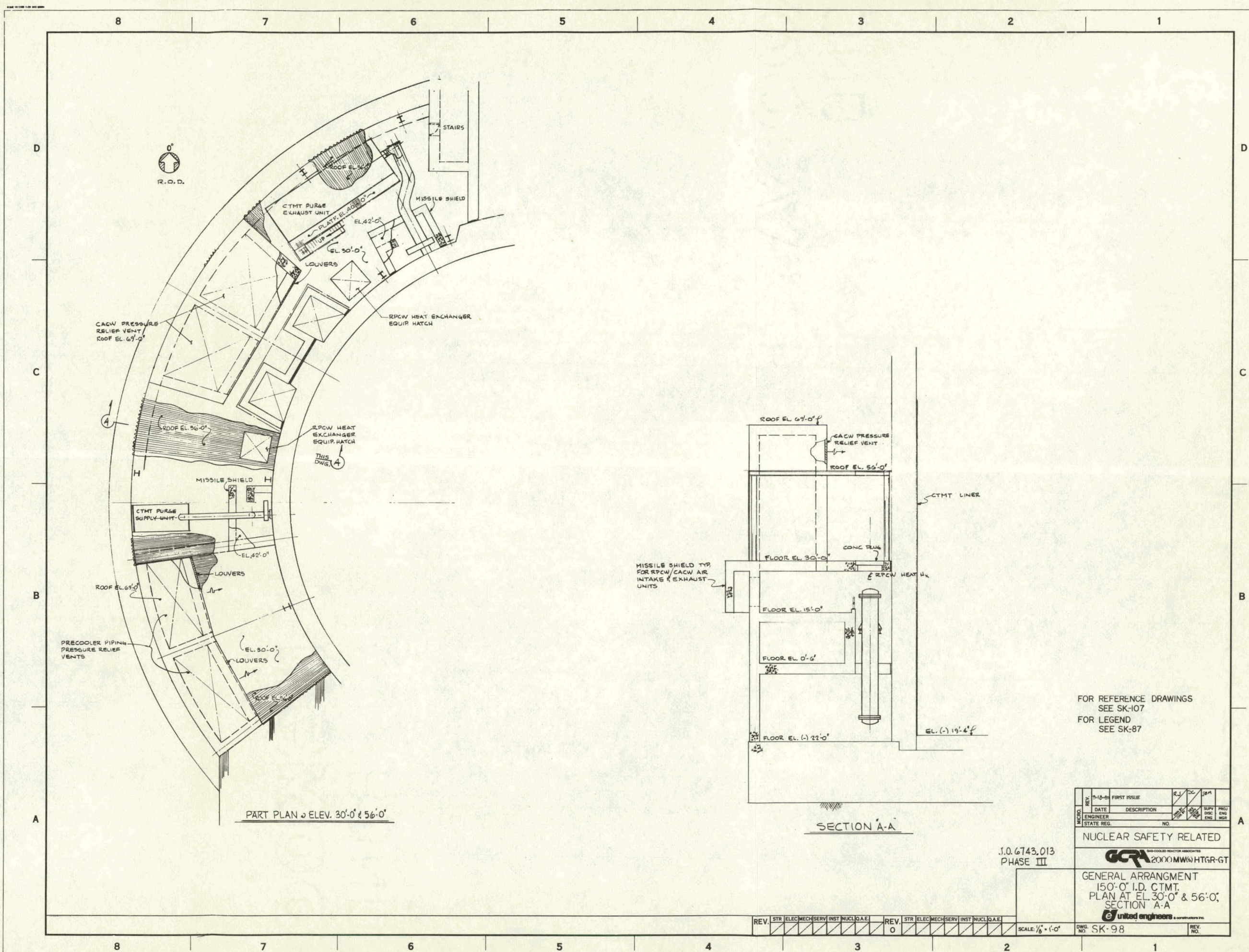
J.O. 6743.012 & 013
PHASE II & III

REV.	STR.	ELEC.	MECH.	SERV.	INST.	NUCL.	0.A.E.	REV.	STR.	ELEC.	MECH.	SERV.	INST.	NUCL.	0.A.E.
								0							

SCALE: NONE
DWG. NO. SK-69B
REV. NO. 0

REV.	DATE	DESCRIPTION	NO.
1			
NUCLEAR SAFETY RELATED			
GCR 2000 MW WHTGR-GT			
KEY ONE LINE DIAGRAM UNIT ELECTRICAL DISTRIBUTION SCHEME - A CLASS-1E SH-2 OF 2			
e united engineers			

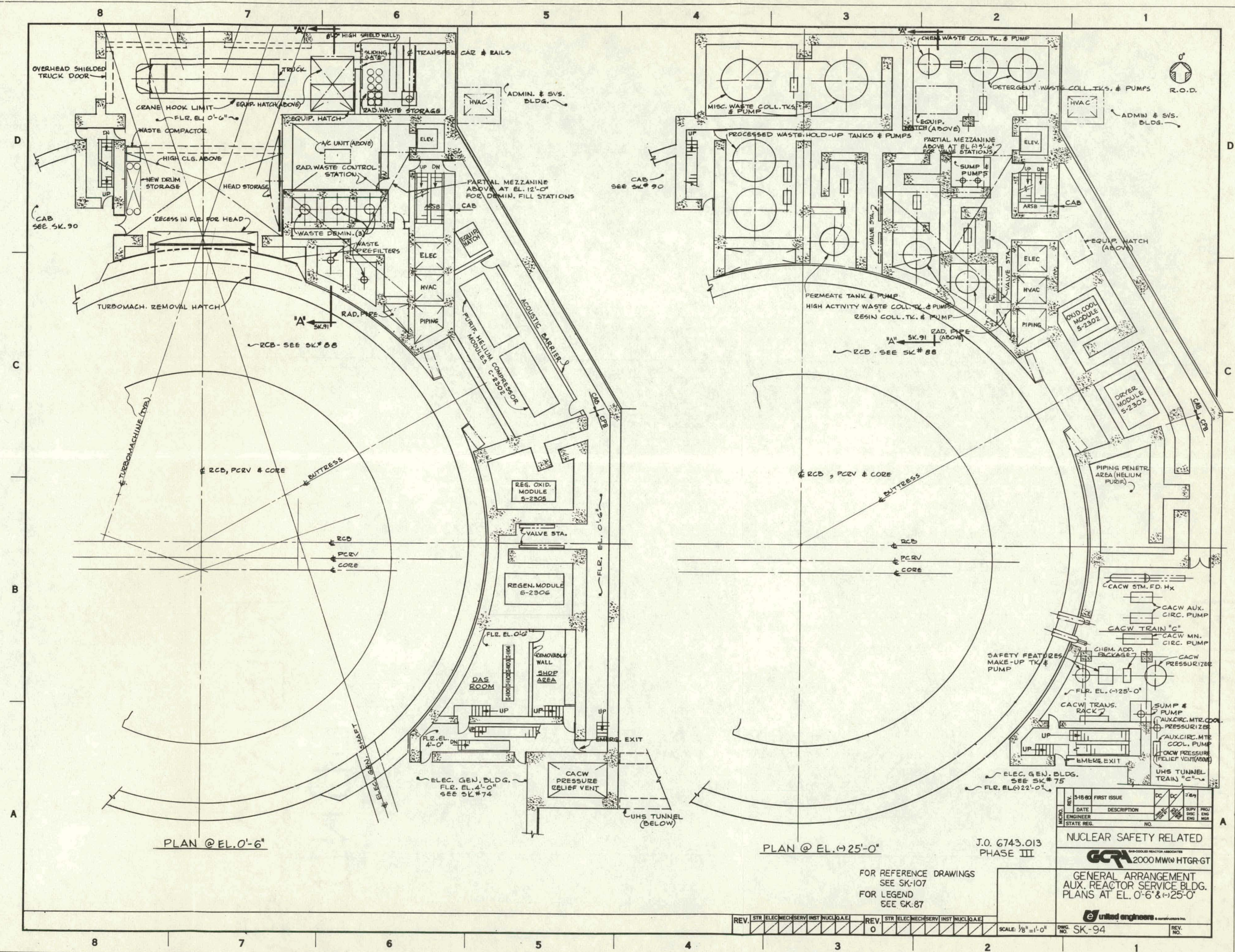




FOR REFERENCE DRAWINGS
SEE SK-107
FOR LEGEND
SEE SK-87

J.O. 6743.013
PHASE III

DATE	DESCRIPTION	NO.	REV.	DATE	DESCRIPTION	NO.	REV.
1-15-01	FIRST ISSUE						
ENGINEER				STATE REG.			
NUCLEAR SAFETY RELATED							
2000 MW WHTGR-GT							
GENERAL ARRANGMENT							
150'-0" I.D. CTMT.							
PLAN AT EL. 30'-0" & 56'-0"							
SECTION A-A							
United engineers & architects, inc.							
REV.	STR	ELEC	MECH	SERV	INST	NUCL	DAE
0							
SCALE	1/8" = 1'-0"						
DWG. NO.	SK-98						REV. NO.



PLAN @ EL. 0'-6"

PLAN @ EL. +25'-0"

FOR REFERENCE DRAWINGS
SEE SK.107
FOR LEGEND
SEE SK.87

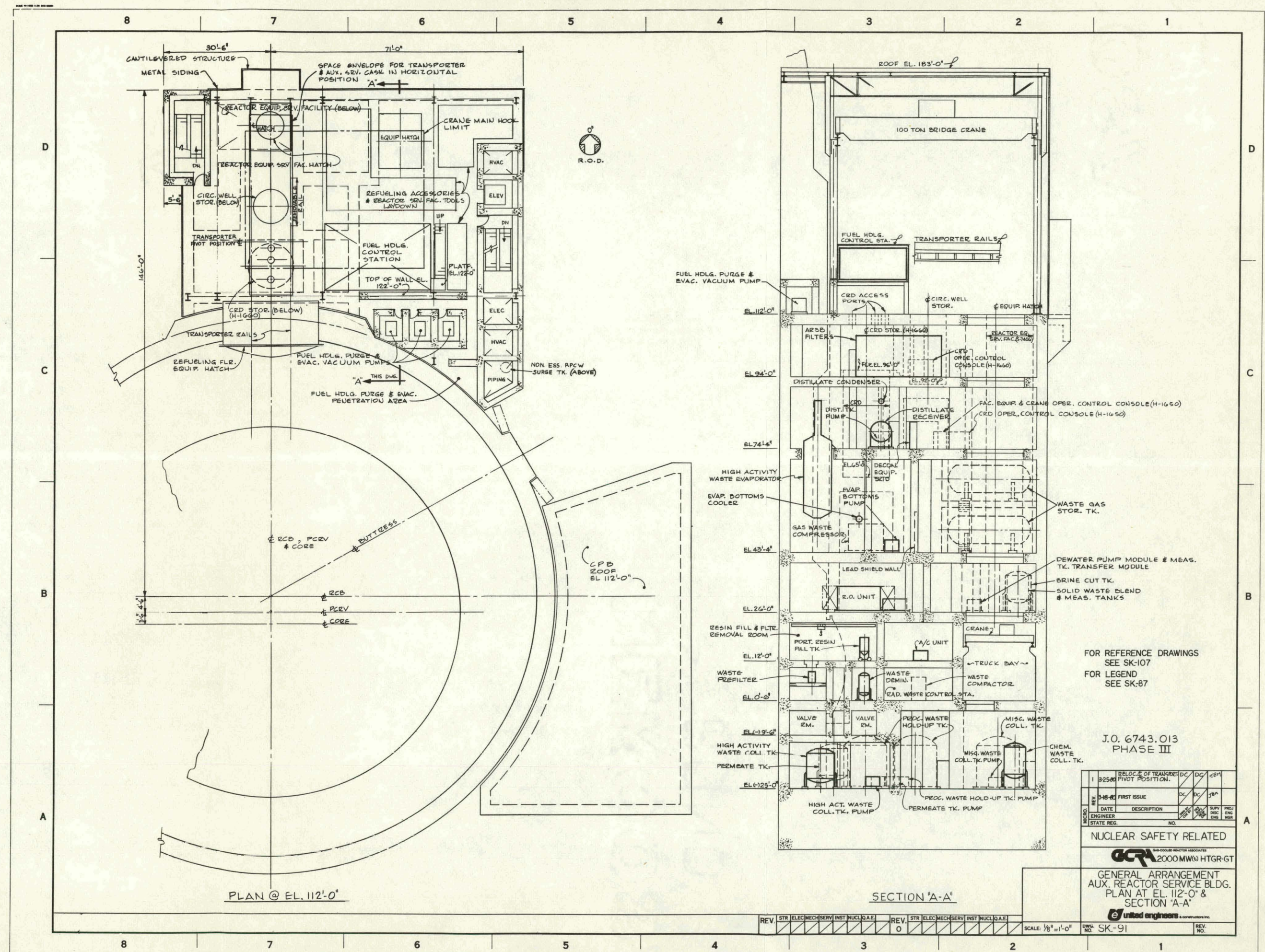
J.O. 6743.013
PHASE III

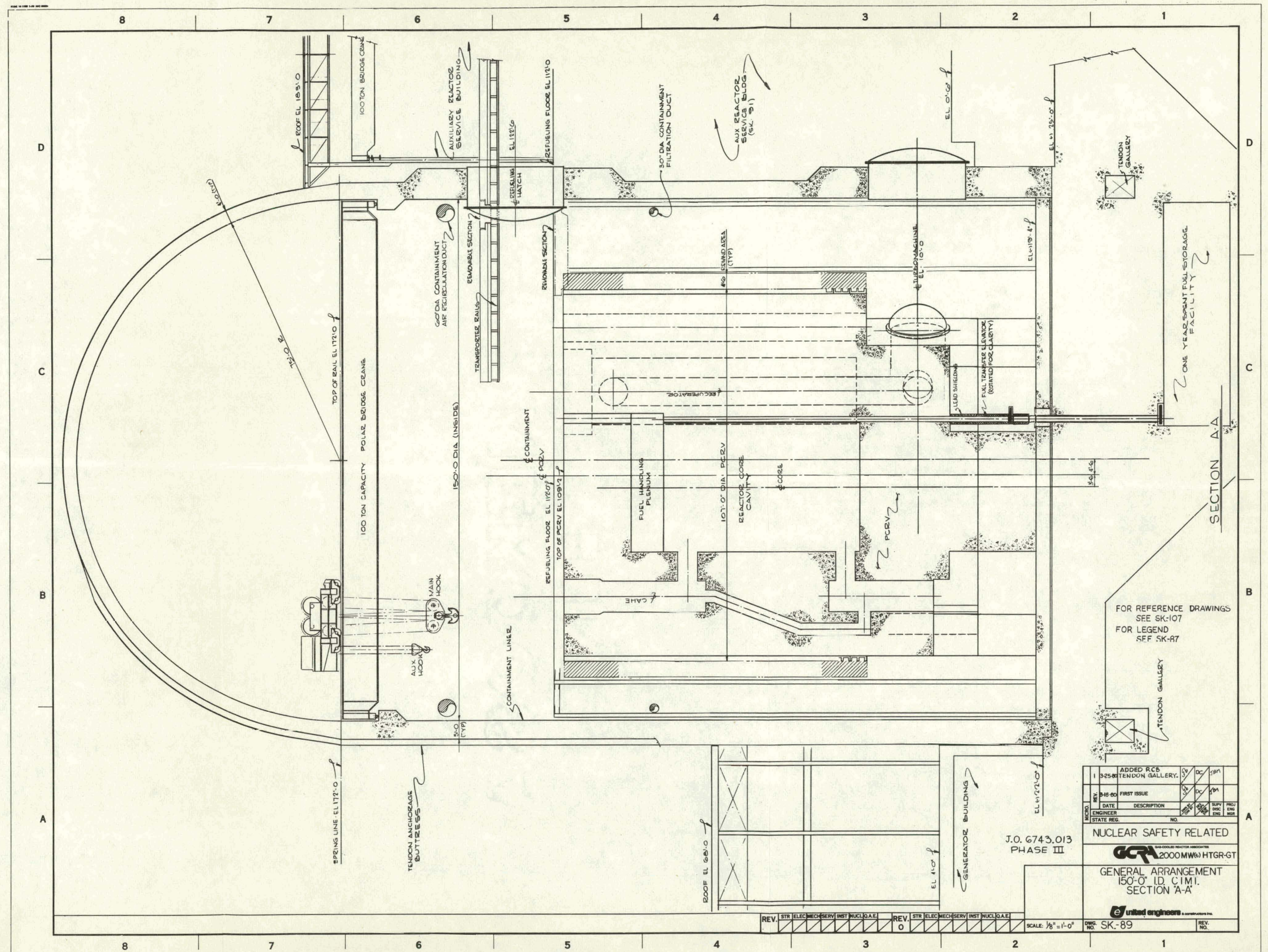
NUCLEAR SAFETY RELATED			
2000MW HTGR-GT			
GENERAL ARRANGEMENT AUX. REACTOR SERVICE BLDG. PLANS AT EL. 0'-6" & +25'-0"			
united engineers			

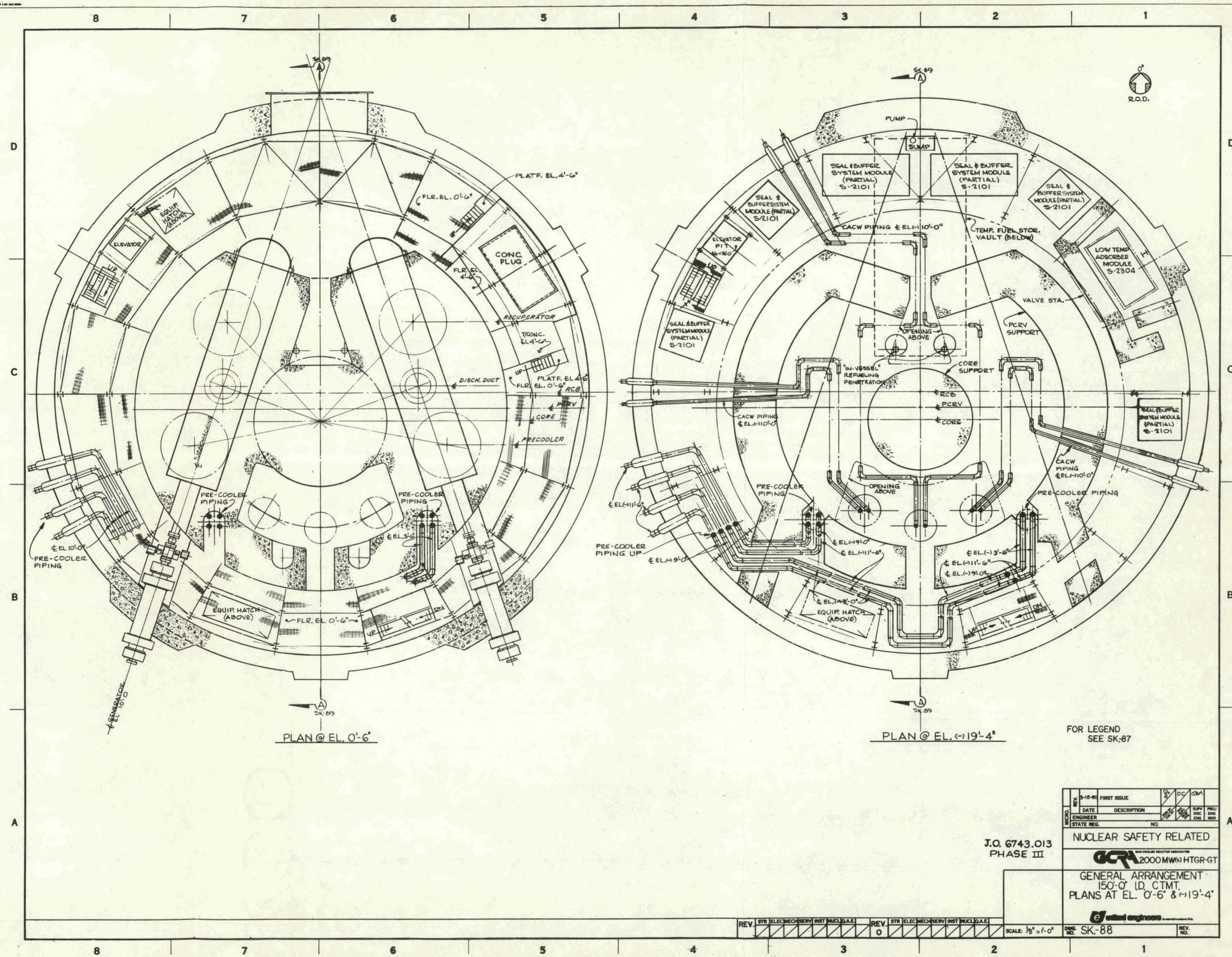
REV	DATE	DESCRIPTION	BY	CHK	APP
1	5-18-80	FIRST ISSUE	DC	DC	729
2					
3					
4					
5					
6					
7					
8					

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

SK. 94







PLAN @ EL. 0'-6"

PLAN @ EL. (-) 19'-4"

FOR LEGEND
SEE SK-87

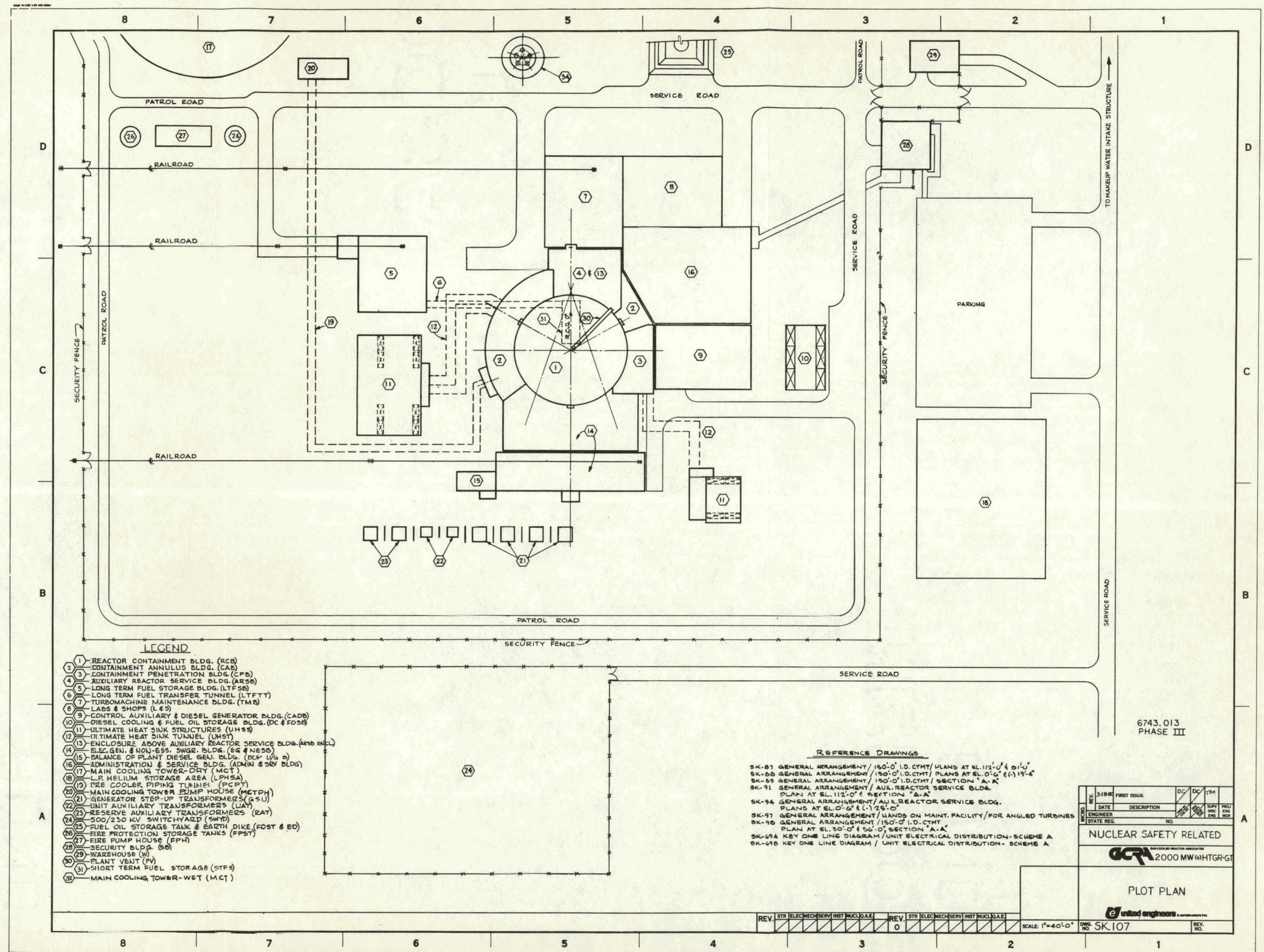
J.O. 6743.013
PHASE III

3-15-90	FIRST ISSUE	DC	CM
DATE	DESCRIPTION	SUPV	PROJ
ENGINEER		ENG	ENG
STATE REG.		NO.	
NUCLEAR SAFETY RELATED			
2000 MW HTGR-GT			
GENERAL ARRANGEMENT 150'-0" ID. CTMT. PLANS AT EL. 0'-6" & (-) 19'-4"			
SK-88			

REV	STR	ELEC	MECH	SERV	INST	NUCL	DATE	REV	STR	ELEC	MECH	SERV	INST	NUCL	DATE

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

SK-88



APPENDIX D

COMMERCIAL PLANT COST DATA

D.0 COMMERCIAL PLANT COST DATA

This Appendix contains base (capital) cost data and payment/cash flow schedules prepared for the commercial plant in the course of this study. Cost data for the RTS were developed by General Atomic Company, and data for the BOP were prepared by United Engineers & Constructors. Table D-1 provides summary definition of capital costs estimated for the commercial HTGR-GT plant. The commercial base plant cost was estimated on the basis of a mature industry where unit cost data appropriate to current LWR projects are applied, all manufacturing learning has been achieved, and non-recurring RTS costs are amortized. To determine the total investment cost presented in Section 3, payment schedules for the RTS and BOP were assumed as shown on Figs. D-1 and D-2 respectively. Given the assumed 6% compound escalation rate and 10% simple interest during construction charge, the total plant cash flow shown on Fig. D-3 was developed. This curve represents the total commercial plant expenditure during the period from construction permit to commercial operation and includes capital, escalation, and interest charges.

TABLE D-1

HTGR-GT COMMERCIAL PLANT BASE COSTS
(1980 \$ x 10⁶)

	<u>Commercial Plant</u>
<u>RTS DIRECT COSTS</u>	
PCRVR Data Acquisition System	1.0
PCRVR Liner and Penetrations	27.7
Thermal Barrier	23.0
Control Rod Drives	8.4
Fuel Handling Equipment	16.3
Control Rod Drive Storage System	*
Reactor Internal Components	13.4
Reactor Core System	11.7
Heat Exchanger System	31.5
Turbomachinery System	30.2
Control Valve System	3.2
Auxiliary Circulator Bearing Water System	*
Helium Service Equipment	0.4
Auxiliary Circulator	5.6
Core Auxiliary Heat Exchanger	2.5
Plant Protection System	2.2
Plant Control System	1.2
Analytic Instrumentation	*
Subtotal	178.3
<u>RTS INDIRECT COSTS</u>	15.7
<u>RTS LEASED EQUIPMENT</u>	1.3
Total RTS	195.3

*In BOP

TABLE D-1
(Continued)

HTGR-GT COMMERCIAL PLANT BASE COSTS

	<u>Commercial Plant</u>
<u>BOP DIRECT COSTS</u>	
<u>STRUCTURES AND IMPROVEMENTS</u>	
Yardwork	8.6
Reactor Containment Building	60.2
Turbomachine Maintenance Building	4.4
Security Building	.4
Auxiliary Reactor Service Building	9.4
Long-Term FSB Building	6.4
Control, Auxiliaries, and D.G. Building	15.7
Administration and Service Building	3.8
Shop and Lab Building	2.1
Fire Pump House	.2
L.P. Helium Storage Area	1.9
Electric Generator Building	2.5
Diesel Cooling and Fuel Oil Storage Building	2.6
Warehouse	.2
Containment Annulus Building	3.6
Containment Penetration Building	5.8
Holding Pond and Control House	.3
Ultimate Heat Sink Structure and Tunnels	6.7
Control Room Emergency Air	<u>.1</u>
Subtotal	134.9
<u>REACTOR PLANT EQUIPMENT</u>	
RTS Options	60.2*
Reactor Equipment	4.3
Main Heat Transport System	3.5
Safeguards Cooling System	6.1
Radwaste Processing	3.7
Nuclear Fuel Handling	5.0
Other Reactor Plant Equipment	15.5
Instrumentation and Control	3.2
Reactor Plant Misc. Items	<u>1.7</u>
Subtotal	103.2

*Includes Electrical Generators

TABLE D-1
(Continued)

HTGR-GT COMMERCIAL PLANT BASE COSTS

	<u>Commercial Plant</u>
<u>ELECTRIC PLANT EQUIPMENT</u>	
Switchgear	8.4
Station Service Equipment	11.9
Switchboards	1.3
Protective Equipment	1.8
Electric Structure and Wiring Centers	9.3
Power and Control Wiring	<u>18.3</u>
Subtotal	51.0
<u>MISCELLANEOUS PLANT EQUIPMENT</u>	
Transportation and Lift Equipment	3.9
Air, Water, and Steam Service Systems	5.7
Communications Equipment	1.6
Furnishings and Fixtures	<u>1.2</u>
Subtotal	12.4
<u>HEAT REJECTION SYSTEM</u>	
Structures	1.0
Mechanical Equipment	<u>26.4</u>
Subtotal	<u>27.4</u>
Subtotal Direct Costs	330.0
<u>BOP INDIRECT COSTS</u>	
Construction Services	77.5
Home Office Engineering and Service	100.0
Field Office Engineering and Service	<u>32.4</u>
Subtotal Indirect Costs	210.0
BOP CONTINGENCY	35
TOTAL BOP COST	575
TOTAL BASE PLANT COST	770

Figure D-1

**2000 MW(t) HTGR-GT
COMMERCIAL PLANT
NSSS CASH FLOW**

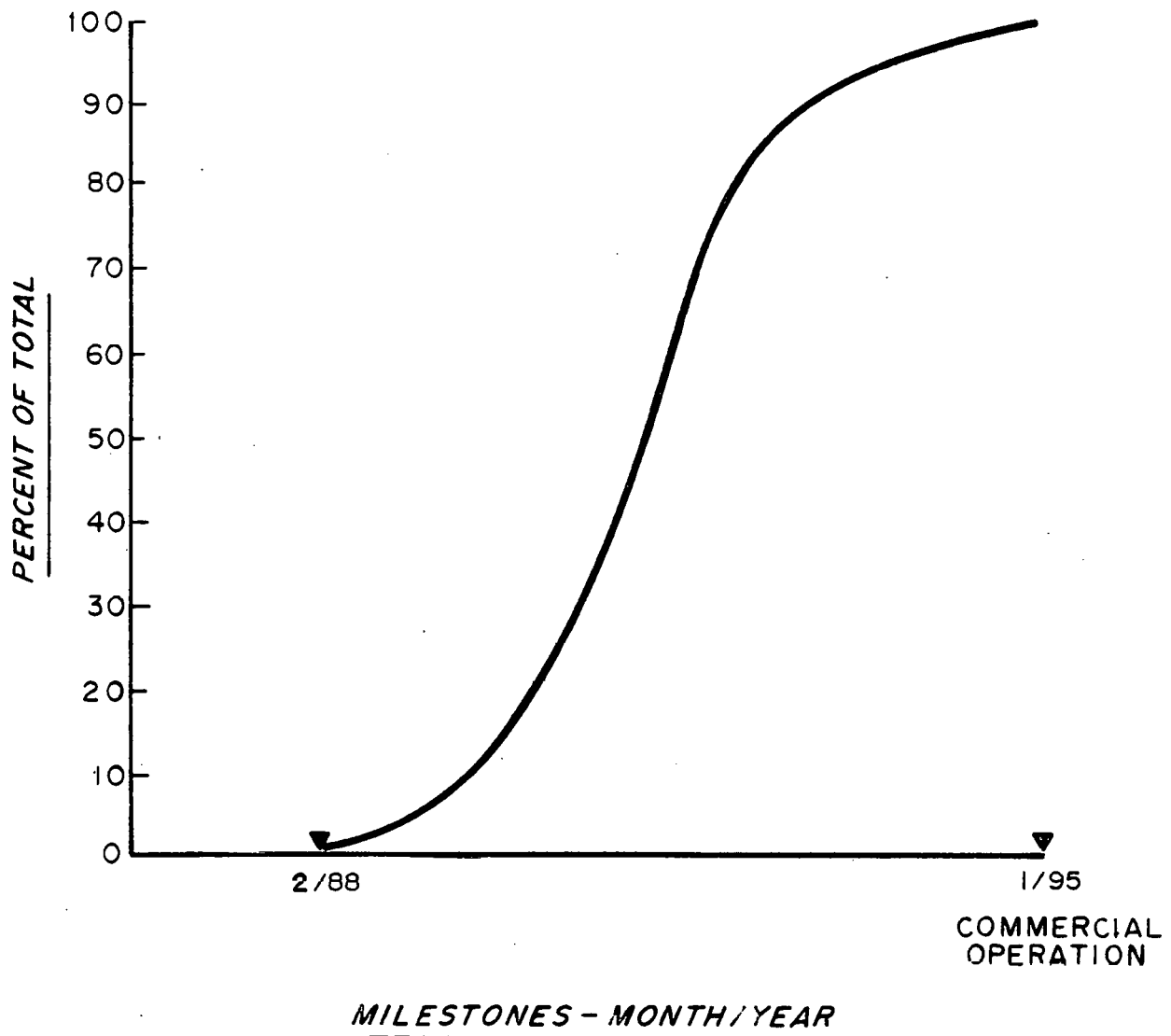
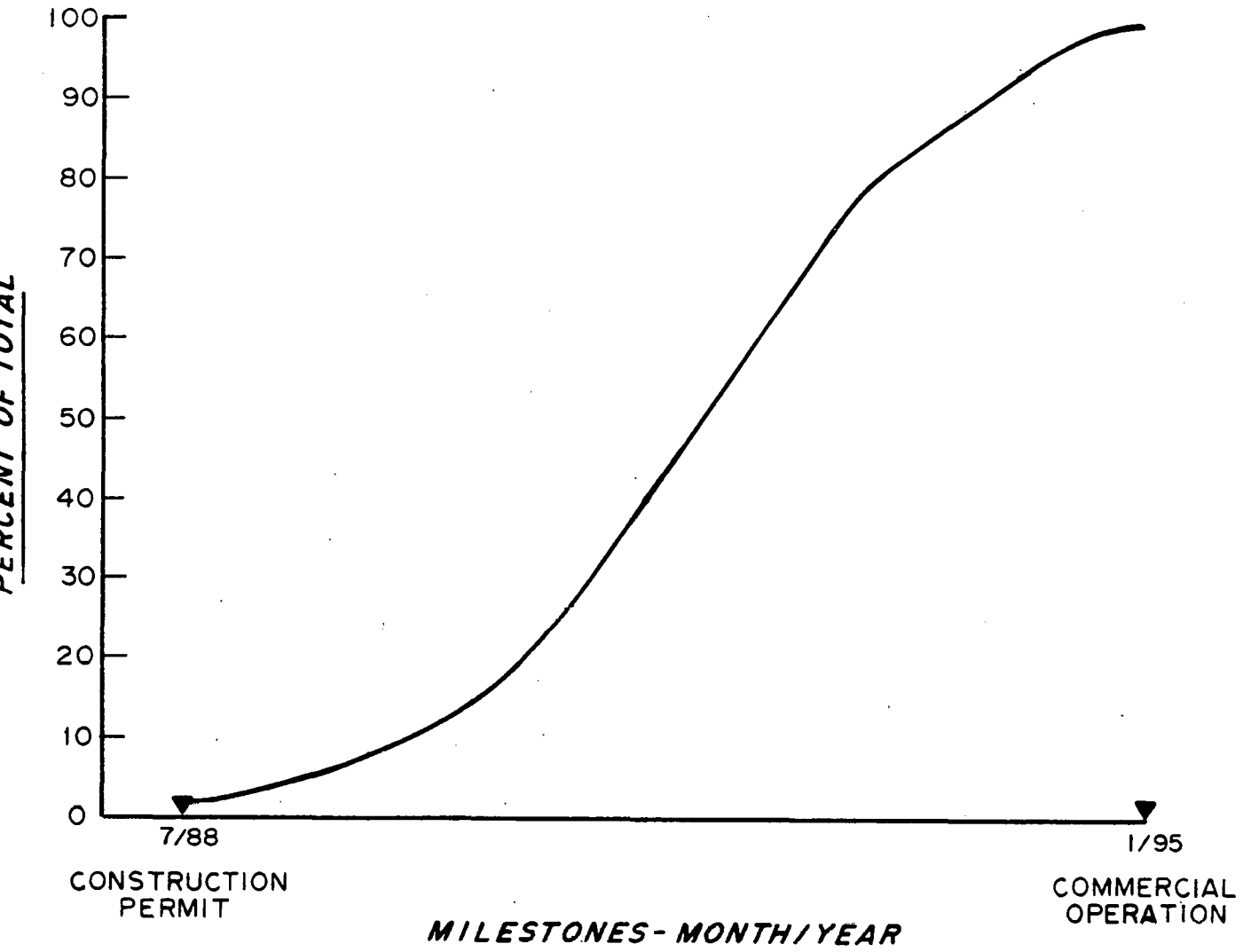


Figure D-2

**2000 MW(t) HTGR-GT
COMMERCIAL PLANT
BALANCE OF PLANT
CASH FLOW**



NOTE INITIAL LUMP SUM AT CP ASSUMED TO BE 2.2 %
WHICH REFLECTS ALL EXPENDITURES PRIOR TO
CP FROM PROJECT START DATE.

Figure D-3

**2000 MW (t) HTGR-GT
COMMERCIAL PLANT
TOTAL CASH FLOW**

