DOE/SF/02034-T10

HIGH TEMPERATURE GAS-COOLED REACTOR

REFORMER APPLICATION STUDY

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LA JOLLA, CALIFORNIA

DECEMBER 1980

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December 1980

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ACKNOWLEDGEMENT

The primary contributors to the HTGR-Reformer Application Study have been:

Gas-Cooled Reactor Associates
General Atomic Company
General Electric Company
United Engineers & Constructors Inc.

In addition, other Program participants and individual utility personnel have provided invaluable review and comment.

The support, encouragement, and active participation by the U.S. Department of Energy in the development of this Application Study are gratefully acknowledged.

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EXECUTIVE SUMMARY

1.0 INTRODUCTION

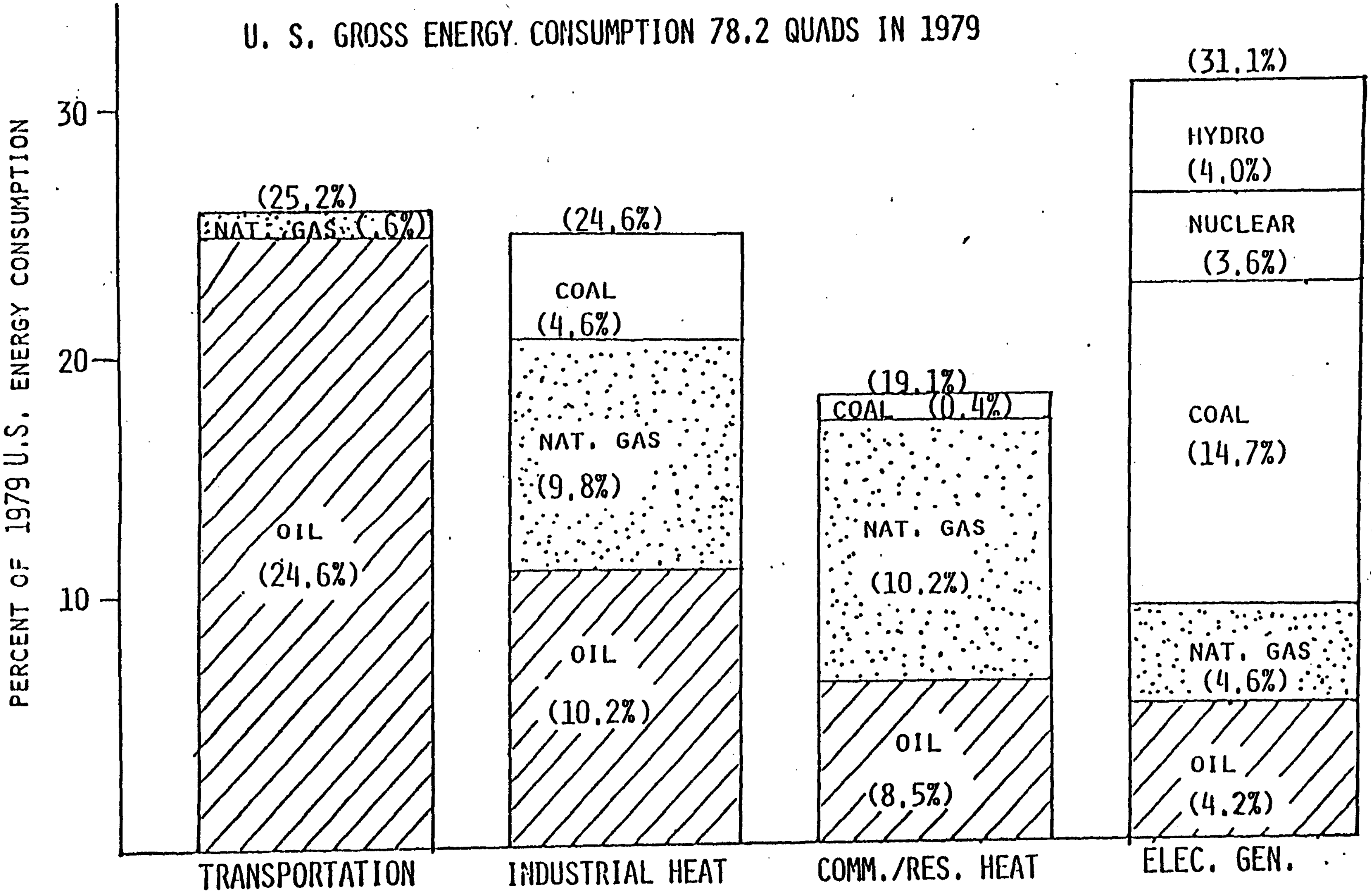
The high-temperature capability of the High-Temperature Gas-Cooled Reactor (HTGR) is a distinguishing characteristic which has long been recognized both within the U.S. and within foreign nuclear energy programs. This high-temperature capability of the HTGR concept leads to increased efficiency in conventional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

Exploiting the high-temperature capabilities of the HTGR for industrial process heat may have profound implications regarding the future energy security of the nation. As shown in Fig. 1, nearly 73% of the energy consumed in the U.S. in 1979 was based upon oil and gas. Nuclear energy accounted for only 3.6% of total energy consumption and in that year was confined to electrical generation. It is evident from Fig. 1 that if nuclear energy is to make a major contribution to reduction of oil and gas use, an expansion into the sectors of transportation, industrial heat, and commercial/residential heat must take place. In this regard, it is probable that current nuclear-electric generation systems, coupled with modern heat pump technology, will make significantly greater inroads within the commercial/residential heating sector. Current nuclear potential within the transportation sector (through the emerging synfuels program) and within the industrial heat sector, however, is restricted by the low-temperature capabilities of current LWR systems.

With the HTGR, the potential for oil and gas displacement in the transportation and industrial sectors is greatly expanded. Using current technology, as typified by Fort St. Vrain, modern steam conditions can be achieved with nuclear energy, and sensible energy transmission and storage systems can be considered. With a further advancement of the state of the art, systems for high-temperature direct heat and thermochemical energy transmission and storage can be developed. Through such systems, the ultimate potential for nuclear fission energy can be realized.

While the unique capabilities of the HTGR 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.

Figure 1


U. S. GROSS ENERGY CONSUMPTION 78.2 QUADS IN 1979
PERCENT OF 1979 U.S. ENERGY CONSUMPTION
30
20
10
(25.2%)
NAT. GAS (.6%)
OIL
(24.6%)
(24.6%)
COAL
(4.6%)
NAT. GAS
(9.8%)
OIL
(10.2%)
(19.1%)
COAL (0.4%)
NAT. GAS
(10.2%)
OIL
(8.5%)
(31.1%)
HYDRO
(4.0%)
NUCLEAR
(3.6%)
COAL
(14.7%)
NAT. GAS
(4.6%)
OIL
(4.2%)
TRANSPORTATION
INDUSTRIAL HEAT
COMM./RES. HEAT
ELEC. GEN.

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 Projects. 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.

In the HTGR-R version of the HTGR, a portion of the reactor thermal energy is converted to a storable, transportable energy form through the use of a highly endothermic, reversible chemical reaction. The concept typifies high-temperature process heat applications of the HTGR and was identified in recent scoping studies as having potential technical and economic benefits when applied to energy distribution and storage and to coupling with emerging synfuels processes. Further, the earlier studies had concluded that economic incentives might exist in reforming applications with core outlet temperatures of 850°C (1562°F) or less.

The HTGR-R Application Study, therefore, was conceived and directed to evaluate the HTGR-R with a core outlet temperature of 850°C as a near-term Lead Project and as a vehicle to long-term HTGR Program objectives. The scope of this effort included evaluation of the HTGR-R technology, evaluation of potential HTGR-R markets, assessment of the economics of commercial HTGR-R plants, and the evaluation of the program scope and expenditures necessary to establish HTGR-R technology through the completion of the Lead Project.

2.0 SUMMARY OF APPLICATION STUDY RESULTS

The time constraints set forth in the HTGR Lead Project Identification Plan permitted consideration of the specified Lead Project configuration, including its commercial potential, but were inadequate for the development of an optimized HTGR-R commercial plant design and cost estimate. As a result, the major findings of this study to date indicate that:

- There is a large potential market for the HTGR-R if institutional and technical barriers can be overcome in a timely manner.
- The significant environmental benefits which are attained through offset of fossil fuels are a key market factor in the projected deployment of HTGR-R systems.

- The 850°C HTGR-R plant as presently configured and applied has limited economic potential.

In order to properly assess the potential of the HTGR-R and the suitability of the HTGR-R as a Lead Project, additional work must be performed before a final judgment is rendered. Design trade-off studies and alternative applications must be investigated to determine if a commercial potential exists for the HTGR-R at 850°C. If commercial incentives are only identified for the HTGR-R with core outlet temperatures greater than 850°C, the design and development program duration and cost and the demonstration path for the HTGR-R must be reassessed. The FY 1981 HTGR Program addresses these issues. In addition to reforming, a potential for the application of high-temperature direct heat to synthetic fuels processes was identified during the course of the FY 1980 investigation. This potential deserves further exploration and will also be included within the scope of the FY 1981 HTGR Program effort.

3.0 SUMMARY TECHNICAL DESCRIPTION

The plant selected for consideration as the HTGR-R Lead Project was identified as an 1170-MW(t) indirect cycle plant with a core outlet temperature of 850°C. The core thermal rating was selected because it was perceived to be within the projected commercial size range and for commonality with the 1170-MW(t) Steam Cycle/Cogeneration (HTGR-SC/C) concept. The indirect cycle heat source and flow diagrams are shown in Figs. 2 and 3. As in other HTGRs, the HTGR-R has its entire primary coolant system contained in a multicavity PCRV which provides the necessary biological shielding in addition to fulfilling the pressure containment function. The multicavity design allows each component (e.g., helium circulators and intermediate heat exchangers) to be located in a separate cavity which facilitates its removal and replacement. The reactor core is cooled by helium, has ceramic-coated fuel particles containing uranium and thorium, and employs graphite as the moderator. Thermal energy is removed from the reactor core by independent primary/secondary helium systems and is supplied to separate process loops. Three independent shutdown loops have been provided, each with the capability of full-core decay heat removal.

The primary helium is heated by the reactor core and transfers its heat to the process plant via the secondary helium loops. Each primary loop includes one intermediate heat exchanger (IHX), an electric motor driven primary helium circulator, and related instrumentation and controls. The primary coolant flows downward through the reactor core, where it is heated from 427°C (800°F) to 850°C (1562°F). The primary helium flows upward through the shell side of the IHX, counter-currently transferring heat to the secondary helium. The secondary helium system transports thermal energy from the IHX to the process plant. The secondary helium loops each consist of a reformer, a steam generator, an electric motor driven secondary helium circulator, and related piping, valves, and instrumentation. The helium thermal energy is split between the reformer and steam generator at a ratio of

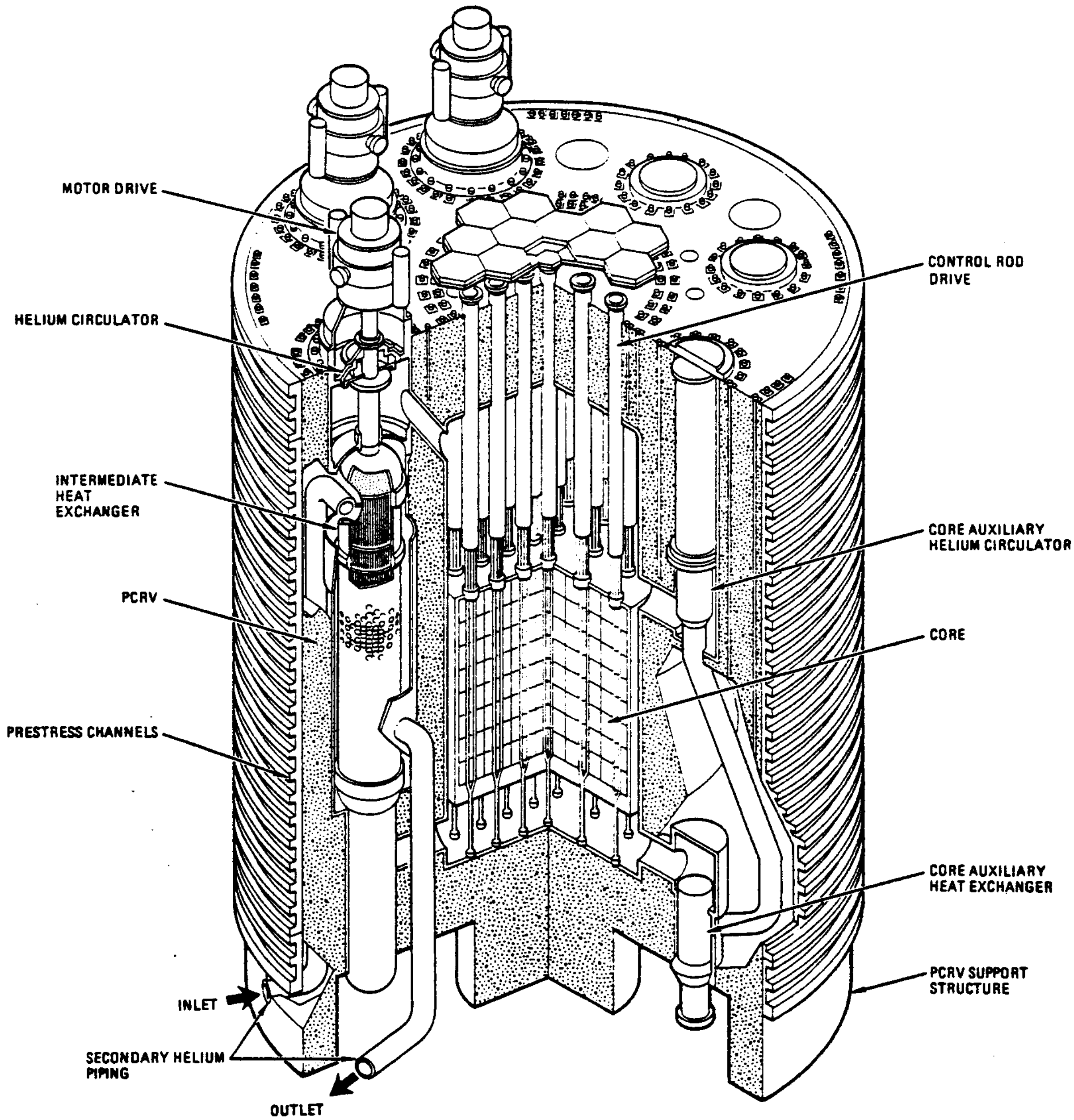


Figure 2 Isometric View of High-Temperature Nuclear Heat Source

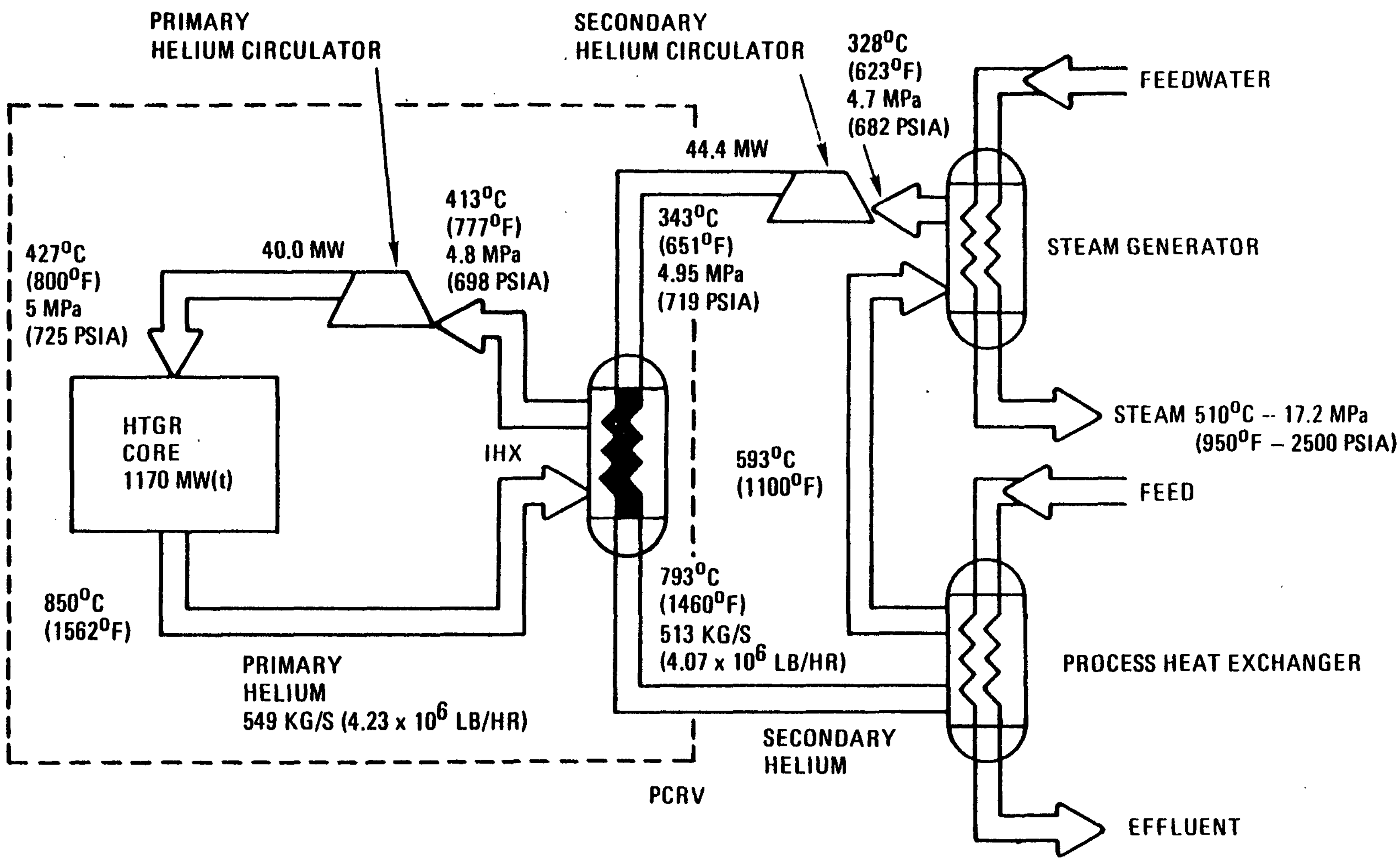


Figure 3 HTGR-R Flow Diagram

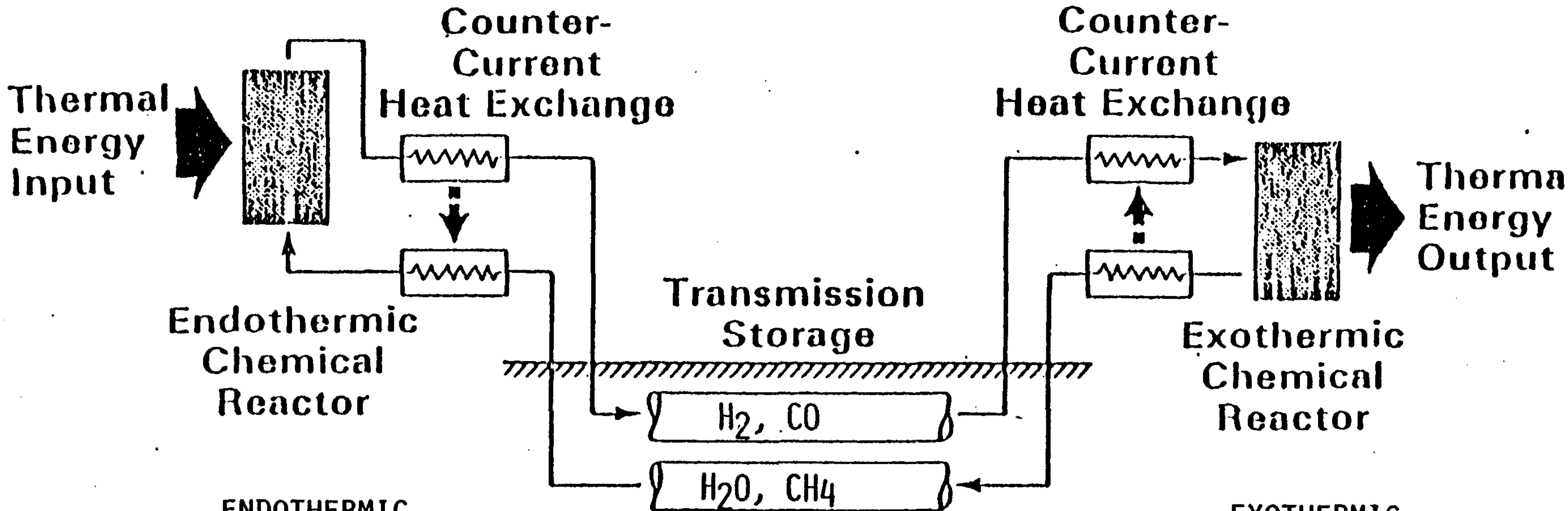
42% and 58%, respectively. The HTGR-R utilizes its high temperature capability to reform a mixture of steam and methane (H_2O and CH_4) in the presence of a catalyst to form hydrogen and carbon monoxide (H_2 and CO). The heat absorbed in this endothermic chemical reaction is supplied by the HTGR. For practical conversion efficiencies, peak reforming temperatures above approximately 705°C/1300°F are required. The steam generator provides steam for electrical power generation, for process plant needs, and for export.

In the Lead Project studied, emphasis has been placed upon energy storage/transmission and subsequent recovery of reactor heat via methanation. This application of the HTGR-R has been referred to as the Thermochemical Pipeline (TCP) and is illustrated in Fig. 4. The reformer effluent gases are cooled by counter-current heat exchange with the reformer inlet gases. Any excess steam is condensed and the effluent gases are compressed for transmission through a pipeline to a user site. At the user site, the hydrogen and carbon monoxide combine in the presence of a catalyst in an exothermic reaction. The high-grade heat released in the methanator delivers 480°C/900°F steam at the user site. The methanator effluent is cooled in a recuperative heat exchanger again preheating the inlet gases. The methane and water are then transmitted back to the reformer via two pipelines, completing the cycle. The pipeline itself may serve as a gas storage mechanism for short time periods and additional storage may be provided using underground caverns. On-site methanation may be utilized for utility load-following applications or remote methanation may be used for remote industrial process heat and cogeneration or for repowering of existing oil and gas-fired electrical power plants. In the above applications, the nuclear heat source is base loaded while the thermochemical transmission/storage system is load following.

An alternative energy transport system coupled with a 750°C/1380°F HTGR heat source was also examined briefly for comparison. This latter plant utilized a plant configuration almost identical to the HTGR-R and a sensible energy transmission and storage concept based upon a molten salt consisting of sodium nitrate, potassium nitrate, and sodium nitrite.

4.0 SUMMARY OF EVALUATIONS

Although the status of HTGR-R plant design is the least advanced of the commercial HTGR concepts under consideration, many conclusions, observations, and recommendations can be made which form a basis for future HTGR Program tasks on the Reformer option. It is obvious that additional work is required to properly assess the HTGR-R concept as well as related high-temperature concepts providing direct nuclear-generated heat via the secondary helium system. This future work will be accomplished as part of the continuation of the HTGR-Reformer Application Study in FY 1981.



$$^*CH_4 + H_2O\ (L) \xrightarrow{\text{ENDOTHERMIC}} CO + 3H_2$$

$$\Delta H^0_{298} = 250 \text{ KJ/MOLE}$$

$$CO + 3H_2 \xrightarrow{\text{EXOTHERMIC}} H_2O + CH_4$$

$$\Delta H^0_{298} = -250 \text{ KJ/MOLE}$$

*PRINICIPAL REACTION

Figure 4 Thermochemical Pipeline (TCP) Concept

4.1 Market Potential

Based upon market assessments conducted through FY 1980, a very large market is projected for industrial process heat, synthetic fuel, and chemical feedstock applications which could be served by the HTGR-R. At present, a large portion of U.S. energy requirements is beyond the scope of applications served by current nuclear technologies. The capability of the HTGR-R to expand the use of nuclear power beyond its current use for electric power generation and to efficiently utilize the uranium/thorium fuel cycle brings forth the potential for nuclear energy to replace a broader spectrum of fossil fuel uses in the U.S. In addition the HTGR-R is expected to have substantial benefits relative to coal, its anticipated competitor in these energy sectors, in terms of impact on air quality (including acid rain and the greenhouse effect), mining and transportation requirements, and solid waste disposal.

However, several institutional and technical barriers must be overcome before commercial deployment becomes a reality. The ownership and operation of commercial nuclear power plants in the United States has historically been vested in the electric utilities. If the HTGR-R is to be applied to industrial markets, utilities must have the incentive and capability to enter these markets or other entities must become involved. If the former, the expanded utility energy role must also be recognized and supported by regulatory bodies in the form of special rate consideration for cogeneration or process heat facilities and acknowledgment of capacity addition requirements. The relative timing of nuclear and industrial facilities also presents a difficult hurdle. The mismatch between long-term (10-12 year) nuclear project lead times with relatively short-term (3-5 year) industrial projects must be overcome. Finally, the reliability of the HTGR-R system must be adequately demonstrated before it can be coupled to a dedicated industrial process. Early plant designs must have the potential to achieve high plant availability in order to justify consideration by industry as a realistic energy alternative.

As noted previously, the commercial version of the HTGR-R Lead Plant exhibited disappointing economic performance relative to competing power generation alternatives in the time frame (1995 startup) studied. However, the potential for economic performance beyond 850°C commercial plants still remains. The full commercial potential must be investigated further through the conduct of design trade-off studies, plant optimization studies, and examination of alternative applications such as hydrogen production for synthetic fuel plants. Improvements in the nuclear plant capital cost and performance are expected. There also is the prospect for large improvements in the capital cost and performance of the thermochemical pipeline concept. These studies and the investigation of designs for alternative applications will be initiated to determine the commercial potential for the HTGR-R and to formulate the basis for the HTGR-R Design and Development Program.

In addition to the potential improvement of HTGR-R concepts, variants of the indirect cycle nuclear heat source to provide direct heat via the secondary helium system deserve further attention. Application of such systems to synthetic fuels (e.g., oil shale development and catalytic gasification processes) is worthy of further investigation.

4.2 Design and Development Requirements and Cost

The development of the HTGR-R represents a further extension of the technology required for HTGR-SC/C plants. Schedules showing design and development activities through completion of the Lead Project were derived in the course of this study. The startup date for the HTGR-R lead plant is projected for 1998. To support this date, an aggressive near-term program will be required to establish the lead plant configuration and host utility/user.

The Design and Development Program required to support the HTGR-R Lead Project amounts to approximately $565 million (1980 $). Therefore, a relatively large program is needed to develop the HTGR-R that will require a long-term government commitment if the concept is to be brought to commercial fruition. In addition, this 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. If for some reason any of these programs are not successful, an extension of the schedule and additional development dollars would be required. Further, if higher core outlet temperatures are required for satisfactory commercial performance and incentives, then a more lengthy and costly development program will evolve.

4.3 Conclusions

The projected economic performance, schedule, and deployment cost for the HTGR-R pose issues of reservation for the consideration of the HTGR-R as a lead project. However, the large potential market, fossil resource conservation, and environmental advantages of the HTGR-R system provide incentives for continued examination of the HTGR-R. The nature of the technical issues confronting this plant and its large deployment cost would indicate that the HTGR-R might be better considered as a follow-on plant to the HTGR-SC/C. This aproach may delay the entry of the commercial HTGR-R somewhat but would provide a more conservative and cost-effective path for the HTGR Program.

This decision cannot be made conclusively until the HTGR-R lead plant design basis is examined in more detail both to better establish its commercial potential and to shape the required design and development program. The following activities should be included in the future scope of the HTGR Program:

- Consideration of potential HTGR-R lead plant performance and cost improvements emphasizing improvements in the reformer, TCP, and methanation plants (energy delivery system). This area currently includes nearly one half of the total lead plant cost.

- Examination of alternative configurations (direct cycle), ranges of core outlet temperatures, and basic performance and cost improvements in the nuclear heat source/balance of reactor plant.

- Evaluation of alternative applications to include the production of synfuels or other direct coupled process heat applications.

- Additional characterization of the potential market for the HTGR-R. The market and the technical/institutional barriers to HTGR-R penetration must be better defined to firmly establish the commercial/national incentives for the HTGR-R system.

1.0 INTRODUCTION

The High-Temperature Gas-Cooled Reactor (HTGR) has the potential to assume a unique role for nuclear energy in an area which has traditionally been the province of fossil fuel applications. The process heat market has long been acknowledged but heretofore has never been seriously considered as a basis for near-term HTGR deployment. Several recent studies, however, have indicated that economic incentives may exist for the HTGR in reforming applications with core outlet temperatures of 850°C (1562°F) or less. Hence, the HTGR Lead Project Identification Plan, which defined the scope of these current studies, included consideration of the HTGR-Reformer (HTGR-R) as an option for Lead Project consideration.

In the HTGR-R concept, a portion of the reactor thermal energy is converted to a storable/transportable energy form through the use of a highly endothermic, reversible chemical reaction. (The balance of the reactor thermal energy is used for baseload electricity through the conventional steam cycle.) It is this distinguishing feature--the capability of storing and transporting reactor energy--which offers the potential for widespread displacement of fossil fuels (notably gas and oil) by nuclear energy in utility and industrial applications. Since peak reforming temperatures in excess of about 705°C (1300°F) are required for suitable conversion efficiencies, the HTGR is uniquely capable of supplying these requirements with nuclear energy. The HTGR-R is also anticipated to ultimately provide the nuclear heat source alternative to fossil energy systems in synthetic fuel production, steelmaking, and hydrogen production, thus providing substantial national incentives for development.

1.1 Purpose

The purpose of the HTGR-Reformer Application Study is to evaluate the suitability of the HTGR-R as a potential near-term Lead Project and to determine the merits and appropriateness of an HTGR Program strategy based upon such a Lead Project. The Lead Project/Program path selected must also obtain the concurrence of all Program participants to ensure a focused rather than diverse design and development effort. The evaluation process must include consideration of the perceived economic and sociopolitical benefits, development costs, status of plant design, anticipated deployment schedule, and user/operator requirements. This document will address these factors as well as identify the potential HTGR-R market size and the barriers facing its deployment. Since the HTGR is envisioned to be utilized for conventional electric generation and central cogeneration as well, the relationship between the HTGR-R and the alternative HTGR systems--HTGR-Gas Turbine (HTGR-GT) and HTGR-Steam Cycle (HTGR-SC)--will also be considered. The Application Study will also provide the basis for definition of the HTGR design and development program if the HTGR-R should be selected as the Lead Project.

1.2 Scope

The HTGR-R Application Study places primary emphasis on the HTGR-R Lead Project and the envisioned near-term series of commercial plants. As the goal of the HTGR Program is to develop the HTGR for eventual commercial deployment, the initial commercial market and program development costs required to reach commercial status are identified. Potential HTGR suppliers have delineated what development and demonstration steps are to be accomplished prior to their expected entry into the commercial HTGR market. These development costs must be offset by economic, environmental, and sociopolitical benefits to the nation, which are also assessed in this document. The value of the Lead Project will be determined as to its capability to demonstrate commercial plant technologies and to establish licensing precedent. Costs and schedules for both the Project and the Program have been developed and are included in the Application Study.

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 generation of high-temperature process heat for industrial and synfuels applications (as described herein) and coupling with the Brayton (Gas Turbine) Cycle.

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.

Process heat options of the HTGR 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 resources 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, which are the principal topic of this report. 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.

The 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 steam 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.

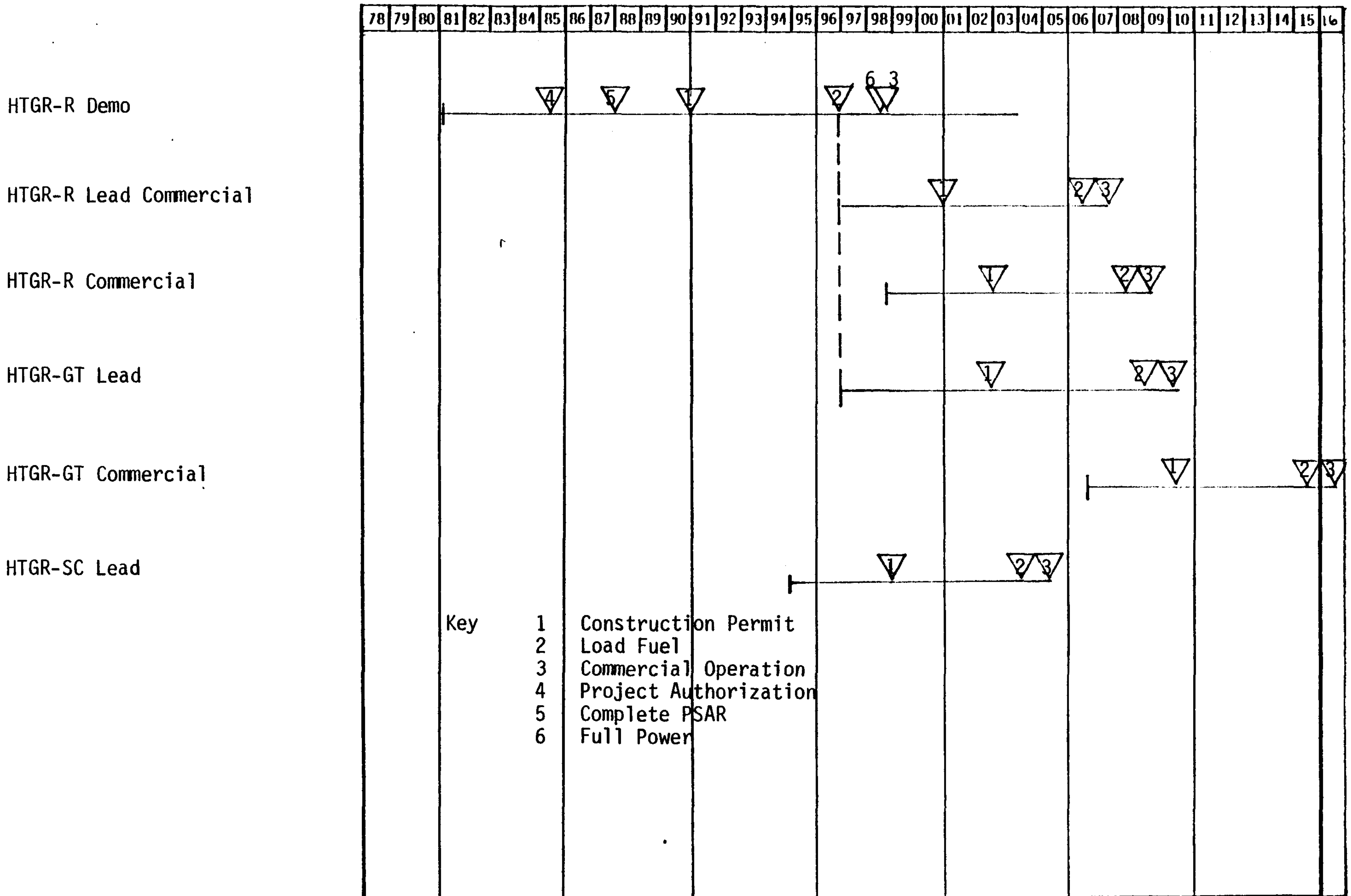
2.2 Program Scenario Based Upon HTGR-R Lead Project

The HTGR-R Lead Project provides the most direct path to deployment of commercial HTGR Reforming plants in the U.S. The Lead Project is envisioned to demonstrate most, if not all, of the commercial plant components, leaving evolution to higher core outlet temperature (>850°C) for later demonstration. The HTGR-R also provides significant development support to the other HTGR options--the HTGR-SC/C and the HTGR-GT. The long-term program schedule based on an HTGR-R demonstration plant is depicted in Figure 2.2-1. The HTGR-SC/C development requirements are, for all practical purposes, enveloped by a design and development program for the HTGR-R. Successful demonstration of the HTGR-R Lead Project could easily lead to a commercial offering of the HTGR-SC/C by one or more U.S. suppliers. In fact, the HTGR-R design, development, and licensing program might lead to commercial HTGR-SC/C offerings prior to demonstration, should the HTGR-SC/C technical issues be resolved and the nuclear power market be revived. The HTGR-R also does much towards the development of the HTGR-GT. With the exception of the turbomachine, most of the plant components are demonstrated either totally or in concept by the HTGR-R Lead Project. As the purpose of the program is to lead to commercial deployment of one or more HTGR plant options, it is important to define the programmatic relationship between the Lead Project and the various HTGR Program elements. These relationships are discussed in greater detail in the following sections.

2.2.1 Relationship to Initial Commercial Plant

The HTGR-R demonstration plant has been shaped to confirm the currently envisioned commercial plant component and system technologies wherever possible. Under the assumption that a commercial market does evolve for an indirect cycle 850°C core outlet temperature HTGR-R, the commercial plant would likely replicate the identified demonstration plant from the nuclear heat source to the reformers and steam generators. If

Figure 2.2-1 Long-Term HTGR Program Schedule with HTGR-R Lead Project



the commercial market cannot be justified until the HTGR-R design achieves a 950°C core outlet temperature and/or the direct cycle configuration, the identified demonstration plant provides a large step towards commercialization. A direct step to 950°C would require significant development in the area of materials technology and could require many additional years to resolve.

2.2.2 Relationship to Advanced Process Heat Systems

In parallel with the initial series of commercial plants, the base technology and advanced system design programs would continue towards definition of higher temperature design and applications. As the majority of the plant design effort for advanced systems and the commercial plant would be similar, development would be required only for the components directly affected by the higher temperatures and/or new applications. Once the materials are selected and the design is in hand, cost estimates can be made to confirm the assumed cost improvements associated with higher core outlet temperatures and advanced processes such as coal gasification or thermochemical water splitting. Depending on the timing, the advanced system design and licensing effort could start within a few years after commencement of successful operation of the HTGR-R demonstration plant. The advanced process heat plant deployment would depend upon a successful development program, the level of HTGR-R commercialization at 850°C, and the relative priority of developing and deploying the other HTGR options--the HTGR-GT and the HTGR-SC/C.

2.2.3 Relationship to Other HTGR Options

Development of the HTGR-R at the 850°C core outlet temperature leads to the development of the HTGR-GT and HTGR-SC/C. The HTGR-SC/C component and system designs are enveloped by the HTGR-R designs and can easily be replicated or extrapolated for a commercial HTGR-SC/C design. The 850°C HTGR-GT would require the additional development of the gas turbine. Most other component and system designs can be replicated or extrapolated from the HTGR-R. The successful gas turbine development program and resolution of the licensing and design issues surrounding turbine rotor failure can result in deployment of the HTGR-GT lead commercial plant after operation of the initial HTGR-R commercial plant. The timing of the HTGR-GT deployment will be dependent on the relative priority of advanced process heat systems, the HTGR-GT, and the level of effort for the HTGR-R commercial plant.

3.0 COMMERCIAL PLANTS

The ultimate goal of the HTGR Program is the commercial deployment of HTGR systems. The commercial market will develop only if there is a vendor for HTGR technology and one or more buyers to consummate purchases at a mutually agreeable price. As such, it is very important for the potential HTGR suppliers to define their objectives and posture toward commercialization. The objectives should be in agreement with potential users of the technology such as the electric and gas utility industry. The following sections provide the perspectives of two potential suppliers of HTGR technology that are current participants in the HTGR Program.

3.1 General Atomic Perspective

3.1.1 Commercial Plant Definition

The commercial plant is anticipated to have the following major features:

1. 1000 to 1500 MW(t).
2. 850°C core outlet temperature.
3. Intermediate helium loop.
4. Prismatic core.
5. Four loops.

The intermediate heat exchangers (IHXs) would be in side cavities surrounding the core with the helium circulators on top of each IHX. The core auxiliary cooling system (CACS) would be either water- or helium-cooled. The CACS units would be in the PCRV or, for greater diversity, one or more might use the IHX and a portion of the intermediate helium loop with an external helium-to-air exchanger.

An isometric view of the PCRV is shown in Fig. 3.1.1-1. The major heat and mass flow parameters are shown in Fig. 3.1.1-2.

3.1.2 Commercial Reforming Applications

3.1.2.1 Products

Early commercial plants are expected to take maximum advantage of current technology and to find application in the production of synthetic fuels from coal and oil shale that can be used to take the place of crude oil and natural gas. Since the technology of synthetic fuel production is itself under intensive development, early commercial applications are likely to be in areas where chemical processes have been or soon will be demonstrated. This approach will lower the investment risk. Processes and products in this category include above-ground oil shale retorting and kerogen upgrading where hydrogen requirements are relatively high. Oil production from shale as high as 8 million barrels/day by the year 2010 has recently been forecast by

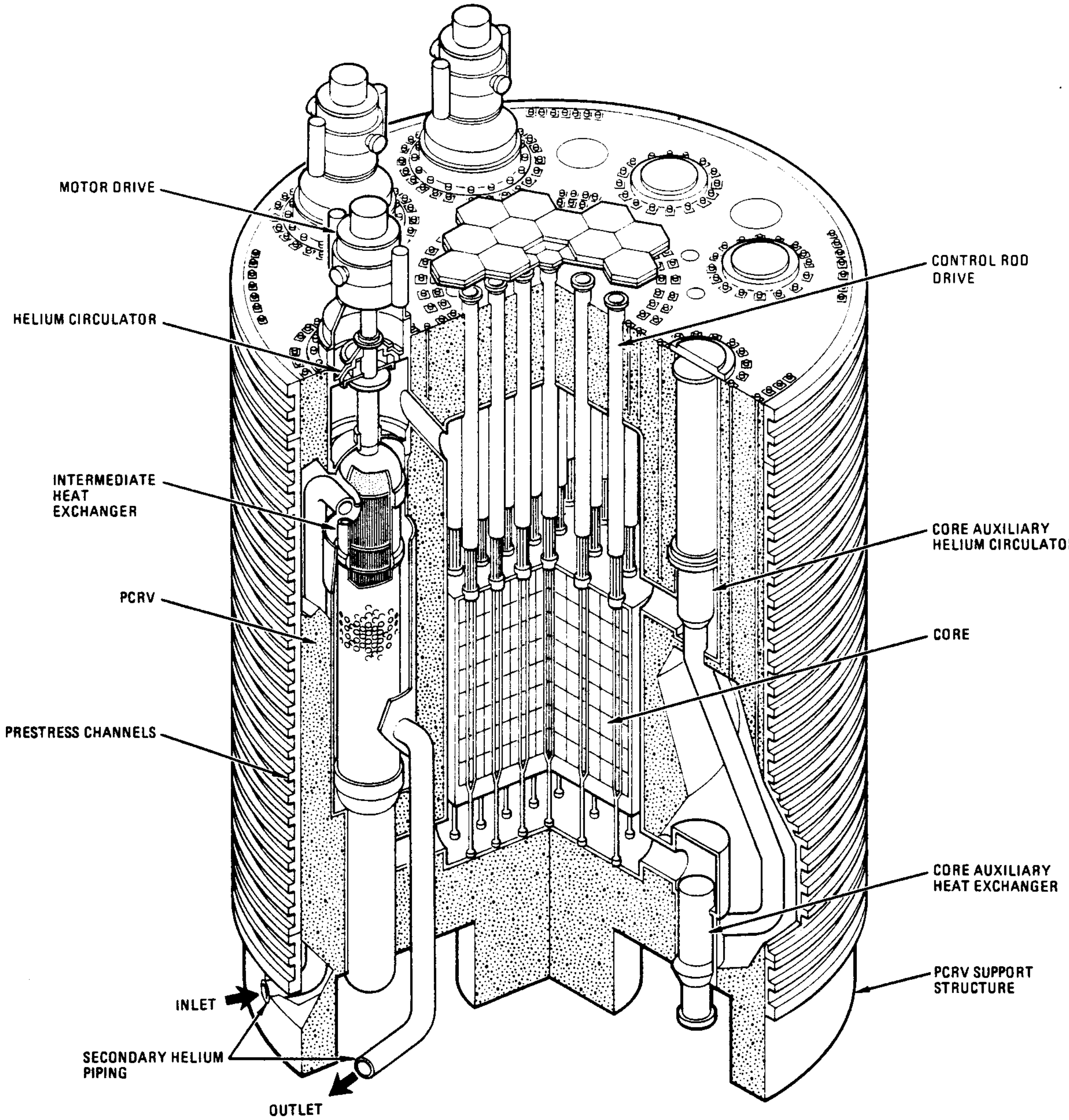


Figure 3.1.1-1 Isometric view of PCRV

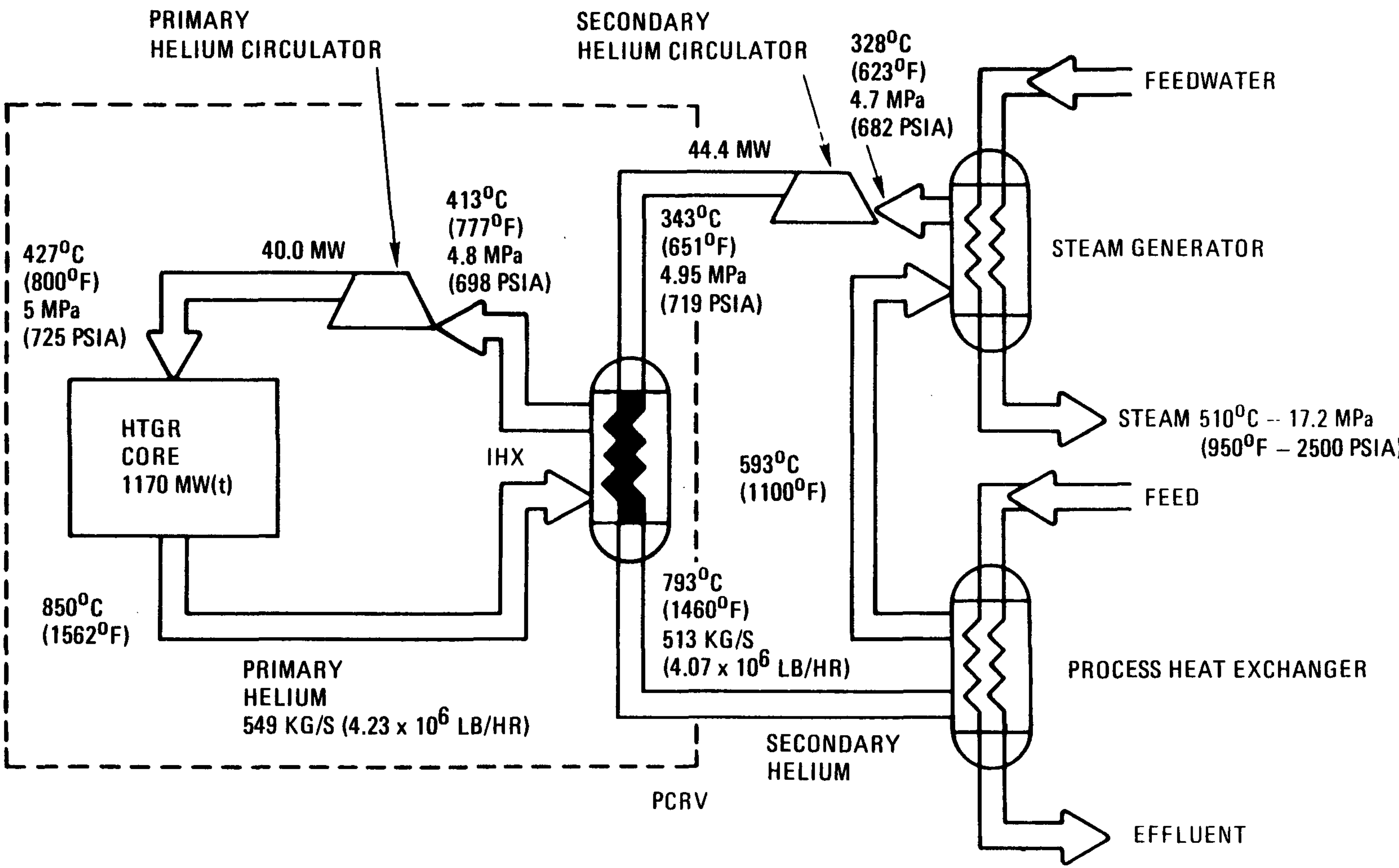


Figure 3.1.1-2 HTGR-R commercial plant system flow diagram

Exxon,* although a more realistic estimate would be 2 to 3 million barrels/day.

Production of liquid products from coal requires extensive quantities of hydrogen, steam, direct heat, and electricity. These processes are currently in the pilot plant stage but are expected to move into the demonstration plant stage soon, with operable plants by 1985 producing up to 20,000 barrels/day. Three processes, SRC-II (Gulf), Donor Solvent (Exxon), and H-Coal (Hydrocarbon Research), are currently moving toward demonstration plants.

Synthesis gas (H_2 + CO) is the raw material for a number of commercially important end products, e.g., methanol and ammonia. The basic coal liquefaction processes can be used to produce an intermediate material. This material can be upgraded by solution hydrogasification to form a light hydrocarbon as reformer feedstock material. Reforming will then produce a synthesis gas.

3.1.2.2 Commercial Market Time Frame

Development of synthetic fuels through the year 2000 will be driven to a large extent by government incentive programs. After that point, synthetics production will be developed to supply the increment between conventional supplies of energy and the U.S. demand. This demand should increase very rapidly after the year 2000, reaching an estimated 28% of the total oil and gas requirements by the year 2020. Coal liquids may command 55% of the synthetic fuels market and oil from shale 25%. The commercial market time frame is most likely limited only by a more competitive method of producing synthetic fuel, e.g., fusion, rather than any lack of demand for the product. For this reason, the development of the reactor plant should continue in order to improve the competitive aspects of the HTGR.

3.1.3. Description of Follow-On Process Heat Objectives

The plant with a 850°C core outlet temperature and a secondary helium loop can be upgraded to 950°C core outlet temperature at a later time and to possibly even higher temperatures in a still later time frame. The prime objective in raising this temperature would be to expand the processes for producing hydrogen. At process temperatures commensurate with a 950° core outlet temperature, the steam-carbon reaction has much better kinetics with a large cross section of U.S. coals. This method directly competes with, and can substitute for, the more conventional partial oxidation process and has the added advantages of (1) not requiring a large oxygen plant and (2) not contaminating the product with numerous oxides that must be separated from the product. Cost comparisons with conventional partial oxidation processes are under way.

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*Los Angeles Times, June 11, 1980.

A second use of the higher temperature available to the process is for thermochemical processes to produce H_2 from water. Major development today is on cycles using sulphuric acid and a halogen such as bromine or iodine. The high-temperature endothermic step is the breakdown of the sulphuric acid. This step requires temperatures of 850°C or higher to avoid large amounts of costly recycle. It is estimated that product cost may be decreased approximately 20% if the reactor temperature can be raised from 850° to 950°C.

A diagram showing progressive development of the HTGR-R is shown in Fig. 3.1.3-1.

3.1.4 Product Economics

Tentative economics for the application of the HTGR to the production of light liquid hydrocarbons, e.g., diesel oil and gasoline, have been prepared and compared with a similar process plant using fossil fuel for the energy requirements. An increase in plant efficiency from 59% to 67% is calculated, primarily because stack losses are eliminated and the relatively inefficient partial oxidation process is reduced in size. The cost of the final product (a light hydrocarbon) is estimated to be reduced by about 15% when an HTGR-R is used as the heat source. A comparison is shown in Table 3.1.4-1. Coal savings of 36% are also realized. A more detailed technical study of this application is currently under way with results expected by the end of FY 1980. In addition, comparative economics for the oil shale application are also under way.

3.1.5 Demonstration/Deployment Strategy

The commercial plant described here is essentially the same as the demonstration/lead plant. It includes an intermediate helium loop with a steam methane reformer. The steam electric plant will provide in-plant needs, and it is unlikely that net electric power will be produced unless the customer so desires.

This plant design offers great flexibility since the process heat exchangers are external to the PCRV and containment. The thermal balance between the reformer and steam generator can be changed. Any advances in reformer design or process technology can be easily accommodated, and other types of process exchangers can be substituted for the reformer. Any of these changes can be accommodated without modifying the nuclear plant; hence, an alternate strategy to commerical plants would not be necessary.

3.1.5.1 Assessment of Risks and Costs

With the concept of the commercial plant virtually identical to that for the demonstration/lead plant, the technical risk is greatly reduced in regard to the nuclear plant. The process plant will have been demonstrated at full size in a non-nuclear facility prior to the commercial plant. The new technology is the marriage between the coal liquids plant and the nuclear plant. Assuming the demonstration/lead

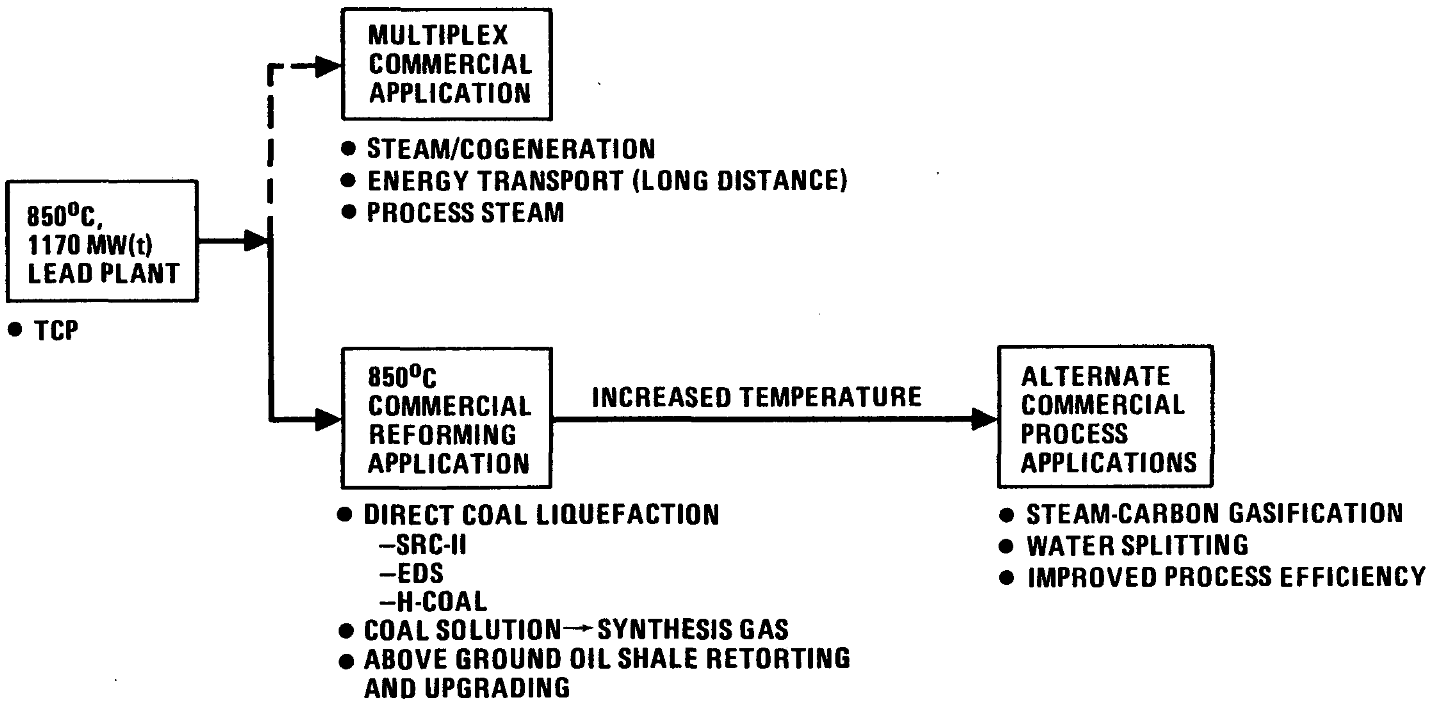


Figure 3.1.3-1 Progression from the HTGR-R lead plant to commercial application

TABLE 3.1.4-1
COMPARISON OF NUCLEAR AND NON-NUCLEAR COAL LIQUEFACTION PROCESSES

	CONVENTIONAL	NUCLEAR
PROCESS	SRC-II	SRC-II NUCLEAR REFORMING
COAL FEED, TONS/DAY	32,210	21,700
NUCLEAR HEAT SOURCE		
REFORMING,[a] MW(t)	--	905
STEAM, MW(t)	--	1155
PRODUCT OUTPUT		
BBL/DAY	90,000	90,000
TONS/YR	4.4×10^6	4.4×10^6
THERMAL EFFICIENCY, %	59	67
PRODUCT/COAL RATIO, BBL/TON	2.8	4.2
HEAT IN PRODUCT/ HEAT IN COAL	0.59	0.95

(a) INCLUDES STEAM PRODUCTION FOR REFORMER.

plant produced steam, electricity, and reformed gases, there is little new equipment - basically only conventional CO_2 stripping and cryogenic H_2-CH_4 separation equipment. Plant startup and control can be fairly well separated by adequate storage of H_2 and/or the use of startup fossil-fired reformers.

In the same manner, the cost risk is minimized since both nuclear and fossil plants have been built to full scale. Cost risk may relate to improvements developed since the demonstration plant, but these risks should be viewed in terms of their anticipated savings.

3.1.5.2 Lead Plant Importance

The lead project is vitally important to this commercial plant since it truly demonstrates the concept and the hardware. The commercial plant will gain advantages of the learning curve in terms of component design and fabrication cost. It may be able to use materials tested in the lead plant and made commercially available at a later date. Licensing and safety issues can be resolved in the lead plant and carried directly through to the commercial plants.

3.2 General Electric Perspective

3.2.1 Commercial Reformer Plant Definition

3.2.1.1 Configuration

In summary, the ultimate nuclear reformer plant configuration is currently expected to be:

• HTGR power	-	1000 MW(t) to 1500 MW(t)
• Configuration	-	Direct cycle (reformer and steam generator within PCRV)
	-	3, 4, or 6 main heat transport loops
• HTGR core	-	Pebble bed or prismatic core
	-	950°C core outlet helium temperature
• Reformer	-	Duplex tube
• Operation	-	Baseloaded at 80% to 95% capacity factor
	-	Part of a multiplant pipeline network

As discussed in Section 3.2.2, this nuclear reformer plant could be coupled with a variety of end-use facilities:

- Thermochemical energy pipeline (TCP) to supply heat for dispersed users.

- Syngas or hydrogen supply for synthetic fuel or other process feedstock, using a light hydrocarbon feed to the reformer (e.g., from coal conversion or shale oil processing).

The following discussion on plant configuration assumes that the first commercial nuclear reformer plant application is the TCP concept.

The reformer plant, which produces principally transportable chemical energy through a thermochemical pipeline (TCP) and also some baseload electricity, is expected to be the first commercial process heat application of the High-Temperature Gas-Cooled Reactor (HTGR). The HTGR-Reformer/TCP serves the industrial heat market as well as the peaking and mid-range electric power market; it is also well suited to provide heat and hydrogen for oil shale processing and other synfuel processes.

In the HTGR-Reformer/TCP, thermal energy of the nuclear reactor is converted to chemical energy through steam reforming of methane. A mixture of steam and methane is heated by the reactor coolant in a steam reformer and partially converted to hydrogen, carbon monoxide, and carbon dioxide. The steam is condensed from the reformed gases, and the resulting mixture is transported at ambient temperature through a pipeline to dispersed users up to 480 km (300 mi) from the reactor plant. The users extract the chemical energy in methanators by converting the hydrogen, carbon monoxide, and carbon dioxide into methane and water to produce steam at temperatures up to 595°C (1100°F), which can be used directly in industrial processes and in the cogeneration of electricity. The methane and water produced by the methanator are then returned in separate pipelines to the multiplex plant. Alternatively, the methane and water could be consumed by the dispersed user if a hydrocarbon source (e.g., coal or methane) and water are available at the multiplex plant. Gas storage (in underground mined formations, e.g., salt caverns), along with pipeline packing, provides capability to serve one- and two-shift users and peaking electricity generation, while the central plant operates continuously.

Helium temperatures below 593°C (1100°F) cannot be utilized in the steam reformer, so a steam generator is placed downstream of the steam reformer to utilize the energy in the lower helium temperature range. The steam produced in the steam generator can be used to heat the process gas feed, and to produce electricity for the plant and for off-site distribution.

- Direct vs. Indirect Cycle and Reactor Outlet Temperature - There is continuing assessment on whether the steam reformers and steam generators should be in the primary circuit (the direct cycle) or in a secondary circuit (the indirect cycle). The FY 1981 work should be focused on a decision on cycle choice. In the direct cycle, the steam reformers and steam generators are heated by the primary helium and may be located within the prestressed concrete reactor vessel (PCRV). In the indirect cycle, the intermediate heat exchanger (IHX) is located within the PCRV and secondary helium is piped through the reactor containment building (RCB) wall to the reformer and steam generators. The technical problems of the IHX design and construction, the licensing problems, and the product costs of the indirect cycle plant must be compared with the technical and licensing problems and product cost of the direct cycle reforming plant. This comparison must be made in the light of ultimate commercial application requirements. Clearly, an IHX is needed for the long-term applications of the endothermic steam-carbon gasifier and for thermochemical water-splitting. GE views these U.S. applications to be so far in the distant future as to practically eliminate their influence on the lead plant configuration choice. An IHX may not be needed for the athermal catalytic coal gasifier (no external heat addition to the gasifier bed), which requires only heat addition to the gasifier feed streams. Development of the IHX may be a more formidable problem than development of a direct cycle (double-wall tube) reformer, and costs of the IHX and intermediate helium system may make the indirect cycle lead plant prohibitively expensive. Licensing the indirect cycle plant may not be any less difficult or less time-consuming than licensing the direct cycle plant. However, once the indirect cycle plant is licensed and built, licensing and building the direct cycle plant may present an insurmountable obstacle because of regulatory perceptions and because of the money and time spent on development of the indirect cycle plant. If the indirect cycle HTGR-Reformer is then found to be noncompetitive with alternative energy sources, the HTGR option may well be closed out. Thus, in GE's view, the selection of direct or indirect reforming cycle for the lead plant requires further and careful consideration.

The direct cycle has the advantage of being able to utilize the approximately 50°C higher temperature of the reactor coolant, which increases the methane conversion. The direct cycle also eliminates the cost and complexity of the intermediate heat exchangers and secondary loops. However, the direct cycle must bring the process gas within the containment. While the process gas itself is not flammable or explosive, it may become so upon mixing with oxygen under accident conditions. To minimize the potential for gas leakage and hydrogen/tritium diffusion between the process gas and reactor helium systems, the duplex-tube steam reformer provides a double-wall barrier between the process gas and the reactor helium. This duplex tube design provides a means to monitor for leakage of either the primary helium or the process gas into the gas gap of the duplex tube. The process gas piping within the RCB can be enclosed in an inert gas tunnel to prevent leakage into the reactor containment building. Further, the process gas can be operated at a slightly higher pressure than the reactor helium to prevent leakage of radioactive primary coolant into the process gas. Operation of the process gas at higher pressures than the reactor helium reduces the methane conversion rate since conversion is inversely proportional to reforming pressure. The ability to improve conversion by operating at reduced process gas pressures is a major potential performance advantage of the indirect cycle; a potential licensing advantage is the placement of the process gas system outside of the reactor containment. The major disadvantages of the indirect cycle are the reduction in peak reformer temperature due to temperature drop across the IHX, the increased IHX and secondary system costs, and the increased cost of the gas system caused by lower reformer gas temperature and pressure. It is believed that the direct cycle plant is licensable. The economic impact of resolving the licensing questions is uncertain for both the direct and indirect cycles. Therefore, the decision on direct/indirect cycle for the commercial plant should be based on performance and cost, which now appear to favor the direct cycle.

As a basis for comparison with estimates produced by future efforts, Table 3.2.1-1 shows GE's current perception of the relative capital costs for the direct and indirect cycle plants at 850°C and 950°C reactor outlet temperature. It is emphasized that the perceptions in Table 3.2.1-1, which show an incentive to develop a 950°C HTGR-Reformer direct cycle, are based on extremely rough estimates and do not include operating and maintenance costs. The uncertainty in these capital costs is believed large, and these values are set forth principally to form a basis for further investigation. The values in Table 3.2.1-1 do not include recent results from the HTGR-R lead plant study, since comparable results on the three other configurations are not available.

TABLE 3.2.1-1

HTGR-REFORMER PLANT CAPITAL COST IMPACT OF REACTOR TEMPERATURE AND CONFIGURATION CHANGES

PLANT CONFIGURATION		REFORMING GAS TEMP./PRESS	CH_4 CONVERS.	COST IMPACT (Excluding Pipeline & Users) GAS PLANT	REACTOR PLANT	BALANCE OF PLANT	TOTAL PLANT	THERMAL PERFORMANCE
DIRECT CYCLE	- 950°C	825°C/40 b.	0.7	0.25	0.4	0.35	1.0	Approx. 85%
INDIRECT CYCLE	- 950°C	800°C/35 b.	0.7	0.25	0.65	0.35	1.25	Equivalent approx. 85%
INDIRECT CYCLE	- 850°C	700°C/15 b.	0.7	0.27	0.65	0.35	1.27	
DIRECT CYCLE	- 850°C	725°C/40 b.	0.3	0.37	0.42*	0.35	1.13	Poorer approx. 80%

*Increased cost over 950° plant caused by increased plant size for equal chemical energy output.

High reactor outlet temperatures are a goal for the HTGR with reformers due to the increased methane conversion rate, which reduces gas plant, pipeline, and methanator costs and may reduce reactor size and cost slightly. It is expected that a direct cycle reformer plant would have a 950°C (1740°F) reactor outlet temperature and operate with a 39 bar (580 psia) primary system pressure.

- Pebble vs. Prismatic Core - Although either a pebble bed or prismatic core can probably be used for this application, the pebble bed reactor may have greater potential for very high temperature operation and high availability; thus, the pebble bed reactor was tentatively selected by GE for the commercial plant configuration. It is recognized that the prismatic core is the current reference U.S. HTGR core design, and the pebble bed core is recommended as a backup to the prismatic design. The greatest difference reflected in the heat transport system is the reactor inlet temperature, which is potentially much lower in the pebble bed reactor (250°C/480°F) than in the prismatic reactor (425°C/800°F); this affects the design of the steam-electric system and changes the split in the power to the reformers and to the steam generators. The prismatic core has a higher primary helium flow rate and corresponding lower core temperature rise because of different core hot channel conditions.

  Because of basic design problems facing both cores, it is recommended that a pebble reactor design effort be continued as a backup to the prismatic design, with focus on the key design issues facing the pebble core. This design effort must be based upon the same design criteria for the two core types.

  Choice of pebble or prismatic core is subject to the future experience to be obtained from operation of Fort St. Vrain, AVR and THTR. The AVR operation at 950°C has given good confidence in the ability of the pebble core to reach this temperature. No comparable operating experience exists for the prismatic core. The decision for the final design selection should be deferred until further operating experience with both designs is available and the development required for 950°C operation is clearly defined.

- Plant Power - The reformer plant size depends on the particular market being served; however, it is expected that the ultimate commercial plant is one of several interconnected plants serving the same market. Many potential users require very high system availability which could not be attained by a single plant. A very preliminary assessment of the new and replacement industrial heat market added in the years 1995 to 2010 indicates that the requirements for a typical system which might be provided by the HTGR-Reformer/TCP would be 3000 MW(t) to 4000 MW(t). If at least four plants are needed in the system to maintain system availability, then each plant would provide 750 MW(t) to 1000 MW(t) of thermochemical energy to the pipeline. The reactor thermal rating would be 50% to 100% higher than this since the plant produces electric power to operate the plant and to distribute to the electric grid.

3.2.1.2 Major System Parameters

Preliminary design studies are being performed on a commercial pebble bed reactor reformer/TCP plant. This plant is a direct cycle 1500 MW(t) pebble bed reactor, with a 950°C (1740°F) helium outlet temperature and a 250°C (480°F) helium inlet temperature. There are six steam reformers and three steam generators in the primary loops; two reformers are connected to each steam generator. A schematic for the plant is shown in Fig. 3.2.1-1. There is a process gas loop for each steam reformer; the loops remain separate until they feed the pipeline. The primary helium supplies a total of 750 MW(t) of heat to the reformers, and the process gas provides 824 MW(t) of thermochemical energy to the pipeline; the difference in reformer power and the pipeline energy is provided by steam from the steam-electric system. The steam-electric system has a standard 163 bar/565°C (2400 psia/1050°F) high pressure turbine with 44 bar (650 psia) back pressure and a separate 2.7 bar (40 psia) double flow low pressure turbine with 5 cm (2 in.) of mercury condenser pressure. The turbines produce 179 MW(e) of electricity, 106 MW(e) of which would be available for the electric grid, after providing power for the helium circulators, turbine auxiliaries, pipeline compressors and steam for heating incoming methane and steam to the reformer.

Helium at 950°C (1740°F) and 39 bar (580 psi) flows from the outlet plenum below the core to the steam reformer through the inner duct of the coaxial hot duct. The helium then passes on the shell side of the shell-and-tube steam reformer, exits at the top of the reformer at 599°C (1110°F), and flows in the inner duct of the coaxial duct to the steam geneator. Each steam generator receives helium from two steam reformers. The steam generator is a once-through shell-and-tube type with helical tubes used for the economizer, evaporator, and superheater sections and a central downcomer for the superheater steam outlet. The helium flows downward on the shell side in the annulus between the helical section shell and the downcomer. At the bottom of the economizer section, the helium turns 180° and flows upward through an annulus (between the helical section shell and the outside shell of the steam generator) to the inlet of the primary circulator (located above the steam generator). The discharged helium from the circulator flows in the outer portion of the coaxial duct from the steam generator to the steam reformer; it then flows around the shell of the reformer and back to core in the outer portion of the coaxial duct. The helium temperature is about 250°C (480°F) when it enters the core cavity. The total helium flow rate is 403 kg per sec (3.2 million lb per hr).

The steam reformer, shown in Fig. 3.2.1-2, is a shell-and-tube heat exchanger using duplex reformer tubes. The steam-methane mixture [at

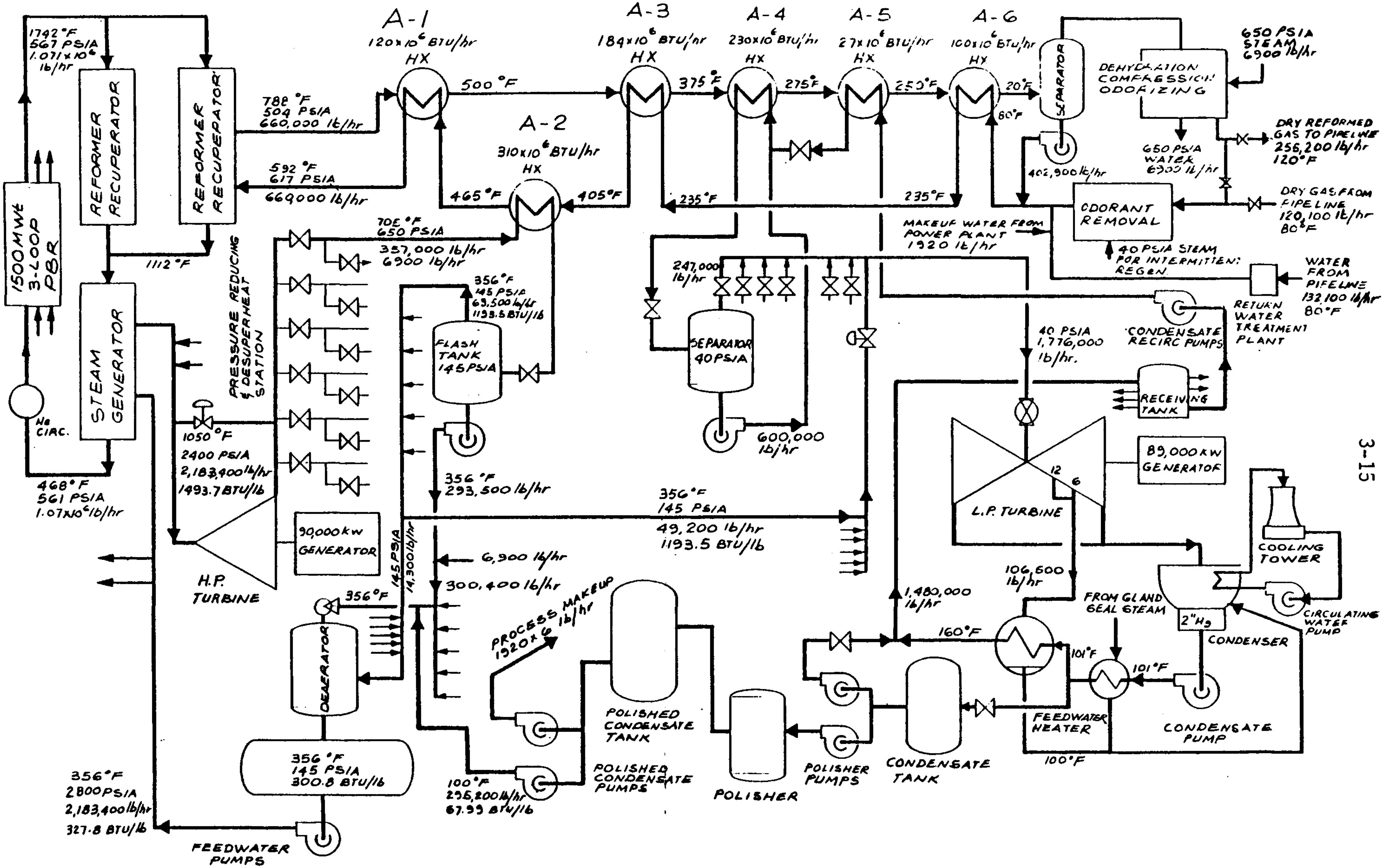


Figure 3.2.1-1 Heat and Mass Balance Diagram
(See 11" x 17" drawing in Appendix C)

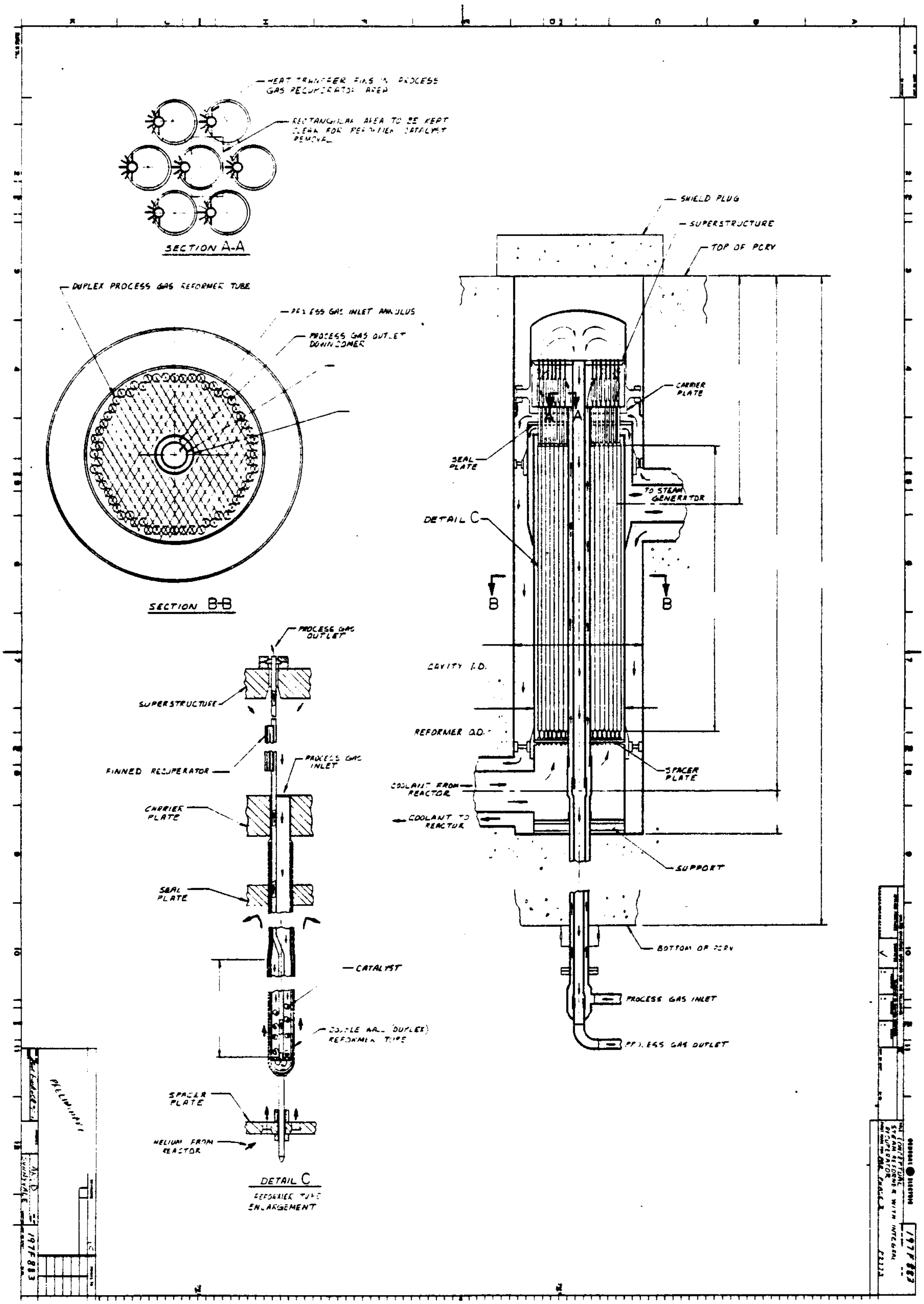


Figure 3.2.1-2 Duplex Tube Steam Reformer
(See 11" x 17" drawing in Appendix C)

310°C (590°F) and 42 bar (617 psia)] enters the reformer at the bottom, flows upward through a pipe in the center of the reformer and enters the recuperator tubes at the top. The duplex reformer tube has a double-wall outer tube with a central return tube. The outer tube has a small gap between the two walls, which is purged by helium. The annulus between the outer double-wall tube and the return tube is filled with a nickel-based catalyst in the form of Raschig rings. The steam-methane mixture flows through the catalyst bed and is heated by the reactor helium and the reformed gas in the return tube. The steam-methane mixture is partially reformed into hydrogen, carbon monoxide, and carbon dioxide as it flows through the catalyst bed. The reformed gas mixture enters the central return tube at the bottom of the duplex reformer tube, and flows to the top of the reformer, giving up heat to the incoming gas. The reformed gas then flows to the bottom of the reformer in an annulus around the central outlet pipe. The reformed gas is at 420°C (790°F) and 34 bar (500 psia) when it leaves the reformer. Besides being cooled by the incoming gas, the reformed gases also produce steam in a bottoming cycle for the steam-electric system. The excess steam is condensed from the reformed gas as it is cooled, and the water is recycled. The reformed gas is then dried, odorized, and compressed before entering the pipeline.

Methane and water are returned to the plant in separate pipelines at an ambient temperature of 27°C (80°F). The return water is mixed with the water condensed from the exiting process gas. The water is mixed with the methane and partially evaporated in a series of mixed-feed evaporators, which are heated by the exiting process gas. The remaining water is evaporated by mixed-feed evaporators, which are heated by the steam from the high pressure turbine. The steam-methane mixture is then heated by the reformed gas mixture to 310°C (590°F) before it enters the reformer.

Feedwater enters the bottom of the steam generator at 180°C (356°F), and flows into the helical tubes of the economizer/evaporator/superheater section. The steam from the helical bundle then flows downward in the central downcomer. The superheated steam exits at the bottom of the steam generator at 565°C (1050°F) and 163 bar (2400 psia).

The steam from the three steam generators is combined and passed through the high pressure turbine. The steam from the high pressure turbine [at 374°C (705°F) and 44 bar (650 psia)] is then piped to the mixed-feed evaporators in the six process gas loops, where heat is transferred from the reformed gas to the steam. The steam return lines from the mixed-feed evaporators feed a 10 bar (145 psia) flash tank, and the steam is extracted for use in the bottoming cycle. The water from the flash tank is mixed with the return water from the flash tank, put through a deaerator and pumped back to the steam generator.

Steam at 27 bar (40 psia) is produced in the bottoming cycle by heat from the reformed gas. Steam from the separator is mixed with throttled steam from the flash tank and expanded through the low pressure turbine to a pressure of 5 cm (2 in.) of mercury. Some of the water from the condenser is recirculated in the bottoming cycle, and the remainder is polished and mixed with the water from the flash tank.

3.2.2 Commercial Reforming Applications

3.2.2.1 Products

The HTGR-Reformer direct cycle plant produces the following basic products:

- Synthesis gas (H_2, CO, CO_2).
- Base load electricity.

When coupled with the TCP, gas storage and methanator plants, the synthesis gas can be used in a closed cycle to produce the following derived products:

- Process heat at 593°C (1100°F) and below for dispersed one-, two- and three-shift industries (including the production of heat for surface oil shale retorts and process steam) and cogenerated electricity.
- Load-following electricity.

When coupled with a source of water and methane (e.g., from catalytic coal gasifier or shale oil process), the synthesis gas (or obtained hydrogen) can be consumed in the production of a wide variety of derived products, including:

- Conversion of synthesis gas directly to motor fuel (indirect liquefaction of coal).
- Upgrading of shale oil to motor fuel through hydrogenation.
- Direct coal liquefaction and upgrading of the coal liquid to motor fuel through hydrogenation.
- Ammonia production from coal.
- Process heat and cogenerated electricity for dispersed users through methanation and combustion of resulting methane.

3.2.2.2 Anticipated Commercial Market Time Frame

The following factors affect the market time frame:

- Economic incentives and competing energy sources.
- Availability of end-use facilities.
- HTGR site availability remote from end-use.

- National security incentives to reduce dependence on imported oil.

- Availability of owner/user institutions.

Because all but the last of these factors appear to significantly favor the HTGR-Reformer/TCP for dispersed process heat and peaking electricity generation, it is believed that a market for the HTGR-TCP exists as soon as the plant is available. However, the lack of an institutional entity to own and operate such a novel energy system for process heat as the TCP presents a formidable obstacle. Providing load-following electricity clearly has the advantage of existing electric utility institutions. Economic and market projections for dispersed process heat and load-following electricity generation are discussed in Section 3.2.4.

Production of heat for dispersed oil shale surface retorts is a market for which an owner/user institution may be more readily developed than for general industrial process heat. However, economic projections have not been made and further work is needed in this area. The economic incentives will depend upon the plant capital cost, increased oil output (or reduced shale input), and reduced cost of imported electric power and other fossil fuel achieved through use of the HTGR-TCP. Section 3.2.4 presents an estimate of HTGR thermal power required for two million barrels/day of western shale oil, which may be the production rate from western shale by the year 2000-2010.

Production of motor fuel through indirect coal liquefication is currently a commercial process, and improvements in this process are under development. The catalytic coal gasifier is being developed on its own merits, independently of the HTGR Reformer. Production of motor fuel by direct liquefaction is in the development stage. Because (1) the price of coal is expected to be low when handled near the mine in the large quantities used in a coal conversion plant and (2) the HTGR-Reformer displaces only approximately 20% and 40% of the coal used in direct and indirect liquefaction, respectively, an economic incentive for the HTGR in coal conversion is anticipated in regions where the price of coal is much higher than the cost of nuclear fuel. However, detailed economic projections have not been made and further work is needed to better define the incentives for the HTGR Reformer. Section 3.2.4 presents estimates of the HTGR thermal power required for two million barrels/day of motor fuel through coal conversion processes.

3.2.3 Follow-on/Advanced Process Heat Application

The discussion in this section assumes that the U.S. Lead Plant is the Indirect Cycle Reformer plant with 850°C reactor outlet temperature coupled with a thermochemical pipeline system probably serving as an electrical load-following generation plant. It is possible that this plant configuration will not be economically competitive with alternate fossil energy sources for process heat applications in the 2000-2020 time frame. Thus, the follow-on plants aimed at achieving economic competitiveness, as well as displacing fossil fuels, are envisioned to be:

1. Direct cycle reformer, 950°C reactor outlet temperature for TCP application to displace fossil fuel use, first used possibly for oil shale processing. Since recovery of useful product from the oil shale may be enhanced by the use of hydrogen in the retorting process (particularly for eastern U.S. shales) the HTGR-Reformer plant might be used both with an open pipeline (hydrogen supply) and a closed pipeline (producing heat for the retorts by methanators). Ultimately, several HTGRs would be installed on a pipeline network, providing redundancy of supply to an oil shale processing and shale oil upgrading complex.

2. Direct cycle reformer, 950°C reactor outlet temperature for application to production of syngas via coal gasification. The catalytic gasifier appears to be favored for application with the HTGR because of several factors:

   a. Heat is added to the 700°C gasifier feed streams (coal, catalyst, steam, recycle H_2 and CO) rather than internally in the gasifier itself (as in the steam carbon gasifier by piping helium through the gasifier bed). In producing methane, the catalytic gasifier itself is athermal, i.e., no heat is added to or removed from the gasifier bed, unlike the hydrogasifier which is exothermal. For the production of syngas, the steam-carbon gasifier is endothermal. In order to heat the gasifier feed streams either (1) an IHX with an intermediate helium loop would be needed, (2) HTGR superheated steam could be used, (3) heat could be recovered from high temperature syngas, (4) methane or other fossil fuel could be burned, or (5) syngas could be methanated to provide the heat.

   b. The methane product of the catalytic gasifier is reformed to produce syngas using the HTGR-Reformer package developed for the TCP [see (a.) above].

   c. The catalytic gasifier can be developed independently of the HTGR-Reformer, then at an appropriate time the HTGR-Reformer package can be coupled with a gasifier plant to reduce the consumption of coal in the production of syngas.

   The syngas is a premium product which has many uses (Ref. 1), including the production of gasoline. Although the development of synfuel processes is at an early stage, indirect coal liquefaction to produce motor fuels may be favored over direct liquefaction because of the requirement to remove impurities from the direct liquefaction product in the process of upgrading syncrude to motor fuel. However, direct liquefaction appears to produce a satisfactory liquid coal for boiler fuel, and the HTGR can be applied to produce both steam and hydrogen (by reforming) for the direct liquefaction and syncrude upgrading process.

The initial gasification step in the indirect coal liquefaction process produces syngas, which is then converted, with the release of heat and water, to motor fuel. The indirect liquefaction process requires heat at high (950°C) temperature to produce the syngas via reforming, and releases heat at lower temperature in the exothermic step to produce motor fuel. Application of the HTGR-Reformer to the indirect liquefaction process reduces the required coal input by 30% to 40%. Since the direct liquefaction process is, in principle, more efficient in its use of coal, application of the HTGR reduces the coal input by only 15% to 20%. However, the problem of impurity removal from the direct liquefaction syncrude in processing to produce motor fuel may result in a total energy input equal to that for indirect liquefaction.

3. Indirect cycle thermochemical water splitting with reactor outlet temperature in the range 850°C to 1100°C. Lawrence Livermore National Laboratory (LLNL) is currently engaged in a three-year program to study the three leading candidate cycles for water-splitting, i.e., the GAC, Ispra, and Westinghouse cycles. Based upon informal information from LLNL, the GA process temperature will be in the range 725°C to 975°C. This tentative range is set at the upper limit by the high cost of heat exchanger materials, and at the lower limit by the large recycle flows and large process vessels required. The time frame for economic development of water splitting is highly uncertain.

3.2.4 Commercial Reforming Projections

The HTGR is potentially applicable to the following energy markets:

- Dispersed industrial heat (non-baseload) - (TCP applications).
- Peaking and mid-range electricity - (TCP applications).
- Oil shale processing - (TCP application plus hydrogen production).
- Coal refining - production of gaseous and liquid fuels.
- Ammonia and methanol production (either with coal or methane feedstock).
- Water splitting to produce hydrogen.

3.2.4.1 The Thermochemical Pipeline System

The concept that combines the HTGR with a thermochemical pipeline (TCP) appears to be an effective way to utilize the HTGR in serving multiple energy markets and is of particular interest in the near term. Markets for end-use methanator heat consist primarily of dispersed industrial heat users (including steam supply for synfuel plants), peaking and mid-range electricity production, and oil shale processing. The methanators can be considered as a replacement for current industrial boilers (oil-fired and gas-fired) and for distillate fuel burning in electric generators (gas turbine or combined cycle).

Analyses indicate that this system, the HTGR-TCP, can supply energy at costs competitive with available alternatives in the 2000-2020 time frame. Two U.S. markets of particular interest are the dispersed industrial heat users (one- and two-shift) plus peaking and mid-range electric power generation systems. In the 2000 to 2020 time period, it is estimated that these potential markets will total approximately 400 GW(t). This amounts to about 10 quads of nuclear energy substitution for fossil fuels.

Based on the information now available, the HTGR-TCP system appears to compare very favorably with other modes of energy supply in the future context. Because much of the technology is novel, reliable cost estimates will be difficult to obtain until more development work is completed. This is also true of the competing technologies; therefore, increased uncertainty in comparing alternatives is likely to characterize energy analysis for some time into the future.

Since a healthy industrial economy growing at a rate commensurate with population growth is necessary if acceptable standards of living are to be maintained, energy supply to industry will have to be ensured. The HTGR-TCP concept appears to be a promising way of achieving that goal.

3.2.4.2 Synthetic Fuel Conversion

The application of HTGR process heat technology to the various coal refining markets initially indicates that product costs in the U.S. are about equal to those evolved from using coal as a heat source and somewhat higher if methane is similarly used. The coupling of the HTGR to the coal gasifier for the eventual export of synthesis gas appears to change this balance in favor of the HTGR in several applications, providing a highly flexible resource--synthesis gas--which allows one to produce hydrogen, ammonia, petroleum, and coal refinery products and to contribute directly to the production of heat.

As coal refining is implemented on a large scale in the U.S., the economics would be expected to shift more strongly in favor of the HTGR. Basically, coal resources would be depleted more rapidly (20% to 40% without the HTGR in most processes) and coal prices would increase relative to nuclear fuels. The potential, therefore, exists to develop optimized systems in which nuclear process heat can be integrated into an economically competitive coal conversion system.

3.2.4.3 Oil Shale

The processing of both eastern (Devonian) and western U.S. oil shales has been considered briefly and is planned to be a continuing subject of future work.

The geographical location of western oil shale deposits lends itself to the use of a central heat and power facility: plants for developing the oil shale would be within a 32.2-40.2 km (20 or 25 mi) radius of a centrally sited HTGR-TCP, located on federal land.

Use of nuclear heat could release for sale substantial (20% to 30% of plant output) amounts of hydrocarbon products which would otherwise be used in recovery and processing. Nuclear heat could also make unnecessary the development of facilities for transporting and handling coal and its waste products after combustion and scrubbing, and could substantially reduce air pollution caused by combustion of oil shale products.

The extent to which various energy forms from the HTGR might be substituted for fossil fuels in oil shale products can only be ascertained by further study of mining, retorting, upgrading, and refinery operations currently required for such processes.

Table 3.2.4-1, Resource Impacts, shows a summary of the current results of this assessment for typical applications. Fig. 3.2.4-1 shows pictorially the relationship of the HTGR to the TCP and coal conversion processes, and summarizes the estimated savings from Table 3.2.4-1 for selected market applications.

3.2.5 Demonstration/Deployment Strategy

3.2.5.1 Reference Demo to Identified Commercial Plant

The 950°C, direct cycle HTGR-Reformer/TCP using pebble fuel is GE's current choice for the identified commercial plant. The current reference U.S. lead plant is the 850°C, indirect cycle reformer plant using a prismatic fuel.

The progression from this lead plant to a prototype 950°C direct cycle reformer plant using a similar prismatic fuel core follows a straightforward development, test, and demonstration program. However, if work on the pebble bed concept demonstrates clear superiority in achieving the necessary process heat temperatures that are identified, the shortest path to commercialization will involve cooperative programs with the FRG. A likely progression from a prismatic core lead plant to an HTGR-R/pebble bed direct cycle 950°C plant is depicted in the following table which shows the assumed FRG, as well as U.S., program. It is expected that the U.S. government will bear the development costs for the U.S. work shown, with industry paying for the commercial value received from the products of the plants.

	U.S.	FRG
1981	Continues work on pebble core as backup to prismatic; other U.S.-FRG joint programs continue.	
1982	Ft. St. Vrain successfully demonstrates core operation at 750°C and full power.	PND steamer project authorized - 750°C, 1500 MW(t), pebble core
1983		
1984	U.S. lead plant authorized - HTGR-R/TCP, indirect cycle, 850°C, 1170 MW(t), prismatic.	THTR begins operations.

Table 3.2.4-1

RESOURCE IMPACTS

Application	HTR-Synfuel Market Scenario(1,2)	Total HTR Capacity (GW_t)	Total Fossil Fuel Displaced(4)	Total Annual Ore Requirements(3) (U_3O_8 Tons/Yr)
A. HTR-TCP	Dispersed Industrial Heat and Peaking Electricity	400	Oil and Methane (10 Quads/yr)	5,600 to 24,000
B. Coal Conversion Processes				
1. SNG (Methane)	6000 MSCF/day	30	Coal (0.8 Quads/yr) (25% of feed)	400 to 1,800
2. Syngas (H_2+CO)	60,000 MSCF/day	60	Coal (1.5 Quads/yr) (40% of feed)	800 to 3,600
3. Gasoline (Indirect liquefaction) (SASOL)	2,000,000 bbls/day	75	Coal (1.8 Quads/yr) (40% of feed)	1,000 to 4,300
4. Gasoline (Direct Liquefaction (H-Coal, SRC, EDS and upgrading)	2,000,000 bbls/day	35	Coal (0.8 Quads/yr) (20% of feed)	600 to 2,700
5. Ammonia	15,000,000 tons/yr	12	Coal (0.3 Quads/yr) (60% of feed)	200 to 800
C. Methane Conversion				
1. Ammonia	15,000,000 tons/yr	7	Methane (0.2 Quads/yr) (40% of feed)	100 to 400
2. Methanol	200,000,000 tons/yr	60	Methane (1.5 Quads/yr) (30% of feed)	800 to 3,600
D. Oil Shale Processing TCP and Hydrogen Production Application)				
1. Western	2,000,000 bbls/day (Arabian Light Crude equivalent)	20	Coal and Shale Oil (0.4 quads/yr) (25% of feed)	300 to 1,200
2. Eastern	2,000,000 bbls/day	40	(0.8 quads/yr) (50% of feed)	600 to 2,400

NOTES

(1) Based upon year 2000 projections

(2) Typical plant sizes are: Ammonia - 10^6 tons/year/plant; SNG-200 MSCF/day/plant; Gasoline-50,000 bbl/day/plant

(3) Range based upon possible HTR and fuel cycle designs. First core load requirement varies from 170 to 250 tons U_3O_8/GW_t.

(4) Composite evaluations of various specific processes considered for each application.

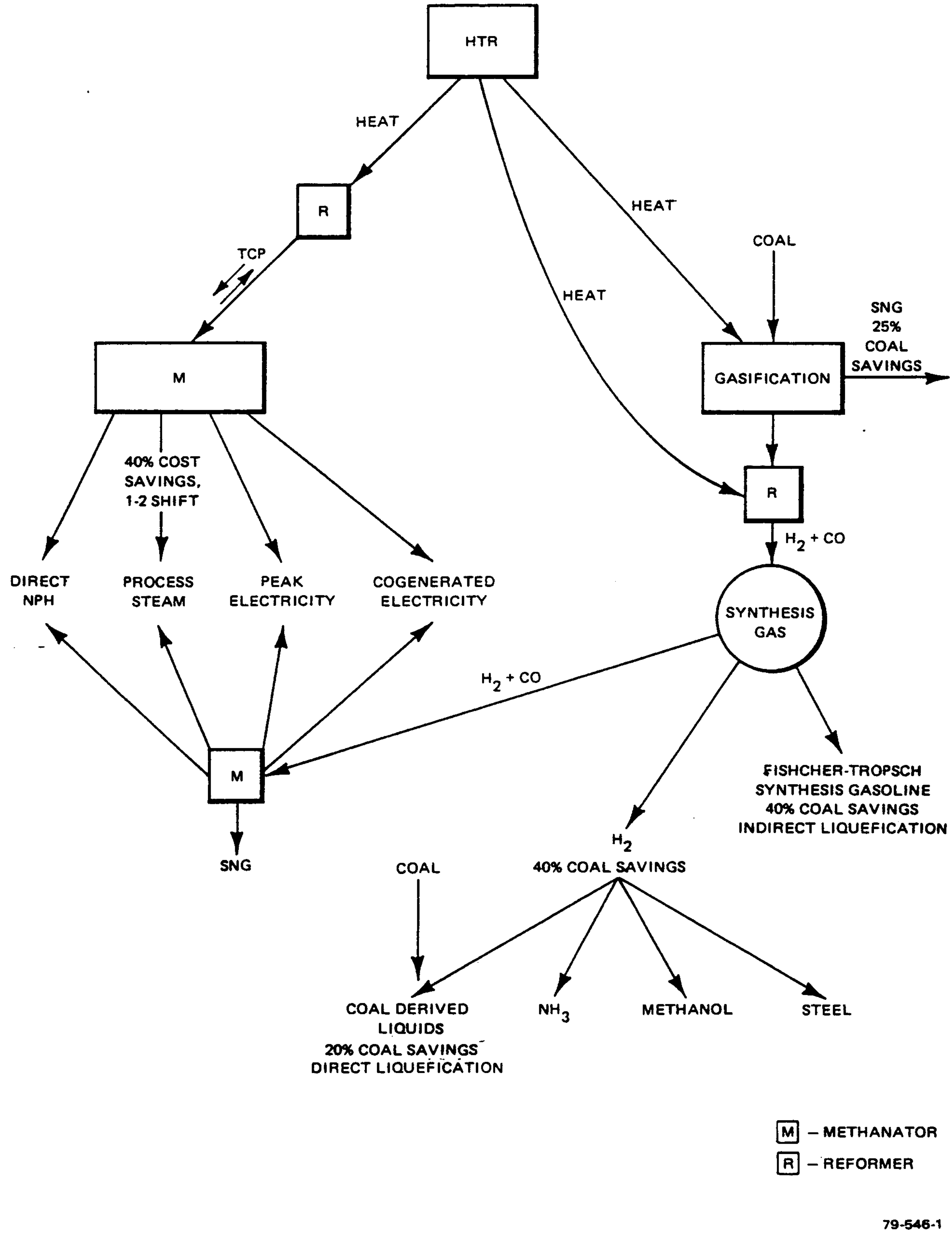


FIGURE 3.2.4-1 APPLICATIONS OF HIGH TEMPERATURE NUCLEAR PROCESS HEAT

	U.S.	FRG
1985	Begin licensing dialog with NRC on pebble core.	
1986		THTR demonstrates successful pebble operation at 750°C.
1987	Complete design of U.S. pebble core for 950°C, 1500 MW(t), direct cycle.	
1988	Decision made making pebble bed, direct cycle reference U.S. design.	PNP reformer project authorized-950°C, 1000 MW(t), pebble core, direct cycle, coal gasification.
1989		
1990	Begin direct cycle reformer test operation.	
1991	Negotiations complete with FRG on Coop programs.	
1992	U.S. HTGR-R/pebble direct cycle 950°C, 1500 MW(t), authorized	PND begins operation.
1993		
1994		
1995		PND demonstrates successful operation.
1996	HTGR-R/TCP prismatic core begins operation.	
1997		
1998	HTGR-R/TCP demonstrates successful operation at 850°C, with direct cycle reformers installed in one intermediate loop.	PNP begins operation.
1999		
2000		PNP demonstrates successful operation at 950°C.
2001		
2002		
2003		
2004	U.S.-HTGR-R/pebble, direct cycle, 950°C begins operation.	

3.2.5.2 Alternative Strategy to Identified Commercial Plant

Since the strategy depends upon information to be developed in the future (e.g., materials solution for 950°C operation, economics and licensing for direct and indirect cycles, prismatic and pebble core operating results, etc.) and the future environment, there seems to be little basis for developing alternative hypothetical deployment strategies at this time. However, a strategy which appears preferable to that illustrated in Section 3.2.5.1 is shown below:

	U.S.	FRG
1981	Continues work on pebble core as backup to prismatic; other U.S.-FRG joint programs continue.	
1982	Ft. St. Vrain successfully demonstrates core operation at 750°C and full power.	PND steamer project authorized-750°C, 1500 MW(t), pebble core.
1983	U.S. concentrates on 950°C, direct cycle HTGR-R/TCP (or H_2 for synfuels).	
1984		THTR begins operations.
1985	Begin licensing dialog with NRC on pebble core.	
1986	Decision made making pebble bed, direct cycle reference U.S. design.	THTR demonstrates successful pebble operation at 750°C.
1987	Complete design of U.S. pebble core for 950°C, 1500 MW(t), direct cycle. U.S. HTGR-R/pebble direct cycle 950°C, 1500 MW(t), authorized.	
1988		PNP reformer project authorized - 950°C, 1000 MW(t), pebble core, direct cycle, coal hydrogasification.
1989		
1990	Begin direct cycle reformer test operation.	
1991	Negotiations complete with FRG on Coop programs.	
1992		PND begins operation.
1993		
1994		
1995		PND demonstrates successful operation.
1996		
1997		
1998		PNP begins operation.
1999	U.S.-HTGR-R/pebble, direct cycle, 950°C begins operation.	
2000		PNP demonstrates successful operation at 950°C.

3.2.5.3 Assessment of (Relative) Risks and Costs

Risks and costs are minimized by:

- Maintaining separate, backup U.S. and FRG HTGR core concepts until operating characteristics and development requirements for 950°C operation are clear.

- Coalescing work on U.S. and FRG HTGR designs after design choice is clear.

The risk for the follow-on direct cycle plant might be slightly reduced by incorporating a direct cycle reformer design in one loop of the lead plant.

3.2.5.4 Importance of the Lead Project

The Lead Project (850°C, indirect cycle, prismatic core) would do the following towards realizing the identified commercial plant:

- Identify problems and solution on gas/steam plant, pipeline, gas storage and methanator system, which are all common to the commercial plant.

- Identify problems and solutions on reformer/steam generator plant operation and control.

- Provide a prestressed concrete pressure vessel for installing a direct cycle reformer design, but only at 800°C.

- Identify solution to carburization and operation of heat transfer components (IHX) at 850°C.

- Provide a dialog with NRC on licensing requirements for gas plant facilities near the reactor containment building, some of which are common to the commercial plant.

- Identify costs of the various plant components.

- Provide first step towards gaining owner/user support.

However, the lead plant would not contribute towards the following commercial plant obstacles:

- Licensing requirements for reformers within the PCRV.

- Licensing requirements for pebble bed core.

- Building pebble core design and fabrication capability in U.S..

- Provide demonstration of heat transfer component operation in reactor helium at 950°C. However, this may be obtained by non-nuclear test facility.

- Provide demonstration of HTGR core operation at 950°C. However, the lead plant might be built with the capability to be operated at 950°C when suitable heat exchange components (IHX) are available.

4.0 COMMERCIAL PLANT MARKET/BENEFITS ANALYSIS

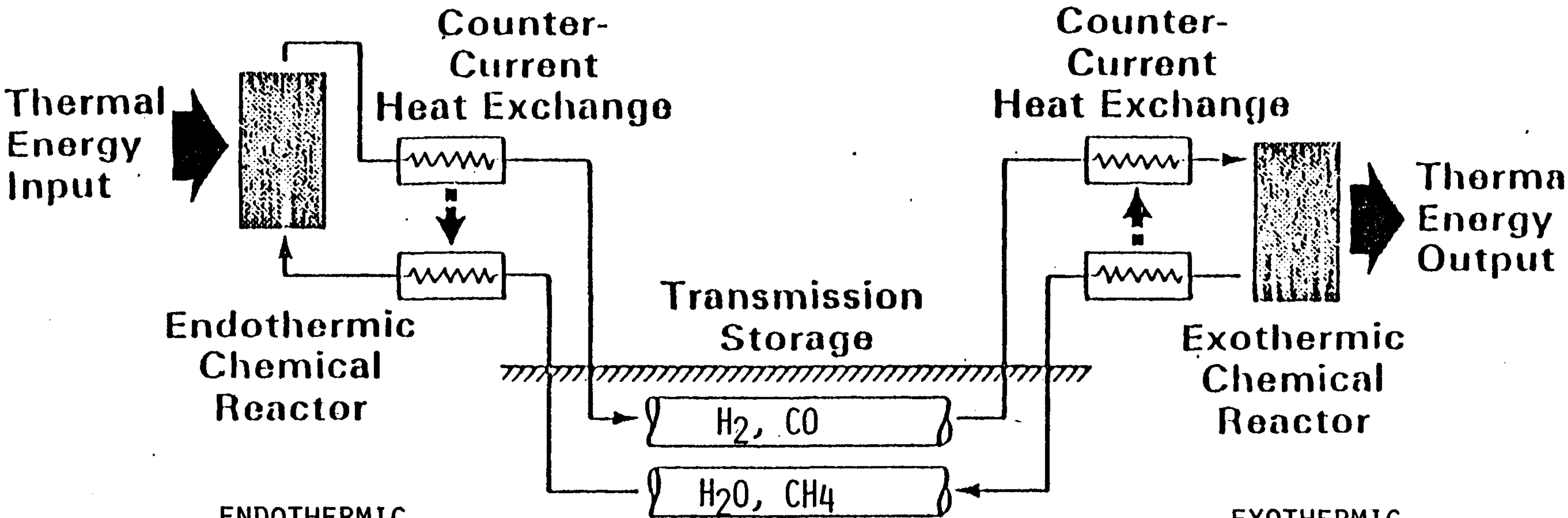
The HTGR-R is projected to have many commercial applications which utilize the reactor energy to cogenerate heat and electricity. All of the envisioned near-term applications deliver heat to a reformer to drive a steam-methane conversion reaction and to a bottoming steam generator for baseload electric power production. The reformer produces a storable/portable synthesis gas which can be utilized for remote, low-capacity-factor process heat or electric power generation; a chemical feedstock for the methanol, fertilizer, ammonia, petrochemical or steel industries; and heat and hydrogen upgrading for the developing synthetic fuel industry. These applications are clearly outside the capability of the current light water reactor (LWR) and present a greatly increased potential for supplying the U.S. and world energy needs with nuclear power. The HTGR is also expected to have advantages over other nuclear power alternatives relative to investment protection, operator exposure, and licensing. Substantial benefits could also be accumulated from the reduced environmental impact of the HTGR relative to coal, its anticipated competition, in the areas of air pollution, mining and transportation requirements, and solid waste disposal. The HTGR-R augments the current use of nuclear power and provides a much-needed alternative to coal for many industrial applications. The contents of this section will attempt to identify and quantify the potential market for the HTGR-R and the benefits associated with its deployment.

4.1 Commercial Plant Market Evaluation - HTGR-R

The HTGR-R can produce energy in many end-use forms including electricity, direct heat, and steam. The various HTGR-R applications utilize the unique capabilities of the plant but deliver energy to the user in different forms. Three major market categories have been identified and are discussed in greater detail below. They are remote energy delivery, synthetic fuel production, and chemical feedstock supply. Since electricity is also generated by the HTGR-R in an intermediate and peaking mode and/or baseload operation, a description of the electric power market is also provided.

4.1.1 Remote Energy Delivery

The HTGR-R may be coupled to a remote energy distribution system through use of the thermochemical pipeline (TCP) concept. This application was selected as the reference for the HTGR-R Lead Project since it was perceived to be an easier demonstration path for the HTGR-R concept. A brief description of the TCP concept is provided; however, more details may be found in Section 5.1.3. The TCP is one of several closed-loop chemical energy systems under study which transport and/or store thermal energy by the use of reversible chemical reactions. The TCP utilizes steam and methane (or other suitable light hydrocarbon) feedstock which is particularly well suited to the temperature capabilities of the HTGR and can deliver high-quality heat to prospective users. The basic concept is shown in Figure 4.1.1-1. In the TCP concept, reactor energy is converted to a chemical form by a catalyzed



*$CH_4 + H_2O\ (L) \xrightarrow{\text{ENDOTHERMIC}} CO + 3H_2$

$\Delta H^{0}_{298} = 250$ KJ/MOLE

$CO + 3H_2 \xrightarrow{\text{EXOTHERMIC}} H_2O + CH_4$

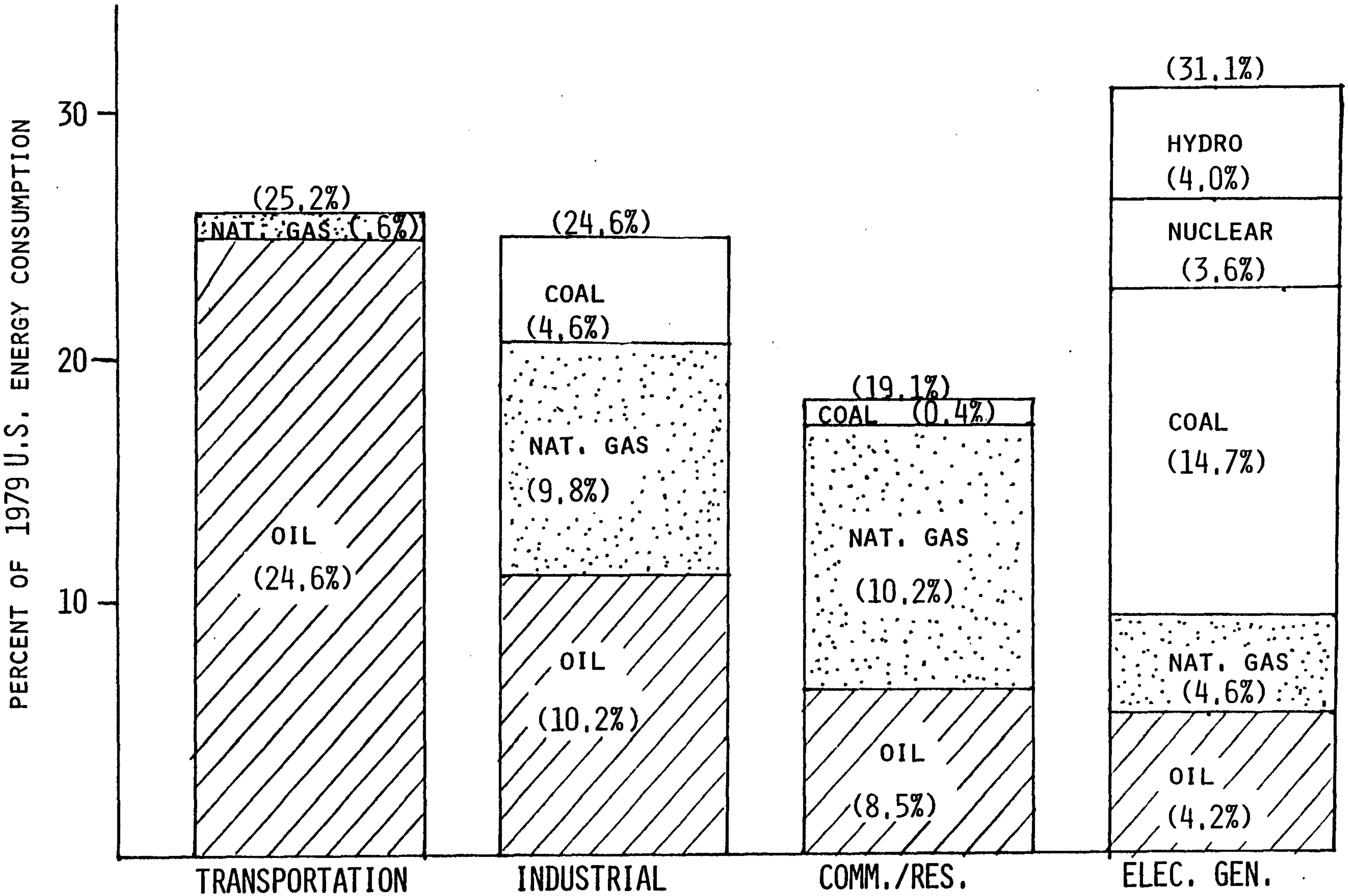
$\Delta H^{0}_{298} = -250$ KJ/MOLE

*PRINICIPAL REACTION

Figure 4.1.1-1 Thermochemical Pipeline (TCP) endothermic chemical reaction in the reformer. The reaction is driven toward completion at higher temperatures and lower pressures. Preheating of the reformer inlet gases to reforming temperatures is accomplished by recuperative heat exchange with the reformer effluent gases. Most of the sensible heat imparted to the effluent gases is thus recovered, permitting near-ambient transmission and storage. The synthesis gas, a mixture of hydrogen and carbon monoxide that is the product of the steam-methane reformer, can be indefinitely stored until its chemical energy is recovered by methanation. A catalyzed exothermic chemical reaction takes place in the methanator which converts the hydrogen and carbon dioxide back to its original steam and methane constituents. As the methanation reaction also takes place at elevated temperatures, once again the sensible heat of the effluent is utilized to preheat the methanator feed gases to reaction temperatures.

The TCP is a closed-cycle system with the only exchanges with the environment being thermal and mechanical energy. Of particular interest is the fact that high reactor thermal utilization is achieved with no releases to the environment at the user site other than waste heat. The TCP medium also retains its energy content during storage and transmission without the degradation experienced in thermal storage and transmission systems. This is due to the nature of the chemical conversion, which does not spontaneously release the stored energy at ambient temperatures. In fact, the large activation energy requirements for the TCP chemical reaction require elevated temperatures and the presence of a reaction catalyst. It is this feature of the TCP which makes consideration of long-term storage and long-distance transmission plausible.

Industrial process heat and energy requirements comprise over one third of the total energy usage in the U.S. Over 50% of this industrial energy is provided by oil or natural gas, which is projected to become in shorter supply and higher priced over the next 20 years. Fig. 4.1.1-2 provides a more detailed breakdown of U.S. energy consumption in 1979 by both sector and fuel. The U.S. industrial sector is characterized in general by geographically dispersed users which require energy at various different temperatures, in different quantities, and for different time periods. The large majority of these industrial users are very small, one-shift operations. The past availability of relatively cheap energy in the form of natural gas and oil provided industry the incentives to adopt these fuels for primary energy sources. However, rapid price increases, projected scarcity, and government regulations are forcing conversion or reconsideration of oil and natural gas use in current and future industrial applications. The Fuel Use Act, one such government regulation, will result in alternate energy sources for nearly all new industrial boilers over 100 million BTUs/hr capacity [30 MW(t)]. Conversion of most existing industrial boilers of this size will also occur. Coal is currently envisioned to be the fuel for these industrial boilers. The bulk of the small industrial users [less than 30 MW(t)] will find conversion to coal more difficult. In order to provide reliable and economical



Ref. 1

Figure 4.1.1-2 U.S. Gross Energy Consumption 78.2 Quads in 1979 energy for low-capacity-factor, low-thermal-output users, an alternative energy source to coal should be identified. The continued use of oil and natural gas by the small industrial users remains a distinct possibility, but this does not help resolve the U.S. dependence on foreign energy sources. Electric heaters may ultimately prove economical and may provide the needed reliability, or the infusion of synthetic oils and gases may provide options to these users.

The TCP is also envisioned to be able to penetrate this market. Its capability to provide clean energy at temperatures up to 538°C (1000°F) to dispersed users may permit consideration of nuclear-generated heat for industrial processes, especially the dispersed, small users which a central steam cogeneration plant could not serve. General Energy Associates, Inc. has prepared industrial fossil energy profiles for 247 utility service areas in the U.S. These profiles did not address the end-use of electrical energy in the industrial sector; however, approximately 750,000 GW-hrs are projected to be used by industries in 1980. A summary of this information is provided in Table 4.1.1-1. This table provides data regarding energy uses by temperature, plant size, hours of operation, and fuel use. As can be seen, over 75% of the industrial energy requirements projected in these areas in 1980 are in the temperature range up to 538°C (1000°F). 56% of this energy is delivered below 260°C (500°F), and the remaining 44% is delivered between 260° and 438°C (500 and 1000°F). The total size of this market is shown to be above 14 quads in 1980 and is projected to be 56 quads by 2020. While the LWR is capable of over 260°C (500°F) steam temperatures, delivery to a process user is more likely limited to 232°C (450°F) or less. The HTGR, on the other hand, could provide all the energy input up to and possibly above 538°C (1000°F). The HTGR can also provide this energy in the form of either steam or direct heat. The industrial heat market can be more simply characterized as a large group of small, one-shift operations and a small number of large, multi-shift operations. The HTGR-R is expected initially to penetrate the low-capacity-factor user market, with the possibility of evolution to multi-shift users farther in the future. It is expected that large coal- or nuclear-powered central cogeneration facilities would provide a more economical energy source to three-shift users.

While there is a large market identified for industrial heat, it is important to look at specific applications and sites to determine the suitability of proposed energy transmission and storage systems. One such study is under way in the New Jersey area within the service territory of Public Service Electric & Gas Company. This study will examine the potential nuclear sites in relation to industrial user locations and geological formations suitable for gas storage.

The reliability of the HTGR-R-driven TCP must be proven before attaining industrial acceptance as a primary energy source. Demonstration through utility operation as an all-electic generator may provide such a vehicle for acceptance and also identifies yet another application for remote energy delivery. Through the use of energy storage and transmission, the TCP concept provides a potential method of smoothing

TABLE 4.1.1-1

SUMMARY OF INDUSTRIAL FOSSIL ENERGY PROFILES
IN 247 UTILITY SERVICE TERRITORIES
(Basis: Ref. 2)

		1980	1990	2000	2010	2020
Energy (QUADS)	Less than 500°F	7.716	10.84	15.68	22.49	32.29
	500 to 1000	5.957	8.296	11.93	17.01	24.37
	1000 to 1700	.5322	.7242	1.038	1.477	2.113
	Less than 1700°F	14.21	19.86	28.65	40.98	58.77
	Above 1700°F	3.670	4.830	6.640	9.030	12.43
	Steam	8.198	11.45	16.49	23.54	33.72
	Direct Heat	9.682	13.24	18.80	26.47	37.48
	Total	17.88	24.69	35.29	50.01	71.20

		Number	Percent
Plant	Less than 2500 hrs	275871	82.3
	2500 to 6000	53605	16.0
	More than 6000	5870	1.8

Thermal Rating		Number	Percent	Number Less than 3000 hrs
Less than .5	Million	255700	76.2	247580
.5 to 1	Watts	30424	9.1	23696
1 to 2	Thermal	21068	6.3	13116
2 to 5		15080	4.5	7935
5 to 10		5585	1.7	2936
10 to 20		3447	1.0	1460
20 to 35		1493	.4	527
35 to 50		708	.2	243
50 to 100		1062	.3	204
More than 100		779	.2	17
Total		335346		297714

1980 Fuel (QUADS)	Oil	3.853
	Coal	3.699
	Gas	9.277
	Other	1.051

utility generating requirements through peaking and mid-range methanation plants powered by the stored gases. The basic nuclear reactor is baseloaded, while the gas storage and transmission systems provide the system load-following capability. When economics and reliability relative to conventional fossil-fueled peaking units are established, this concept could very well expand the nuclear energy contribution within the electric power production market.

4.1.2 Synthetic Fuel Production

The market for commercial synthetic fuel plants, whether they be coal gasification, coal liquefaction, or shale oil production, is very speculative at this time. Depending on the size and availability of domestic and foreign crude oil production, U.S. demand, the cost of energy, and the level of government support for the synthetic fuel industry, widely differing projections of synthetic fuel production exist. The President's proposal of an $88 billion synthetic fuels development program last year has stimulated tremendous interest in synthetic fuels in government and industry. The government interest is driven by the strategic and economic implications of U.S. reliance on foreign energy sources. Although the rapidly increasing energy costs are bringing synthetic fuel production closer to a commercial reality, the development of synthetic fuels will be driven primarily by government policy and incentives through the year 2000. At some time after the turn of the century, it is expected that a truly commercial synthetic fuel industry will emerge to meet energy supply shortfalls in portable fuels.

Although a great deal of uncertainty exists regarding the potential size of the synthetic fuels market, there are no technical barriers facing its deployment. In fact, there are several commercial synthetic fuel processes, and a viable synthetic fuels industry has existed in the world since the early 1800s in the form of coal gasification. Synthetic fuel research and development activities have been under way for 30 years in the U.S., and several first-generation technologies have been deployed commercially. In addition, several second-generation technologies have attained the pilot plant stage, with plans under way for demonstration plants. The financial, regulatory, and political barriers facing commercial deployment of synthetic fuel plants are formidable but surmountable, leading to commercial deployment in the early 2000s.

Several studies (Refs. 3, 4, and 5) have examined the synthetic fuel market potential in the past two years and have indicated promise for the HTGR. Fig. 4.1.2-1 depicts TRW's conclusions (Ref. 3) regarding the market penetration of the HTGR. The TRW study assumed electricity generation applications of the HTGR to be a fall-out of the synfuel and process heat applications. This HTGR-synfuel market was assessed to be forty 3000-MW(t) plants by 2020 in the nominal case and five 3000-MW(t) plants in a pessimistic case based upon early introduction (mid-1990s) of an 850°C core outlet indirect cycle HTGR-R. These conclusions were based on a nominal projection of a 40%

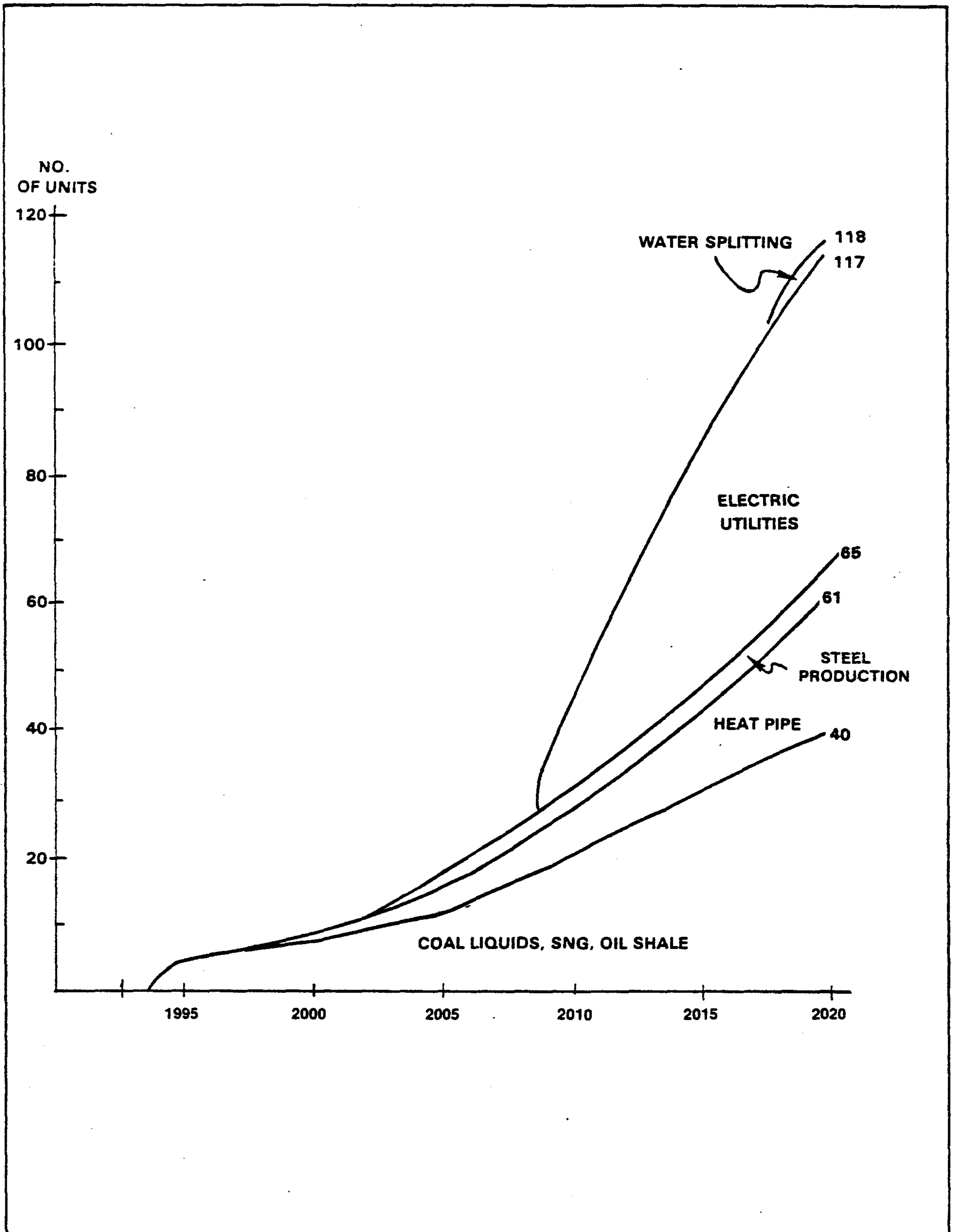


Figure 4.1.2-1 Gas-Cooled Reactor Market Penetration

Ref. 3

market share in synthetic fuel plants and a pessimistic projection of a 10% market share in coal liquids respectively. This study was updated recently (Ref. 6) to account for changes in the perceived deployment of synthetic fuels plants and the more pessimistic schedule assumed in Fig. 2.2-1 (commercial deployment in 2008). For these changes, the HTGR market penetration projection through 2020 was eleven 3000-MW(t) nominally and two 3000-MW(t) plants in the pessimistic case. The recent Pace study (Ref. 5) confirms the size of the overall synthetic fuel market in 2020 but has substantially different conclusions regarding the product mix and timing of that market. As interest has been focusing on the synthetic fuel market, considerations of the future U.S. needs and capabilities have been taken into account. These considerations have led to the following points:

- The commercial synthetic fuel industry is not likely to emerge on a large scale until after the year 2000. The timing matches the projected deployment of the commercial HTGR-R.

- Oil shale conversion is likely to be the first technology developed on a commercial scale.

- The shortfall in U.S. demand will likely be initially felt in liquids, resulting in demands for synthetic liquids from oil shale and coal.

- The deployment of commercial synthetic fuel plants could ultimately be limited by industry's capability to design, manufacture, and construct. Public and political resistance to large centralized energy projects and their developers could also loom as a pitfall to large-scale deployment (Ref. 5).

- The synthetic gas market will be slower to develop as increased projections on the availability of natural gas are made.

The HTGR-R holds a great deal of promise to capture a broad segment of the synthetic fuel market. The HTGR-R may provide financial incentives, certainly in some areas of the country relative to transported coal. Of greater significance is the HTGR's capability for higher resource utilization and reduction in plant effluents. These capabilities can lead to substantial environmental savings with regard to mining, transportation, and waste disposal requirements as well as air and water pollutants. The environmental impact is discussed in Section 4.3.3. Deployment of a large-scale synthetic fuel industry may well mandate tight environmental restrictions on plant siting and operation. These restrictions could very well lead to a large marketplace for HTGR-R-powered synthetic fuel plants.

4.1.3 Chemical Feedstock Supply

Traditional reforming industries have utilized steam reforming of natural gas to produce the desired commercial end-products. The primary products of these industries are ammonia and methanol. In addition to using the natural gas as a feedstock to the reformer, it is also utilized to fire the reformer. The HTGR can be used to directly displace this natural gas which is burned to fire the reformer and provide the steam and electric needs of the process as well. Consumption of ammonia is based primarily on its agricultural role, and about 75% of the U.S. ammonia production is used in this area. If the U.S. is to continue to be the bread basket of the world, increased agricultural yields via increased use of fertilizer will call for an increasing demand for ammonia. As natural gas becomes costly and in shorter supply, alternative feedstock sources will undoubtedly be examined. Similarly, the HTGR-R can supply merchant hydrogen or synthesis gas for direct iron reduction processes in steelmaking or a feedstock for chemical industries.

Not only can the HTGR directly displace the natural gas burned in the reforming plant with relatively inexpensive nuclear fuel, but it can also be coupled with a coal gasification process to supply a substitute natural gas (SNG) feedstock. The HTGR-R can also eliminate the air pollutants, carbon and nitrogen oxides, associated with the combustion of natural gas in air. The HTGR market potential in chemical feedstock supply is likely to be smaller than for synthetic fuel production, although there is the prospect for a chemical energy complex where a large central coal conversion plant powered by several HTGRs can produce chemical feedstock as well as synthetic coal gases and liquids.

4.1.4 Electricity

Although baseload electricity is not the primary output of the HTGR-R, some net power to the grid will likely be provided in the design of the plant. This baseload contribution is expected to be rather small and, therefore, not have a large impact on the projected baseload electric market. However, the capability of the HTGR to be coupled to energy storage concepts permits consideration of nuclear power in the intermediate and peaking electric markets. These systems may prove competitive against battery or compressed-air storage concepts now under consideration for current baseload power plants. As 30-40% of the current installed generation capacity serves the intermediate and peaking market, there is a substantial replacement market which will develop as fossil fuels become increasingly expensive and subject to environmental or political restrictions. The results of a GCRA study (Ref. 7) have indicated a substantial system savings associated with the availability of energy storage options. A study funded by Electric Power Research Institute (EPRI) (Ref. 8) concluded that up to 5% of our electrical energy consumption could be supplied by energy storage on a weekly cycle and 3% on a daily cycle. Further, the study indcates that up to 12% or 17% of annual peak load capacity can be supplied by energy storage (75% efficiency) on daily or weekly cycles respectively. Installed electric generation capacity projections made by Hanford Engineering Development Laboratory (HEDL) (Ref. 9) indicate that between 1566 (low growth) and 2466 (high growth) GW(e) will be on line in 2020, with the median being 2034 GW(e). Assuming that the EPRI projections are correct and that the HTGR is capable of penetrating one half of the energy storage market, a market of 100-200 GW(e) of installed capacity by the year 2020 is indicated. This translates to a market of at least 100 1170-MW(t) HTGR plants.

The projections are speculative at best and certainly assume that the reliability and availability of the HTGR and its energy storage system are proven and are economically attractive relative to competing energy storage systems. More work needs to be done on the HTGR plant design and cost estimates, and the progress of alternative storage mechanisms needs to be monitored as well.

4.2 Commercial Incentives

The HTGR-R plant design and cost estimates are at a developmental stage and, therefore, contain uncertainties/risk relative to commercially available options. It is clear that additional design and development activities are required to confirm plant design and cost assumptions; however, there are many other incentives for the deployment of the HTGR-R which are better defined. These incentives relate to the inherent design characteristics of the HTGR, which exhibit advantages in the areas of investment protection, operational exposure, public acceptance, institutional compatibility, licensing, and evolutionary capability.

4.2.1 Investment Protection

The HTGR concept presents a nuclear power option with the capability to substantially reduce the risk of loss of capital investment. This feature has always been a part of the HTGR plant design, but little emphasis was placed on it prior to the Three Mile Island (TMI) plant incident. The repercussions of TMI have caused concern to stockholders and company officials regarding the safety of their investment in nuclear power reactors. Although the final impact of the TMI plant will not be known for several years at best, it is clear that purchased power requirements alone can lead to substantial financial impact on a utility. The HTGR has many features which would minimize the probability of repeating the severity of the outage caused by a TMI-type incident. In addition, the HTGR has greater operational flexibility to manage abnormal plant transients.

The HTGR utilizes a graphite core structure, ceramic fuel, a gaseous coolant, and a concrete reactor vessel. These features serve to limit the possibility of a catastrophic event but also provide the forgiving nature of the HTGR with respect to plant transients such as loss of forced primary circulation. The high core heat capacity and low power density result in very slow and predictable plant temperature transients. A complete loss of flow can be tolerated for a period of minutes to hours without damage to the core or components. This results in adequate time for plant operators to identify the problem and respond properly to the initiating event. The graphite structural elements of the core tend to even temperature distributions across the core because of its high thermal conductivity and because its structural integrity is maintained up to temperatures of 3000-3300°C (5500-6000°F). The ceramic-coated fuel particles provide a fission product barrier, which leads to a slow, controlled release of volatile nuclides under accident conditions. The primary and secondary containment systems in conjunction with plant cleanup systems can then limit offsite releases. The strong negative temperature coefficient of the core provides a self-shutdown mechanism, and the core does not require a fast-acting shutdown system.

The gaseous helium coolant provides several advantages as a cooling medium. Since helium acts only in a single phase, there is no flashing. The single-phase coolant also leads to greater certainty of pressure measurements, no requirement for level measurements, no primary circulator cavitation problems, and no additional coolant inventory for plant cooldown. In an accident condition, there is no loss of coolant since helium and/or air can be circulated through the core to cool it down. Also, helium is neutronically transparent and chemically inert and, therefore, has no reactivity effects or interactions with the fuel. The low level of stored energy in the single-phase coolant also results in lower energy releases to the containment building in the unlikely event of primary pressure boundary failure.

The prestressed concrete reactor vessel (PCRV) has many structural features which also limit the probability of primary system pressure boundary failure. The multiplicity of tendons makes failure of individual structural members inconsequential. The massive concrete structure provides shielding, not only for the structural tendons but also for personnel access to the containment building. The effects of neutron embrittlement on structural components in the PCRV are, therefore, essentially eliminated. The fact that the concrete is under compression makes cracks self-sealing, and the concrete does not sustain crack propagation.

These inherent features provide additional time for operator response to accidents and severe plant transients, making operator errors less likely. The forgiving nature of the core also provides for greater protection of the owner's investment in that major damage to the plant is less likely and takes longer periods of time to occur.

4.2.2 Reduced Operational Exposure

The fission product retention characteristics of the HTGR coated-particle fuel along with low circuit activity from primary system corrosion products are projected to result in exceptionally low primary system activity levels. These low primary system activity levels lead to a significant reduction in exposure rates to maintenance personnel. Since individual exposure limits are set by federal regulations, a reduced number of personnel are, therefore, required to accomplish a given maintenance task, with attendant reduction in maintenance cost. Experience in operating the early HTGRs supports this expectation as described below.

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 4.2.2-1, which was taken from Ref. 10. Because Peach Bottom was a 40-MW(e) prototype reactor, it can be compared with early, low-power LWRs. Exposure data for Big Rock, Humbolt and Lacrosse are presented in Fig. 4.2.2-1, 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.

At the time of its first refueling in February 1979, Fort St. Vrain (FSV) had generated 953 GW(e) hours of net power. Personnel exposure data collected indicate that FSV has exposure characteristics similar to those shown by Peach Bottom 1. Table 4.2.2-2 shows the FSV man-rem exposure data for the years 1977 and 1978, which are then compared to similarly sized plants in Fig. 4.2.2-2. Although it is still relatively early in its life, it can be seen that FSV 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 FSV 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. 11).

As an example, for the reference 900 MW(e) HTGR-SC plant, the refueling operation is expected to result in 5.5 man-rem of exposure, which is consistent with the FSV data when it is extrapolated for reactor size and the time delay that occurred at FSV 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. 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 FSV reactors both utilize highly enriched uranium (HEU-93%) fuel. The reference HTGR fuel is currently low enriched uranium (LEU-20%), 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

TABLE 4.2.2-1

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. 10

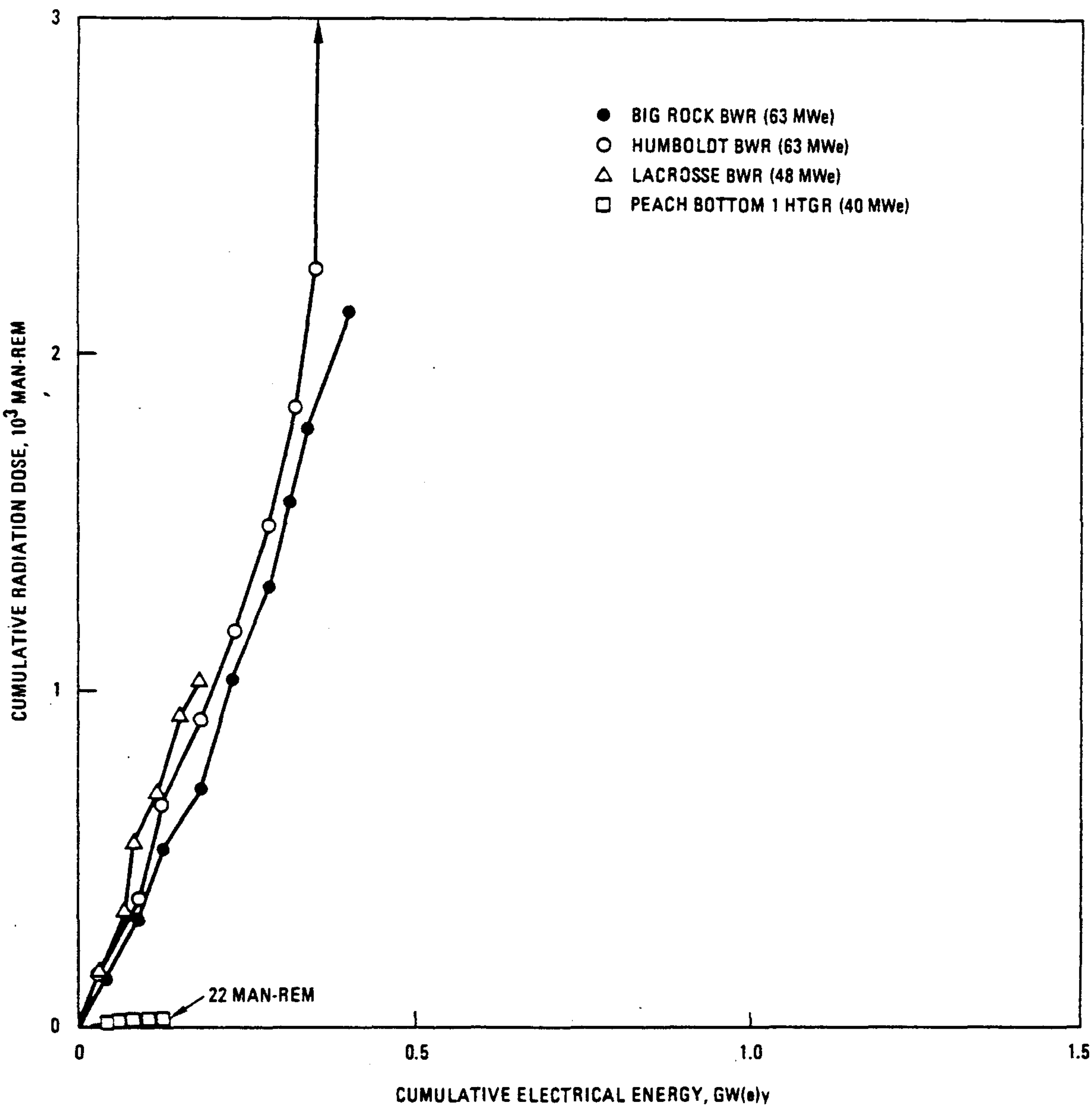


Figure 4.2.2-1 Cumulative Occupational Exposures for Early, Low-Power Nuclear Plants

Source: Ref. 10

TABLE 4.2.2-2

FSV MAN-REM EXPERIENCE

Personnel	Exposure	Averaged Man-Rem	Net Power Generation [GW(e)y]	Rate of Accumulation [man-rem/GW(e)y]
1977				
946	None	0		
55	<100 mrem	2.75		
1	100-250 mrem	0.175		
		2.9	0.0256	113
1978				
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. 10

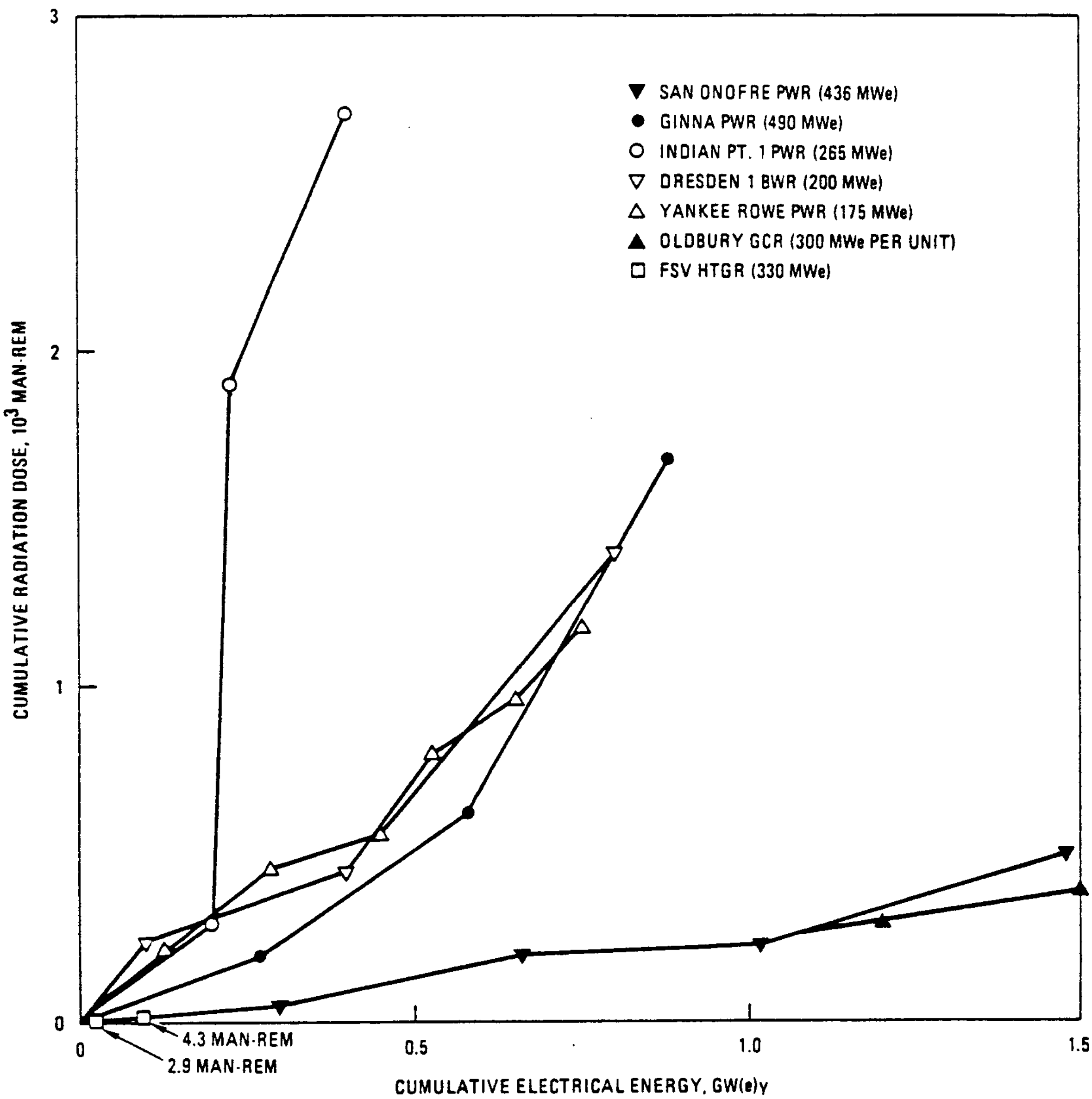


Figure 4.2.2-2 Cumulative Occupational Exposures for Medium-Power Nuclear Plants

Source: Ref. 10

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 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 rates and that the best means of accomplishing this would be through fuel performance.

In summary, the HTGR is expected to have substantially reduced occupational exposure rates. This difference in exposure can lead to lower maintenance costs and/or provide a margin if occupational exposure guidelines should be lowered in the future. These factors can be of great importance to the prospective owner and operator in considering future deployment of nuclear power.

4.2.3 Institutional Compatibility

Deployment of the HTGR-R or any nuclear concept in the industrial process heat market requires that many economic and institutional barriers be surmounted. The Midland Project has provided a current example of the problems involved in large central nuclear cogeneration plants. There are several issues which cause utilities and industries to take vastly different approaches to energy expansion and to make joint ventures difficult under the current financial and regulatory climate.

A major problem facing the deployment of nuclear power in the process heat market is the aspect of timing. Industrial firms typically have shorter planning horizons than utilities and require much shorter payback periods to justify their capital expenditures. If an industry is contemplating expansion of its facilities or retirement of existing boilers, the decision is not likely to be made in the minimum of 10-12 yrs in advance currently required for deployment of a nuclear reactor. Even if industries were to plan this far in advance, the uncertainty of the nuclear construction schedule associated with the future regulatory and financial conditions can easily cause slippage in the original schedule. These difficulties are being faced by a mature nuclear technology and would be no less a problem for a developmental technology such as the HTGR-R. Industry is not likely to jeopardize the operation of a production plant through use of an unproven technology to provide the steam or heat it needs. However, with the proper government support and incentives in the way of licensing reform, financial considerations, and the establishment of clear federal and state regulations, the deployment of nuclear cogeneration or process heat facilities may be possible. It is certain, however, that the industrial firms must have a reliable source of energy to continue operating. That mandates a backup energy supply, preferably one already in operation, which can cover any gap between the scheduled and actual operation dates of the nuclear facility.

In addition to the timing difficulties, the financial arrangements to construct the plant bring on further barriers. Utility financing provides access to less costly capital relative to private industry financing and thus potentially cheaper energy and product costs. However, the utilities are regulated by state and/or local bodies which approve rates of return and are increasingly involved in the evaluation of the timing and need for new facilities. In cogeneration facilities, utilities would like to see a higher rate of return granted to justify the additional risk associated with serving two "masters." Little movement has been seen to date; however, with future emphasis on resource utilization and lower energy costs, there could well be a breakthrough in the near future.

Electric utilities are currently the only private entities which have the capability to license, finance, and operate commercial nuclear power facilities in the U.S. It is quite likely that this role will continue since most industries would prefer to see utility ownership and operation of central cogeneration facilities. However, the industrial firms do want to maintain some control over the operation of the power plant in areas such as scheduling shutdowns, etc. Since the utility and industry are likely to compromise, neither party will achieve the reliability and flexibility of a dedicated power plant. This is the very reason why the utilities are seeking a higher rate of return on cogeneration facilities.

In summary, the deployment of dedicated nuclear process heat plants or central cogeneration facilities will require resolution of many issues. There must be a move by government agencies at the federal, state, and local levels to establish uniform regulations and provide the appropriate incentives for cogeneration. The streamlining of the licensing process together with firmly established regulation and "one-stop" permitting could lead to large savings. The reliability of the proposed energy source must be well established and appropriate backup systems provided. The timing and financial arrangements must also be firmly defined before nuclear cogeneration projects can become a reality. Although it is nearly impossible to foresee such a project in today's regulatory and economic climate, major changes in policy are likely to occur if the current U.S. standard of living is to be maintained. As the HTGR-R is not to be considered for commercial deployment until after the turn of the century, it could well serve this evolving cogeneration market.

4.2.4 Licensing

This section will assess the various characteristics of the HTGR which affect 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.

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 as well as the Fort St. Vrain prototype. The Fulton and Summit applications were withdrawn prior to granting of the construction permits, but Summit had received a Limited Work Authorization.

The design characteristics which set the HTGR apart are discussed briefly below. First, the HTGR core is constructed exclusively of ceramic materials, primarily graphite, which maintains its 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 mins can be tolerated without any damage to primary system components.

Another inherent safety characteristic of the HTGR 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 ambient air, 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 PCRV, 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 PCRV are provided by a large number of steel tendons that run axially through and circumferentially around the vessel. The concrete acts as a radiation 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 sequences which might result in release of radioactivity from a large HTGR Steam Cycle plant. Unfortunately, no such study has yet been performed specifically for the HTGR reforming plant. However, the results of this study are for the most part applicable to the HTGR-R. A summary table of results of the AIPA is presented in Table 4.2.5-1

TABLE 4.2.5-1

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. 12

from Ref. 12 and is compared with the results of the WASH-1400 analysis of the LWR. Because the studies were performed by different groups, the uncertainties associated with the two analyses are not the same. 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, FRG, 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.

The HTGR-R adds consideration of the release of toxic and explosive gases and the resulting impact on the nuclear plant. By design, the reformer feedstock is a light hydrocarbon, e.g., methane, and the effluent consists of a mixture of carbon monoxide and hydrogen. As carbon monoxide is a toxic gas, its release cannot be allowed to impair the safe shutdown of the reactor. Similarly, the potential explosion resulting from the release and detonation of the hydrocarbon or hydrogen must also be addressed. The consequences of the maximum credible explosions must be designed for in such a manner that safe shutdown of the reactor is also not impaired. In addition, there are several generic licensing issues facing the HTGR.

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. 13. The major topics and their responses are reviewed below:

- Use of Graphite as a Structural Material - The design criteria for graphite structures has 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. The reference indirect cycle configurations would be less susceptible to water ingress as the steam generator is removed from the primary system.

- 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/1382°F helium environment. The reference fuel for the Lead Project is LEU (20% enriched). As a result, much experimentation remains to be done on LEU fuel particles and their properties, including fission product retention. As higher temperature applications are pursued, i.e., 850°C/1562°F 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 in^2 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 that 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.

- 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. Core auxiliary cooling system testing criteria still must be developed 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.

In summary, the HTGR has the potential to experience a given probability accident with lower consequences, which may be a significant advantage in the undefined nuclear licensing process of the future. There are, of course, uncertainties which must be addressed in the design, development, and licensing phases of the Program. There also remain the open licensing issues discussed above which must be resolved prior to commercial deployment. Not one of these issues appears, however, to preclude the licensing of a commercial HTGR reactor.

4.2.5 Capability to Meet Evolving Energy Requirements

The HTGR-R has the potential to supply the requirements of all four U.S. energy-use sectors: residential, commercial, industrial, and transportation. The current contribution of nuclear power is confined to electric utility use. In the U.S., electricity currently has a limited role in supplying the national energy needs and provides only a portion of the residential, commercial, and industrial requirements and virtually none of the transportation requirements. Although electricity's contribution is projected to increase in the future, it is not likely to have a major impact on the transportation sector unless electric-powered cars and mass transit systems are utilized to a great extent in the future. Even here, the nuclear contribution is limited to the baseload electric power generation and shares this market with fossil-powered units.

Electricity now supplies over 11% of the U.S. end-use energy requirements; its actual contribution is over seven quads (10^{15} BTU). This contribution is projected to more than double by 1995. In the longer term, nuclear might provide up to 50% of the total electrical energy input. Forecasts from the DOE (Ref. 14) show the electric contribution growing to approximately 23% of the end-use energy by the year 2020, or about 24 quads (Series C Projection). The conversion losses associated with electric power generation and distribution, however, would require nearly half of the total U.S. energy supply in 2020. Improved fuel utilization through use of cogeneration and higher system efficiencies could have important positive implications on the future U.S. energy picture.

In contrast to the future electric market, there are potential nuclear process heat contributions for high-temperature heat in the industrial and transportation (synthetic fuel production) sectors. The industrial market alone is projected to be nearly twice the electric market. Depending on timing, the commercial synthetic fuel market could develop during the development of the HTGR, allowing the HTGR to contribute to the transportation sector.

Through these applications, it is very possible that the HTGR could provide the energy requirements in these expanded markets and hence increase the contribution of nuclear power.

4.2.6 Siting

The coupling of the HTGR to energy storage and transmission systems provides added incentive for deployment of HTGR systems in terms of siting. If nuclear power is to provide a larger contribution to future U.S. energy requirements, the formation of large nuclear energy parks may well take place. The HTGR can greatly aid their formation. First, the HTGR with its higher temperature capability achieves greater energy utilization and hence rejects less heat. Many energy park sites can be limited as to capacity because of cooling water availability. Secondly, the HTGR coupled with an energy storage and transmission system is capable of transmitting a large percentage of the reactor output to a remote location, very likely an urban area. Here the energy can be utilized for electric power production and/or industrial process heat. The bulk of the waste heat can then be rejected at the remote power production sites, and power is produced without power plant effluents such as CO_2, SO_2, particulates, or radiation. In these applications, energy pipelines would displace requirements for overhead transmission lines, which are also becoming more and more difficult to deploy.

4.3 Commercial Plant National Incentives - HTGR-R

The HTGR-R has the capability to supply a large portion of the U.S. energy requirements in the form of electricity, steam, or direct heat. The high temperatures which can be achieved by the HTGR permit consideration of nuclear power for applications which have been historically based on fossil fuels. The potential market expansion for nuclear power, coupled with the perceived HTGR advantages in attaining both public and investor acceptance, could lead to a healthy political and social environment for deployment of nuclear power in the future. As the U.S. has painfully learned through past experience, undue reliance on any one energy resource can result in economic upheaval when loss of political, public, or user acceptance because of environmental effects, safety, cost, or availability of fuel affects the viability of that option. Therefore, the HTGR can provide a much needed alternative to coal in applications such as industrial process heat, synthetic fuel production, and baseload intermediate and peaking electric production. The HTGR becomes an even more important option as coal prices increase due to ever expanding demand and utilization, acting as an insurance policy against higher fuel costs.

4.3.1 Fossil Resource Conservation

The U.S. is moving rapidly to coal and away from oil and natural gas based on strategic and economic considerations. The present role of nuclear power is very much limited though recognized as necessary in the overall energy mix. The increased reliance on coal projected by the government could easily tax the U.S. capability to mine and transport coal. The environmental and health effects associated with the combustion and conversion of coal could further exacerbate the problems associated with its increased use in the U.S. The HTGR could provide an alternative which has reduced environmental impact while at the same time providing a hedge against increasing coal prices. As is the case today in baseload power plants, the coal or nuclear option is likely to be selected on a site-specific basis, taking into account fuel costs, environmental restraints, capital costs, operating and maintenance costs, and perceived risk. By making the nuclear alternative available in this large new market segment, conservation of vital fossil resources can be attained. With commercial products such as drugs, fertilizers, fabrics, etc. being made from hydrocarbon resources, there may well be a point in time when it is realized that fossil fuels should no longer be combusted in large boilers to provide energy. In this manner, a vital resource can be conserved for the use of future generations. Deployment of the HTGR-R could drastically reduce the drain on U.S. hydrocarbon resources and at the same time diversify the U.S. energy supply situation.

4.3.1.1 Oil and Natural Gas

In recent years, the U.S. has become increasingly aware of the limited availability of clean fossil fuels. There have been numerous studies to predict resources of oil and natural gas. While there is considerable uncertainty as to the range of validity of different methods for estimating undiscovered natural resources, there is a general agreement among all such studies that the projected overall energy demand exceeds the anticipated fossil energy supplies, and even the most optimistic projections concede that clean fossil fuels will provide an ever-decreasing fraction of the nation's needs.

These two conclusions appear to be true not only for the U.S. but for the rest of the world as well. These problems were demonstrated most dramatically by the oil embargo a few years ago and by the natural gas shortages in the winter of 1976-1977. The response to this projected energy deficit has been aimed at a variety of goals consisting of an increased energy production (exploration, recovery of oil, harvesting newer sources of energy), conservation, and fuel substitution.

A review of past energy consumption patterns shows that approximately 75% of the total energy needs in the U.S. are met by the use of petroleum products and natural gas (Ref. 14). The only possible way in which these needs can continue to be met by prime fossil fuels is through massive imports. Either for balance of trade or security reasons, this "heavy foreign imports" scenario is generally judged neither desirable nor viable in the long run. The only other alternative for the near future, apart from a drastic reduction in energy consumption, appears to be a switch to alternative sources of energy consisting predominantly of coal and nuclear power. Solar, wind, and geothermal sources will play a larger role when they become technically available and economically attractive. Since there are considerable differences in the ease of substitution among the various uses of prime fuels, an examination of the nature of problems associated with this shift in energy sources from oil and gas to coal and nuclear power is in order.

After the transportation sector, the next largest use of oil and gas is in the industrial sector (Ref. 14). While some of the industrial use is in the form of chemical feedstocks and, therefore, is very difficult to directly replace, the rest is consumed as fuel to provide either steam or process heat.

The use of oil and gas in the residential and commercial sectors is predominantly for providing space heat and hot water. In principle, any source of low-grade heat can be substituted for this usage of prime fuel. The potential benefits of using power plant reject heat for this purpose are increasingly recognized in Europe. Whether or not district heating concepts are equally viable in this country is a matter of debate. At present, these consumers are considering primarily oil (domestic or imported), coal-derived clean fuels, heat pumps, or low-temperature solar heat for their future needs.

The electric utilities also rely on oil and natural gas primarily for the generation of peak and intermediate electricity. The rising fuel costs have led them to an increasingly intensive search for alternatives. The two main approaches to obviate the need for oil and gas are load management by regulation or pricing policies and the use of energy storage to deliver intermediate and peak electricity from baseload generation.

The HTGR-R can directly displace these current uses of oil and natural gas in energy storage and distribution applications which produce intermediate/peaking electricity and/or process heat and steam. The eventual coupling to district heating systems holds even greater potential for displacement of these prime fuels.

4.3.1.2 Coal

The forecast of dwindling domestic supplies of oil and natural gas, coupled with the prospect of exorbitant prices for foreign oil and gas, has focused a great deal of attention on the largest U.S. energy resource, coal. Coal has been considered a rather "dirty" fuel, and the availability of relatively inexpensive and clean-burning oil and natural gas has displaced its use in many instances. This displacement resulted primarily from environmental restrictions placed by evolving state, local, and federal regulations as well as convenience and economics. The growing cost of oil and natural gas is now justifying retrofit of older plants with the appropriate environmental protection equipment to burn coal. In addition, coal is sought to be a source of clean-burning synthetic fluids and gases through coal liquefaction and gasification processes. The HTGR-R, coupled to synthetic fuel processes, can increase the yield of liquid and gaseous products per ton of feed coal, thus stretching the availability of this valuable hydrocarbon resource.

Certainly, coal will play a large part in the overall U.S. and world energy supply picture in the 21st century. However, to achieve this enlarged role, a dramatic expansion of U.S. coal mining and transportation systems must take place. The HTGR is projected to displace 20-40% of the coal feedstock (Ref. 4) in synthetic fuel applications and can directly displace coal requirements in utility or industrial power generation roles. The corresponding reduction in coal demand which follows would substantially lessen the burden on coal mining and transportation. Of course, the HTGR is also projected to have significantly lower environmental impact than coal for a given application, and this is discussed further in Section 4.3.3.

4.3.2 Uranium Conservation

Although coal is the largest currently available energy resource in the U.S., the deployment of breeder reactors and recycle technology greatly expands the energy equivalent of our uranium resources. Tables 4.3.2-1 and 4.3.2-2 depict the recoverable reserves and resources of conventional mineral fuels in the U.S. and the world respectively. As uranium is utilized only for energy generation while coal can be a valuable hydrocarbon resource, the significance of pursuing the nuclear power option is clear. The HTGR provides a vehicle to penetrate

TABLE 4.3.2-1

U.S. RECOVERABLE RESERVES AND RESOURCES
OF CONVENTIONAL MINERAL FUELS

Fuel	Identified		Undiscovered hypothetical resources	Total	Quads of Btu equivalent*
	Reserves	Inferred resources			
Coal (billion short tons)	260	648	895	1,803	37,863
Oil (billion barrels)	34	23	82	139	806
Natural gas liquids (billion barrels)	6	6	16	28	115
Gas (trillion cubic feet)	209	202	484	895	917
Uranium (thousand short tons)	890	1,395	1,515	3,800	1,140 (LWR) 68,400 (FBR)
Total (quadrillion Btu)	6,163	14,391	20,287		40,841 (LWR) 108,101 (FBR)

Note: The adjective "recoverable" indicates that the estimates refer to how much material may possibly be recovered and not to how much material is in place However, the intention is to go beyond what could be recovered with today's technological and economic means. The coal estimates include thin beds, which cannot be recovered economically at the present time; and the uranium estimates include materials recoverable only at costs above present market prices. Although the hydrocarbon estimates refer to quantities that could be recovered economically today, improved conditions are unlikely to affect them substantially. LWR = Light Water Reactor; FBR = Fast Breeder Reactor.

*One quad is 10^{15} Btu Heat equivalents used here are: coal, 21 million Btu per short ton; oil, 5.8 million Btu per barrel; NGL, 4.1 million Btu per barrel; gas, 1.025 Btu per cubic foot; uranium, 300 billion Btu per short ton (if used in LWRs) or 18 trillion Btu per short ton (if used in fast breeders).

Ref. 15

TABLE 4.3.2-2

WORLD RECOVERABLE RESERVES AND RESOURCES
OF CONVENTIONAL MINERAL FUELS

Region or nation	Coal (billion metric tons coal equivalent)		Oil (billion barrels)		Gas (trillion cubic feet)		Uranium (thousand metric tons U)	
	Reserves	Resources	Reserves	Resources	Reserves	Resources	Reserves	Resources
United States	178	1,285	29	110-185	205	730-1,070	643	1,696
Canada	9	57	6	25-40	59	230-380	182	838
Mexico	1	3	16	145-215	32	350-480	5	7
South and Central America	10	14	26	80-120	81	800-900	60	74
Western Europe	91	215	24	50-70	143	500	87	487
Africa	34	87	58	100-150	186	1,000	572	772
Middle East	—	—	370	710-1,000	731	1,750	—	—
Asia and Pacific	40	41	18	90-140	89	—	45	69
Australia	27	132	2	—	31	500	296	345
Soviet Union	110	2,430	71	140-200	910	2,850	n.a.	n.a.
China	99	719	20	—	25	—	n.a.	n.a.
Other Communist areas	37	80	3	—	10	—	n.a.	n.a.
Total	636	5,063	642	1,450-2,120	2,502	8,710-9,430	1,894	4,288
Quintillion (10^{18}) Btu	17.7	140.6	3.7	8.4-12.3	2.6	8.9-9.7	7.4(LWR) 443.2(FBR)	16.7(LWR) 1,003.4(FBR)

Note: All resource figures are cumulative. They include reserves. The figures for international coal reserves and resources in this table are given in metric tons of fixed heat content rather than actual metric tons, while the domestic coal reserves and resources are given in actual tons with an average heat content of 21 million Btu per ton. Similarly, 300 million Btu per short ton U_3O_8 corresponds to 390 million Btu per metric ton U in this table.

energy markets other than baseload electric power production and expands our energy resources even further through deployment of the uranium/thorium fuel cycle.

4.3.2.1 Uranium/Thorium Fuel Cycle

Two basic fuel cycles have generally been considered for thermal-spectrum reactors: the low-enrichment uranium (LEU) cycle and the LEU/thorium cycle (LEU/Th). While the LEU cycle has traditionally appeared more attractive for LWR plants, the LEU/Th 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 the comprehensive NASAP study recently completed by DOE.

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.

4.3.2.2 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 4.3.2-3 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

TABLE 4.3.2-3

U_3O_8 REQUIREMENTS AND Pu_f DISCHARGE
FOR ALTERNATIVE FUEL CYCLES IN LWR AND HTGR PLANTS*
(LOAD FACTOR = 70%; ENRICHMENT TAILS = 0.1%)
(Basis: Ref. 7)

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%.

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 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 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.

4.3.2.3 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 (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 <u>not</u> 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.

4.3.2.4 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 denatured 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.

4.3.2.5 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 fuels 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.

4.3.3 Reduced Environmental Impact

The HTGR-R provides a much-needed nuclear alternative in industrial process heat applications. Currently, the use of oil and gas predominates in industry, but the cost and availability of these fuels, together with government regulations, will force selection of alternative fuels. Coal is expected to be that alternative. The abundance of coal and its present cost favor the transition toward coal in the future. However, this transition may have pitfalls. Substantial increases in the use of coal may cause price escalation due to mining, transportation, and conversion. The availability of a nuclear process heat option provides an insurance policy against outlandish increases in the cost of coal. Of equal or greater importance is the corresponding reduction in effluents achieved with deployment of the HTGR-R. Not only are air pollutants associated with combustion of coal (SO_2, NO_x, CO, hydrocarbons) reduced or eliminated, but solid wastes are also substantially reduced. The availability of the HTGR could, therefore, have a very profound effect on the U.S. capability to maintain its standard of living without degradation of the environment.

4.3.3.1 Air Pollution

The quality of air in the U.S. has come under increased private and government scrutiny during the 1970s, leading to a rash of regulations at the federal, state, and local level designed to ensure our air quality. The primary source of air quality degradation has been the combustion of fossil fuels. Fig. 4.1.1-2 indicated the tremendous contribution of fossil fuels to total U.S. energy consumption. Therefore, in order to maintain air quality, the U.S. has the following alternatives:

1. Deploy a "clean-burning" substitute energy source,

2. Develop improved methods for fossil fuel combustion which reduce emissions, or

3. Accommodate a reduction in U.S. energy use.

The first solution is by far the most desirable, but the search for an emission-free economical energy source has not been easy. Nuclear power can eventually displace a large portion of fossil fuels in electric power generation, but current nuclear technologies do little for energy requirements in the industrial, transportation, residential, or commercial energy use sectors. However, the HTGR-R is capable of delivering emission-free energy at the point of use for industrial application. The second solution is the one most often pursued in recent years. However, each level of emission reduction has typically been achieved at higher and higher cost. The effect of diminishing returns will result in marginal reductions in emissions and higher fossil energy costs. A total reduction in point-of-use emissions will require the elimination of fossil fuel combustion. That leads to the third solution, which many experts feel could drastically change the U.S. standard of living.

It is important to summarize the current regulations affecting air quality in order to provide a perspective as to their impact on fossil fuel combustion in the future. Industry, in particular, has been greatly affected by the Clean Air Act and its amendments, Environmental Protection Agency (EPA) regulations, the states, and air quality regions.

EPA is required to establish National Ambient Air Quality Standards (NAAQS) and has primary enforcement authority. States, however, are required to develop implementation plans for attainment and maintenance of the ambient standards, for EPA's approval. Among the aspects of attainment and maintenance of NAAQS which vitally concern industrial plants are nonattainment areas, prevention of significant deterioration, the Offset Policy, new source performance standards, national emission standards for hazardous air pollutants, best available control technology, lowest achievable emission rate, and reasonably available control technology.

In 1975, as a statutory deadline for attainment approached, at least 160 of the nation's 247 air quality control regions had monitored violations. A strict interpretation of the law would have prohibited new sources from locating in any area which had failed to attain the ambient standard for the pollutant or pollutants it would emit. EPA was forced in 1976 to develop a procedure for permitting growth--the Offset Policy. Its essence is that major new growth is permitted in nonattainment areas only if air quality is <u>improved</u> as a result of that growth. The impact on industrial growth in nonattainment areas is tremendous.

The Offset Policy requires major new or modified sources seeking permits to expand in and around nonattainment areas to reduce emissions to the lowest achievable emission rate (LAER); to certify that all sources which it owns or controls in the same state are in compliance; to obtain emission reductions from existing area sources to more than offset the pollution to be added by the new LAER-controlled source; and to demonstrate that a net air quality benefit in the affected area will result.

The Clean Air Act requires that each State Implementation Plan include a comprehensive, accurate, current inventory of actual emissions from all sources in nonattainment areas. The purpose is to quantify the origins of the nonattainment problem and, by periodically revising and updating the inventory, to ensure that reasonable further progress toward attainment is demonstrated. To achieve an "attainment inventory" may require very substantial reductions in the "baseline inventory" associated with nonattainment. State Implementation Plans must include "reasonable further progress schedules" and provisions for new source review to ensure that the NAAQS will be met on time--by 1982 to 1987.

Nonattainment provisions are intended to achieve and then to maintain ambient standards. Other parts of the Clean Air Act are intended to prevent significant deterioration of air already cleaner than the ambient standards. The prevention of significant deterioration regulations originally prohibited construction of stationary sources in many of 19 specified categories unless EPA, or a state to whom responsibility had been delegated, issued a permit evidencing that the source would apply best available control technology for SO_x and particulates and that emissions of these pollutants would not cause significant deterioration of clean air. The 1977 amendments are along the same line but more comprehensive and restrictive.

Without delving further into the many complex aspects of air quality standards and regulations and their trends, it is safe to say with regard to the future that existing regulations will continue to make industrial growth very difficult and expensive even without a shift from oil to gas to coal. As industry is forced by regulation or by cost and scarcity to switch from burning oil to burning coal, prevention of significant deterioration of air quality would require expensive new facilities, stack gas cleanup, very careful handling of coal, and acceptable provisions for handling and disposing of fly ash, grate ash, and sludge from desulfurizing, which contain hazardous materials.

At present, stringent federal emission-control requirements are not imposed on small industrial plants. As attainment deadlines approach and emission inventories must be reduced, the expectation must be that even small plants cannot escape.

Clean fuels of the future might include coal-derived synthetics or hydrogen. While coal-derived organic fuels presumably would contain no ash and negligible amounts of sulfur, the combustion process would require careful control to ensure that emissions of nitrogen oxides, carbon monoxide, and hydrocarbons are within allowable limits. Hydrogen, often called the cleanest of fuels, produces oxides of nitrogen when combusted in air at boiler temperatures. Closed-loop methanation of TCP gas would produce no pollutant emissions.

Table 4.3.3-1 provides a summary of environmental effects of various process heat sources delivering 1000 MW(t). It can be seen that substantial reductions in daily emissions can be achieved at the point of use through remote energy delivery. A similar result is shown for HTGR-powered synthetic fuel plants in Table 4.3.3-2. Based upon TRW's market penetration assumption, the cumulative environmental benefit associated with HTGR synfuels plants is included in Table 4.3.3-3.

The HTGR-R has the capability to provide clean energy in urban and industrialized areas of the country through the transmission and delivery of thermal energy from remote reactor sites. The benefits of such a system are two-fold. First, the reduction in effluents in urban industrial areas associated with the HTGR-R can permit siting of expanded industrial capacity without further degradation of air quality. It is these very urban areas which are experiencing the greatest deterioration of air quality.

TABLE 4.3.3-1

ENVIRONMENTAL EFFECTS OF 1000-MW(t) PROCESS HEAT SOURCES

(Basis: Ref. 4)

	Natural Gas	Residual Oil	Coal	TCP
Present Use for Process Heat (percent)	43	22	35	
Projected Future Use	sharply down	down	up	
Emissions (tons/day)				
Particulates	0.45	0.16	2.3	--
SO_x	0.02	22.10	34.2	--
NO_x	6.11	9.18	28.5	--
CO	0.01	7.90	1.05	--
CO_2	4425	4742	6300	--
Hydrocarbons	1.16	0.98	281	--
Aldehydes	0.09	0.20	--	--
Organics	--	--	0.003	--
Heavy Metals	--	4.11	0.2	--
Radioactivity (m Ci/day)	--	0.55	3.67	? (a)
Solid Wastes (tons/day)				
Bottom Ash	--	1.48	59.3	--
FGD Scrubber Sludge	--	--	816	--
Fly Ash Retained	--	--	235	--

(a) Assumed to be very low.

TABLE 4.3.3-2

HTGR REDUCTION IN EFFLUENTS OVER NON-NUCLEAR-POWERED SYNFUELS PLANT - SINGLE PLANT BASIS

(Basis: Ref. 6)

PROCESS	DIRECT BASIS: PRODUCT OUTPUT	NET REDUCTION IN EFFLUENTS (TONS/YR)				
		SO_2	NO_x	SOLID WASTE (ASH)	CO	HC
Methanol from Coal	58,300 tons/day (Reference 3-1)	30,500	12,500	136,750	N/A	N/A
Ethanol from Biomass	50×10^6 gallons/yr (Reference 6-2)	612.5	340.4	9189.2	54.1	16.1
Oil from Shale	254,000 Bbl/day (Reference 1-1)	757	3953	4,164,000	803	284
Direct Liquefaction	186,000 Bbl/day (Reference 1-1)	29,857	11,200	227,000	N/A	N/A
Indirect Coal Liquefaction (Mobil M)	177,800 Bbl/day gasoline +30,800 Bbl/day LPG (References 1-1, 3-2)	15	N/A	154,750	1245	114
High-Btu Gasification	540×10^6 scf/day (Reference 1-1)	83,000	139,300	7.4×10^6	N/A	N/A
Reference emissions from a 500 MWe	8.2×10^{12} Btu/yr energy production	45,649	9,405	89,191	525	156
Conventional Eastern Coal Boiler (Reference 3-3)	Same boiler meeting revised NSPS	5,945	3,305	89,191	525	156

TABLE 4.3.3-3

CUMULATIVE HTGR-SYNFUELS ENVIRONMENTAL IMPACTS*

(Basis: Ref. 6)

	TONS OF EMISSION AVOIDED			
	2015		2020	
	NOMINAL	PESSIMISTIC	NOMINAL	PESSIMISTIC
OIL SHALE				
SO_2	757	0	6,056	0
NO_x	3,953	0	31,624	0
Solid Waste	4.164×10^6	0	33.312×10^6	0
CO	803	0	6,424	0
HC	284	0	2,272	0
DIRECT LIQUEFACTION				
SO_2	29,857	0	298,570	29,857
NO_x	11,200	0	112,000	11,200
Solid Waste	0.277×10^6	0	2.27×10^6	0.227×10^6
CO	NA	0	NA	NA
HC	NA	0	NA	NA
INDIRECT LIQUEFACTION				
SO_2	30	0	450	15
NO_x	NA	NA	NA	NA
Solid Waste	0.309×10^6	0	2.321×10^6	0.155×10^6
CO	2,490	0	18,675	1,245
HC	228	0	1,710	114
HIGH BTU GAS				
SO_2	83,000	0	664,000	0
NO_x	139,300	0	1,114,400	0
Solid Waste	(7.4×10^6)	0	59.2×10^6	0
CO	NA	NA	NA	NA
HC	NA	NA	NA	NA
TOTAL				
SO_2	113,644	0	969,076	29,872
NO_x	154,453	0	1,258,024	11,200
Solid Waste	12.1×10^6	0	97.103×10^6	0.382×10^6
CO	3293	0	25,101	1,245
HC	512	0	3,982	114

*Assumes all plants operational only in latest year of period shown.

4.3.3.2 Radioactive Wastes

Although little has been determined regarding the HTGR-R plant radioactive effluents to date, information can be extrapolated from past steam cycle designs. In general, the amount of gaseous, liquid, and solid radioactive waste is expected to be lower than for comparable LWRs of the same power rating. Among the conditions noted in Ref. 16 are the reduction in liquid and gaseous wastes, lower dilution requirements for gaseous (100 times) and liquid (10 times) wastes, the potential for large quantities of solid waste in the form of spent fuel and reflector blocks, and the need for consideration of C-14 disposal. The plant design will meet current regulations regarding releases and disposal of radioactive material and has the capability to meet even tighter regulations should they evolve. Coupled with the remote siting capability, radioactive wastes are not expected to be a concern.

4.3.3.3 Solid Wastes

Tables 4.3.3-1, 4.3.3-2, and 4.3.3-3 also depict the savings in solid waste disposal associated with deployment of the HTGR-R in TCP or synthetic applications. These comparisons do not include any assessment of solid waste disposal associated with mining coal or uranium/thorium. As the nation turns toward expanded combustion of coal in the U.S. for electric power production, industrial process heat, or synthetic fuel production, the ash disposal problem will become exacerbated. There is growing concern over the content of this waste, which contains carcinogens and radioactive materials, leading to possible treatment in the future as a hazardous waste. The deployment of the HTGR-R system can reduce or eliminate this potential problem, which can be especially important to small industrial users where the expense of ash disposal could become a tremendous burden.

4.3.4 International Relations/U.S. Policy

Active HTGR design and development programs are currently in progress in many foreign countries. These programs are discussed in Section 5.4.2. The existence of these programs provides some indication of the anticipated international potential for the HTGR. While there is interest in the electric power applications of the HTGR, there is unanimous support for process heat applications. The selection of the HTGR-R Lead Project could help establish the U.S. as the leader in advanced nuclear technologies, a role which has not been enjoyed for several years.

Although the U.S. is in a far better position to achieve energy self-sufficiency than many other nations, an understanding of the energy problems facing other nations is of paramount importance. Lacking the immense coal resources of this country, many European and Asian countries are pursuing nuclear power more intensely than the U.S. The application of nuclear power in the past has been limited to electric power generation. The HTGR is envisioned to be one nuclear system which can expand the role of nuclear power to provide heat and energy for industrial processes. The development of the HTGR could, therefore, have many favorable impacts on overall energy consumption in the world.

As other countries are actively pursuing this technology, it is in the best interest of the U.S. to encourage technical exchange and/or joint projects in reactor development and demonstration. It is clear that the major ingredients for the successful transfer of foreign technology include a commitment in the U.S. to the construction of a demonstration plant and a commitment of U.S. technical expertise. And of even greater importance, the commitment to an HTGR-R Lead Project will help reassert the intent of the U.S. to be an international leader in the development of peaceful nuclear energy and to preserve U.S. influence in world nuclear safeguards.

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5.0 HTGR LEAD PROJECT AND PROGRAM

The HTGR-R was selected as one of four Lead Project options to be considered under the HTGR Lead Project Identification Plan. At the time the plan was formulated, the cumulative level of HTGR-R design and development was at a much earlier stage than either the HTGR-SC/C or HTGR-GT options. In order to meet the ambitious schedule set forth in the HTGR Lead Project Identification Plan, several assumptions and ground rules were established to facilitate completion of the HTGR-R plant definition and cost estimate. The five-month period allotted for plant design did not permit the execution of tradeoff studies or design optimization. In order to better understand the evolution of the HTGR-R lead plant design effort, the following background information is provided.

The HTGR-R plant was initially identified as an 1170-MW(t) indirect cycle plant with a core outlet temperature of 850°C. The core thermal rating was selected because it was perceived to be within the projected commercial size range and for commonality with past 1170-MW(t) HTGR-SC/C designs. The core outlet temperature of 850°C was selected as the highest temperature within reason for a Lead Project targeted for operation before the turn of the century. The indirect cycle is the current reference, although the direct cycle is still under study. While the decision on lead and commercial plant configuration is slated for FY 1981, the indirect cycle was selected for the current evaluation on the basis of reduced licensing risk, which is particularly desirable for a Lead Project.

As the design of the HTGR-R evolved, the split of energy delivered to the reformer and steam generators was initially established as 42% reforming, 58% steam generation. This energy split was carried as a design basis and is still believed to be reasonable. However, it should be recognized that this energy split has not been optimized and may be different for the various different applications such as synthetic fuels or remote energy delivery via the thermochemical pipeline. Four primary loops were selected to optimize PCRV packing and to maintain IHX thermal rating within reason. The concept of four totally independent and separated loops was continued in the secondary system and process plant. Two reformers and one steam generator per secondary loop, a total of eight and four respectively per 1170-MW(t) plant, are enclosed in one prestressed concrete pressure vessel. The number of reformers was selected based upon size limitations imposed by fabrication and shipping requirements. The following plant design requirements were also established:

1. Plant capability to operate at reduced power with fewer than four loops.

2. Maintenance and inspection of reformer or steam generator can be accomplished with that loop out of service and other loops at power.

3. The steam plant and reformer plants are to remain independent such that the loss of one does not require shutdown of the other.

4. Plant capability for electric power operation (only) with reduced core power level.

5. Plant capability for reformer operation without turbine generator operation.

The application selected for the HTGR-R evaluation utilizes the thermochemical pipeline concept to deliver energy for industrial process heat or for utility load-following electric power generation. The plant design up to the pipeline was identical for both applications. The utility application was selected for consideration as fewer institutional obstacles are faced in a non-industrial application. Electricity as an initial demonstration plant application is further indicated when one considers that the reliability of the system must be established before industry could consider the thermochemical pipeline concept as a viable alternative energy source. The HTGR-R Lead Project might also be applied to repowering of an existing oil- or natural-gas-powered facility with methanators. Although this concept is not specifically addressed, the associated design and cost information may be extracted from the contents of this report.

While the resulting HTGR-R Lead Project design reflects a non-optimized design in what may prove to be a less-than-optimum application, the FY 1980 design effort provides an excellent basis for future work. The following sections describe the plant, the associated design and development program, and the technical issues which must be addressed in order to implement the HTGR-R Lead Project.

5.1 Lead Plant Technical Definition

5.1.1 Nuclear Heat Source

This section is a brief description of the HTGR-R nuclear heat source (NHS). The NHS is capable of providing 1170 MW(t) via a high-temperature high-pressure helium loop to a compatible reforming/steam generation process.

The NHS (see Fig. 3.1.1-1) is characterized by a graphite-moderated, helium-cooled thermal reactor core, which is located within a PCRV. The NHS consists of five major systems and a number of support systems. The five major systems are the PCRV, reactor internals, reactor core, primary cooling system, and the core auxiliary cooling system (CACS).

The NHS scope extends to the secondary helium inlet and outlet nozzles of the intermediate heat exchanger (IHX) and to the cooling water inlet and outlet nozzles of the CACS core auxiliary heat exchanger (CAHE). A brief description, including the functions of the major systems, is provided below. The major functions for most of the other NHS systems are also included.

5.1.1.1 Prestressed Concrete Reactor Vessel

The PCRV system includes the PCRV structure; the cavity liners, penetrations, and closures; the thermal barrier; and a pressure relief system. The overall functions of the PCRV system are to provide the primary reactor coolant pressure boundary, to house the NHS components, and to provide a biological shield around the reactor. Plan and elevation views of the NHS PCRV system are shown in Figs. 5.1.1-1 and 5.1.1-2, respectively.

The PCRV structure is a multicavity vessel of prestressed concrete characterized by a central core cavity and peripheral cavities, which house the primary cooling system components and the CACS components. The vessel is prestressed circumferentially by wound strand cables and vertically by linear strand tendons. 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, the closures at the penetrations, and the IHX supports form the continuous gas-tight primary coolant boundary of the PCRV. The liner and penetration anchors transmit loads from the internal equipment supports to the concrete structure. The liners are cooled by circulating water in tubes attached to the liners at their interfaces with the concrete.

The thermal barrier minimizes heat losses from the primary coolant and maintains the liner and concrete temperatures within acceptable limits. Different types of thermal barrier are used in the various cavities and

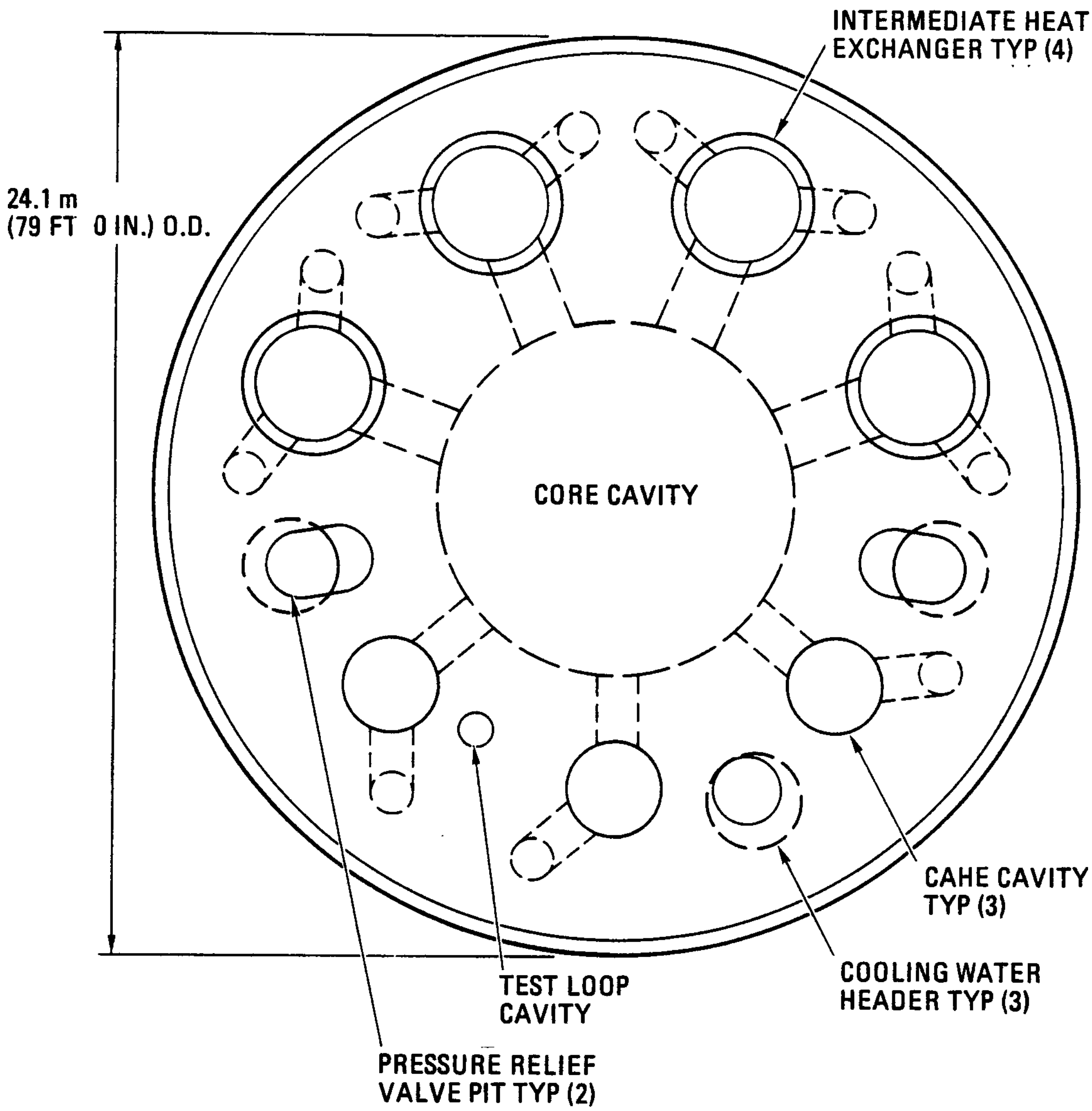

Figure 5.1.1-1 HTGR-R PCRV plan view

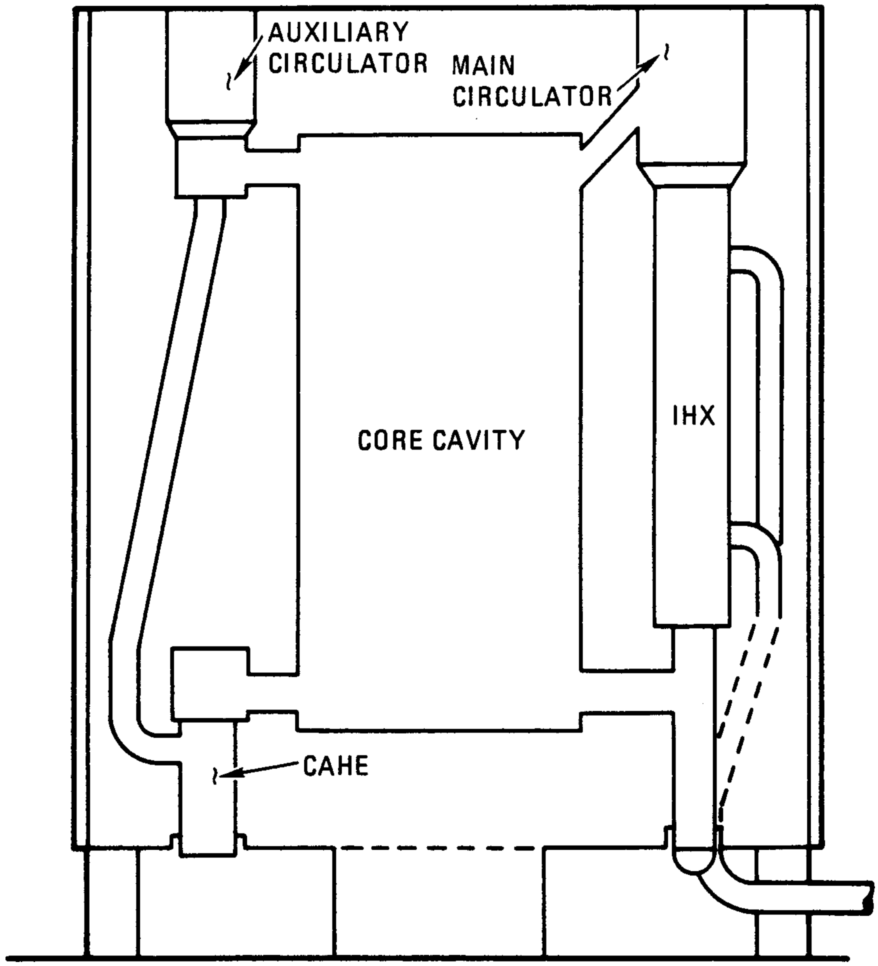

Figure 5.1.1-2 HTGR-R PCRV elevation view

TABLE 5.1.1-1
HTGR-R PCRV MAJOR PARAMETERS

Type	Multicavity PCRV
Overall dimensions	
Diameter, m (ft)	24.08 (79.00)
Height, m (ft)	28.04 (92.00)
Core cavity	
Quantity	1
Diameter, m (ft)	9.60 (31.50)
Height, including in-vessel refueling, m (ft)	20.65 (67.75)
Intermediate heat exchanger cavity	
Quantity	4
Diameter, m (ft)	
Mid-height	3.00 (9.83)
Top	3.63 (11.92)
Height (above the centerline of hot duct to top of PCRV), m (ft)	23.55 (77.25)
CAHE cavity	
Quantity	3
Circulator cavity diameter, m (ft)	
Mid-height	2.03 (6.67)
Top	2.44 (8.00)
CAHE bundle cavity diameter, m (ft)	
Mid-height	1.83 (6.00)
Bottom	2.51 (8.25)
Maximum cavity pressure, MPa (psig)	5.35 (776)
PCRV support	Ring support with central core

zones within the PCRV, depending primarily upon the local gas temperatures. Typically, the thermal barrier consists of layers of fibrous insulation material held against the liner by metal cover plates. These plates are anchored to the liner via attachment fixtures, which are designed to minimize the thermal conduction to the liners.

A pressure relief system limits the pressure of the primary coolant in the PCRV to a specified safe value and limits the rate of pressure relief flow from the PCRV. The design allowable maximum pressure is given in Table 5.1.1-1.

5.1.1.2 Reactor Internals System

The reactor internals system consists of five major components: the core support floor structure, the core lateral restraint, the permanent side reflector, the core peripheral seal, and the upper plenum structures of the in-vessel refueling system. Each of these major components and their specific functions is described below.

The core support floor consists of graphite core support blocks supported by graphite posts, which in turn are supported on graphite seats atop ceramic bases on the PCRV bottom head. The major function of the core support floor is to provide vertical support for the reactor core and reflector elements, while also serving as the plenum into which primary coolant is discharged and mixed before passing out of the core cavity. The core support floor also provides a seat for the peripheral seal.

The core lateral restraint is composed of metal support assemblies located in a regular array between the permanent side reflector and PCRV liner; it also includes the neutron side shield. The primary function of this component is to provide lateral support for the reactor core, the support floor structure, and the reflectors.

The permanent side reflector consists of columns of stacked graphite blocks that form a cylinder surrounding the hexagonal reflector columns of the reactor core. The primary functions of this component are to reflect neutrons back into the core and to attenuate the neutron flux to surrounding components of the internals and PCRV.

The core peripheral seal is formed by triangular cross section graphite logs, which fit in the annular space between the core support floor and the thermal barrier. A sloping shelf in the outer face of each peripheral core support floor block provides the inner seal seat. The outer seat is provided by a metal structure, which is supported by cantilever beams from the PCRV liner enclosed within the thermal barrier. The primary function of the seal is to limit the coolant flow that bypasses the reactor core and flows between the permanent side reflector and the PCRV liner.

The upper plenum structures of the in-vessel refueling system are steel structures. These are 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 elevator in the PCRV side wall.

5.1.1.3 Reactor Core System

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

The fuel elements and hexagonal reflector elements are arranged in columns supported on the core support blocks. Each support block under the major portion of the active core corresponds to one refueling region, which has a central control column and six surrounding fuel columns. Each refueling region (Fig. 5.1.1-3) has its own flow control valve. The fuel regions are surrounded by two rows of hexagonal reflector columns which are, in turn, surrounded by the permanent side reflector. All elements in a fuel region are loaded and unloaded at the same time.

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

Table 5.1.1-2 summarizes the major reactor core system design parameters, and Fig. 5.1.1-4 shows the core layout.

The fuel element (Fig. 5.1.1-5) is a graphite block with the dual function of containing the fuel and acting as a moderator. Each fuel element consists of a hexagonal graphite block containing drilled coolant passages and parallel fuel channels into which individual fuel rods are inserted. The individual fuel rods contain fuel particles distributed in a graphite matrix. Control rods and reserve shutdown holes are included in the central fuel column of a regicn.

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 material in a suitable container, are inserted into the core fuel elements to provide a source of neutrons of sufficient strength to ensure a safe, controlled approach to reactor criticality.

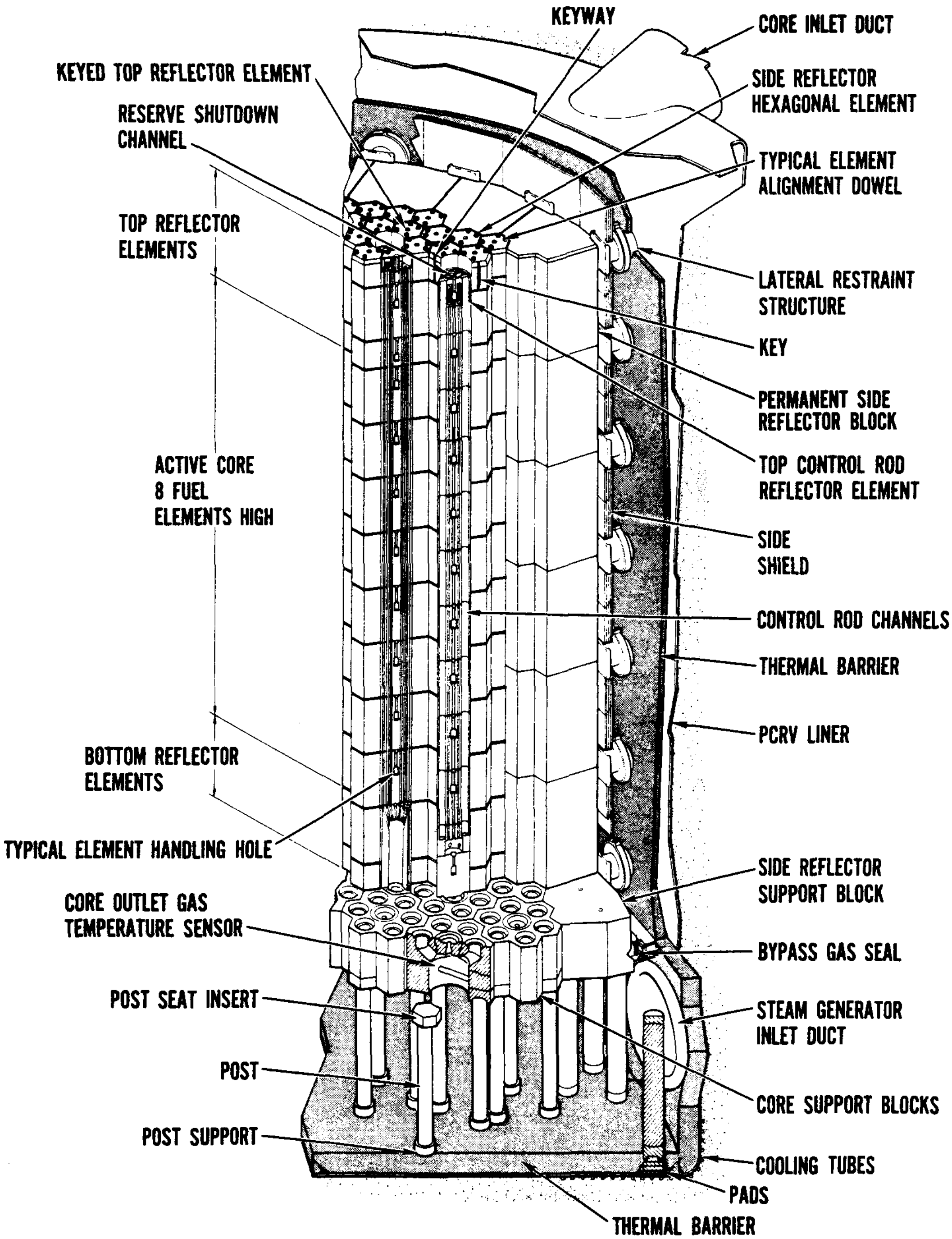

Figure 5.1.1-3 Core and support arrangement

TABLE 5.1.1-2
HTGR-R BASIC CORE PARAMETERS

Nominal core power, MW(t)	1170
Nominal core power density, W/cm^3	6.6
Number of fuel blocks per column	8
Number of fuel columns	247
Number of seven-column regions with variable orifices	31
Number of five-column regions with variable orifices	6
Number of control rod pairs	37
Number of power rods	37
Number of reserve shutdown hoppers	33
Core volume, cm^3 (ft^3)	1.77×10^8 (6250.7)
Fuel cycle	
Initial fuel cycle	LEU/Th
Refueling cycle	3-year cycle, 33% reloaded each year ("thick buffer")
Fissile material/particle	UC_2/TRISO
Fertile material/particle	ThO_2/TRISO
Fuel enrichment, % U-235	19.9%

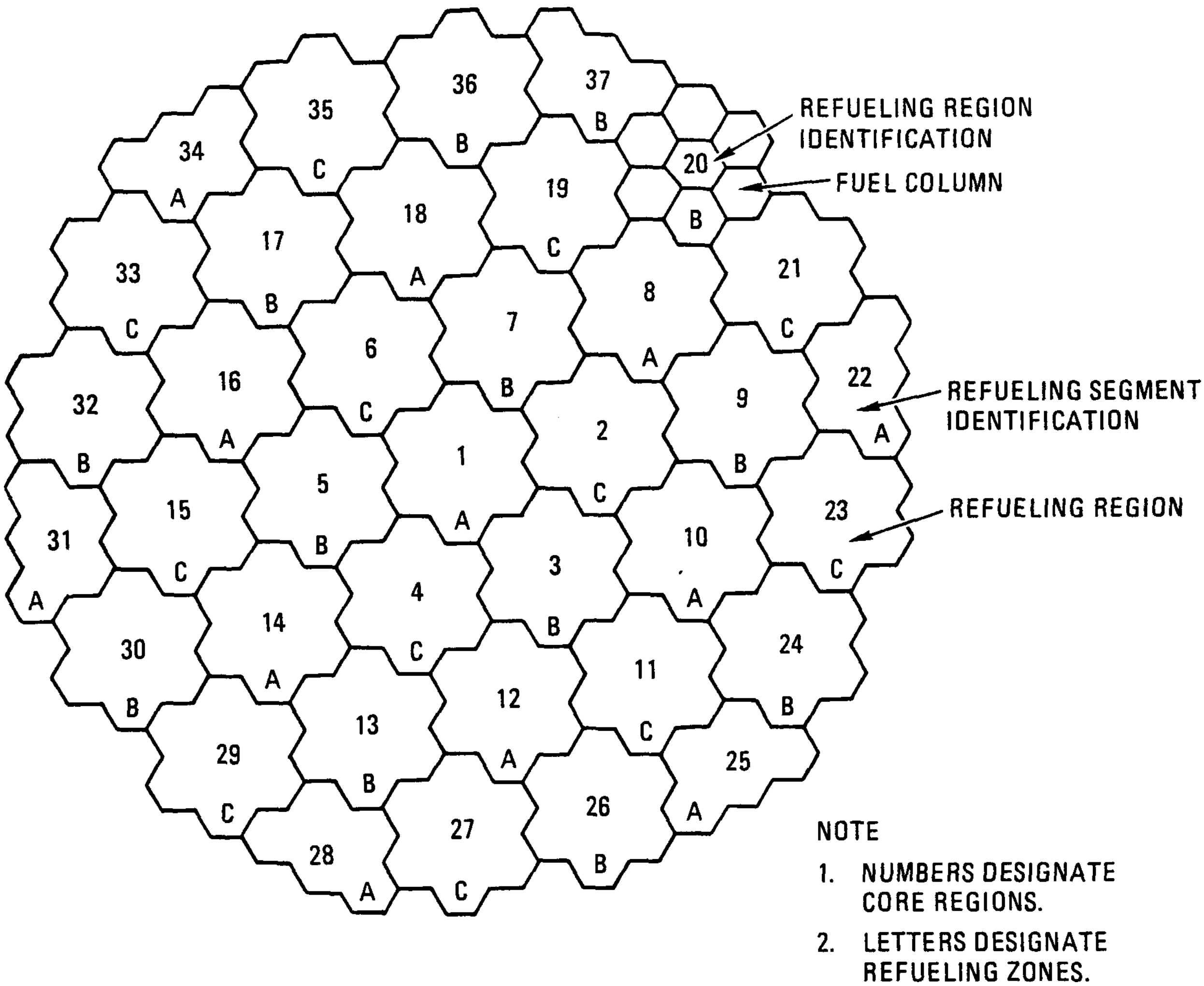

Figure 5.1.1-4 HTGR core layout

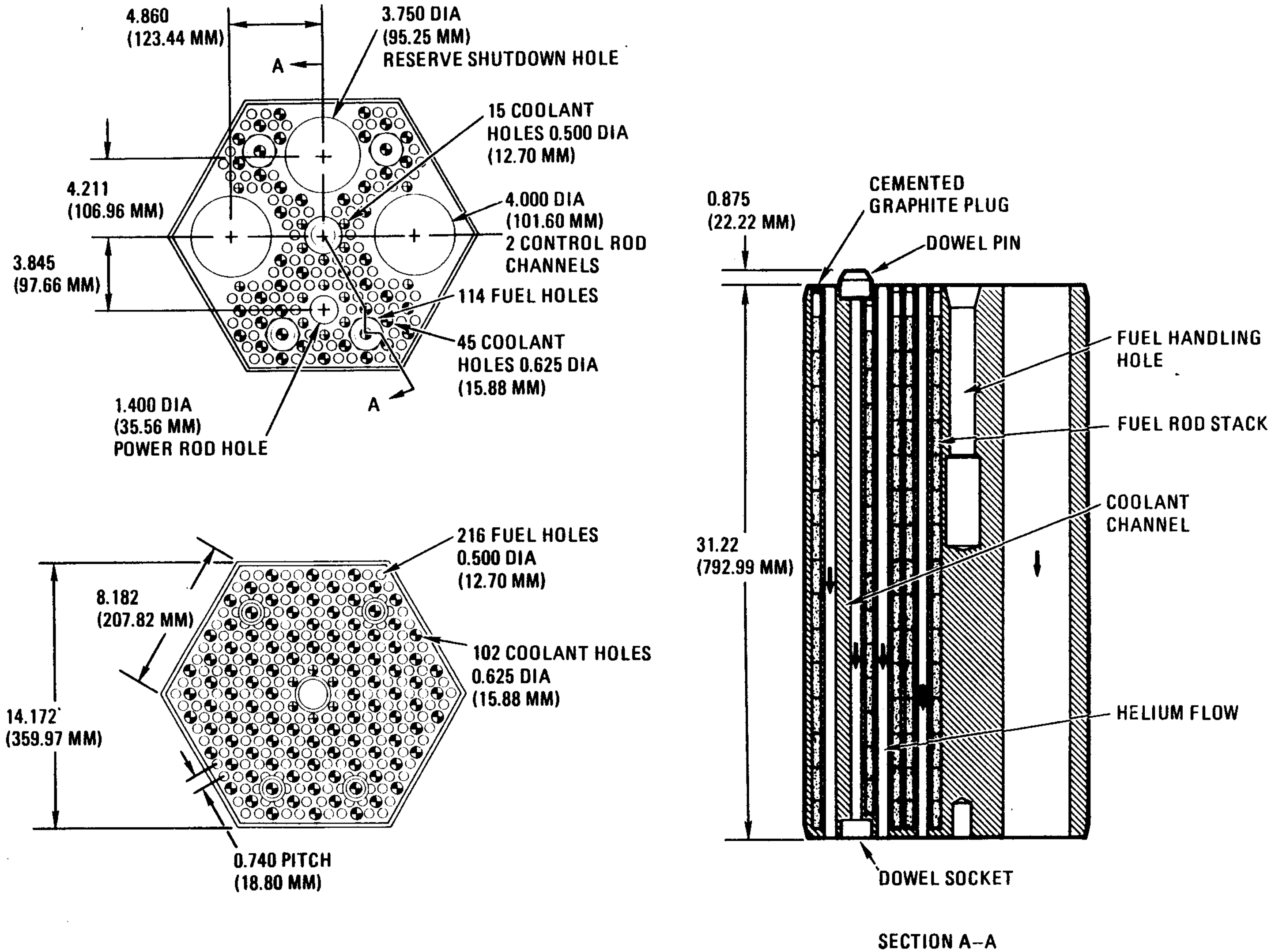

Figure 5.1.1-5 HTGR fuel element

The HTGR core is designed to accommodate many different cycles, including those with fully enriched uranium and thorium (HEU/Th) and those with low-enriched uranium (less than 20% U-235) with and without thorium (LEU/Th and LEU cycles).

For the cycle with high-enriched fuel, 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 the same type of coatings as the fissile kernels.

In the low-enrichment fuel cycles, the fuel element design would be essentially unchanged except for the fuel rod diameter, which might be adjusted to accommodate the different fuel loadings of this cycle. The fuel particle designs would be similar in character to those used in the high-enrichment cycle.

The fuel cycle is based on a batch loading scheme in which a certain fraction of the core, known as a fuel segment, is refueled on an annual basis. Thus, on a 3-yr cycle, approximately one third of the fuel regions would be reloaded each year, and each region would remain in the core for 3 yr. The refueling scheme is chosen so that the regions to be refueled are symmetrically distributed throughout the core. The region distribution for a 3-yr fuel cycle is illustrated in Fig. 5.1.1-4. Other refueling schemes are also feasible, for example, a 4-yr cycle in which about 25% of the core is refueled annually. Mixed cycles are also feasible; for example, all fuel regions on the boundary could be reloaded on a 6-yr cycle and all inner regions on a 3-yr cycle. This scheme, known as the "thick buffer" fuel cycle, minimizes boundary power peaks while providing reasonable fuel economics. The actual refueling scheme chosen will depend on optimization studies, including both fuel cycle costs and plant operation.

5.1.1.4 Primary Cooling System

The NHS primary cooling system consists of four parallel forced circulation cooling loops, each containing an IHX, a main helium circulator and drive motor, and a loop shutoff valve. As shown in Fig. 3.1.1-1, these loop components are located in cavities peripheral to the central reactor core cavity. The peripheral cavities are connected to the core cavity by upper and lower cross ducts.

The function of the primary system is to transfer the heat generated by the reactor core to the secondary system coolant (helium) during normal plant operations. Circulator discharge helium flows to the core cavity upper plenum via the upper cross ducts. The flow is then directed downward through the core, where it absorbs the nuclear heat and exits into the lower core plenum. The flow then proceeds to the IHX cavities via the lower cross ducts, through the IHX where it transfers heat to the secondary coolant (helium), and finally to the main helium circulator, completing the loop. The major components of the primary system are described below.

- Intermediate Heat Exchanger - The function of the IHX is to transfer thermal energy from a primary coolant loop to a secondary coolant loop and, in addition, to provide a barrier for egress of fission products, circulating with the primary coolant into the secondary loop.

  The IHX for the NHS (Figs. 5.1.1-6 and 5.1.1-7) is a straight tubular gas-to-gas counterflow heat exchanger. The heat transfer bundle tubes are welded at each end to a tubesheet assembly, which comprises a tubesheet and spherical head. A circular shroud welded to one of the tubesheet assemblies encloses the bundle; the shroud is perforated at the top and bottom for radial secondary gas flow. The tubes are supported laterally by horizontal low pressure drop "egg crate" type grids, which transfer tube loads into the shroud.

  The IHX is located entirely in the PCRV and is welded at the lower end to a liner extension support. The upper end of the unit is attached to a primary/secondary gas boundary dome via a bellows/seal assembly, which compensates for IHX axial thermal expansion. A secondary gas bypass seal is located in the annulus between the IHX and the cavity liner. Primary gas flow restrictors are provided at each end of the unit to guard against the unlikely simultaneous failure of the tubesheet/head weld and the secondary piping outside of the PCRV.

  Primary helium from the core enters the IHX at the bottom, flows upward through the tubes, and exits at the top to the circulator located in the same cavity, where it is compressed and returned to the core. The secondary helium enters the IHX cavity at the top, flows radially through the shroud perforations to the top of the bundle, and then turns 90° and flows downward over the outside of the tubes in counterflow to the primary gas. The helium exits the bundle radially through the lower shroud perforations and carries heat to the reformer. Geometric and thermodynamic characteristics are shown in Table 5.1.1-3.

- Main Helium Circulation Equipment - The main helium circulator is a vertically oriented, single-stage centrifugal compressor. It is driven by a directly coupled vertical electrical motor. The motor is a 3-phase synchronous unit with a brushless excitation system. Variable speed is provided by use of a solid-state adjustable frequency power supply. Figure 5.1.1-8 shows the main features of the unit, and Table 5.1.1-4 lists the major circulator parameters. The compressor is inside the primary cooling system pressure boundary and the motor is in the containment environment; a multistage mechanical face seal is used to provide pressure separation.

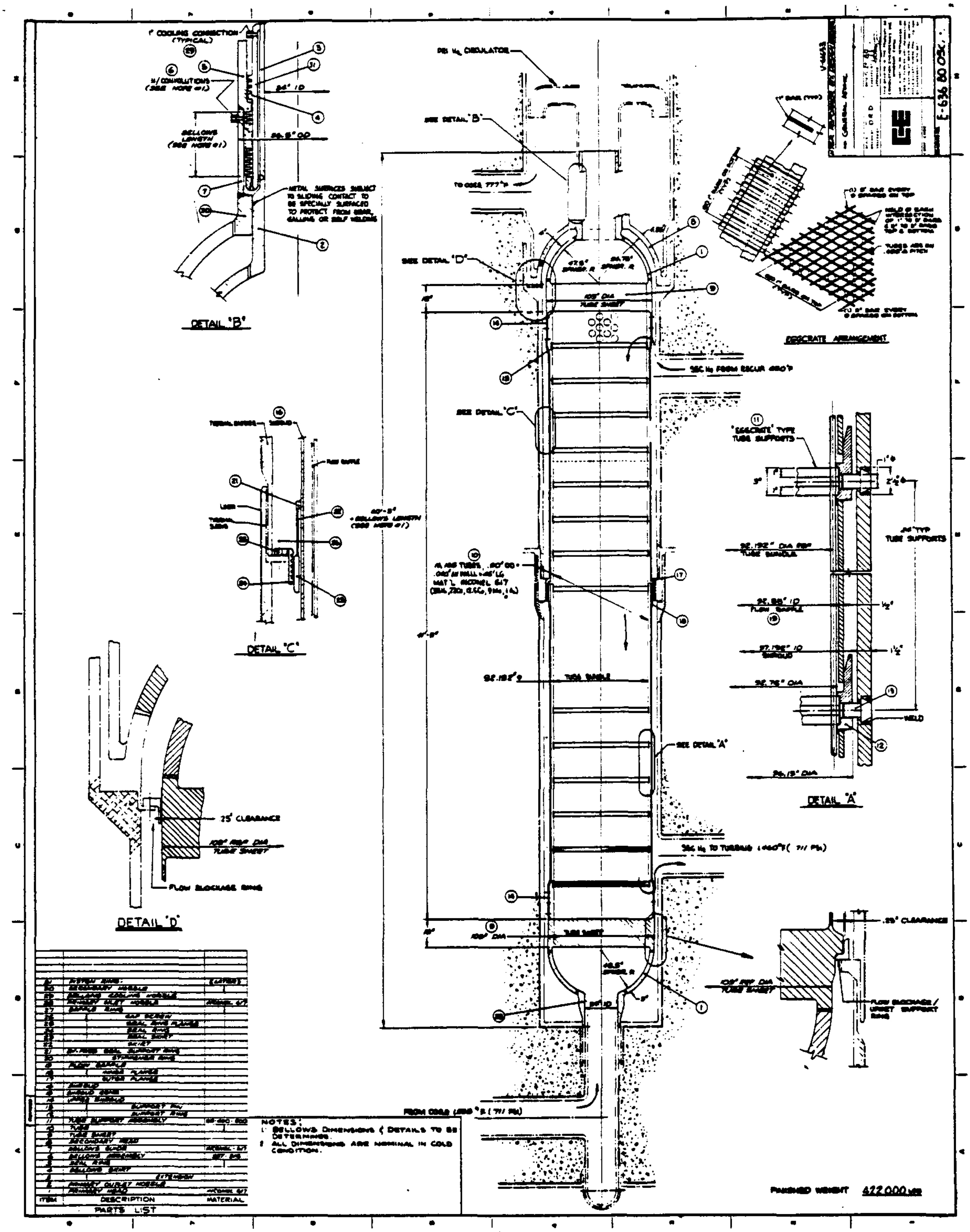

Figure 5.1.1-6   HTGR-R IHX Design

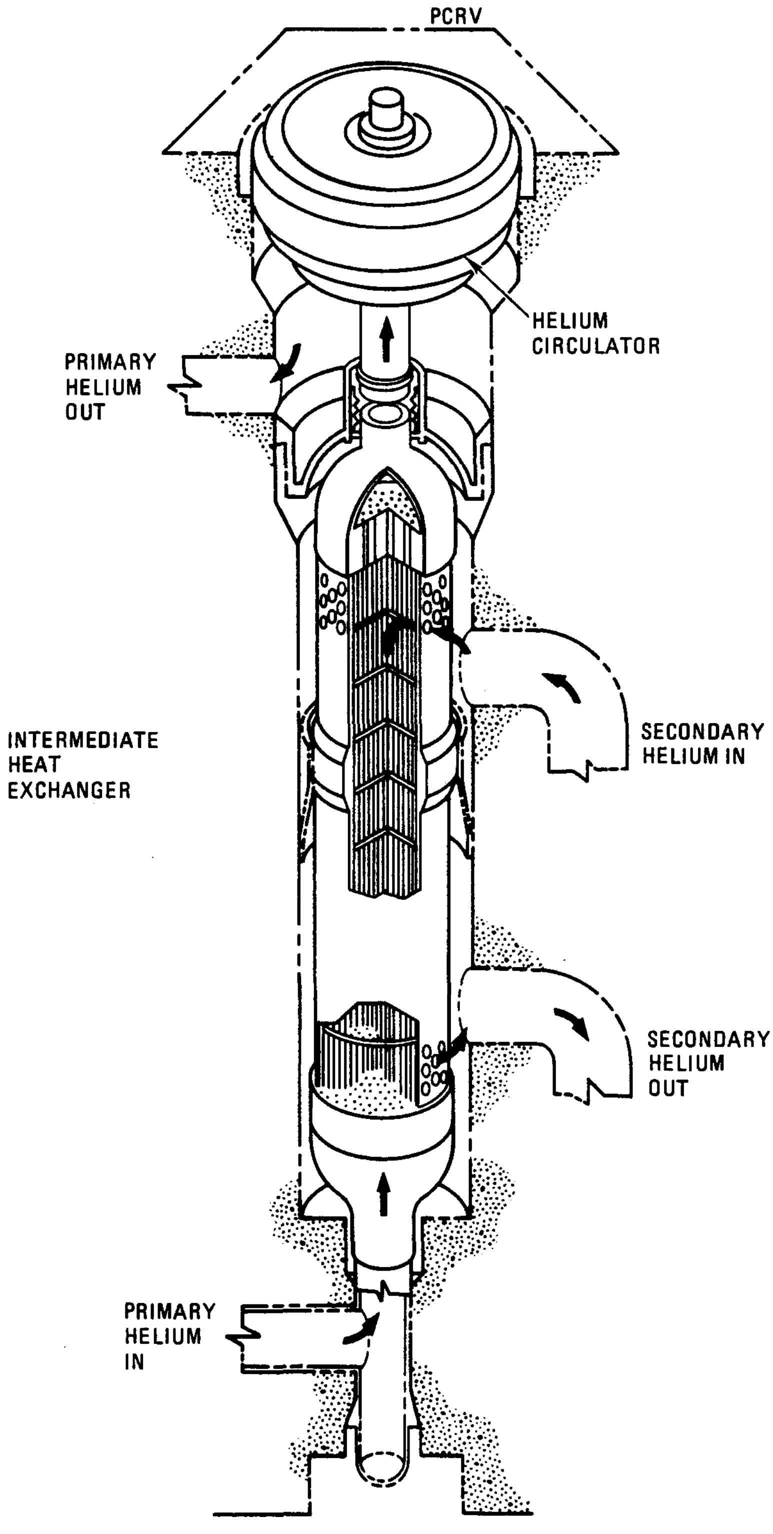

Figure 5.1.1-7 Intermediate heat exchanger

TABLE 5.1.1-3
HTGR-R INTERMEDIATE HEAT EXCHANGER DESIGN DATA
(VALUES FOR ONE HEAT EXCHANGER)

Power rating, MW(t)	290
Primary helium flow, kg/s (lb/hr)	133.50 (1.06 x 10^6)
Primary helium inlet temperature, °C (°F)	846 (1555)
Primary heliium outlet temperature, °C (°F)	414 (777)
Secondary helium flow, kg/s (lb/hr)	128.04 (1.02 x 10^6)
Secondary helium inlet temperature, °C (°F)	343 (650)
Secondary helium outlet temperature, °C (°F)	793 (1460)
Primary helium inlet pressure, MPa (psia)	4.9 (711)
Primary helium outlet pressure, MPa (psia)	4.81 (698)
Secondary helium inlet pressure, MPa (psia)	4.95 (718)
Secondary helium outlet pressure, MPa (psia)	4.9 (711)
Number of tubes	16,160
Tube outside diameter, mm (in.)	12.7 (0.50)
Wall thickness, mm (in.)	1.016 (0.040)
Active tube length, m (ft)	12.57 (41.25)
Tube material	Inconel 617
Module weight, kg (lb)	153,042 (337,400)
Module length, m (ft)	16.54 (54.25)
Module maximum diameter, m (ft)	2.56 (8.42)

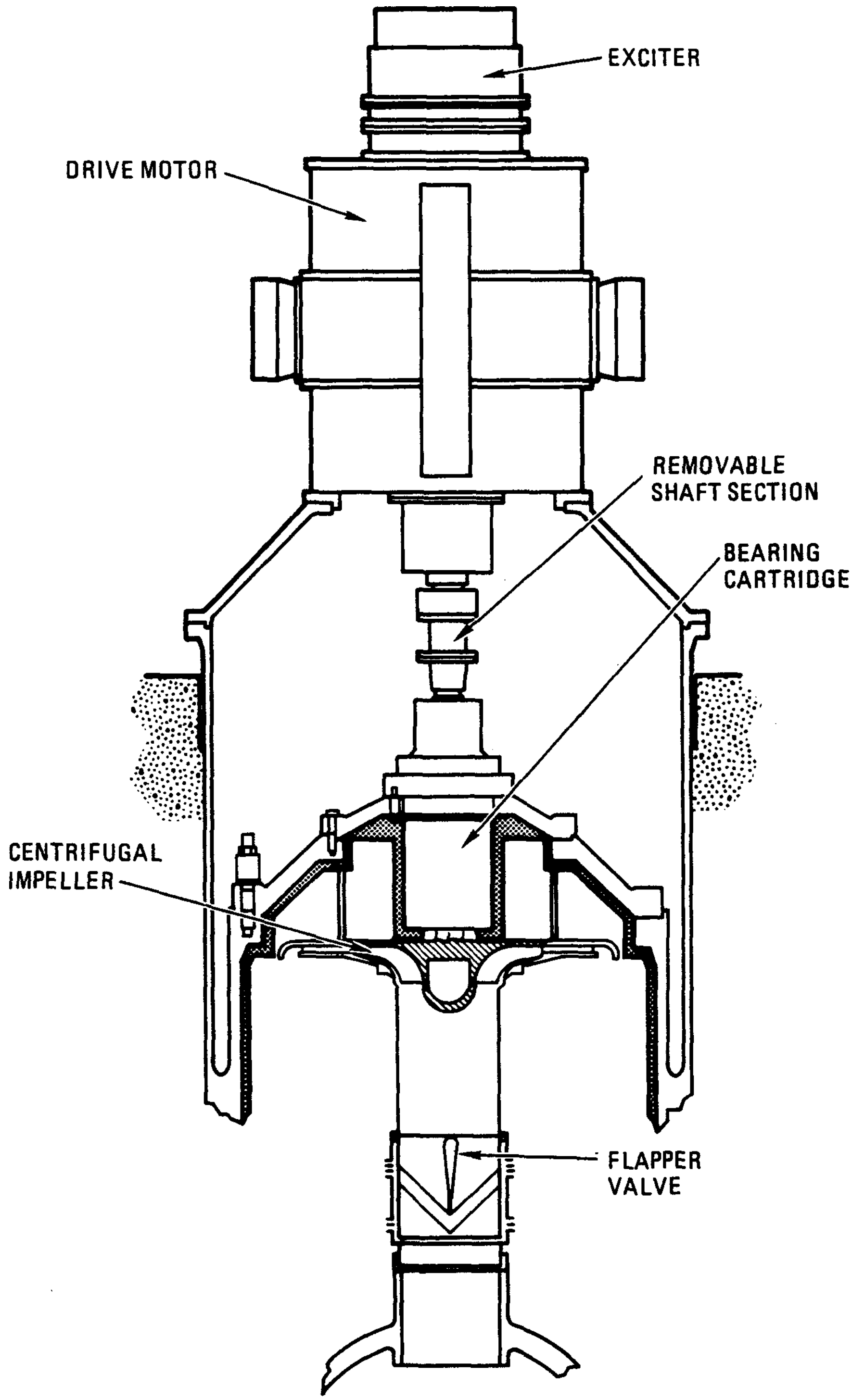

Figure 5.1.1-8 Main circulator

TABLE 5.1.1-4
HTGR-R MAIN CIRCULATOR DESIGN DATA

Inlet pressure, MPa (psia)	4.8 (698)
Inlet temperature, °C (°F)	414 (777)
Outlet pressure, MPa (psia)	5.0 (725)
ΔP, MPa (psi)	0.186 (27)
Head, m (ft)	5,639 (18,500)
Flow, kg/s (lb/s)	135.2 (298)
Volume flow, m^3/s (ft^3/s)	40.07 (1415)
Wheel diameter, m (in.)	1.55 (61.16)
Speed, rpm	3393
Specific speed, rpm	80
Power, MW (hp)	10 (13,300)
Efficiency, %	82.5

The rotating shaft system, i.e., motor and compressor, is supported on three main radial bearings and a double-acting thrust bearing. Two radial bearings and the thrust bearing are a part of the electric motor; they are pivoted shoe bearings utilizing oil as the lubricant. A single-shrouded, step water bearing is provided in the compressor near the impeller. This type bearing is utilized in the compressor to preclude the possibility of large quantities of oil entering the primary cooling system. In addition, the water bearing is unaffected by high-temperature soak conditions and radiation damage. Water is prevented from entering the primary cooling system through the use of buffer helium in conjunction with limited leakage clearance seals and labyrinths. Water pressure for the journal bearing is provided by a self-actuated bearing pump. The compressor shaft has a two-stage centrifugal pump impeller mounted between the journal bearing and the high-pressure shaft seal. The mechanical shaft seal is similar to units used on PWR coolant pumps. In addition to the shaft mechanical seal, two static shutdown seals are provided. These seals can be used to isolate the compressor bearing cartridge outboard of the compressor journal bearing. The seals are bellows-operated and can only be used when the shaft is stationary. A primary coolant shutoff valve is located at the inlet to each compressor. This valve is a split-flapper flow-actuated check valve.

5.1.1.5 Core Auxiliary Cooling System

The function of the CACS is to provide an independent means for cooling the reactor core following a loss of main loop cooling when the PCRV is pressurized or depressurized. The CACS cooling capability is such that the temperatures of all components in the PCRV are maintained within safe limits. This is accomplished by forced circulation of primary coolant through the auxiliary cooling loops. Residual heat and afterheat are removed by the coolant flow as it passes in a downward direction through the reactor core. This heat is delivered to the CAHE where it is transferred to the core auxiliary cooling water system (CACWS) for ultimate dissipation to the atmosphere via an air blast heat exchanger. Each of the three CACS loops is fully capable of removing the core residual and decay heat for safe cooldowns from 102% of reactor power level under pressurized conditions. Two loops are required for cooldown from the same operating conditions when the reactor is in the depressurized condition.

Table 5.1.1-5 gives the CACS design base data for the HTGR-R. The components of the CACS are described below.

TABLE 5.1.1-5
HTGR-R CACS DESIGN DATA

Number of loops	3
CAHE	
Bundle length/diameter, m (ft)	4.1/1.1 (13.5/3.5)
Heat transfer surface area,[a] m^2 (ft^2)	189.6 (2041)
Outer tube o.d./i.d., mm (in.)	35/27.9 (1.38/1.10)
Inner tube o.d./i.d., mm (in.)	22.9/12.7 (0.90/0.50)
Tube spacing (triangular array), mm (in.)	50.8 (2.00)
Number of tubes	420
Auxiliary circulator data	
Maximum power requirement, shaft kW/loop (shaft hp/loop)	176 (236)
Pressure rise, Pa (psi)	5 (0.73)
Mass flow, kg/s/loop (lbm/hr/loop)	11.14 (88,425)
CACWS	
Heat duty, MW/loop (Btu/hr/loop)	31.2 (1.064×10^8)
Air blast heat exchanger bare tube heat transfer area, m^3/loop (ft^2/loop)	394.8 (4250)
Water flow rate at pump inlet, m^3/s/loop (gpm/loop)	0.045 (720)
Water pumping power, shaft kW/loop (shaft hp/loop)	84 (113)
Water nominal pressure, MPa (psia)	10.34 (1500)
Water subcooling margin, mJ/kg (Btu/lbm)	0.163 (70)
Air flow rate, kg/s/loop (lb/hr/loop)	193.3 (1.534)
Air approach velocity, m/s (ft/min)	5.69 (1120)
Air pumping power, shaft kW/loop (shaft hp/loop)	248 (333)
Number of fans per loop	2

The CAHE (Fig. 5.1.1-9) is a straight bayonet-tube, water-cooled heat exchanger. Hot gas from the lower cross duct enters the top of the straight bayonet-tube bundle and flows downward through the tube bundle parallel to the tubes. Water enters and exits the CAHE through a penetration in the bottom of the PCRV. The water entering a bayonet tube flows upward in the annulus between the outer tube and the inner tube wall of the bayonet configuration, gaining heat through the outer tube wall. At the top of the tube, the flow direction is reversed and the water flows downward through the inner tube and the tubesheet.

The auxiliary circulation equipment (Fig. 5.1.1-10) consists of an auxiliary circulator and drive, shutoff valve, and motor controls. The auxiliary circulators are electric-driven axial flow compressors. The auxiliary circulator drives are variable-speed induction motors, which include static inverters for variable frequency speed control. The shutoff valve, which is the butterfly type, functions to limit reverse flow through the system when the main loops are operating.

5.1.1.6 Shutdown Cooling And Afterheat Removal

Shutdown cooling and afterheat removal (hereafter referred to simply as shutdown cooling) are accomplished primarily through use of the main loops. This preferred means of shutdown cooling is described below.

Off-site power is used to maintain continued operation of the primary and secondary helium circulators, the condensate and feed pumps, and the condenser cooling water pump. Steam generated during residual heat removal is routed through a turbine bypass line, desuperheated, and then condensed. As the residual heat supply diminishes, the steam generators eventually flood. Core cooling can be maintained indefinitely by rejecting heat to the condenser via the primary and secondary helium and the feedwater systems. The cooling effect of the reformer on the effluent secondary helium from the IHXs is required to prevent overtemperature of the steam generator tubes. Off-site power is used to maintain circulation of the process fluids.

The plant includes a three-loop CACS (Section 5.1.1.5). This is an engineered safety system that meets all of the shutdown cooling requirements set forth in the plant design criteria. It has the capability to provide adequate shutdown cooling under both normal and accident conditions.

5.1.1.7 Support and Service Systems

The systems described below are a part of the NHS but are not a direct part of the NHS power transfer equipment. They are, in general, support or service systems.

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

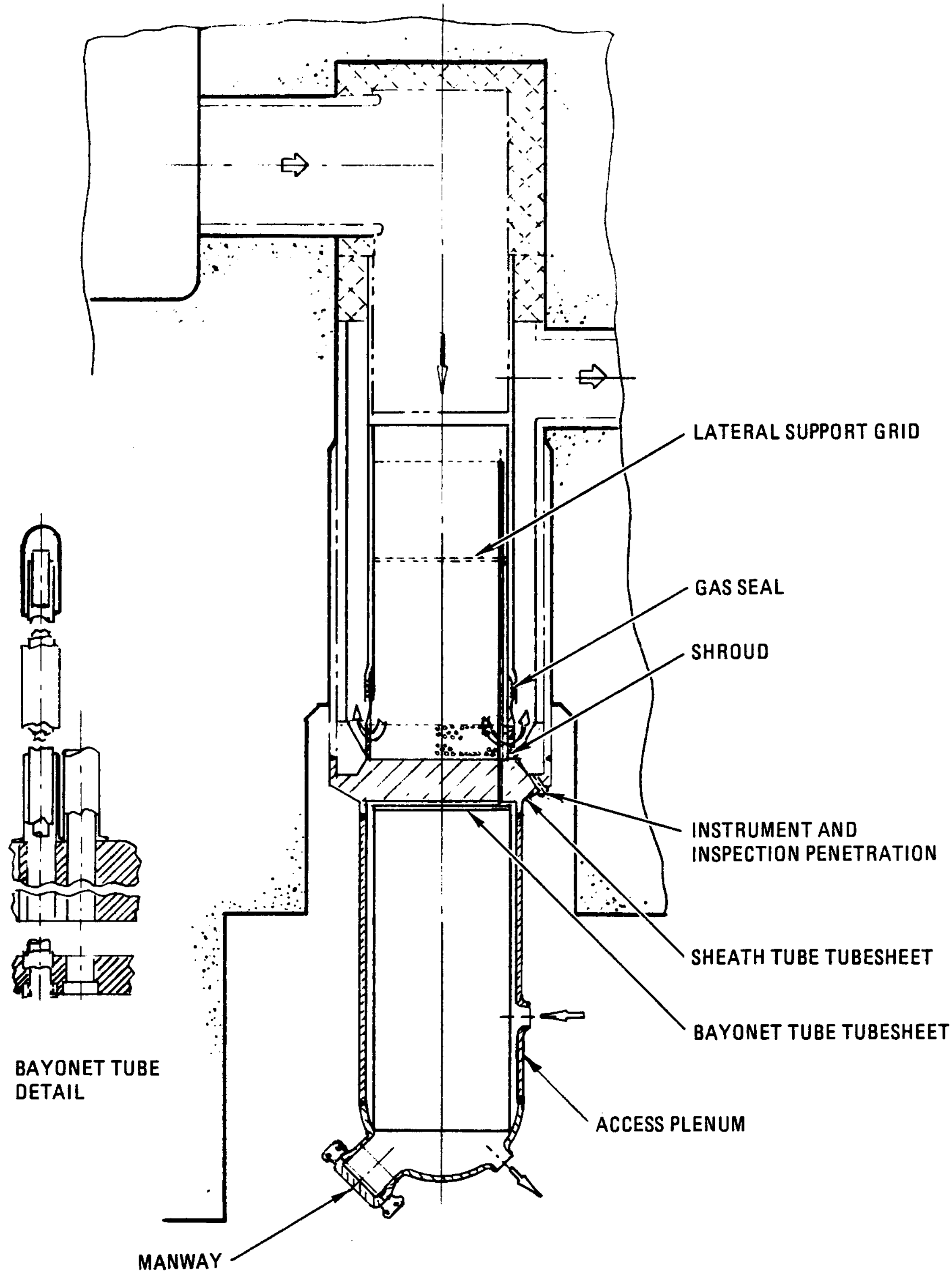

Figure 5.1.1-9 Bayonet-tube CAHE

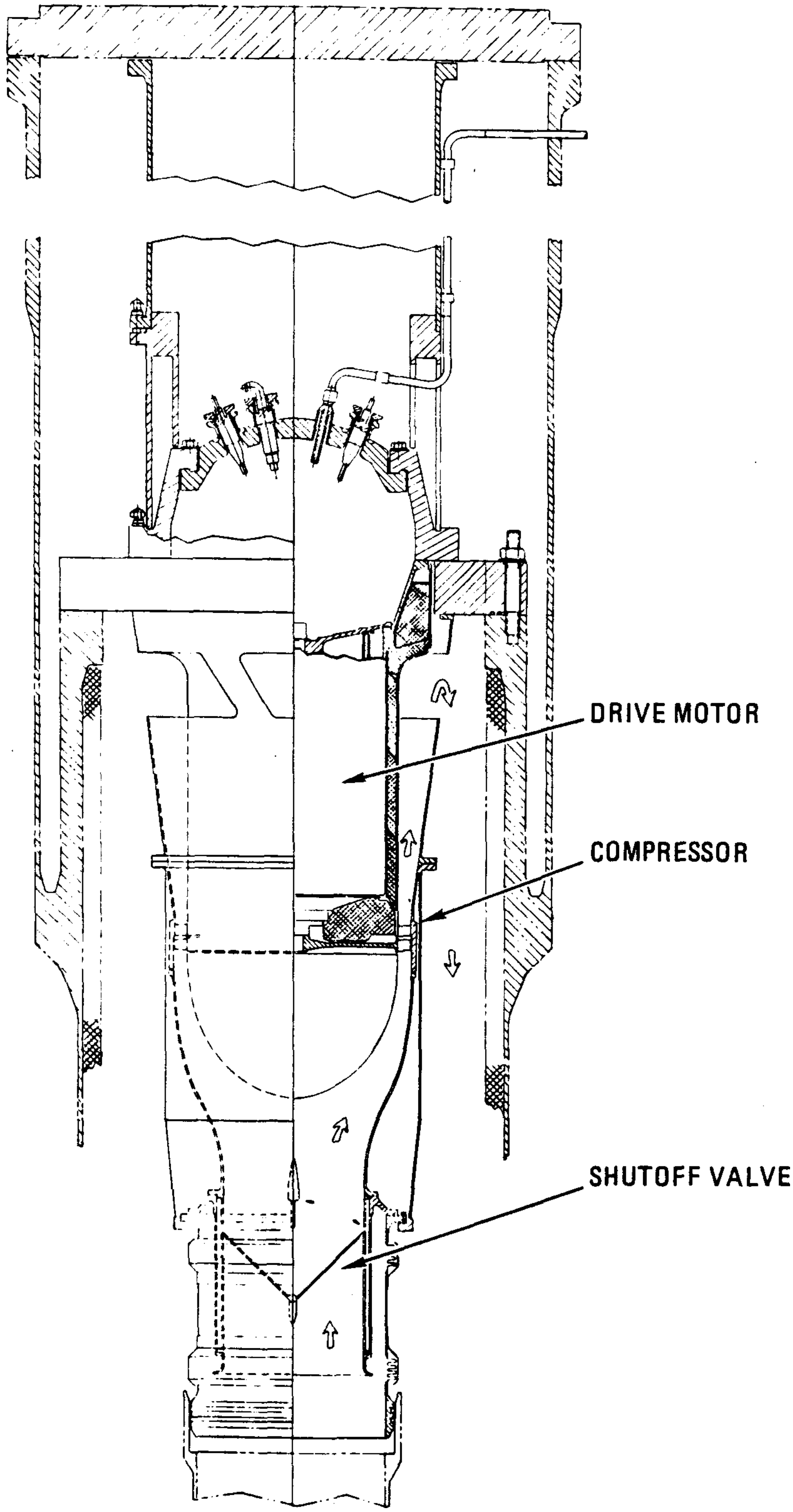

Figure 5.1.1-10 Auxiliary circulator

The main circulator service system provides high-pressure water for circulator bearing lubrication and cooling. The system also supplies buffer helium to prevent bearing water ingress to the primary system, to prevent leakage of primary coolant into the circulator, and to actuate the circulator static seals during circulator shutdown.

The auxiliary circulator service system provides buffer helium to prevent ingress of circulator motor bearing lubricant to the primary system or leakage of primary coolant to the motor casing.

The helium service system removes helium from the primary system and processes it to remove particulates, chemical impurities, and radioactivity such that the resulting gas can be safely used as a purge and seal gas throughout the plant.

The plant protection system prevents unacceptable releases of radioactivity that could constitute a hazard to the health and safety of the public. The system initiates actions to protect the fission product release barriers and limits release of radioactivity should failures occur in the barriers.

The plant control system provides safe plant operation and high plant availability. The system is designed to regulate reactor power and to control the pressure and temperature of the helium produced by the NHS, based on appropriate interface relationships with the BOP.

The remaining systems not directly involved with NHS power operation include the fuel handling, fuel shipping, reactor services, analytical instrumentation, and gas waste management systems.

5.1.2 Balance of Plant

This section presents the plant description for the 1170 MW(t) HTGR-R balance of reactor plant and balance of plant. 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. General arrangement drawings of major site structures, the plot plan, the heat balance, and the electrical single line diagrams developed for the HTGR-R lead plant are provided in Appendix C.

5.1.2.1 Structures and Improvements

The selected site arrangement is shown in drawing SK-147. This arrangement achieves a high degree of close-coupling between major structures while allowing for minor increases in the size of each structure. The auxiliary reactor service building is located directly adjacent to the reactor containment building to minimize travel distance for refueling equipment and the length of piping runs. The control auxiliary and diesel generator building is located near both the reactor containment building and the auxiliary reactor service building to minimize cable runs. The reactor containment building incorporates the containment annulus building, the containment penetration building, and the auxiliary reactor service building on a common mat.

The containment annulus building contains all of the major safeguards equipment for the reactor plant cooling water system, the core auxiliary cooling water system, and the auxiliary circulator motor cooling water system, with the exception of the core auxiliary cooling water system air blast heat exchangers. The air blast heat exchangers and the nuclear service water system equipment are located in the ultimate heat sink structures. The containment annulus building also houses related heating, ventilating, and air conditioning (HVAC) equipment and the piping penetrations for the safeguards as well as the secondary helium system. The containment penetration building houses some equipment for the radioactive waste management system and includes two separate cable penetration areas which link the control auxiliary and diesel generator building and the reactor containment building.

The reforming train buildings and the reformer prestressed concrete pressure vessel (PCPV) building are sited an appropriate distance (assumed to be 200') from all safety-related structures. The reformer PCPV building was placed between the reforming train building and the reactor containment building to minimize the length of the secondary helium pipe runs and to mitigate the effects of any problems which might occur in the reforming train buildings.

The steam turbine building is located in close proximity to the reformer PCPV building to minimize the length of steam piping runs from the steam generators located in the reformer PCPV building to the turbine generator. The steam turbine building orientation precludes any interaction between Seismic-Category-I structures and postulated turbine missiles as illustrated on the site plan.

TABLE 5.1.2-1

1170 MW(t) HTGR-REFORMER PLANT

STRUCTURE DIMENSION SUMMARY

	STRUCTURE	SEISMIC CATEGORY	NUMBER OF STORIES	HEIGHT FEET	WIDTH FEET	LENGTH FEET	VOLUME $10^3 FT^3$
212	Reactor Containment Building	Cat. I	NA	228'	131'I.D. 140'O.D.	NA	2776
213	Steam Turbine Building	Non-Cat. I	3	102'	112'	248'	2833
214	Security Building	Non-Cat. I	1	14'	69'	72'	70
215	Auxiliary Reactor Service Building	Cat. I	5	109'	80'	118'	1025
216	Main Circulator Controller Bldg.	Non-Cat. I	1	20'	55'	110'	121
217	Fuel Storage Building	Cat. I	1	76'	83'	97'	725
218A	Control Aux. & Diesel Gen. Building	Cat. I	6	126'	115'	138'	2000
218B	Administration & Services Building	Non-Cat. I	2	26'	160'	240'	998
218C	Auxiliary Boiler and Makeup Demineralizer Building	Non-Cat. I	1	40' 20'	60' 50'	80' 50'	192 50
218D	Fire Pump House	Non-Cat. I	1	12'	30'	75'	27
218E	L.P. Helium Storage Area	Non-Cat. I	NA	NA	150'	200'	NA
218F	Non-Vital Switchgear Building	Non-Cat. I	1	26'	30'	78'	61
218H	Diesel Cooling & Fuel Oil Storage	Cat. I	2	72'	52'	68'	255

TABLE 5.1.2-1
(Continued)

1170 MW(t) HTGR-REFORMER PLANT

STRUCTURE DIMENSION SUMMARY

	STRUCTURE	SEISMIC CATEGORY	NUMBER OF STORIES	HEIGHT FEET	WIDTH FEET	LENGTH FEET	VOLUME 10^3FT^3
218I	Warehouse	Cat. I	1	20'	50'	70'	70
218J	Containment Annulus Building	Cat. I	2	48'	140'I.D. 200'O.D.	NA	494
218K	Containment Penetration Building	Cat. I	5	109'	37'	93'	429
218L	Secondary Circulator Building (4)	Non-Cat. I	1	42'	64'	69'	186
218M	Reformer PCPV Building (4)	Non-Cat. I	1	114'	64'	64'	467
218N	Reforming Train Building (4)	Non-Cat. I	1	40'	64'	243'	622
218S	Holding Pond	Non-Cat I	NA	8'	80'	80'	51
	and Control House		1	10'	8'	10'	1
218T	Ultimate Heat Sink Structures Train A & B	Cat. I	2	61'	56'	62'	425
	and Train C		1	37'	25'	62'	57
218U	Control Room Emergency Air Intake Structures (2)	Cat. I	1	15'	12'	12'	2

The nuclear island for the HTGR-R incorporates building arrangements previously developed by UE&C for HTGRs with larger foundation base mats. This arrangement minimizes nuclear island building sizes and is consistent with the other HTGR options. Detailed analysis to verify the acceptability of bending moments on the non-symmetrical foundation mats will be the subject of future work.

- Reactor Containment Building (SK-148 through SK-150) - The reactor containment building houses the prestressed concrete reactor vessel (PCRV) and other nuclear heat source (NHS) 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 release 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 Category-I, reinforced-concrete structure composed of a foundation mat, cylindrical shell, and hemispherical dome with a design pressure of 60 psia. 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 two hatches: a 28' I.D. equipment hatch at the refueling floor level at the top of the PCRV and an 8' personnel hatch. The PCRV is concentrically located and is supported on a 12'6"-high, reinforced-concrete ring wall and pedestal bearing on the reactor containment mat. Within the annular space between the PCRV and 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. Below the containment floor, a temporary fuel storage facility is located in the containment mat.

- Steam Turbine Building (SK-151) - The steam turbine building houses the turbine generator, condenser, condensate system, feedwater system, service and instrument air system, and other associated equipment. The steam turbine building is a non-Category-I, three-story, metal structure supported on reinforced-concrete spread footings and framed with structural steel. The ground floor is a concrete slab on grade, and the upper floors are grating, except for the operating floor in the turbine hall, which is a concrete slab. The roof is metal decking with insulation and built-up roofing. The walls are covered with insulated metal siding. An overhead traveling crane is located in the turbine hall, as is the turbine-generator pedestal, a high-tuned, reinforced-concrete structure supported on a mat foundation. A rail/truck bay is located at one end of the building on the ground floor.

- 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 transport of the fuel handling equipment, 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.

- Main Circulator Controller Building - The main circulator controller building houses electrical controllers and associated equipment for the main helium circulators.

- 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 alongside 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 wells in the fuel storage pool, also located in this building. A rail/truck shipping bay is located at one end of the building and extends outside the fuel storage building to accommodate oversized rail cars.

- Control Auxiliary and Diesel Generator Building - The control auxiliary and diesel generator building houses the control and electrical equipment required for plant operation, which includes the main control room, cable spreading areas, switchgear area, and diesel generators. 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, shops, warehouse, and storerooms. Also included are a health physics complex and a checkpoint to control entry to nuclear island structures.

- Auxiliary Boiler and Makeup Demineralizer Building - The auxiliary boiler and makeup demineralizer building is a framed-structural-steel building which houses the auxiliary boiler and demineralized water system and associated equipment.

- 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 both primary and secondary plant helium inventories.

- Non-Vital Switchgear Building - The non-vital switchgear building is a framed-structural-steel building which houses electrical switchgear for non-vital systems.

- 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 (SK-148 through SK-150) - The containment annulus building houses the reactor plant cooling water system pump, heat exchangers and surge tank, the core auxiliary cooling water system pumps, the auxiliary circulator motor cooling water system pumps, safety-related HVAC equipment, secondary helium system valve rooms, main circulator electric service area, electrical piping, and 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, horseshoe-shaped, annular, two-story, reinforced-concrete structure that surrounds the reactor containment building and 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 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 (SK-148 through SK-150) - The containment penetration building houses the containment electrical penetration areas, instrument areas, data acquisition system room, the remote safe-shutdown room, including switch-gear and ventilation areas, and portions of the liquid and solid waste management system. The containment penetration building is a Category-I, five-story, reinforced-concrete structure that abuts the reactor containment building and utilizes the projected portion of the reactor containment building base mat as a common foundation.

- Secondary Circulator Building (SK-153 and SK-154) - There are four intermediate circulator buildings, each one housing two secondary helium circulators and motors and associated lube and seal oil systems. The secondary circulator building is a non-Category-I, single-story, metal structure supported on reinforced-concrete spread footings and framed with structural steel.

- Reformer PCPV Building (SK-153 and SK-154) - There are four reformer PCPV buildings, each one housing a single prestressed concrete pressure vessel containing two reformers and one steam generator associated with a single reforming train. Secondary helium piping, feedwater and steam piping, and process gas piping are routed as needed to the PCPV. The HVAC system for this building is designed to maintain acceptable PCPV temperatures and is instrumented to detect process gases which may be present in the building environment. The HVAC system includes both continuous slow purge and emergency fast purge equipment to prevent buildup of combustible gas mixtures.

  The reformer PCPV building is a non-Category-I, single-story, metal structure supported on a reinforced-concrete foundation mat and framed with structural steel. Surrounding the PCPV is a steel structure extending up to the top of the PCPV to allow personnel access to that area. Hatches are provided in the reformer PCPV building roof to allow for removal of the reformer and steam generators using an external mobile crane.

- Reforming Train Building (SK-153 and SK-154) - There are four reforming train buildings, each one housing the heat exchangers, charcoal bed, helium compressors and motors, and associated equipment which constitute a single reforming train. The HVAC system for this building is designed and instrumented to detect process train gases which may be present in the building environment. Buildup of combustible process gas is prevented by continuous purge of the building environment. The reforming train building is a non-Category-I, single-story, metal structure supported on reinforced-concrete spread footings and framed with structural steel.

- 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 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 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 and reactor plant cooling areas in the containment annulus 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 core auxiliary and diesel generator building.

5.1.2.2 Reactor Plant Equipment

The reactor plant is designed to meet the interface requirements of the two-loop 1170 MW(t) HTGR-Reformer Nuclear Heat Source (NHS) provided by General Atomic Company. The NHS scope includes the reactor internal components, reactor core, neutron and region flow control system, fuel handling system, PCRV service system, primary coolant system, control rod drive storage wells, special shipping equipment, main and auxiliary circulator service systems, helium purification system, core auxiliary cooling system, reactor plant protection system, overall plant control system, and plant data acquisition and processing system. 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 motor during periods of full core auxiliary cooling system operation. The safeguards cooling system incorporates sufficient redundancy and capacity to ensure adequate motor 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 System - This system utilizes the in-vessel refueling and fuel handling 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 mat, provides storage for one refueling. Fuel storage 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 cooling 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, maintain primary coolant chemistry, 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 for remote operation of the plant. 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.

5.1.2.3 Turbine Plant Equipment

The energy conversion system consists of a tandem-compound, two-flow turbine with 33-1/2" last-stage blading with reheat and a two-pole, hydrogen-cooled generator and rotating exciter with a synchronous speed of 3600 RPM. The turbine-generator is calculated to deliver 288,410 KW(e) gross output with throttle steam conditions of 2415 psig/950°F/ 2,236,015 lb/hr flow and 2.5 in Hga exhaust pressure while operating in a regenerative feedwater heating cycle having steam-turbine-driven boiler feed pumps and five stages of feedwater heating. Turbine generator accessories include the lube oil supply and purification system, hydraulic oil system, stator cooling system, gland sealing system, gas storage system, and associated instrument and control systems.

A heat balance for this plant (SK-155) was developed using a conceptual feed heating cycle, the specified main steam conditions, and the thermal performance data for an existing two-flow machine obtained from a turbine vendor. Calculated gross turbine-generator output of approximately 288.4 MW(e) was adjusted for house auxiliary power load of approximately 258.1 MW(e) [approximately 95.6 MW(e) primary and secondary circulator electric requirement, approximately 10.5 MW(e) other auxiliary loads, and approximately 152 MW(e) for process gas compressors], resulting in approximately 30.3 MW(e) net station output.

The condensing system includes the main thermal cycle condenser, condensate pumps, condensate booster pumps, air removal pumps, condensate storage tank, steam bypass to condenser, and condensate polishing system. The condenser consists of a single shell under the double-flow LP turbine, with two-pass, single-circuit circulating water flow.

Five-stage feedwater heating is utilized to heat the feedwater being pumped into the steam generators to 356°F. Heaters number 1, 2, 3, and 5 are closed-type, horizontal heaters with U-tube bundles. The number 4 heater is a direct-contact, deaerating type with a five-minute-capacity storage compartment. A process gas to feedwater heat exchanger is utilized to transfer approximately 23 MW(t) from the reformer train to the turbine plant. This heater is installed in parallel to the low pressure regenerative feedwater heaters. Two 60%-size, turbine-driven boiler feed pumps and three 60%-size, motor-driven booster pumps are provided.

The main steam lines from each steam generator individually penetrate their respective PCPVs and are headered in the reformer PCPV building. Each steam line has an isolation valve, a non-return valve, and two safety-relief valves. Block valves are provided in each lead to the main turbine to preclude water entry during startup. For startup, shutdown, and other conditions of off-normal operation, a main steam bypass to condenser is provided. This bypass uses two steam dump valves and one water dump valve and also a flash tank to generate low-energy steam during periods when the steam generator is flooded.

A demineralized water makeup system provides the required high-purity water for makeup to the secondary cycle and miscellaneous cooling loops and laboratory processes and for initial plant cleaning and flushing operations. An all-volatile chemical treatment system automatically maintains feedwater chemistry.

The turbine plant instrumentation and control system is designed to ensure that the turbine and the related plant systems can be safely and efficiently operated during all normal and off-normal conditions. The instruments and controls are located in the main control room and function in conjunction with the reactor plant controls.

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 safety 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. The key one-line diagram is shown on SK-156.

A generator circuit breaker is provided to facilitate rapid disconnection of the generator from the offsite power system, allowing the auxiliary power system to be fed through generator stepup and station auxiliary transformers. There are ten 15-KV and two 5-KV non-Class-1E metalclad 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 in the Class-1E buses being fed from the diesel generators in the event of a loss of offsite power supplies.

Non-Class-1E and Class-1E 460 V 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 three station auxiliary transformers feeding into two 13.8-KV and two 4.16-KV non-Class-1E buses, three 4.16-KV Class IE buses, and three 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 regulation during extreme plant operating conditions. Appropriate protections are provided for the transformers.

There are four unit auxiliary transformers feeding into eight 13.8-KV non-Class-IE buses. Unit substations are provided to furnish power sources to the low-voltage (460 V), 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 chargers. During emergency operation, d-c power is supplied from the batteries in the absence of any a-c source to the battery chargers. During startup and shutdown, d-c power is supplied from whichever source is available. Non-Class-1E, 125/250 V d-c buses are fed from two non-Class-1E batteries and two non-Class-1E chargers.

Three independent diesel generators are provided to furnish the onsite a-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 uninterruptible power sources for miscellaneous vital and non-vital a-c plant loads.

Switchboards, protective equipment, appropriate electrical structure, 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 system, 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 Waste Heat Rejection System

The waste heat rejection system provides cooling for the main thermal cycle and all plant service water during normal plant operation. This system includes the main cooling tower, circulating water system piping, pumps, and structures; the makeup and blowdown system piping, pumps, and structures and associated instrumentation, controls, and chemical feed systems.

Four 25%-capacity, horizontal, centrifugal circulating water pumps are provided. Cooling is accomplished with a single mechanical draft, wet tower, which is capable of providing cooling requirements for condenser heat loads as stated above, plus normal service water heat load. Meteorological conditions typical of an Eastern Pennsylvania site were assumed for purposes of sizing these towers.

Two 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.

5.1.2.7 Secondary Helium System

The secondary helium system includes all of the piping, compressors, valves, and reactor containment building piping penetrations required to circulate the secondary helium coolant between the intermediate heat exchangers located in the PCRV and the reformers in the reformer PCPV as well as facilities for helium storage and purification. There are four secondary helium loops, one corresponding to each of the four reforming trains.

Eight 5.6-MW(e) secondary helium circulators with 9000-hp motors are provided with associated lube and seal oil skids. Two circulators are provided in each secondary loop. Secondary helium piping is carbon steel with an internal thermal barrier to keep the carbon steel piping below about 600°F. The internal thermal barrier consists of a fibrous insulating material such as saffil encased in Inconel 713 LC and is designed to be fabricated in cast modular cylinders which could be installed in and removed from the carbon steel pipe with relative ease.

Double-valve isolation for each loop of the secondary helium system is provided in the piping just outside of the reactor containment building penetrations. These valves are located in the containment annulus building. Pipe whip restraints are provided inside the reactor containment building as required to protect safety-related equipment in case of a helium pipe rupture.

5.1.3 Process Plant and Delivery System

The process plant and delivery system is comprised of the reformer plant, pipeline and storage, and the methanator plant. The purpose of this system is to convert reactor thermal energy to chemical form, transmit this energy, and recover thermal energy at a remote location via methanation. The proposed chemical reactants system selected for the thermochemical pipeline utilizes steam and methane. Steam-methane reforming to hydrogen and carbon monoxide (syngas) is driven to greater conversion rates with increasing temperature, diminishing pressure, and increasing steam-methane ratios. Preliminary assessment of the achievable conversion ratio with a helium inlet temperature of 800°C (1472°F) and a reformer tube performance based on prior GE test experience suggested that a 70% methane conversion ratio could be achieved with a 750°C (1382°F) maximum reforming temperature, 15-bar reformer inlet pressure, and a 3:1 steam-methane ratio. The effect of increasing conversion ratio is to reduce pipeline and methanator costs at the expense of increased reformer cost.

Plant design factors also arise from desired or imposed pipeline, storage, and methanation plant conditions. The most significant methanation plant condition is that syngas enters the methanators at 61-bar (900 psia). This higher pressure minimizes the number of methanator trains (5 to 6) for the 450 MW(e) peaking turbine-generator because the component size is inversely proportional to the pressure and because of upper limits on component size. The 61-bar (900-psia) level of pressure is practicable from the standpoint of the pipeline and gas storage facilities.

The compressors to initially achieve a 82-bar (1200-psia) transmitting pressure level at the nuclear plant site are installed in the process gas heat exchanger train. Siting of compressors upstream of the mixed-feed evaporators markedly improves the functioning of these devices since condensation of the steam in the process gas stream occurs at high pressure and the associated elevated temperature.

Two pipeline considerations have also influenced the reformer plant equipment design. A requirement has been assumed to introduce an odorant to the gas lines as a safety measure, and a requirement to transport both the process gas and the methane with low levels of moisture to prevent pipeline corrosion and/or freeze damage. To eliminate the odorant and storage-absorbed impurities (both catalyst poisons), contaminant-removal beds are provided, and excess moisture is to be removed from the process gas stream by glycol driers. A water treatment system is provided to maintain boiler-quality water returning from the pipeline and recirculating from the process gas to the methane stream.

Other startup process gas subsystems include (1) a nitrogen purge system for reformer shutdown and startup operation, (2) a gas-flaring system to provide for burning of the low conversion process gas obtained during reformer startup, which can neither be piped offsite or recirculated, and (3) a startup steam system to generate steam for initial feed to the reformers (following the nitrogen purge gas) and which recirculates this steam until methane is introduced and the reforming reaction commences. In addition, methane and water makeup subsystems are needed for initial charging as well as replacing losses.

5.1.3.1 Reformer Plant

The HTGR-R lead plant includes a steam-methane reforming system which converts a portion of the nuclear-generated thermal energy to chemical energy and transports the syngas to a pipeline system for storage and methanation. The reformer plant consists of three basic systems: the reformer, the heat exchanger train, and the auxiliary gas subsystems.

5.1.3.1.1 System Heat Balance

Fig. 5.1.3-1 shows the reference HTGR lead plant configuration and heat balance, including the major nuclear island components and the power-generation system. The steam-methane mixture is normally reformed at a pressure of 15 bar (220 psia) and a 3:1 steam-methane ratio. Energy division of core power supplied to the reformer and the steam generators is 42% and 58%, corresponding to about 515 MW(t) and 716 MW(t), respectively. There are four completely separated gas system heat exchanger trains; the steam-electric system is also independent of the process gas system, reflecting an emphasis on plant operability, controllability, and availability.

Methane-rich gas is delivered to the reformer plant from storage at 16°C and 20-bar (60°F and 300 psia). The gas is desulfurized through a series of activated charcoal beds. Feedwater, returned from the methanator plant at 16°C (60°F) and about 1.4-bar (20 psia), is pumped to reforming pressure [about 16-bar (230 psia)] before mixing with methane and being delivered to the gas system heat exchanger trains. The process of heat exchange between the incoming feed steam and the outgoing stream is presented in Fig. 5.1.3-2. The reformed gas leaves the heater at 193°C (380°F) and is further cooled with turbine feedwater to 149°C (300°F) to minimize the gas compression required for the steam gas separation.

To separate steam from the syngas and to recover the latent heat of the steam, heat must be transferred from the reformed gas to the feed mixture through steam condensation in the reformed gas. This requires the partial pressure of steam in the outgoing stream to be higher than that of the feed stream. The condensation of steam in the outgoing stream and the evaporation of incoming feedwater takes place in a mixed-feed evaporator. While the feedwater provides a cooling mechanism for steam condensation in the syngas stream, the release of latent heat of steam condensation is also required to provide the heating mechanism for feedwater evaporation. The temperature difference required for heat transfer between the two streams is maintained by compressing the syngas to raise the partial pressure of steam in the gas stream so that steam will start condensing at a higher

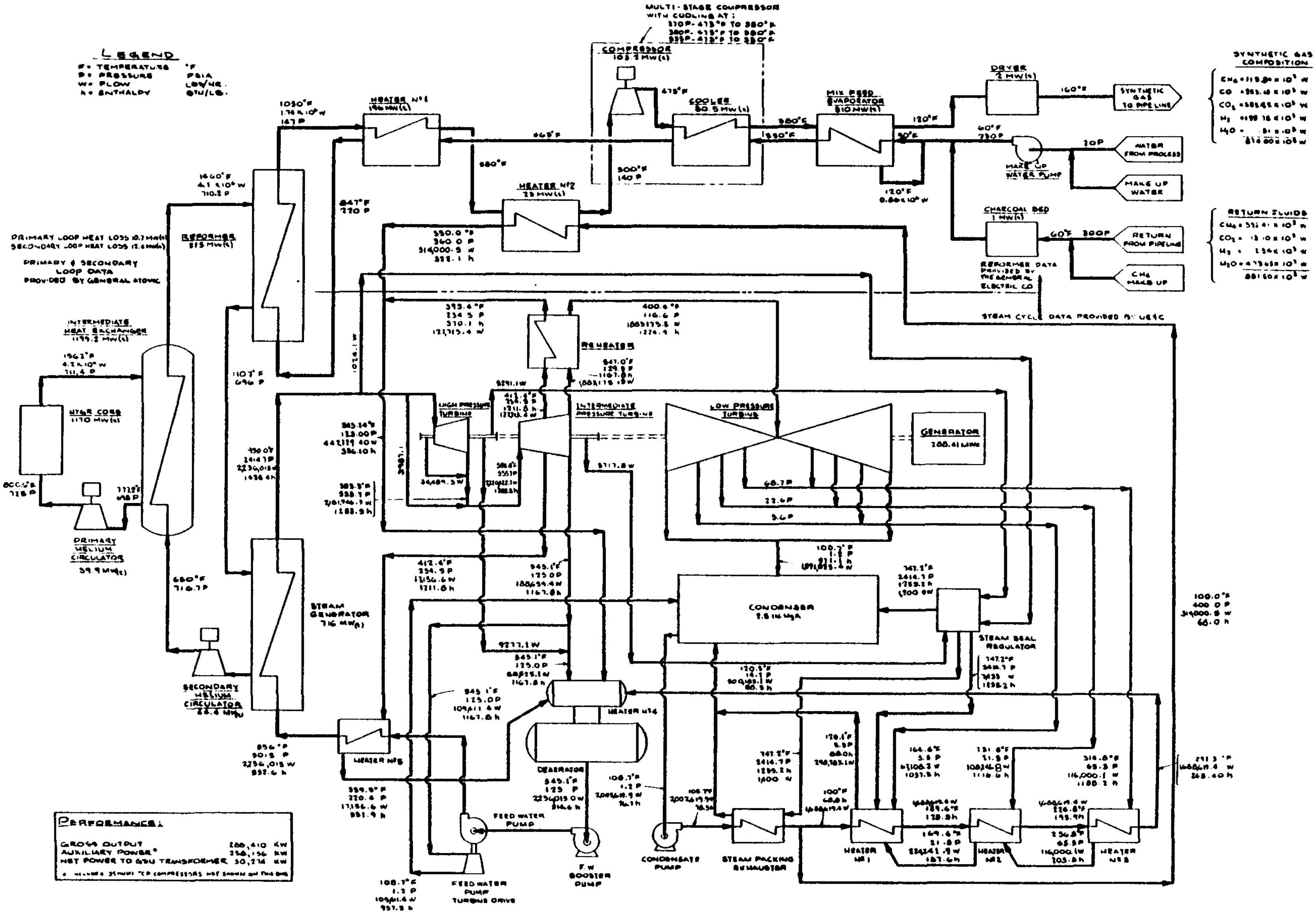

Figure 5.1.3-1 HTGR-R Heat Balance

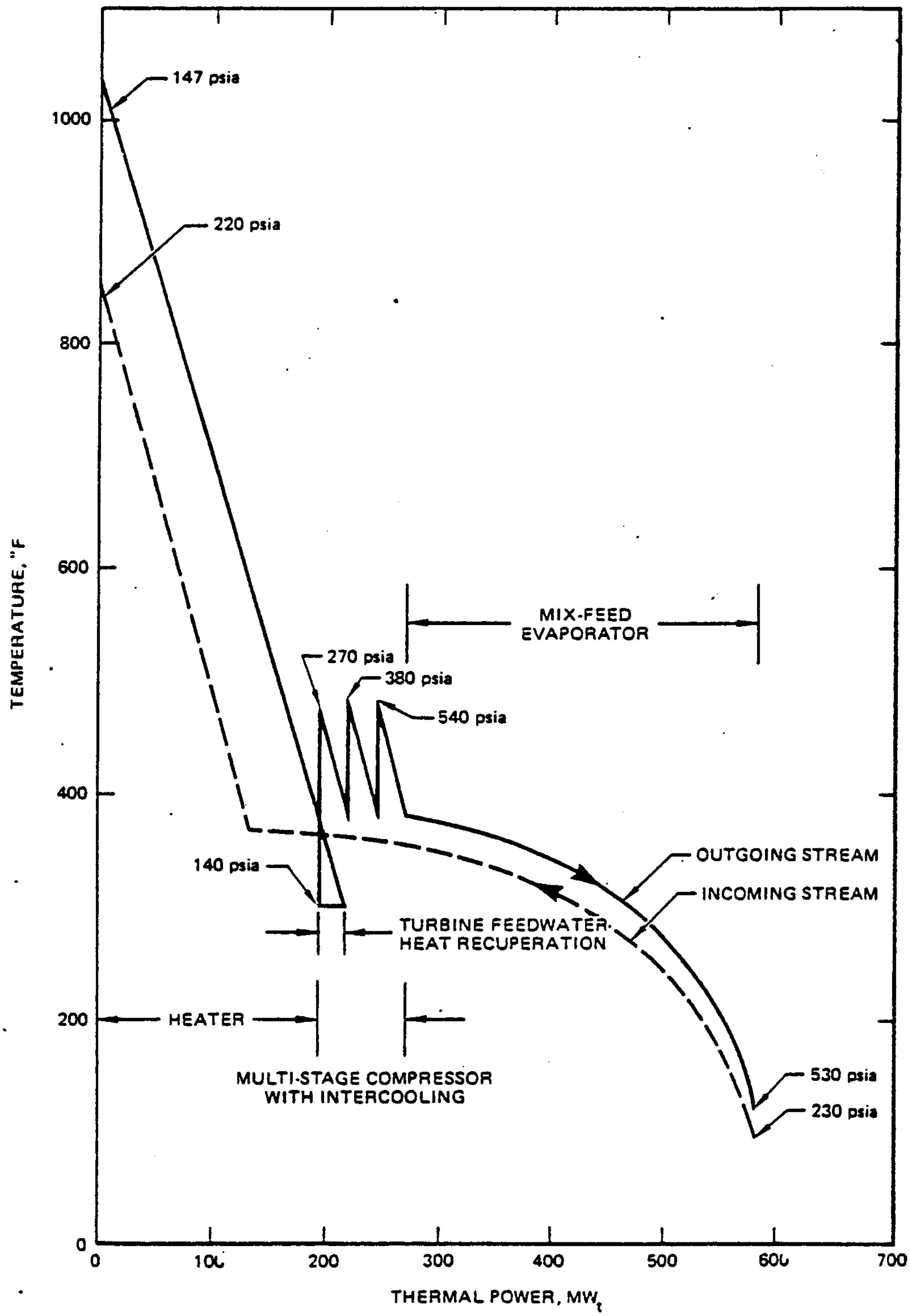

Figure 5.1.3-2 HTGR-R Process Gas Heat Train

temperature. The compressor system is a three-stage gas compressor with intercooling provided by the feed stream. The reformed gas is compressed at each stage to 245°C (473°F) before cooling to 193°C (380°F) to remove heat added during compression. The corresponding discharge pressure at each stage of the compressor is 18-bar (270 psia), 26-bar (380 psia), and 37-bar (540 psia), respectively. The electric motor will require 117 MW(e) to operate with 103.2 MW(t) added into the stream as heat. The total load for the coolers is about 80 MW(t).

A minimum of eight mixed-feed evaporators is required for the reformer plant to minimize the thermal duty and the physical size of the heat exchangers. Two different mixed-feed evaporator designs are arranged in series in each train to accommodate the feedwater heating/condensate cooling at one end and steam expansion/steam condensing at the other end of the streams. The condensate extracted at 49°C (120°F) is allowed to throttle into the feed stream to make up the reforming ratio of about three parts of steam to one part of methane-rich gas. The remaining moisture in the syngas is removed through a series of glycol driers. The dry gas is sent to the pipeline at about 49°C, 36-bar (120°F, 530 psia).

Several desirable plant features were identified and incorporated into the reference design, including independence of the turbine-generator, independence of the reformer gas plant, minimum waste heat rejection, and flexibility of plant operation.

The methane reformer process is provided with subsystems for startup, shutdown, and gas stream preparation. These subsystems are:

- <u>Gas Purification</u> - Gas purification is required to clean up the odorant placed in the pipeline to meet pipeline transmission requirements. The odorant selected is a 50-50 mixture of tert-butyl mercaptan and dimethyl sulfide. The odorant is present at a concentration equivalent to 5 ppmv of sulfur. The steam-hydrocarbon process is carried out using a nickel catalyst that is sensitive to sulfur. Concentrations in excess of 1 ppmv will cause a significant activity loss. The production of reformed gas requires, therefore, the thorough desulfurization of the reformer feed.

  Chemically treated activated carbon is used for reformer feed desulfurization. The odorant-containing feed is contacted with the treated carbon at ambient temperatures and the sulfur removed by chemical reaction and absorption to less than 1 ppmv. The carbon is regenerated periodically by stripping with steam to remove the odorant.

  The gas purification subsystem for each of the four process lines consists of nine activated carbon beds connected in parallel. The beds are sequentially regenerated with superheated steam such that one bed is always off-line in the regeneration mode while the remaining eight beds perform the purification function. Each bed is housed in a 1.2 meter (4-ft) diameter x 2.4 meter (8-ft) high carbon steel tank with removable 10-20-mesh rigid screens top and bottom to contain the carbon. The operation of each individual bed may be represented as shown in Fig. 5.1.3-3.

- Gas Drying - In order to avoid pipeline corrosion, no condensation of water from the gas being transported can be allowed. The gas is, therefore, dried to a water content of 0.02%, which is equivalent to a dewpoint of 4.4°C (40°F) at 36-bar (530 psia). The process gas is dried using a glycol dehydration process. In this process, water is absorbed from the process gas by a countercurrent flow of triethylene glycol (TEG). The dried gas exits the process through a gas-to-gas heat exchanger that heats the wet inlet gas and cools the dry outlet gas; the wet TEG flows to a reboiler and distillation column where the water is driven out of the TEG. The dry TEG is pumped back to a contactor to dry more process gas while the water separated from the TEG is discarded due to traces of TEG present.

  The equipment required for each of the four process lines will consist of the following:

  4 gas inlet scrubbers

  2 glycol-gas contactors

  6 glycol reconcentrators, each of which includes:

  1 glycol reboiler
  1 stripping still with reflux condensor
  1 heat exchanger surge tank
  1 glycol recirculation pump

  6 glycol flash separators

- Nitrogen Purge - Nitrogen purge is required to avoid catalyst degradation and formation of toxic nickel carbonyl. The reformer catalyst is heated to its initial reforming reaction temperature in a nitrogen atmosphere. The nitrogen purge system for each of the four process lines must be capable of supplying sufficient nitrogen volume to purge all reformer-related equipment up to the pipeline--equal to approximately 1.84 million liters (65,000 cubic feet). Purging with nitrogen is a nonroutine operation that occurs only during initial startup, shutdown, and equipment maintenance. When specific equipment items are undergoing repair, they are isolated and purged separately to minimize nitrogen usage and operational problems. The purge system consists of a tube trailer, either thermochemical-pipeline-owned or leased, containing up to 3.17 million liters (112,000 standard cubic feet) of nitrogen gas. The trailer is mobile and can be moved to various major equipment items. Where practical, gas manifolds are in place so that specific equipment items may be isolated.

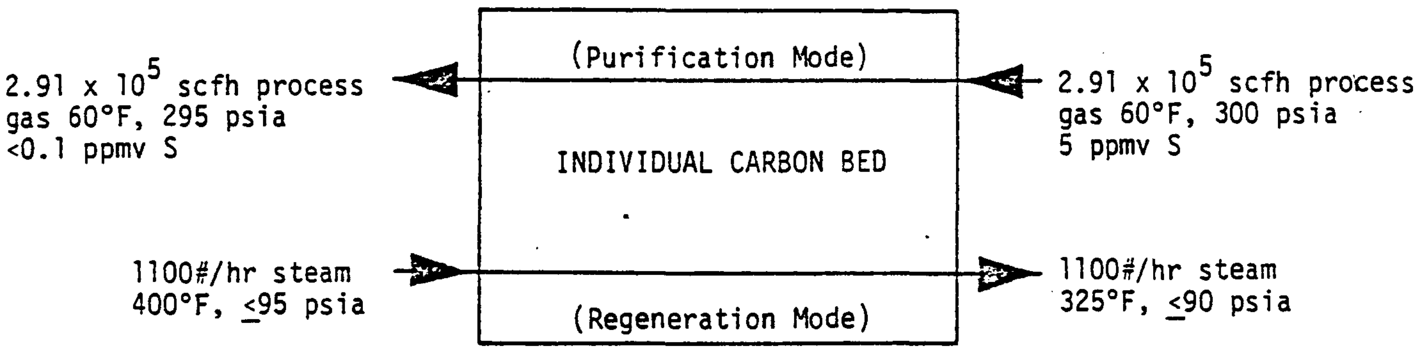

Figure 5.1.3-3 Gas Purification System

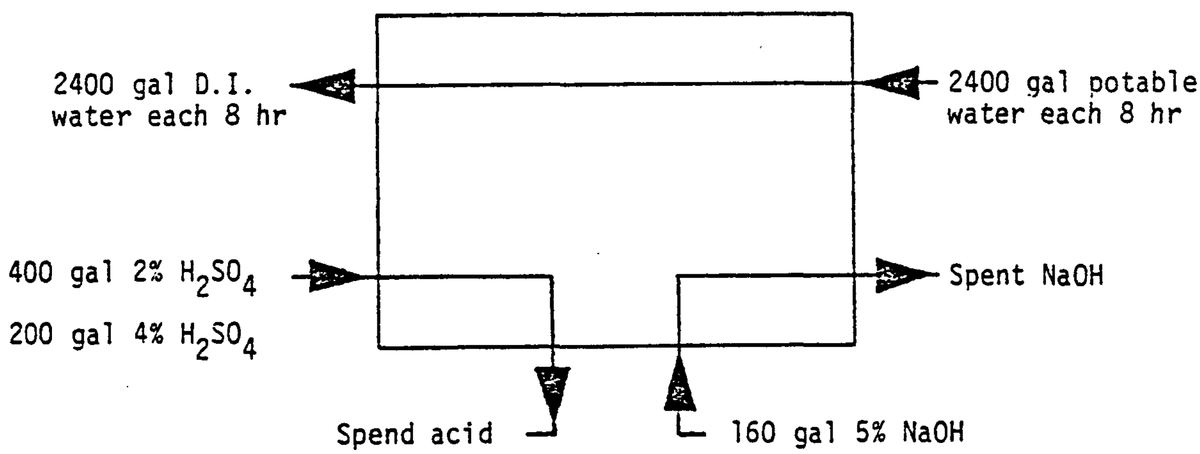

Figure 5.1.3-4 Water Purification System

- Water Treatment - Water treatment is required for the makeup water added to the thermochemical pipeline system and must be of a quality suitable as steam for use in steam turbines and that will cause no catalyst desensitization or equipment corrosion. The water quality will be assured by use of an ion exchange process to generate deionized water from domestic potable water. Each of the four process lines requires two .3 meter (1-ft) diameter x 4.6 meter (15-ft) high beds of ion exchange resin. Each bed is one-half anion exchange resin and one-half cation exchange resin. The beds are plumbed in parallel such that one bed is in the purification mode while the other bed is being regenerated. The resin is regenerated with 2% and 4% sulfuric acid and with 5% sodium hydroxide. Each bed operates in the purification mode for 8 hrs, then is off-line for regeneration for 8 hrs. The operation of each bed may be represented as shown in Fig. 5.1.3-4.

5.1.3.1.2 Reformer Design Description

The reformer design concept is illustrated in Fig. 5.1.3-5, with cross-section details on Fig. 5.1.3-6. The reformer design data are summarized in Table 5.1.3-1. The reformer tubes containing the catalyst are suspended from a flat tubesheet, which in turn is suspended from a thermal sleeve upon which is imposed the temperature gradient to the prestressed concrete pressure vessel (PCPV). The tubes are free to expand axially but are laterally supported by six flow baffles mounted in a flow shroud.

The helium enters the tube bundle from below and flows in a serpentine path to the exit nozzle. The process gas enters a 25.4-cm (10-in) inlet nozzle to the dome plenum; the steam-methane gas mixture enters each reformer tube where the reforming process takes place in the presence of nickel catalyst. The process gas continues through the catalyst bed and then returns through the straight pigtail to a segmented manifold and the outlet nozzle.

The catalyst tubes are made from Alloy 800H, which was selected because of its relatively good rupture strength, long-term creep strength, and weldability, and because it is a code-approved material for high-temperature applications with accepted allowable stresses. The reformer tube size was based on Boiler and Pressure Vessel Code (B&PV) Section III Case 1592 primary stress criteria for Alloy 800H for the limiting condition of normal operation for 300,000-hr (40-yr) life. The tube wall thickness to outside diameter (t/D_o) for the design conditions was determined to be 0.115. The tube diameter and number of tubes were selected based on the required heat transfer to achieve the optimum performance for the least amount of tubing material required. The reformer tubes neck down slightly to pass through the tubesheet; this is done to achieve reasonable tubesheet ligament efficiencies while maintaining simple and inexpensive tube-to-tubesheet weld joints. Since the tubesheet is fabricated from 316H stainless steel, the reformer tube requires a transition weld and a 316H spool piece for

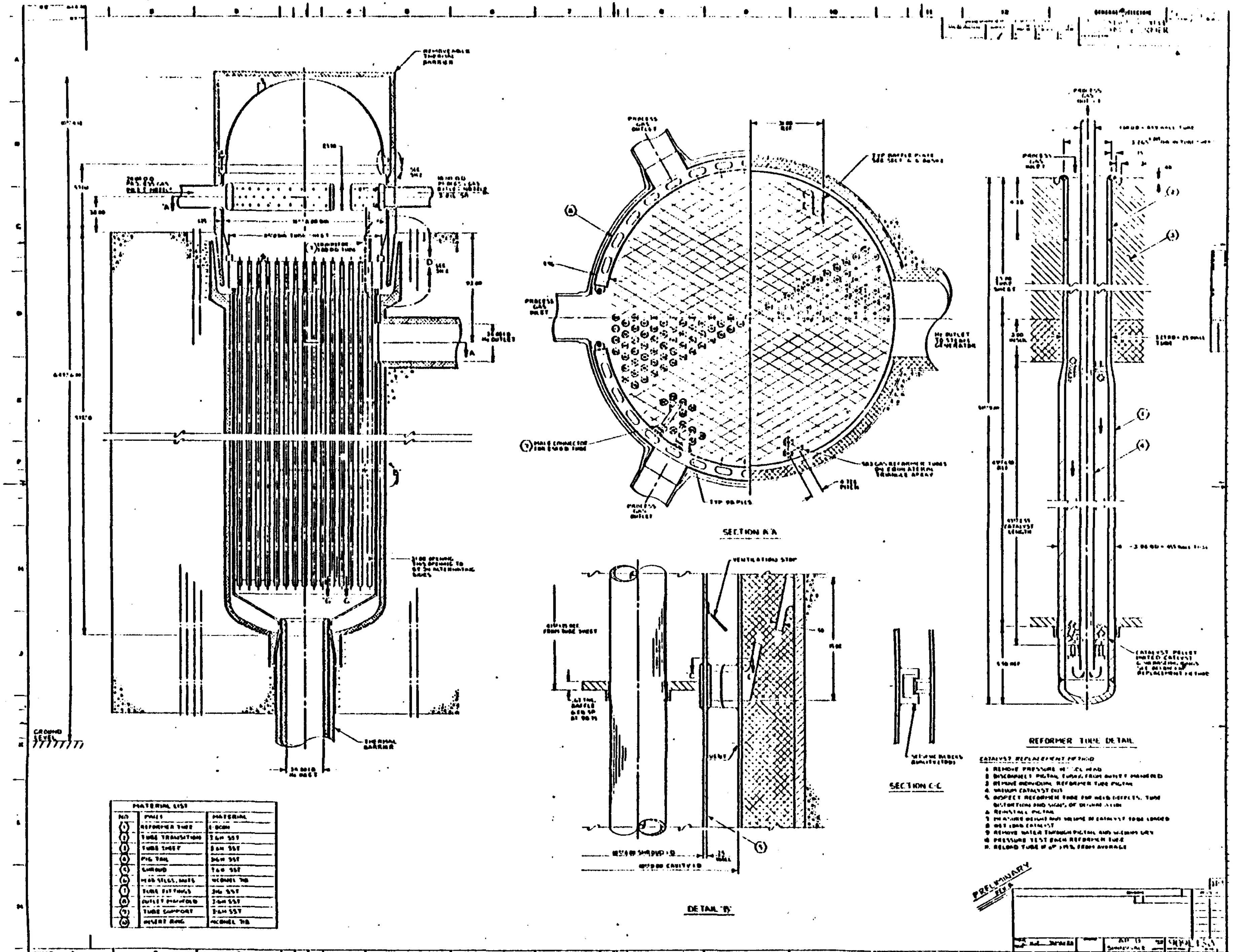

Figure 5.1.3-5   Reformer Design Concept (See 11" x 17" drawing in Appendix C)

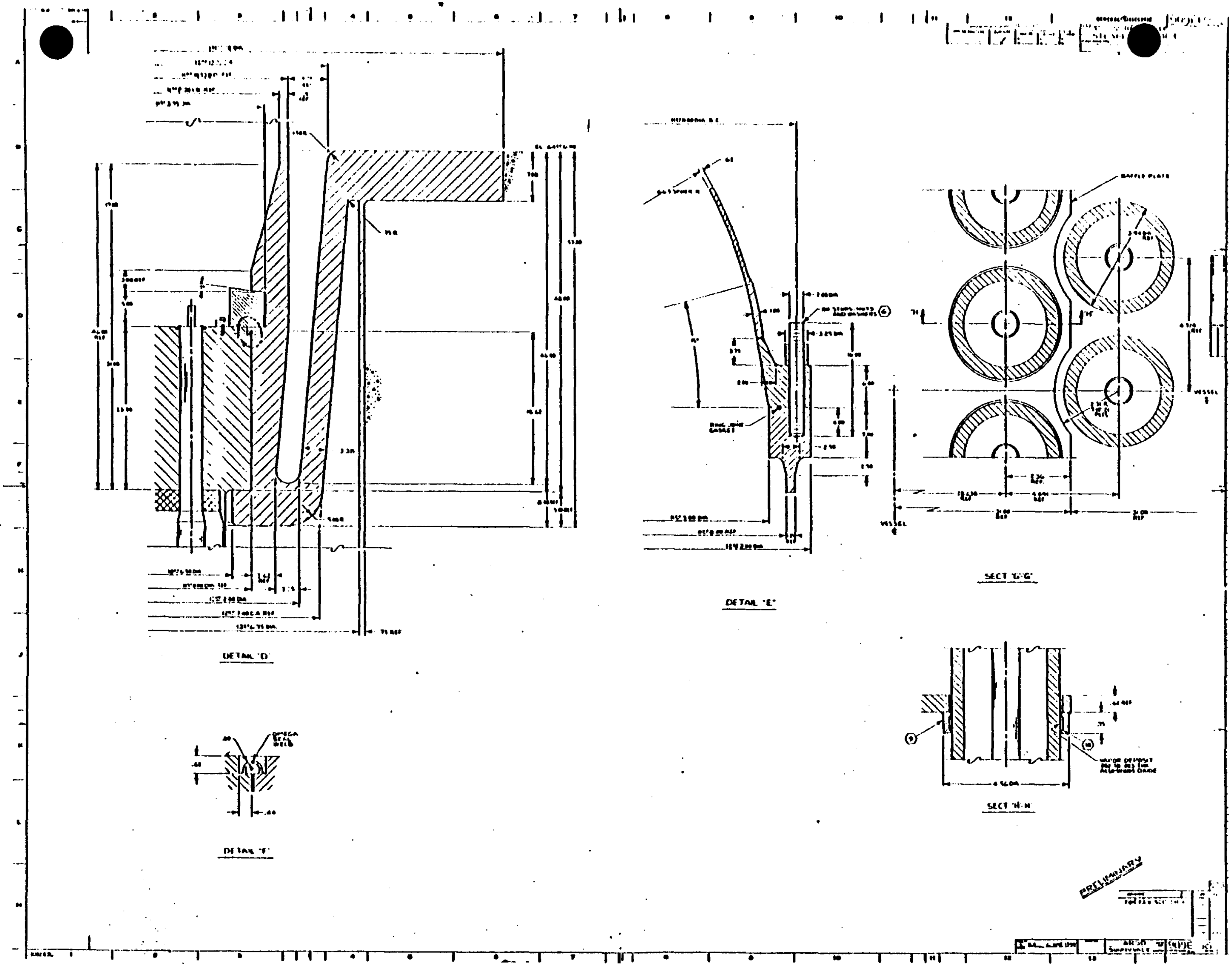

Figure 5.1.3-6 Details of Reformer Design Concept

(See 11" x 17" drawing in Appendix C)

TABLE 5.1.3-1

DESIGN SUMMARY FOR 1170-MW HTGR-R 850°C INDIRECT CYCLE STEAM REFORMER

	HELIUM		PROCESS GAS	
FUNCTIONAL AND STRUCTURAL REQUIREMENTS				
TOTAL HEAT TRANSFERRED AT 100% POWER, MW	511.4		511.4	
FLOW RATES, kg/s (10^6 lb/hr)	513.6	(4.07)	216	(1.71)
INLET TEMPERATURE, °C (°F)	791	(1456)	450	(842)
OUTLET TEMPERATURE, °C (°F)	598	(1108)	564	(1047)
INLET PRESSURE, bar (psia)	48.7	(716)	15	(220)
PRESSURE DROP, bar (psia)	0.67	(9.8)	4.5	(66)
NUMBER OF REFORMERS	8 (2 per loop)			
NUMBER OF REFORMER TUBES (per reformer)	583			
REFORMER TUBE DIMENSIONS				
DIAMETER, OD, mm (in.)	100	(3.9)		
WALL THICKNESS, mm (in.)	14.8	(0.6)		
LENGTH (active catalyst), m (ft)	15	(49)		
REFORMER TUBE PITCH, mm (in.)	120	(4.7)		
REFORMER DIMENSIONS				
DIAMETER (cavity), m (ft)	3.31	(10.9)		
HEIGHT, m (ft)	17.54	(57.5)		
REFORMER TUBE MATERIAL	Ni-Fe-Cr Alloy 800H			
REFORMER TUBE DESIGN LIFE ESTIMATE, hr (yr)	280,000 (40)			
PIGTAIL DIAMETER, mm (in.)	25.4	(1.0)		
PIGTAIL MATERIAL	316H SS			
CATALYST TYPE	G-90 United Catalyst or Equivalent			
CATALYST DIMENSIONS, mm (in.)	6.3 x 6.3 (.25 x .25)			
CATALYST VOID FACTION	0.50			
CATALYST HEAT TRANSFER CORRELATION	Modified Kunii			
CATALYST LIFE, yr	7-10			
CATALYST REPLACEMENT METHOD	Remove Pigtail/Vacuum			
METHANE CONVERSION, %	69-70.0			
ASME CODE CLASSIFICATION	Section III, Class 3 (Design Based on cc N-47)			
BUNDLE REMOVAL CAPABILITY	Yes			
MAXIMUM TIME ALLOWED FOR CATALYST, days/yr REPLACEMENT AND TUBE INSPECTION	21			

a reliable tube-to-tubesheet joint. The tube is welded to the tubesheet utilizing a trepan type weld. This joint is expected to provide long reliable life under transient service conditions. This joint also provides for easy access for weld inspection and repair.

To restrict seismically induced loads on the tube bundle, seismic motion restraints are installed in the stagnant helium annulus between the shroud and the PCPV cavity liner. These restraints permit vertical and radial thermal growth of the shroud and liner without permitting lateral deflection of the bundle. This design also permits removal of the tube bundle and shroud for rework and repair should it be necessary. If a reformer tube weld should fail in service, the catalyst in that tube would be removed and the tube capped at the tubesheet.

The loading on the tubesheet results primarily from the unbalanced pressures on opposite sides [49 bars (716 psia) on the helium side, 15 bars (220 psia) on the process gas side]. This normal operating pressure unbalance is restrained with a shear ring arrangement. In addition, an omega-shaped seal weld is utilized to seal the secondary helium from the process gas. The tubesheet, shear ring, and omega seal are designed to accommodate the short-term upset conditions of full pressure differential should the process gas pressure be suddenly lost.

The reformer sizing was accomplished using the DSR1 computer code (Ref. 1) to model the steady-state behavior of each reformer bundle. The accuracy of this code was verified by single tube tests conducted at the Nuclear Research Center in Juelich, Federal Republic of Germany (Ref. 2). The catalyst employed in the reformer was assumed to be similar to that which was used in the tests in Germany, raschig rings of nominal dimensions 6.3-mm diameter by 6.3-mm long.

The process gas enters the reformer at 440°C and 15 bar. Heat exchange between the process gas and the product gas return tubes in the upper plenum raises the process gas temperature to 450°C before it reaches the catalyst. The reforming temperature and pressure (at the bottom of the catalyst bed) are 751°C and 11 bar. This results in a methane conversion of 70.0%. The maximum theoretical conversion obtainable at this temperature and pressure is 72.8% based on equilibrium. This product gas then moves up the internal return tube and provides 120 MW of regenerative heat exchange to the process gas in the catalyst bed. The product gas exits the reformer at 554°C and 10.25 bar. Helium enters the reformer at 791°C and 48.7 bar and exits at 599°C and 48.0 bar. A total of 511 MW is transferred to the process gas in the reformer, and an additional 80 MW is transferred to the process gas via the process gas compressors.

5.1.3.1.3 <u>Prestressed Concrete Pressure Vessel</u>

The reformers and steam generator in each secondary loop are enclosed in a prestressed concrete pressure vessel (PCPV). There are four separate process loops in the process plant. Each loop contains a PCPV system consisting of the PCPV structure, the steel cavity liners, the steel penetration and closures, and the thermal barrier. The function of the PCPV system is to house the two reformers and the steam generator of a process loop. The major components of a PCPV system are discussed below. Fig. 5.1.3-7 shows plan and elevation views of the PCPV.

The PCPV structure is a multicavity vessel of prestressed concrete characterized by two reformer cavities and a steam generator cavity arranged in a triangular array. The vessel is prestressed circumferentially by wound strand cables and vertically by linear strand tendons. These two prestressing systems provide sufficient precompression in the concrete to resist the secondary system pressure loads during the vessel life. Table 5.1.3-2 gives PCPV data for the process loop application.

The steel liners and the closures at the penetrations form the continuous gas-tight boundary of the PCPV. Penetrations and closures act with the concrete to resist secondary coolant pressures. The liner and penetration anchors transmit loads from internal equipment supports and steel closures to the PCPV concrete structure. A liner cooling system is included to remove the heat which passes through the thermal barrier.

The thermal barrier minimizes heat losses from the secondary helium system and maintains the PCPV liner and concrete temperatures within acceptable limits. Typically, the thermal barrier consists of layers of fibrous insulation compressed against the PCPV liner by metal cover plates, which are in turn attached to the PCPV liner. Different types of thermal barrier are used throughout the inner surfaces of the PCPV, depending primarily on local gas temperatures.

The pressure relief system consists of relief valves, piping, and other equipment required to limit the PCPV maximum cavity pressure (MCP) to a specified value and to limit the rate of pressure relief flow from the PCPV. The MCP value is given in Table 5.1.3-2.

5.1.3.2 Pipeline/Storage System

The major factors governing the design of the gas pipeline and storage facilities are (1) the rate of generation of process gas on a continuous basis, (2) the rate of usage, and (3) the gas composition. The composition of the gases, as noted earlier, emerged from the reformer studies at a 70% conversion of methane. For the process heat application, the rate of usage was specified as equivalent to the liberation of 1500 MW(t)/hr for eight hrs/day, seven days/wk. In the electrical load-following application, consumption of gas energy was specified as equivalent to the generation of 450 MW(e) for eight-hrs/day.

The quantity of gas storage was specified as approximately 8000 MW(t) hrs of storage for all cases, sufficient to meet the eight-hrs/day maximum gas flow demands. Three possible variations of storage and user locations along the length of a 160-km (100-mi) pipeline were

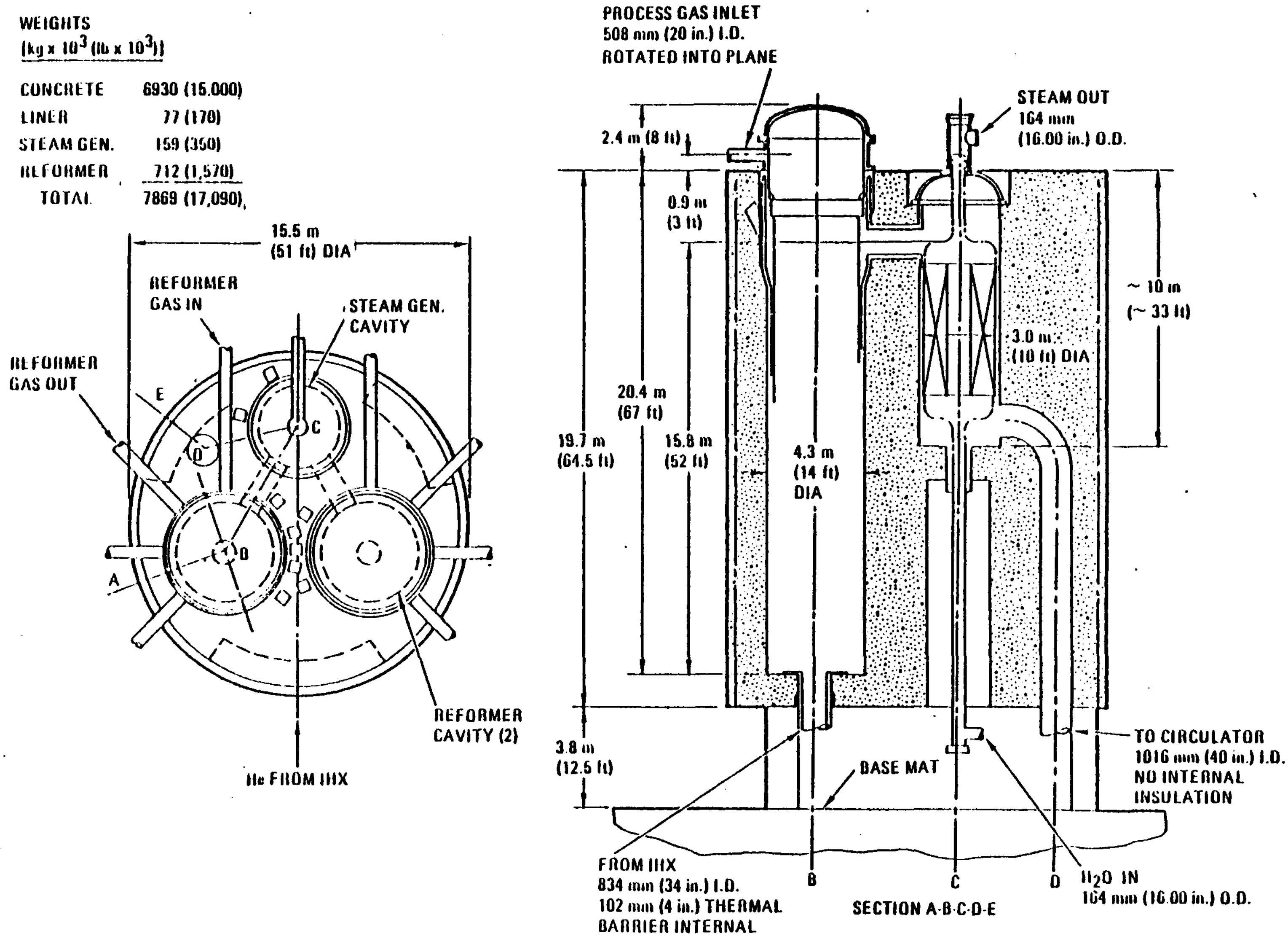

Figure 5.1.3-7 HTGR-R reformer loop PCPV

TABLE 5.1.3-2

PCPV MAJOR PARAMETERS

Type	Multicavity
Overall dimensions	
Diameter, m (ft)	15.5 (51.0)
Height, m (ft)	20.4 (67.0)
Reformer cavity, number	2
Diameter, m (ft)	4.3 (14.0)
Depth, m (ft)	18.49 (60.67)
Steam generator, number	1
Diameter, m (ft)	3.05 (10.00)
Depth, m (ft)	~10.06 (~33.00)
Maximum cavity pressure, MPa (psia)	5.35 (776)

were specified for the process heat user application. The peaking electricity plant was specified to be within 32 km (20 mi) from the nuclear site so that a sufficiently large area was available within which to select the best geologic storage formations. These four system configurations are presented in Fig. 5.1.3-8. The three variations in storage location for process heat applications are considered to envelope the range of technical difficulties and costs to be expected over a range of possible gas system sites.

5.1.3.2.1 Pipeline Systems Design

Three transmission pipelines are required for the HTGR-R closed-loop thermochemical pipeline. These pipelines are a synthesis gas pipeline from the reformer to the methanator, a methane-rich gas pipeline from the methanation facility to the reformer, and a water-return line from the methanator to the reformer.

To determine the optimal system configurations and hence optimized system costs, a gaseous and liquid transmission pipeline computer model was used by the Institute of Gas Technology to simulate the three transmission pipelines. Only standard size steel pipeline diameters were used to calculate the optimum pipe diameters and pipeline pressures. The pipeline design optimization computer model input is a combination of technical and economic factors. The technical factors of flow and gas composition constraining the range of pipeline parameter variations are given on Figs. 5.1.3-9 and 5.1.3-10.

Reciprocating compressors, either single or multistage, common in pipeline service, were selected for service for both the synthesis gas and methane-rich gas transmission because of their simplicity and adaptability.

The pipeline system was designed based on each unique section of the pipeline system for each of the four system configurations studied. A summary of the pipeline specifications for the peaking electricity plant located within 32 kilometers (20 miles) of the nuclear site (Configuration I) and the process heat user application with industrial park systems sited 53 kilometers (33 miles) apart along a 160-kilometer (100-mile) pipeline and storage (Configuration IV) is given on Figs. 5.1.3-11 and 5.1.3-12, respectively.

5.1.3.2.2 Storage Design

The selection and design of a cavern for gas storage is extremely site-dependent. No firm site has been selected for the purposes of this study. Therefore, it was assumed that proper geologic structures will occur at any desired site and depth. Base-case storage parameters for synthesis gas and methane-rich gas facilities are shown in Table 5.1.3-3. The upper limit for gas pressure was assumed to be 82 bar (1200 psia), which is set to minimize potential hydrogen embrittlement effects for the syngas mixture. The lower pressure limit for gas pressure was assumed to be 27 bar (400 psia), which was set to minimize potential salt creep when salt dome storage is used.

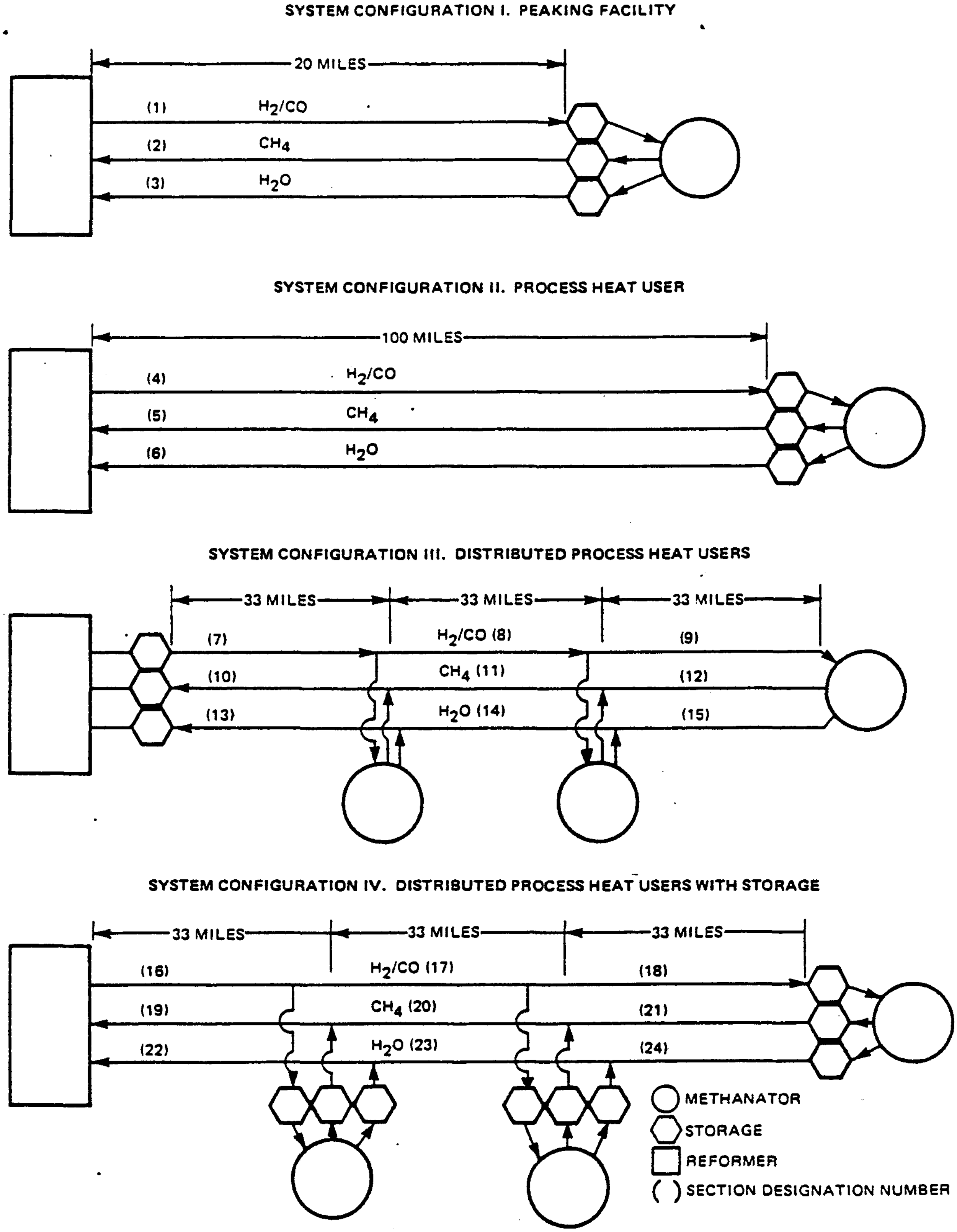

Figure 5.1.3-8 HTGR Closed-Loop Thermochemical Pipeline System Configurations

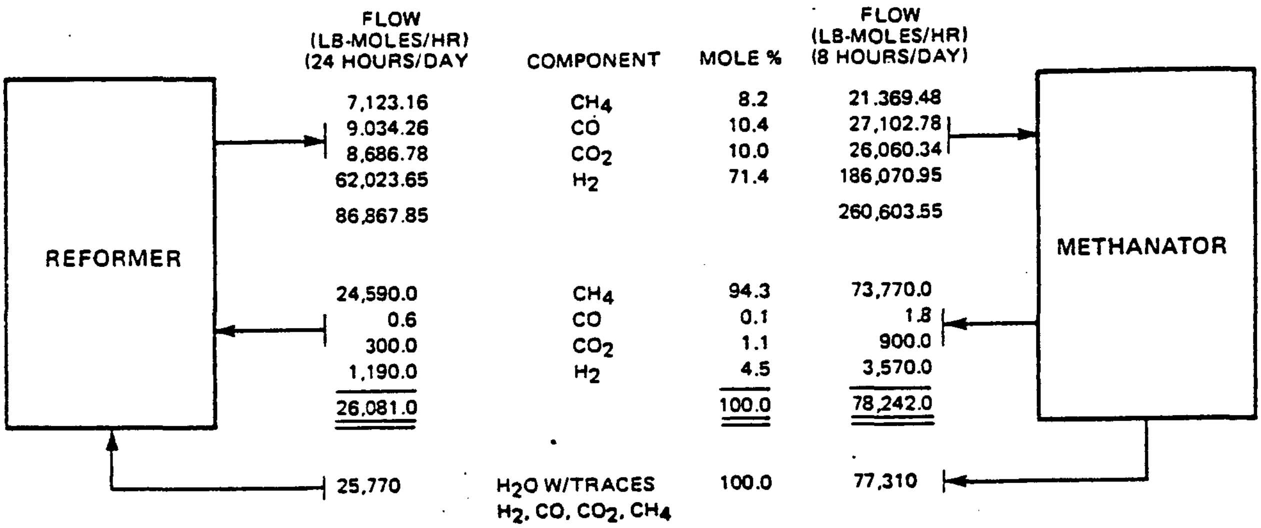

Figure 5.1.3-9 Pipeline and Storage System Design Constraints (Configurations I and II)

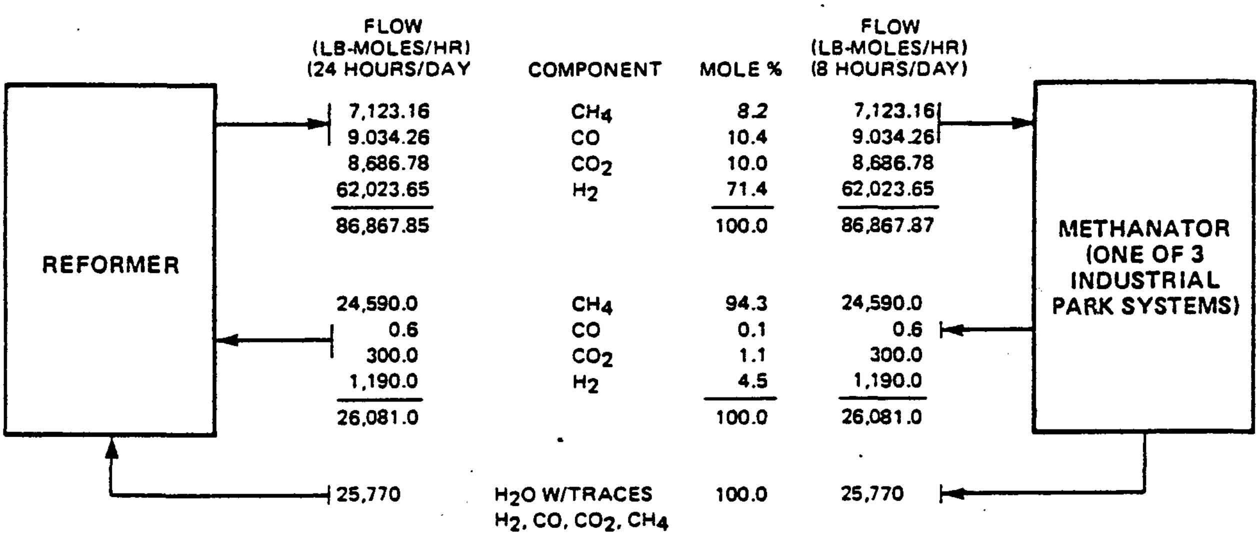

Figure 5.1.3-10 Pipeline and Storage System Design Constraints (Configurations III and IV)

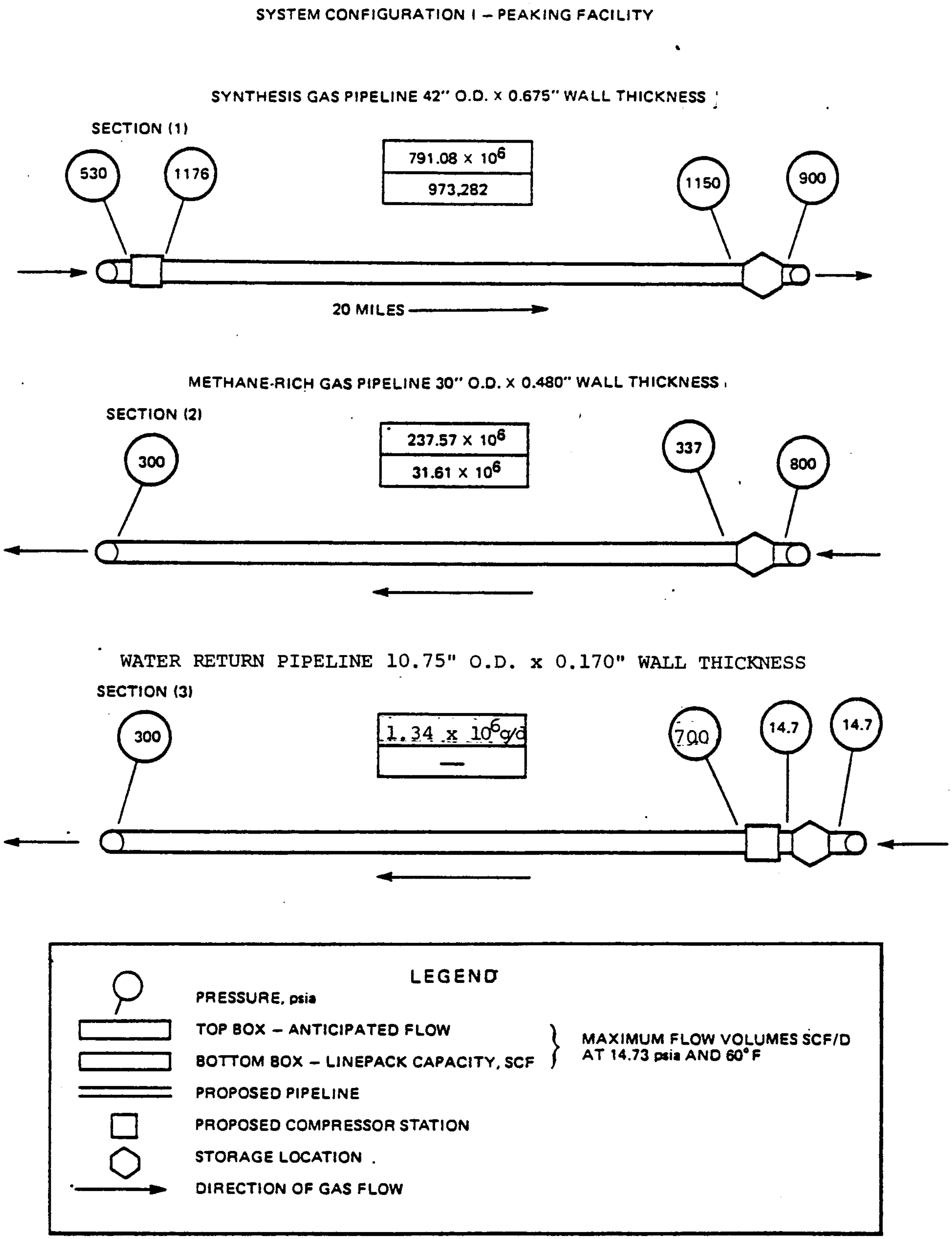

Figure 5.1.3-11 Pipeline and Storage System Design for the Peaking Electricity Plant (Configuration I)

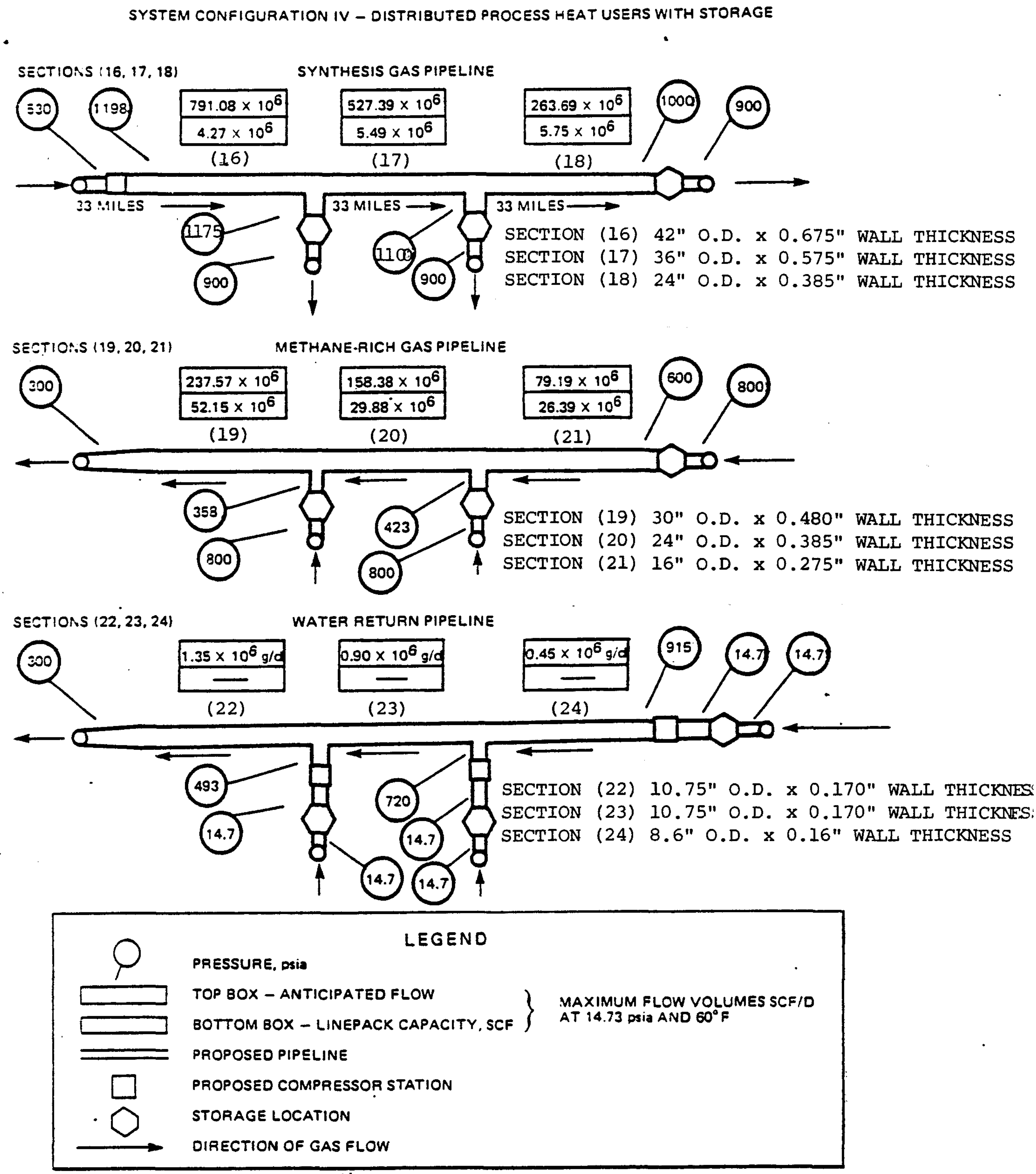

Figure 5.1.3-12 Pipeline and Storage System Design for the Process Heat User Application (Configuration IV)

TABLE 5.1.3-3

BASE-CASE STORAGE PARAMETERS

Gas Type	Working Gas, lb-mol	Line Pressure for Injection Withdrawal, psia	Cycle Times, hr Injection	Cycle Times, hr Withdrawal	Cavern Temperature, °F
Syngas	1.36×10^6	1000/1000	16	8	100
Methane	$.417 \times 10^6$	600/500	8	16	

TABLE 5.1.3-4

WATER STORAGE REQUIREMENTS

Demand/Day From Storage in 10^6Gal	Storage Facility Size in 10^3bbl (oil)
2.67	63.5
0.89	21.2

Water storage requirements are noted in Table 5.1.3-4. Two thirds of total water demand will come from onsite storage cycled each day. Because a hydraulic head is not required for subsequent distribution as in municipal water storage tanks, ground-erected structures similar to API oil storage tanks were selected.

5.1.3.3 Methanation Plant

Methanation plant facilities for both the load-following electricity application and the process heat application are similar in arrangement and the number of components in a train (six methanator units/train). Fig. 5.1.3-13 shows the arrangement of the six methanator units to form one methanation train for the load-following electricity application. A similar arrangement is shown for the process heat application in Fig. 5.1.3-14. The load-following electricity plant consists of six such methanation trains (Fig. 5.1.3-15) to produce the superheated steam for driving the two 225-MW(e) turbine generators. The process heat methanation plant consists of a single methanation train (Fig. 5.1.3-15, note Case II) which has one fifth of the capacity (and hence syngas flow) of an individual train (one of six) incorporated in the load-following electricity plant. The process heat plant generates superheated steam [68 bar/482°C (1000 psia/900°F)] for process heat use. The load-following electricity plant plot plan, layouts of the turbine-generator plant and facilities, and heat balance and flow diagrams are shown in Figs. 5.1.3-16 through 5.13-19.

The syngas feed and the resulting product gas for the methanation plant facilities are:

	Feed	Product
Mol %		
H_2	71.36	4.56
CO	10.67	0.002
CH_4	8.13	94.30
CO_2	9.84	1.14
	100.00	100.00
Pressure, psia	900	800
Temperature, °F	100	100

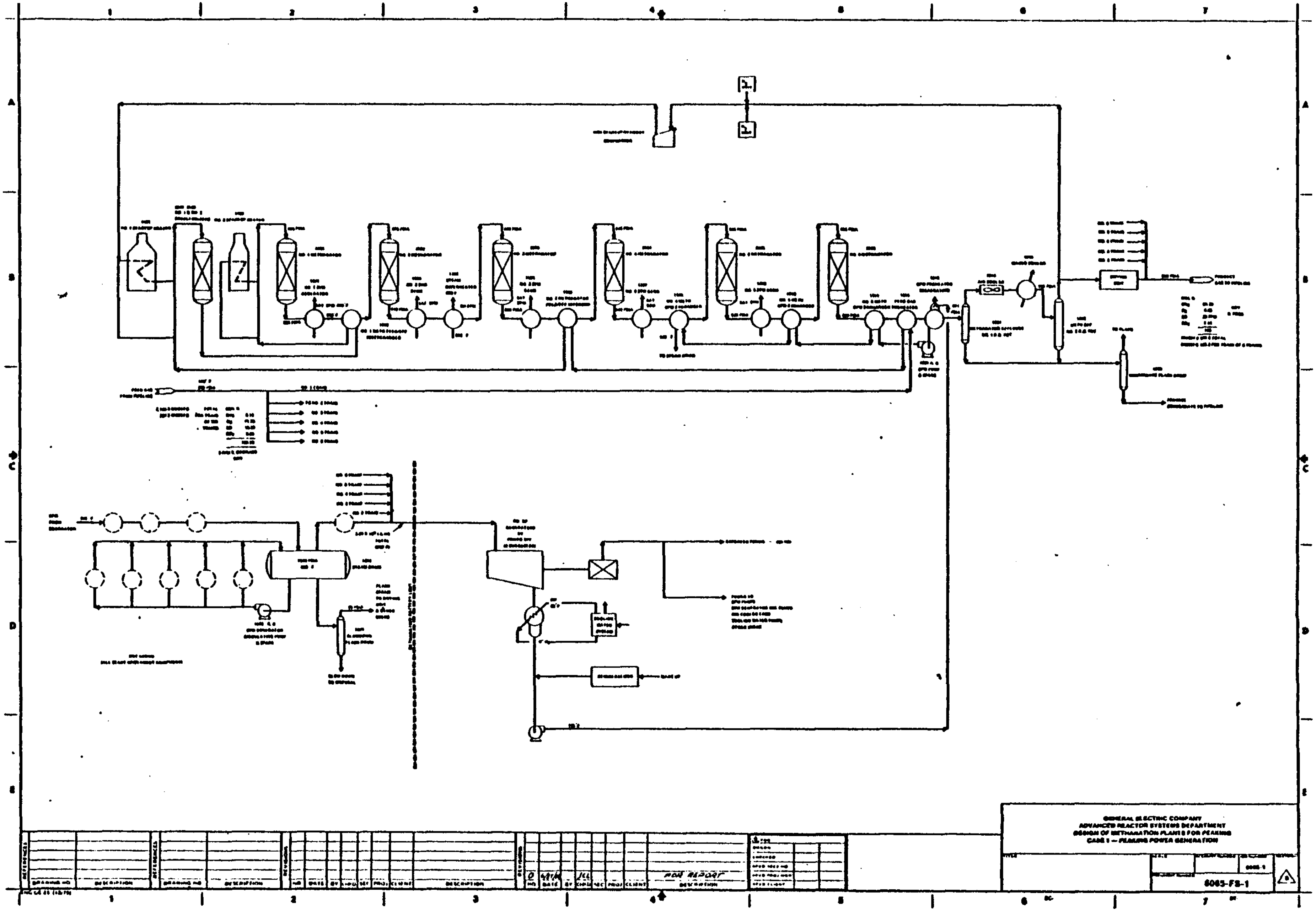

Figure 5.1.3-13 **Methanation Trains with Six Methanators per Train**

(Load-Following Electricity Plant)

(See 11" x 17" drawing in Appendix C)

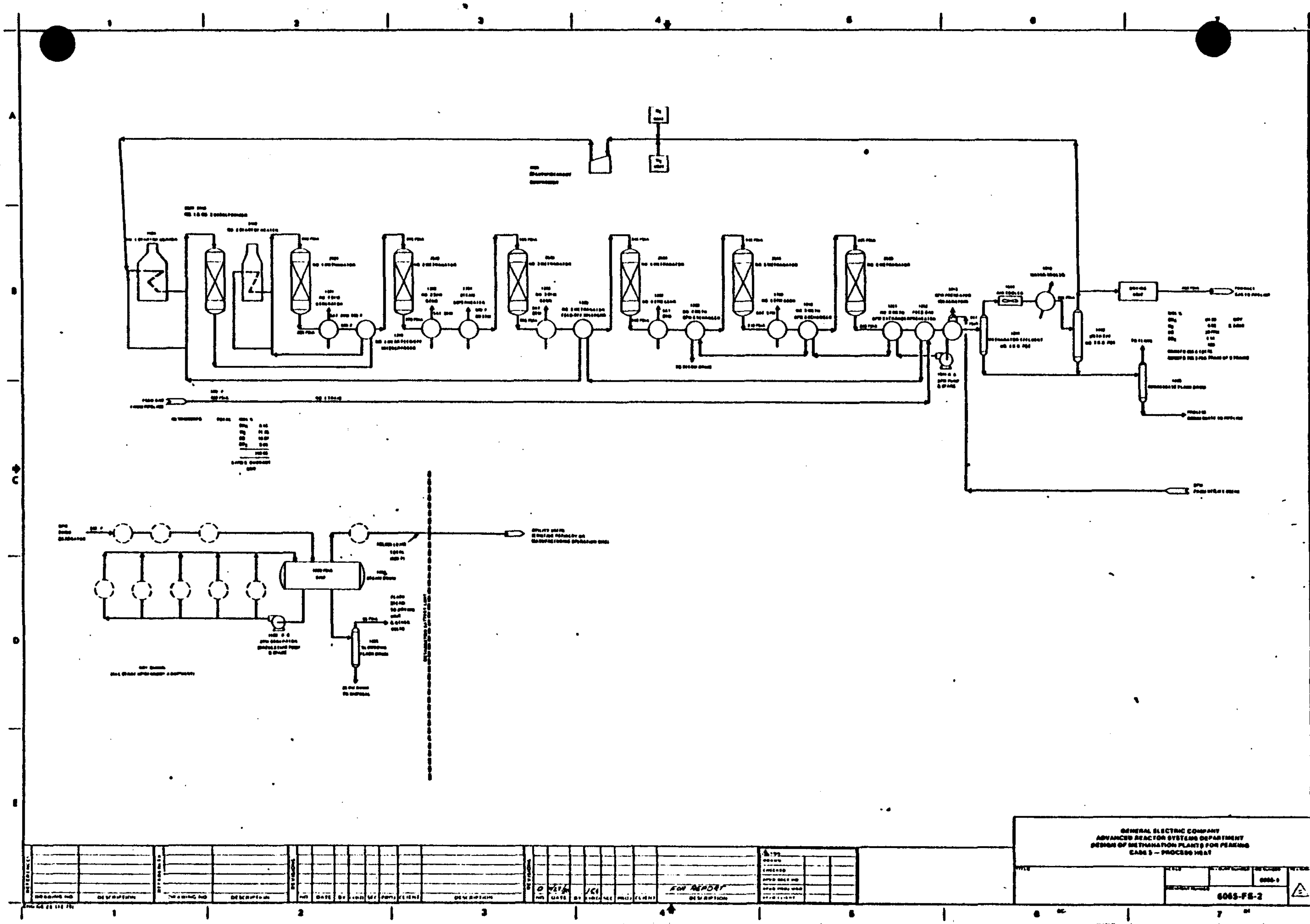

Figure 5.1.3-14 Single Methanation Train with Six Methanators (Process Heat Plant)

(See 11" x 17" drawing in Appendix C)

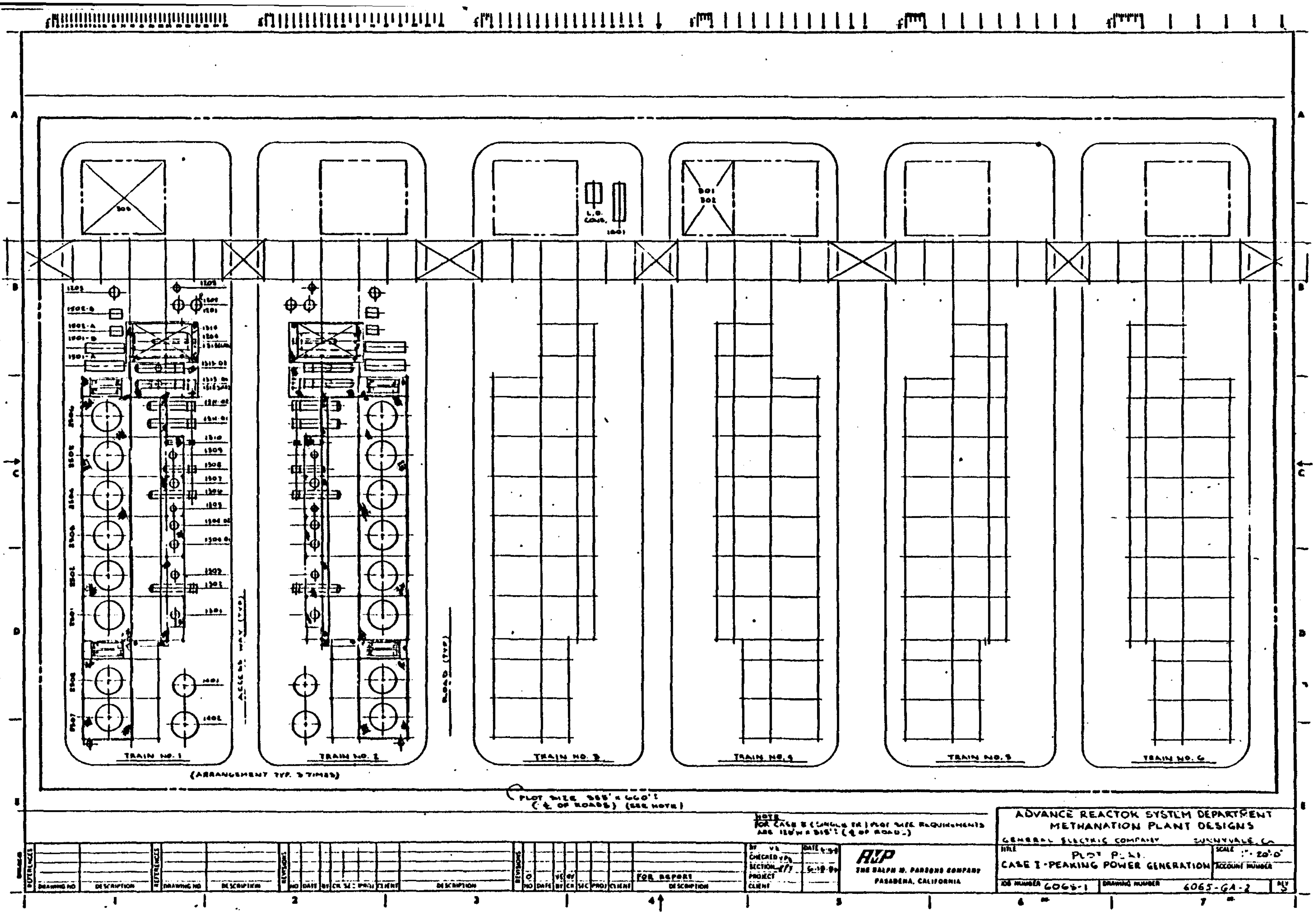

Figure 5.1.3-15   Plot Plan for Load-Following Electricity Plant

(See 11" x 17" drawing in Appendix C)

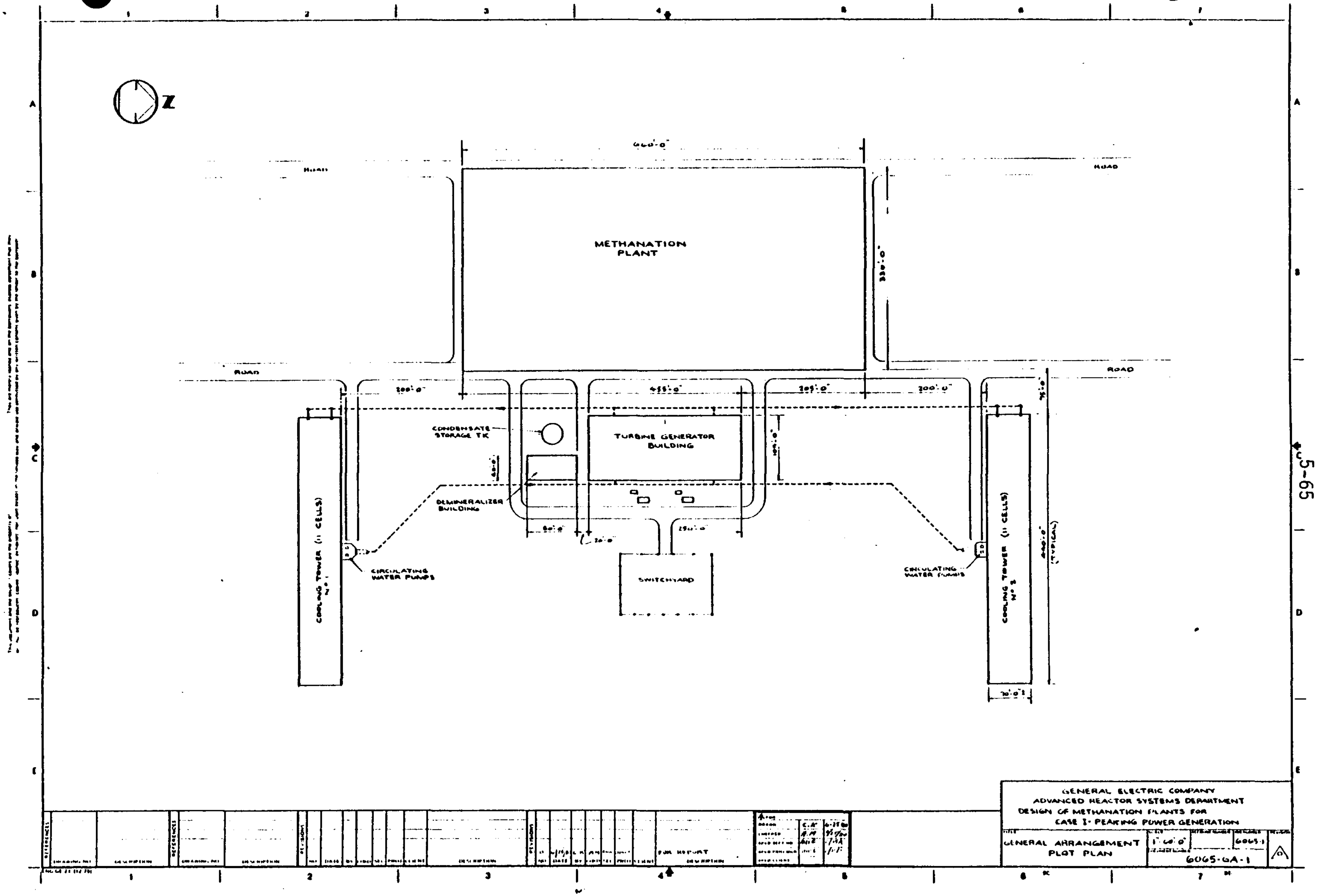

Figure 5.1.3-16 General Arrangement Plot Plan of Load-Following Electricity Plant (See 11" x 17" drawing in Appendix C)

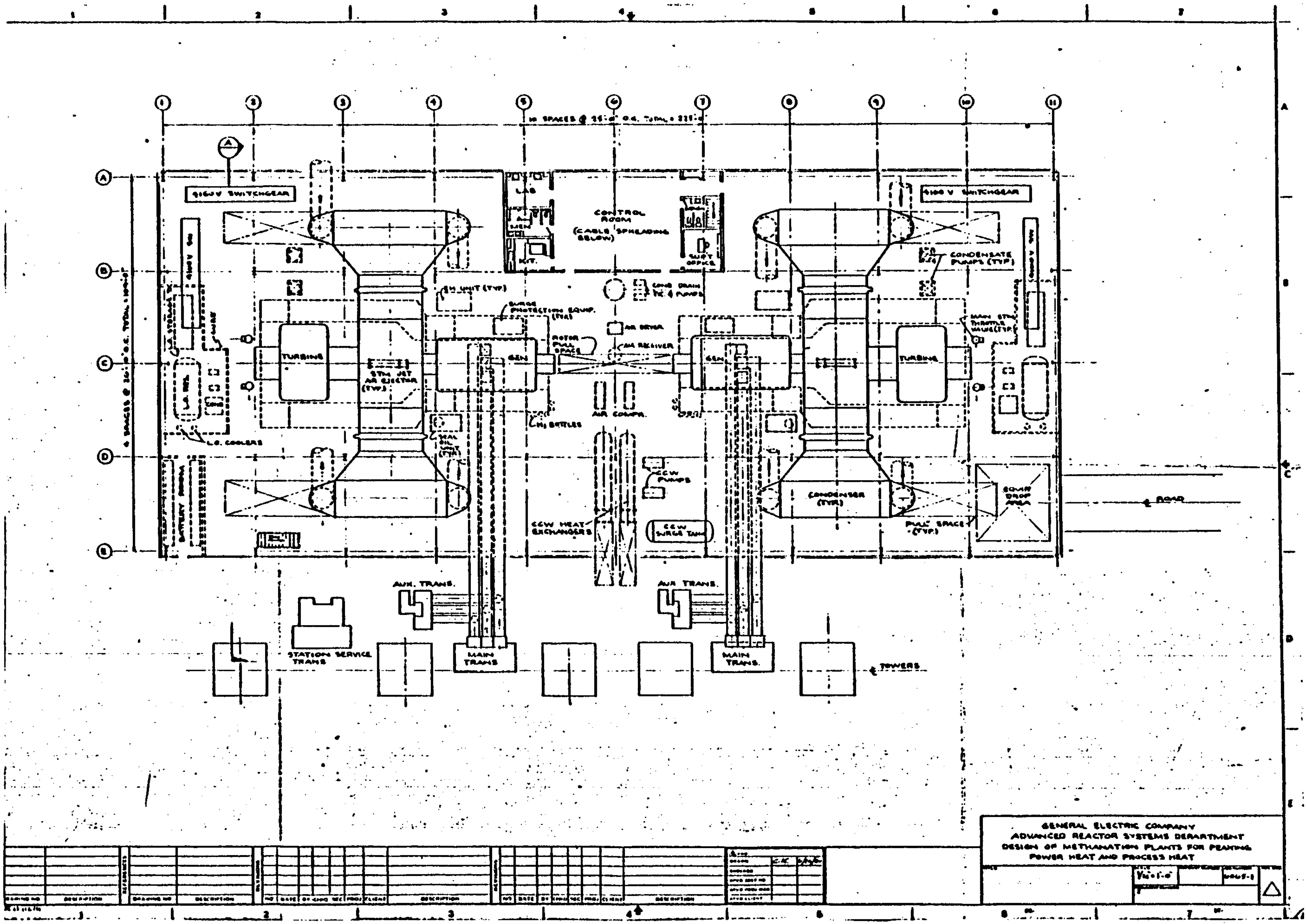

Figure 5.1.3-17   General Arrangement Turbine Building Plan
(See 11" x 17" drawing in Appendix C)

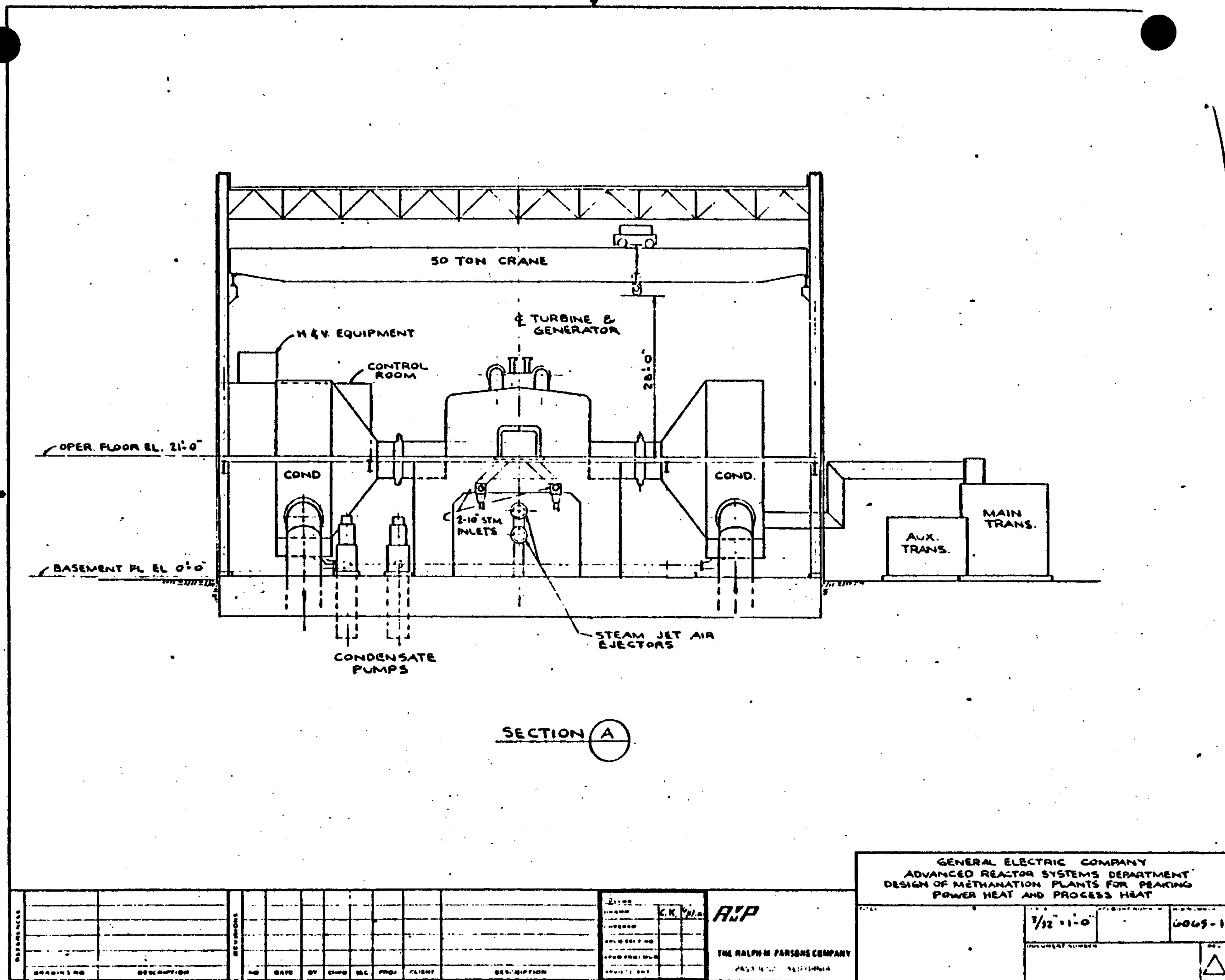

Figure 5.1.3-18 General Arrangement Turbine Building Section
(See 11" x 17" drawing in Appendix C)

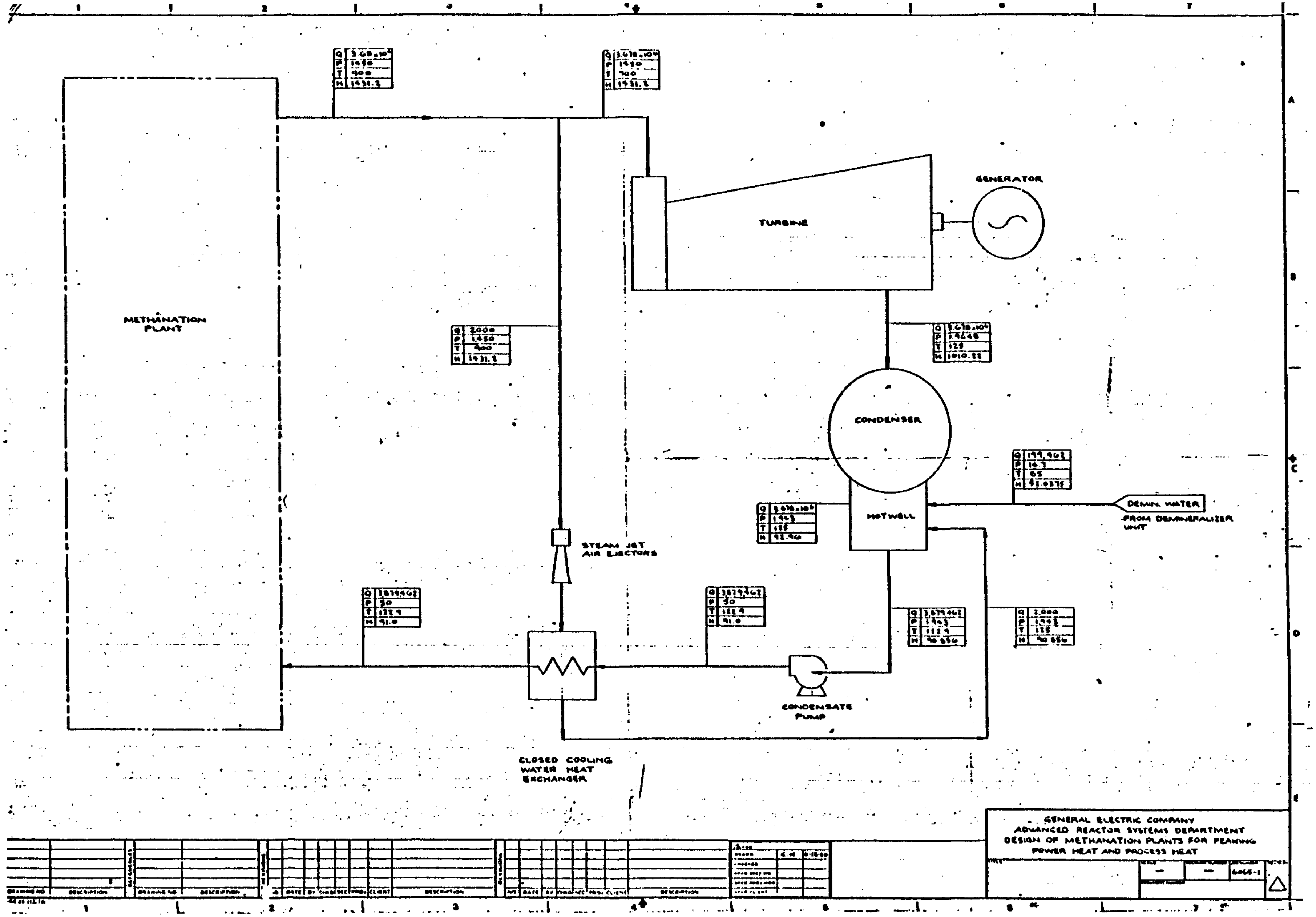

Figure 5.1.3-19 **Heat Balance Full Load**

(See 11" x 17" drawing in Appendix C)

5.1.3.3.1 Methanation Train

The heat released in methanation of the syngas feed to the product methane gas is used:

- To produce high pressure superheated steam to generate electric power in the load-following application.

- To produce high pressure superheated steam to be exported to a variety of one-shift process heat users in the process heat application.

In addition to the six methanation units, each methanation train consists of feed preheat exchangers and vessels containing zinc oxide to remove the odorant (5 ppm) added to the gas in the pipeline. In operation, the desulfurized syngas feed flows through a series of fixed-bed adiabatic catalytic reactors (methanator units). Between these methanator reactors, exothermic heat of reaction is removed from the system by the generation of high-pressure steam and by conventional heat exchange between the feed and effluent streams. As the flow progresses through the six reactors and exchangers and the bulk of the syngas is methanated, the temperature of the syngas feed is progressively lowered, finally resulting in an adequately reduced temperature favorable for achieving a high conversion of hydrogen and carbon oxides to methane (product gas).

Between methanator units and after the sixth methanator, heat is recovered for feed preheat and to recover heat into the steam system, that is, for boiler water heating, steam generation, and steam superheating. After final cooling in an air cooler and water cooler, the condensate is removed and the remaining product gas is dried in a drying unit. The product gas (methane) is then returned to the pipeline.

The methanation trains are assumed to be designed in accordance with refinery and chemical plant codes. Complete ancillary equipment is provided for startup, shutdown, and other procedures. Fired heaters are provided for initial heatup, catalyst reduction, startup and subsequent operation in hot standby mode, and for any subsequent cold startups. A compressor and its auxiliaries are provided for circulating gases for these various operations--nitrogen for heatup, hydrogen for catalyst reduction, and syngas for standby operation. Hot standby operation for peaking electricity will consume pipeline syngas at an approximate rate of 10.4 MMSCFD and power at an approximate 5,300 KW. A package hydrogen unit is included for converting reformed gas to hydrogen for catalyst reduction. Nitrogen facilities for plant purging and startup are included.

The methanation units contain flow and temperature control systems in accordance with the usual refinery designs. Emergency trip systems are included as required, for instance, to protect fired heaters against low flow conditions.

5.1.3.3.2 Methanator Units

The vessels of all six bulk methanator units (fixed-bed adiabatic catalytic reactors) are refractory lined to reduce the temperature of the steel shell, to prevent corrosion of the steel shell, and to reduce heat loss from the shell. The linings consist of two layers of refractory, a layer of lightweight castable for thermal insulation, and a layer of dense and impermeable material for corrosion protection and wear resistance. A thin layer of external insulation is installed on the shell to further reduce heat loss from the shell.

The methanator unit is provided with an inlet gas baffle plate. The catalysts are supported by 1/2"-3/4" alumina balls, which are on a refractory bed support dome. A catalyst dump manway is provided at the top level of the catalyst bed. Two multipoint thermocouples are provided in the catalyst bed for monitoring bed temperatures. A differential pressure cell is provided to indicate pressure drop across the methanator.

5.1.3.3.3 Methanation Catalysts

Methanation catalysts consist of nickel oxide supported on an inert carrier in the range of 15 to 30 weight percent nickel. Catalysts for the methanators are shown in the table below:

	Nos. 1 & 2 Methanators	Nos. 3 & 4 Methanators	Nos. 5 & 6 Methanators
Size	5/8" x 1/4" x 5/16"	1/4" x 1/4"	1/4" x 1/4"
Shape	rings	tablets	tablets

The catalyst is activated by the reduction of nickel oxide to nickel using a reducing gas mixture of approximately 20 to 100% hydrogen. Reducing temperatures are usually in the range of 750°F to 1000°F. When preheating a catalyst before reduction, the heating rate is limited to 100-150°F per hour to avoid damage to the catalyst due to the expansion of gases retained in the pores. Active methanation catalyst is pyrophoric and should be oxidized under controlled conditions before exposure to the atmosphere. The spent catalysts may then be removed and put into containers for sale through the metal recovery market.

5.2 Plant Performance

This section summarizes those design and operational aspects for the HTGR-R plant that are the major contributors to plant performance.

5.2.1 Design Performance

Table 5.2.1-1 presents a summary of the performance parameters that characterize the 1170-MW(t) HTGR-R plant. The plant parameters include a heat balance and system operating conditions for the primary NHS and the secondary BOP systems at the reference design conditions.

A simplified flow and energy distribution diagram for the 1170-MW(t) HTGR-R plant is shown in Section 3.1.1, Fig. 3.1.1-2. The thermal power delivered to the secondary system process heat exchanger is derived from the power generated by the reactor core and the primary helium circulator less NHS heat losses.

The design hot helium temperature at the core outlet is 850°C (1562°F), and the design pressure at the primary helium circulator outlet is 5 MPa (725 psia). Design life of the major components of the plant is 40 yr except for the IHX, which is limited by the operating conditions. The primary side (NHS) helium pressure is balanced with the secondary side (BOP) helium pressure at the IHX hot end to maintain a near-zero pressure differential on the tube wall. Relative thermal energy supply to the process heat exchanger and the steam generator is based on the heat (and temperature) necessary to satisfy the endothermic requirements of the reformer and to produce steam at appropriate conditions to meet the steam and power needs of the reformer plant and produce electricity for export.

The energy block diagram of the HTGR-R plant for both the cases of load-following power generation and process heat supply is shown in Fig. 5.2.1-1. In normal operation, 1170 MW(t) of nuclear energy is produced in the reactor. The primary and the intermediate circulators require 95.6 MW(e) of power to transport the thermal energy to the power plant and the reformer plant. However, 84.3 MW(t) of the circulating power is returned as heat added to the heat transport loops. With a total heat loss of 23.1 MW(t) in the heat transport systems, the thermal energy input to the power plant and the reformer plant is 1231.2 MW(t).

Based on the temperature partition of the intermediate circuit temperature drop, the energy split to the reformers and the steam generators will be 42% and 58%, respectively. The gross turbine generator output is 288.4 MW(e), and 451 MW(t) will be rejected to the cooling tower as waste heat. Approximately 550 MW(t) of the reformer plant input energy will be converted into chemical energy through steam-methane reforming. However, it takes 117 MW(e) of gas compression to separate steam from reformed gas and to transport the syngas to the pipeline; another 35 MW(e) is required to pump syngas to the pipeline pressure of 1200 psia. While most compressor power can be recovered in the gas system heat exchanger train, the pipeline pumping, however, will be lost as heat dissipated into the pipeline and the surrounding environment.

TABLE 5.2.1-1
1170-MW(t) HTGR-R SYSTEM PARAMETERS

NHS heat balance

Core power, MW(t)	1170
Thermal power added by circulators, MW(t)	39.9
Heat losses, MW(t)	10.7
Power to IHX, MW(t)	1199.2

NHS system parameters

	Pressure or Pressure Drop [MPa (psia)]	Temperature [°C (°F)]	Flow [kg/s (lb/hr x 10^6)]
Reactor inlet	4.997 (724.5)	427 (800)	532.2 (4.224)
Reactor outlet	4.906 (711.4)	850 (1562)	532.2 (4.224)
IHX inlet	4.900 (710.3)	842 (1554.4)	532.9 (4.235)
IHX outlet	4.823 (699.3)	412 (775.0)	541.0 (4.294)
Circulator inlet	4.814 (698)	413 (777.2)	541.0 (4.294)
Circulator outlet	5.000 (725)	427 (800.6)	541.0 (4.294)
Pressure drops			
Core	0.090 (13.1)		
IHX	0.076 (11)		
NHS loop total	0.186 (27)		
Bypass flows			
CAHE, hot ducts, etc.			1.39 (0.011)
Purge for cooling IHX structure			7.43 (0.059)

BOP process loop heat balance

IHX power, MW(t)	1199.2
Thermal power added by circulators, MW(t)	44.4
Heat losses, MW(t)	12.4
Power to steam generator, MW(t)	707
Power to reformer, MW(t)	524.2

BOP process loop system parameters

	Pressure or Pressure Drop [MPa (psia)]	Temperature [°C (°F)]	Flow [kg/s (lb/hr x 10^6)]
IHX inlet	4.942 (716.7)	343 (650)	513.6 (4.075)
IHX outlet	4.898 (710.2)	793 (1460)	513.6 (4.075)
Reformer inlet	4.873 (706.4)	791 (1456.0)	513.6 (4.075)
Reformer outlet	4.802 (696.4)	594 (1102)	513.6 (4.075)
Steam generator inlet	4.797 (695.6)	594 (1100.5)	513.6 (4.075)
Steam generator outlet	4.701 (681.6)	327 (623.1)	513.6 (4.075)
Circulator inlet	4.696 (681.0)	327 (621.3)	513.6 (4.075)
Circulator outlet	4.959 (681.0)	344 (650.1)	513.6 (4.075)
Pressure drops			
IHX	0.045 (6.5)		
Reformer	0.069 (10)		
Steam generator	0.097 (14.0)		
Process loop total	0.262 (38.0)		

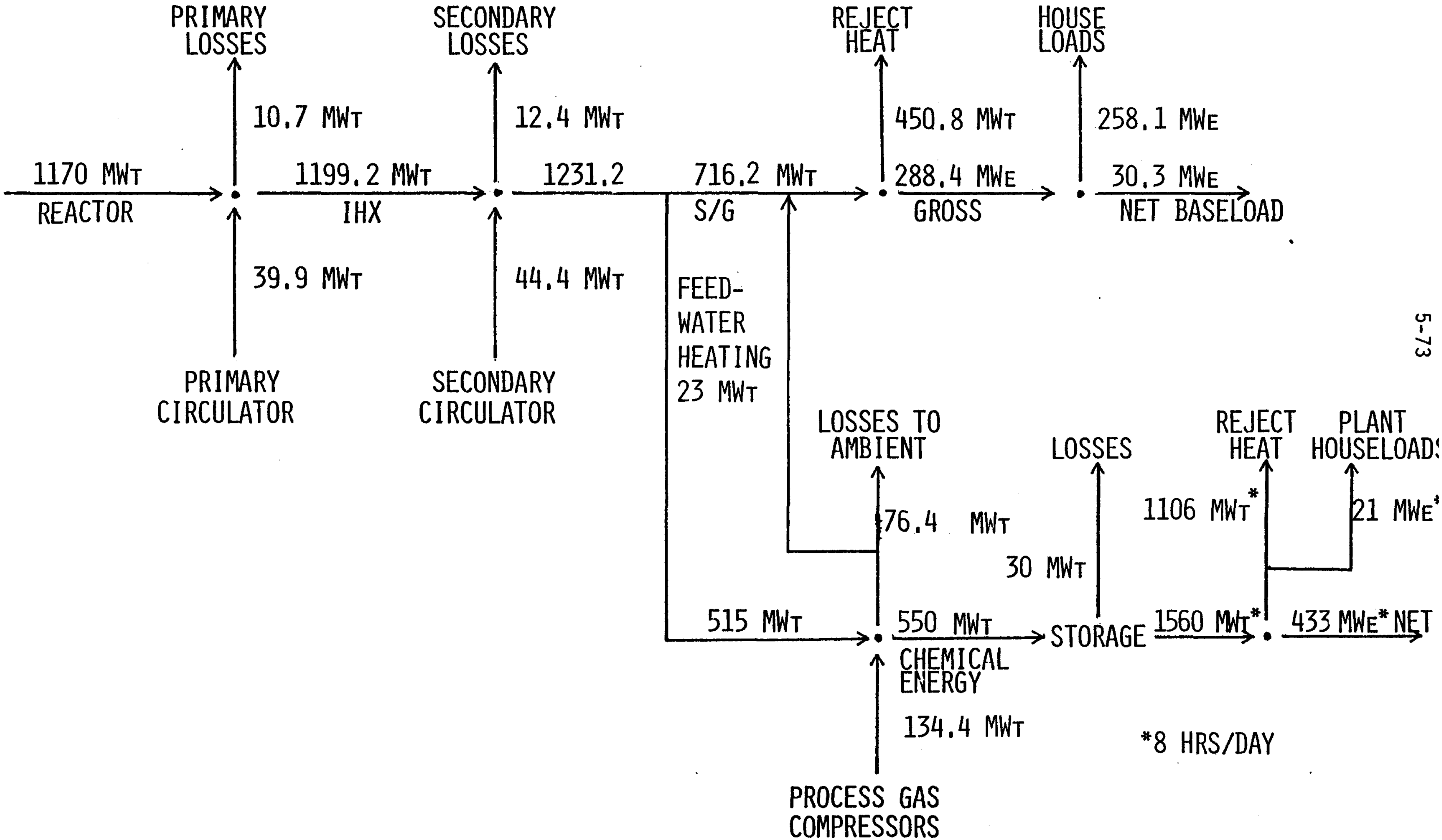

Figure 5.2.1-1 HTGR-R Energy Balance

Table 5.2.1-2 summarizes the overall plant performance of the reformer-pipeline-methanator system. In the case of load-following power generation, syngas is retrieved for methanation through a normal cycle of 8-hr operation in each day. The twin-turbine generator will generate 433 MW(e) net to produce a total peak power output of 463.3 MW(e).

For process heat supply, higher efficiency of energy conversion results in the production of 1560 MW of process steam during normal cycle methanation. The power plant, however, will generate a net continuous power of 30.3 MW(e).

5.2.2 Control and Dynamics

Plant control/protection systems provide the control, protection, and monitoring functions for ensuring safe and reliable operation of the plant over a wide range of plant conditions. These systems are designed to accommodate all planned modes of plant operation and to ensure the integrity of the fission product barriers and major plant components in the event of equipment malfunction, failure, or other abnormal condition.

Calculated plant transient performance, based on computer simulation of representative plant transient events and their design number of cycles, provides a basis for plant design. Results of the transient analyses are used in equipment design and selection, development of plant operating strategy, and validation of process control and plant protection schemes.

5.2.2.1 Overall Plant Control System

The plant control system (PCS) is an integrated system comprised of the overall plant control loops, which regulate the reactor and process system conditions; the analytical instrumentation subsystem, which provides the necessary equipment for monitoring plant operation and performance; and the component protection systems, which provide for the nonsafety protection of plant components and which serve as the first level of protection for incidents that could otherwise result in the need for safety-related plant protection system (PPS) action. The PPS is independent from the PCS and provides the safety-level protective functions and systems that prevent any unacceptable releases of radioactivity that could constitute a hazard to the health and safety of the public.

The HTGR-R plant control system shown on Fig. 5.2.2-1 coordinates the HTGR reactor plant heat source with two individual energy conversion systems: (1) the reformer and (2) the turbine-generator system. These two systems share the reactor power in fixed proportions determined by the limited allowable temperature range of the helium entering the steam generator (leaving the reformer). While there are many possible plant control options, the selection of a plant in which the reactor leads and the reformer and turbine systems follow will be an appropriate selection for a base load operating philosophy. Under this premise, the turbine-generator and reformer systems can operate at maximum

TABLE 5.2.1-2

OVERALL PLANT PERFORMANCE

	THERMAL	ELECTRIC
Reactor Power, MW(t)	1170.0	
Circulator Power Required, MW(e)		- 95.6
Circulator Heat Added, MW(t)	84.3	
Reformer Compressor Power, MW(e)		-117.0
Reformer Compressor Heat Added, MW(t)	103.2	
Miscellaneous Plant Power		- 10.5
Pipeline Pump Power, MW(e)		- 35.0
Gross Power Output, MW(e)		288.4
Net Power Output, MW(e)		30.3
Load-Following Power Generator:		
Load-Following Power Generation, MW(e)		433*
Total Maximum Power Generation, MW(e)		463.3
Process Heat Supply:		
Net Power Generation, MW(e)		30.3
Net Process Heat Supply, MW(t)	1560*	

*Normal cycle 8-hr operation per day, 7 days a week.

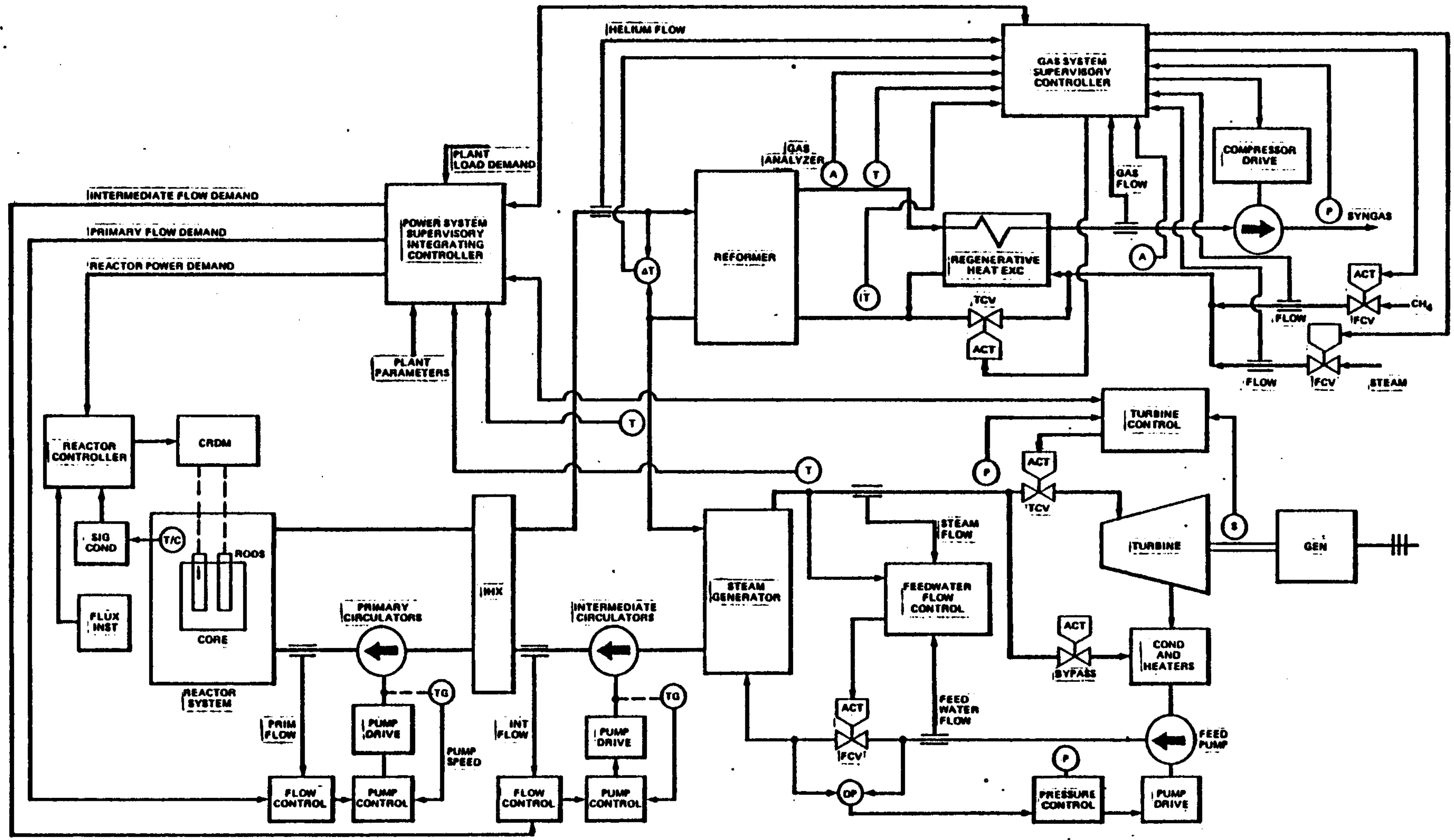

Figure 5.2.2-1 Power Range Control System

capacity without the need to constantly adjust load between them as would be necessary if either system were to load follow.

The control system operating philosophy is to set the reactor power as desired and to have the reformer and turbine-generator plants automatically follow reactor load while proportioning power to maintain desired plant parameters. The system will be capable of manual operation, many alternative partially automatic and manual modes, and fully automatic operation. The control system shown by Fig. 5.2.2-1 is a two-level control system. The top level is the supervisory control, which receives as input the operator-determined plant load demand. The supervisory controller generates many programmed set points which are a function of desired plant load. The second level controllers receive the supervisory controller output and maneuver the plant systems to the desired load and parameters. The plant supervisory controller is a real-time system. The reformer plant and its heat exchanger train also are controlled by a supervisory controller system equal to a Bristol 3000 microprocessor in combination with an H-P 9845 set point terminal.

The second level or local inner loop controllers include controller systems to control the reactor, primary and intermediate circulators, feedwater system, turbine, boiler feed pump and reformer plant compressors, and feedgas and syngas composition. For a given load, the primary variables to be controlled are the turbine throttle steam temperature and pressure, and the gas composition and temperature in and out of the reformer.

The supervisory controllers of the power and reformer systems maintain the plant at the demand power level and predetermined parameters. The turbine-generator is controlled to take whatever load the steam generators provide. This is accomplished by operating the turbine under control of its initial pressure regulator. This is the simplest method to maintain main steam header pressure and coordinate allotted steam system load. Turbine speed, except during starting and synchronization periods, is controlled by the synchronous speed corresponding to the connected electrical system frequency. With the control systems provided, other pressure control options such as trimming steam pressure by intermediate flow modulation are possible if desired. Main steam temperature is controlled by maintaining predetermined set points of plant parameters, and is trimmed by varying feedwater flow rates. In once-through steam generators, outlet steam temperature is very sensitive to feedwater flow rate.

Control of the reformer plant consists of matching the reformer load allotted while maintaining the helium temperature into the steam generator. An on-line heat balance is calculated for the helium in and out of the reformers and the feedgas flow and composition are maintained in response to the allotted load. Similarly to the turbine-generator plant, the reformer plant also follows the reactor load. Temperature control of reformer feedgas can be trim controlled by flow variation in a bypass line around the highest temperature regenerative heat exchanger. A gas analyzer at the reformer outlet is provided to determine the gas composition; analysis of this composition determines the nature of signals to the supervisory controller which adjusts inlet temperature and/or inlet gas composition to maintain syngas quality.

In addition to accommodating plant system perturbations resulting from routine plant operation, the PCS provides for the control of transients imposed on the system during reactor trip, loop shutdown, feedpump trip, and loss of process steam/reformer load. Under these conditions, reactor power, helium flow, and feedwater/reformer flow are reduced at predetermined rates in order to minimize thermal transients imposed on the heat exchangers and reactor components.

The PCS also provides automatic actions for protection of major plant components and protective actions during certain incidents that otherwise would result in the need for safety-level PPS action. The PCS protective actions include those required as a result of failure of an active nuclear system component. Failure of these protective actions will not jeopardize public health or safety.

5.2.2.2 Plant Operation

This section addresses HTGR-R plant operational methods, procedures, and plant characteristics necessary to achieve desired operational modes. Operations are divided into normal conditions (startup, power operation, and shutdown) and off-normal operations (N-1 loop operation, steam-electric generation plant only, and reformer plant only). The current assumptions regarding plant characteristics and features are the following:

1. No reformer helium bypass will be provided.

2. The secondary loops will be independent, i.e., no piping connections between loops.

3. In the steam generator mode, the reformers will be kept warm on the gas side by steam bleed flow.

4. The reformers can operate at low loads such as below 25% of full power.

5. The reformers cannot be operated with gas alone; there must be a mixture of steam and gas to prevent carbon deposition on the reformer surfaces. Steam/methane ratios larger than a factor of two are required.

6. Reformer temperature increase of 100°F per hr is acceptable.

7. Reformer action starts when the temperature reaches 500° to 600°C, assuming proper gas mixtures.

8. The reformer system can be purged with nitrogen.

9. The steam generator on an individual loop cannot be operated if the reformers of that loop are not operational.

10. The steam turbine can be placed on load when the throttle temperature steam reaches 100°F superheat.

- Startup - The basic procedure to start the plant is to first bring the plant to a relatively low power level with only the reactor and steam-electric plant in operation. The reformer plant is started only after the temperature of the hot-leg helium of the HTGR system reaches about 1000°F to permit reforming to begin. Since the reformer plant regenerative heat exchanger train must be phased in, the procedure is to operate the reformer by supplying steam from either an auxiliary boiler or the steam generators, and to supply methane from a storage tank. The reformed gas is initially flared to the atmosphere until the regenerative system is warmed and in steady state at low power. Starting at low reactor power and building up to full system temperatures is necessary to minimize heat losses and steam and methane losses during this period.

  Figure 5.2.2-2 shows the load range temperature profiles (helium systems) versus reactor power. The startup sequence is as follows:

  1. The sequence starts from plant standby conditions with all systems at a temperature level of approximately 650°F, which is determined by the saturation temperature of the water in the steam generator corresponding to the steam generator operating pressure. All heat-up rates should be held to 100°F per hour.

  2. The once-through steam generator is started in the conventional way and the plant is brought to a power level of approximately 15% to 20%. The hot-leg temperatures are about 1100°F and the reformer is isothermal with intermediate helium on the shell side and some heating steam bled through the tube side if needed to achieve uniform heating.

  3. At this point reactor plant hot-leg temperatures are increased and reformer action is initiated. Reactor power is added while the helium flow remains relatively constant. Steam and methane are admitted to the reformer in the ratio of steam/methane greater than a factor of four. The reformed gases and steam products are flared to the atmosphere. Heat removal in the reformer matches the additional heat added by the reactor as the system hot-leg temperatures increase.

  4. The HTGR system is heated to obtain full helium system ΔTs while maintaining helium flows at about 20% of full flow. The reformer plant regenerative system is started with heating provided by the partial diversion of syngas being flared from

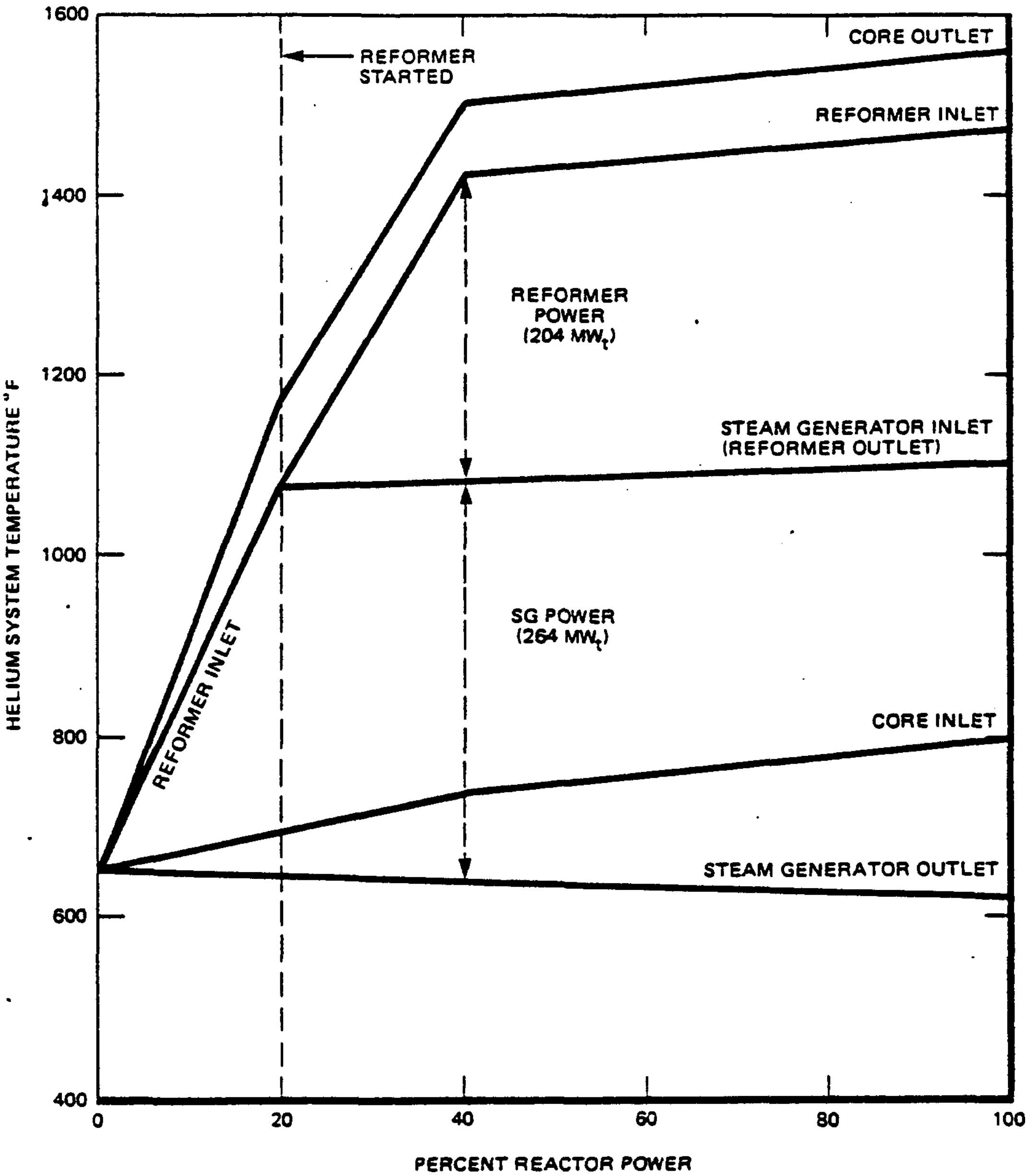

Figure 5.2.2-2 Helium Temperature Through the Reformer and Steam Generator as Reactor Power Increases

the reformer. This is initiated as soon as significant reforming begins. As heating and water evaporation in the mixed feed evaporator begins, the regenerative heat exchanger chain and gas pipeline become operational.

5. At the completion of the startup phase, the reactor power is about 40%, and the power split between the reformer and steam plant is about 50/50. This provides for about 40% of full steam load on the steam generator which is considered about minimum load to maintain boiling stability for the once-through units.

- Power Operation - The normal power operating range is between 40 to 100% reactor power with the helium system temperature profiles shown by Fig. 5.2.2-2. The plant normally operates at 100% power in a base load process heat-electric power mode.

- Shutdown - Shutdown occurs either in a normal power rundown or a reactor trip. The reactor trip presents some potential difficulties, specifically in terms of thermal shock to various components. The reformer response to thermal shock is of principal concern. The specific characteristics need to be examined in more detail by further study. However, the procedure will be to cool down as slowly as possible by reducing helium flow to minimum values and thus remove only enough heat to maintain acceptable helium temperatures at the steam generator inlet. Relatively high steam to methane ratios (3 or 4/1) will be maintained to prevent carbon formation during the cool down process. As the helium temperature of the HTGR hot legs falls to 1000°F, the reformer can be taken out of service and filled with dry hot nitrogen to prevent moisture accumulation and carbon formation. The reformer cool down rate is then governed by the rate at which the helium of the HTGR systems cools down. These cool down rates need to be determined by further transient analyses of the HTGR plant trip event.

- N-1 Operation - N-1 operation refers to operating the plant with one entire loop system out of service. Three loop operation would allow three reformer loops to operate at their normal energy output. The steam-electric plant would operate at three-fourths power. For outages such as a primary or intermediate circulator or an IHX, N-1 operation is an obvious event. Outage of a component downstream of the reformers causes N-1 outage because it does not appear possible to remove and reject sufficient heat to lower the intermediate helium temperature to an acceptable level for entry to the steam generator. There are other operating circumstances that could be considered to maintain the reformer in operation for a short period of time, perhaps a matter of hours. If the problem causing the outage in the reformer plant is located downstream of the reformer, it appears possible to isolate the reformer plant and run the reformer on steam from the steam generator and make up methane gas mixture from storage. This could be vented for short periods (and flared) to provide time for a relatively quick fix of items such as compressors or valves. Since the steam leaving the SG is at about the same temperature as the reformer feed, and the steam to gas ratio is 3 or 4/1, the feed mixture would not require much additional heating if any.

This operation would require reduced electric output and augmented feedwater heating, perhaps in the deaerator or last stage feedwater heater. Considerable makeup water to the condenser would be needed.

A complete outage of a reformer loop would require a complete plant shutdown, startup on three loops, and a subsequent shutdown and startup again on four loops after repairs had been completed.

- Steam-Electric Generation Only - The steam-electric plant operates in this solo mode by reducing both the normal reactor power and outlet temperature. The power reduction would be equal to the 515 MW(t) capacity of the reformer plant. The reactor temperature reduction is necessary to achieve an intermediate hot-leg temperature of 1107°F, equal to the normal steam generator inlet temperature. Since there is no reformer helium bypass system, the reformers will have helium flowing through them isothermally during steam-electric plant operation. The approximate heat balance for this mode of operation is shown by Fig. 5.2.2-3. There appears to be no difficulty in operating the plant in the electric generation mode only. When the reformers require hands-on maintenance, however, the entire plant or affected loop must be taken out of service.

- Reformer Plant Only - Operation of the reformer plant only is possible, if the steam turbine has a full-load steam bypass system and the condenser and heat balance of heat rejection equipment are sized to reject the full load [733 MW(t)].

  It is not too difficult to size for full-load heat rejection because the condenser can operate in this mode at slightly higher pressures which provide higher Δt's to facilitate greater heat transfer rates. In addition to the above items, the steam bypass system will need to have a desuperheater for the steam entering the condenser.

  The advantage of reformer plant only operation is that heat can be supplied through the reformer plant during periods when the T-G plant is tripped out due to electrical faults, etc., and thus avoid periodic shutdowns and restarts of the entire plant. Various repairs to the T-G plant can also be made providing they can be made under the circumstances which must exist to use the T-G plant for bottoming cycle heat rejection. It is assumed here that steam plant heat rejection will always be needed to reduce the HTGR cold-leg helium temperatures to acceptable values.

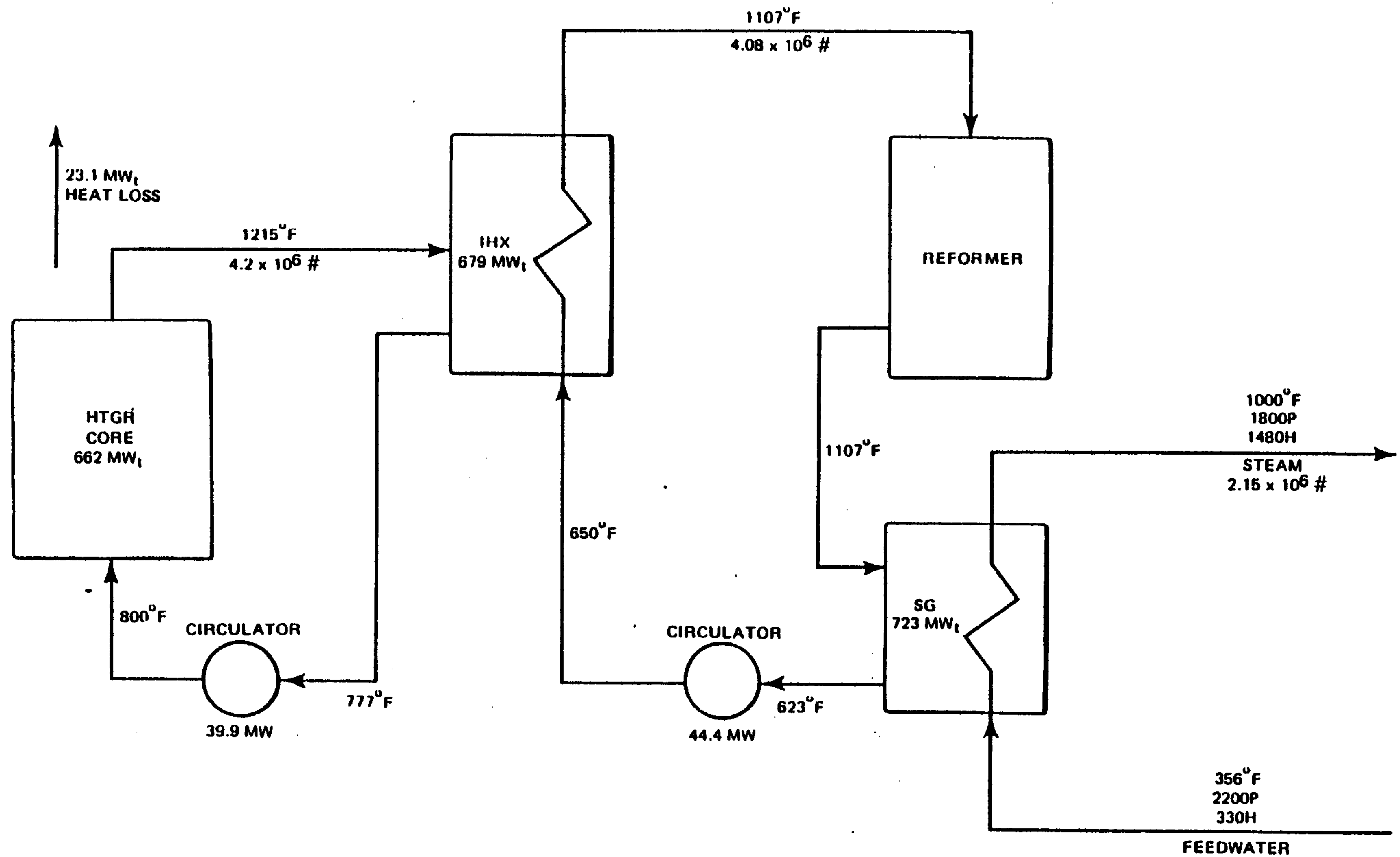

Figure 5.2.2-3 Heat Balance for Plant Operating with Electrical Power Generation Only

The state of the plant during reformer plant operation only would be as follows:

1. The T-G is sealed with mainline steam, the T-G is continuously rotated on turning gear, and the condenser is operational.

2. Steam is being generated by the steam generator.

3. Steam bypasses the T-G, goes through the desuperheater and into the condenser.

4. The feedwater train is in operation, feedwater heating is accomplished with live steam from the main steam header.

5. Auxiliary electric power would be supplied by the incoming lines.

One of the difficulties in the above mode of operation is feeding live steam to the feedwater heaters. While this is necessary to get the feedwater temperature up to an acceptable value, it has the potential to drive the turbine if the extraction line isolation valves should leak. It is standard practice to double valve these lines. Although the turbine manufacturers do not like the practice of live steam feed to heaters, it must be accepted for startup and thus would be acceptable for higher load operation.

5.2.2.3 Response to Critical Transients

Critical transients establish design requirements for the major plant components and systems. A qualitative discussion of anticipated critical transients is presented below. Preliminary representative plant transients have been identified (Table 5.2.2-1) and the critical transients have been selected. The representative transients will be subjected to control and dynamic systems analysis to verify the adequacy and completeness of the critical transient selection. The following critical transients have not yet been computed and analyzed since overall plant modeling is still in progress; however, anticipated responses are presented:

1. Reactor trip: Sets thermal stress requirements and cooldown rates for the reactor support structure, IHX, reformer, and steam generator; sets main loop aftercooling and PCS control action requirements.

   Response: Reactor trip system initiates reactor shutdown by insertion of scram rods; PCS initiates programmed feedwater, reformer, and helium flow reduction with subsequent trip of the process plant; and steam generators provide for afterheat removal. Main steam bypasses the process plant and is desuperheated before final heat rejection through the condenser. Main cooling loops should provide prolonged afterheat removal before CACS cooling is required.

TABLE 5.2.2-1
HTGR-R REPRESENTATIVE PLANT TRANSIENTS[a]

Group 1 (planned operation)

- Startup from refueling status
- Startup with full helium inventory
- Shutdown to refueling status
- Shutdown with full helium inventory
- Normal process load increase
- Rapid process load increase
- Normal process load decrease
- Rapid process load decrease

Group 2 and 3 [F[b] $\geq 10^{-2}$)

- Reactor trip from design power[c]
- Reactor trip from minimum load
- Reformer/process loop shutdown[c]
- Process steam load rejection
- Sudden reduction of feedwater/reformer flow[c]
- Turbine-generator trip
- Steam ingress to secondary helium loop (small leak)[c]
- Control rod insertion
- Reactor overcooling
- Slow secondary helium depressurization

Group 4 ($10^{-2} > F \geq 10^{-4}$)

- Slow primary helium depressurization
- Rod withdrawal with core power-to-flow trip
- Loop shutdown with primary helium valve failure
- Failure of primary circulator speed control
- Failure of secondary circulator speed control
- Steam ingress to secondary helium loop (moderate leak)
- Loss of primary helium flow
- Loss of secondary helium flow
- Shutdown of all main loops
- Total loss of feedwater
- Operating basis earthquake (OBE) with reactor trip

Group 5 ($10^{-4} > F \geq 10^{-6}$)

- Steam ingress to secondary helium loop (large leak)
- Rapid primary helium depressurization[c]
- Rapid secondary helium depressurization[c]
- Rod withdrawal with backup trip on high IHX inlet temperature

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(a) Transients are grouped by their probability of occurrence.
(b) F = expected frequency per reactor year.
(c) Denotes critical design transient.

2. Sudden reduction of feedwater/reformer flow: Sets PCS reactor power runback rate and imposes thermal stress requirements on steam generators/reformers.

   Response: Main steam pressure control action to maintain pressure; reactor power runback and reduction of primary and secondary helium flow; continued operation of process plant at reduced load (or shutdown if below minimum load).

3. Rapid primary helium depressurization: Sets CACS sizing requirements and imposes maximum IHX tube wall differential pressure; depressurization rate established by maximum PCRV penetration area.

   Response: PPS initiates reactor trip on either low primary system pressure or high IHX primary inlet temperature, depending upon the depressurization area. The PCS initiates programmed feedwater and reformer flow reduction with subsequent trip of the process plant. Low primary helium flow results in PPS main loop shutdown and CACS initiation; CACS provides for adequate afterheat removal.

4. Steam/process ingress to secondary helium loop: Sets helium monitoring system requirements and steam generator and reformer isolation/dump system requirements; sets secondary loop pressure relief system set point and main loop isolation requirements.

   Response: Detection of moisture/effluent; PPS initiates reactor trip, isolation and dump of faulty steam generator/reformer loop, and concurrent shutdown of the primary loop. PCS initiates programmed feedwater and reformer flow reduction with subsequent trip of the process plant. Intact main loops should provide initial afterheat removal before CACS cooling is required.

5. Rapid secondary helium depressurization: Imposes maximum tube wall differential pressure for steam generator and reformer, adverse pressure gradient across IHX tube wall; sets helium monitoring system requirement (potential for formation of an explosive air-methane mixture).

   Response: PPS initiates reactor trip on low secondary loop helium pressure and isolates the leaking process loop(s). PCS initiates feedwater and reformer flow reduction with subsequent trip of process plant. PPS initiates main loop shutdown and CACS startup upon low main loop helium flow. CACS provides for adequate afterheat removal.

5.2.3 Inservice Inspection

The basis for development of an inservice inspection 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 inservice inspection and testing on plant availability can be minimized by scheduling inservice inspection operations to coincide with refueling and other planned outages. While some inservice inspection operations can be conducted during normal plant operations, the bulk can only be performed 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 inservice inspection operations will reduce plant availability. A program of inspection and testing for NHS components and BOP system components 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 HTGR-R plant. Previous studies have identified plant performance aspects of inservice inspection and testing for generic elements. Quantitative differences in the selection of generic elements, such as core size and number of loops, will impact proportionately on inservice inspection scope and duration. Major components unique to this concept and cycle are the four IHXs within the PCRV with top-mounted electric drive circulators.

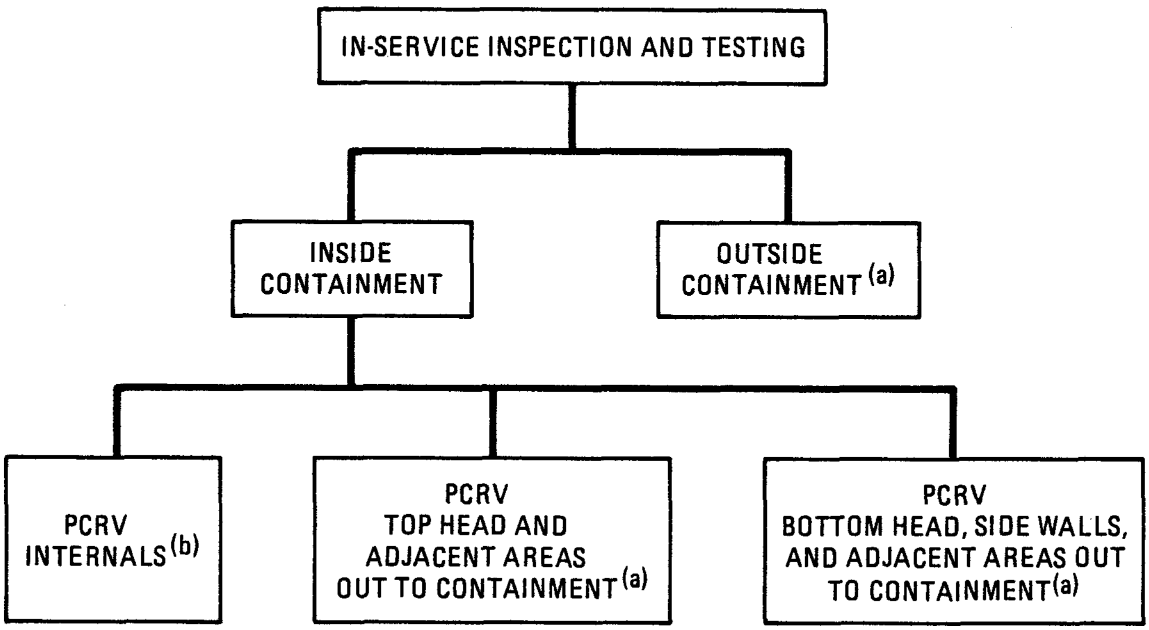
Inspections that have a direct impact on the refueling duration are those that are conducted from the top head area and which interrupt the refueling sequences. Components of this category include reactor internals, top head penetrations, and top head portions of piping systems extending from the PCRV. All other components within the containment and external to the PCRV, including portions of piping systems extending to containment boundaries, can be inspected without significant impact on refueling time frames. This category includes some 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 and, in many cases, can be conducted during normal plant operation. The scope of inservice inspection operations for nuclear components of the HTGR-R plant located outside the containment is of little consequence to availability. This condition results from the lower level of inservice inspection requirements imposed by code for Class 3 pressure boundary components, which are assumed to constitute the major portion of nuclear systems located outside the containment.

Although no specific studies relating to inservice inspection for this plant concept have been conducted, it is estimated that tasks to be accomplished during annual shutdown will not exceed 12 days (average for 10-yr inspection interval), with about 3 days impacting 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 NHS



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

Figure 5.2.3-1 Typical inservice inspection work areas

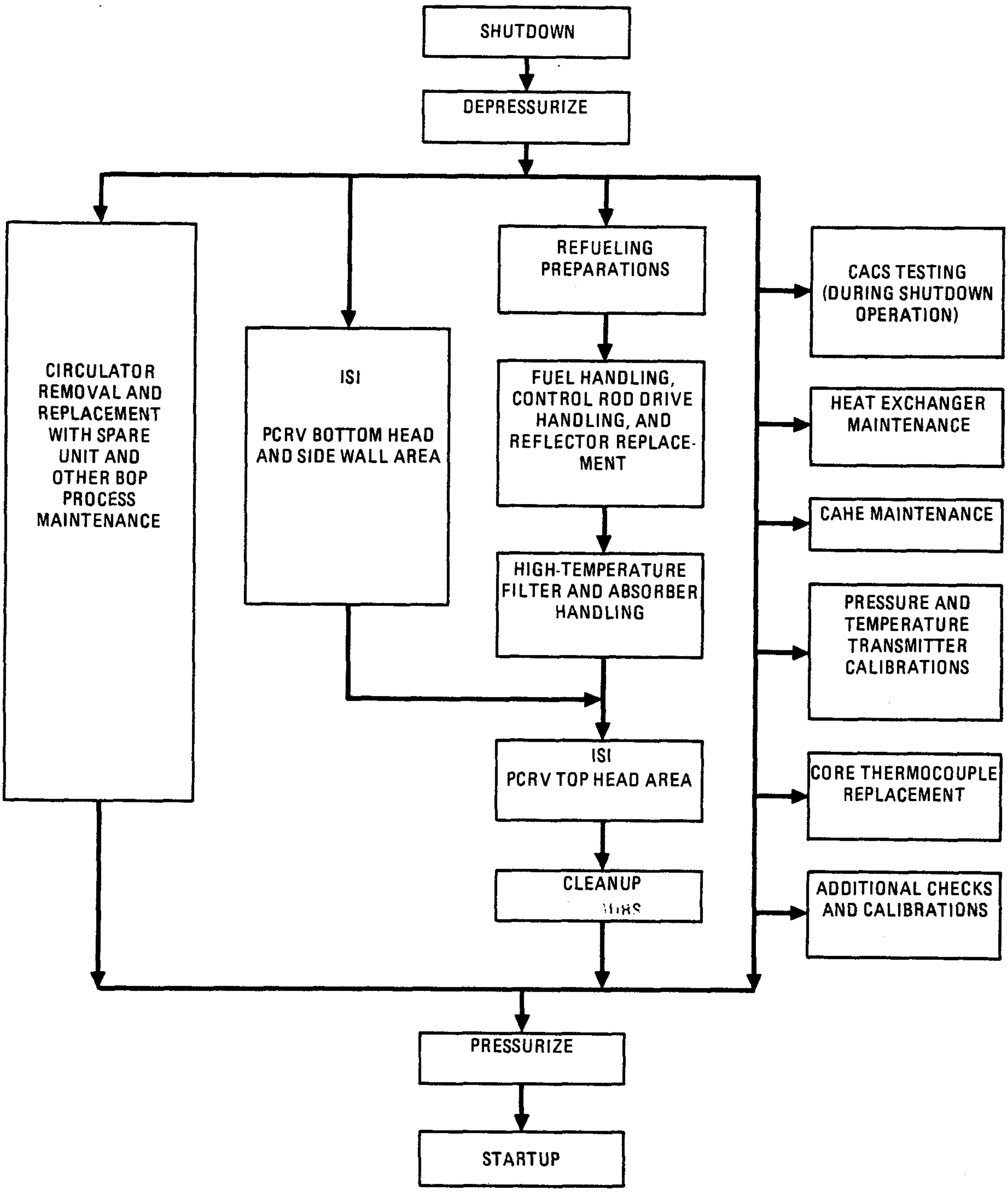


Figure 5.2.4-1 Typical HTGR-R maintenance activities during refueling outage

maintenance tasks that are included within the overall plant maintenance activity include:

1. Refueling.

2. Inservice inspection (Section 5.2.3).

3. Component and system servicing.

The major portion of the scheduled maintenance work is conducted during the annual refueling outage, with some inservice inspection 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.

The critical path for the NHS scheduled maintenance outage encompasses refueling, high-temperature filter and adsorber handling, and inservice inspection 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 mechanisms, the remaining NHS 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 power operation. Scheduling of control rod drive mechanism servicing fairly soon after the annual refueling will provide an early indication of any problems that might be developing with the drives.

Although no specific maintenance studies have been conducted for this plant configuration, methods and time and motion estimates developed for the NHS portion of other HTGR plants would indicate that the scheduled annual outage duration may be governed more by the BOP area tasks, particularly turbine-generator maintenance, than by those associated with NHS servicing. The maintenance philosophy related to the secondary and process system components are not established at this time.

The major unscheduled maintenance tasks that would promote a plant shutdown for restoration, or replacement, of NHS components include the IHXs, CAHEs, and primary loop circulator. Other than tube plugging of the heat exchangers, no other major servicing of those components can be performed in-situ. The removal of the primary loop circulators has been provided for, i.e., cask and handling capabilities. Temporary structures, handling requirements, etc., would have to be provided for removal and replacement of the IHXs.

The HTGR-R brings with it the requirements for reformer maintenance. As these requirements can also affect overall station availability and annual operating and maintenance costs, their impact has also been examined.

The catalyst replacement is readily accommodated by removing the reformer head and disconnecting the pigtail tubing from the outlet manifold. The straight pigtail is then individually removed from each reformer tube and the catalyst vacuumed out. The reformer tubes would then be visually and ultrasonically inspected on the bore side for weld defects, tube distortion, fouling deposits, and other signs of tube degradation. The catalyst is then replaced by reinserting the pigtail and wet loading the catalyst. The tubes are filled with demineralized water, the volume and weight of the catalyst is measured and then poured into the tube. The water is removed through the pigtail and vacuum dried. The catalyst load is then tested using the pressure drop method to assure uniform catalyst loading between tubes. To perform the pressure drop test an inflatable bladder would be installed (with a nitrogen connection through the bladder); nitrogen would be purged through the catalyst bed and the pressure drop measured for a given flow rate. Deviations in pressure drop of 15% or greater from the average for all the tubes would be rejected and reloaded.

Based on commercial reformer experiences at reduced heat fluxes, it is expected that replacement of catalyst would occur approximately every 10 years or less depending to a large degree on the type of upset conditions placed on the reformer. For the reformer systems, the equipment inspection and preventive maintenance can be performed during periods when the equipment has been taken off-line for regeneration or normal nuclear reactor outage.

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 these values 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 with other heat sources, both nuclear and non-nuclear. Furthermore, to be attractive as a power source for industrial processes, a reactor system and the overall process facility must exhibit high availability. The availability goal for the mature (>5-yr operation) HTGR-R 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 averge downtime is divided ultimately into allocated goals for plant systems 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.

The availability of the HTGR-R plant is considered in two broad categories: (1) that associated with the production and delivery of heat, and (2) that related to the use of the heat, i.e., the reformer process. The heat source, up to and including the IHXs, is thus viewed as an entity for availability evaluation and contains many reactor plant systems and components similar to those for which goals and predictions have been considered in other studies. The reformer process, however, presents two new aspects of availability evaluation. First, it contains elements such as the high-temperature catalytic heat exchanger reformer, the reliability and maintainability of which have not been previously estimated. The second aspect relates to the fact that four separate and isolatable reformer process loops are provided. There appears to be no reason why a loop could not be valved off or flanged off for major replacement or repair while the other three loops continue to operate. Thus, a shutdown of a loop for repairs only reduces the capacity factor to 75% during the repair time and would allow relaxation of the downtime allocation per process loop.

Table 5.2.5-1 presents a preliminary goal allocation for the HTGR-R plant based on the above reasoning. The estimated scheduled downtime is governed by the time for refueling and other NHS annual maintenance and totals 260 h. Any scheduled maintenance, such as IHX testing or inspection, replacement of catalyst in the reformer, or steam generator servicing, is assumed to be carried out concurrently with the above and, hence, will not increase the total scheduled downtime. Unscheduled downtime for the reactor is allocated at 195 h for NHS systems common to other HTGR plant concepts plus 85 h for the IHX. The total allocated reactor downtime is 540 h. To achieve a goal of 90% availability, which allows a total of 876 h/yr downtime, 336 h remain for allocation of unscheduled downtime to the reformer process. If the whole plant complex were assumed to be down during forced repairs of the reformer process, the allocated 336 h/yr for the four loops is only 84 h/loop-yr, and unscheduled downtime (for the reformer, the steam generator, the secondary circulator, and the loop isolation valves plus a number of heat exchangers and other process equipment) would be required to fit within this time. If, however, three loops are operated while the fourth is being repaired, each loop can have an average unscheduled outage of 336 h/yr and meet the effective availability goal. These allocations represent a preliminary goal based on previous reactor goal allocations and are subject to revision with more detailed concept definition and evaluation. Based on the allocation shown in Table 5.2.5-1, the effective availability of the reactor plant portion of the facility is 94%. Similarly, that of the reformer plant is 93%. A net facility availability of 90% is achievable since the 260 h/yr scheduled maintenance for both the NHS and the process plant is assumed to be conducted concurrently.

5.2.6 Operator Exposure

Radiation exposures to plant personnel include the exposures arising from reactor operation and surveillance, refueling, waste processing,

TABLE 5.2.5-1
PRELIMINARY GOALS FOR AVAILABILITY OF HTGR-R PLANT

	Downtime [h/yr (%)]		
	Reactor	H_2 Reformer Process	Total
Scheduled downtime			
Shutdown and startup	30		
Refueling	160		
Inservice inspection	60		
Filter absorber replacement	10		
IHX	<230[a]		
Reformer		<260+	
Steam generator			
Total	260 (3.0)	260 concurrent with reactor scheduled outage	260 (3.0)
Unscheduled downtime			
Systems common to other HTGR concepts	195		
IHX	85		
Requirements exclusive[b] to HTGR-R (total for 4 loops)		336	
Total	280 (3.2)	336 (3.8)	616 (7.0)
Total scheduled and unscheduled	540 (6.2)	596 (6.8)	876 (10.0)
Percent availability	94	93	90

[a] Performed during refueling and other scheduled downtimes.

[b] Reformer, steam generator, secondary circulator, and loop isolation valves.

inservice inspection, and special (or unplanned) maintenance. Occupational exposures for the HTGR-R option are yet to be assessed. A slight increase in the man-rem exposure is expected in comparison with an equivalent size HTGR-SC plant because of the impact of the higher reactor outlet temperature and maintenance associated with the IHXs.

As a design basis, the annual occupational dose for the 1170-MW(t) HTGR-R plant will be limited to 70 man-rem. The breakdown by the type of operation is provided in Table 5.2.6-1. This design basis is derived from the information presented in Ref. 1 for the 2240-MW(t) HTGR-SC plant. The in-vessel refueling scheme is assumed.

Occupational exposures at HTGR plants are projected to be considerably lower than those at LWR facilities with similarly rated powers. Actual man-rem exposures at the Peach Bottom HTGR and at the FSV HTGR have been exceptionally low (less than 10 man-rem/yr), providing assurance that HTGR personnel exposure will be maintained as low as reasonably achievable.

5.2.7 Radioactive Effluents and Waste

During normal operation of the HTGR-R, 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 graphite coatings into the graphite of the fuel elements and finally diffuse into the primary helium coolant. It is expected that reactor core components could be contaminated with graphite dust and lightly adherent films of plateout activity. It is also expected that the primary coolant and its attendant fission products could leak at a very low rate from the operating reactor to the containment and secondary systems and subsequently to the environment. The HTGR-R plant will have installed waste treatment systems designed to collect and process the gaseous, liquid, and solid waste that will be 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 plant have been calculated. The following discussion briefly summarizes the information on HTGR-R radioactive wastes and compares these waste quantity estimates with LWR plants of similar thermal power rating.

5.2.7.1 Liquid Wastes

During operation of the HTGR-R plant, a number or radioactive liquid wastes are generated, collected, and subsequently processed by the

TABLE 5.2.6-1
MAN-REM DESIGN BASIS FOR 1170-MW(t) HTGR-R

Type of Operation	Design Basis (man-rem/y-unit)
Refueling	5
Reactor operation and surveillance	10
NHS system maintenance and inservice inspection	15
BOP maintenance	25
Special maintenance	15
Total	70

liquid waste processing system. The expected sources of the liquid radioactive waste are as follows:

1. Decontamination system fluids.

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.

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 1170-MW(t) HTGR-R are summarized in Table 5.2.7-1. Also provided in the table are similar estimates of the liquid waste effluents for LWR (both BWR and PWR) plants. It is apparent that anticipated HTGR-R radioactive liquid waste discharges to the environment are but small fractions of those for similarly sized LWR plants in terms of both activity levels and quantities.

5.2.7.2 <u>Gaseous Wastes</u>

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, fuel shipping cask off-gas).

6. Liquid and solid radioactive waste processing systems off-gas.

TABLE 5.2.7-1
ANNUAL RADIOACTIVE WASTE GENERATION - NORMALIZED TO A REACTOR POWER OF 1170 MW(t)

Reactor/Data Source	Annual Release of Radionuclides (Ci)				Discharge to Waste Storage Facilities, Solid Wastes	
	Liquid Wastes		Airborne Wastes			
	Mixed Fission Products (No Tritium)	Tritium	Noble Gases	Iodine and Particulate	Volume (m^3)	Total Activity (Ci)
HTGR-R						
Continuous purge option[a]	0.002[b]	0.0[c]	100[d,e,f]	0.007	33	7400[g]
Intermittent purge option[h]	0.002[b]	0.0[c]	0.9[d,e,f]	0.003	33	7400[g]
BWR						
Operating plant average - 1976[i]	2.2	15	110,000	0.4	750	2600
Improved treatment capability[j]	0.15	7	24,000	0.15	300	600
Most advanced treatment capability[k]	0.15	7	1,500	0.003	~300+	~600
PWR						
Operating plant average - 1976[i]	2.6	370	7,400	0.07	370	150
Improved treatment capability[j]	0.01	130	1,500	0.02	74	2200
Most advanced treatment capability[k]	0.0007	130	480	0.002	~74+	~2200+

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 extrapment 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 26 m^3 as low level waste (330 Ci), 2 m^3 as titanium sponge waste (4400 Ci) and 6 m^3 as replaceable reflector block waste (2200 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" 1170 MW(t) 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.

7. If a wet cooling tower option is selected, and if radioactive liquids are discharged to the cooling tower blowdown stream, some evaporation of activity would be anticipated. Dry cooling towers would not release airborne radionuclides.

In addition, tritium can diffuse through the HX tube walls and mildly contaminate the process gas. It is estimated that tritium contamination levels in process gas (H_2 + CO) for the 1170-MW(t) HTGR-R will be on the order of 12 pCi/liter. This level of contamination is well below U.S. guidelines. Total tritium diffusion to the product gas is expected to amount to approximately 270 Ci/yr. The use of an intermediate loop effectively reduces tritium transport to the product gas. This reduction, in contrast to an integrated reformer concept, is due primarily to decreased tritium diffusion prcperties of the HX materials, as well as the use of an intermediate loop gas purification system and intermediate loop chemistry control. Small amounts of tritium may also be released as a result of diffusion through the steam generator tube walls and subsequent release through a condenser air ejector.

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-R, 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 1170-MW(t) HTGR-R operating with a once-through air purge basis. The table also gives the expected gaseous release estimates for an HTGR-R closed containment/intermittent purge* option. As would be expected, the intermittent purge option greatly reduces the release of gaseous waste, since containment ventilation to the environment is precluded between purges, thus allowing radioactive decay of the contained noble gases.

For comparison purposes, the LWR gaseous waste (scaled to power level) is also given in Table 5.2.7-1. 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 release with moderate and extensive liquid and gaseous waste treatment system installed.**

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*Intermittent purge for the HTGR-R 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.

**See WASH-1258 for additional information on BWR and PWR gaseous and liquid radioactive waste systems.

It is apparent that HTGR-R 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-R releases would continue to be less than LWR discharges.

5.2.7.3 Solid Waste

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

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

2. Spent resins resulting from demineralizer use and CO_2 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.

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-R operations are also summarized in Table 5.2.7-1. Additionally, this table summarizes (1) reported LWR solid radioactive waste generation for the year 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-R operation is significantly less than that produced in LWR operations; however, the total activity present in HTGR-R 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 the HTGR total activity in solid waste somewhat larger than for LWRs.

5.2.8 Investment Risk

The accident at the Three Mile Island nuclear power plant has focused public attention on reactor safety and, in addition, has dramatized the serious financial impact of such accidents on the utility. The inherent features unique to the HTGR provide increased assurance to the utility industry that even in the event of accidents, major equipment malfunctions, multiple failures, and operator errors, the damage to the plant, cleanup time, and restoration time will be held to a minimum. HTGR inherent features, such as the graphite core and helium coolant, define a forgiving reactor concept as exemplified by FSV operating experience.

5.3 Safety and Licensing

5.3.1 Inherent Safety Features

The HTGR design has inherent and passive features that make gas-cooled reactors a low risk to the utility and to the public as well as to reactor operating personnel. These features require less reliance on complex active systems and also help maintain the integrity of the reactor core and 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 summarized in Table 5.3.1-1 and described below.

- Helium Gas as a Coolant - A fundamental property of a noncondensable gas is that it totally occupies the space it is in and, so confined, obeys a simple linear temperature-pressure relationship. Because there is no liquid-gas interface to be considered, unambiguous measurements of temperature and pressure indicate the state and location of the coolant.

  A loss of coolant cannot occur; depressurization only can occur, and this is accommodated without any concern for the consequences of a change in phase, which results in a degradation of fuel cooling capability. Adequate core cooling is possible even at atmospheric pressure.

  Helium coolant is also chemically and neutronically inert; helium cannot react with core components and it does not contribute to or affect the nuclear chain reaction. In contrast, at Three Mile Island the uncovering of the core and subsequent fuel cladding heatup caused the zirconium-water chemical reaction that apparently resulted in damage to the fuel rods, as well as the extensive liberation of hydrogen gas.

- Ceramic Core and Reflector - The core and reflector structure is composed of graphite, a material that sublimes at about 3800°C and retains good strength to above 2500°C. The structure weighs almost 3 million pounds, and the associated heat capacity, together with the high temperature capability and low power density, ensures that reactor temperature transients will proceed very slowly. The slow thermal response provides a forgiving reactor since the behavior of the system is more readily predictable and more time is available to prevent transients from progressing into major accidents. Time is available for equipment repair, system adjustment, or other corrective action. For example, extended interruptions in core cooling system operation of the order of 30 minutes can be tolerated before damage to the core flow orifices and control rods would occur.

TABLE 5.3.1-1
SAFETY SIGNIFICANCE OF KEY INHERENT FEATURES

Inherent or Passive Feature	Relevant Properties	Safety Significance
Helium coolant	• Single phase	• No boiling, bubbles, liquid level, or pump cavitation • Coolant injection system not required • No ambiguity of signal indicating presence of coolant
	• Neutronically inert	• No reactivity effects
	• Chemically inert	• No fuel/helium chemical interaction
Graphite core	• High heat capacity, low power density	• Slow transient response • Time for prevention and mitigation of accidents
	• Graphite cannot melt but may locally sublime	• Strength maintained to over 3000°C (5432°F)
Coated particle fuel form	• Ceramic material	• Maintains integrity at very high temperature
	• Multiple "pressure vessels"	• Slow controlled release of volatile nuclides under no cooling condition
PCRV and associated liner	• Multiplicity of tendons	• Failure of individual structural members inconsequential
	• Tendons shielded	• No change in properties
	• Tendons removable	• Inservice inspection possible
	• Integral arrangement	• Primary system pipe/duct ruptures eliminated • Multiple structural failure required for air ingress

- Coated Particle Fuel - Another area of concern to both safety and plant maintenance relates to the possible migration of fission products. The coatings on the fuel particles constitute tiny independent pressure vessels, which contain the fission products. A total interruption of the core cooling systems would have to continue for about 3 h before any fuel damage and about 20 h before 50% of the core radioactivity would be released, providing time for fission product decay and for mitigating operator actions.

- Prestressed Concrete Reactor Vessel - The safety advantages of using a PCRV for containment of the entire primary system, a feature facilitated by the use of a noncondensable coolant, stem primarily from the redundancy of the load-bearing steel tendons. The independence and redundancy of these tendons provide a barrier to fault propagation within the vessel. The tendons are shielded from the effects of irradiation by the concrete. The steel liner functions as a non-load-bearing seal that is always held in compression by the surrounding prestressed concrete, a design feature which greatly limits the possibilities of fault propagation in the liner. The necessary liner cooling arrangements, moreover, furnish an additional available heat sink.

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 an 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 Protection System - The plant protection 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 protection system initiates safety functions such as reactor trip, CACS startup, and containment isolation.

- 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, 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.

- HTGR-R Safety Features - The HTGR-R 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. Pressure relief, system which prevents PCRV overpressurization by relieving excess pressure to the containment

  2. Automatic isolation of the secondary loops in response to low pressure or high radioactivity level.

  3. Core auxiliary heat exchanger isolation system, which detects conditions that indicate a leak between the CACWS and the primary coolant system.

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 HTGR-R 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 issues for the HTGR-R are expected to be (1) the IHX and the potential for direct release of primary coolant outside the containment due to major failures in both the secondary loop and the IHX and (2) the explosion hazards of the reformer.

- Issues from Previous NRC Reviews - An application has not been made for review of a plant of the design described herein. However, the HTGR-R 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.

  - 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 is an important function 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," is being 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 design having electrically driven main circulators greatly alleviates the problem of achieving sufficiently high gas flow rates for meaningful testing; i.e., the large auxiliary steam plant to drive circulators is not required.

- 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, backpressure 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 tasks 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 the HTGR-R - The HTGR-R employs an IHX operating at high temperature and high pressure. Although the primary and secondary sides are near a pressure balance during normal operation, accidents on both the primary and secondary sides can cause a rapid pressure imbalance and thereby subject the IHX to sudden pressure loads.

  Because the IHX will be designed to withstand normal transients as well as external accidents, the probability of these events will be low. However, to ensure acceptable consequences in the event that any one of them should occur, design features to mitigate the consequences, such as flow restrictors, will be investigated.

  For economic reasons, it is desirable to site the reformer as close to the reactor as possible. Even though the probability of reformer explosion or combustible gas leaks will be minimized via design and safeguards, the impact of postulated explosions and fires must be investigated. Appropriate design measures, such as hardening of the containment, will be undertaken if necessary.

  The nuclear island must be situated an adequate distance from the reformer system and storage field so that nuclear safety is unaffected by a detonation of the gases or a sudden release of a lethal amount of carbon monoxide contained in the synthesis gas. There is a risk of a detonation associated with the reformer and the storage field. The detonation-related parameters of the reformer are presented in Table 5.3.3-1. The TNT equivalent of the chemical energy in the reformer is estimated to be about 5800 lb. Because there are four reformer modules (PCPVs), it is a good assumption that only one-fourth of this TNT equivalent (about 1500 lb) would present a risk to the nuclear plant. Based on scaling laws developed by the military, which generally follow a cube root relationship between equivalent weight and distance and a 100% detonation yield, it is estimated that at distances greater than 260 ft from the reformer overpressures will be less than the 2.3 psi containment tornado design standard for the HTGR-R plant.

  It is more realistic to assume that the oxygen needed for detonation displaces some of the gas; under this assumption, the worst detonation that could occur is at the upper detonability limit of

TABLE 5.3.3-1

DETONATION-RELATED PROPERTIES OF THE REFORMER INVENTORY

Component	lb-mol	mole %	Higher Heating Value, 10^6 Btu	Upper Detonability Limit, %
CH_4	19.8	15.5	7.50	13.5
CO	4.1	3.2	0.51	59.0
CO_2	4.2	3.3	-	-
H_2	28.8	22.5	3.55	59.0
H_2O	71.0	55.5	-	-
Total or Average	127.9	100.0	11.56	39.0

TABLE 5.3.3-2

DETONATION-RELATED PROPERTIES OF THE STORED GASES

	Component	Volume, 10^6 SCF	HHV, 10^9 Btu
Synthesis Gas Field	CH_4	43.2	43.2
	CO	54.8	17.8
	CO_2	52.7	-
	H_2	376.1	122.2
	Total	526.8	183.2

At 24% yield = 22 million lbs TNT equivalent.

	Component	Volume, 10^6 SCF	HHV, 10^9 Btu
Methane Gas Field	CH_4	149.1	149.1
	CO	0.0	0.0
	CO_2	1.8	-
	H_2	7.2	2.3
	Total	158.1	151.4

At 24% yield = 18 million lbs TNT equivalent.

about 39%. This TNT equivalent is about 570 lb and requires a distance of approximately 190 ft before overpressures are less than 2.3 psi. It should be noted that current nuclear standards would require the reformer subsystem (and other processes) to be located at least 200 ft from the nuclear plant, which is the current basis for HTGR-R plant layout and design.

There is also a risk of a detonation of the stored gas. The maximum volumes of synthesis gas and methane in storage are about 530 and 160 million SCF, respectively. Detonation-related properties of the stored gases are shown in Table 5.3.3-2. If the synthesis gas storage field were to catastrophically release its contents, a detonation equivalent to 22 million lb of TNT could take place, assuming a yield of 24% as suggested by nuclear guidelines.

Overpressures from such a detonation would not decrease to 2.3 psi until a distance of about 6400 ft (1.2 miles) from the center. If the methane field were to catastrophically leak and detonate, it is estimated that it could have a TNT equivalent of about 18 million lb. The corresponding safe distance to the point where overpressures are less than 2.3 psi is about 6000 ft (1.1 miles). If it is assumed that the clouds of gases are moved and dispersed by winds, the safe distances would increase considerably.

- <u>Issues Specific to the TCP</u> - The synthesis gas components also require siting consideration because of the toxic nature of carbon monoxide. If one of the four reformer modules were to leak its product, a lethal concentration of CO would exist 190 ft away after about 1.7 hr, and 260 ft away in about 4.4 hr. This analysis assumes CO displaces air in a spherical fashion, but does not disperse; dispersion considerations would increase the lethal distance or conversely decrease the time to reach a distance. The synthesis gas from the four modules must be collected in a manifold where there is a risk that all the gas produced could leak. If this leaking manifold was also 190 ft away from the nuclear plant, a lethal concentration would be reached in about 0.4 hr, assuming a simple displacement model. If the manifold was located 260 ft away, a lethal concentration would exist at the nuclear plant in 1.1 hr.

Sectionalizing block valves are installed in pipelines to isolate sections of the line during an emergency. IGT has estimated the interval from the reformer side of both the synthesis gas and methane pipelines that would limit the hazard from a gas leak to a detonation with an overpressure less than 2.3 psi at 190 ft. For the synthesis gas pipeline, a valve must be installed within 150 ft of the reformer end of the pipeline. This interval is based on a nominal pipeline diameter of 42 inches and a pressure of 120 psi. For the methane pipeline, the estimated interval is 460 ft, based on a nominal diameter of 30 inches and a pressure of 300 psi. Block valves must be spaced thrcughout the remainder of the pipeline to meet minimum Federal standards. For Class 3 locations, the minimum interval is four miles, for Class 2 it is 2-1/2 miles. Pipelines must also be equipped with blowdown valves in each section that are sized to empty the pipeline as fast as is practicable.

The principal radiation concern is tritium from the primary coolant diffusing through materials of construction and leaking past connections until it establishes an equilibrium in the heat pipe system itself. There is also a possibility of uranium daughter products, such as radon-222, entering the gas from underground storage; however, IGT is unaware of any reports of radon contamination of stored gas by this mode. The IGT analysis of the radiation hazard is based on the maximum permissible concentration in air $(MPC)_a$ for a 168-hr week. For tritium, this concentration is 4×10^{-4} $\mu Ci/cm^3$; for radon, it is 1×10^{-8} $\mu Ci/cm^3$. If gas in the pipeline were to leak, it would be diluted with air. Odorants are added so that gas can be detected at one-fifth of its lower detonability limit (LDL). Thus, gas is considered "safe" at this concentration and leaks that would cause this concentration could go unnoticed. For methane, the LDL is 67.3%, and for synthesis gas, it is about 17%. For purposes of this analysis the synthesis gas value is the more conservative one to use (i.e., it errors on the safe side). One-fifth of 17% is 3.4%, or an air-to-gas ratio of about 29:1. The concentration of tritium or radon in the pipeline diluted with this proportion of air should not exceed the $(MPC)_a$. Based on this analysis, the allowable concentration of the radionuclides in the gas is twenty-nine times the $(MPD)_a$; for tritium this is about 1×10^{-10} $\mu Ci/cm^3$ and for radon-222, it is about 3×10^{-7} $\mu Ci/cm^3$.

The air emissions, water effluents, solid wastes, thermal discharges, and noise associated with the reformer, pipeline and storage fields were found to have inconsequential effect on the environmental acceptability of the heat thermochemical pipeline (TCP) concept. Perhaps the major environmental consequence is the requirement for careful disposal of the brine produced during solution mining of the salt cavern storage facility.

The major water effluent is a brine solution produced during the construction of the gas storage field. This is probably the most serious environmental consequence of the TCP concept. The large volume of brackish water (about 400,000 bbl) must be disposed of without contaminating fresh water sources or damaging ecosystems. This water is perhaps best permanently handled by injecting it below groundwater through a system of disposal wells. The injection formation must be chosen carefully to assure the brackish water could not mix with fresh water. State-of-the-art well drilling and completion techniques must be utilized to prevent brine leakage during injection operations. Disposal wells are currently being designed up to 10,000 bbl/day capacities. With the disposal well sealed off, the brine is effectively isolated from fresh water sources and ecosystems and permanently disposed.

5.3.4 Siting Flexibility

The siting flexibility of the HTGR-R plant was investigated and compared to an LWR of the same size. From the standpoint of radiological impact, the HTGR, in general, has greater flexibility in siting than the LWR. This fact is borne out when the radiological wastes projected for the HTGR-R are compared to those of the BWR and PWR, as was shown in Table 5.2.7-1.

The radiological impact of the HTGR-R and LWR effluents has been estimated for this study in terms of boundary dose levels for the corresponding airborne and liquid effluents reported in the Table 5.2.7-1 summary of radiological wastes. Table 5.3.4-1 provides the estimated site boundary airborne effluent dose levels for normal and off-normal conditions for both HTGR-R and LWR operations.

For normal operations (annual release), the HTGR-R limiting airborne radiological impact is whole body exposure from immersion in the gaseous airborne wastes, primarily noble gases. Nevertheless, the HTGR-R limiting pathway exposure ranges from 17 to 1700 times below the 10CFR50 Appendix I "as low as reasonably achievable" limit of 5 mrem whole body exposure. Additionally, the projected HTGR-R gaseous dose levels are significantly below BWR and PWR comparable doses for the HTGR-R 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. From Table 5.3.4-1 it is evident that the most advanced radioactive-waste treatment systems of the LWR satisfy the infant-thyroid dose criterion of 10CFR50, Appendix I. The HTGR-R infant-thyroid dose, however, is negligible in comparison to LWR values.

For severe accidents such as the HTGR-R DBDA, the HTGR-R 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 5.3.4-1. Both the HTGR-R and the representative PWR conform to 10CFR100 dose limits; however, the HTGR-R exhibits significantly more dose margin than does the LWR.

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

TABLE 5.3.4-1

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 1170 MW(t)]

Dose Category	10CFR50 Appendix I Limits	Annual Site Boundary Dose[a] (mrem)			
		HTGR-R[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	0.3	0.0007	0.35	0.01
Thyroid inhalation - adult	15	3×10^{-6}	0.7×10^{-6}	0.05	0.03
Thyroid - infant (grass/cow/milk/infant pathway)	15	0.001[d]	0.0002[d]	4.1[e]	4.0[e]

Severe Accident Conditions - Design Basis/Siting Events (Normalized to 1170 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-R						
Design basis depressurization[f] accident (DBDA)	25	300	0.0005	0.01	0.00015	0.002
Maximum hypothetical[g] fission product release (MHFPR)	25	300	0.0005	0.01	0.19	9.0
PWR						
Loss of coolant accident[h] (LOCA)	25	300	1.0	52	0.7	45

TABLE 5.3.4-1 (Continued)

(a)For the HTGR, exclusion area boundary (EAB) distance at 425 m and low population tone (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 x 10^{-5} s/m^3 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; instant aneous 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.)

5.4 Status of Design and Development

5.4.1 U.S. Program

Work at GA on nuclear process heat was initiated in the mid-1960s. The program has drawn heavily on HTGR-SC technology and operation of the Peach Bottom and FSV HTGR plants. The process heat application is a natural extension of steam cycle work, since many of the components can be used without modification. Since the fuel elements have no metal cladding, core temperatures can be raised to levels where processes competitive with fossil-fired heat sources can be considered. However, due to the increased core outlet temperature required for efficient process heat applications, an extension of current metallurgical technology is necessary for certain components.

Formal contract work was started in 1971 with a coal gasification study performed by GA and Stone and Webster Engineering Company for the State of Oklahoma. The most promising process studied was a coal solution hydrogasification process where the coal is first solubilized and then hydrogasified. A portion of the methane product is reformed with steam to produce the required hydrogen. The HTGR supplies the endothermic heat of reaction for the steam reforming as well as the steam and electricity for the entire plant. In a later phase of this program, testing was done with a 0.9 kg/h (2 lb/hr) experimental unit that investigated the process step of hydrogasifying the coal liquid to produce light aromatics and substitute pipeline gas. The investigation of the coal liquid step has been under development by other companies for a number of years. In later work for the National Aeronautics and Space Administration, this basic process was modified to produce hydrogen rather than methane and light aromatics from coal.

In 1974 ERDA (now DOE) initiated support for the HTGR for process heat. This work, which is continuing, has focused primarily on the design and development of the nuclear heat source. Particular studies have been directed at the variation in product cost with reactor outlet temperature, the need for an intermediate helium loop between the primary helium and the process streams, and the cost effect of reactor size. Many different process applications in the area of coal gasification, coal liquefaction, oil shale retorting and upgrading, and transport and storage of nuclear heat via heat transfer salt have been investigated. Thermochemical water splitting has also been investigated, and technical work on the sulfur-iodide cycle has been in progress since 1972.

In the early 1970s, GE studies identified the HTGR as being an attractive nuclear heat source for producing synthesis gas and hydrogen by means of chemical processes such as steam/methane reforming and thermochemical water splitting. This early GE effort included cooperative studies with GHT and KFA in the Federal Republic of Germany (FRG), which provided GE the opportunity to access the German HTGR pebble bed reactor technology. From that time to the present, GE has been actively engaged, under USDOE contract, in work oriented toward use of the HTGR for hydrogen (or synthesis gas) production, including:

1. Assessing and defining markets for HTGR applications.

2. Developing conceptual HTGR plant and component designs, and identifying R&D needs.

3. Evaluating and developing advanced metallic materials for operation in the 900°C to 1000°C helium temperature range needed for hydrogen production.

4. Assessing the HTGR technology developed in the FRG for potential U.S. applications.

One result of GE's effort has been to identify the HTGR-Thermochemical Pipeline (HTGR-TCP) as having a market incentive relative to fossil fuels in the 1995-2010 time frame for serving dispersed, one- and two-shift, industrial heat users and for generating load-following (daily-peaking) electricity. In addition, a full-size duplex (tube-within-a-tube) steam reformer tube was designed by GE, fabricated by Foster Wheeler (under subcontract to GE), and successfully tested during June 1979 in the EVA I facility at Julich, FRG. GE has recently begun testing metallic materials for high temperature components in a new materials facility in Schenectady.

5.4.2 International Program

5.4.2.1 Background

Development of gas-cooled reactors has been pursued on a worldwide basis for at least 25 yr. The British Calder Hall was built in 1956, and since then several major industrial countries have become involved. The largest programs in the free world, totaling billions of dollars, have centered in England, Germany, and the U.S., but major programs have also been carried out in France, Switzerland, and Japan. Funding for these countries for the year 1979 is shown in Table 5.4.2-1. The HTGR programs ongoing internationally are summarized below.

5.4.2.2 Germany

The largest helium-cooled reactor program in the world is in the FRG. Two HTGRs are operating or are under construction. The 15-MW(e) AVR prototype pebble-bed power plant has operated successfully since its startup in 1967. The 300-MW(e) pebble-bed Thorium Hoch Temperatur Reaktor (THTR) is currently under construction. Major changes in safety and licensing requirements, together with late delivery of some components, have caused schedule delays, and startup is now expected in 1982-1983. Hochtemperatur Reaktorbau (HRB) is supplying the NSSS for the THTR, and its parent corporation, Brown Boveri & Cie (BBC), is supplying the turbine and BOP.

TABLE 5.4.2-1
1979 INTERNATIONAL GCR FUNDING
($ Millions)

Germany	
HHT	42
PNP	59
Process	22
THTR	140
Recycle	9
Other HTGR	5
GCFR	3
	280
Switzerland	
HHT	7
GCFR	2
	9
France	
Generic	12
Japan	
VHTR	16
Process	8
	24
U.S.	
HTGR	33
Recycle	9
GCFR	26
	68

Until recently, the German HTR program has been directed toward the development of the HTGR-GT (HHT) and HTGR-PH (PNP), and significant technical development has taken place on those two projects. For PNP development, Gesellschaft fur Hochtemperaturreaktor Tecnik (GHT) had overall project responsibility supported by HRB, Kernforschungsanlage (KFA), and the coal companies RBW and BF. The program has been guided by a user organization formed in March 1978 that consists of German coal and gas companies.

Recently, a joint User's Group, formed between the German HTGR Utilities Organization and the German Ruhrkolhe and Ruhrgas Companies, has been contracted by the German Government Ministry of Technology (BMFT) to evaluate an HTGR-SC as the next potential project in Germany. The application under consideration is to supply electricity as well as process steam for coal gasification and liquefaction processes. This near-term project is viewed as an intermediate step in the demonstration and development of the HTGR for more advanced, higher temperature applications.

5.4.2.3 France

A close collaboration between the Commissariat a l'Energie Atomique (CEA) and GA on HTGR plant and fuel technology development has been in progress since the early 1970s. Recently, the program has moved toward increased focus on process heat applications. The French program includes tests in several unique large-scale facilities, such as the Carmen flow test loop and the fuel irradiation facilities, and has made major contributions toward reducing technical risks in HTGR development. General Atomic has licensed the HTGR system and fuel to the CEA and HTGR components to SAHTR/Novatome (the French advanced reactor company).

5.4.2.4 Japanese HTGR Program

The Japanese envision the HTGR in a process heat role as a key element in reducing their dependence on imported fossil fuels. Until recently, nuclear steelmaking was expected to be the primary application. However, changing energy economics have led them to include a broader range of HTGR applications, including synthetic fuel production. The Japanese HTGR development program, under way since 1969, is directed toward a 50-MW(t) Very High Temperature Reactor (VHTR) with a nominal helium outlet temperature of 1000°C. Organization of the Japanese effort is shown in Fig. 5.4.2-1. Cooperation with the Japanese organizations involved in the program is a definite possibility. Again, the extent of cooperation will depend on the future course of the U.S. program.

5.4.2.5 European Cooperation (Umbrella Agreement)

This agreement, signed in 1977, between the U.S. government and the FRG, Switzerland, and France establishes the formal basis for any

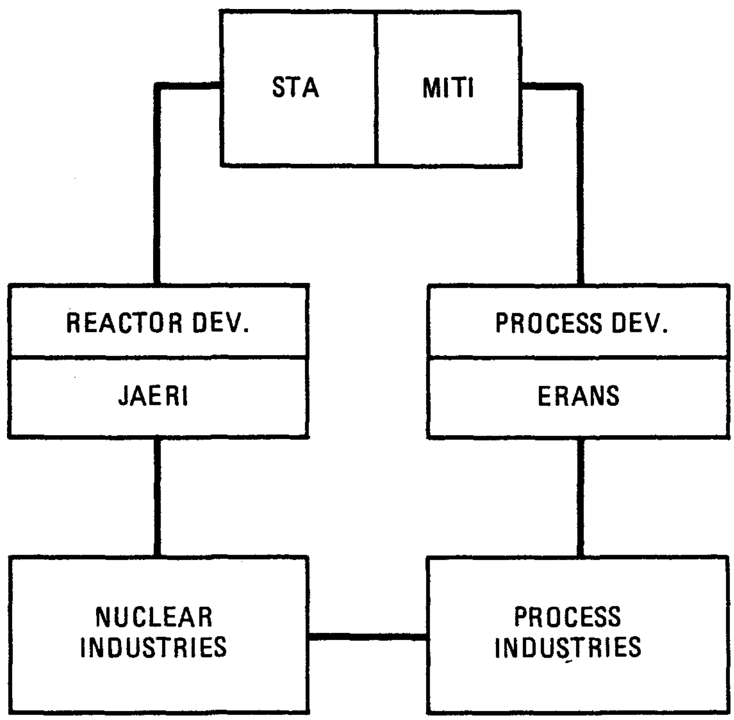

Figure 5.4.2-1 Japanese process heat program

specific cooperative efforts that are undertaken which involve public funds. It is written to enable each party to benefit by the exchange of knowledge and know-how with the other parties for the purposes of identifying areas of common interest, defining joint planning and programs (including conceptual design efforts), eliminating duplication of R&D efforts, advancing the state of HTR technology for all parties, and increasing user and supplier confidence in HTR systems. The most significant cooperative efforts under the Umbrella Agreement to date have centered in the HTGR-GT/HHT and HTGR Generic Technology programs. There is also a cooperative arrangement between U.S. and German utility organizations for the exchange of information and derivation of plant functional specifications.

The scope of future cooperation with the German HTR program will depend to a large extent on the direction of the U.S. HTGR program. The Umbrella Agreement provides a framework for further specific areas of information exchange.

5.5 Priority Technical Issues

The status of design work on the HTGR-R 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. These issues also give 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. The effects of high operating temperature on the IHX, thermal barrier, metallic core internals, and reformer.

2. Fuel element graphite stress analysis uncertainty.

3. Core support graphite stress and oxidation.

4. Reformer plant safety/licensing criteria for process gas releases/ explosions, tritium release/product contamination, and secondary helium piping failures.

5. Core region temperature fluctuations.

6. Water ingress.

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

As the definition of HTGR-R 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 characterization of acoustic vibrations, pressure transients, and coolant mixing and dispersion phenomena; the design of internally insulated piping systems and isolation valves; IHX and reformer design and performance verification; 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 Effects of High Operating Temperatures on Primary System Components

5.5.1.1 Core Restraint and Peripheral Seal

- Issue - Increased helium temperatures (relative to the original design for the HTGR-SC/C) require reevaluation 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/C conditions. The nickel alloy materials used in the different parts of the spring packs and peripheral seal were near their limit for the steam cycle conditions. At higher temperatures, embrittlement due to carburization and aging becomes a problem.

- 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 alternative materials to the existing designs will be evaluated.

5.5.1.2 Thermal Barrier

- 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 at GA have identified Inconel 713LC as the candidate for casting alloys. Handbook property data for this alloy indicate acceptable strength and carburization resistance compatible with an 850°C mixed core outlet temperature. GA has conducted initial discussions with several foundries, which indicate castability of desired component sizes and geometries. An order has been placed for thermal barrier integral cover plate attachment fixture castings for further evaluation. A comprehensive data base for the most promising thermal barrier casting materials will be established. This multiyear program is designed to provide basic materials design data in HTGR environments and to subsequently develop suitable design criteria.

5.5.1.3 Intermediate Heat Exchanger

- Issue - The current IHX design for the 850°C reactor outlet temperature HTGR-R plant requires the use of Inconel 617 or a similar wrought high-temperature alloy. At these design temperatures, carburization of alloys of this type due to interaction with primary coolant impurities can be rapid, and means must be found to prevent such carburization by coatings, cladding, coolant doping, or other techniques. Several techniques are currently under study, and considerable effort will be needed to find viable solutions.

Current IHX designs assume a solution to carburization of tubes and support structures. If no solution is found, the design would require cast materials with resulting increases in envelope size and cost and questionable feasibility.

The IHX will be a Code stamped component; however, existing rules and materials extend only to 760° to 815°C and do not cover some of the candidate materials for the IHX. Accordingly, considerable effort will be required to obtain Code approval for IHX materials and design temperatures.

- Program for Resolution - The IHX design is relatively new; therefore, programs to resolve material property concerns and Code qualification have yet to be started.

  Carburization tests, however, are in progress and various methods of stopping or limiting carburization are being investigated. These include modifying the gas environment, developing coatings or claddings, and developing modified alloys. The ongoing development program projects that a solution to carburization will be identified within 2 years. The selection of the IHX material may involve a tradeoff between component design life and component replacement/replaceability.

  In addition to the carburization program, programs are planned to produce information necessary for Code qualification of Inconel 617. This effort would take approximately 5 years.

  Resolution of the material issue, i.e., identification of an acceptable material with limited carburization rates, is required early in the conceptual design phase (by the end of 1982) of the IHX. Complete resolution for material ordering would be required by the end of 1985.

5.5.1.4 Reformer

- Issue - The current 850°C reactor outlet temperature indirect cycle plant design leads to 790°C reformer inlet temperatures. The reference material for the HTGR-R reformer was selected by GE to be Alloy 800H. At these temperatures, it is important to establish mechanical properties and life expectancy for materials and weldments. Also, design allowances for material degradation in the helium environment must be determined. Friction and wear data, fabrication techniques, and nondestructive examination techniques must also be examined and factored into the reformer design.

There is also the potential question of ASME Code application to the reformer tubes, in which case allowable stress criteria would need to be developed out to 1600°F. Section III of the current ASME Boiler and Pressure Vessel Code has established allowable stresses for time to rupture and creep strength up to 1400°F. Extrapolations for design lifetimes of 300,000 hr need to be verified experimentally and approved by the Code. However, Code qualification may not be mandatory and conventional cast reformer materials may be acceptable for use although component lifetime could remain an issue.

- Program for Resolution - The design of convectively heated reformers for the HTGR is at an early stage, and only a portion of the materials information has been generated to date. Programs to resolve the material property concerns, material selection, and Code qualification requirements are at an early stage and are under way.

5.5.2 Fuel Element Graphite Stress Analysis Uncertainty

- Issue - Stresses in the fuel element blocks are difficult to analyze, 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 but 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 are 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.

5.5.3 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.

5.5.4 Plant Safety/Licensing Criteria

- Issue - The definition of plant accident scenarios and limiting system/component loading conditions must occur early in the design and licensing process to verify concept feasibility. The HTGR-SC has been the primary basis for past licensing interface with the NRC and for HTGR plant safety studies. As such, the HTGR-R has received only cursory examination of licensing/safety issues which are specific to its design. In order to initiate system and component designs and confirm their ability to meet imposed criteria, plant safety goals must be identified and potential plant accident scenarios must be defined. Among the unique HTGR-R events to be considered are process gas explosions, toxic gas releases, secondary helium pipe failures, and tritium transport/product contamination.

- Program for Resolution - A program has been initiated in FY 1981 to extend technical studies of explosion risks and plant design solutions identified during the FY 1980 application studies. The preparation of a probabilistic risk assessment safety report is also being undertaken during FY 1981. This assessment will consider all the HTGR-R specific incidents such as process gas explosions, toxic gas releases, secondary system pipe breaks and blowdown, tritium transport/product contamination, etc. Efforts will also be initiated to delineate HTGR-R licensing review requirements and plans for implementation.

5.5.5 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 10 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-R 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 over 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.

5.5.6 Water Ingress

- Issue - The HTGR-R applications have the potential for water ingress into the primary coolant system. Operating experience at Fort St. Vrain (FSV) has shown that the primary helium circulator with water-lubricated bearings is a potential major source of water ingress during operating transients. This operating experience has also shown that water ingress can have a major impact on plant availability. Plant technical specification limits on allowable moisture and total oxidants in the primary system are set at a low value, i.e., 10 PPMV of total oxidants ($CO + CO_2 + H_2O$) considering graphite oxidation and fuel hydrolysis. Once water ingress occurs, it is a slow process to remove it and reestablish this low technical specification limit.

- Program for Resolution - Several significant design improvements in the seal and bearing design have been made to reduce both the volume of water which can be introduced and the probability of such ingress occurring. However, further efforts are needed to test and verify the generic design of the water-related rotating machinery service system for the main circulators. The need and criteria for a water cleanup system must be investigated based on an analysis of the probability of water ingress and time for cleanup.

SECTION 5 REFERENCES

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6.0 TECHNICAL BASIS FOR FOLLOW-ON/ADVANCED SYSTEMS

6.1 Process Heat Systems

6.1.1 Objectives

Objectives for advanced process heat systems are to (1) raise the core outlet temperature to 950°C, (2) develop the direct reforming version if commercially indicated, and (3) develop new interface equipment to match with synthetic fuel processes.

6.1.2 Approach

Based on the 850°C core outlet temperature, indirect cycle lead plant, a series of commercial plants similar in design could be produced without further development. The next step would be to develop all components necessary for the higher core outlet temperature in the configuration(s) identified for commercial plants (direct and/or indirect versions). For the two commercial versions being considered, common items of development would include the fuel particles, fuel blocks, core internals, and thermal barrier under the core, in the cross ducts, and surrounding the high-temperature heat exchanger.

For the secondary helium loop configuration (indirect reforming), the IHX becomes the main development item from both a materials and mechanical design and analysis standpoint. In the secondary loop, the helium stop valves may require rework because of the higher imposed temperatures. The steam-methane reformer is within current design and material practice.

For the direct reforming version, the reformer becomes a significant development item to be considered. As one approach, the lead plant could be utilized to test a prototypical primary loop reformer (duplex tube) unit in the secondary helium loop.

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

1. High-temperature (950°C) helium flow loop to test a section of the IHX or duplex tube reformer. [This facility could be the Helium Component Test Facility (HCTF) required for HTGR-GT development.]

2. Expanded metallurgical test facilities for new alloys and nonmetallic materials.

For the direct reforming version, a demonstration plant is expected to be required for final testing of the integrated system and for confirmation of licensability. Based on the high-temperature helium loop work for IHX testing and the nuclear plant testing of the core and thermal barrier, no demonstration plant is likely to be necessary for the indirect reforming version, and a commercial plant could be constructed assuming the process had been properly demonstrated.

Additional design and development, particularly with regard to heat exchange equipment, would be required for deployment of other process heat options such as steam carbon gasification and water splitting. These options typically employ intermediate helium loops and would, therefore, require development of high-temperature IHXs and process heat exchangers. The demonstration and test facilities described above, along with continued lead plant operation, would comprise the primary basis for deployment.

6.2 Other HTGR Options

The HTGR-GT could also gain substantially from the development of the reformer version with a significant portion of the core, reactor internals, thermal barrier, and CACS directly applicable. If an indirect cycle gas turbine should ultimately be chosen, the HTGR-R demonstration/lead plant would serve closely as the demonstration plant, requiring only gas turbine development and demonstration in the HCTF. Major development areas would be the HCTF-GT facility (which could be combined with the non-nuclear reformer demonstration facility).

In the sequence of reformer development, the steam cycle would gain considerable knowledge from the higher temperature requirements imposed by the reformer application. At any time, commercial development could be pursued as a separate activity. The amount of time saved and the development shared with the reformer would depend on how far reformer development had progressed at the time of the spin-off.

7.0 LEAD PROJECT/PROGRAM SCHEDULE

The HTGR-R requires a significant extension of the state-of-the-art of nuclear power plant technology. The temperature regimes encountered in the nuclear heat source and the complex process systems required for reforming 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-R 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-R Lead Project.

The schedule presented in Fig. 7-1 is viewed as a "reasonable target" to design, license and build an HTGR-R plant. Developmental and regulatory issues will be dealt with in the interval from project inception and the receipt of a construction permit. The schedule for this interval is predicated on the initiation of a pre-application licensing review program by the end of FY 1982 and the identification of the utility/site for early site review by mid FY 1983. The pre-application licensing review will provide for early NRC involvement in the process of evolving criteria for the resolution of safety issues and give guidance to the formulation of testing programs required to verify design adequacy of key components. The early site review program will initiate development of siting criteria unique to the application of the HTGR nuclear heat source to industrial reforming. It is projected that this early discourse with regulatory agencies will facilitate the formal review of the PSAR which is estimated to require 36 months.

The construction period specified on the schedule is 75 months. The bases for the construction schedule are target LWR construction schedules, previous studies by UE&C for the 2240-MW(t) and 3360-MW(t) HTGR-SC plants, and a factor for the complexity of the HTGR-R Lead Plant. This added complexity entails the parallel construction of the pipeline/storage/methanation plants, reforming plant, and the secondary helium piping systems. The startup period between fuel load and full power commercial operation is projected to be 15 months. This extended period is based on the added complexity associated with coupling the HTGR with the reformer plant and the associated secondary systems. Full power commercial operation is projected to be 1998.

Figure 7-1
HTGR-R LEAD PLANT DESIGN &
CONSTRUCTION MILESTONE SCHEDULE

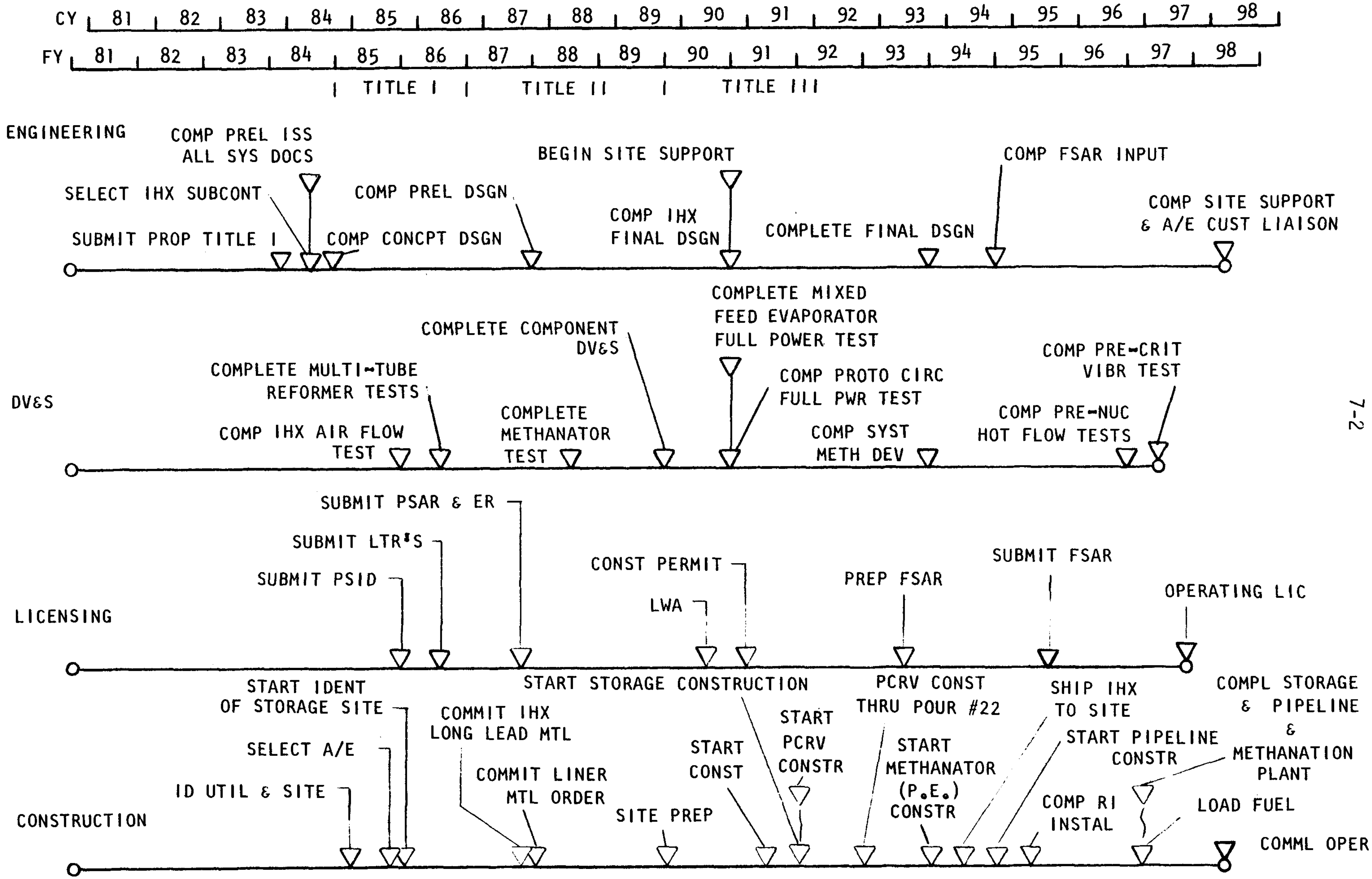
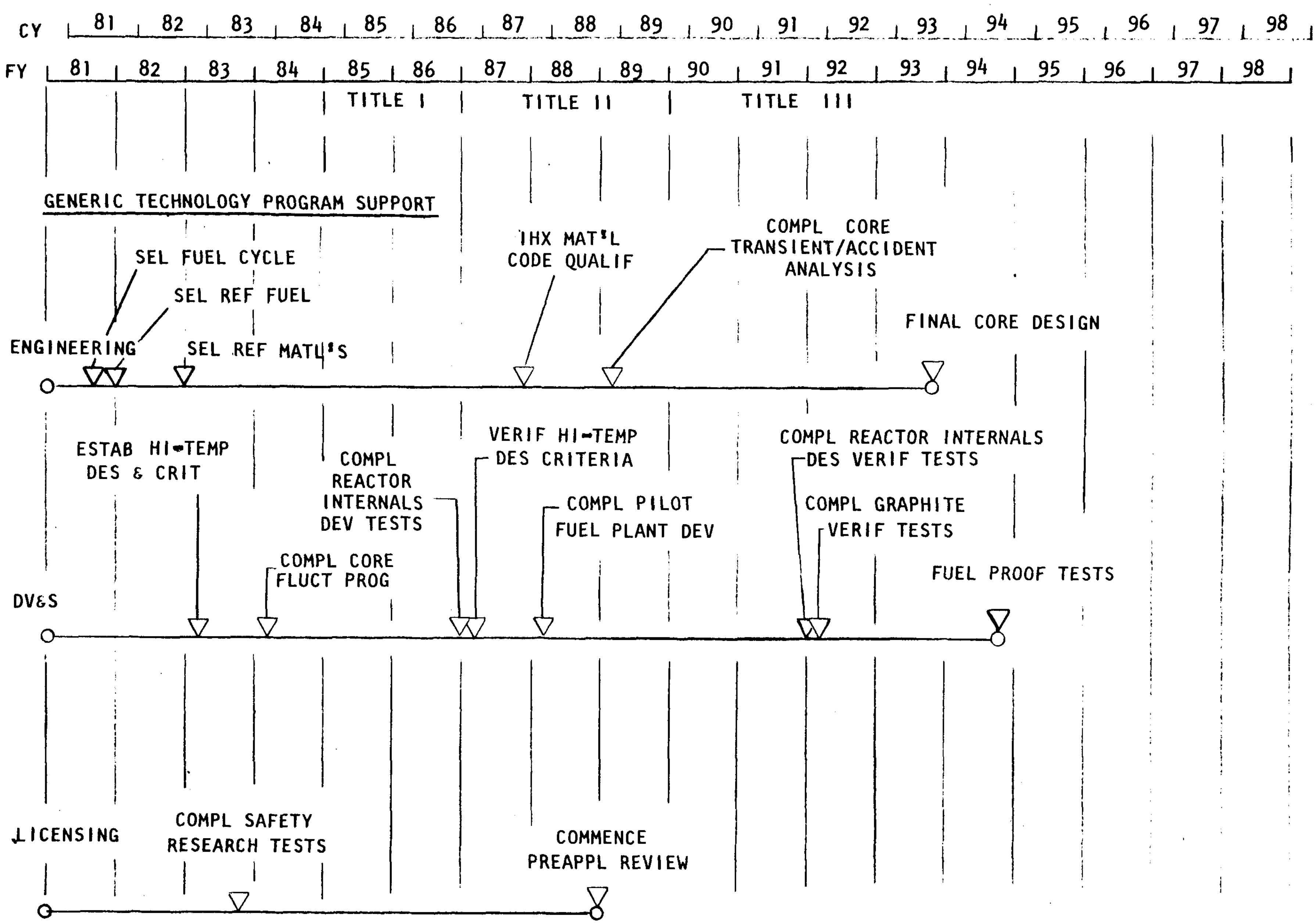


Figure 7-1 (continued)
HTGR-R LEAD PLANT DESIGN &
CONSTRUCTION MILESTONE SCHEDULE



8.0 LEAD PROJECT/PROGRAM AND COMMERCIAL PLANT ECONOMIC EVALUATIONS

The purpose of this section is to present the HTGR Program cost estimate based upon an HTGR-R Lead Project in the 1998 time frame. Among the contributors to total program costs are the Lead Project design and development costs, the Lead Project capital costs, and follow-on/advanced systems design and development costs. An assessment of expected commercial (equilibrium) plant operating costs is also provided to assist in market evaluation and program benefits analysis. A cash flow diagram is provided for the equilibrium plant.

8.1 Lead Project Design and Development Cost Estimate

The design and development program cost estimate has been generated as a result of the cooperative efforts of GA, GE, UE&C and GCRA. The split of responsibility for designating the HTGR-R design and development requirements fell along the line of design responsibility for the HTGR-R Lead Project Study. GA provided the design and development requirements and cost for the nuclear heat source (NHS) related systems and components. The NHS-related development costs were segregated into technology development and nuclear heat source design and development areas. The breakdown of the costs by area is provided in Table 8.1-1 for total program cost to complete. The process systems and components design and development costs were developed by GE, UE&C, GCRA, and GA and are also presented in Table 8.1-1. All of the cost information presented is in 1980 dollars.

The total design and development program cost to support the HTGR-R Lead Project is estimated to be $565M. The major development areas involve fuels and high-temperature materials, which comprise nearly 35% of the total development cost. Work on the fuels area emphasizes the development of medium enriched fuel fuel for the temperature regimes of the HTGR-R. Materials development is oriented to address the high-temperature requirements of this plant, most notably in the intermediate heat exchanger. The details of the NHS design and development program are provided in Appendix A.

The HTGR-R Lead Project cost summary is provided in Table 8.1-2. This summary was developed from design and cost estimate work performed by GA on the NHS, by UE&C on the balance of plant (BOP), and by GE on the process plant/delivery system. The basis for the HTGR-R Lead Project cost estimate was an 1170-MW(t) plant in a utility load-following application with a 1998 commercial operation date. The capital costs and cash flow projections are provided in 1980 dollars.

To determine the lead plant costs shown in Table 8.1-2, a subjective assessment was made to determine the degree of uncertainty that may be encountered in designing and constructing the first HTGR-R plant. The NHS cost for the lead plant reflects judgments by GA to account for soft tooling and non-recurring engineering and licensing costs. For the balance of reactor plant (BORP), the following factors were applied to the UE&C commercial plant cost estimate entries (see Section 8.2) to

TABLE 8.1-1

HTGR-R DESIGN AND DEVELOPMENT COSTS

(1980 $M)

TECHNOLOGY DEVELOPMENT	
FUEL	80
MATERIALS	115
PLANT TECHNOLOGY	22
TECHNOLOGY TRANSFER	6
CAPITAL EQUIPMENT	22
SUBTOTAL	245
NUCLEAR HEAT SOURCE SYSTEMS AND COMPONENTS DESIGN AND DEV.	
SYSTEMS AND DESIGN SUPPORT	38
SAFETY/LICENSING/GES	22
REACTOR VESSEL	40
REACTOR INTERNALS	17
REACTOR CORE AND FLOW CONTROL	34
CIRCULATOR	21
HEAT EXCHANGERS	35
CORE AUXILIARY COOLING SYSTEM	14
MISC. REACTOR SERVICES	16
FUEL HANDLING	8
CONTROL/ELECTRICAL	10
SUBTOTAL	255
PROCESS SYSTEMS AND COMPONENTS DESIGN AND DEVELOPMENT	
SYSTEMS AND DESIGN SUPPORT	25
PRESTRESSED CONCRETE PRESSURE VESSEL	3
STEAM GENERATOR	8
CIRCULATOR	1
REFORMER	15
METHANATOR	6
PIPING	5
CONTAINMENT ISOLATION	2
SUBTOTAL	65
TOTAL DESIGN AND DEVELOPMENT COSTS	565

TABLE 8.1-2

1170-MW(t) HTGR-R LEAD PLANT COSTS
(1980 $M)

NUCLEAR PLANT	
Structures and Improvements	132
Reactor Plant Equipment	309
Turbine Plant Equipment	48
Electric Plant Equipment	65
Miscellaneous Plant Equipment	14
Main Condenser Heat Rejection System	7
Secondary Helium System	75
Reforming Plant Equipment	195
Subtotal Directs	845
Indirects	370
Contingency	65
Total	1280
PIPELINE AND STORAGE	60
METHANATION PLANT	400
TOTAL BASE CAPITAL COST(1)	1740
PROJECT DEVELOPMENT(2)	30
TOTAL LEAD PROJECT	1770

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

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

account for first-of-a-kind (FOAK) engineering and design effects as well as inexperience with construction techniques unique to the HTGR-R. The TCP and storage system and the methanation plant cost entries provided by GE were assigned factors of 1.00 and 1.20 respectively.

Account	Lead Plant Factor
Structures and Improvements	1.10
Reactor Plant	1.50
Turbine Plant	1.00
Electric Plant	1.10
Miscellaneous Plant	1.05
Heat Rejection System	1.05
Secondary Helium System	1.50
Reforming Plant	1.50
Construction Services	1.25
Engineering Services	1.50

The FOAK uncertainties produce an estimated cost differential between the lead and equilibrium plants of $340M. In the interest of presenting major cost elements in a comparable format, other costs including owner's costs, interest during construction, and escalation have not been accounted for. 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 startup phases. Obviously, capital-intensive projects are particularly vulnerable to the effects of interest during construction and escalation. Such uncertainties in project schedules may result in substantially greater economic risk than the $340M ascribed to FOAK costs.

In Table 8.1-3, costs of major elements are distributed in accordance with the Project phases described in the project schedule (Fig. 7-1). Estimated expenditures for the Lead Project Definition Phase total $186M and reflect the effort to establish reactor outlet temperature, plant configuration, and plant application to develop the appropriate bases for reactor design. Estimated expenditures for the Preliminary Design/Licensing Phase are $143M. Funding requirements for the Detailed Design/Licensing interval are $225M. Expenditures during the Construction and Startup Phase are estimated at $1781M.

Table 8.1-3 also projects a plausible degree of utility financial support. Utility group support at a modest level may be anticipated throughout the course of the Program. Lead utility financial involvement is not projected until the onset of Title I with the offering of a site. Total in-progress financial support by utilities has been projected at a level that is slightly over 30% of the lead plant cost. The lead utility cost contribution (25%) is based on the commercial value of the electricity produced at full-power operation. The lead utility is also expected to pay the fuel and operating and maintenance costs fully, which along with the capital cost contribution

TABLE 8.1-3

HTGR-R LEAD PROJECT/PROGRAM COST PROJECTIONS (1980 $M)

	PROGRAM/PROJECT DEFINITION				PREL. DES./ LIC.		DETAILED DES./LIC.			CONST./ STARTUP	
COST ELEMENTS	81	82	83	84	85	86	87	88	89	1990 - 1998	TOTALS
LEAD PROJECT											
• TECHNOLOGY DEV.											
DOE	20.0	22.5	27.5	31.0	33.5	31.0	23.0	17.5	13.0	26	245
• NHS COMPONENT DEV.											
DOE	3.0	7.0	16.5	20.5	22.0	26.0	32.0	34.5	27.5	66	255
• P/CP COMPONENT DEV.											
DOE	.5	1.5	2.5	3.5	5.0	6.0	7.0	9.0	9.0	21	65
• LEAD PLANT											
DOE	6.5	6.0	5.0	5.0	5.0	6.0	7.0	8.5	11.0	1155	1215
GCRA/EPRI	1.0	1.5	2.5	2.5	3.5	4.0	6.0	6.0	8.0	85	120
Lead Util.	--	--	--	--	0.5	0.5	1.5	1.5	3.0	428	435
Subtotal	7.5	7.5	7.5	7.5	9.0	10.5	14.5	16.0	22.0	1668	1770
• TOTAL LEAD PROJECT											
DOE	30.0	37.0	51.5	60.0	65.5	69.0	69.0	69.5	60.5	1268	1780
Utility	1.0	1.5	2.5	2.5	4.0	4.5	7.5	7.5	11.0	513	555
Totals	31.0	38.5	54.0	62.5	69.5	73.5	76.5	77.0	71.5	1781	2335
ALTERNATE SYSTEMS											
DOE	7.0	10.1	6.7	6.0	6.0	6.0	7.0	8.0	10.0		
GCRA/EPRI	1.0	1.0	1.0	1.0	1.0	1.0	1.5	1.5	2.0		
FUEL RECYCLE										Further Projections Dependent on Deployment Scenario for Follow-on Projects	
DOE	2.8	4.3	5.4	6.0	6.5	7.0	7.5	8.0	8.5		
TOTAL PROGRAM											
DOE	39.8	51.4	63.6	72.0	78.0	82.0	83.5	85.5	79.0		
Utility	2.0	2.5	3.5	3.5	5.0	5.5	9.0	9.0	13.0		

corresponds to a 40% contribution on the annual operating cost of the HTGR-R lead plant.

Table 8.1-3 also projects the total HTGR Program costs, which include the Lead Project expenditures, follow-on HTGR systems, and fuel recycle. The cost projections for the alternate on follow-on systems are projected through 1989 on a near-level basis. The size and schedule for these activities after FY 1989 will be determined by the incentives identified by the overall HTGR Program, and no attempt to assess these costs is included. Fuel recycle cost projections are also provided through FY 1989. It is noted that major commitments for fuel recycle are not required until multiple commitments have been made for HTGR plants.

8.2 Equilibrium Plant Economic Evaluation

The HTGR equilibrium plant cost estimate was established for a remote energy distribution system serving industrial customers with 900°F steam via a 100-mi TCP. For the purposes of comparison with other HTGR plant options, the commercial plant product costs have been provided in 1995 dollars. Table 8.2-1 provides the overall plant capital cost in 1980 dollars. Table 8.2-2 presents the HTGR-R plant total investment cost in 1995 dollars, including escalation, interest during construction, and an assessment of annual operating costs; determines annual levelized power costs and plant products; and estimates thermal energy costs based upon a baseload electric cost of a large dedicated coal electric generator (159 mills/KW-hr). The thermal energy costs of the HTGR-R were calculated on this basis to be 90 $/MBTU as compared to costs of 59 $/MBTU for fluidized bed coal combustion (FBC) and 63 $/MBTU for oil. The energy cost of the FBC unit is based on a 50-MW(t) plant size with industrial financing. The energy cost of oil represents fuel cost only.

Table 8.2-2 presents economic information which indicates little incentive for the commercial HTGR-R. However, that conclusion may be premature as there is believed to be a potential for capital cost and performance improvement as time permits the design to evolve. Also, a large percentage of the capital cost increase observed, as compared to previous studies, has occurred in the process and methanation plants. The basic nuclear island, in contrast, has seen very little change. Therefore, economic potential remains not only for an improved thermochemical pipeline application but also for synthetic fuel or chemical feedstock production applications. Due to resource limitations, the bounds of this study did not include a technical and economic evaluation of the synthetic fuels application.

An integrated cash flow curve for the equilibrium plant is included in Fig. 8.2-1. The cost of energy from the HTGR-R and alternative energy sources was based upon the financial and fuel cost assumptions identified in Table 8.2-3.

TABLE 8.2-1

1170-MW(t) HTGR-R EQUILIBRIUM PLANT BASE CAPITAL COSTS
(1980 $M)

NUCLEAR PLANT	
Structures and Improvements	120
Reactor Plant Equipment	244
Turbine Plant Equipment	48
Electric Plant Equipment	59
Miscellaneous Plant Equipment	14
Main Condenser Heat Rejection System	6
Secondary Helium System	50
Reforming Plant Equipment	162
Subtotal Directs	703
Indirects	263
Contingency	51
Total	1017
PIPELINE AND STORAGE	137
METHANATION PLANT	550
TOTAL PLANT BASE CAPITAL COST	1704

TABLE 8.2-2

HTGR-R EQUILIBRIUM PLANT ENERGY COSTS

	HTGR-R BASELOAD ELECTRIC AND REMOTE ENERGY DELIVERY
PLANT COSTS ($M)	
TOTAL BASE COST	1704
ESCALATION	1761
INTEREST DURING CONSTRUCTION	965
TOTAL INVESTMENT (1995 $)	4430
ANNUAL LEVELIZED POWER COSTS (1995 $) ($M)	
CAPITAL	797
O&M	90
FUEL (THROWAWAY)	118
CASE 1 - TOTAL	1005
FUEL (RECYCLE)	88
CASE 2 - TOTAL	975
PRODUCTS	
BASELOAD ELECTRIC 24 HRS/DAY [MW(e)]	30.3
THERMAL ENERGY 8 HRS/DAY [MW(t)]	1560
ANNUAL LEVELIZED ENERGY COSTS (1995 $)	
ELECTRICITY (MILLS/KW-HR, 70% PLANT CF)*	159
STEAM ($/MBTU, 70% PLANT CF)	90
ALTERNATIVE LEVELIZED ENERGY COSTS (1995$)	
50 MW(t) FBC STEAM ($/MBTU) (1 SHIFT PER DAY OPERATION)	59
OIL (FUEL PORTION ONLY, $/MBTU)	63

*Equivalent to electricity cost estimates for large 800 MWe coal plant developed for 1995 startup.

Figure 8.2-1

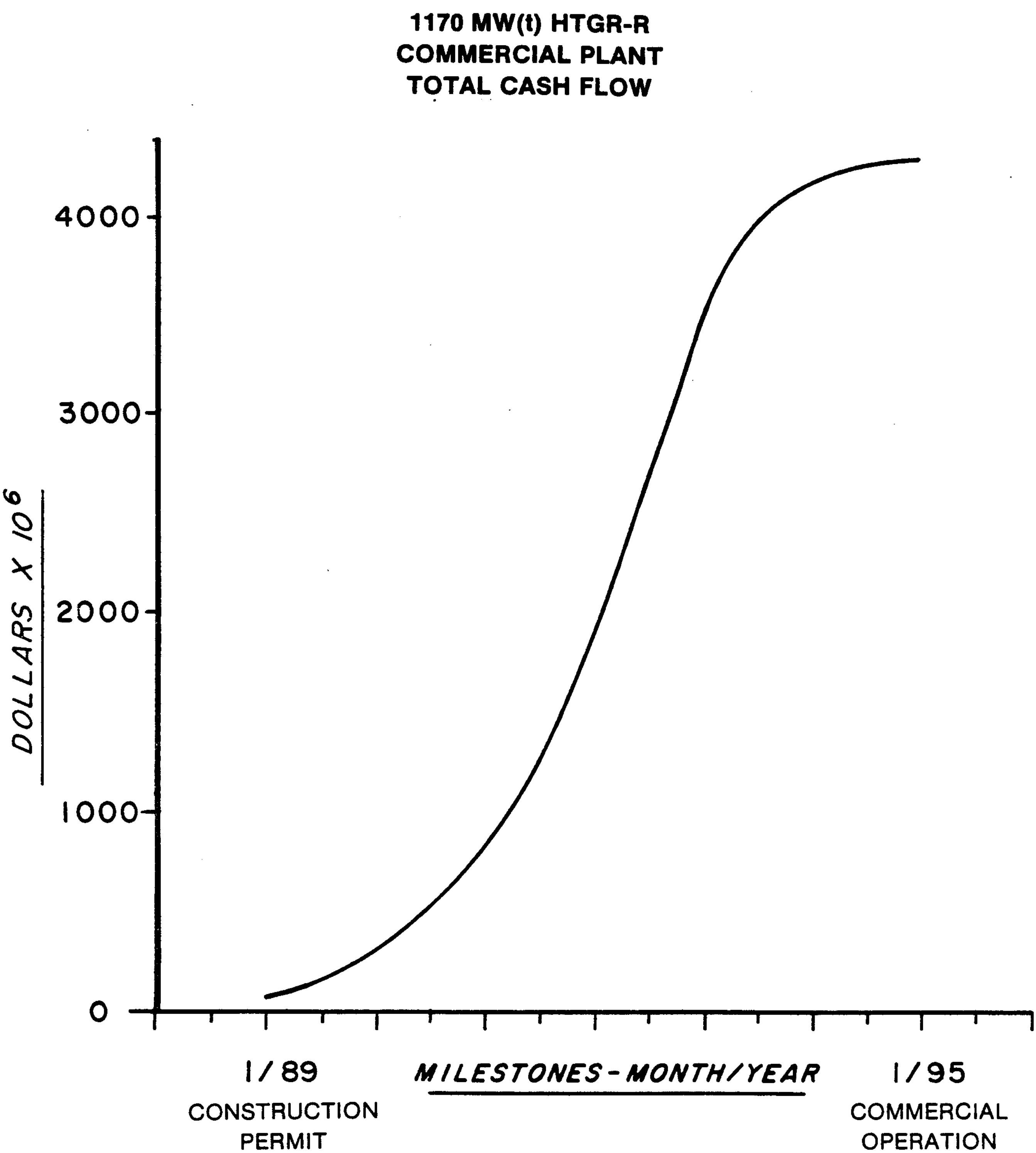

1170 MW(t) HTGR-R
COMMERCIAL PLANT
TOTAL CASH FLOW
4000
3000
2000
1000
0
DOLLARS X 10^6
1/89
CONSTRUCTION
PERMIT
MILESTONES - MONTH/YEAR
1/95
COMMERCIAL
OPERATION

TABLE 8.2-3

HTGR-R COST ASSUMPTIONS

Commercial Plant Basis	Nth Plant
Base Date for All Costs	1/80
Date of Operation for All Plants	1995
Book Life for All Plants	30 yr
Plant Size	1170 MW(t)

Fuel Input Costs (1/80 $)

Coal	
Range	$0.70 - 1.60/MBTU
Average	$1.36/MBTU
Oil	$5.40/MBTU ($27 barrel)
Uranium	
1990	45 $/lb U_3O_8
2000	45
2010	75
2020	120
2030	120

Tails 0.2%
Conversion $5/KG
Enrichment $100/SWU

Financial Factors

Discount Rate	10%
Fixed Charge Rate	18%
Interest During Const. (Simple)	10%
Escalation	
Labor and Materials	6%
Coal	8%
Oil	9%

Fuel Cycle Costs: Based on detailed analysis at GA

O&M Costs:

O&M costs were developed based on information described in ORNL Report "A Procedure for Estimating Nonfuel Operation and Maintenance Costs for Large Steam-Electric Power Plants." HTGR-R O&M costs were assumed to be approximately 30% higher than LWR O&M costs.

Comparative Alternatives

- PWR - Costs based on extrapolated 800 MW(e) design from Reference 1200 MW(e) developed by UE&C and application of equivalent capacity factor
- Coal - Costs based on Reference 800 MW(e) Coal Plant for electric power and a 50 MW(t) FBC for steam production developed by UE&C and application of equivalent capacity factor. Fixed charge rate for FBC assumed to 25% (private financing).
- Oil - Fuel costs only. 85% boiler efficiency assumed for process steam. 12,000 BTU/KW-hr heat rate assumed for peaking electricity.

9.0 RECOMMENDED APPROACH TO HTGR-R DEVELOPMENT

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

- In Sections 3 and 4, environmental and resource utilization advantages relative to competing systems were identified. There is a large potential process heat market for the HTGR if the institutional and regulatory barriers can be overcome within the time frame of commercial deployment (post-2000). Confirmation of the capabilities of the HTGR-R systems to penetrate the process heat market and to provide an economic alternative energy source will require further study. However, both GA and GE project economic incentives for the commercial HTGR-R systems, as identified in Sections 3.1 and 3.2 respectively.

- In Section 5, the HTGR-R lead plant was described and the major technical obstacles were identified. The development program required to support the lead plant effort is in excess of $550M. Consideration of the technical issues, especially the development and code qualification of the metallic heat exchanger components (IHX), results in a lead plant startup schedule targeted for 1998, as depicted in Section 7.

- In Section 8, projections for program development costs, lead project capital costs, and commercial plant product costs are provided. The total design and development cost is expected to be $565M while the lead project cost is estimated to be $1770M. The product cost estimate for the energy delivered from the methanator system is 50-60% more expensive than for projected fossil alternatives in the same time frame.

The projected economic performance, schedule, and deployment cost for the HTGR-R pose issues of reservation for the consideration of the HTGR-R as a lead project. However, the large potential market, fossil resource conservation, and environmental advantages of the HTGR-R system provide incentives for continued examination of the HTGR-R. The nature of the technical issues confronting this plant and its large deployment cost would indicate that the HTGR-R might be better considered as a follow-on plant to the HTGR-SC/C. This aproach may delay the entry of the commercial HTGR-R somewhat but would provide a more conservative and cost-effective path for the HTGR Program.

This decision cannot be made conclusively until the HTGR-R lead plant design basis is examined in more detail both to better establish its commercial potential and to shape the required design and development program. The following activities should be included in the future scope of the HTGR Program:

- Consideration of potential HTGR-R lead plant performance and cost improvements emphasizing improvements in the reformer, TCP, and methanation plants (energy delivery system). This area currently includes nearly one half of the total lead plant cost.

- Examination of alternative configurations (direct cycle), ranges of core outlet temperatures, and basic performance and cost improvements in the nuclear heat source/balance of reactor plant.

- Evaluation of alternative applications to include the production of synfuels or other direct coupled process heat applications.

- Characterization of the potential market for the HTGR-R. The market and the technical/institutional barriers to HTGR-R penetration must be better defined to firmly establish the commercial/national incentives for the HTGR-R system.

APPENDIX A

NUCLEAR HEAT SOURCE
DESIGN AND DEVELOPMENT PROGRAM

A.0 NUCLEAR HEAT SOURCE DESIGN AND DEVELOPMENT PROGRAM

The design and development program for the HTGR-R plant provides the support activities necessary to design, construct, and operate a reformer plant within the defined scope, objectives, and schedule established for the plant. These activities are grouped in two categories consisting of the specific program and the generic technology program. The specific program activities relate directly to particular design aspects of the HTGR-R plant, while the generic technology program activities are applicable to all HTGR design applications.

A.1 Specific Program

The major design and development activities related to the nuclear heat source (NHS) of the HTGR-R plant (and some secondary loop components within the NHS supplier's responsibility) are as follows:

1. Licensing.
2. Safety and reliability.
3. Systems engineering.
4. Reactor vessel and components.
5. Main helium circulators.
6. Intermediate heat exchanger.
7. Main and auxiliary circulator service system.
8. Plant protection system.
9. Plant control system.
10. Plant data acquisition and processing system.
11. Analytical instrumentation system.
12. Other miscellaneous NHS design support.
13. Steam generator.*
14. Containment isolation valves.*

The major design and development activities within each of the work areas are routine in that they are needed to develop and document any design. These activities are as follows:

*Secondary components for which the NHS supplier is responsible.

1. Design, analysis, proof testing, and documentation. Documentation includes the preparation of system descriptions, process flow diagrams, piping and instrumentation diagrams, specifications, operating manuals, and service manuals.

2. Safety and reliability support for the design effort through analysis and probabilistic risk assessment.

3. Licensing support for the design effort through interpretation of regulatory requirements and establishment of strategies for compliance with these regulations.

4. System design support through optimization studies, static and dynamic performance analyses, and acoustic analysis.

5. Technical support of project management, cost development, and procurement activities.

6. Liaison with the architect-engineers, customers, suppliers, plant operators, and governmental agencies.

7. Project management and support to provide overall project management of design and development of the HTGR-R 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, reactor turbine system equipment list, etc.

8. Quality Assurance 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. This includes other 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.

9. 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 Special Development Activities

Two of the large development areas in the HTGR-R specific program that do not fall into the routine design requirement categories or entail extensive development are the IHX and the primary helium circulator. The design/development and testing associated with these are described below.

- IHX Design and Development

  - Scope - The scope of work consists of establishing and executing a program that provides for the design, development, and manufacture of IHXs and associated auxiliary equipment for an HTGR-R plant. The program includes accomplishment of all the activities required to meet this objective beginning with the conceptual design of the component and extending through technical support during manufacture, shipment, installation, startup, and plant acceptance by the owner. The program provides for subcontracting the IHX final design and manufacture to a heat exchanger supplier and therefore includes preparation of a bid package for potential suppliers' quotation. Technology exchange technical review and coordination between the selected supplier and the NHS supplier are also included in the program.

  - Major Activities - Figure A.1.1-1 illustrates the overall schedule of the major activities for the IHX design and development program. The emphasis in the early part of the program will be on the engineering design and development work required for formulation of a viable design of the component. 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 analysis, mechanical design, and high-temperature structural analysis associated with steady-state and transient operation. System and physical interfaces will be addressed, as well as cost and preparation of the design specification for the heat exchanger supplier bid package.

    Testing to be performed will include flow distribution, high-temperature materials design data, heat transfer, fretting and wear, vibration, seismic, and acoustic tests; methods associated with high-temperature heat exchanger design, sizing, performance, and structural analysis will be developed.

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

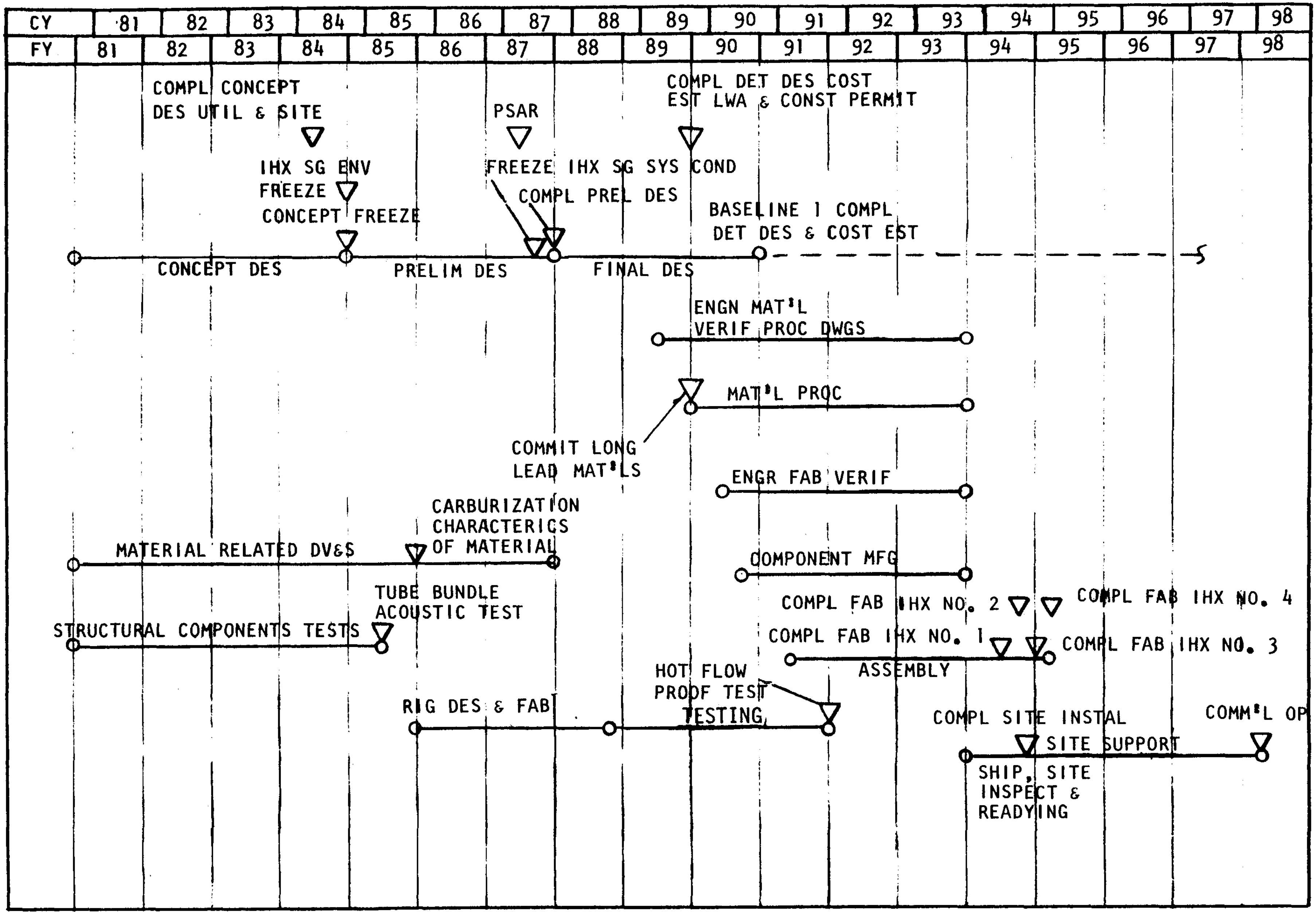


Figure A.1.1-1 HTGR-R IHX design and development

- Status - The IHX design is still in its early preconceptual phase. Concept selection is ongoing and current component evaluations associated with short-term plant studies include minimal supporting mechanical design, thermal, and structural analyses.

  A list of required test and methods development programs has been prepared. These programs are considered essential to heat exchanger development to obtain basic technology or to verify the design. The bulk of this test and methods work has not been started.

- Main Circulator Design and Development

  - Scope - The main circulator design includes detail preparation of drawings and supporting analyses of the main circulator, its electric motor driver, and the primary closure. The drawings will be taken from the conceptual layout phase to the detailed manufacturing level. The supporting analyses will include fluid flow, aerodynamic, rotor dynamics, stress, vibration, thermal, and electrical analyses. In addition, supporting safety and reliability data will be prepared.

    The development program comprises three major phases. The first will be a full-size test of the water bearings and seals. The second phase will be an atmospheric air test of the machine aerodynamics. This test will be performed on a 1/3-size test rig. The final phase will be a full-scale test of a prototype circulator under reactor operating conditions.

    The objective of the program is to produce a main circulator that will meet plant performance requirements. To meet these requirements, the unit must be reliable and relatively maintenance free over the life of the plant. Figure A.1.1-2 illustrates the overall schedule for the main helium circulator design and development program.

  - Major Design Activities Planned - The design work will be broken into three broad areas: conceptual, preliminary, and final. In the conceptual and preliminary areas, the design will be developed in sufficient detail to allow overall interaction with other interfacing systems and components. These interactions will include rotor dynamics, off-design operation, plant transients, and electrical response. After the preliminary design phase, the final design and manufacturing will be carried out.

    A test program will be carried out in parallel with the design work. The objective of the test program is to verify the design and analysis of the helium circulator in logical

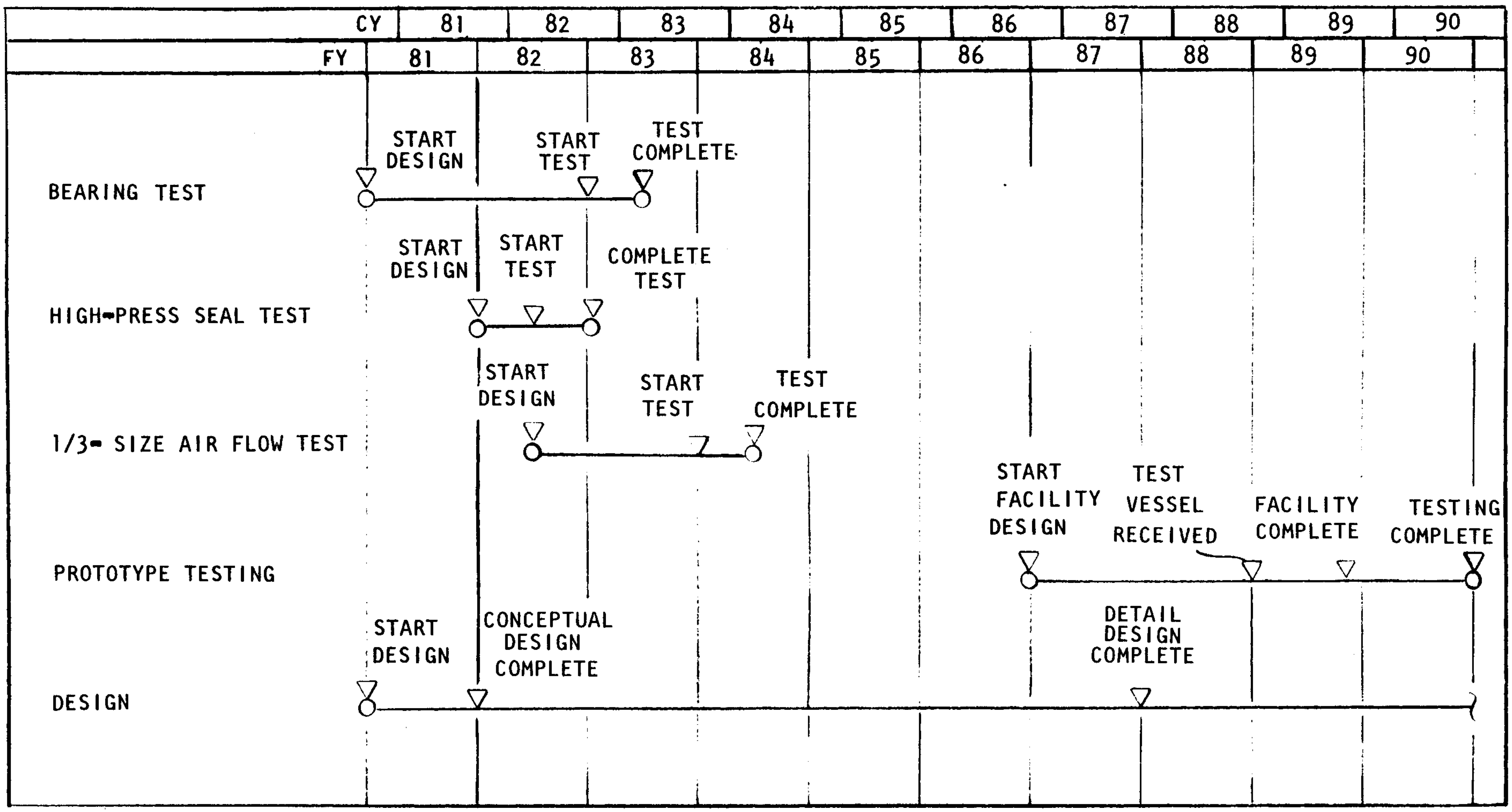


Figure A.1.1-2 HTGR-R Main Circulator Design and Development steps. The program will be conducted in three major areas. First, the bearing test rig will be used to confirm the bearing spring rates. This test will also verify the performance characteristics of the integral bearing pump, seal system, and the auxiliaries. Next, a 1/3-scale air flow test rig will be used to verify the aerodynamic performance of the compressor and diffuser blading. It will also be used to study the shutoff valve transient response and aerodynamic characteristics. The full-size prototype test will allow the evaluation of a typical compressor under full-power reactor operating conditions. This test will be conducted in a helium test facility. It will allow full control of the major circulator parameters, i.e. flow, temperature, pressure, speed, and pressure rise. This program will also test the motor controller and the auxiliary systems.

- Design Status - The proposed circulator differs from the circulators used in the FSV plant and the LHTGR programs mainly because it is electrically driven. Much of the test and design experience from those programs will be utilized. Major changes in the bearing water and service system have been incorporated in order to improve the reliability of the shaft seal system and thus prevent future water ingress from the circulators.

- Major Development Activities Planned - A test program will be conducted in parallel with the design. The objective of the test program is to verify the design and analysis of the helium circulator in logical steps.

  First, the bearing test rig will be used to confirm the bearing spring rates. This test will also verify the performance characteristics of the integral bearing pump. It will also prove the shaft seal and the backup seal functions, as well as verify the performance of various components in the auxiliary system and their interaction during transients.

  Next, a 1/3-scale air flow test rig will be used to verify the aerodynamic performance of the compressor and diffuser blading. It will also be used to study the shutoff valve transient response and aerodynamic characteristics.

  The full-size prototype test will allow the evaluation of a typical compressor under reactor operating conditions. This test will be conducted in a helium test facility. Testing will be done in a closed-loop vessel with a variable flow resistance. It will allow full control of the major circulator parameters, i.e., flow, temperature, pressure, speed, and pressure rise.

- <u>Development Status</u> - A detailed bearing test rig design has been developed. In addition, a number of major components have been procured. The major tasks to be done are manufacturing and testing.

  The air flow test rig is only developed in terms of conceptual drawings. The full size prototype test program has only reached the planning stage.

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 1981 to 1982 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

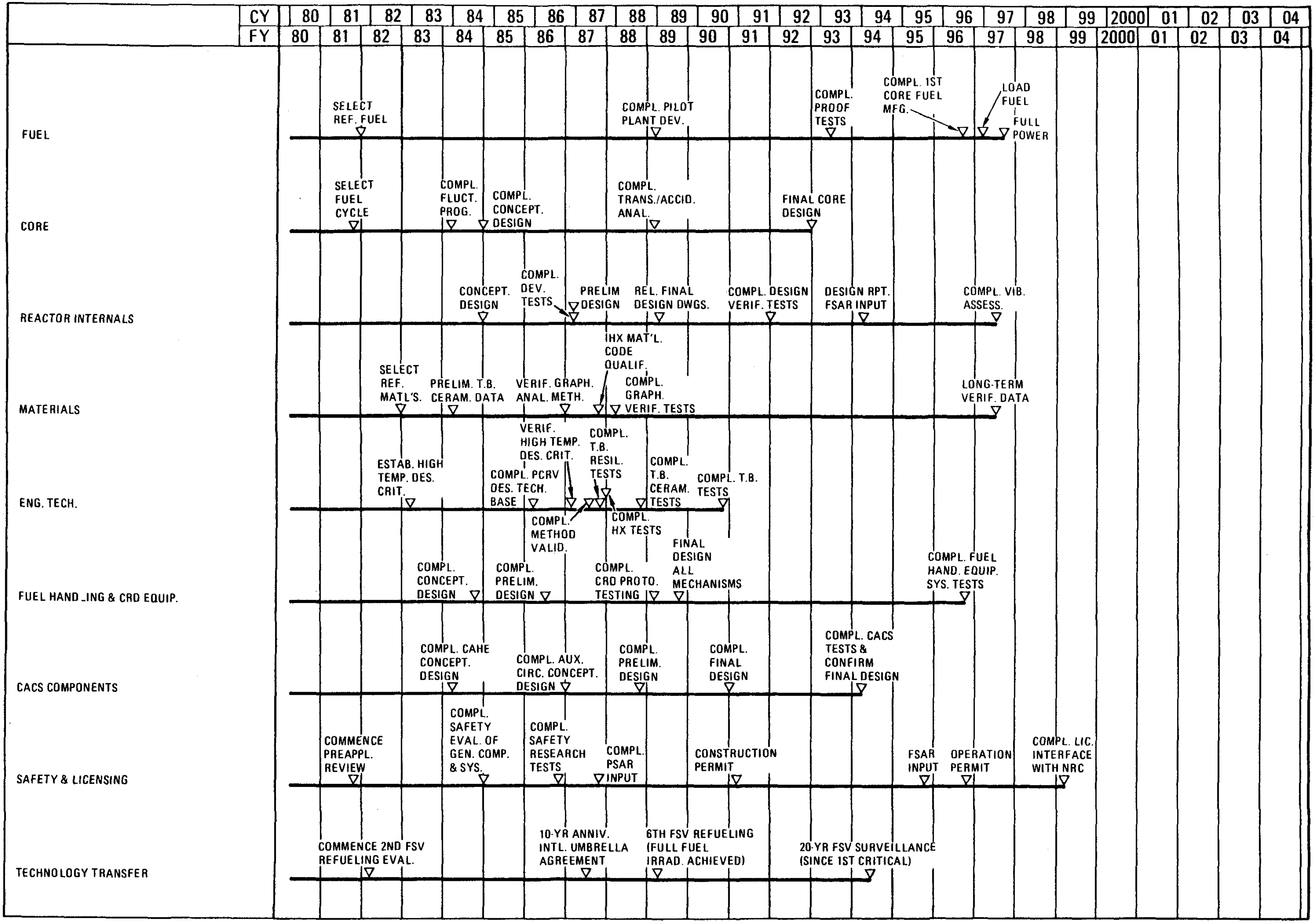


Figure A.2-1 HTGR Generic Technology Program: HTGR-R Lead Plant

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 September 1981 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 postirradiation 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

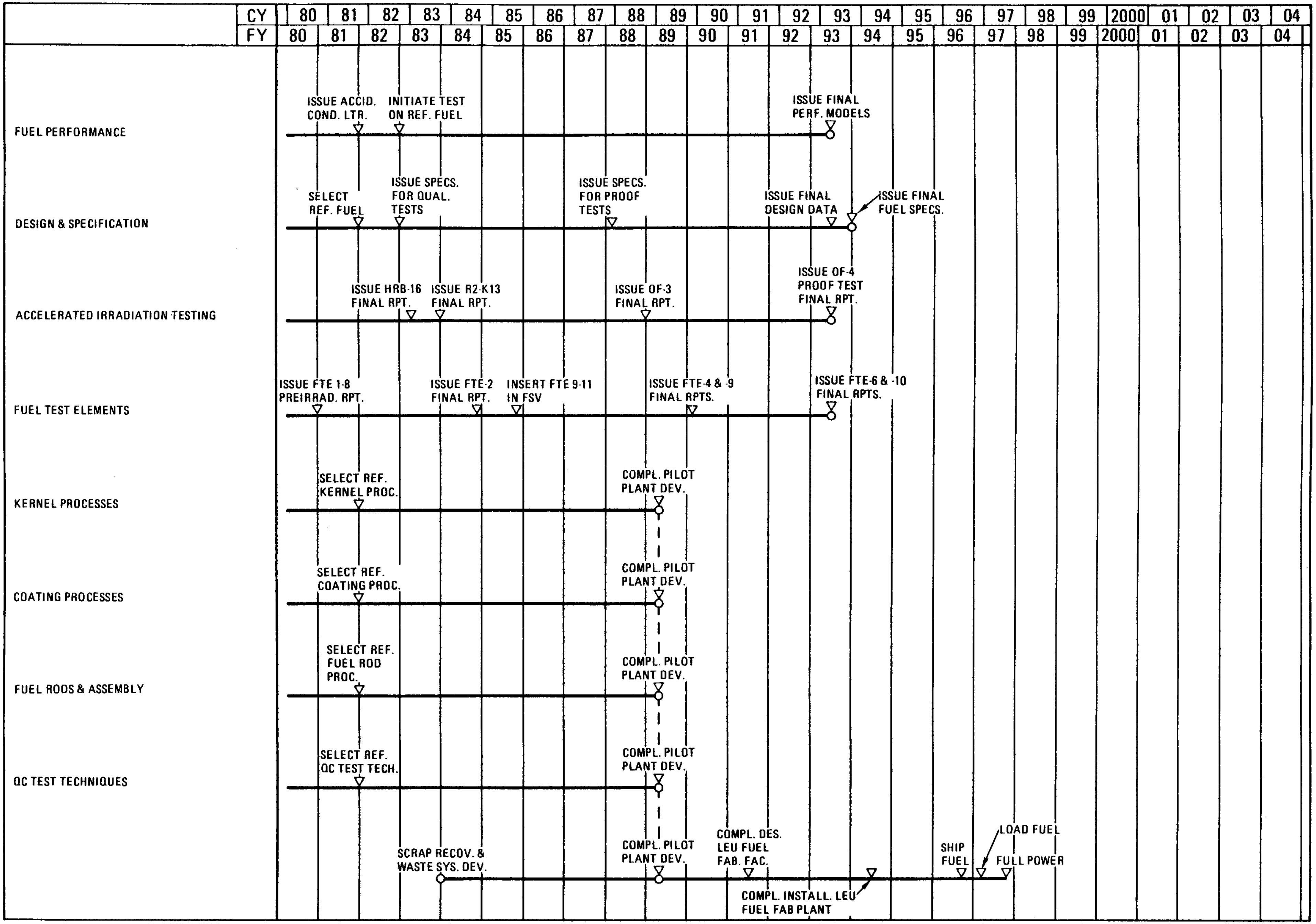


Figure A.2.1-1 HTGR Generic Technology Program Detailed Fuel Milestone Schedule: HTGR-R fuel under isothermal, thermal gradient, and simulated unrestricted core heatup conditions will be performed to generate the data required to develop and verify these performance models. Initial tests will be performed on candidate fuels to support the reference fuel selection in September 1981. 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 September 1981, 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)C_2 fuel kernels produce less uniform product size distributions than GSP processes. Moreover, the dry-mix process is not applicable for UCO and UC_2/ThO_2. 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.

UF_6-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 UF_6-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

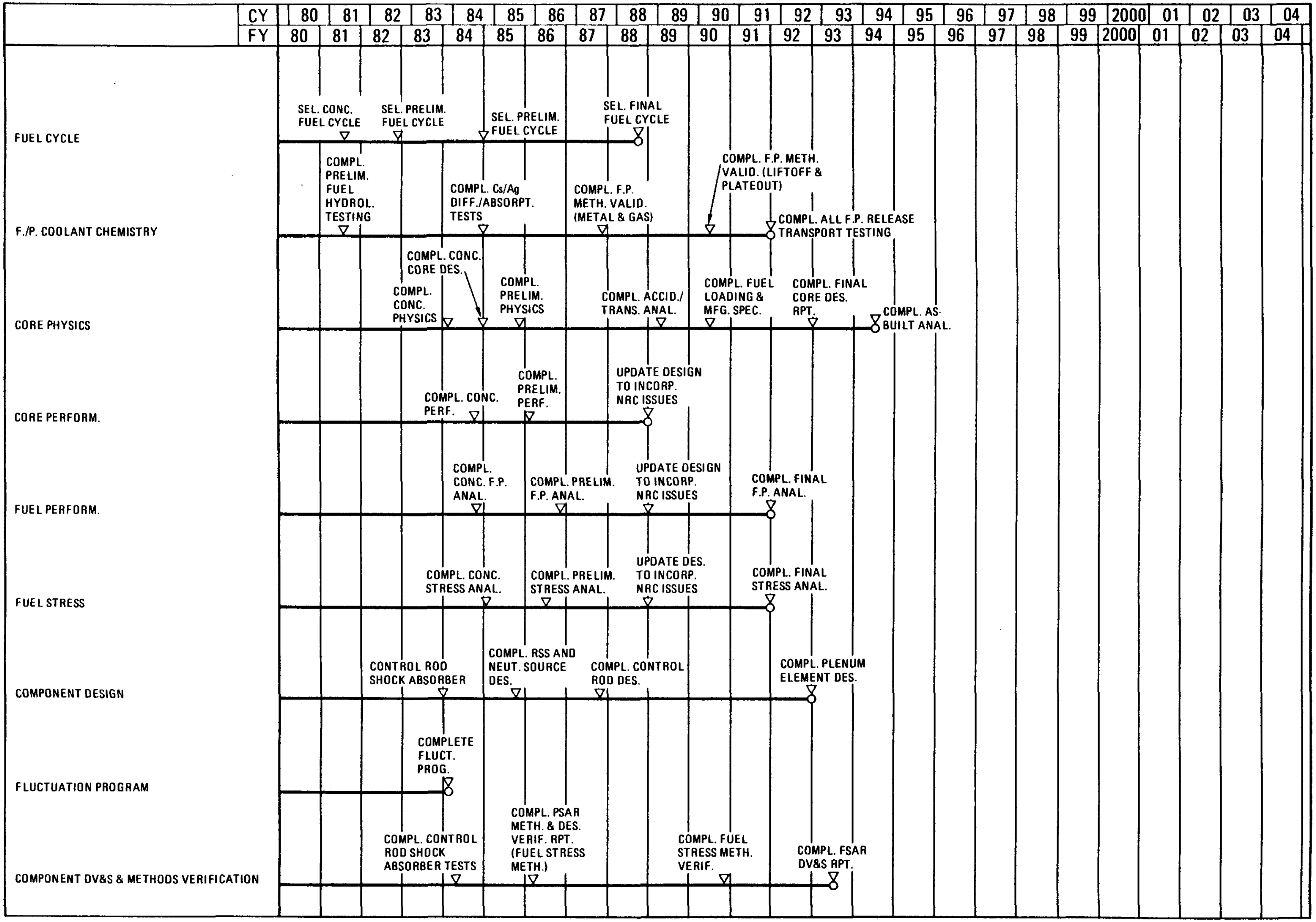


Figure A.2.2-1 HTGR Generic Technology Program Detailed Core Milestone Schedule: HTGR-R components on a schedule consistent with the plant licensing and construction program.

- 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: UC_2, UCO, and zirconium-buffered UO_2.

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 cooperation 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 PCRV. The PSR is keyed to the core barrel, which is in turn radially keyed to the PCRV cavity wall to prevent lateral motion of the entire core assembly.

  In the LHTGR, the core and CSF are supported directly through the PCRV 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 PCRV 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

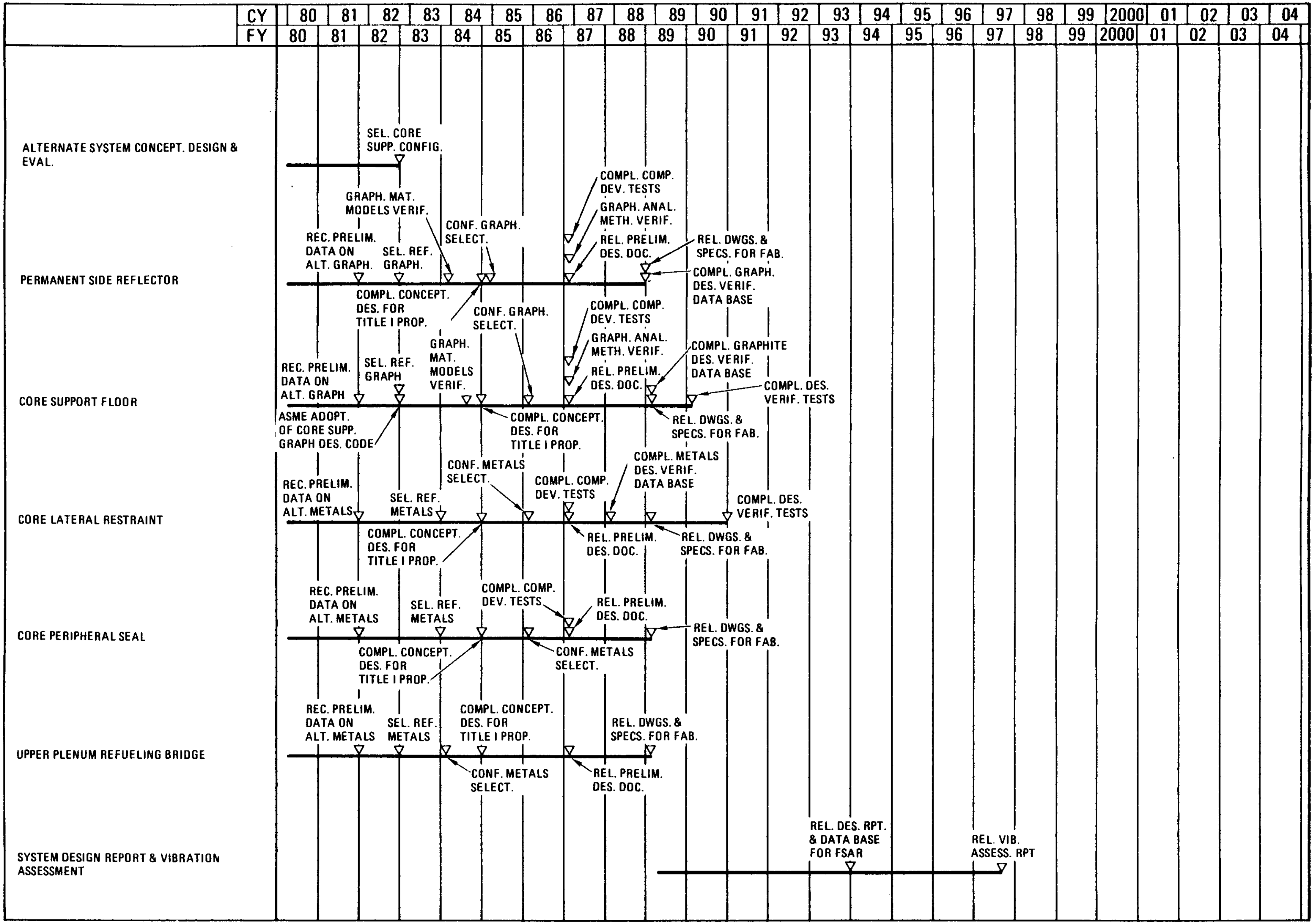


Figure A.2.3-1 HTGR Generic Technology Program Detailed Reactor Internals Milestone Schedule: HTGR-R and safe shutdown earthquake. A basic concept has been tentatively identified, but further design and analysis will be required to confirm that or any other alternative design. The potential benefits 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.

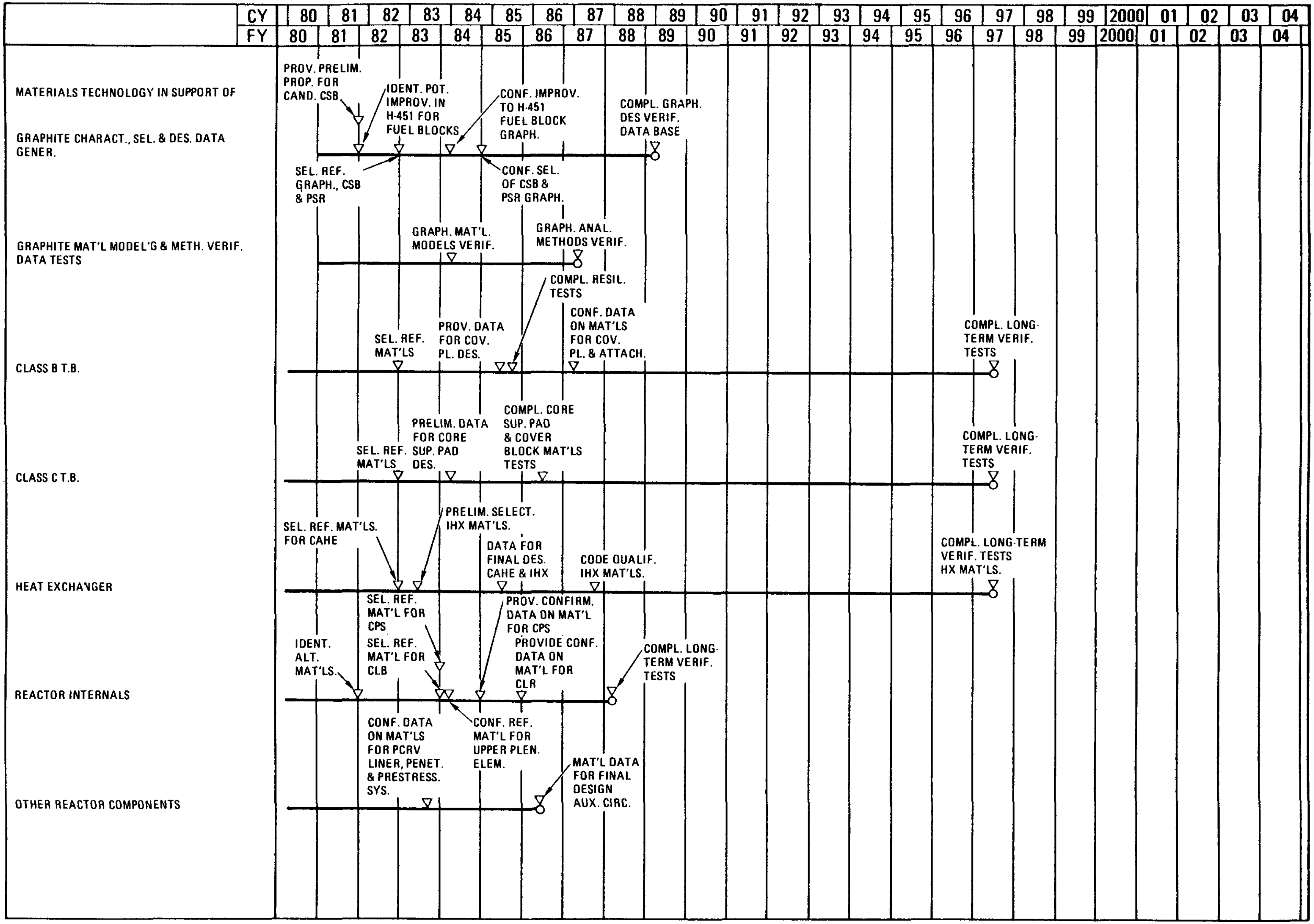


Figure A.2.4-1 HTGR Generic Technology Program Detailed Materials Milestone Schedule: HTGR-R

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.

- <u>Planned Program</u> (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 [≥850°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

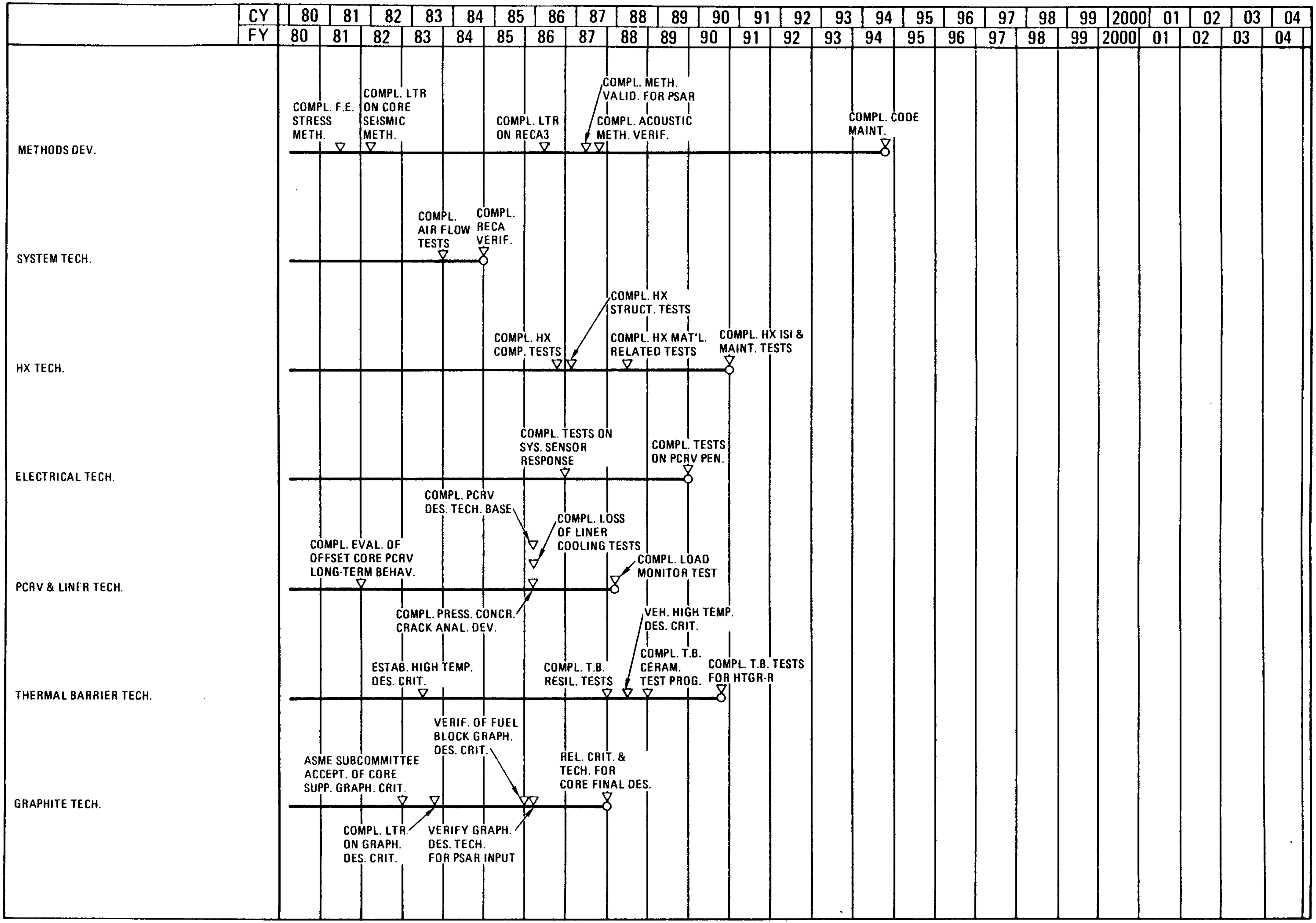


Figure A.2.5-1 HTGR Generic Technology Program Detailed Engineering Technology Milestone Schedule: HTGR-R studies, and a performance analysis computer program for helical coil finned tubes and axial flow heat exchangers.

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 PCRV 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 CO_2-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 FIRL 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 fusable links that operate the reserve shutdown system hoppers and for test circuitry to check fusable 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:

<u>Phase 1</u> - A general checkout in air for assembly, installation, and operating characteristics in a simulated refueling penetration and core region mock-up.

<u>Phase 2</u> - Cyclic operating tests of 630,000 jogs and 5400 scrams, performed in simulated reactor conditions of helium and temperature.

<u>Phase 3</u> - 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

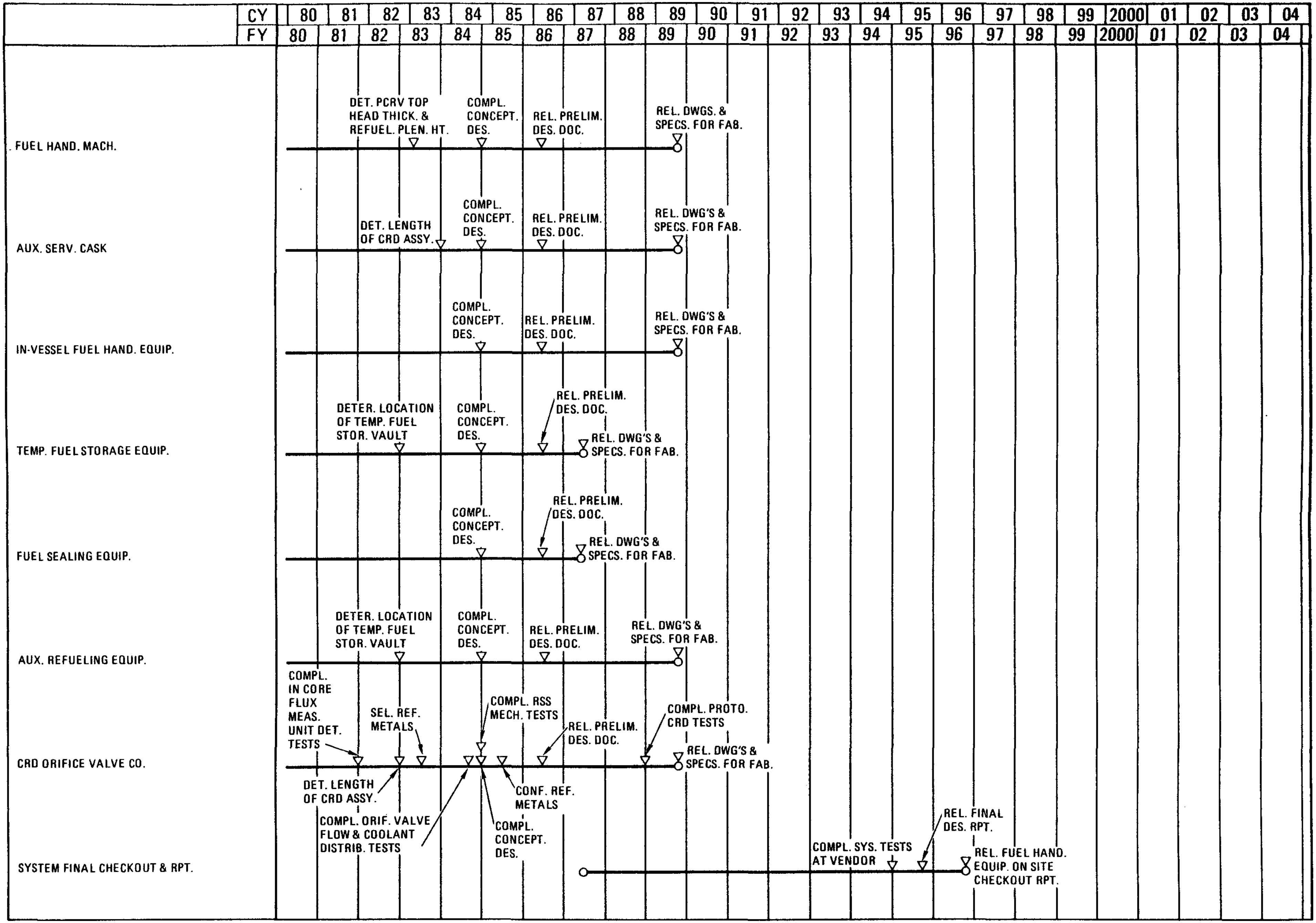


Figure A.2.6-1 HTGR Generic Technology Program Detailed Fuel Handling and CRD Equipment Milestone Schedule: HTGR-R will be assembled. Each major operational component will be tested individually and in a full system test before the fuel handling system is delivered to the site. Once installed at the 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 x 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 PCRV blowdown gas mixing and heat transfer in the containment on the peak containment pressure response to PCRV 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 plateout 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. Interpetation 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 DBDAs 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

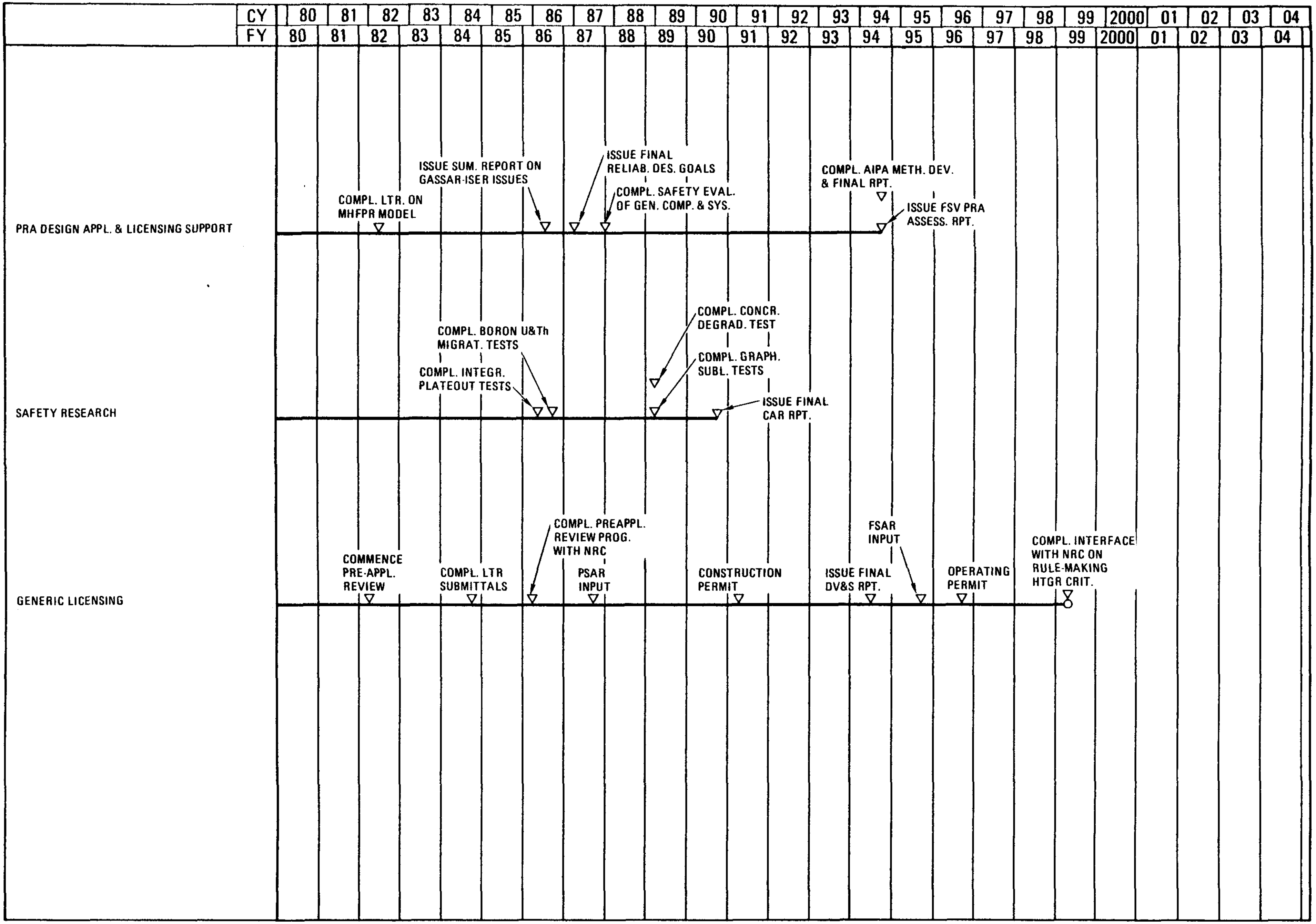


Figure A.2.7-1 HTGR Generic Technology Program Detailed Safety and Licensing Milestone Schedule: HTGR-R development. These include core support and fuel element graphite burnoff, fission product source terms and circulating activity, primary coolant impurity levels, carbon deposition, and fission product 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 BC_4 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.

- <u>Licensing</u> - 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 intiation 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.

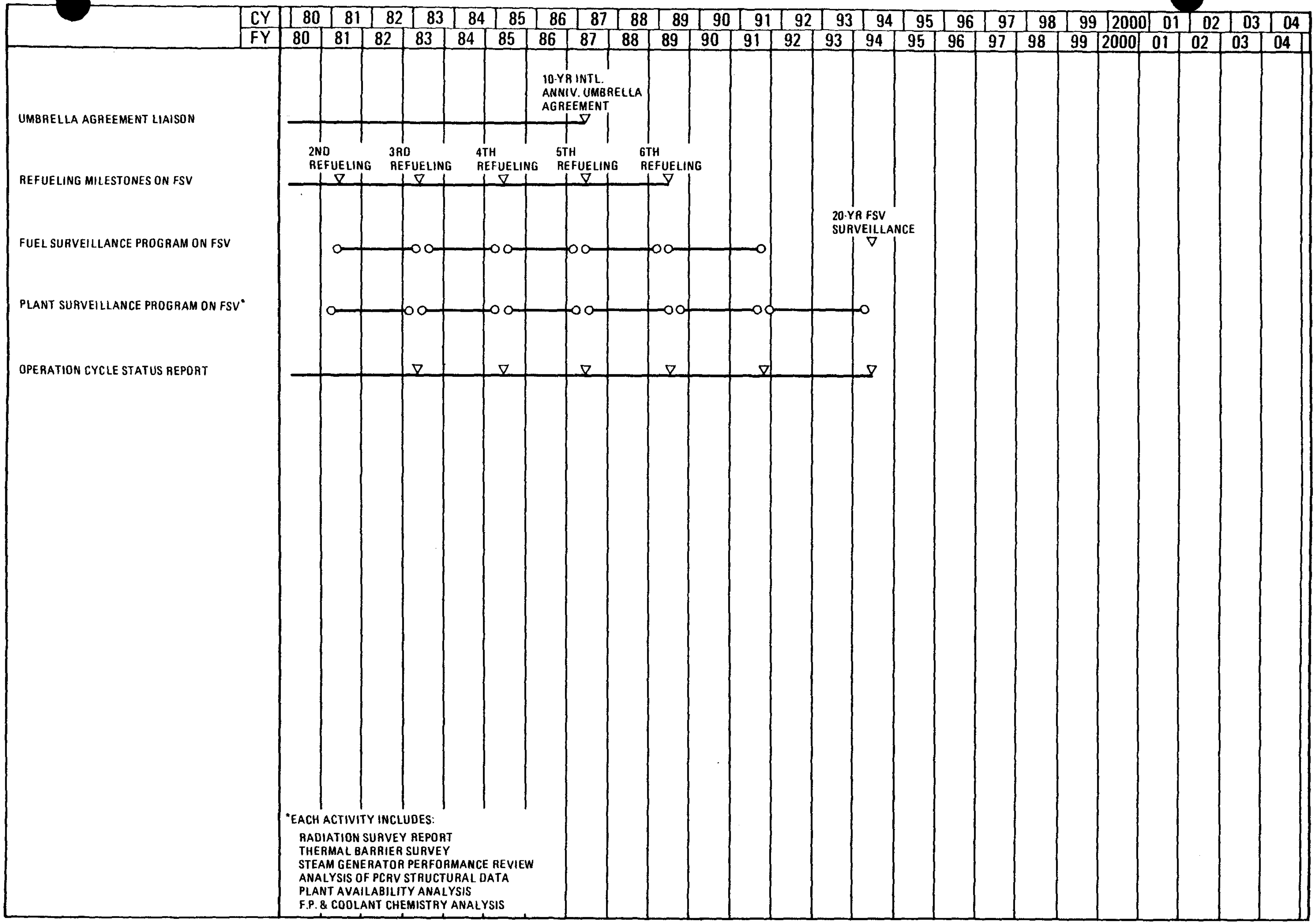


Figure A.2.8-1 HTGR Generic Technology Program Detailed Technology Transfer Milestone Schedule: HTGR-R

- 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 nondestructive 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 develoment 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 progam 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 feedthrough, 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.

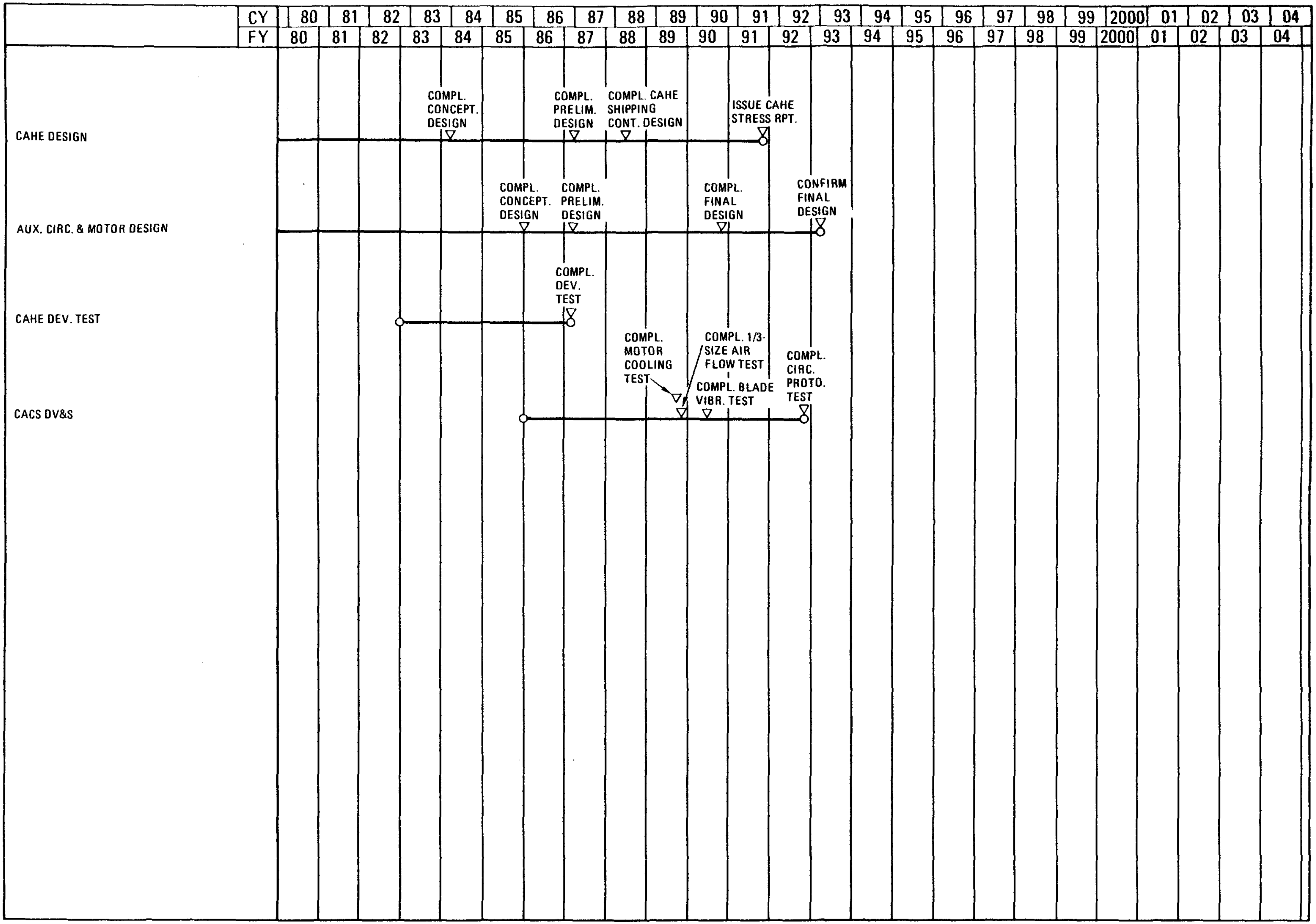


Figure A.2.9-1 HTGR Generic Technology Program Detailed CACS Components Milestone Schedule: HTGR-R

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.

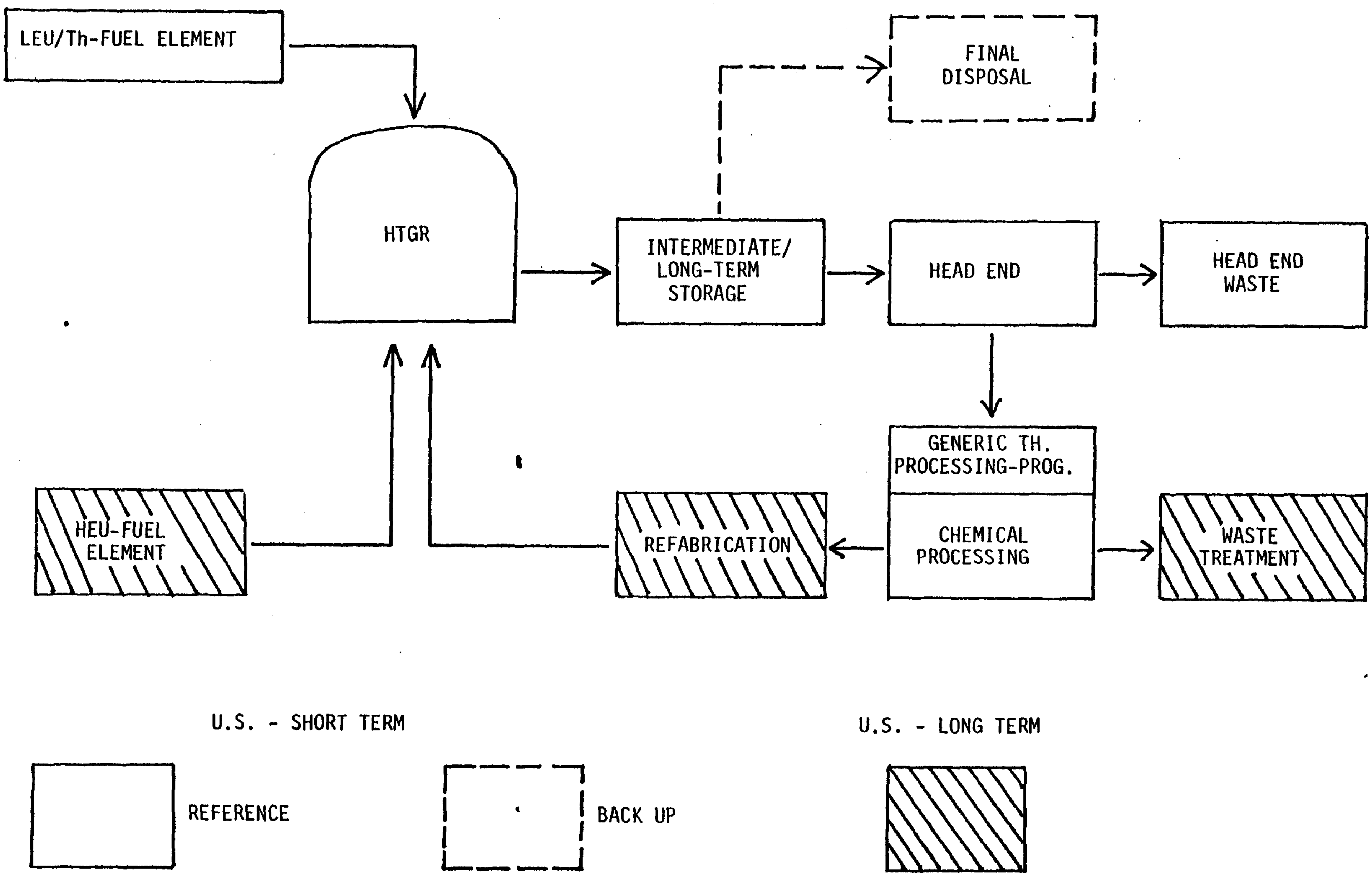


Figure A.3-1 U.S. Reference Fuel Cycle Strategy

- 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. <u>Off-Gas Treatment</u>: 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. <u>Solvent Extraction Pilot Plant</u>: This work includes pulse column operation and data acquisition to verify computer codes for predicting pulse column performance.

5. <u>Laboratory Studies</u>: 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. <u>HEF Technical Support</u>: This work includes continuing technical support tasks which address generic design aspects for a spent fuel treatment plant.

B. <u>Refabrication</u>

It is proposed that refabrication development be resumed as a major activity as the national need for recycle becomes more clearly defined.

C. <u>U.S./FRG Cooperative Program</u>

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.

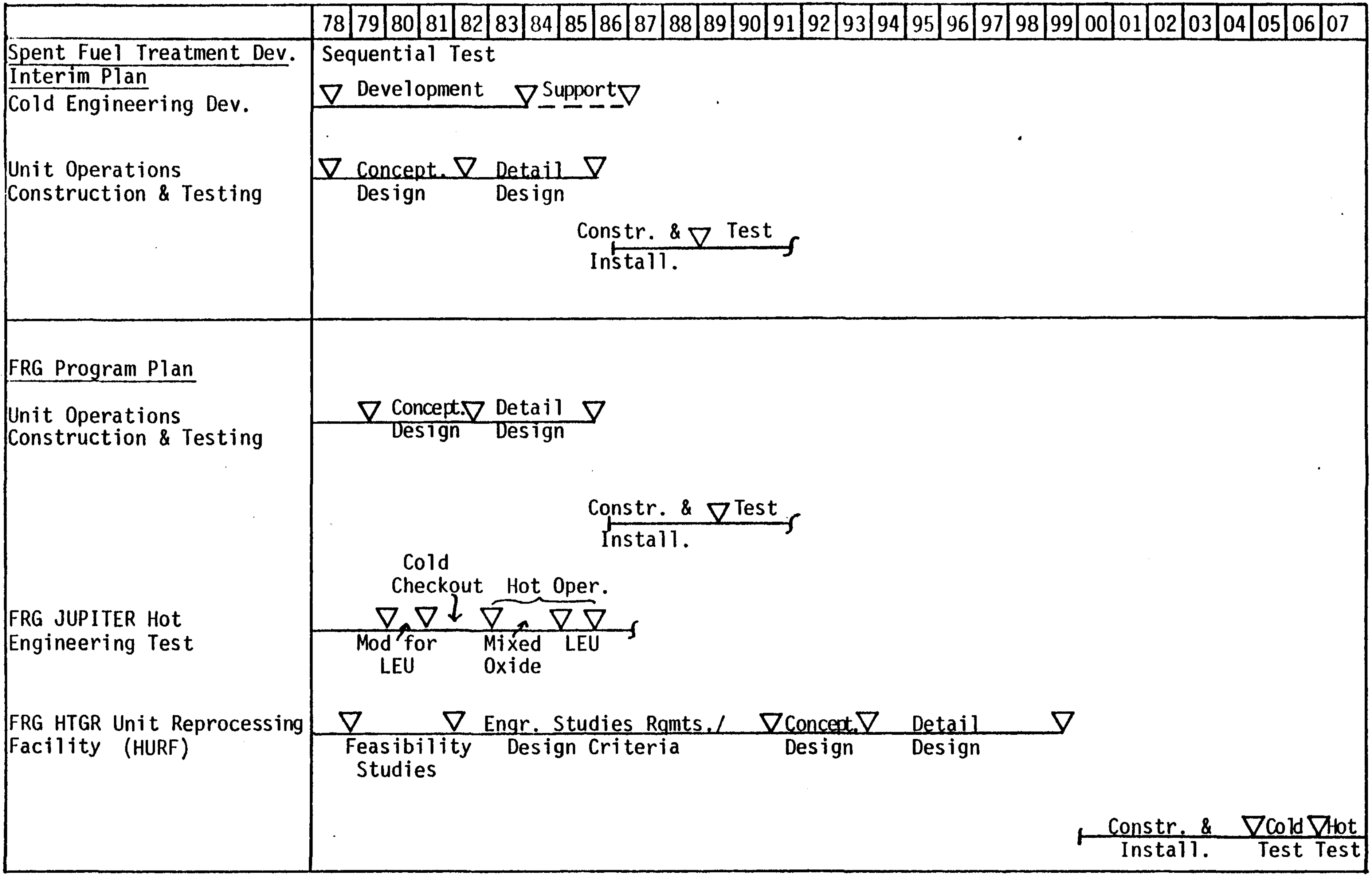


Figure A.3-2 HTGR Spent Fuel Treatment Development Program Schedules

APPENDIX B

ALTERNATE PLANT STUDY -
HTGR-SALT

B.0 ALTERNATE PLANT STUDY - HTGR-SALT

B.1 Introduction And Summary

This appendix documents the results of a preliminary engineering study of the HTGR-SALT concept, which combines a nuclear heat source (NHS) with a base-loaded electrical power station and a remotely sited peaking power plant. The particular HTGR-SALT application studied is a potential alternative to the thermochemical pipeline (TCP) concept.

The HTGR-SALT was specifically configured to match the capabilities of the HTGR-R lead plant. The same ground rules established for the HTGR-R lead plant (TCP) were applied to the HTGR-SALT design. These consisted of the following:

1.	Base-load steam conditions	950°C, 2400 psia
2.	Pipeline distance	32.2 km (20 miles)
3.	Hours of peaking plant operation	8 h/day
4.	Gross base-load electric output	~290 MW(e)
5.	Peaking turbine plant arrangement	No reheat, minimum use of extraction steam for feedwater heating

A preconceptual cost estimate was prepared for the HTGR-SALT system and is compared with the HTGR-R lead plant cost estimate in Table B.1-1. A product cost assessment for these two plants was also made, along with extrapolated product costs for similar plants coupled to a 160.9-km (100-mile) pipeline. These results are provided in Table B.1-2. These estimates are very preliminary in nature, and thus the application of these data should be confined to a comparison of alternative HTGR concepts. In the case of both plant designs, improvements in plant performance and possibly capital cost are expected as the design develops further. These preliminary results identify an advantage for the HTGR-SALT at 32.2 km (20 miles) transmission distance and an advantage for the HTGR-R at 160.9 km (100 miles). Further design and optimization are required to quantify these relative advantages.

The HTGR-SALT contains three basic systems: (1) the NHS primary helium circuit, (2) a secondary helium circuit, and (3) a thermal transport circuit. The latter two circuits are external to the NHS. The secondary helium circuit provides heat to a conventional base-loaded steam cycle power plant. The thermal transport system is a molten salt loop that bridges a 32.2-km (20-mile) physical separation between the NHS base-loaded plant complex and a steam cycle peaking power station. Nuclear heat not used for base-loaded plant operation is delivered via the secondary helium circuit to high-temperature thermal storage tanks in the salt loop. During periods of peak power demand, this thermal storage capacitance is discharged to provide the heat input to the peaking power plant. The thermal transport fluid is a commercially available heat transfer salt (HTS) based on a eutectic composition of potassium nitrate, sodium nitrate, and sodium nitrite.

TABLE B.1-1
1170-MW(t) HTGR-SALT/1170-MW(t) HTGR-R COST COMPARISON
(1980 Dollars)

Account No.	Account Description	1170-MW(t) HTGR-R	1170-MW(t) HTGR-SALT	Remarks (HTGR-SALT Versus HTGR-R)
21	Structures and improvements	117.6	100.0	Smaller containment, no reformer prestressed concrete pressure vessel or heat exchanger buildings
22	Reactor plant equipment			
	Nuclear steam supply system (NSSS) base	186.9	156.8	Pricing provided by GA
	NSSS in balance of plant (BOP)	26.1	25.6	Pricing provided by GA
	Balance of reactor plant	33.7	33.7	Same nuclear island design
	Total	246.7	216.1	
23	On-site turbine plant	47.4	50.0	Less main steam piping, more expensive steam generator/reheater, same steam conditions
24	Electric plant equipment	58.8	50.0	70% reduction in compressor/pumping power
25	Miscellaneous plant equipment	13.5	13.5	Same miscellaneous equipment
26	On-site cooling towers	6.4	6.4	Same waste heat rejection
27	Secondary helium system	49.8	38.0	60% smaller compressors, less piping
28	Reforming plant equipment	147.5	20.0	He/salt heat exchanger versus reformer and heat exchanger trains; pumps versus compressors
31	TCP/molten salt transfer system	60.0[a]	200.0	Install high-temperature salt storage, piping, and concrete trench versus cold syngas piping
41	Methanation/remote turbine plant	400.0[a]	125.0	Same steam production, no methanation equipment
	Total direct costs	1147.7	819.0	
	Total indirect costs	285.0	300.0	819.9 versus 687.7 direct cost
	Contingency	54.0	61.0	1119.0 versus 972.7 direct plus indirect cost
	Total base cost	1486.7	1180.0	

[a] Includes indirects and contingency.

TABLE B.1-2
COMPARISON OF COST OF ELECTRICITY
(Millions of Dollars)

	HTGR-R 32.2 km (20 miles)	HTGR-R 160.9 km (100 miles)	HTGR-SALT 32.2 km (20 miles)	HTGR-SALT 160.9 km (100 miles)
Total base plant cost (1980)	1490	1565	1180	1880
Escalation	1540	1620	1220	1945
Interest during construction	840	880	665	1060
Total project investment cost (1995)	3870	4065	3065	4885
Annual levelized power costs (1995)				
Capital Fixed charge rate = 0.18 (0.14)	697 (542)	732 (569)	552 (429)	879 (684)
Operation and maintenance	86	86	86	86
Fuel (recycle)	115 (87)	115 (87)	112 (81)	112 (81)
Total	898 (715)	933 (742)	750 (596)	1077 (851)
Net output, MW(e)				
Baseload	30	27	213	154
Peaking	430	430	485	420
Cost of peaking with base load pegged at 123 mills/kWh, mills/kWh	996	1039	594	1119

The main advantages of the HTGR-SALT concept are:

1. Load flexibility. The HTGR-SALT plant can be tailored to specific utility demands through adjustments in its base-to-peak load split and thermal storage facilities.

2. Site flexibility. The molten salt thermal transport system makes it possible to site a reactor a considerable distance from an industrial complex and still be able to deliver heat to this complex for electricity and/or process steam generation purposes without unacceptable transmission loss.

3. Resource conservation/utilization. The HTGR-SALT concept permits continuous operation of the NHS at full power, taking full advantage of lower fuel cost.

4. Technology utilization. The HTGR-SALT concept can utilize a high-technology heat source, such as the HTGR.

A more detailed description of the HTGR-SALT plant and its technical and economic aspects are presented in the following sections.

B.2 Plant Description

B.2.1 System Schematic

A schematic flow diagram of the HTGR-SALT plant showing the major components and key system design parameters is shown in Fig. B.2.1-1. Heat generated in the reactor is transported by the primary coolant helium to the intermediate heat exchanger (IHX), which transfers it to the externally located secondary helium system. Four primary/secondary helium loop combinations are required for the plant. Parallel flow circuitry is employed to route the hot secondary helium simultaneously to the steam generating equipment for the base-load plant and to an array of He/salt heat exchangers that provides the thermal input to the molten-salt-heated peaking plant.

In the secondary helium circuit, hot helium from the IHX is introduced to three parallel legs containing the reheater, superheater, and He/salt heat exchangers. The combined discharge helium from the reheater and superheater is then routed to the boiler, after which it joins the helium flowing from the He/salt heat exchanger and is pumped back to the IHX by the secondary helium circulator. Initially, the HTGR-SALT studies considered four completely independent secondary helium circuits, as reflected in Figs. B.2.1-1, B.2.2-2 (see Section B.2.2), and B.2.2-3 (see Section B.2.2). When component design investigations revealed that the base-loaded plant heat duty could be handled with a single steam generator/reheater set, the secondary helium circuit concept was modified to permit manifolding of the four helium loops to one steam generator/reheater interface. This change is reflected in the helium-to-steam component designs and the HTGR-SALT cost estimate.

The water/steam circuitry and state points for the base-loaded power plant are consistent with conventional steam cycle technology. The steam plant is based on a simple reheat/regenerative cycle with conventional condensate polishing and deaerating provisions. The base-loaded plant has a 290-MW electrical output.

The third parallel leg of the secondary helium system contains the He/salt heat exchanger, which transfers the remaining secondary loop thermal output to the molten salt transport system. Hot salt leaving the He/salt heat exchanger is used to charge the hot storage tanks sited at both the reactor and the peaking plant sites. During peak demand periods the salt is circulated from hot storage to cold storage through steam generators that power the peaking plant. The peaking plant is separated from the He/salt heat exchangers at the reactor site by 32.2 km (20 miles) of hot and cold molten salt piping with approximately ten pipeline booster stations sited periodically along its length. A set of hot and cold storage tanks is provided at each end of the pipeline to permit the piping to be maintained continuously at operating temperature, thus reducing thermal shock problems during startups and shutdowns. The steam peaking power plant assumed for this study is based on conventional boiler technology and can deliver 505 MW(e) of peaking power for 8 hours continuously each day.

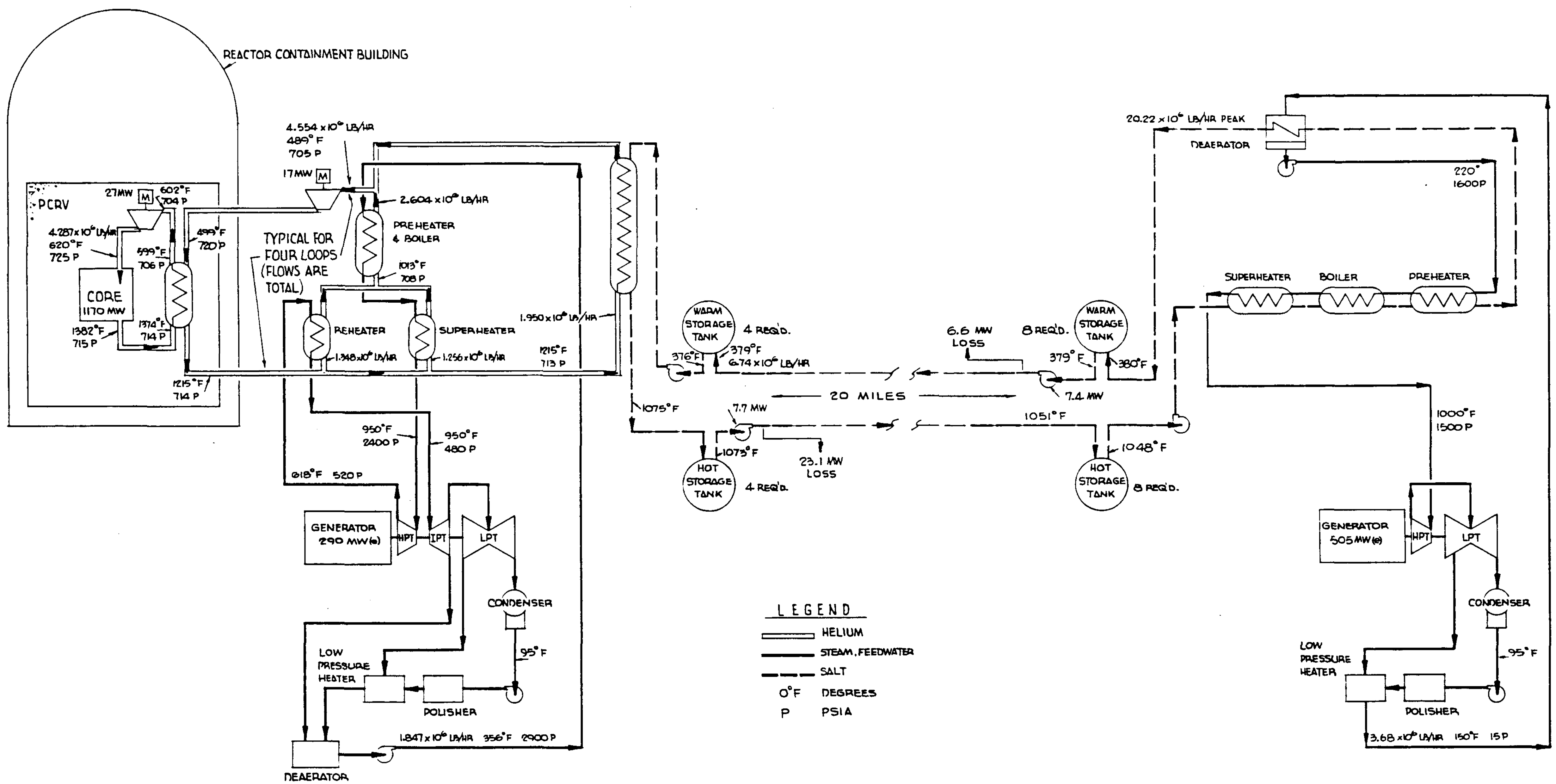


Figure B.2.1-1 HTGR-SALT flow diagram

B.2.2 Plant Arrangement

The following sections provide a brief description of important features in the HTGR-SALT plant arrangement. The information is based upon the results of a brief study made to provide more definition of the HTGR-SALT concept for conceptual cost estimating. Specific areas addressed include the NHS, the secondary helium system, the balance of reactor station, the peaking power station, and the salt transmission system.

B.2.2.1 Nuclear Heat Source

The NHS is contained in a multicavity prestressed concrete pressure vessel (PCRV) with the reactor cavity in the center and four IHXs and their auxiliary core cooling units surrounding it.

The PCRV has the following design parameters:

PCRV diameter	21.6 m (71 ft)
PCRV height	27.7 m (91 ft)
Core cavity diameter	9.6 m (31.5 ft)
Core cavity height (minimum)	18.3 m (60 ft)
Operating pressure	5.0 MPa (725 psi)
Maximum cavity pressure	5.2 MPa (750 psi)
Core outlet temperature	750°C (1382°F)
Number of IHXs	4
IHX cavity diameter	2.7 m (9 ft)
IHX effective length	11.1 m (36.3 ft)
Concrete strength	44.8 MPa (6500 psi)
Number of core auxiliary cooling system (CACS) units	3

The PCRV for the NHS (Fig. B.2.2-1) contains four IHXs and three CACS units. In the top view, the four IHX cavities are on one side of the PCRV and the CACS units are on the opposite side. The secondary inlet and outlet ducts of the IHXs are integral with the PCRV and exit at the bottom. The routing of these lines from the PCRV exit to the containment wall is similar to that shown in UE&C Drawing SK 149.*

The diameter of a multicavity PCRV is determined by the cavity diameters, the maximum cavity pressure, and the duct routing for the required gas flow path. The relatively short IHX permits a horizontal duct from the circulator, which is located in the top section of the IHX cavity, into the slightly extended top plenum of the core cavity. This arrangement minimizes the ligament between the core and IHX cavities and results in a PCRV diameter of 21.6 m (71 ft).

The PCRV is prestressed longitudinally by longitudinal tendons and circumferentially by wire winding. The PCRV cooling water header pits are

*"General Arrangement, RCB HTGR-R," UE&C Drawing SK 149, sheets 1 and 2.

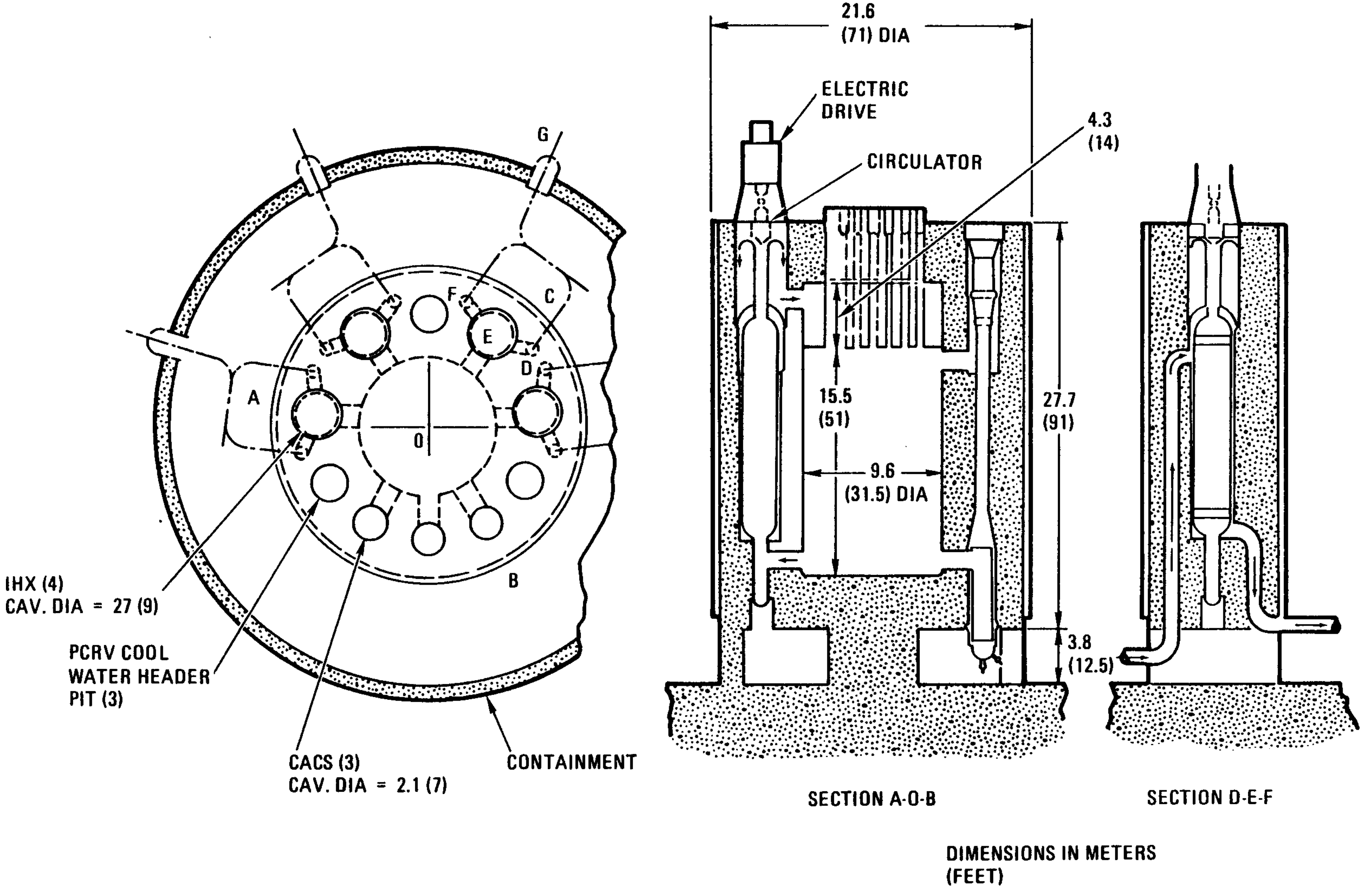


Figure B.2.2-1 HTGR-SALT conceptual 1170-MW(t) layout interspersed between the IHX and CACS cavities, as shown in Fig. B.2.2-1. The 27.7-m (91-ft) PCRV height is determined by the installation of the IHX and circulator in their combined cavity.

The secondary helium piping within the containment determines the diameter of the containment building. An annulus approximately 7.01 m (23 ft) wide between the PCRV and the containment is required. The pipes penetrate the containment wall into the four valve housings.

B.2.2.2 Intermediate Loops

Figure B.2.2-2 shows one of the four intermediate loops, which consists of a steam generator, a helium/steam reheater, a He/salt heat exchanger, a circulator, and the required piping. The helium flow from the IHX through the loop components is shown in the flow diagram (Fig. B.2.1-1).

B.2.2.3 Balance of Reactor Station

Figure B.2.2-3 shows one-half of the complete reactor station. Figure B.2.1-1 shows that the hot salt solution frcm the He/salt heat exchanger of each intermediate loop goes into the hot storage tanks and from there into the transmission line routing the hot salt to the peaking power station. Warm salt returning from the peaking power station is routed via the transmission line into the warm storage tanks and from there back into the He/salt heat exchanger.

The steam turbine generator building for the base-loaded station is located between the hot storage tanks in order to minimize the hot and warm helium pipe lines to and from the steam generation equipment.

B.2.2.4 User Peaking Power Station

In addition to the hot and warm salt storage facility, the user peaking power station (Fig. B.2.2-4) consists of two sets of salt/water heat exchangers, which produce the steam for the steam turbine generator station at this station.

Two parallel trains of heat exchangers are provided for steam generation. Each train contains a preheater, a boiler, and a superheater arranged in series to produce countercurrent salt/water-steam circuitry.

B.2.2.5 Salt Transmission System

The 32.2 km (20 mile) long transmission system for the hot and warm salt mixture is shown schematically in Fig. B.2.2-5. The transmission system includes ten pumping stations each for the hot and warm lines. This limits internal pressures consistent with acceptable long-term stresses in the piping materials at their service temperatures.

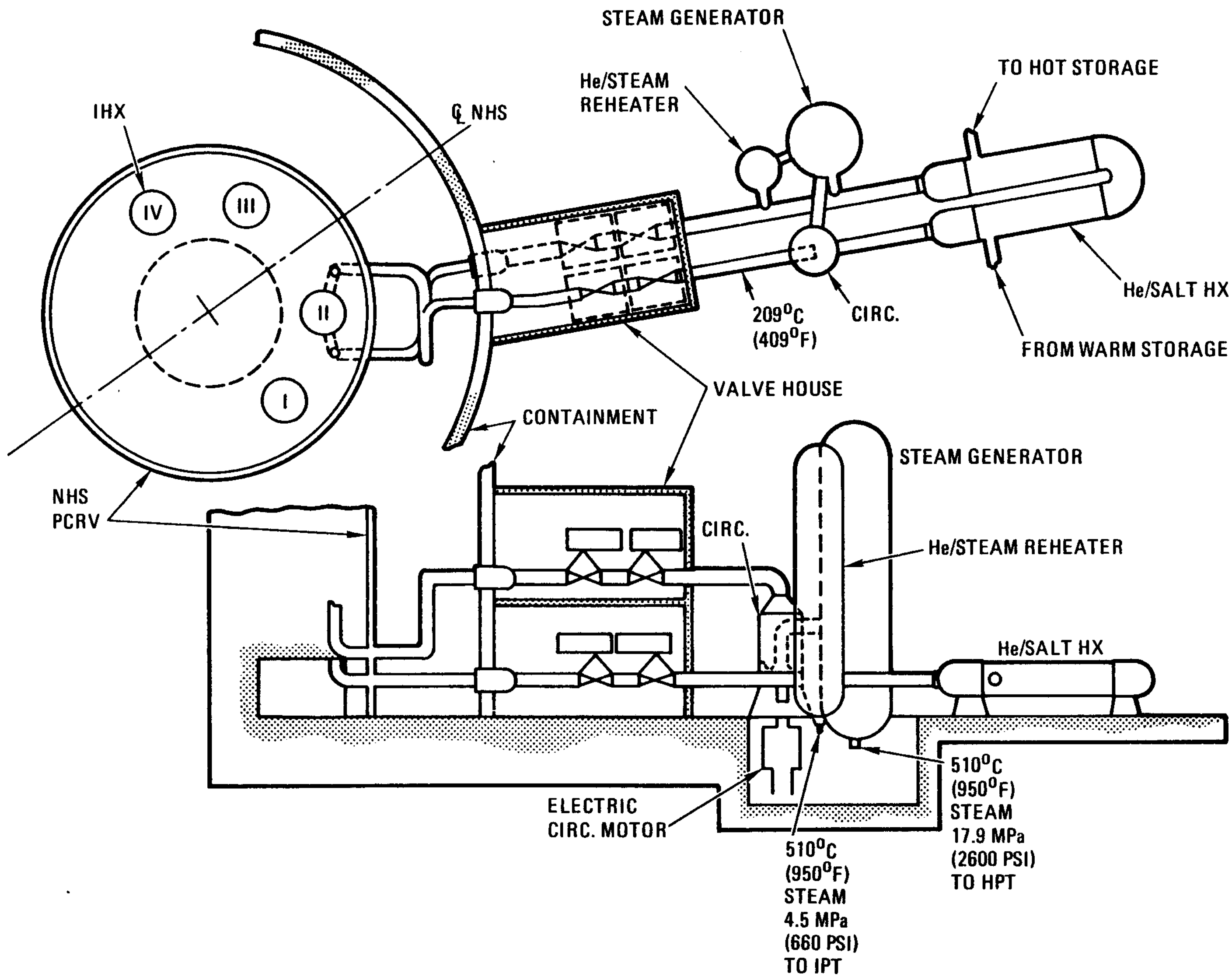


Figure B.2.2-2 1170-MW(t) HTGR-SALT intermediate loop conceptual design

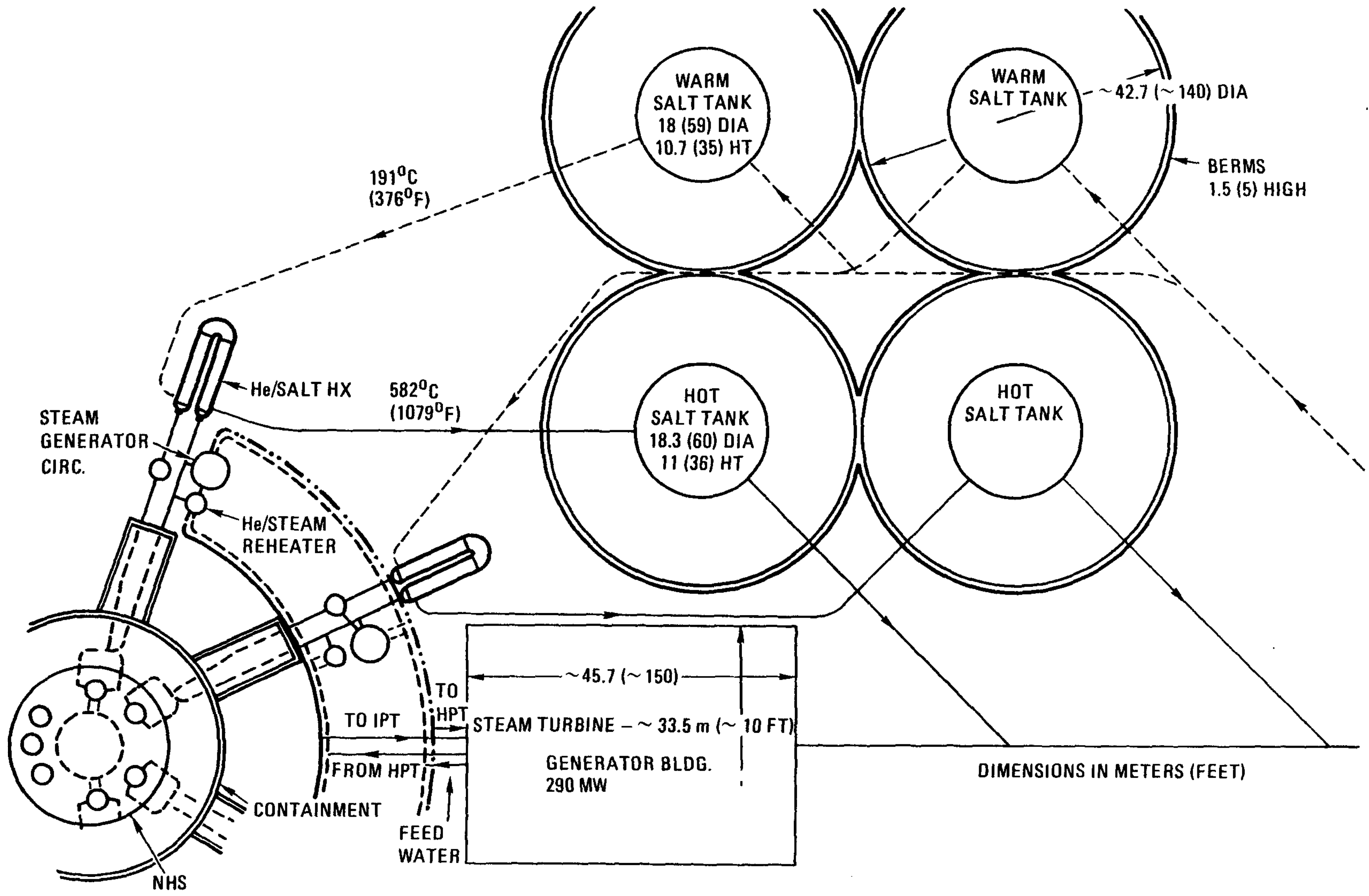


Figure B.2.2-3 Conceptual design, 1170-MW(t) HTGR-SALT, reactor station

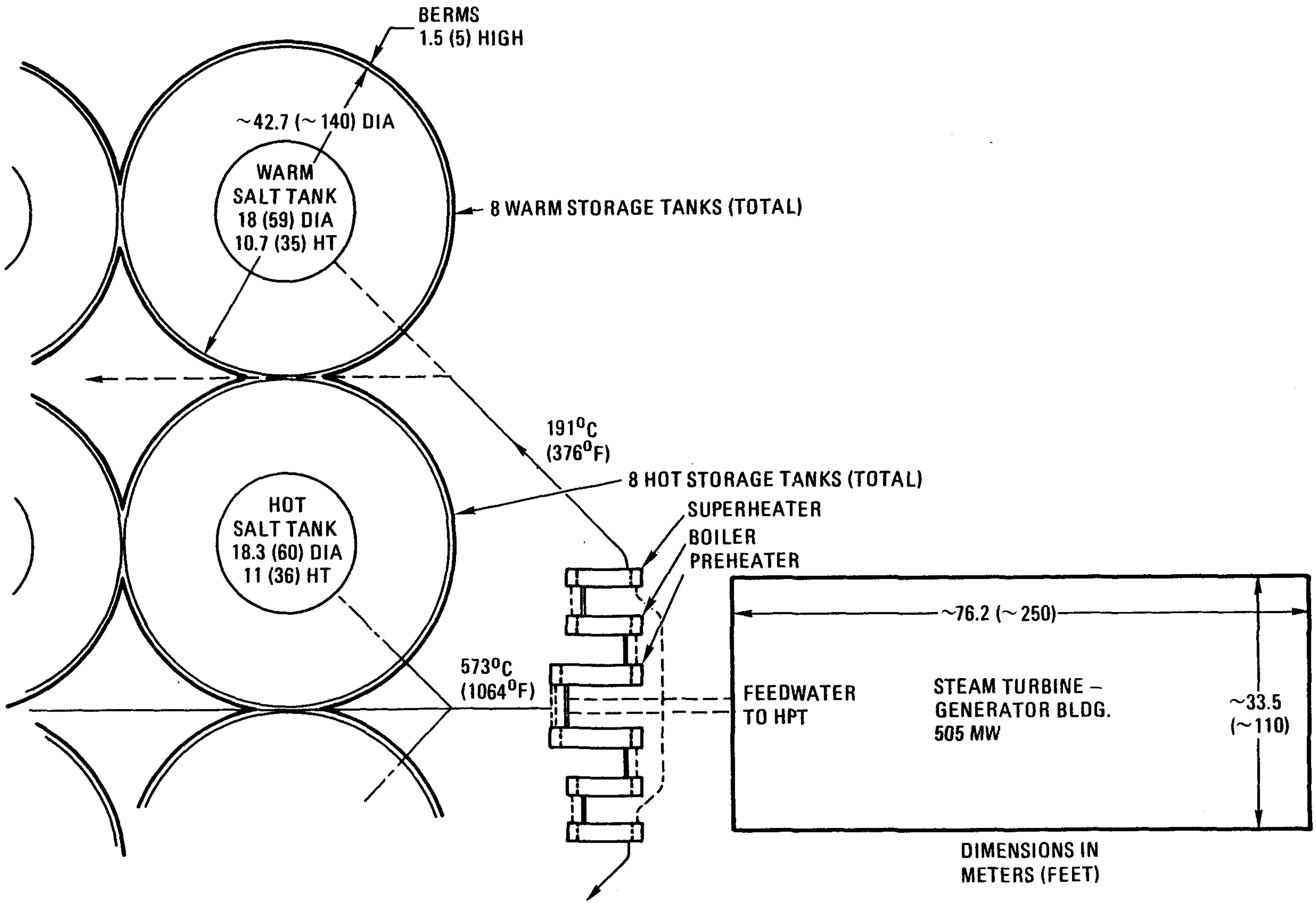


Figure B.2.2-4 Conceptual design, 1170-MW(t) HTGR-SALT, user station

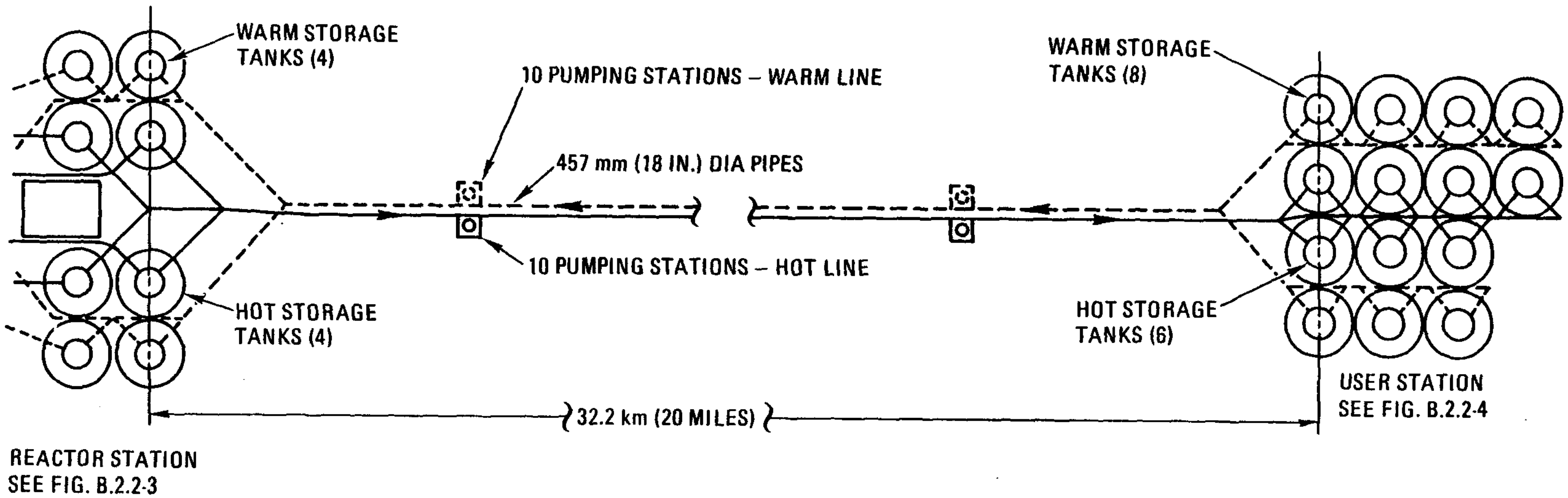


Figure B.2.2-5 Salt transmission system

B.3 System Design Considerations

The specifications of the system, given in Table B.3-1 and Fig. B.2.1-1, were selected to attain the following objectives:

1. Provide a close comparison to an 1170-MW(t) HTGR-R plant utilizing a TCP to transport gas to a methanating (user) station located 32.2 km (20 miles) away. This remote station is a power peaking plant that operates for 8 hours per day.

2. Use design information previously developed for the HTGR, the hot helium lines, the He/salt and salt/steam heat exchangers, and the salt lines and tanks.

3. Use parameters achievable with current HTGR technology.

The mean core outlet temperature of 750°C (1382°F) is the same as the currently operating HTGR at Fort St. Vrain, Colorado. The temperature rise across the core of 405.6°C (762°F) gives the same helium flow rate as for the HTGR-R. The resulting core inlet temperature of 326.7°C (620°F) is satisfactory for the eutectic mixture of HTS.

The remaining primary system parameters were determined by appropriately scaling heat losses and pressure drops and rises from the HTGR-R data. Four primary helium loops are used, consistent with the reference HTGR-R, although a fewer number of loops might be more cost effective. There are four IHXs, coupled to four secondary helium loops. Each primary and secondary loop has its own circulator so that they can operate independently of each other.

The secondary helium parameters selected give approximately a 25% increase in the log mean temperature differential (LMTD) across the IHX relative to the HTGR-R. This permits a smaller heat exchanger and a smaller PCRV. Pressure drops and heat losses in the secondary helium loops are based on a 30.5-m (100-ft) equivalent pipe run and a 762-mm (30-in.) o.d. pipe assuming external insulation of 152.4 mm (6 in.) of calcium silicate. This intermediate piping length is shorter than for the HTGR-R because the He/salt heat exchanger may be placed safely quite close to the reactor.

The He/salt heat exchanger pressure drop (helium side) was scaled from previous values (Ref. 1). The heat exchanger design is also based on these reference data and utilizes one unit. The He/ steam generator is basically the 560-MW(t) Mark IV steam generator scaled up to the 690-MW(t) requirements of this plant.

The transport salt parameters give as large a temperature difference as practical consistent with an acceptable margin above freezing for the warm salt line. They also give an adequate LMTD so that the salt heat exchangers are of acceptable size. The HTS melting point is 142.2°C

TABLE B.3-1
1170-MW(t) HTGR-SALT PARAMETERS

NUCLEAR HEAT SOURCE HEAT BALANCE

Core power, MW(t)	1170.0
Thermal power added by circulators, MW(t)	27.2
Compressor efficiency, %	80.0
Heat losses, MW(t)	9.1
Power to IHX, MW(t)	1188.1

NUCLEAR HEAT SOURCE SYSTEM PARAMETERS

	Pressure or Pressure Drop [MPa (psia)]	Temperature [°C (°F)]	Helium Flow [kg/s (lb/hr x 10^6)]
Core inlet	4.996 (724.6)	326.7 (620.0)	540.2 (4.287)
Core outlet	4.930 (715.1)	750.0 (1382.0)	--
IHX inlet	4.923 (714.1)	745.6 (1374.5)	540.2 (4.287)
IHX outlet	4.864 (705.5)	314.9 (598.8)	540.2 (4.287)
Circulator inlet	4.857 (704.4)	316.7 (602.0)	540.2 (4.287)
Circulator outlet	4.999 (725.0)	326.3 (619.4)	--
Core ΔP	0.06557 (9.51)		
IHX ΔP	0.05916 (8.58)		
NHS loop total ΔP	0.1422 (20.63)		

TABLE B.3-1 (Continued)

INTERMEDIATE LOOP HEAT BALANCE

IHX power, MW(t)	1188.1
Thermal power added by circulators, MW(t)	16.8
Compressor efficiency, %	80.0
Heat losses, MW(t)	
Hot-side piping	0.07
Warm-side piping	0.02
HX shells	0.11
Total	0.2
Heat through He/salt heat exchanger, MW(t)	515.0
Heat through steam generator, MW(t)	
Reheater	99.3
Superheater	92.5
Boiler/preheater	498.0
Total	689.8

INTERMEDIATE LOOP SYSTEM PARAMETERS

	Pressure or Pressure Drop [MPa (psia)]	Temperature [°C (°F)]	Helium Flow [kg/s (lb/hr x 10^6)]
IHX inlet	4.961 (719.6)	259.3 (498.8)	573.8 (4.554)
IHX outlet	4.924 (714.1)	657.5 (1215.5)	573.8 (4.554)
Reheater inlet	4.916 (713.0)	657.5 (1215.5)	169.8 (1.348)
Superheater inlet	4.916 (713.0)	657.5 (1215.5)	158.2 (1.256)
Reheater/superheater outlet	4.882 (708.0)	486.4 (907.5)	328.1 (2.604)
Boiler/preheater inlet	4.882 (708.0)	486.4 (907.5)	328.1 (2.604)
Boiler/preheater outlet	4.871 (705.0)	253.8 (488.8)	328.1 (2.604)
He/salt HX inlet	4.916 (713.0)	657.5 (1215.5)	245.7 (1.950)
He/salt HX outlet	4.861 (705.0)	253.8 (488.8)	245.7 (1.950)
Circulator inlet	4.861 (705.0)	253.8 (488.8)	573.8 (4.554)
Circulator outlet	4.966 (720.2)	259.4 (498.9)	573.8 (4.554)

TABLE B.3-1 (Continued)

SALT LOOP HEAT BALANCE

He/salt heat exchanger power, MW(t)	515.0
Thermal power added by line pumps, MW(t)	12.9
Pump efficiency, %	85.0
Heat losses, MW(t)	
Hot-side piping	23.1
Warm-side piping	6.6
Hot salt storage tanks	4.0
Warm salt storage tanks	0.65
Salt/steam generator shell	0.03
Total	34.4
Heat input through salt/steam generator, MW(t)	493.5 avg (1/3 of 1480.5 peak)

SALT LOOP SYSTEM PARAMETERS

	Pressure or Pressure Drop [MPa (psia)]	Temperature [°C (°F)]	Salt Flow Rate [kg/s (lb/hr x 10^6)]
Helium/salt HX inlet	1.55 (225)	191.1 (376.0)	849 (6.74)
Helium/salt HX outlet	1.38 (200)	579.4 (1075.0)	849 (6.74)
RS[(a)] hot storage inlet	0.103 (15.0)	579.4 (1075.0)	849 (6.74)
RS hot storage outlet	0.103 (15.0)	578.4 (1073.2)	849 (6.74)
US[(b)] hot storage inlet	0.103 (15.0)	575.2 (1051)	849 (6.74)
US hot storage outlet	0.103 (15.0)	573.2 (1048)	2548 (20.22)
Salt/steam generator inlet	1.55 (225)	573.2 (1048)	2548 (20.22)
Salt/steam generator outlet	1.38 (200)	191.1 (380)	2548 (20.22)
US warm storage inlet	0.103 (15.0)	191.1 (380)	2548 (20.22)
US warm storage outlet	0.103 (15.0)	190.4 (379)	849 (6.74)
RS warm storage inlet	0.103 (15.0)	192.6 (378.6)	849 (6.74)
RS warm storage outlet	0.103 (15.0)	191.1 (376.0)	849 (6.74)

(a) RS = reactor station.
(b) US = user station.

TABLE B.3-1 (Continued)

HELIUM-TO-STEAM LOOP HEAT BALANCE

Heat through steam generator, MW(t)	689.7
Net power (after generator auxiliaries), MW(e) (Net generator efficiency of 42%)	289.7
Heat rejected plus generator auxiliaries, MW	400.0

HELIUM-TO-STEAM LOOP PARAMETERS

	Pressure or Pressure Drop [MPa (psia)]	Temperature [°C (°F)]	Water/Steam Flow [kg/s (lb/hr x 10^6)]
Preheater inlet	19.99 (2900)	180 (356)	232.7 (1.847)
Preheater outlet	19.72 (2860)	--	232.7 (1.847)
Boiler inlet	19.65 (2850)	--	232.7 (1.847)
Boiler outlet	18.41 (2670)	358 (677)	232.7 (1.847)
Superheater inlet	18.34 (2660)	358 (677)	232.7 (1.847)
Superheater outlet	16.55 (2400)	510 (950)	232.7 (1.847)
HP turbine inlet	16.48 (2390)	510 (950)	232.7 (1.847)
HP turbine outlet	3.59 (520)	326 (618)	232.7 (1.847)
Reheater inlet	3.52 (510)	326 (618)	232.7 (1.847)
Reheater outlet	3.31 (480)	510 (950)	232.7 (1.847)
IP turbine inlet	3.24 (470)	510 (950)	232.7 (1.847)
IP turbine outlet	--	--	--
LP turbine inlet	--	--	--
LP turbine outlet	0.00138 (0.2)	37.8 (100)	--
Condenser inlet	0.00103 (0.15)	35 (95)	--
Condenser outlet	0.689 (100)	35 (95)	--

TABLE B.3-1 (Continued)

SALT-TO-STEAM HEAT BALANCE

Heat through steam generator, MW(t)	1480.5
Net power (after generator auxiliaries), MW(e) (Net generator efficiency of 34.2%)	506
Heat rejected plus generator auxiliaries, MW	974.5

SALT-TO-STEAM LOOP PARAMETERS

	Pressure or Pressure Drop [MPa (psia)]	Temperature [°C ([F)]	Water/Steam Flow [kg/s (lb/hr x 10^6)]
Preheater inlet	11.03 (1600)	65.6 (150)	463.7 (3.68)
Preheater outlet	10.69 (1550)	316.1 (601)	463.7 (3.68)
Boiler inlet	10.69 (1550)	316.1 (601)	463.7 (3.68)
Boiler outlet	10.62 (1540)	315.6 (600)	463.7 (3.68)
Superheater inlet	10.55 (1530)	315.6 (600)	463.7 (3.68)
Superheater outlet	10.34 (1500)	538 (1000)	463.7 (3.68)
HP turbine inlet	10.20 (1480)	538 (1000)	463.7 (3.68)
LP turbine outlet	0.001380 (0.2)	37.8 (100)	463.7 (3.68)
Condenser inlet	0.001034 (0.15)	35 (95)	463.7 (3.68)
Condenser outlet	0.6895 (100)	35 (95)	463.7 (3.68)

(288°F), and the 190°C (374°F) warm return temperature offers a suitable margin. The hot salt temperature of 579.4°C (1075°F) provides an approach temperature of 77.8°C (140°F), which is adequately large; further, the 579.4°C (1075°F) value is close to the long-term thermal stability limits of the salt. The temperature difference of 388.3°C (699°F) gives an acceptable salt flow rate, pipe diameter, pumping power, and salt pressure.

The He/salt heat exchangers give an output at the reactor station integrated over a 24-hour period equivalent to that of the thermochemical heat pipe, namely 12,360 MW hours or 515 MW(t) continuously. After summing the heat losses and power added through the circulators, a net 690 MW(t) passes through the helium to the steam generators. This heat duty and the available temperature permit using the steam turbine power conversion system developed for the HTGR-R plant. This single reheat cycle generates 290 MW(e) output at the generator terminals and provides the base power supply.

The heat losses and pressure drops in the salt piping are based on a 32.2-km (20-mile) length plus 40% allowance for bends and expansion loops. Piping 457.2 mm (18 in.) in diameter is used with 152.4 mm (6 in.) of calcium silicate insulation on the hot pipe and 76.2 mm (3 in.) on the warm pipe. In order to use thin-walled piping, it is necessary to limit pressure levels in the piping to less than 1.72 MPa (250 psi). Multiple pumping stations are therefore employed. Approximately 10 hot and 10 warm pumps are located at the same stations so they use the same right-of-way and common pump station maintenance equipment.

The thermal storage system design is discussed in Section B.4.3. The storage tanks used in this system are identical to those described in Ref. 2. The thermal losses from these tanks are based on a 152.4 mm (6 in.) thick layer of calcium silicate insulation.

The total salt/steam generator rating (for two trains of heat exchangers) is 1480.5 MW(t). These units operate as a power peaking unit for 8 hours out of each 24 hours. Because of this cycling power requirement, each steam generator was configured into three separate parts: a preheater, a drum-type boiler, and a separate superheater. There is no reheater because it is impractical to fit a reheat cycle to the selected hot and warm salt conditions. Only modest feedwater heating is possible. Therefore, the power conversion efficiency of the associated steam turbine train is only 34.2%, substantially less than the base power conversion plant efficiency of 42%. A salt-heated deaerator is used since steam heating is not practical.

In retrospect, the plant cost effectiveness might be improved if a higher warm salt temperature were used, since this would permit a reheat cycle with feed heating. However, the practicality of using reheat turbines for cyclic duty needs examination. The system parameters in Table B.3-1 appear reasonable and adequate for the objectives given previously.

B.4 Component Design

Preliminary studies of the important components in the HTGR-SALT plant were made to identify basic concepts and to support the plant cost estimate. The results of this work follow.

B.4.1 Primary System Components

The NHS primary system in the HTGR-SALT plant contains the IHX, primary helium circulator, reactor, and interconnecting ducting. The technical approach, general arrangement, and physical dimensions of this equipment were established by scaling a conceptually similar HTGR-R plant design. The primary system in the HTGR-SALT plant operates at considerably lower temperatures than the HTGR-R NHS, which permits the use of less expensive, more conventional materials in the hot zones.

The important influence of the IHX on plant performance and economics requires analysis in greater depth, as discussed below.

The IHX is conceptually similar to that for the HTGR-R. Figure B.4.1-1 shows a schematic arrangement of the IHX. The IHX is a straight tubular gas-to-gas counterflow heat exchanger, which transfers heat from the primary to the secondary helium loop.

The heat transfer bundle tubes are welded at each end to a tubesheet assembly, which comprises a tubesheet and spherical head. A circular shroud welded to one of the tubesheet assemblies encloses the bundle and is perforated at the top and bottom for radial secondary gas flow. The tubes are supported laterally by horizontal low pressure drop "egg crate" type grids, which transfer tube loads into the shroud.

The IHX is located entirely in the PCRV and is welded at the lower end to a liner extension support. The upper end of the unit is attached to a primary/secondary gas boundary dome via a bellows/seal assembly, which compensates for IHX axial thermal expansion. A secondary gas bypass seal is located in the annulus between the IHX and the cavity liner. Primary gas flow restrictors are provided at each end of the unit to guard against the unlikely simultaneous failure of the tubesheet/head weld and the secondary piping outside of the PCRV.

Primary helium for the core enters the IHX at the bottom, flows upward through the tubes, and exits at the top of the circulator located in the same cavity, where it is compressed and returned to the core. The secondary helium enters the IHX cavity at the top, flows radially through the shroud perforations to the top of the bundle, turns 90°, and flows downward over the outside surfaces of the tubes in counterflow to the primary gas. The helium exits the bundle radially through the lower shroud perforations and carries heat to the external salt loops.

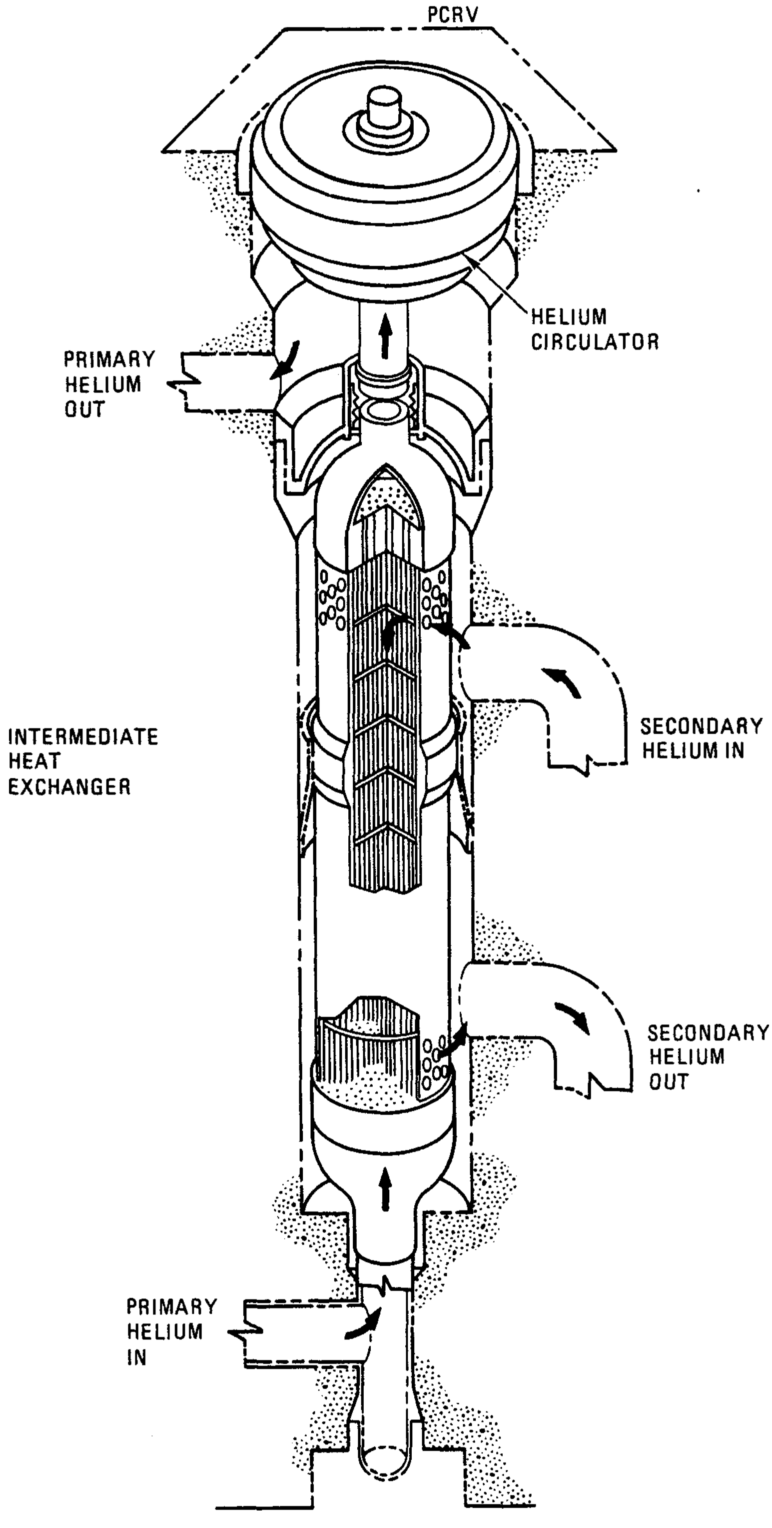


Figure B.4.1-1 Intermediate heat exchanger

The physical size of this IHX is nearly the same as its HTGR-R counterpart. Table B.4.1-1 summarizes the pertinent statistics. Bundle details were designed using the GA library code NUSIZE and by design considerations for minimum tube diameter and wall thickness. The design shown in the Fig. B.4.1-2 parametric survey is governed by (1) minimum practical tube outside diameter, (2) minimum practical tube pitching, and (3) maximum allowable primary-side pressure loss.

Due to lower primary temperatures in the HTGR-SALT NHS, this IHX is all Incoloy 800H, which is less expensive than Inconel 617. Stainless steel (SS) may also be acceptable, but confirmation of this would require more detailed study.

B.4.2 Secondary System Components

The secondary helium system components include the steam generator and reheater equipment, the He/salt heat exchanger, and the secondary helium circulator. The design of the steam equipment and circulator is based on HTGR-SC technology with helium pressure containment provided by conventional vessel techniques. The He/salt heat exchanger is a custom design based on conventional industrial heat exchanger practice. These component studies were done to the level required for the HTGR-SALT conceptual cost estimate. This equipment is described in more detail in the following sections.

B.4.2.1 Steam Generator

The He/steam generator in the secondary loop is the same general type as that proposed for the HTGR-SC and NHSDR plants. It consists of a helical coil economizer, evaporator, and one stage of superheat (EES) with a finishing straight tube superheater (STSH). A bimetallic weld (BMW) located in a quiescent zone connects the Alloy 800 STSH to the 2-1/4Cr - 1Mo EES.

Water enters the EES below the coil at the feedwater tubesheet and is heated as it rises to the top of the coil. It next flows through the expansion loops (and BMW) to the center of the unit and then flows downward through the STSH to the superheat tubesheet at the bottom of the unit. One-half of the helium enters at the bottom end, enters the STSH section, and flows upward to the top, where it turns 180° to enter the EES section. At this point it is joined by the remaining one-half helium mass flow (the exhaust from the reheater). The total mass flow then passes downward over the coil to the bottom, where it exits the unit. The design is identical to the 560-MW(t) Mark IVA steam generator shown in Fig. B.4.2-1 except for the following:

1. An additional helium inlet nozzle is added at the top of the unit, and some expansion loops are re-routed.

2. The EES outer flow shroud thickness is increased and becomes the pressure-retaining shell.

TABLE B.4.1-1
HTGR-SALT IHX PRELIMINARY DATA

Fluid Flow Parameters		
Circuit	Primary Helium	Secondary Helium
Flow (plant), kg/s (lb/hr)	540 (4.287×10^6)	573.9 (4.554×10^6)
Inlet temperature, °C (°F)	745 (1374)	259.4 (499)
Outlet temperature, °C (°F)	315 (599)	665 (1229)
Inlet pressure, MPa (psia)	4.92 (714)	4.96 (720)
Pressure loss, MPa (psid)	0.055 (8)	0.041 (6)
Effectiveness	0.886	0.818
LMTD, °C (°F)	67.2 (121)	67.2 (121)
Heat duty, MW per plant	1209	1209

Construction Details	
Number of units per plant	4
Type of construction/flow arrangement	Tubular/counterflow
Fluid routing: primary/secondary	Tube-side/shell-side
Tube bundle arrangement	Monolithic
Number of tubes per IHX	1600
Tube o.d. x wall, mm (in.)	12.7 x 1.02 (0.5 x 0.040)
Effective tube length, m (ft)	11.1 (36.3)
Approximate HX o.d. (inside thermal barrier), m (ft)	2.67 (8.75)
Material	Incoloy 800H
Approximate IHX weight, tonne (tons)/HX	141 (156)
ASME code classification	Section III, Class 1
Similar unit reference	025785
Active heat transfer surface area, m^2 (ft^2)/plant	28,253 (304,106)

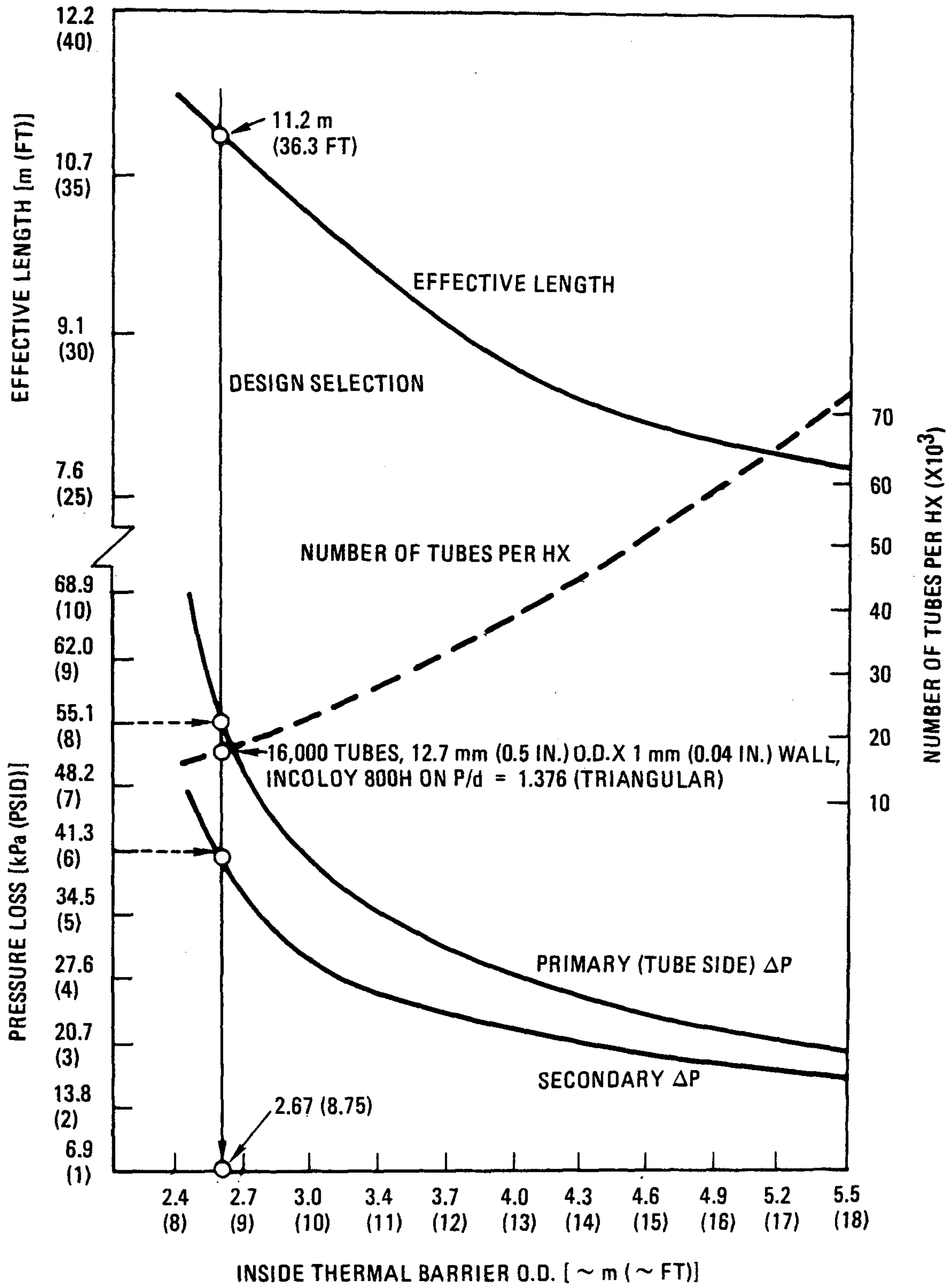
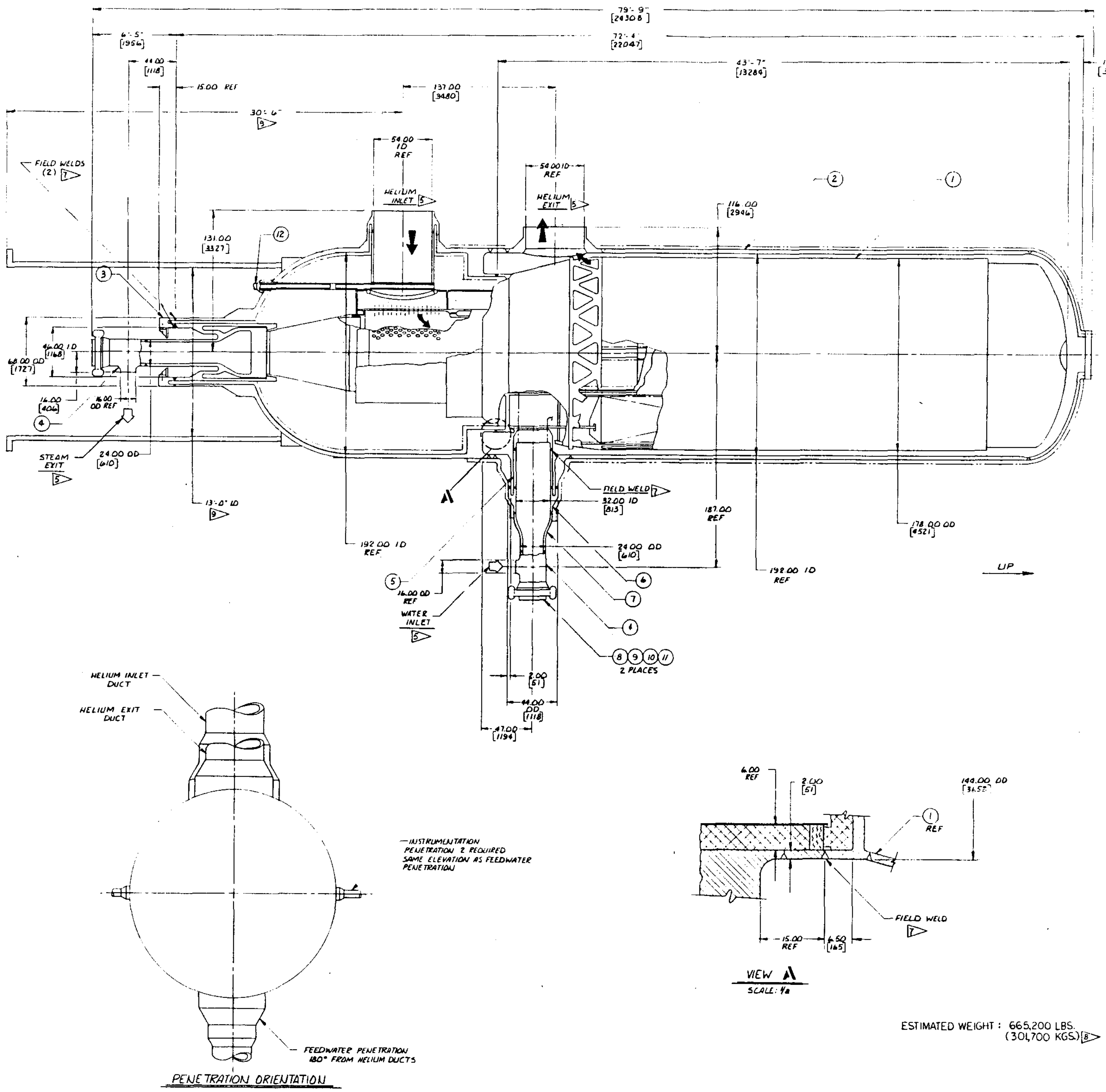


Figure B.4.1-2 IHX thermal sizing study: counterflow design, monolithic bundle, NUSIZE results



NOTES:

1. ALL DIMENSIONS ARE IN THE NOMINAL COLD CONDITION
2. DIMENSIONS IN [ ] BRACKETS ARE IN MILLIMETERS
3. ADDITIONAL MAN ACCESS PENETRATIONS LOCATED ON HELIUM INLET AND EXIT DUCTS
4. STEAM GENERATOR TYPE:
   MARK IV-A TYPE, 590 MW(t), COUNTER FLOW, HELICAL COIL BUNDLE, STRAIGHT SUPERHEAT.
5. DESIGN CONDITIONS:
   NUMBER OF TUBES: 462
   TUBE SIZE: EES: .866 OD x .13/.114 WALL, STSH: 1.26 OD x .173 WALL
   PRIMARY COOLANT: HELIUM (SHELL SIDE)
   INLET TEMP: 1266°F (686°C)
   OUTLET TEMP: 601°F (316°C)
   PRESSURE: 1050 PSIA
   SECONDARY COOLANT: WATER (TUBE SIDE)
   INLET TEMP: 430°F (221°C)
   OUTLET TEMP: 1005°F (541°C)
   PRESSURE: 2515 PSIA
6. STEAM SIDE PRESSURE BOUNDARY (TUBE BUNDLE, TUBESHEET, ETC.) SHALL BE DESIGNED, FABRICATED, TESTED AND STAMPED IN ACCORDANCE WITH THE ASME BOILER AND PRESSURE VESSEL CODE SECTION III, CLASS 1, AND CODE CASE N-47.
   THE HELIUM PRESSURE BOUNDARY (SHELL, ETC.) SHALL BE DESIGNED AND FABRICATED IN ACCORDANCE WITH SECTION VIII, DIVISION 1.
7. FINAL FIT-UP WELD OF HEAT EXCHANGER TO VESSEL.
8. ESTIMATED WEIGHT DOES NOT INCLUDE VESSEL WEIGHT.
9. DIMENSIONS TO BE CONFIRMED BY LINERS BRANCH.

	ITEM	PART NO	DESCRIPTION	MATL	MATL SPEC
LOT	12		INSTRUMENTATION		
2	11		GRAYLOC CLAMP - 24"	CARBON STL	
2	10		GRAYLOC SEAL RING - 24"		
2	9		GRAYLOC BLIND HUB - 24"		
2	8		GRAYLOC BUTT WELD HUB - 24"		
1	7		REDUCER - 36" x 24" x 24" LG		SA-106
1	6		FORGING, FW 44" OD x 32" ID x 24" LG		SA-508 CL 1
1	5		PIPE, FW 32" ID x 36" LG x 2" THK		SA-106
2	4		TEE, REDUCING 24" x 24" x 16"	CARBON STL	SA-106
1	3	000214-001	TORUS		SA-508 CL.1
1	2	026231	VESSEL ASSEMBLY		
1	1	000838	MK IV-A STEAM GENERATOR ASSY		
			ASSEMBLY		

ESTIMATED WEIGHT: 665,200 LBS. (301,700 KGS.) 8

Figure B.4.2-1 Steam generator general arrangement

3. The unit is increased approximately 23% in capacity, and therefore size, from 560 MW(t) to 690 MW(t).

B.4.2.2 Reheater

The reheater design (Fig. B.4.2-2) is straight tube counterflow. One-half of the helium mass flow enters the shell side of the unit at the bottom, penetrates the bundle radially, and flows parallel to the tubes upward to the top of the unit, where it exits to flow over to the top above the upper tubesheet. It then flows down inside the tubes to the bottom tubesheet, where it leaves the unit.

An expansion bellows is provided at the cold upper end between the shell (at a location of reduced diameter) and the tube bundle to accommodate the differential expansion. For simplicity and to reduce cost, the pressure-retaining shell is also a flow shroud (except at the hot bottom end where the shell must be insulated from the gas). An assessment of tolerance accumulations in this area was beyond the scope of this preliminary study.

The unit is supported at the bottom with a skirt attached to the bottom dome (not shown). Differential expansion in this area where the reheater helium outlet connects with the steam generator was not studied; however, this is a typical piping flexibility problem involving large-diameter piping.

Since the temperature at the hot end is marginal for ferritic alloys, the tubing, hot tubesheet, and most of the shell are 316 SS. The cold tubesheet and parts of the shell upper end are carbon steel.

B.4.2.3 He/HTS Heat Exchanger

The He/HTS heat exchanger shown in Fig. B.4.2-3 is a U-tube or "hair pin" type heat exchanger with the helium flowing through the tubes. It is a cross-counterflow arrangement with the helium making a single pass through the tubes and the salt making a single counterflow pass on the shell side. The shell-side pass is broken into 20 cross-flow paths across the tubes.

The HTS is a high-density fluid and thus requires rather small flow passages to promote good heat transfer. Therefore, cross-flow is used to give adequate flow velocity and Reynolds number, in spite of a low limit on the tube spacing and number of tubes. The selected distance between the cross-flow baffles gives the desired shell-side flow area and velocity.

The U-tube or "hair pin" arrangement minimizes tube thermal expansion problems. The crossover section of the U-tube bundle is long enough to give sufficient tube flexibility to cope with hot and cold leg differential expansion.

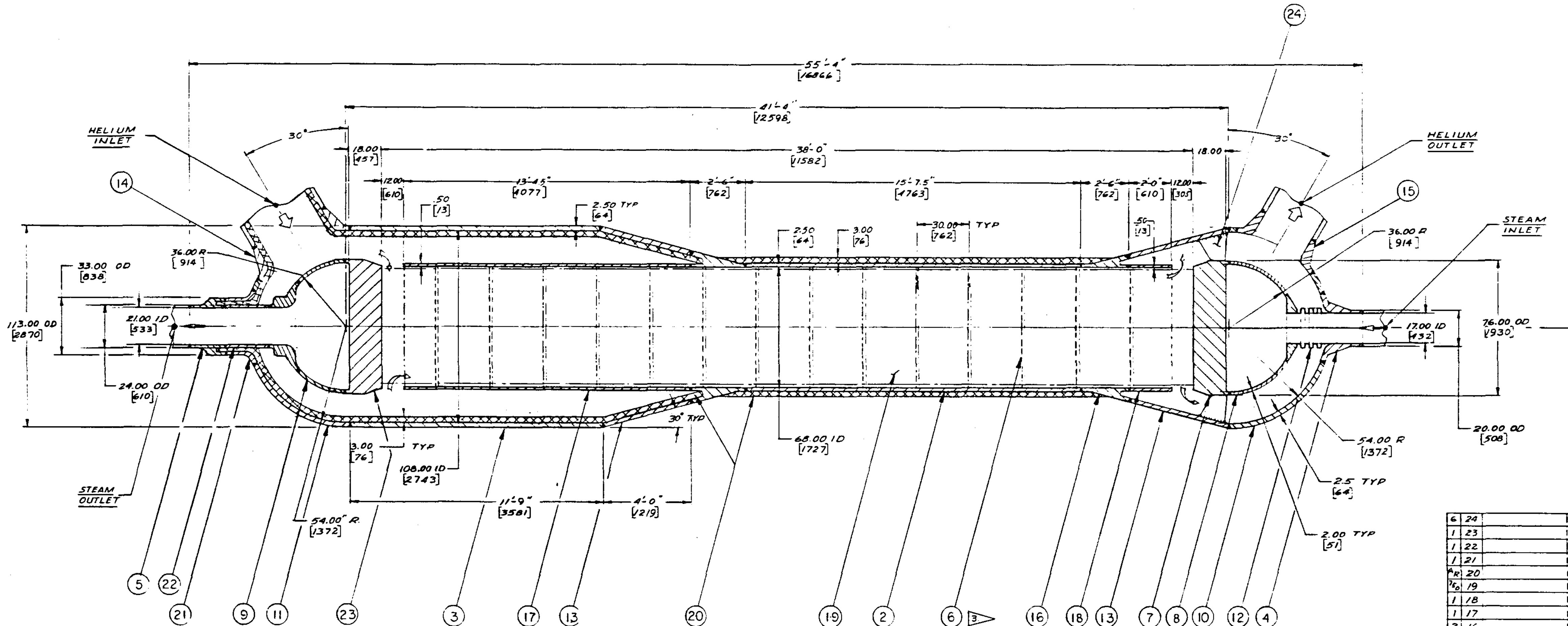
NOTES: UNLESS OTHERWISE SPECIFIED

1. ALL DIMENSIONS ARE NOMINAL COLD CONDITION. DIMENSIONS IN [ ] BRACKETS ARE IN MILLIMETERS.

2. DESIGN CONDITIONS:

SECONDARY HELIUM	INLET TEMP	1215.5 °F
	EXIT TEMP.	917.5 °F
	PRESSURE	713 PSI
STEAM	INLET TEMP.	356 °F
	OUTLET TEMP.	950 °F
	PRESSURE	2900 PSI @ FW TUBESHEET 2400 PSI @ SH TUBESHEET

3. ITEM 6 SIMILAR TO C-E CONCEPT WITH 40% BLOCKAGE.



QTY	ITEM	PART NO	DESCRIPTION	MATL	MATL SPEC
6	24		GUSSETS .75 THK	2¼Cr 1 Mo	SA-387Gr22CL1
1	23		TUBE SHEET 76"Ø×18" THK (HOT)	ALLOY 800H	SB-564
1	22		PIPE 24"OD×1.5WALL×2'Lg	ALLOY 800H	SB-407Gr1
1	21		FLANGE 33"OD	ALLOY 800H	SB-564
AR	20		INSULATION KAOWOOL		
7560	19		TUBES 1.25Ø .131 WALL × 38'Lg.	ALLOY 800H	SB-163Gr1
1	18		UPPER SHROUD 69"Ø×½WALL	2¼Cr 1 Mo	SA-387Gr22CL1
1	17		LOWER SHROUD 69"Ø×½WALL	ALLOY 800H	SB-409Gr1
2	16		Y TORUS-FORGING	2¼Cr 1 Mo	SA-336 F22a
1	15		He OUTLET NOZZLE 36"OD	2¼Cr 1 Mo	SA-336 F22a
1	14		He INLET NOZZLE 48"OD	ALLOY 800H	SB-564
2	13		CONE	2¼Cr 1 Mo	SA-387Gr22CL1
1	12		BELLOWS	ALLOY 800H	SB-407Gr1
1	11		BOTTOM DOME	2¼Cr 1 Mo	SA-387Gr22CL1
1	10		TOP DOME	2¼Cr 1 Mo	SA-387Gr22CL1
1	9		LOWER HEAD	ALLOY 800H	SB-564
1	8		UPPER HEAD	2¼Cr 1 Mo	SA-387Gr22CL1
1	7		TUBE SHEET 76"Ø×18" THK	2¼Cr 1 Mo	SA-336 F22a
14	6		GRID 1.75 DEEP × .125 THK. WEB	ALLOY 800H	SB-408Gr1
1	5		OUTLET NOZZLE 24"OD	ALLOY 800H	SB-564
1	4		INLET NOZZLE 20"OD	2¼Cr 1 Mo	SA-336 F22a
1	3		SHELL 113"Ø×2½"WALL×11'-9"Lg.	2¼Cr 1 Mo	SA-387Gr22CL1
1	2		SHELL 72"Ø×2"WALL×15'-7.5"Lg	ALLOY 800H	SB-409Gr1
	1		ASSEMBLY		

Figure B.4.2-2 He/steam reheater

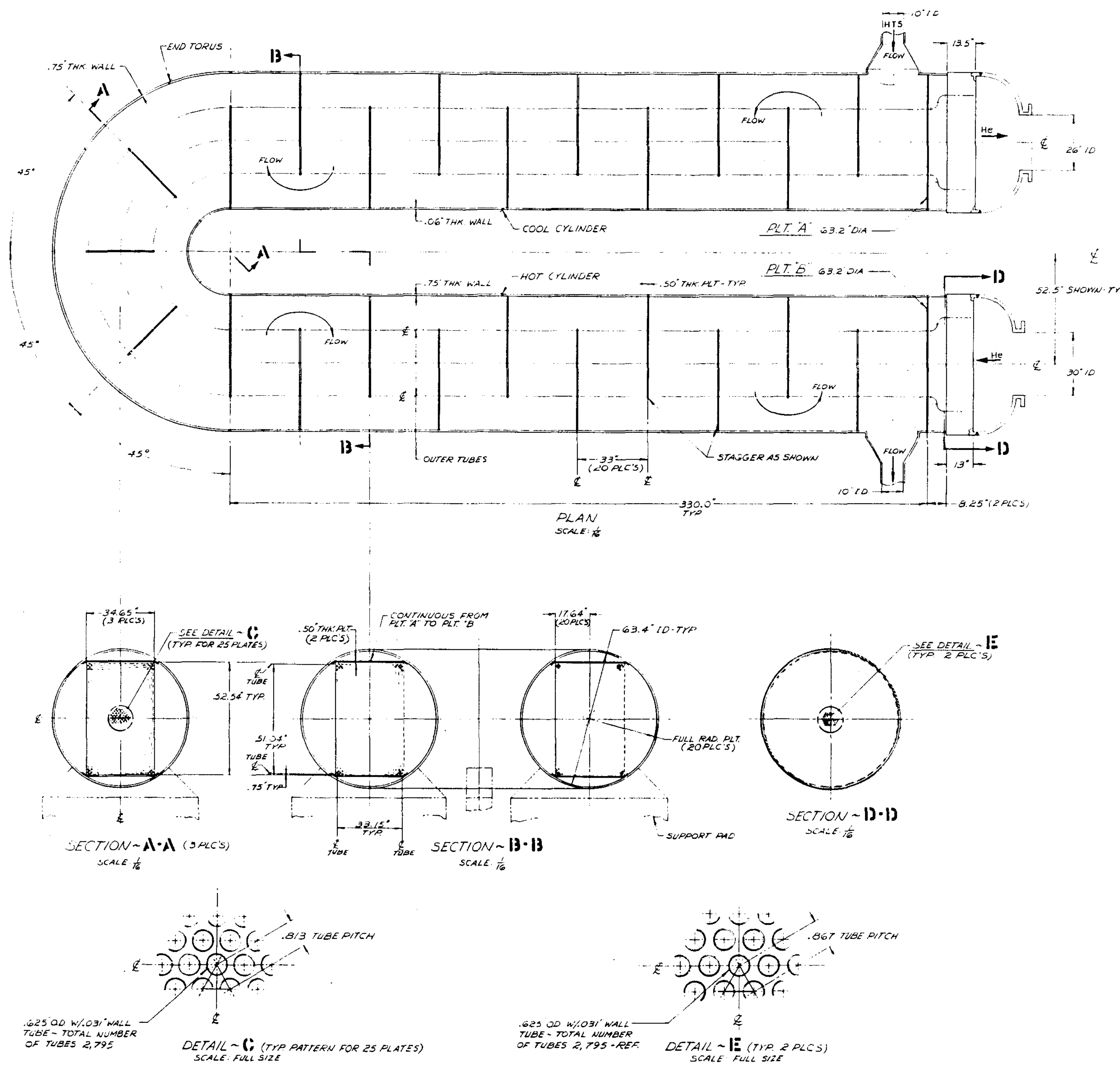


Figure B.4.2-3 He/HTS heat exchanger

Tubesheets are used at each end of the tubes. The tubes are arranged in a triangularly pitched, rectangular cross-section bundle pattern in the bulk of the tube bundle. They are, however, flared out into a wide circular cross-section bundle pattern at the tubesheets. This provides a higher tube plate ligament efficiency and minimizes the thickness of the tube plates.

Individual cylindrical pressure shells are used for each of the two legs. They are connected by a half toroidal section at the "bottom" of the "U". The cylinders have an i.d. of 1.6 m (63.6 in.), which just circumscribes the rectangular tube bundle. Longitudinal baffles run the length of the tube bundle to prevent the shell-side flow from bypassing the bundle. The cross-flow baffles cause the HTS to sweep the bundle 20 times in the two cylindrical sections. The toroidal section is not included as heat transfer area since the baffles do not restrict the flow to a cross-flow pattern. In this section the baffles only restrain the tubes so that the bundle is free to move with thermal expansion.

The heat exchanger is predominantly Incoloy 800H and type 316 SS. Carbon steel could be used instead of 316 SS in the cooler regions [below 489°C (840°F)], except for possible corrosion and BMW problems. Further study may show that less expensive materials (Cr-Mo or carbon steels) can be used in the cooler sections.

Table B.4.2-1 shows the thermodynamic and mechanical features of the He/salt heat exchanger.

B.4.2.4 Secondary Helium Circulator

The secondary helium circulator in the HTGR-SALT plant is a motor-driven multistage centrifugal compressor using standard commercial components where possible. For costing, this unit has been scaled from its counterpart in the HTGR-R plant.

B.4.2.5 Secondary Helium Ducting

The HTGR-SALT plant secondary helium ducting requirements were studied to determine the major physical requirements that influence the system cost. These parameters included pipe outside diameter, wall thickness, expansion capability, and pipe material. The requirements are based on the specified HTGR-SALT plant secondary helium flow rate, the IHX inlet and discharge temperatures and presssures, a straight line separation distance of 21.3 m (70 ft), and specified total pressure losses of 0.0076 MPa (1.1 psi) for the hot leg and 0.0041 MPa (0.6 psi) for the return leg.

The low pressure loss requirements impose rather severe design limitations on the pipe if conventional expansion loops are used. With a "Z" expansion loop having two 90° long radius elbows in the hot line, the

TABLE B.4.2-1
He/SALT HEAT EXCHANGER
(VALUES SHOWN FOR ONE UNIT)

Type	Hair pin tube - Cross-counterflow He in tubes
Helium flow rate, kg/s (lb/hr)	61.612 (4.875×10^5)
Helium inlet temperature, °C (°F)	657 (1215.3)
Helium exit temperature, °C (°F)	254 (488.8)
Helium inlet pressure, MPa (psia)	4.92 (713)
Helium exit pressure, MPa (psia)	4.86 (705)
HTS flow rate, kg/s (lb/hr)	212.3 (1.658×10^6)
HTS inlet temperature, °C (°F)	191 (376)
HTS exit temperature, °C (°F)	579 (1075)
HTS inlet pressure, MPa (psia)	1.55 (225)
HTS exit pressure, MPa (psia)	1.38 (200)
LMTD, °C (°F)	70 (126.05)
Effectiveness	0.866
Overall heat transfer coefficient (U_o), W/m^2-K ($Btu/hr\text{-}ft^2\text{-}°F$)	851.7 (150)
Number of tubes	2795
Tube o.d., mm (in.)	15.88 (0.625)
Tube i.d., mm (in.)	14.3 (0.563)
Tube pitch (in bundle), mm (in.)	20.78 (0.813)
Bundle geometry	Triangular pitched rectangular array
Number of tubes per row	65
Number of rows per shell-side pass	43
Distance between baffles, m (ft)	0.838 (2.75)
Minimum shell-side flow area in row, m^2 (ft^2)	0.26 (2.78)
Shell-side frontal area, m^2 (ft^2)	1.12 (12.1)
Turning flow area (in plane of baffle), m^2 (ft^2)	0.27 (2.9)
Number of cross-flow passes	20
Tube-side flow area, m^2 (ft^2)	0.45 (4.83)
Active tube length, m (ft)	16.76 (55)
Actual average tube length, m (ft)	21.5 (70.5)
Actual surface area, m^2 (ft^2)	2336 (25153)
Tubesheet tube pitch, mm (in.)	22.02 (0.867)
Ligament efficiency	0.242
Tube pattern maximum radius, m (in.)	0.67 (26.2)
Tubesheet diameter, m (in.)	1.54 (60.5)
Tubesheet thickness (cold end), mm (in.)	342.9 (13.5)
Tubesheet thickness (hot end), mm (in.)	330.2 (13.0)
Pressure shell thickness (cold cylinder), mm (in.)	15.24 (0.6)
Pressure shell thickness (hot cylinder), mm (in.)	19.05 (0.75)
Pressure shell thickness (torus), mm (in.)	19.05 (0.75)
Helium inlet dome thickness, mm (in.)	38.1 (1.5)
Helium exit dome thickness, mm (in.)	38.1 (1.5)

TABLE B.4.2-1 (Continued)

Helium inlet nozzle i.d., m (in.)	0.762 (30)
Helium exit nozzle i.d., m (in.)	0.66 (26)
HTS inlet nozzle i.d., m (in.)	0.23 (9)
HTS exit nozzle i.d., m (in.)	0.23 (9)
Shell-side baffle thickness, mm (in.)	19.05 (0.75)
Materials of construction:	
Tubes	Incoloy 800H
Cold tube sheet	304 SS
Hot tube sheet	Incoloy 625
Pressure shell	316 SS
Cold helium dome	Carbon steel
Hot helium dome	Incoloy 800H
Shell-side baffles	316 SS

loss due to turning in the elbows is approximately 75% of the total. The center portion of the "Z" loop is about 12.2 m (40 ft) long to cope with the 254 mm (10 in.) expansion in the return line pipe. This section adds approximately 7.6 x 10^{-4} MPa (0.11 psi) of friction loss to the total. A flow diameter of over 1.04 m (41 in.) is then necessary for both the hot and cold legs in order to limit the pressure losses to the desired levels. The wall thickness for this diameter is about 57.2 mm (2-1/4 in.) for the hot pipe using type 316 SS.

If the expansion loops can be eliminated by using expansion joints, such as the Hyspan model 3500 externally pressurized series, the pipe size could be reduced to about 711 mm (28 in.) and the hot pipe wall thickness to 40.6 mm (1.6 in.). This would represent a cost reduction in the pipe of about 4921 \$/m (1500 \$/ft) and would reduce the run by about 12.2 m (40 ft) and eliminate the elbows. The cost of each expansion joint is about \$18,000, but its qualification remains to be addressed.

Another alternative is to use the smaller pipe with the expansion loops and to increase the allowable system pressure loss. If 762 mm (30 in.) i.d. ducts are used, the loss in the hot leg with two elbows and a 10.7 m (35 ft) center leg is about 0.021 MPa (3.0 psi).

B.4.3 Salt System Components

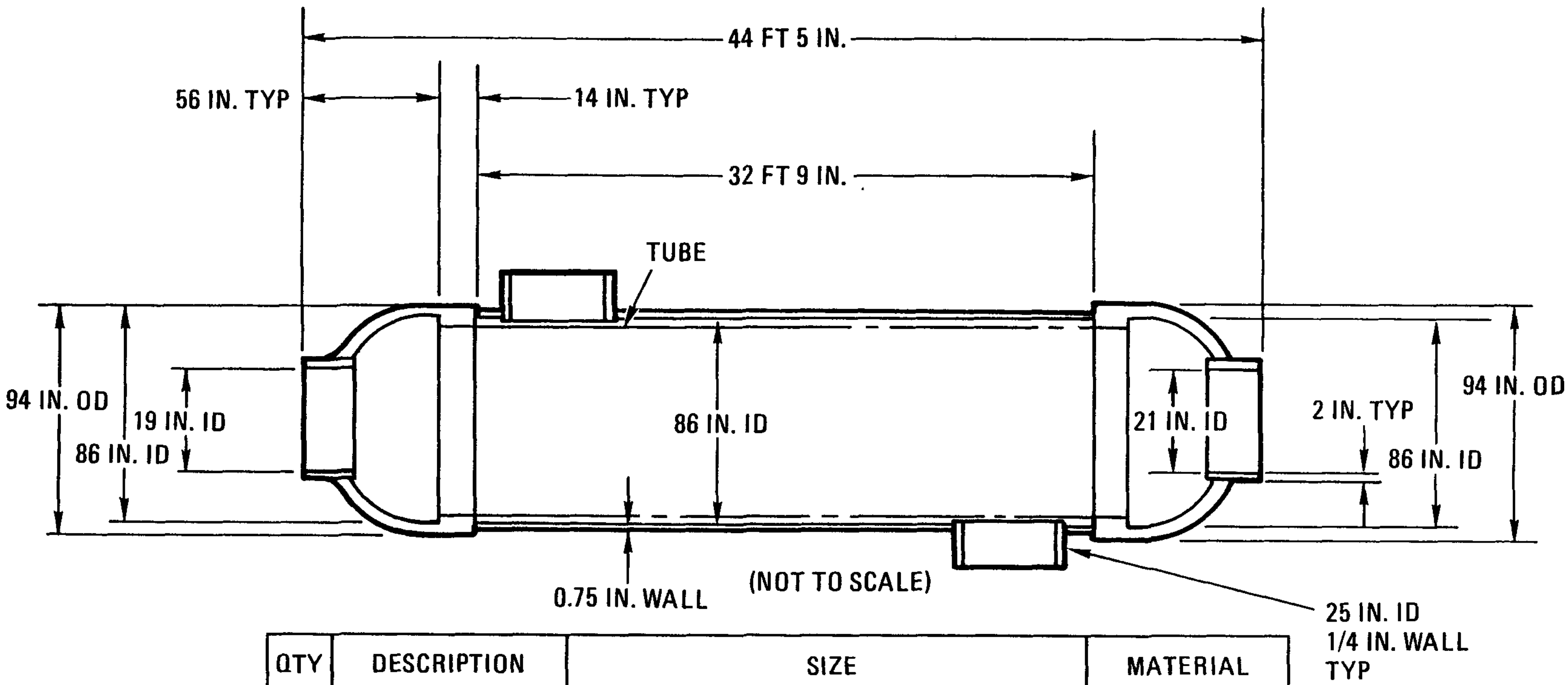
The main components of the salt system downstream of the He/salt heat exchanger are the salt piping and pumps, salt storage, and the salt/steam heat exchanger. These heat exchangers, which are discussed below, are Tubular Exchanger Manufacturer Association (TEMA) type BEM shell-and-straight-tube units, ASME Section VIII, Division 1, similar in construction to high quality process equipment.

B.4.3.1 Salt/Steam Heat Exchangers

Figures B.4.3-1 through B.4.3-3 are typical views of the type of construction employed in the salt/steam heat exchangers. There are two trains of three heat exchangers each. Each train has an economizer, boiler, and superheater; the boiler and superheater units have a steam drum between them.

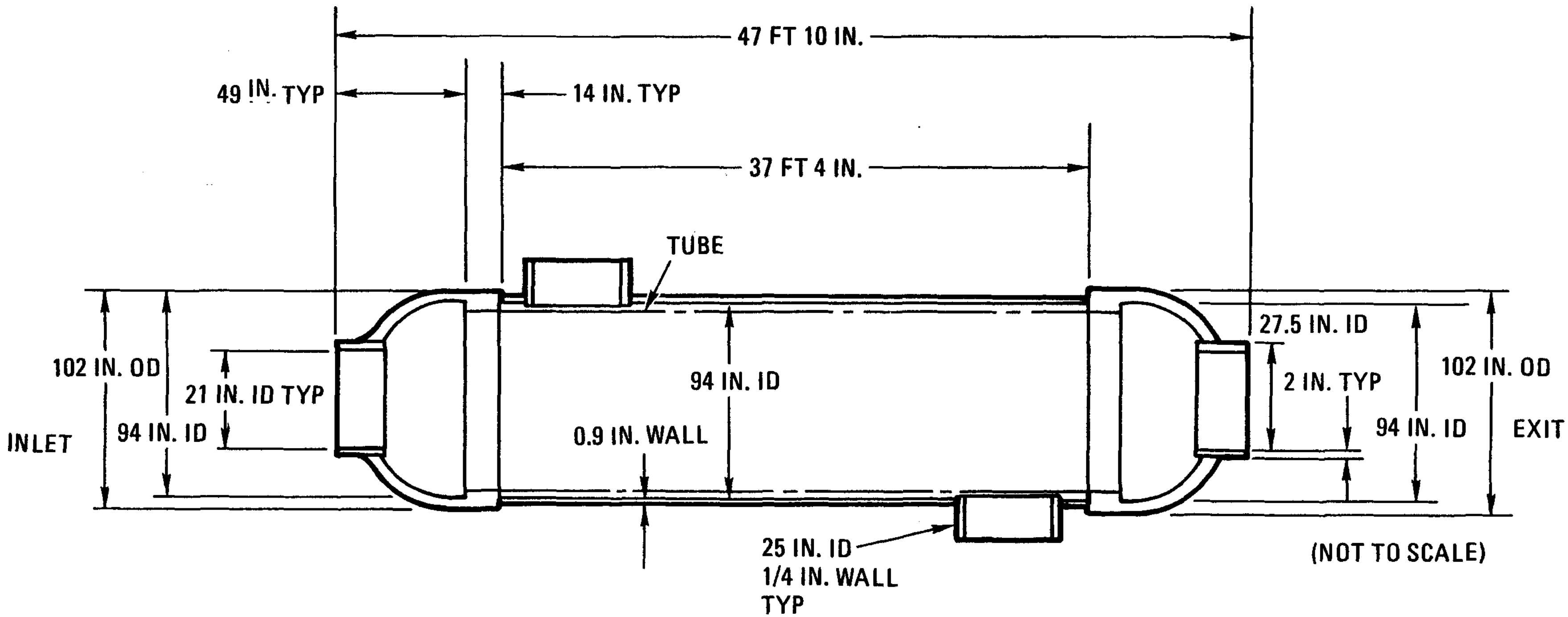
Steam flows through the tube side to minimize shell and tube wall thicknesses. All of these are straight tube cross-counterflow units. Warm salt enters the shell at one end, makes several passes across the tube bundle, and exits at the other end. Steam (or water) enters the tubesheet at the salt exit, flows through the tubes, and exits at the other tubesheet.

Thermal expansion has not been studied, but can be accommodated with relatively conventional bellows similar to those used in the He/steam reheater discussed in Section B.4.2.2.



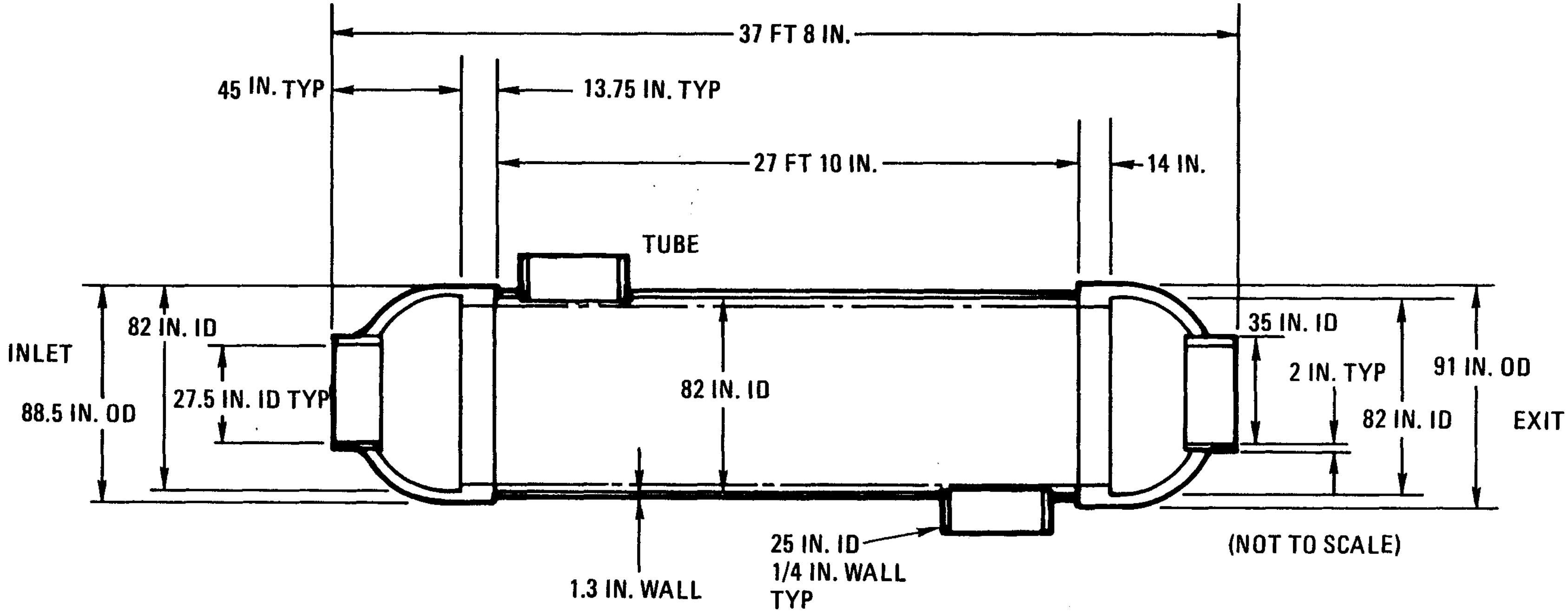
QTY	DESCRIPTION	SIZE	MATERIAL
2	PIPE	25 IN. ID X 1/4 IN. WALL X 18 IN. LG	CARBON STEEL
1	NOZZLE, FORGING	23 IN. OD X 19 IN. ID X 15 IN. LG	CARBON STEEL
1	NOZZLE, FORGING	25 IN. OD X 21 IN. ID X 15 IN. LG	CARBON STEEL
2	HEAD, ELLIPTICAL	2:1 ELLIPTICAL HEAD, 86 IN. ID X 4 IN. THK	CARBON STEEL
2	TUBESHEET	94 IN. OD X 14 IN. THK	CARBON STEEL
1	SHELL	86 IN. ID X 0.75 IN. WALL X 32 FT 9 IN. LG	CARBON STEEL
9151	TUBE	0.50 IN. OD X 0.035 IN. WALL X 35 FT LG	CARBON STEEL

Figure B.4.3-1 HTGR-SALT preheater



QTY	DESCRIPTION	SIZE	MATERIAL
2	PIPE	25 IN. ID X 1/4 IN. WALL X 18 IN. LG	2-1/4 Cr - 1 Mo
1	NOZZLE, FORGING	25 IN. OD X 21 IN. ID X 12 IN. LG	CARBON STEEL
1	NOZZLE, FORGING	31.5 IN. OD X 27.5 IN. ID X 12 IN. LG	CARBON STEEL
2	HEAD, ELLIPTICAL	2:1 ELLIPTICAL HEAD, 94 IN. ID X 4 IN. THK	CARBON STEEL
2	TUBESHEET	102 IN. OD X 14 IN. THK	CARBON STEEL
1	SHELL	94 IN. ID X 0.75 IN. WALL X 37 FT 4 IN. LG	2-1/4 Cr - 1 Mo
11,874	TUBE	0.50 IN. OD X 0.035 IN. WALL X 40 FT LG	2-1/4 Cr - 1 Mo

Figure B.4.3-2 HTGR-SALT boiler



QTY	DESCRIPTION	SIZE	MATERIAL
2	PIPE	25 IN. ID X 1/4 IN. WALL X 18 IN. LG	316 SS
1	NOZZLE, FORGING	27.5 IN. ID X 31.5 IN. OD X 15 IN. LG	CARBON STEEL
1	NOZZLE, FORGING	35 IN. ID X 39 IN. OD X 15 IN. LG	316 SS
1	HEAD ELLIPTICAL	2:1 ELLIPTICAL HEAD, 82 IN. ID X 3.75 IN. THK	CARBON STEEL
1	HEAD, ELLIPTICAL	2:1 ELLIPTICAL HEAD, 82 IN. ID X 4.8 IN. THK	316 SS
1	TUBESHEET, COLD	88.5 IN. OD X 13-3/4 IN. THK	CARBON STEEL
1	TUBESHEET, HOT	91 IN. OD X 14 IN. THK	316 SS
1	SHELL	82 IN. ID X 1-1/4 IN. WALL X 27 FT 10 IN. LG	316 SS
6469	TUBE	0.50 IN. OD X 0.049 IN. WALL X 30 FT LG	316 SS

Figure B.4.3-3 HTGR-SALT superheater

Temperatures in the economizer permit carbon steel throughout, but the higher salt temperatures in the boiler and superheater require 316 SS. Table B.4.3-1 summarizes the key features of the salt-to-steam units generated for costing.

B.4.3.2 Salt Storage and Transport

The salt is transported from the reactor to the user station by a single hot line and a single warm return line. These lines are 32.2 km (20 miles) long, plus a 40% allowance for expansion loops. The hot line is 457 mm (18 in.) diameter, Schedule 10, 316 SS pipe, insulated externally with 152.4 mm (6 in.) of calcium silicate with an aluminum jacket for weather protection. It is supported at 10.97 m (36 ft) intervals so that the maximum deflection between supports does not exceed 6.35 mm (1/4 in.). The warm line is 457 mm (18 in.) diameter, Schedule 10, carbon steel pipe, insulated externally with 76.2 mm (3 in.) of calcium silicate with an aluminum jacket and supported at 10.97 m (36 ft) intervals.

Multiple pump stations are required to limit internal pressures to less than 1.72 MPa (250 psi). This pressure gives acceptable stresses in the thin-walled piping under long-term creep. The hot and warm pumping stations are placed at the same location for convenience in servicing and access. The pumps are single vertical multistage centrifugal units. Based on pump efficiencies of 85%, the pump powers are 0.77 MW(e) for each hot unit and 0.740 MW(e) for each warm unit. Bypasses are provided so that if one pump fails, the flow can be circumvented around the failed unit and the pumping duty can be picked up by the remaining operational units. Alternatively, dual pumps at each stage or a backup pump at each stage can serve either the hot or warm leg. System reliability requirements will be established in later studies.

Calculations show that the warm line will take about 20 hours to cool to 170°C (338°F), which is 27.8°C (50°F) above the melting point. Thus, the system has time following an outage before action must be taken to prevent pipe freeze-up. Sumps are provided at each pump station as an emergency storage for the fluid in each segment of the line associated with that pump.

The piping is not trace heated because that is costly and is believed to be unnecessary. The salt in the piping has a large thermal inertia and there is time to respond should a fault occur. If a fault of some duration should occur, a regular shutdown would be initiated.

The shutdown is like the startup in that the salt is diluted with water to lower the piping system temperature and maintain fluidity at ambient temperatures. The American Hydrotherm Company sells these dilution systems.

Purity of the salt should be maintained since impurities such as carbonates and hydroxides can increase the melting point or the relative

TABLE B.4.3-1
SALT-TO-STEAM HEAT EXCHANGER STATISTICS

	Superheater	Boiler	Preheater
Type construction	Tubular counterflow TEMA BEM		
Number required per plant	2 in series	1	2 in series
Shell i.d., mm (in.)	2082 (82)	2388 (94)	2184 (86)
Shell length, m (ft-in.)	8.5 (27-10)	11.4 (37-4)	10 (32-9)
Shell thickness, mm (in.)	33.4 (1-5/16)	22.9 (0.9)	19.1 (0.75)
Tube o.d. x wall, mm (in.)	12.7 x 1.2 (0.5 x 0.049)	12.7 x 0.9 (0.5 x 0.035)	12.7 x 0.9 (0.5 x 0.035)
Tube length, m (ft)	9.1 (30)	12.2 (40)	10.7 (35)
Number of tubes per HX	6,469	11,874	9,151

corrosiveness of the hot fluid. In addition, there is likely to be some change in the salt composition from thermal degradation (nitrate to nitrite and nitrite to oxide and NO_2). Therefore, an on-line salt purification system is installed, which continually processes a fraction of the flow stream. This same purification system might be utilized to transform draw salt into HTS during the initial startup of the plant if this is found to be more cost effective than external (chemical plant) production of HTS.

The thermal storage has the following general design criteria:

1. There must be sufficient hot storage for energy to be generated for 24 hours and then utilized for 8 hours in a power peaking plant at the user station.

2. There must be sufficient warm storage to handle all the inventory of the pipes plus hot storage.

3. The reactor must be able to continue to operate should the user station be down.

4. The user station must be able to continue to operate should the reactor be down.

5. The hot and warm storage tanks must be located and utilized so as to prevent user station or reactor outages from causing sharp thermal transients to the pipe lines.

6. A 10% ullage allowance is to be added to the storage volume to allow for gas coverage and unavailability of a single tank.

The thermal storage concept is based upon using the sensible heat of the hot molten salt. This salt has a high volumetric heat capacity and low cost. It is a simpler design than latent heat systems, which have intervening heat exchangers and are subject to availability losses due to temperature drops across these heat exchangers. The above design criteria are based on storage tank designs taken from Ref. 1 and specifications given in Table B.4.3-2.

The tanks have a height-to-diameter ratio of 0.6, which reduces the turning moment due to seismic acceleration and the magnitude of concomitant waves on the surface of the stored molten salt. The low turning moment from earthquakes also reduces the required design soil bearing stress. The tanks have external insulation. The warm tank wall is made of carbon steel and the hot tank is 316 SS.

TABLE B.4.3-2
THERMAL STORAGE SPECIFICATIONS

Hot Salt Tanks	
Dimensions	18.3 m (60 ft) diameter x 11 m (36 ft) high, dished roof
Material	SA-240-316 SS
Plate thickness	
Wall	30 mm (1-3/16 in.) bottom 3.66 m (12 ft) 21 mm (13/16 in.) middle 3.66 m (12 ft) 11 mm (7/16 in.) top 3.66 m (12 ft)
Floor	6 mm (1/4 in.) 19 mm (3/4 in.) outer 0.2 m (1/2 ft)
Roof	6 mm (1/4 in) with stiffeners
Connections	305 mm (12 in.) diameter and 356 mm (14 in.) diameter, each with Sparger diffusers
Insulation	152.4 mm (6 in.) calcium silicate with aluminum jacket
Foundation	Compacted dry soil
Warm Salt Tanks	
Dimensions	18 m (59 ft) diameter x 10.7 m (35 ft) high, dished roof
Material	SA-516-Gr 70 carbon steel
Plate thickness	
Wall	19 mm (3/4 in.) bottom 3.66 m (12 ft) 14.3 mm (9/16 in.) middle 3.66 m (12 ft) 7.9 (mm (5/16 in.) top 3.35 m (11 ft)
Floor	7.9 mm (5/16 in.) 15.9 mm (5/8 in.) outer 0.6 m (2 ft)
Roof	6.4 mm (1/4 in.) with stiffeners
Connections	305 mm (12 in.) diameter and 356 mm (14 in.) diameter, each with Sparger diffusers
Insulation	76.2 mm (3 in.) calcium silicate with aluminum jacket
Foundation	Compacted dry soil

The storage is split into several tanks to satisfy the general design criteria given earlier as follows:

1. Four hot storage tanks are located at the reactor station.

2. Four warm storage tanks are located at the reactor station.

3. Eight hot storage tanks are located at the user station.

4. Eight warm storage tanks are located at the user station.

The heat transport and thermal storage system uses current technology with state-of-the-art components. Reasonably accurate cost estimates can be made from the present overall plant conceptual design.

APPENDIX B REFERENCES

1. HTS Thermal Storage Peaking Plant, General Atomic Company, GA-A14160, April 1977.

2. Line Focus Solar Central Power Systems, Phase 1, General Atomic Company, GA-A15580, September 1979.

APPENDIX C

ENGINEERING DRAWINGS

LIST OF DRAWINGS

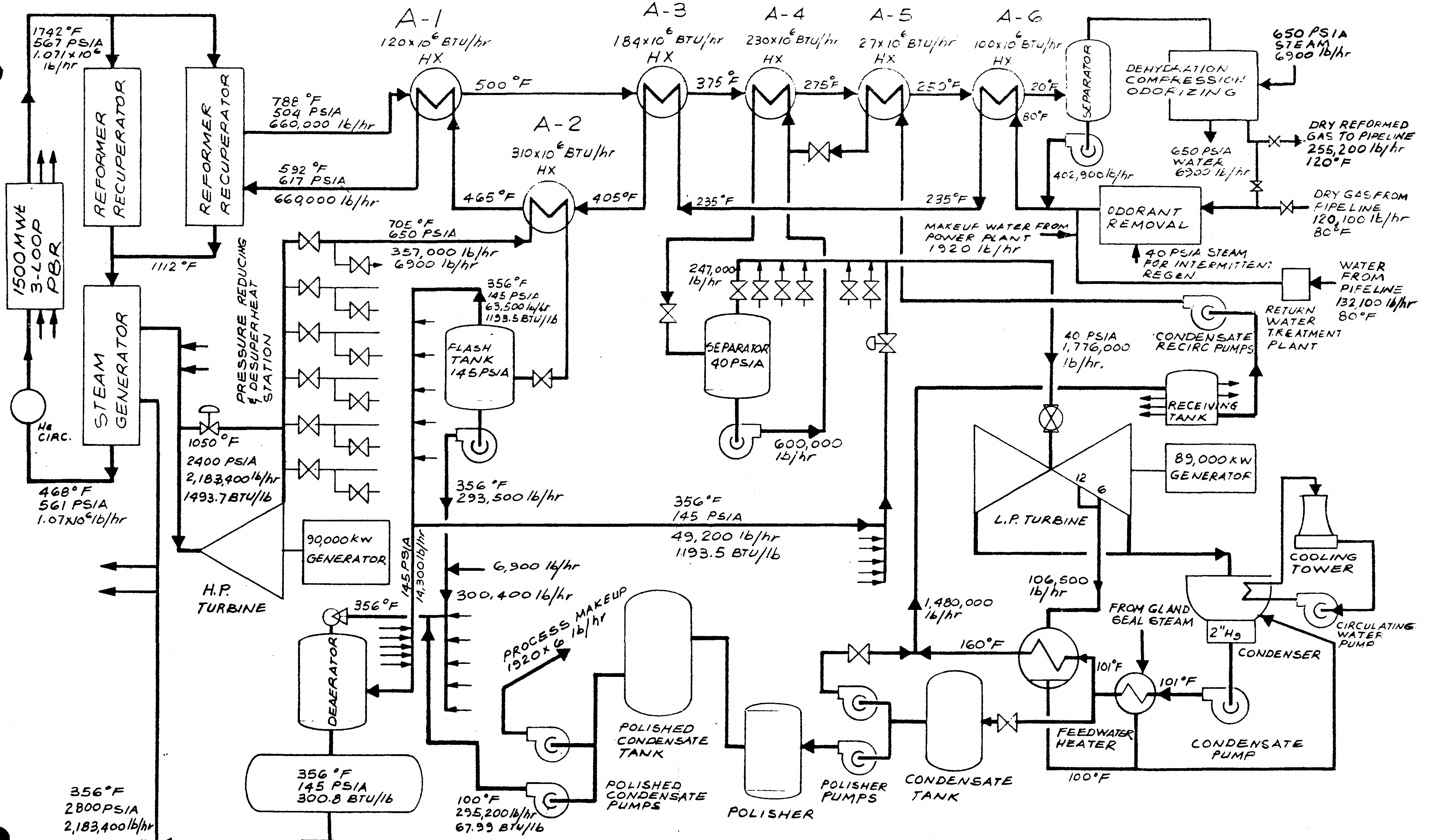


FIGURE 3.2.1-1 HEAT & MASS BALANCE DIAGRAM

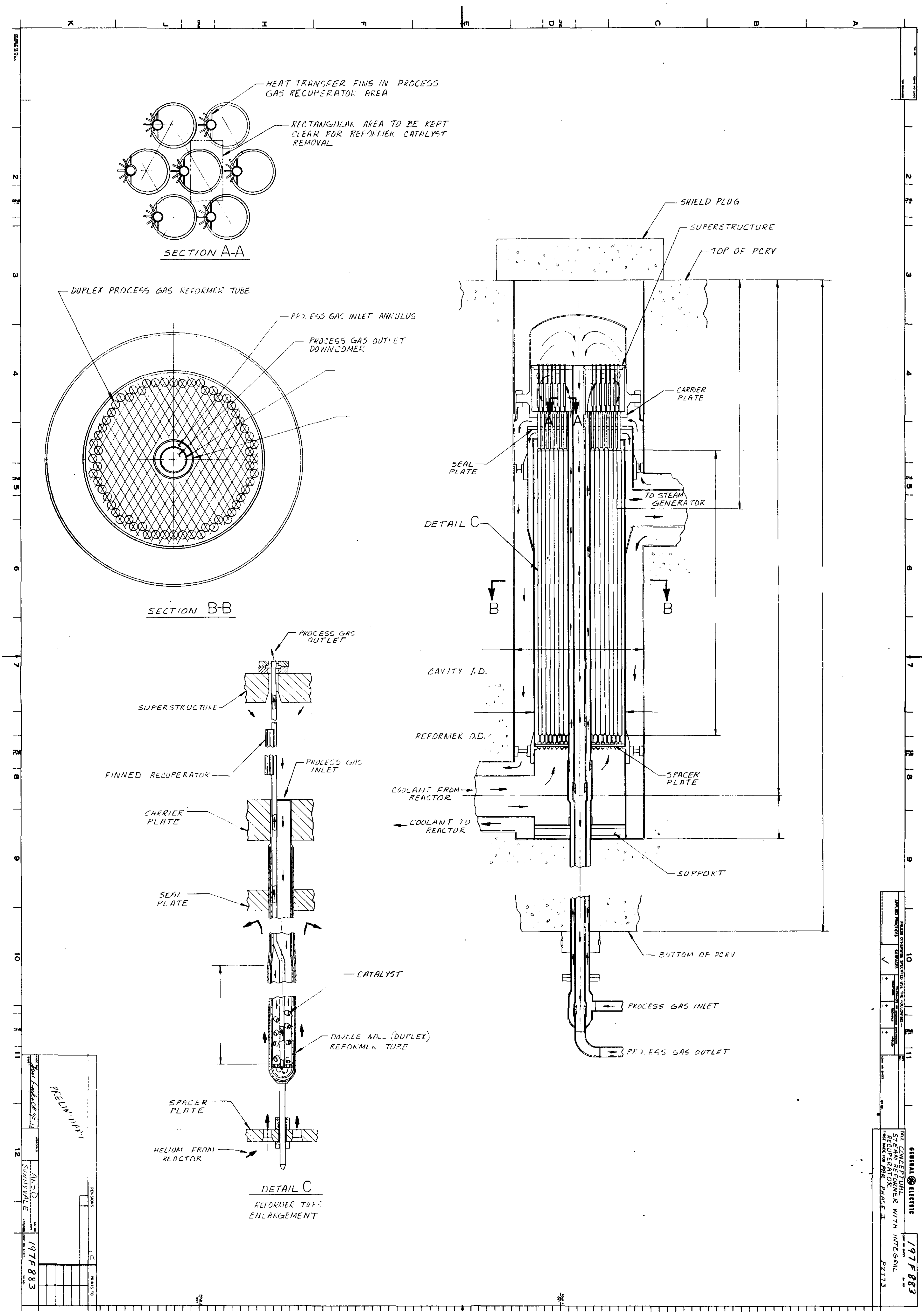


FIGURE 3.2.1-2 Duplex Tube Steam Reformer

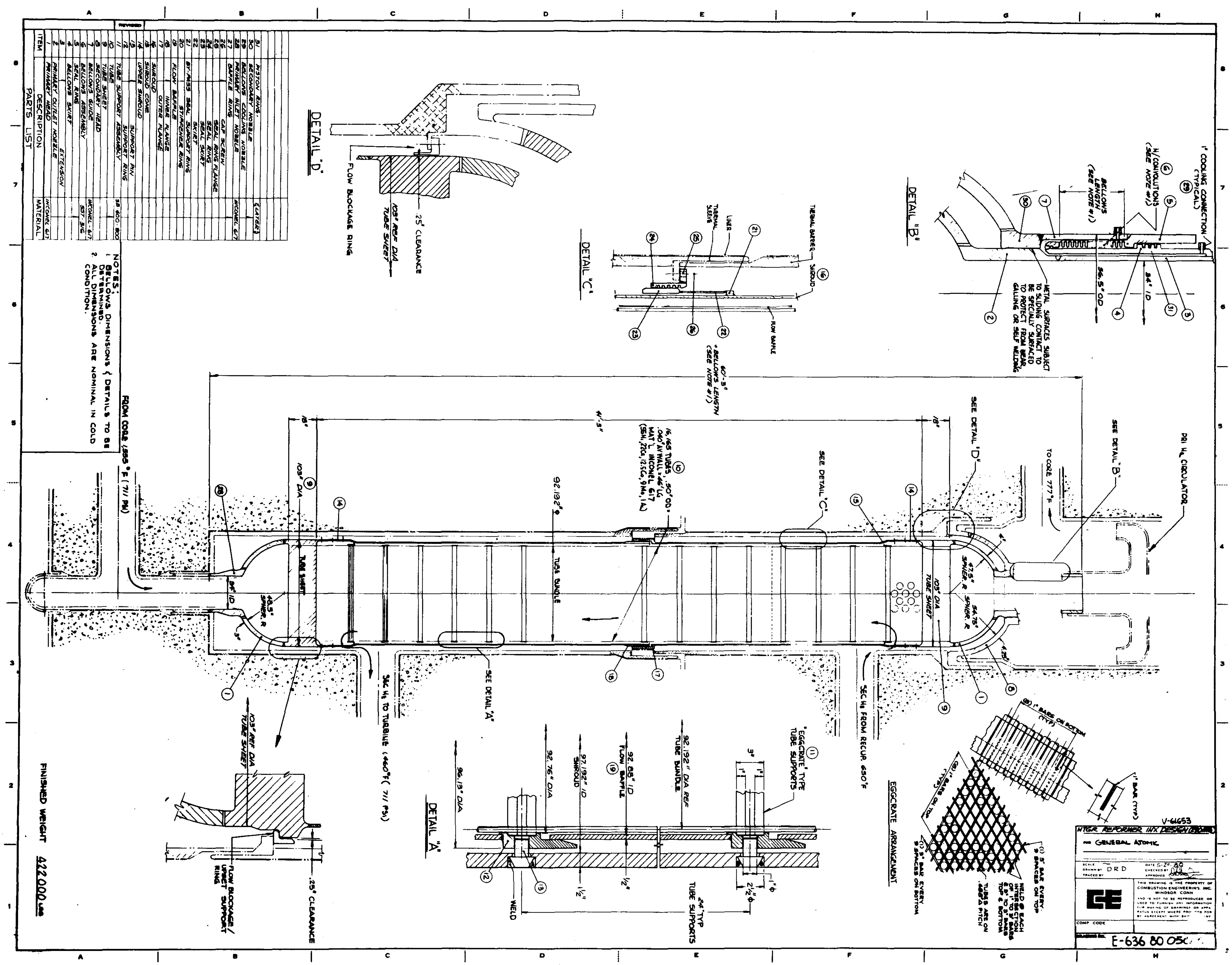


Figure 5.1.1-6 HTGR-R IHX design

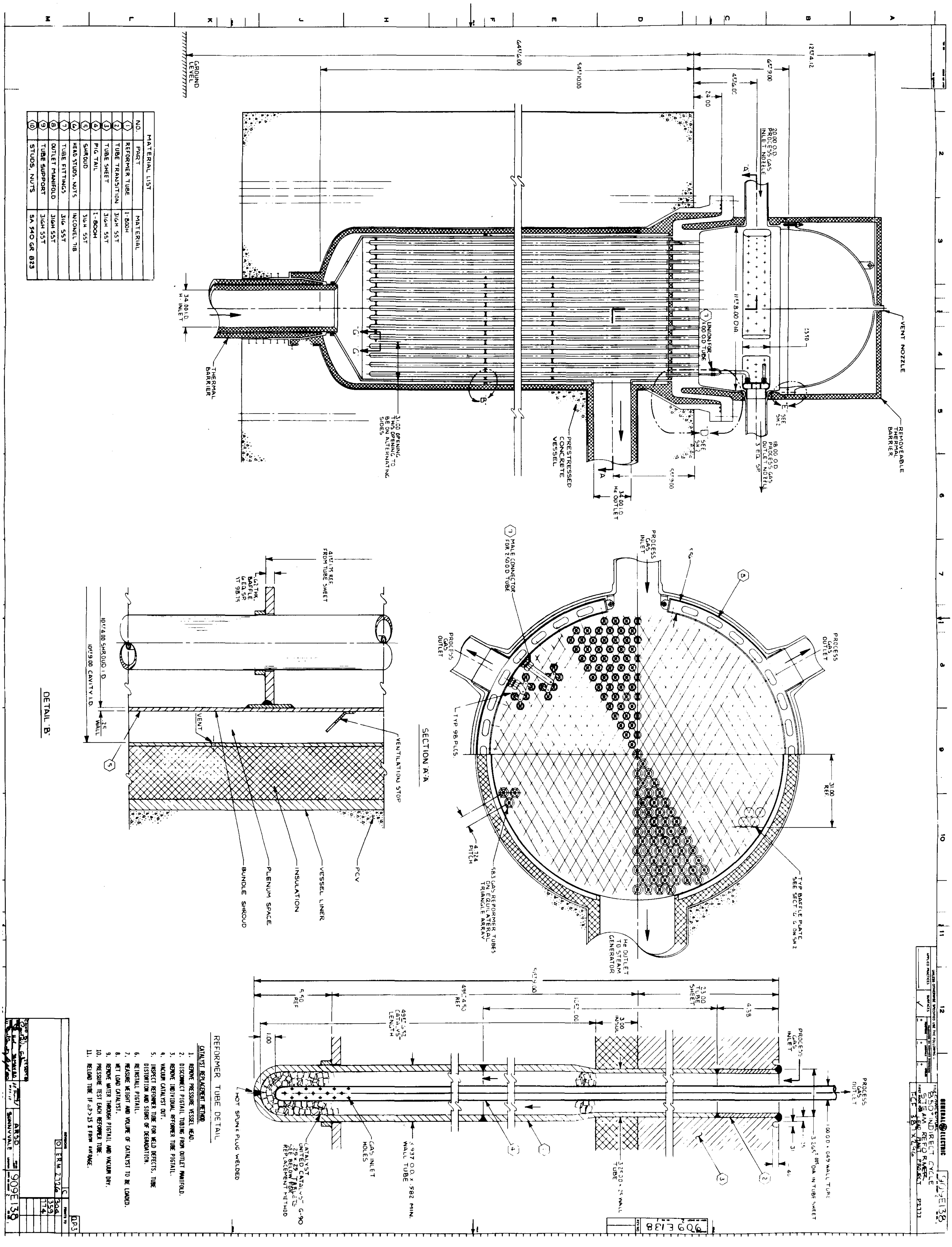


Figure 5.1.3-5 Reformer Design Concept

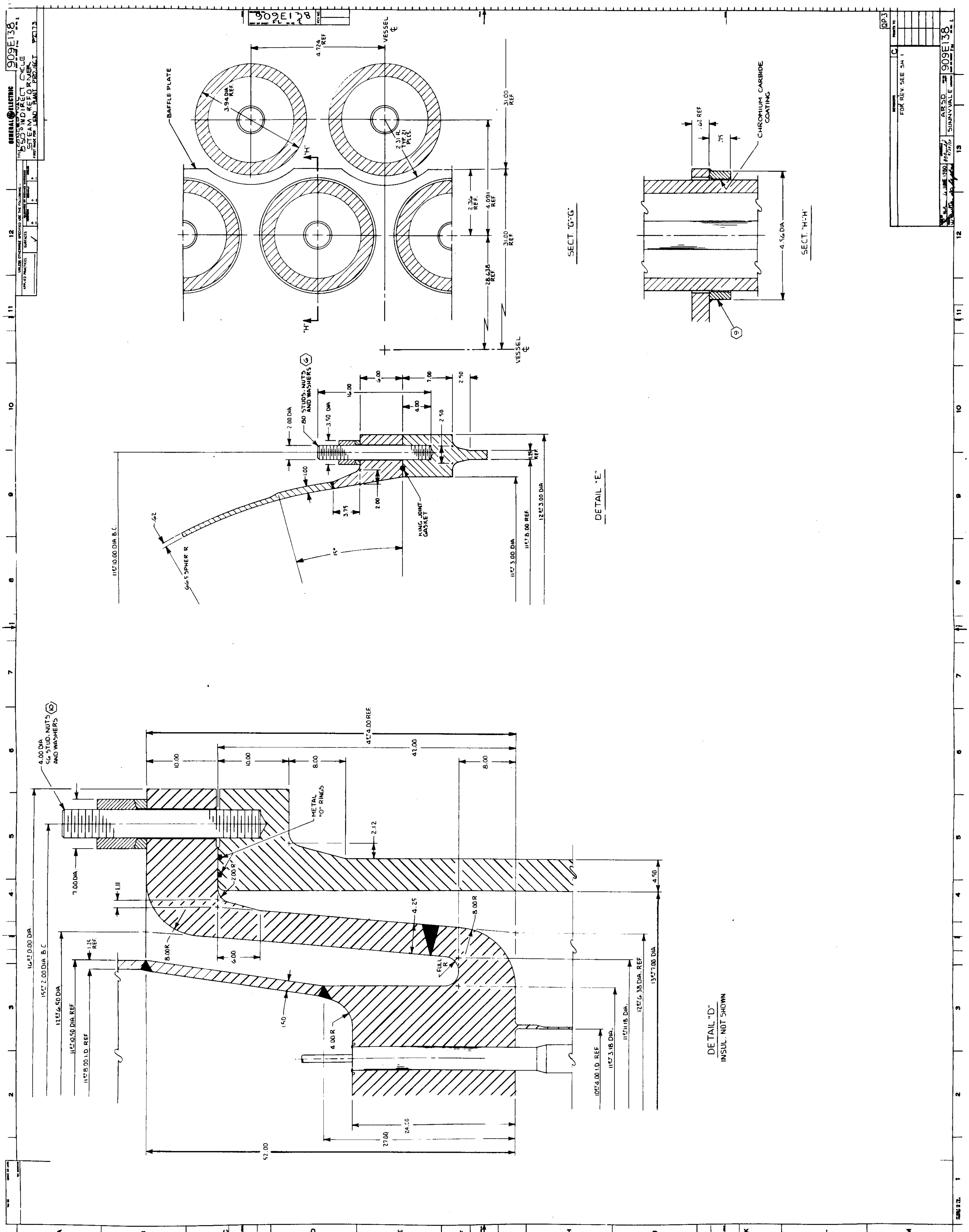


Figure 5.1.3-6 Details of Reformer Design Concept

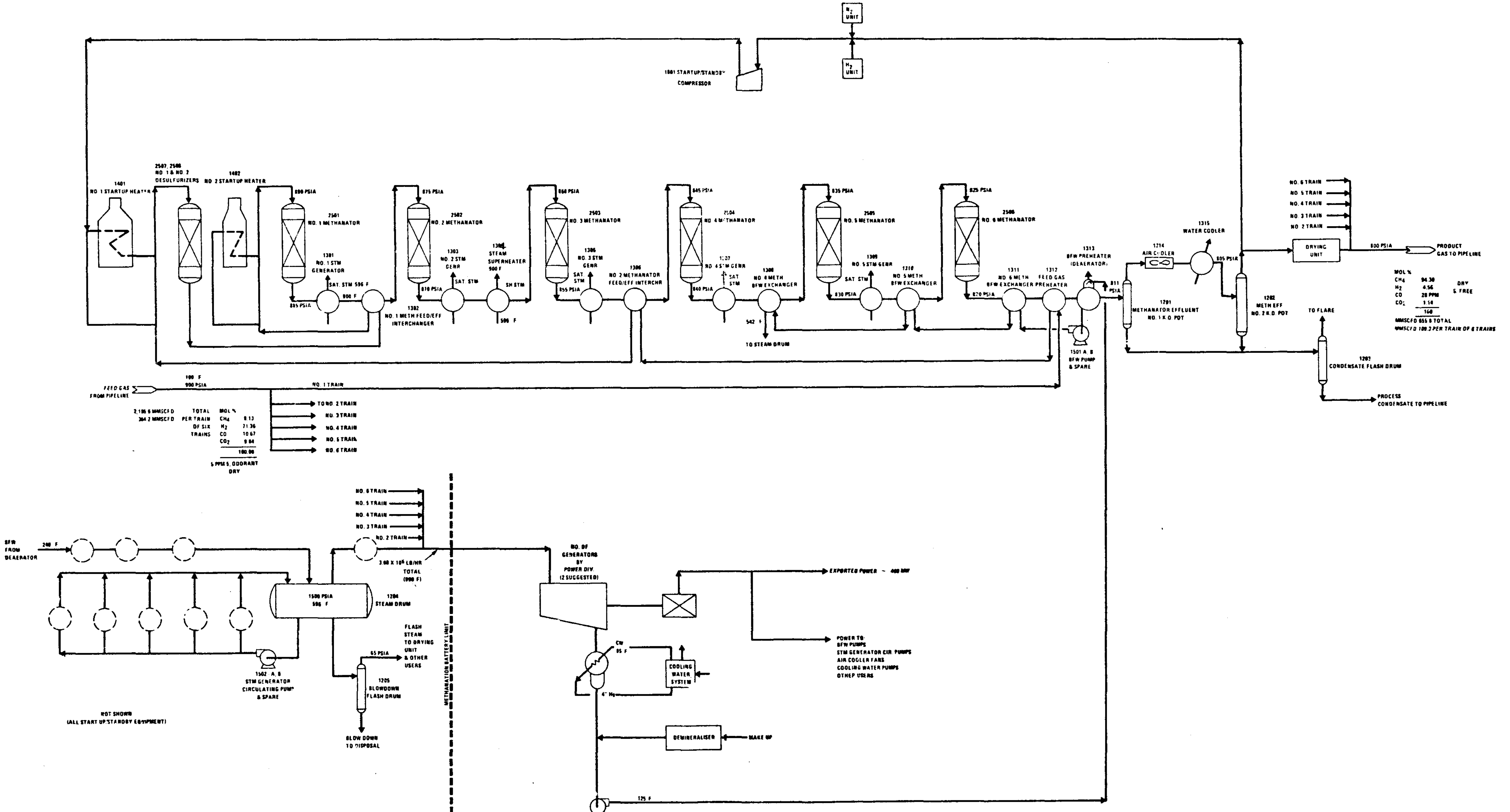


Figure 5.1.3-13 Methanation Trains with Six Methanators Per Train (Load-Following Electricity Plant)

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GENERAL ELECTRIC COMPANY
ADVANCED REACTOR SYSTEMS DEPARTMENT
DESIGN OF METHANATION PLANTS FOR PEAKING
CASE 1 — PEAKING POWER GENERATION

6065-1

6065-FS-1

NO	DATE	BY	CHKD	SEC. PROJ.	CLIENT	DESCRIPTION
0	6/27/[illegible]	JCL				FCR REPORT

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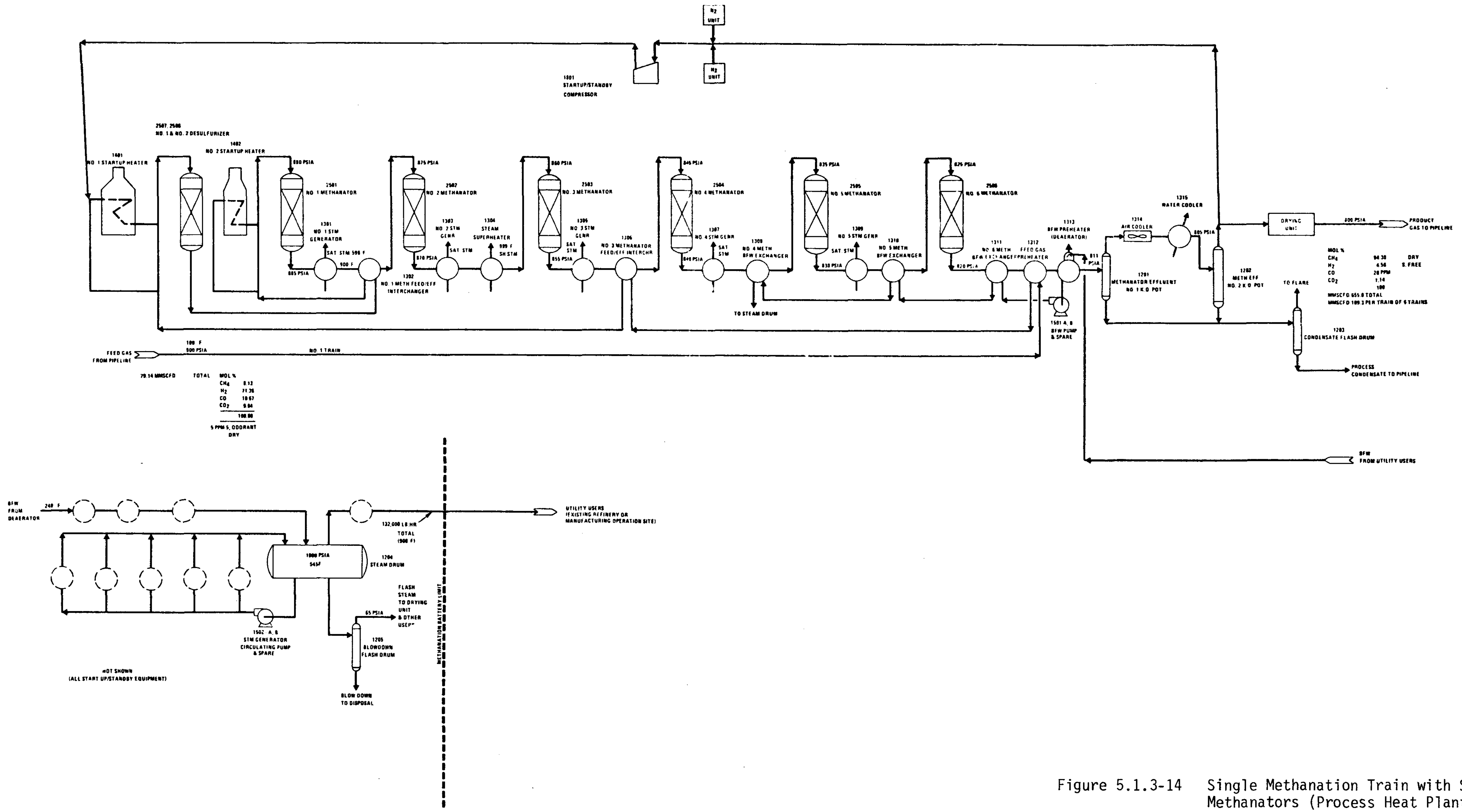

Figure 5.1.3-14 Single Methanation Train with Six Methanators (Process Heat Plant)

GENERAL ELECTRIC COMPANY
ADVANCED REACTOR SYSTEMS DEPARTMENT
DESIGN OF METHANATION PLANTS FOR PEAKING
CASE 2 — PROCESS HEAT

6065-1

6065-FS-2

NO	DATE	BY	CHKD	SEC	PROJ	CLIENT	DESCRIPTION
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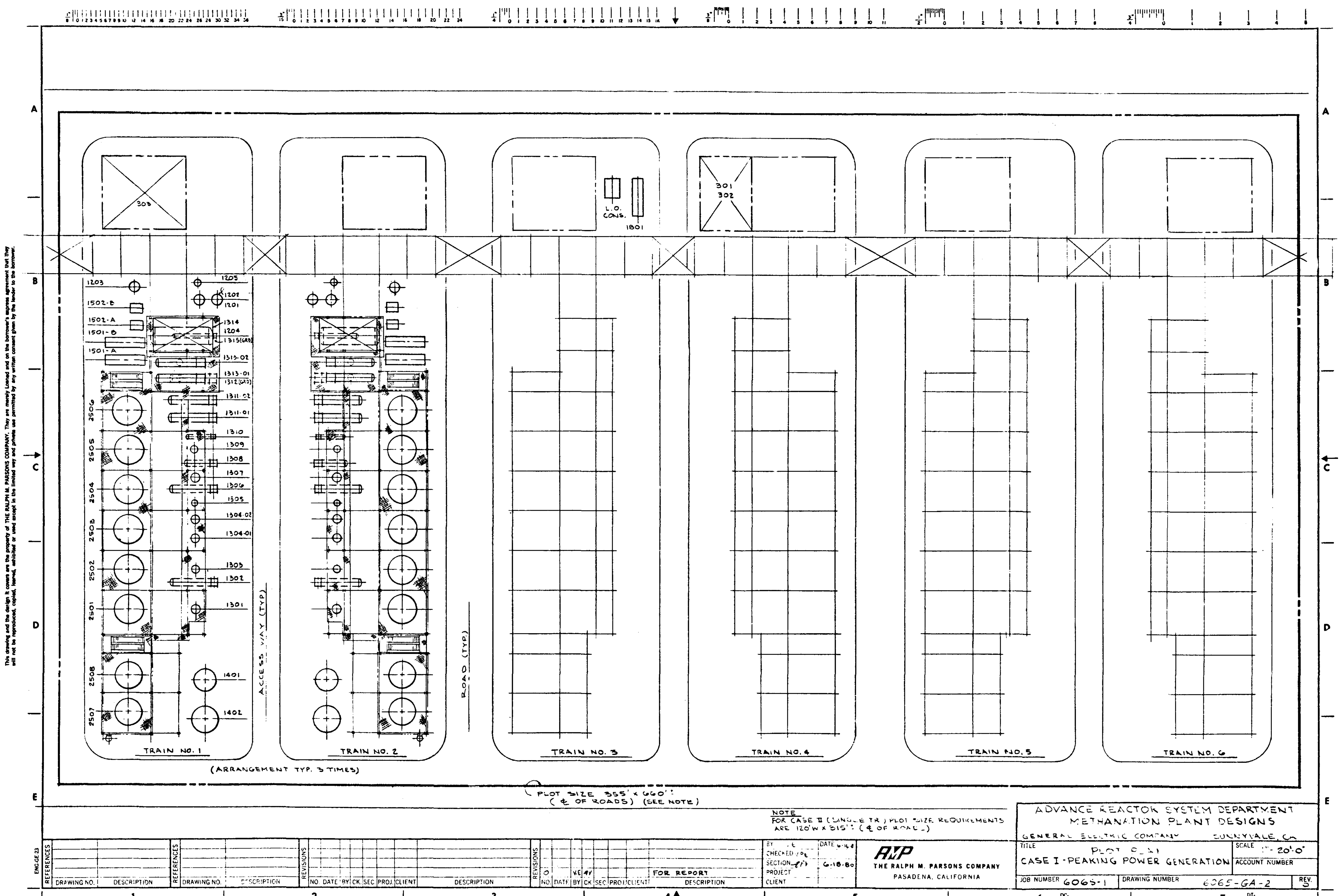

Figure 5.1.3-15 Plot Plan for Load-Following Electricity Plant

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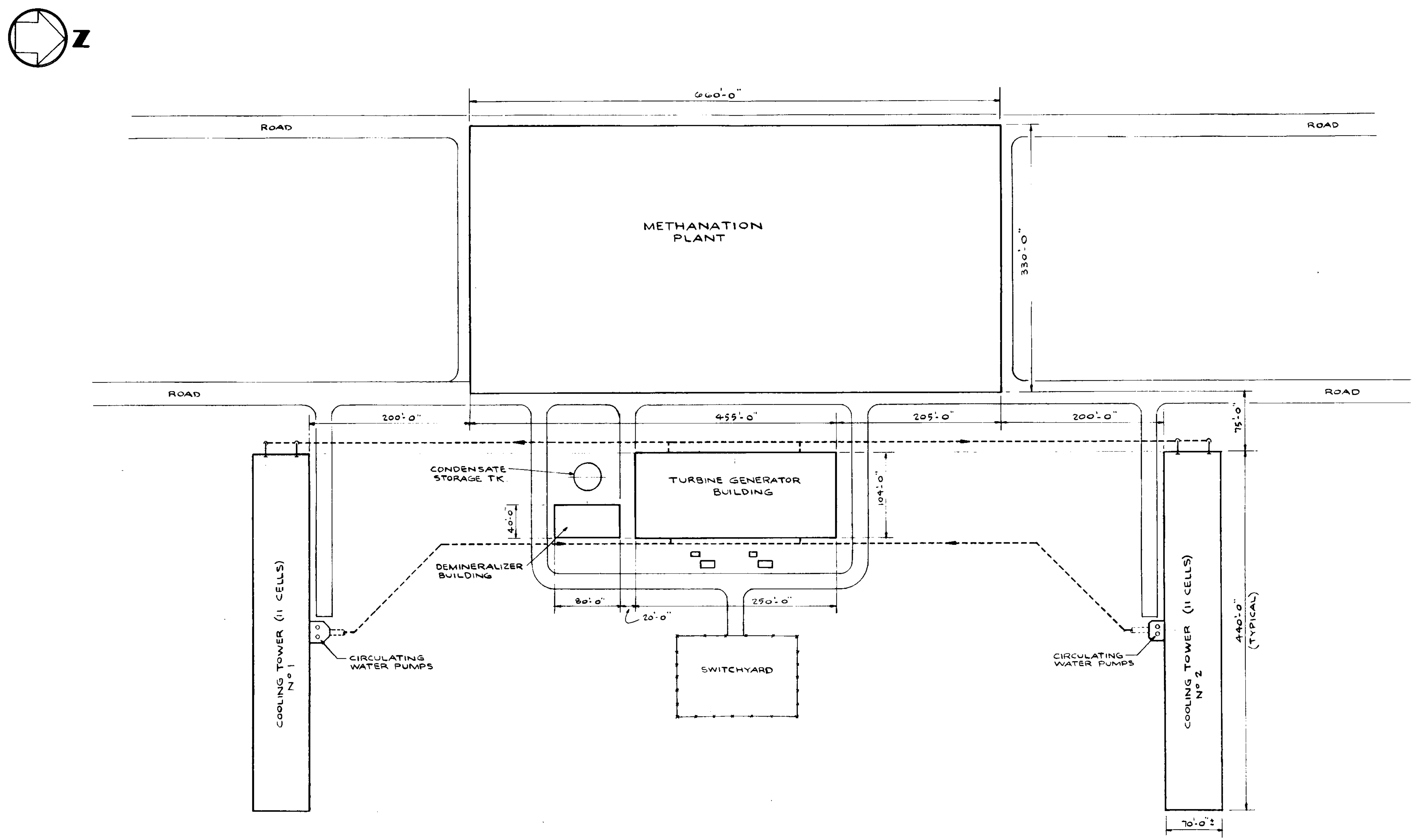

Figure 5.1.3-16 General Arrangement Plot Plan of Load-Following Electricity Plant

GENERAL ELECTRIC COMPANY
ADVANCED REACTOR SYSTEMS DEPARTMENT
DESIGN OF METHANATION PLANTS FOR
CASE I- PEAKING POWER GENERATION

TITLE: GENERAL ARRANGEMENT PLOT PLAN

SCALE: 1":60'-0"

JOB NUMBER: 6065-1

DOCUMENT NUMBER: 6065-GA-1

DRAWN: C.K. 6-25-80

REVISION 0: FOR REPORT

ENG GE 23 (12 79)

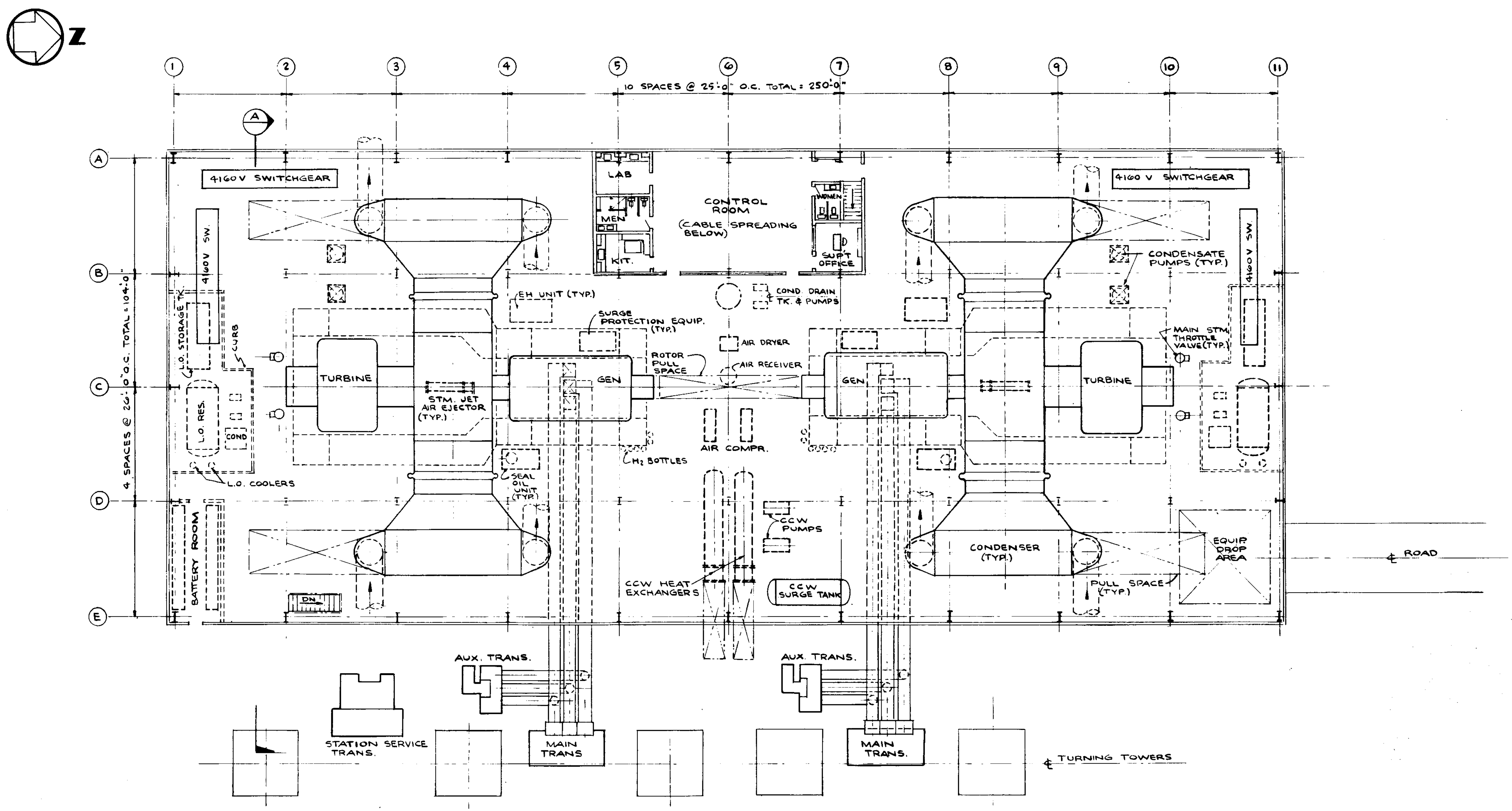

Figure 5.1.3-17 General Arrangement Turbine Building Plan

GENERAL ELECTRIC COMPANY
ADVANCED REACTOR SYSTEMS DEPARTMENT
DESIGN OF METHANATION PLANTS FOR
CASE I - PEAKING POWER GENERATION

DRAWN	C.K.	6/16/80
CHECKED	AM	6/25/80
APVD SECT HD		6/25/80
APVD PROJ MGR		6/11/81
APVD CLIENT		

NO.	DATE	BY	CHKD	SEC	PROJ	CLIENT	DESCRIPTION
0	6/25/80	C.K.	AM				FOR REPORT

RMP THE RALPH M. PARSONS COMPANY PASADENA, CALIFORNIA

TITLE: GENERAL ARRANGEMENT TURBINE BUILDING PLAN
SCALE: 3/32" = 1'-0"
JOB NUMBER: 6065-1
DOCUMENT NUMBER: 6065-GA-3
REVISION: 0

ENG-GE-23 (12/79)

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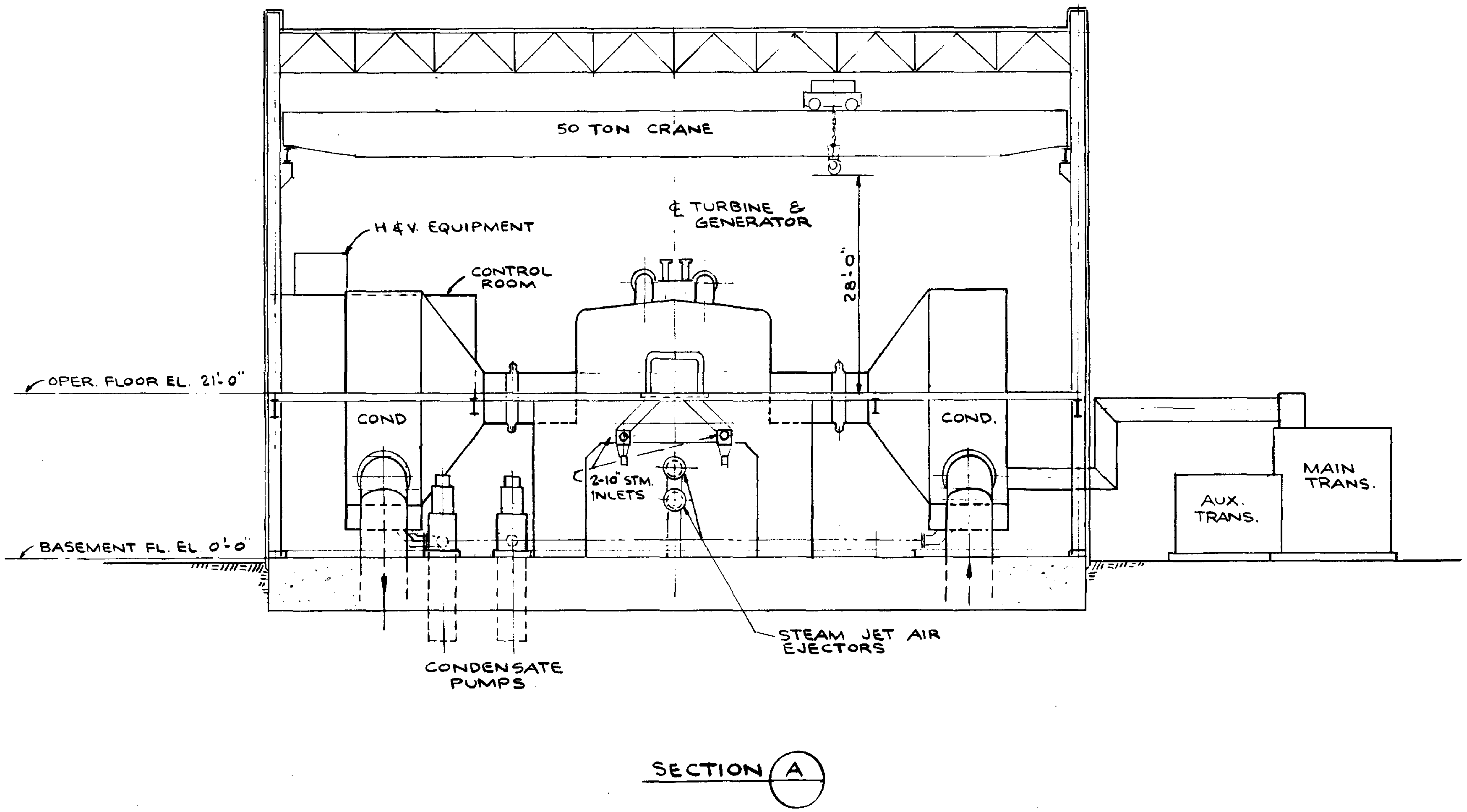

Figure 5.1.3-18 General Arrangement Turbine Building Section

GENERAL ELECTRIC COMPANY
ADVANCED REACTOR SYSTEMS DEPARTMENT
DESIGN OF METHANATION PLANTS FOR
CASE I - PEAKING POWER GENERATION

TITLE: GENERAL ARRANGEMENT TURBINE BUILDING SECTION "A"

SCALE: 3/32" = 1'-0" | JOB NUMBER: 6065-1 | DOCUMENT NUMBER: 6065-GA-4 | REV: 0

NO	DATE	BY	CHKD	SEC	PROJ	CLIENT	DESCRIPTION
0	6/25/80	C.K	AM				FOR REPORT

REFERENCES: DRAWING NO | DESCRIPTION

FOR: DRAWN C.K. 6/17/80; CHECKED AM; APVD SECT HD; APVD PROJ MGR; APVD CLIENT

RMP THE RALPH M. PARSONS COMPANY PASADENA, CALIFORNIA

ENG-GE-22 (11/78)

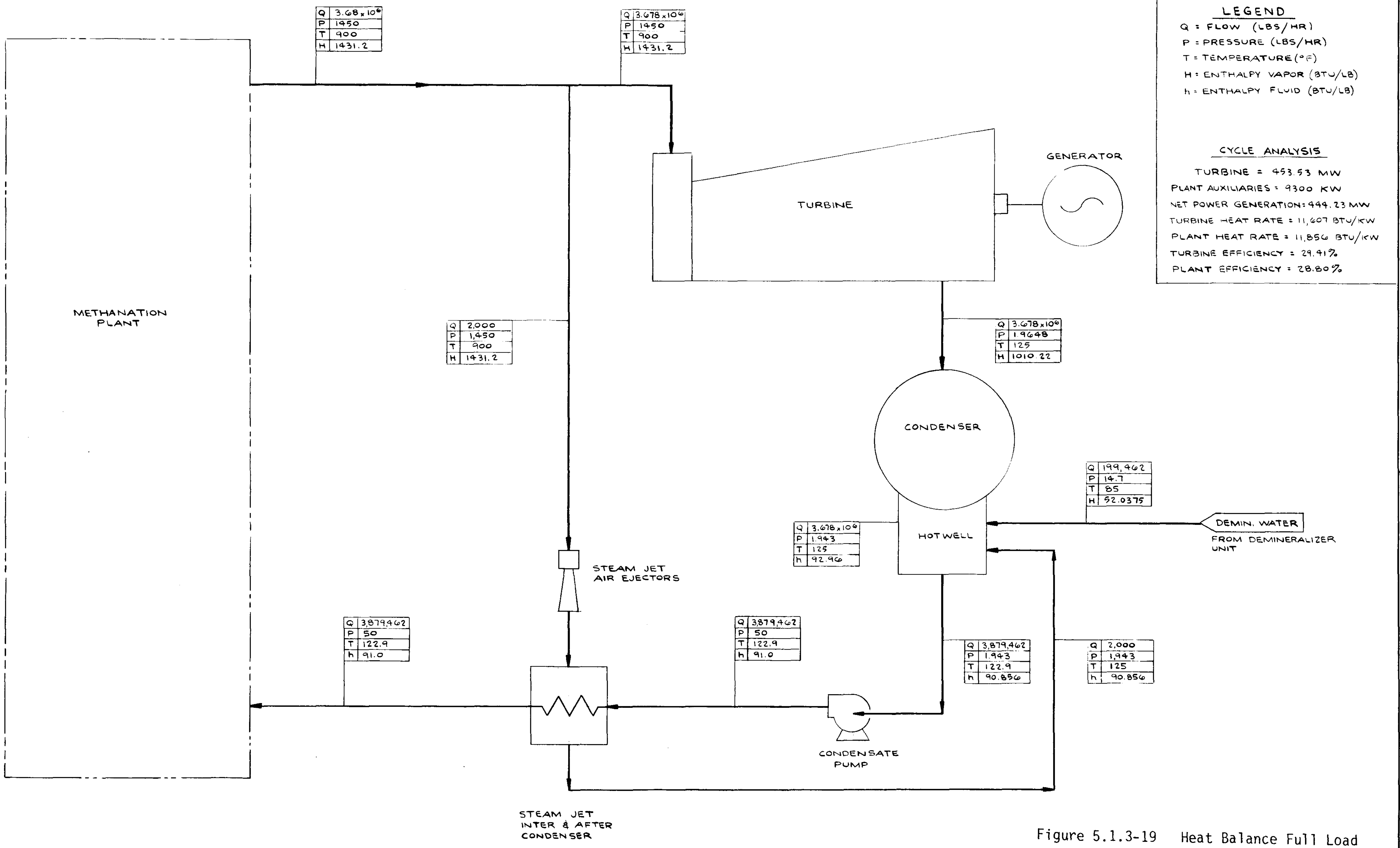

Figure 5.1.3-19 Heat Balance Full Load

GENERAL ELECTRIC COMPANY
ADVANCED REACTOR SYSTEMS DEPARTMENT
DESIGN OF METHANATION PLANTS FOR
CASE I - PEAKING POWER GENERATION

THE RALPH M. PARSONS COMPANY
PASADENA CALIFORNIA

HEAT BALANCE
FULL LOAD

DRAWN	C.K.	6-18-80
CHECKED	AM	6/25/80
APVD SECT HD		6/25/80
APVD PROJ MGR		6/25/80
APVD CLIENT		

NO	DATE	BY	CHKD	SEC	PROJ	CLIENT	DESCRIPTION
0	6/25/80	C.K.	AM				FOR REPORT

SCALE — ACCOUNT NUMBER — JOB NUMBER 6065-1 REVISION 0

DOCUMENT NUMBER 6065-FS-3

ENG GE 23 (12/79)

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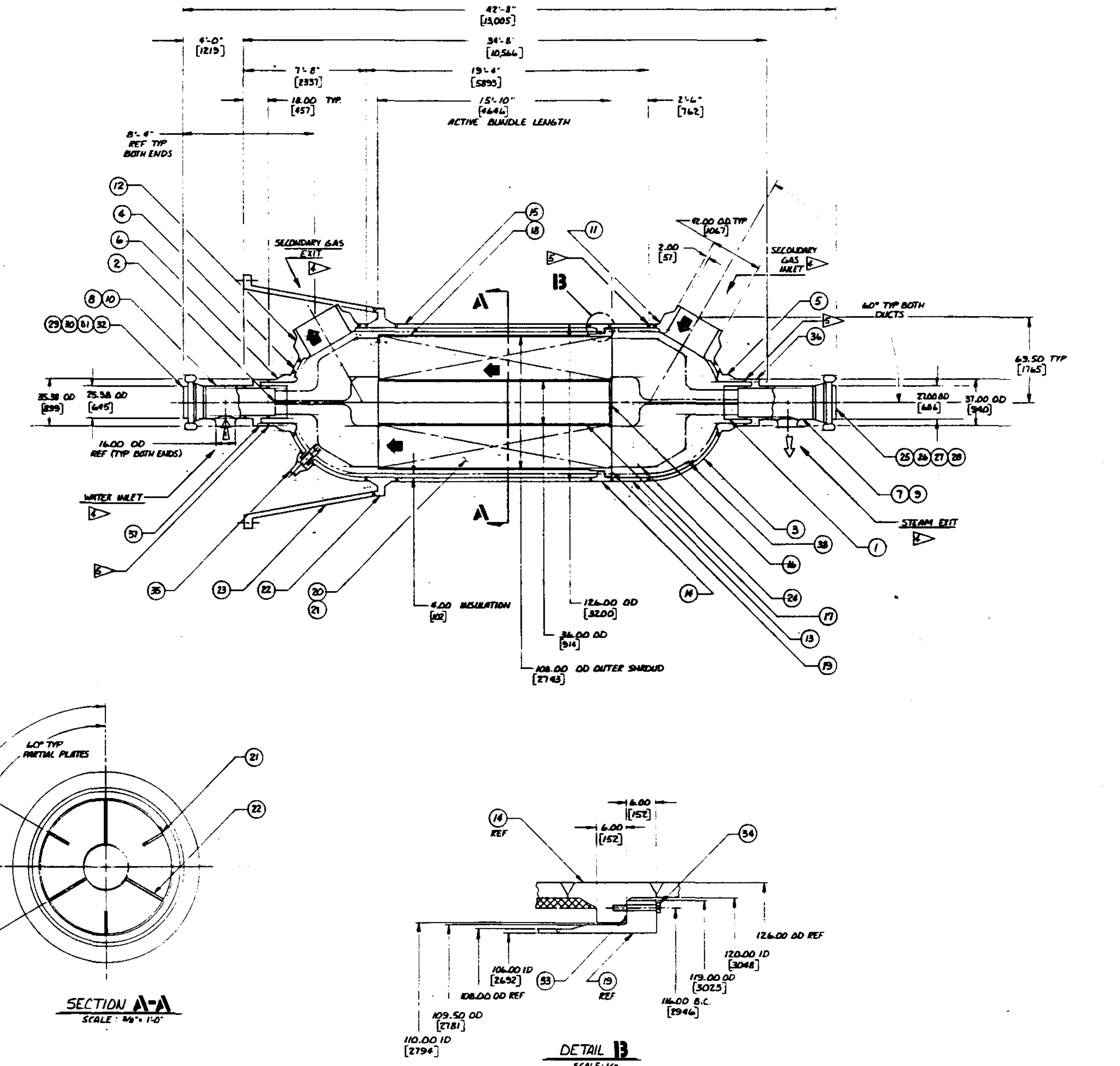

SECTION A-A
SCALE: 3/8"=1'-0"

DETAIL B
SCALE: 1/2

NOTES:

1. ALL DIMENSIONS ARE NOMINAL COLD CONDITION.
2. DIMENSIONS IN [ ] BRACKETS ARE IN MILLIMETERS
3. MANUFACTURE IN ACCORDANCE WITH ASME BOILER AND PRESSURE VESSEL CODE SECTION VIII DIVISION I.
4. DESIGN CONDITIONS:
   - NUMBER OF TUBES : 180 (HELICAL BUNDLE)
   - TUBE SIZE : 1.00 OD x .191 WALL
   - TRANSVERSE PITCH : 1.452
   - LONGITUDINAL PITCH : 1.45
   - SECONDARY COOLANT : HELIUM (SHELL SIDE)
     - INLET TEMP : 1100 °F (593 °C)
     - EXIT TEMP : 608 °F (320 °C)
     - PRESSURE : INLET : 688 PSIA; OUTLET : 673 PSIA
     - FLOW RATE : 1.0175×10^6 LB/HR
   - SECONDARY COOLANT : WATER (TUBESIDE)
     - INLET TEMP : 356 °F (180 °C)
     - EXIT TEMP : 950 °F (510 °C)
     - PRESSURE : INLET : 2925 PSIA; OUTLET : 2500 PSIA
     - FLOW RATE : 5.615×10^5 LB/HR
5. FIELD WELD FOR FINAL FIT-UP OF VESSEL AND HEAT EXCHANGER (OPTIONAL)
6. SECONDARY GAS NOZZLES AND INSTRUMENTATION PENETRATION MAY BE AT OPTIONAL LOCATIONS ON SAME ELEVATIONS

REQD/ASSEMBLY	ITEM	PART NO	DESCRIPTION	MATL	MATL SPEC
AR	38		INSULATION - 4" THK		
1	37		TORUS - 35 3/16" OD x 31 3/16" ID x 18" LG	CARBON STL	SA-508 CL 1
1	36		TORUS - 37" OD x 33" ID x 18" LG	2¼ Cr - 1Mo	SA-336 F22
LOT	35		INSTRUMENTATION		
48	34		BOLT - 1.00 - 8UN - 2B x 9" LG		
1	33		SEAL, FLEXITALIC 111" ID x 113" OD	LATER	
1	32		GRAYLOC CLAMP - 25 3/8"	CARBON STL	
1	31		GRAYLOC SEAL - 25 3/8"	↓	
1	30		GRAYLOC BLIND HUB - 25 3/8" OD	↓	
1	29		GRAYLOC WELD HUB - 25 3/8" OD	CARBON STL	
1	28		GRAYLOC SEAL - 27"	2¼ Cr - 1Mo	
1	27		GRAYLOC CLAMP - 27"	↓	
1	26		GRAYLOC BLIND HUB - 27"	↓	
1	25		GRAYLOC WELD HUB - 27"	↓	
4	24		TUBE - 1.00 OD x .191 WALL x 536' LG	2¼ Cr - 1Mo	
1	23		CYLINDER SUPPORT - 9' LG x 4" THK	CARBON STL	SA-516 GR70
1	22		RING SUPPORT - 120" ID x 150" OD x 24"	CARBON STL	SA-508 CL 1
3	21		PLATE, SUPPORT, PARTIAL - 15'-10" x 18" x 1.75"	2¼ Cr - 1Mo	SA-387 GR22
3	20		PLATE, SUPPORT, FULL - 15'-10" x 36" x 1.75"	↓	SA-387 GR22
1	19		FLANGE, SHROUD - 106" ID x 118 OD x 20" LG	↓	SA-336 F22
1	18		SHROUD, OUTER - 108" OD x 1" THK x 14'-1" LG	↓	SA-387 G22
1	17		SHROUD, INNER - 36" OD x 1" THK x 15'-9 1/2"	↓	SA-387 GR22
1	16		CAP - 36" DIA x 3/4" THK	↓	SA-387 GR22
1	15		SHELL, OUTER - 126" OD x 3" THK x 15'-4" LG	↓	SA-387 GR22
1	14		FLANGE, RING - 126" OD x 110" ID x 18" LG	↓	SA-336 F22
1	13		SHELL, UPPER - 126" OD x 3" THK x 30" LG	2¼ Cr - 1Mo	SA-387 GR22
1	12		NOZZLE, EXIT - 53" OD x 38" ID x 24" LG	CARBON STL	SA-508 CL 1
1	11		NOZZLE, INLET - 53" OD x 38" ID x 24" LG	2¼ Cr - 1Mo	SA-336 F22
1	10		WELDOLET - 16" OD x 13" ID	CARBON STL	SA-508 CL 1
1	9		WELDOLET - 16" OD x 13" ID	2¼ Cr - 1Mo	SA-336 F22
1	8		PIPE, INLET - 25 3/8" OD x 22 3/8" ID x 32" LG	CARBON STL	SA-516 GR70
1	7		PIPE, EXIT - 27" OD x 22 3/4" ID x 32" LG	2¼ Cr - 1Mo	SA-387
1	6		NOZZLE, INLET - 45" OD x 31 5/8" ID x 24" LG	CARBON STL	SA-508 CL 1
1	5		NOZZLE, EXIT - 48" OD x 33" ID x 24" LG	2¼ Cr - 1Mo	SA-336 F22
1	4		DOME, LOWER - 60" R INSIDE x 3" THK	CARBON STL	SA-516 GR70
1	3		DOME, UPPER - 60" RAD (INSIDE) x 3" THK	2¼ Cr - 1Mo	SA-387
1	2		TUBESHEET, INLET - 25.38 OD x 9" THK	CARBON STL	SA-508 CL 1
1	1		TUBESHEET, EXIT - 27" OD x 11" THK	2¼ Cr - 1Mo	SA-336 F22
			ASSEMBLY		

NEXT USED ON: 904813, 904661

DISC	SYSTEM	TYPE	PROJECT
H	29	DSK	0000

PARTS LIST

ESTIMATED WEIGHT: 449,000 LBS (203,680 KGS)

HTGR-REFORMER 1170 MW(t) PLANT & NHSDR PLANT

DIST.

GENERAL ATOMIC COMPANY
SAN DIEGO, CALIFORNIA

LTR	DATE	ISSUE DESCRIPTION	DR.	CHK.	DES.	APPD	APPD
/2		INCORP BOP (INTERFACE REQM'TS)					
/1		ORIG. FOR PROJS. 6435 & 7200					
		PRELIMINARY ISSUE					

ISSUED 1
ISSUED 1

UNLESS OTHERWISE SPECIFIED
DIMENSIONS ARE IN INCHES
NO BURRS OR SHARP EDGES
DIMENSIONS PER ANSI-Y-14.5
TOLERANCE:
DECIMAL FRACTION ANGLE
.XXX = ± ± ±
.XX = ± .12
ALL MACHINED SURFACES
DO NOT SCALE PRINT

PROPRIETARY INFORMATION
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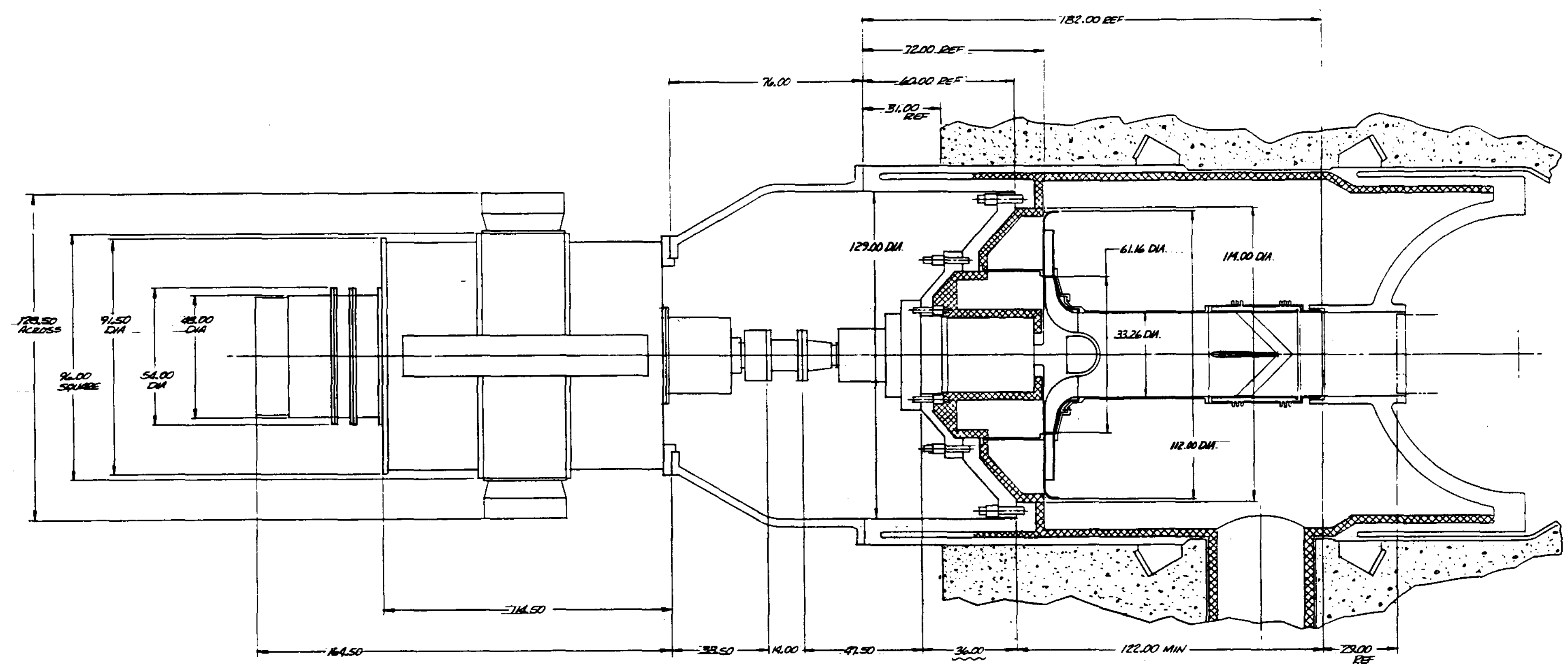
TITLE REFORMER 179 MW(t) STEAM GENERATOR GENERAL ARRANGEMENT

SIZE	CODE IDENT NO	DWG. NO.	ISSUE
E	32334	025784	/2

SCALE 3/8"=1'-0" SHEET 1 OF 1

025784 /2



ITEM	PART NO	DESCRIPTION	MATL/MATL SPEC
	904813	PARTS LIST	

REQD/ASSEMBLY M 21 DSK 720C

DIST.

GENERAL ATOMIC COMPANY

UNLESS OTHERWISE SPECIFIED
DIMENSIONS ARE IN INCHES
NO BURRS OR SHARP EDGES
DIMENSIONS PER ANSI-Y-14.5
TOLERANCE
DECIMAL FRACTION ANGLE
.XXX = ± ± ±
.XX = ±
ALL MACHINED SURFACES
DO NOT SCALE PRINT

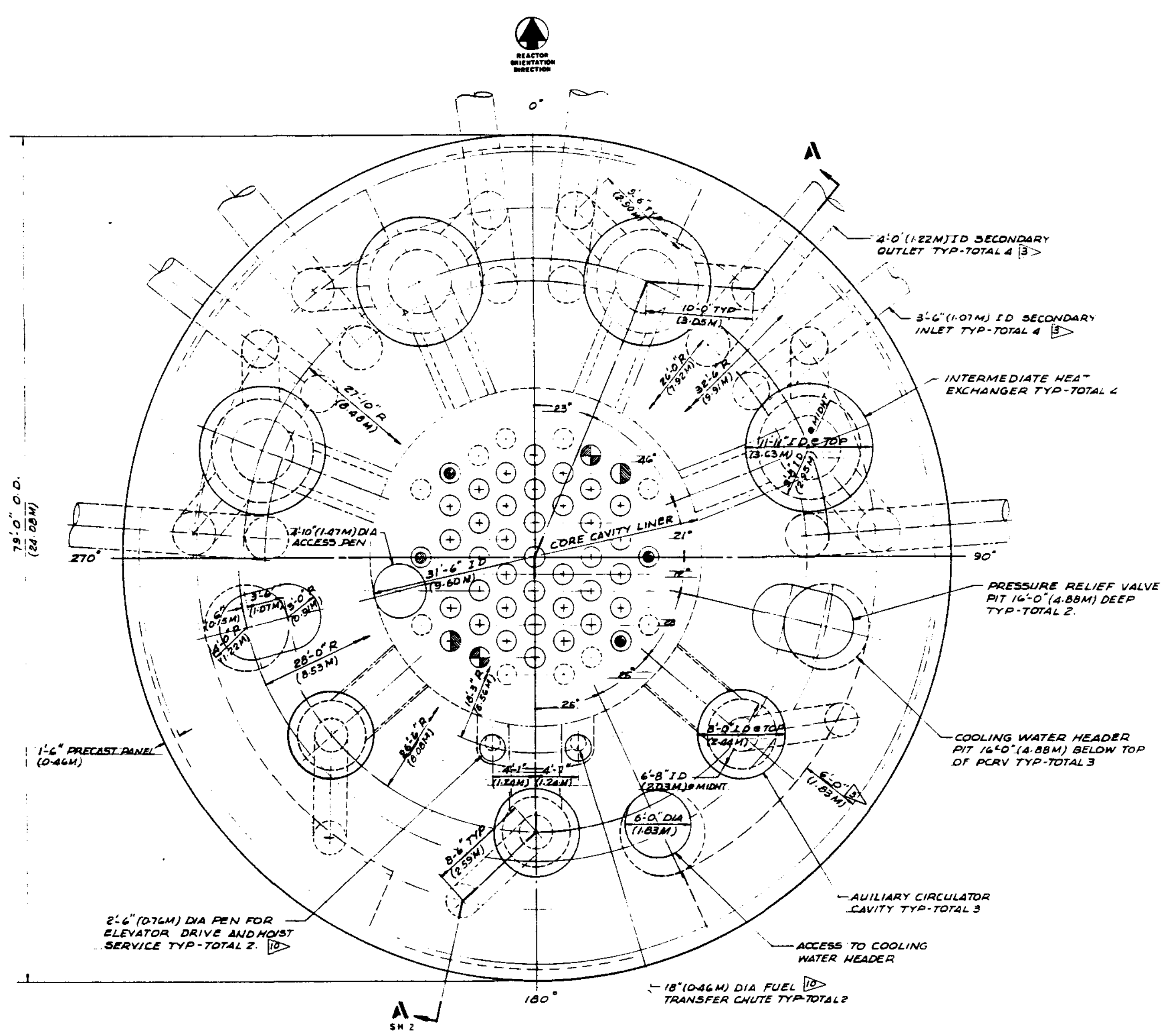
LTR	DATE	ISSUE DESCRIPTION	DR.	CHK.	DES.	APPD	APPD
3		REV. DIMENSIONS					
2		REVISED DIMENSIONS					
1		PRELIMINARY	DET				

TITLE MAIN CIRCULATOR, REFORMER

SIZE	CODE IDENT NO.	DWG. NO	ISSUE
E	32334	025779	3

SCALE 3/4"=1'-0" QAL SHEET 1 OF

025779 /3



PLAN - PCRV TOP HEAD

LEGEND

- REFUELING PENETRATION - 37
- HIGH TEMPERATURE FILTER - 2
- HIGH TEMPERATURE ADSORBER - 2
- ELEMENT TRANSPORTER PEN - 4
- COLUMN SUPPORT FOR FHM - 10

GENERAL NOTES:

1. THIS PCRV DESIGN IS IN GENERAL COMPLIANCE WITH "CONCRETE REACTOR VESSELS AND CONTAINMENTS," ACI 359, ASME SECTION III DIVISION 2 CODE
2. THIS PCRV GENERAL ARRANGEMENT INCORPORATES THE FOLLOWING DESIGN PARAMETERS:
   A. 6500 PSI (44.82 MPa) CONCRETE FOR PCRV AND PRECAST PANEL
   B. 2500 KIPS TENDONS FOR LINEAR PRESTRESSING
   C. 41.3 KIPS (½" Φ CABLES) STRANDS FOR CIRCUMFERENTIAL PRETRESSING
   D. 776 PSIG (5.35 MPa) MAXIMUM CAVITY PRESSURE
   E. IN-VESSEL REFUELING.
3. DIMENSIONS AND OPENINGS IN THE PCRV SUPPORT STRUCTURE & SECONDARY HELIUM INLET & OUTLET DUCT EXIT ARE SHOWN FOR SUGGESTED ARRANGEMENTS ONLY. FINAL CONFIGURATION AND LOCATION ARE PART OF THE BOP DESIGN.
4. THE FEASIBILITY OF THIS GENERAL ARRANGEMENT IS SUBJECT TO THE FOLLOWING.
   A. CONCRETE LIGAMENTS TO BE VERIFIED BY ANALYSIS.
   B. TENDON INTERFERENCE AND LAYOUT STUDIES.
   C. THE CAVITY ENVELOPES AS PER REFERENCE DRAWINGS:
      IHX DWG NO. 025786/1
      MAIN CIRC DWG NO. 025779/3
      AUX CIRC DWG NO. 026217/1
      CAHE DWG NO. 025791/1
      IN-VESSEL REFUELING DWG NO. 025499
5. LINER MATERIALS ARE CARBON STEEL EXCEPT AS NOTED:
   - PLATE TO BE SA-537, CLASS 2 } FOR THICKNESSES > 1
   - FORGINGS TO BE SA-508, CLASS 2A } $RT_{NDT} \leq -30°F$
   WHERE MATERIAL IS INDICATED AS 2¼ CHROME:
   - PLATE TO BE SA-384, GR 22, CLASS 1
   - FORGINGS TO BE SA-336, GR 22
   WHERE MATERIAL IS INDICATED AS ALLOY 800H:
   - PLATE TO BE SB-409, ALLOY 800H
   - FORGINGS TO BE SB-564, CONDITION 2.
6. LINER THICKNESS TO BE ½" (0.01M) EXCEPT WHERE NOTED OTHERWISE.
7. COOLING TUBE PITCH TO BE 5¾" (0.15M) EXCEPT WHERE NOTED OTHERWISE.
8. ALL LINER STUDS TO BE ¾" (0.02M) DIA x LENGTH SHOWN ON DRAWING. PROVIDE ONE STUD PER 50 SQ INCH (0.03 SQ M) OF LINER SURFACE WHERE SHOWN ON THE DRAWING.
9. DETAILS AND MINOR PENETRATIONS NOT SHOWN TO BE SIMILAR TO HTGR-SC PCRV.
10. REFER TO ALTERNATE REFUELING STUDY FOR ARRANGEMENT AND COST INFORMATION.
11. EMBEDDED COOLING AND INTERCONNECTING PIPING TO CONSIST OF 150,000 FEET OF 1¼" (0.03M) SCHEDULE 40 PIPE.

QUANTITIES:

1. CONCRETE (6500 PSI):
   - PCRV — 11,500 CY
   - PRECAST PANELS — 760 CY
2. REINFORCEMENT (ASTM A615 GR 60).
   - PCRV — 435 TONS
   - PRECAST PANELS — 95 TONS
   - NO. OF MECHANICAL SPLICES IN THE PCRV — 2500
3. PRESTRESSING, LPS:
   - NO. OF TENDONS — 160
   - LENGTH OF TENDONS BETWEEN CONCRETE SURFACES* — 92 LF
   - *ALLOW 5 LF ADDITIONAL PER TENDON FOR TENSIONING.
4. PRESTRESSING, CPS:
   - ½" Φ CABLES TOTAL LENGTH — 3.0×10^6 LF
   - NO. OF STRAND LAYERS — 232
   - NO. OF 18" PANELS — 234
   - NO. OF LOAD MONITOR PANELS — 6
5. INSTRUMENTATIONS.
   - NO. OF LPS LOAD CELLS — 8
   - NO. OF CPS LOAD CELLS — 6
   - NO. OF PCRV SENSORS — 300
   - NO. OF LINER SENSORS — 275
   - NO. OF PCRV SPARE SENSORS — 30
   - NO. OF LINER SPARE SENSORS — 30

ITEM	PART NO	DESCRIPTION	MATL/MATL SPEC
REQD/ASSEMBLY	9048 S; DISC S; SYSTEM 11; TYPE DLO; PROJECT 7200	PARTS LIST	

HTGR-R 1170 MW (t)

DIST.

GENERAL ATOMIC COMPANY

UNLESS OTHERWISE SPECIFIED
DIMENSIONS ARE IN INCHES
NO BURRS OR SHARP EDGES
DIMENSIONS PER ANSI-Y-14.5
TOLERANCE:
DECIMAL FRACTION ANGLE
XXX = ± ± ±
XX = ±
ALL MACHINED SURFACES
DO NOT SCALE PRINT

LTR	DATE	ISSUE DESCRIPTION	DR.	CHK	DES	APPD.	APPD
ISSUED 1 / 2		REVISED FOR COST ESTIMATING					
ISSUED 1 / 1		CONCEPTUAL					

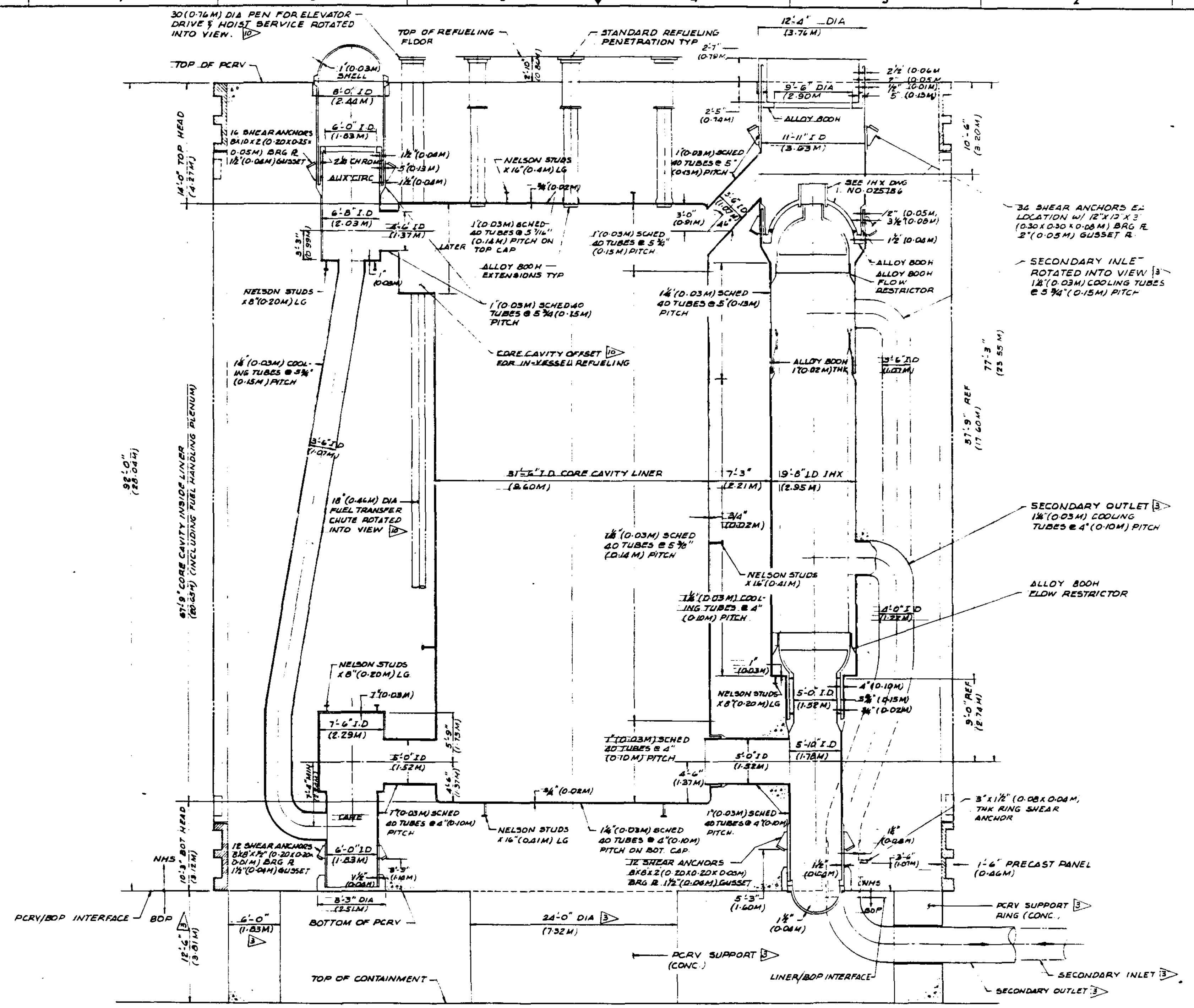
TITLE: PCRV GENERAL ARRANGEMENT 4-LOOP PLANT

SIZE	CODE IDENT NO	DWG. NO	ISSUE
E	32334	026220	1/2

SCALE 1/4"=1'-0" QAL-1 A/L 2 CAT-1 SHEET 1 OF 2

SC-1

026220 1/2



SECTION A-A

ITEM	PART NO	DESCRIPTION	MATL/MATL SPEC

PARTS LIST

HTGR-R 1170 MW(t)

GENERAL ATOMIC COMPANY

UNLESS OTHERWISE SPECIFIED DIMENSIONS ARE IN INCHES NO BURRS OR SHARP EDGES DIMENSIONS PER ANSI Y-14.5 TOLERANCE DECIMAL FRACTION ANGLE ALL MACHINED SURFACES DO NOT SCALE PRINT

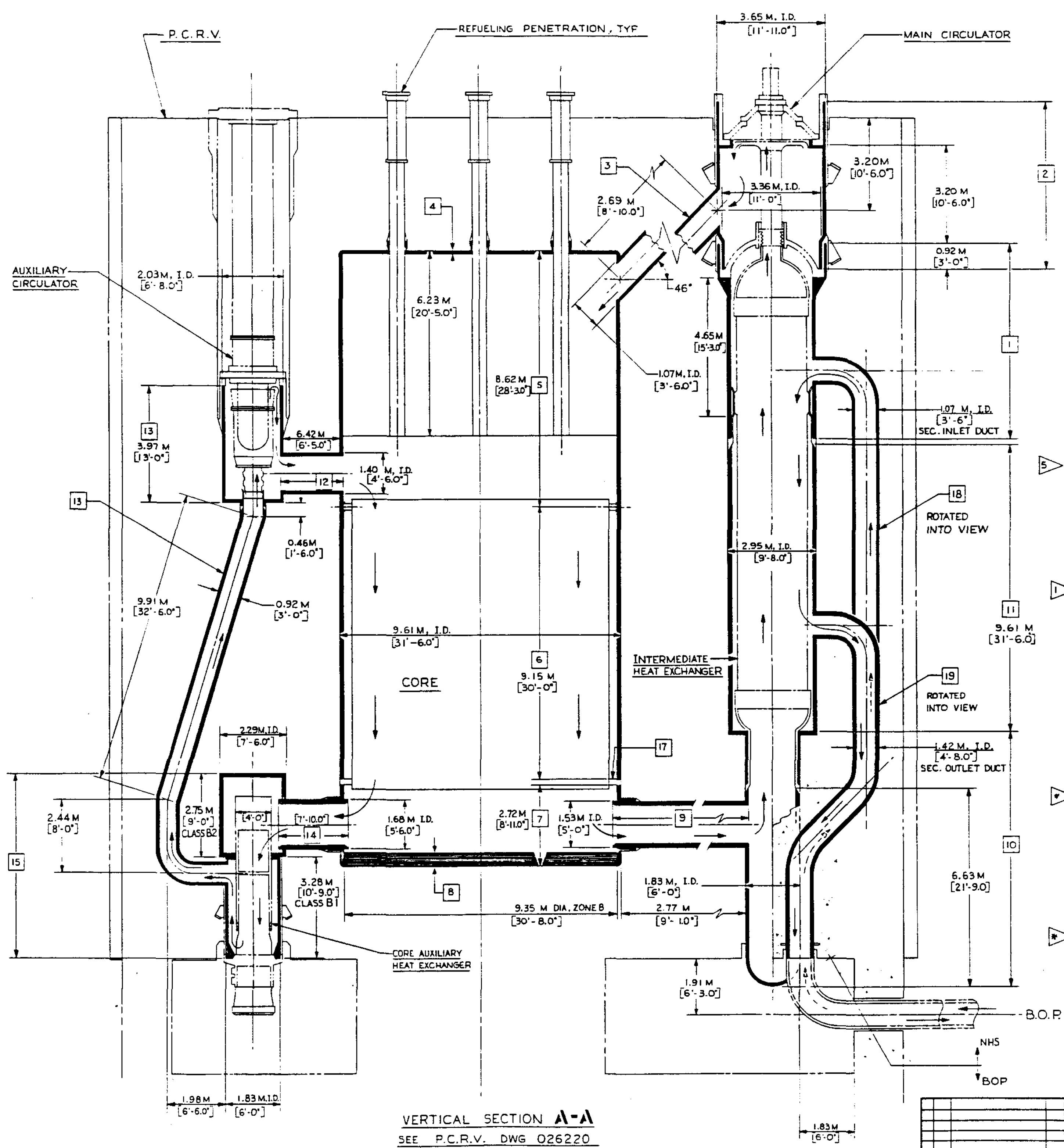
SEE SHEET 1

TITLE PCRV GENERAL ARRANGEMENT 4-LOOP PLANT

SIZE E CODE IDENT NO 32334 DWG NO 026220 ISSUE

SHEET 2 OF 2

026220



VERTICAL SECTION A-A
SEE P.C.R.V. DWG 026220

NOTES:

1. ZONE 9 THERMAL BARRIER ALSO CONSISTS :
   INLET FAIRING.
   INLET OMEGA SEAL.

   THERMAL SHIELD SEGMENT.
   THERMAL SHIELD COUPLING.
2. DIMENSIONS ARE IN METERS, M. OR MILLIMETERS, MM. ENGLISH UNITS SHOWN IN PARENTHESIS, [ ].
3. INSTALLATION PER GA SPEC. NO. 90000B.
4. THERMAL BARRIER THICKNESS DOES NOT INCLUDE ATTACHMENT FIXTURE PROTRUSION.
5. FUEL TRANSFER CHUTE (QTY 2) NOT SHOWN.
6. CLASS B1 : TYPE 316 ST. STL.
   " B2 : IN 713 LC
   " C : (a). GRAPHITE AND SLIP CAST FUSED SILICA FOR FLOOR BLOCKS.
   (b). PYROLYTIC GRAPHITE, ALUMINA AND FUSED SILICA FOR SUPPORT PADS.
   (c). TYPE 316 ST. STL COVERPLATE AND CONTAINER; SAFFIL & KAOWOOL INSUL.

ZONE	CLASS	THICK-NESS (4)	AREA PER CAVITY	NO. OF CAVITIES	AREA PER REACTOR	DRAWING NO.	REMARKS
1	B1	76 MM [3.00]	42.15 M^2 453 FT^2	4	168.6 M^2 1813 FT^2		
2	B1	76 MM [3.00]	52.53 M^2 565 FT^2	4	210.12 M^2 2260 FT^2		
3	B1	76 MM [3.00]	9.03 M^2 97 FT^2	4	36.13 M^2 389 FT^2		
4 (5)	B1	76 MM [3.00]	72.48 M^2 780 FT^2	1	72.48 M^2 780 FT^2		
5 (5)	B1	127 MM [5.00]	252 M^2 2710 FT^2	1	252 M^2 2710 FT^2		
6	B1	127 MM [5.00]	239.3 M^2 2573 FT^2	1	239.3 M^2 2573 FT^2		
7	B2	127 MM [5.00]	68.12 M^2 733 FT^2	1	68.12 M^2 733 FT^2	--	
8	C	469 MM [18.47]	68.7 M^2 739 FT^2	1	68.7 M^2 739 FT^2		
9 (1)	B2	127 MM [5.00]	13.27 M^2 143 FT^2	4	53.08 M^2 570 FT^2		
10	B2	127 MM [5.00]	40.92 M^2 440 FT^2	4	163.67 M^2 1760 FT^2		
11	B2	127 MM [5.00]	87.38 M^2 940 FT^2	4	349.5 M^2 3758 FT^2		
12	B1	76 MM [3.00]	8.55 M^2 92 FT^2	3	25.64 M^2 276 FT^2		
13	B1	76 MM [3.00]	69.2 M^2 744 FT^2	3	207.6 M^2 2233 FT^2		
14	B2	127 MM [5.00]	12.59 M^2 136 FT^2	3	37.76 M^2 406 FT^2		
15	B1	76 MM [3.00]	20.23 M^2 218 FT^2	3	60.7 M^2 653 FT^2		
15 (*)	B2	127 MM [5.00]	24.56 M^2 264 FT^2	3	73.68 M^2 792 FT^2		
17	B2	NO INFORMATION		1			PERIPHERAL SEAL
18	B1	76 MM [3.00]	75.33 M^2 810 FT^2	4	301.32 M^2 3240 FT^2		SECONDARY INLET DUCT
19	B2	127 MM [5.00]	62.26 M^2 670 FT^2	4	249.06 M^2 2678 FT^2		SECONDARY OUTLET DUCT

(*) ZONE 16 IS NOT USED.

AREA SUMMARY : [AREA IS MEASURED AT LINER SURFACE.]

CLASS B1	1574.21 M^2	16 927 FT^2
CLASS B2	994.92 M^2	10698 FT^2
CLASS C	68.73 M^2	739 FT^2
TOTAL	2 637.86 M^2	28 364 FT^2

ITEM	PART NO	DESCRIPTION	MATL/MATL SPEC

REQD/ASSEMBLY
NEXT INDENTURED DOCUMENT 904815
DISC M | SYSTEM 11 | TYPE DLO | PROJECT 7200
PARTS LIST

HTGR – REFORMER, 850°C, 1170 MW (t), 4 LOOP
DIST.

GENERAL ATOMIC COMPANY

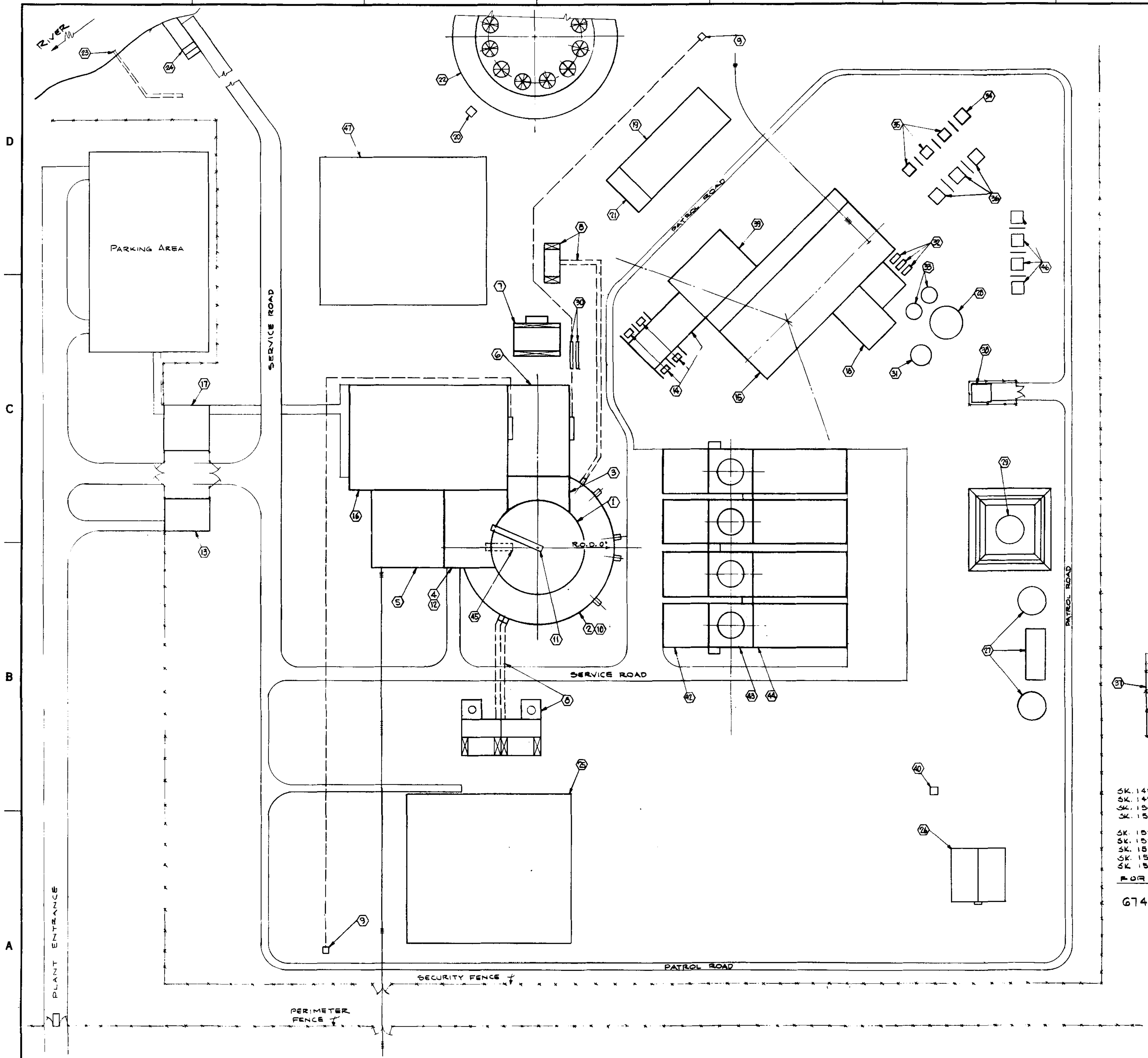
UNLESS OTHERWISE SPECIFIED
DIMENSIONS ARE IN INCHES
NO BURRS OR SHARP EDGES
DIMENSIONS PER ANSI-Y-14.5
TOLERANCE:
DECIMAL FRACTION ANGLE
.XXX = ± ± ±
.XX = ±
ALL MACHINED SURFACES
DO NOT SCALE PRINT

A PRELIMINARY ISSUE
LTR | DATE | ISSUE DESCRIPTION | DR. | CHK. | DES | APPD | APPD

TITLE
THERMAL BARRIER
GENERAL ARRANGEMENT

SIZE E | CODE IDENT NO 32334 | DWG. NO 026227 | ISSUE A
SCALE 1/4" = 1 FT | SHEET 1 OF 1

026227 A SH 1 OF 1



LEGEND

1. REACTOR CONTAINMENT BLDG.
2. CONTAINMENT ANNULUS BLDG.
3. CONTAINMENT PENETRATION BLDG.
4. AUXILIARY REACTOR SERVICE BLDG.
5. FUEL STORAGE BLDG.
6. CONTROL, AUXILIARY & DIESEL GENERATOR BLDG.
7. DIESEL COOLING & FUEL OIL STORAGE BLDG.
8. ULTIMATE HEAT SINK STRUCTURE & TUNNEL
9. CONTROL ROOM EMERG. AIR INTAKE STRUCTURE
10. ENCLOSURE ABV. CONTAINMENT ANNULUS BLDG.
11. PLANT VENT
12. ENCLOSURE ABV. AUXILIARY REACTOR SERVICE BLDG.
13. WAREHOUSE
14. MAIN CIRCULATOR CONTROLLER BLDG. & TRANSFORMER
15. TURBINE BLDG.
16. ADMINISTRATION & SERVICE BLDG.
17. SECURITY BLDG.
18. AUX. BOILER & MAKEUP DEMIN. BLDG.
19. CIRCULATING WATER PUMP HOUSE
20. CHLORINATION BLDG.
21. COOLING TOWER SWITCHGEAR BLDG.
22. COOLING TOWER (WET)
23. COOLING TOWER BLOWDOWN DISCH. PIPE.
24. MAKEUP WATER INTAKE STRUCTURE
25. SECONDARY L.P. HELIUM STORAGE AREA
26. HOLDING POND
27. FIRE PUMP HOUSE & STORAGE TANK
28. CONDENSATE STORAGE TANK
29. FUEL OIL STORAGE TANK & EARTH DIKE
30. H.P. HELIUM STORAGE TANK
31. DEMINERALIZD WATER STORAGE TANK
32. REAGENT STORAGE TANK
33. POLISHER WASTE RECEIVER TANK
34. GSU TRANSFORMER
35. SAT TRANSFORMER
36. RAT TRANSFORMER
37. 500/230 KV SWITCHYARD
38. H_2, CO_2, N_2 STORAGE AREA
39. NON-ESS SWITCHGEAR BLDG.
40. SEWAGE TREATMENT PLANT
41. SETTLING BASIN
42. SECONDARY CIRCULATOR BLDG.
43. REFORMER PCPV BLDG.
44. REFORMING TRAIN BLDG.
45. TEMPORARY FUEL STORAGE
46. UNIT AUXILIARY TRANSFORMERS
47. PRIMARY L.P. HELIUM STORAGE AREA

REFERENCE DRAWINGS

SK. 148 GEN. ARRGT. - R.C.B., C.P.B. AND A.R.S.B. - PLAN AT EL. 87'-4"
SK. 149 GEN. ARRGT. - R.C.B., C.P.B. AND A.R.S.B. - PLAN AT EL. 20'-0" & 22'-0"
SK. 150 GEN. ARRGT. - 131'-0" I.D. CTMT - SECTION "A-A"
SK. 151 GEN. ARRGT. - TURBINE BLDG. - OPERATING/MEZZANINE FLOOR - EL. 50'-0"/22'-0"
SK. 152 GEN. ARRGT. - TURBINE BLDG. - ELEVATION "A-A"
SK. 153 GEN. ARRGT. - REFORMER TRAIN.
SK. 154 GEN. ARRGT. - REFORMER TRAIN - ELEVATION "A-A"
SK. 155 HEAT BALANCE
SK. 156 KEY ONE LINE DIAGRAM UNIT ELECTRICAL DISTRIBUTION

FOR LEGEND SEE SK.-148

6743.052

REV.	DATE	DESCRIPTION			SUPV DISC ENG	PROJ ENG MGR
	6/25/80	FIRST ISSUE	RJ		JBM	

ENGINEER

STATE REG. NO.

NUCLEAR SAFETY RELATED

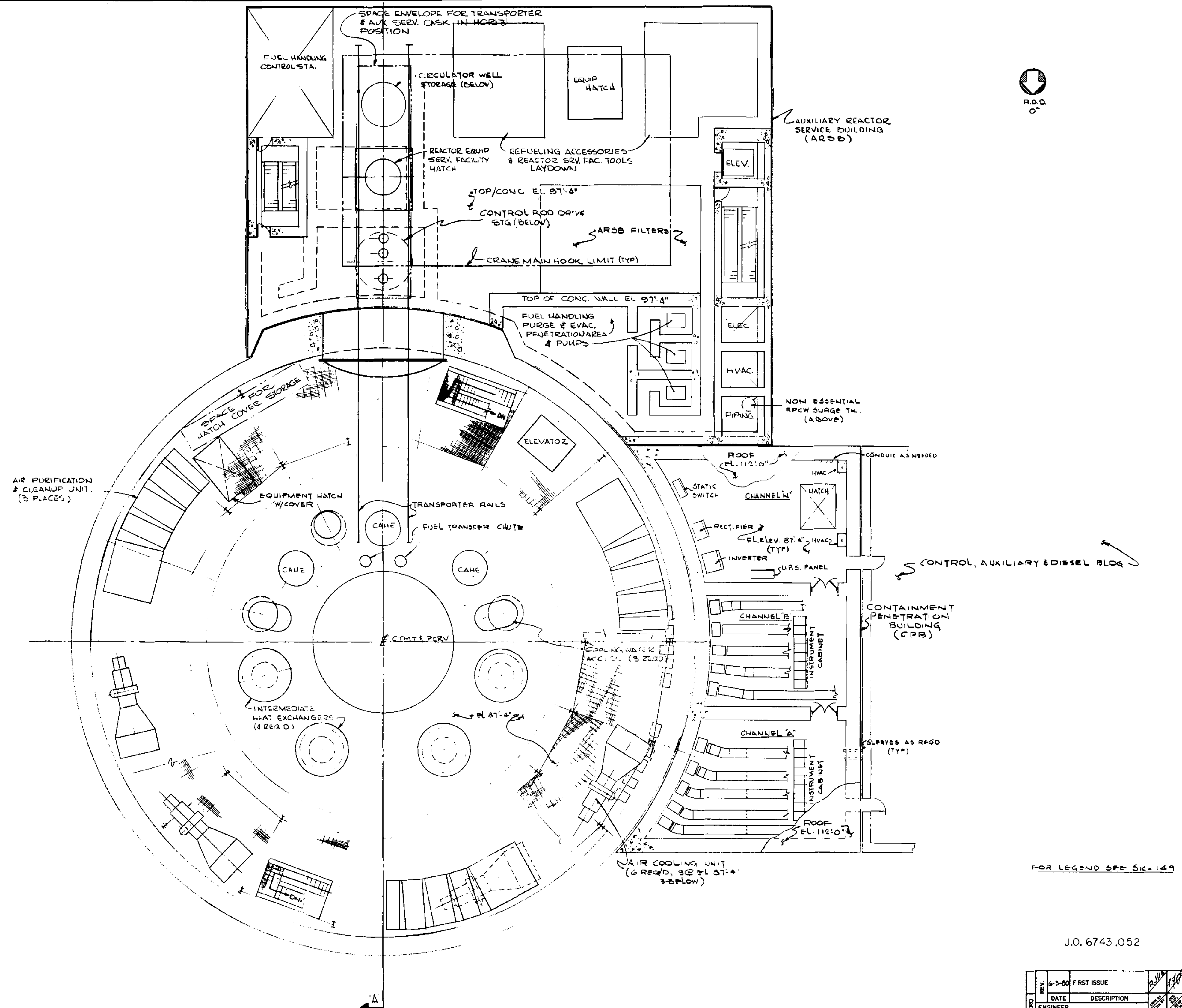
GCRA GAS-COOLED REACTOR ASSOCIATES 1170 MW(t) HTGR-R

PLOT PLAN

united engineers & constructors inc.

SCALE: 1" = 50'-0" DWG. NO. SK-147 REV. NO.

REV. STR ELEC MECH SERV INST NUCL Q.A.E. REV. 0 STR ELEC MECH SERV INST NUCL Q.A.E.



FOR LEGEND SEE SK-149

J.O. 6743.052

REV.	DATE	DESCRIPTION			
	6-3-80	FIRST ISSUE			

MICRO
ENGINEER
STATE REG. NO

NUCLEAR SAFETY RELATED

GCRA GAS-COOLED REACTOR ASSOCIATES 1170 MW(t) HTGR-R

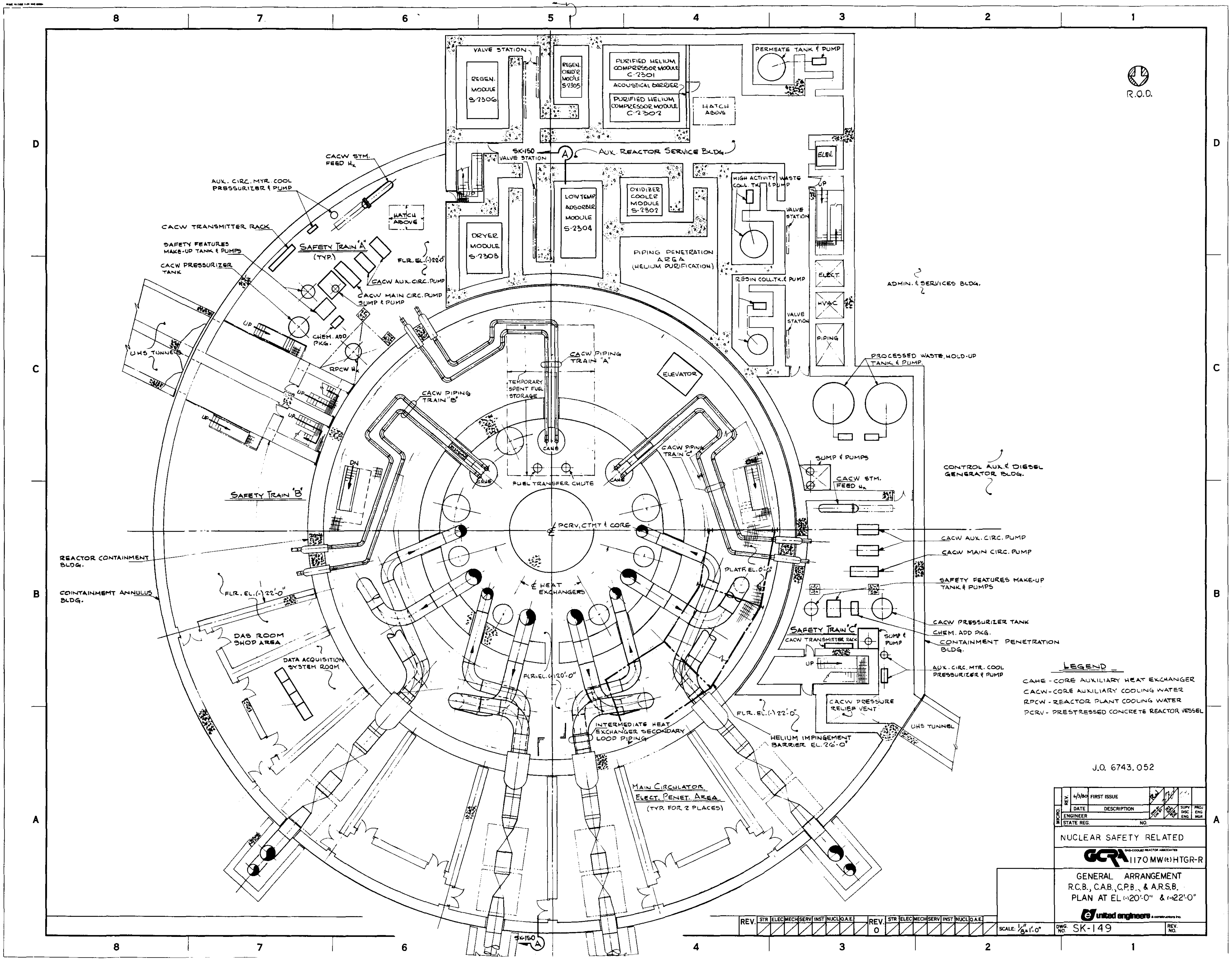
GENERAL ARRANGEMENT
R.C.B., C.P.B. AND A.R.S.B.
PLAN AT EL. 87'-4"

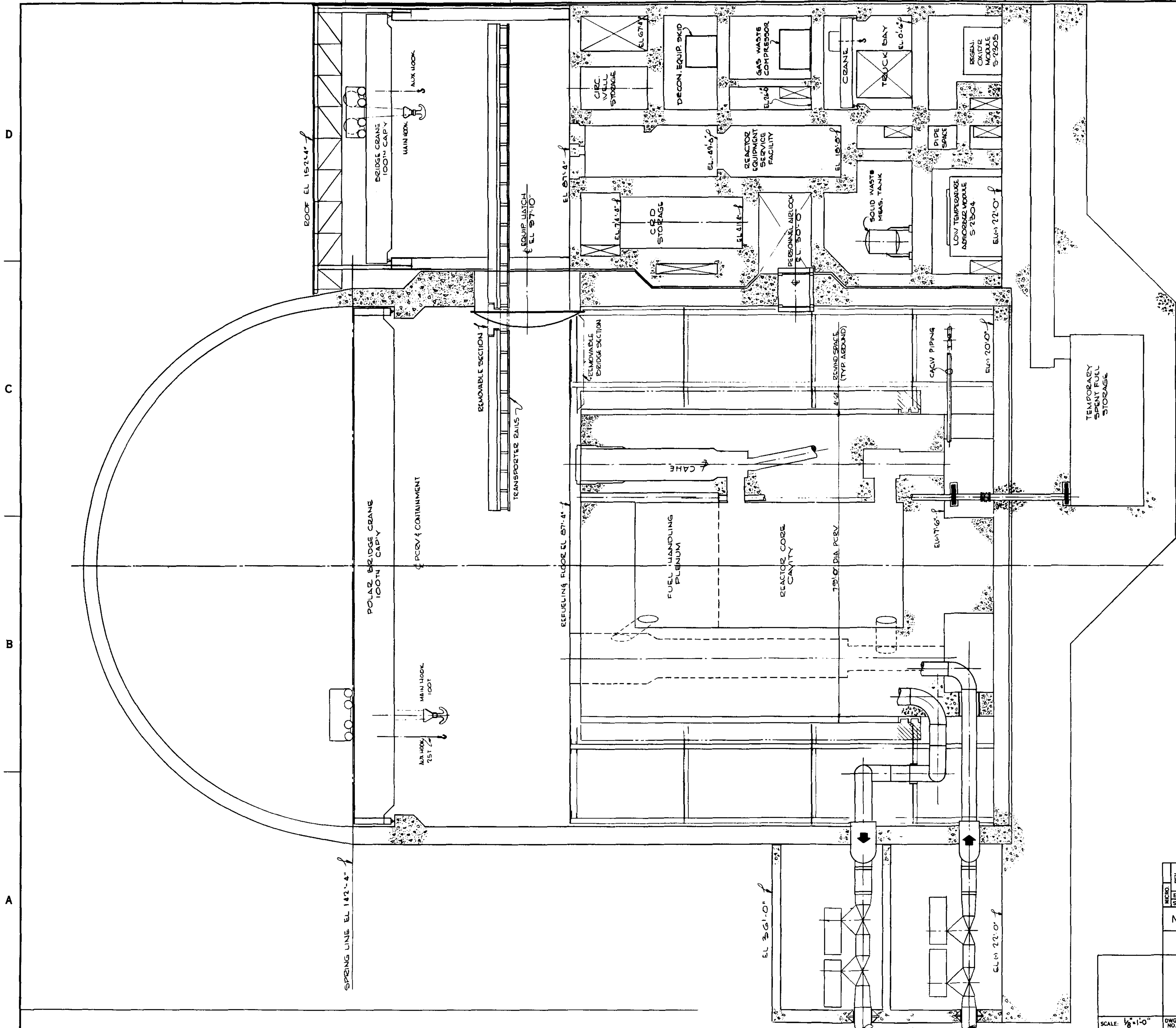
united engineers & constructors inc.

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

DWG. NO. SK-148

REV. NO.


VALVE STATION
REGEN. MODULE S-2306
REGEN. OXID'R MOD'L S-2305
PURIFIED HELIUM COMPRESSOR MODULE C-2301
ACOUSTICAL BARRIER
PURIFIED HELIUM COMPRESSOR MODULE C-2302
HATCH ABOVE
PERMEATE TANK & PUMP
R.O.D.
SK-150 A VALVE STATION
AUX. REACTOR SERVICE BLDG.
ELEV.
LOW TEMP ADSORBER MODULE S-2304
OXIDIZER COOLER MODULE S-2302
DRYER MODULE S-2303
HIGH ACTIVITY WASTE COLL TK & PUMP
VALVE STATION
PIPING PENETRATION AREA (HELIUM PURIFICATION)
RESIN COLL.TK.& PUMP
ELECT.
HVAC
PIPING
ADMIN. & SERVICES BLDG.
CACW STM. FEED Hx
AUX. CIRC. MTR. COOL PRESSURIZER & PUMP
HATCH ABOVE
CACW TRANSMITTER RACK
SAFETY TRAIN 'A' (TYP.)
SAFETY FEATURES MAKE-UP TANK & PUMPS
FLR. EL. (-)22'-0"
CACW PRESSURIZER TANK
CACW AUX. CIRC. PUMP
CACW MAIN CIRC. PUMP SUMP & PUMP
CHEM. ADD PKG.
RPCW Hx
UHS TUNNEL
UP
CACW PIPING TRAIN 'A'
ELEVATOR
TEMPORARY SPENT FUEL STORAGE
CACW PIPING TRAIN 'B'
PROCESSED WASTE HOLD-UP TANK & PUMP
CAHE
FUEL TRANSFER CHUTE
CACW PIPING TRAIN 'C'
SUMP & PUMPS
CONTROL AUX. & DIESEL GENERATOR BLDG.
CACW STM. FEED Hx
DN
SAFETY TRAIN 'B'
℄ PCRV, CTMT & CORE
CACW AUX. CIRC. PUMP
CACW MAIN CIRC. PUMP
REACTOR CONTAINMENT BLDG.
PLATF. EL. 0'-0"
SAFETY FEATURES MAKE-UP TANK & PUMPS
CONTAINMENT ANNULUS BLDG.
FLR. EL. (-)22'-0"
℄ HEAT EXCHANGERS
CACW PRESSURIZER TANK
CHEM. ADD PKG.
SAFETY TRAIN 'C'
CACW TRANSMITTER RACK
SUMP & PUMP
CONTAINMENT PENETRATION BLDG.
DAS ROOM SHOP AREA
DATA ACQUISITION SYSTEM ROOM
FLR. EL. (-)20'-0"
AUX. CIRC. MTR. COOL PRESSURIZER & PUMP
LEGEND
CAHE - CORE AUXILIARY HEAT EXCHANGER
CACW - CORE AUXILIARY COOLING WATER
RPCW - REACTOR PLANT COOLING WATER
PCRV - PRESTRESSED CONCRETE REACTOR VESSEL
CACW PRESSURE RELIEF VENT
FLR. EL. (-)22'-0"
INTERMEDIATE HEAT EXCHANGER SECONDARY LOOP PIPING
HELIUM IMPINGEMENT BARRIER EL. 26'-0"
UHS TUNNEL
J.O. 6743.052
MAIN CIRCULATOR ELECT. PENET. AREA (TYP. FOR 2 PLACES)
SK-150 A
FIRST ISSUE
DATE DESCRIPTION
ENGINEER
STATE REG. NO.
NUCLEAR SAFETY RELATED
GCRA 1170 MW(t) HTGR-R
GENERAL ARRANGEMENT
R.C.B., C.A.B., C.P.B., & A.R.S.B.
PLAN AT EL (-)20'-0" & (-)22'-0"
united engineers & constructors inc.
SCALE: 1/8"=1'-0"
DWG. NO. SK-149
REV. NO.
REV. STR ELEC MECH SERV INST NUCL Q.A.E.
1 2 3 4 5 6 7 8
A B C D



J.O 6743.052

REV.	DATE	DESCRIPTION
	6-3-80	FIRST ISSUE

ENGINEER

STATE REG. NO.

NUCLEAR SAFETY RELATED

GCRA GAS-COOLED REACTOR ASSOCIATES 1170 MW(t) HTGR-R

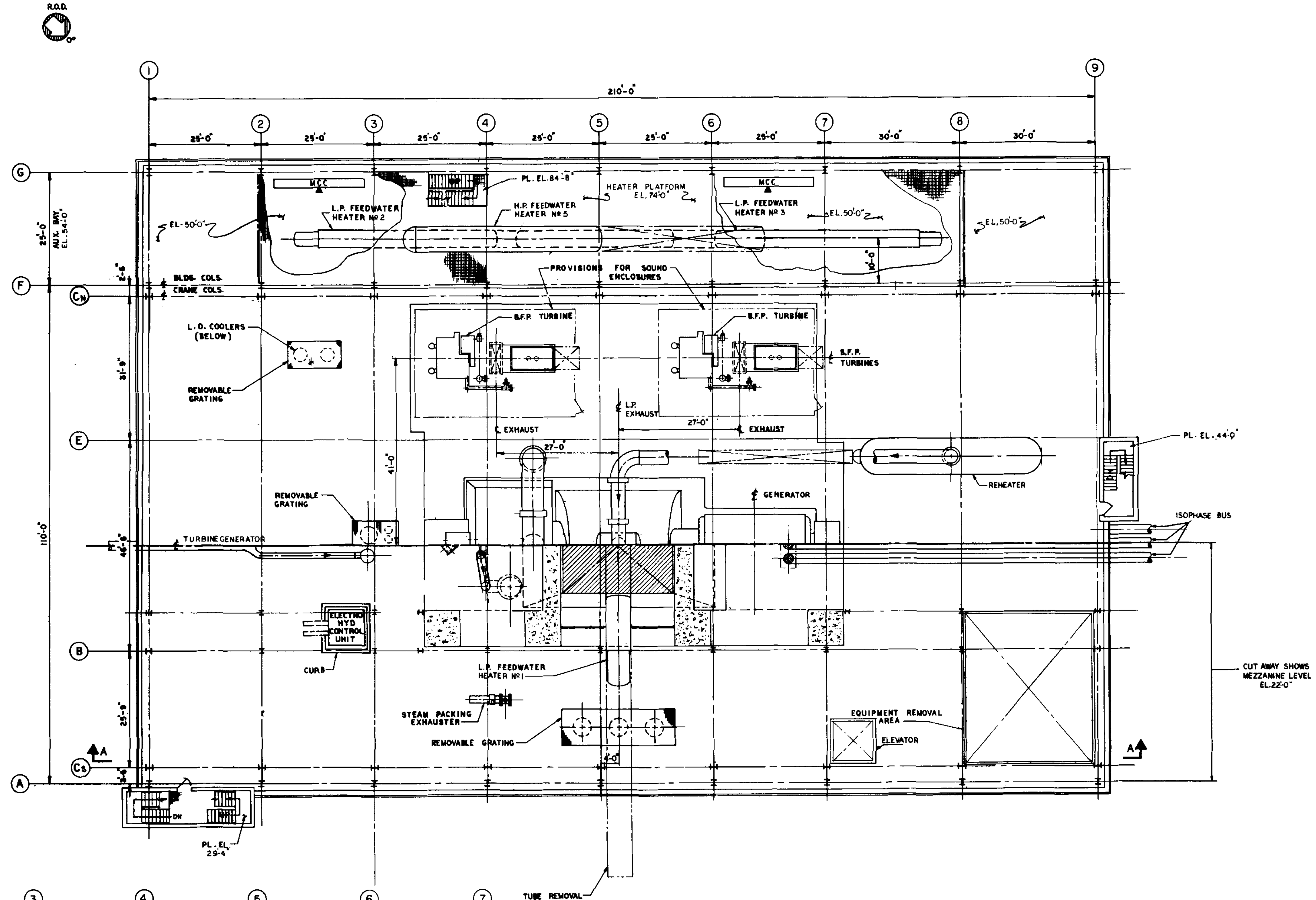
GENERAL ARRANGEMENT
131'-0" I.D. CTMT.
SECTION "A-A"

united engineers & constructors inc.

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

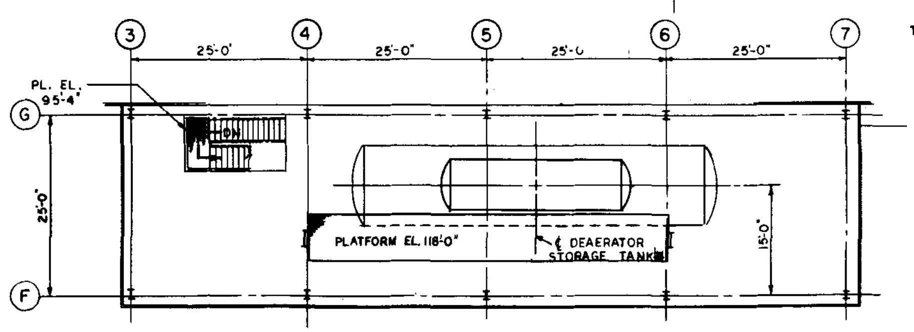
DWG NO. SK-150

REV. NO.



OPERATING FLOOR-EL. 50'-0"

MEZZANINE FLOOR-EL. 22'-0"



PLAN EL. 106'-0"

J.O. 6743.052

REV.	DATE	DESCRIPTION
	7/9/80	FIRST ISSUE

ENGINEER
STATE REG. NO.

NOT NUCLEAR SAFETY RELATED

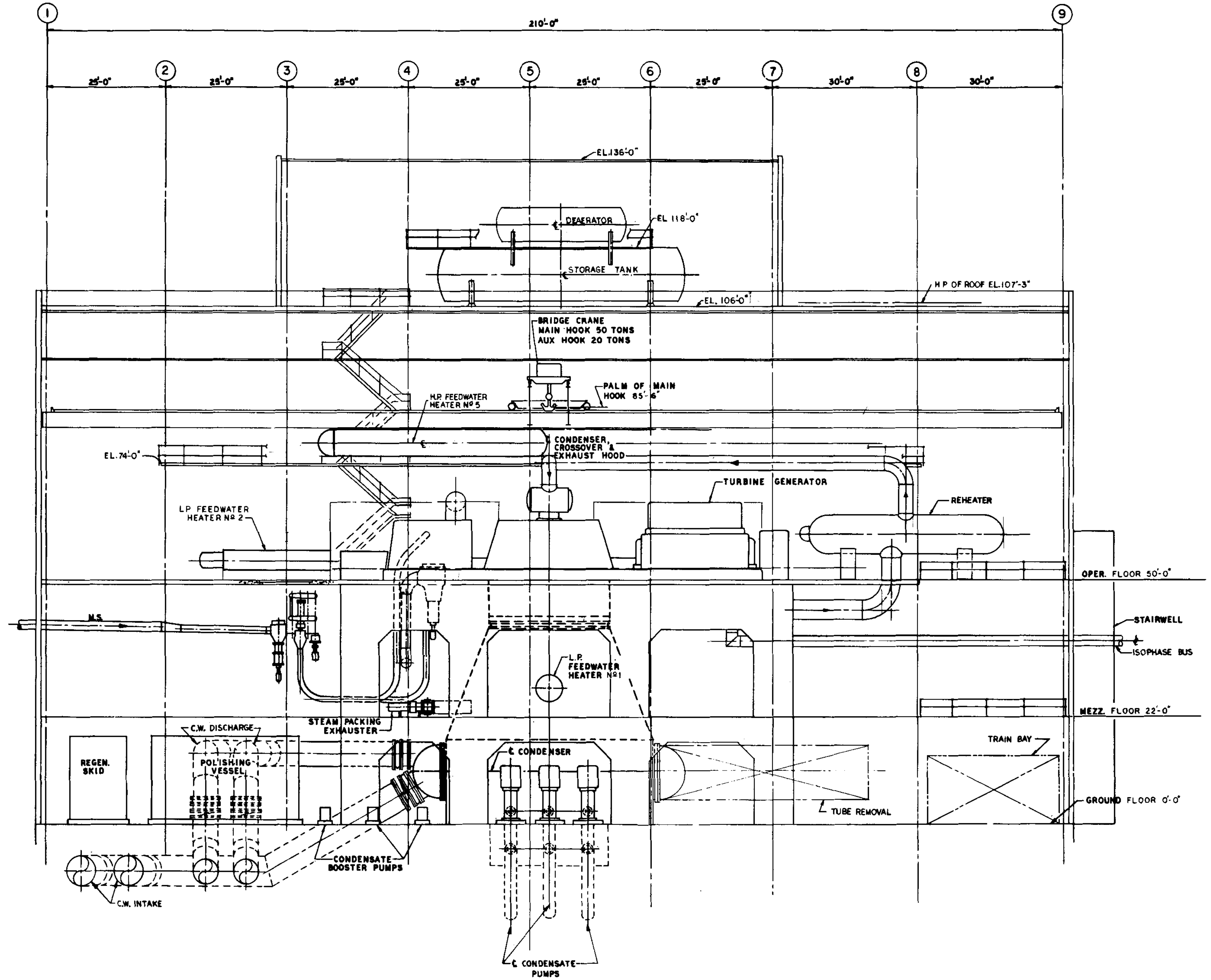
GCRA 1170 MW(e) HTGR-R

GENERAL ARRANGEMENT
TURBINE BUILDING
OPERATING/MEZZANINE FLOOR
EL. 50'-0"/22'-0"

united engineers & constructors inc.

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

DWG. NO. SK.-151



ELEVATION
"A-A"

J.O. 6743.052

REV.	DATE	DESCRIPTION
		FIRST ISSUE

ENGINEER
STATE REG. NO.

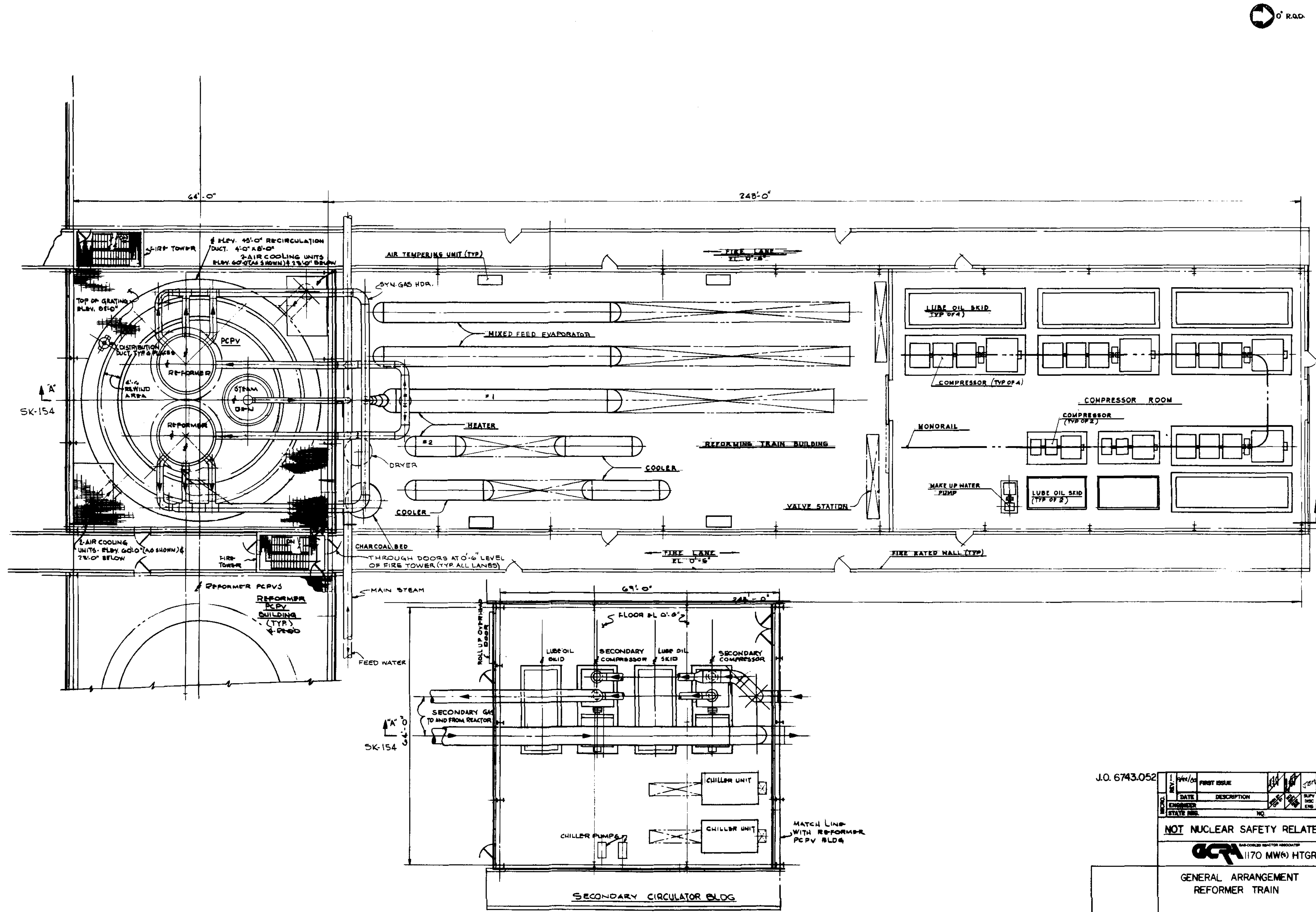
NOT NUCLEAR SAFETY RELATED

GCRA 1170 MW(e) HTGR-R

GENERAL ARRANGEMENT
TURBINE BUILDING
ELEVATION "A-A"

united engineers

SCALE: 1/8"=1'-0"
DWG. NO. SK.-152
REV. NO.



J.O. 6743.052

REV.	DATE	DESCRIPTION
1		FIRST ISSUE

NOT NUCLEAR SAFETY RELATED

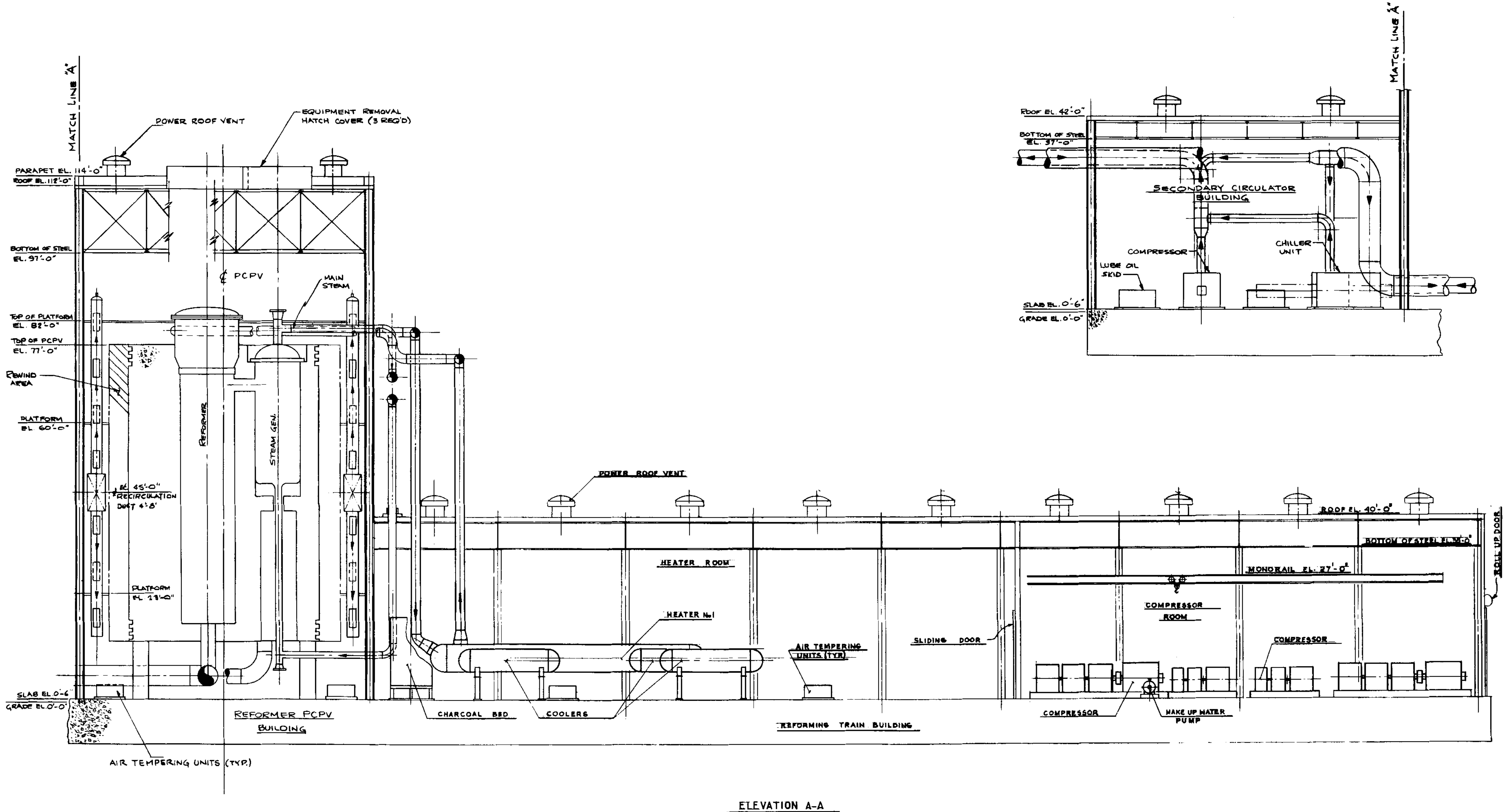
GCRA 1170 MW(e) HTGR-R

GENERAL ARRANGEMENT
REFORMER TRAIN

united engineers

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

DWG. NO. SK-153 REV. NO. 1



ELEVATION A-A

J.O.-6743.052

REV.	DATE	DESCRIPTION				
1	9/22/80	FIRST ISSUE			JBM	
			DSGN BY / CHK BY		SUPV DISC ENG	PROJ ENG MGR

MICRO.
ENGINEER
STATE REG. NO.

NOT NUCLEAR SAFETY RELATED

GCRA GAS-COOLED REACTOR ASSOCIATES 1170 MW(e) HTGR-R

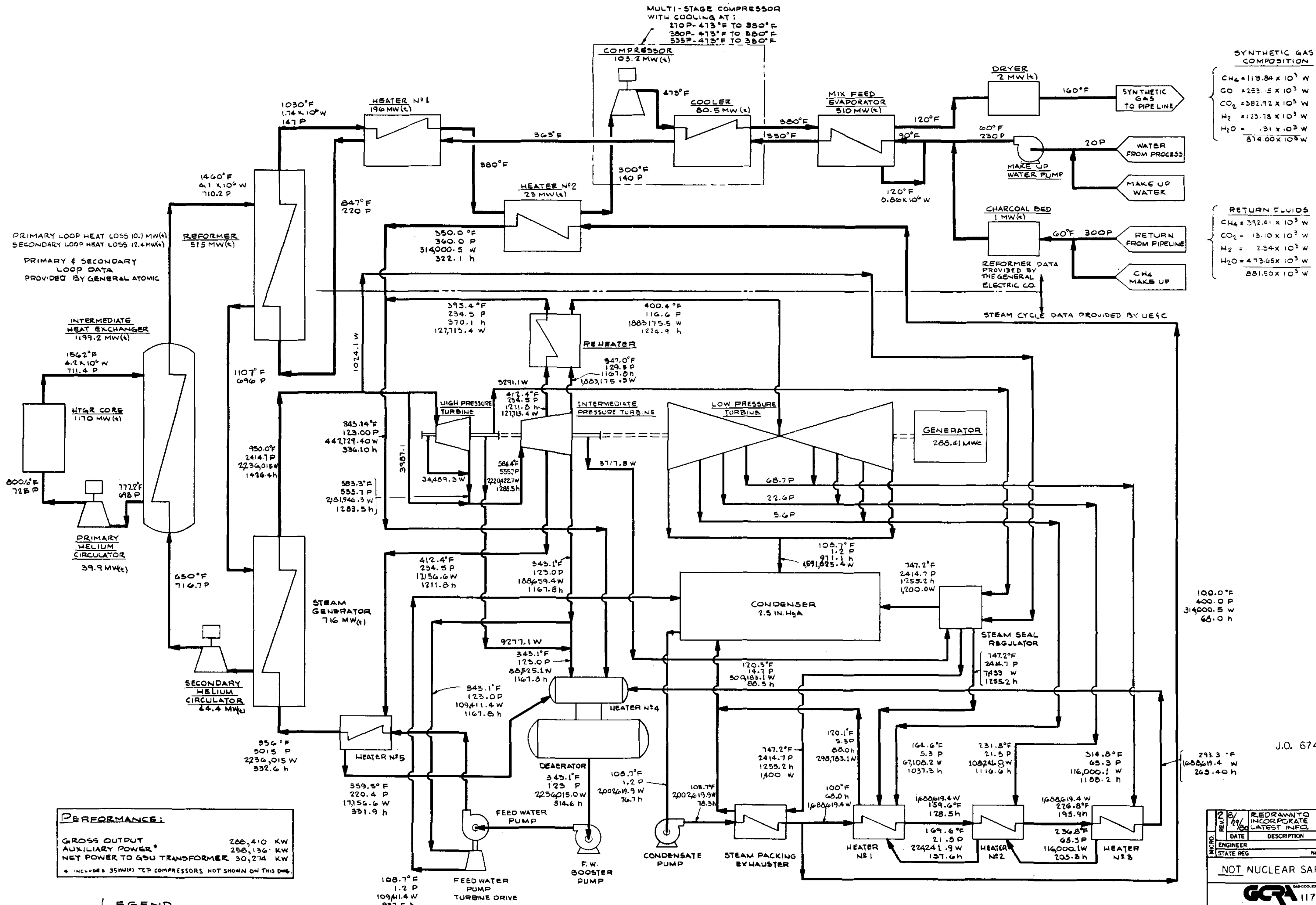
GENERAL ARRANGEMENT
REFORMER TRAIN
ELEVATION A-A

united engineers & constructors inc.

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

DWG. NO. SK-154

REV. NO. 1



PERFORMANCE:

GROSS OUTPUT	288,410 KW
AUXILIARY POWER*	258,136 KW
NET POWER TO GSU TRANSFORMER	30,274 KW

* INCLUDES 35MW(e) TCP COMPRESSORS NOT SHOWN ON THIS DWG.

LEGEND

F =	TEMPERATURE	°F
P =	PRESSURE	PSIA
W =	FLOW	LBS/HR.
h =	ENTHALPY	BTU/LB.

J.O. 6743.052

REV.	DATE	DESCRIPTION
2	8/29/80	REDRAWN TO INCORPORATE LATEST INFO.

ENGINEER

STATE REG. NO.

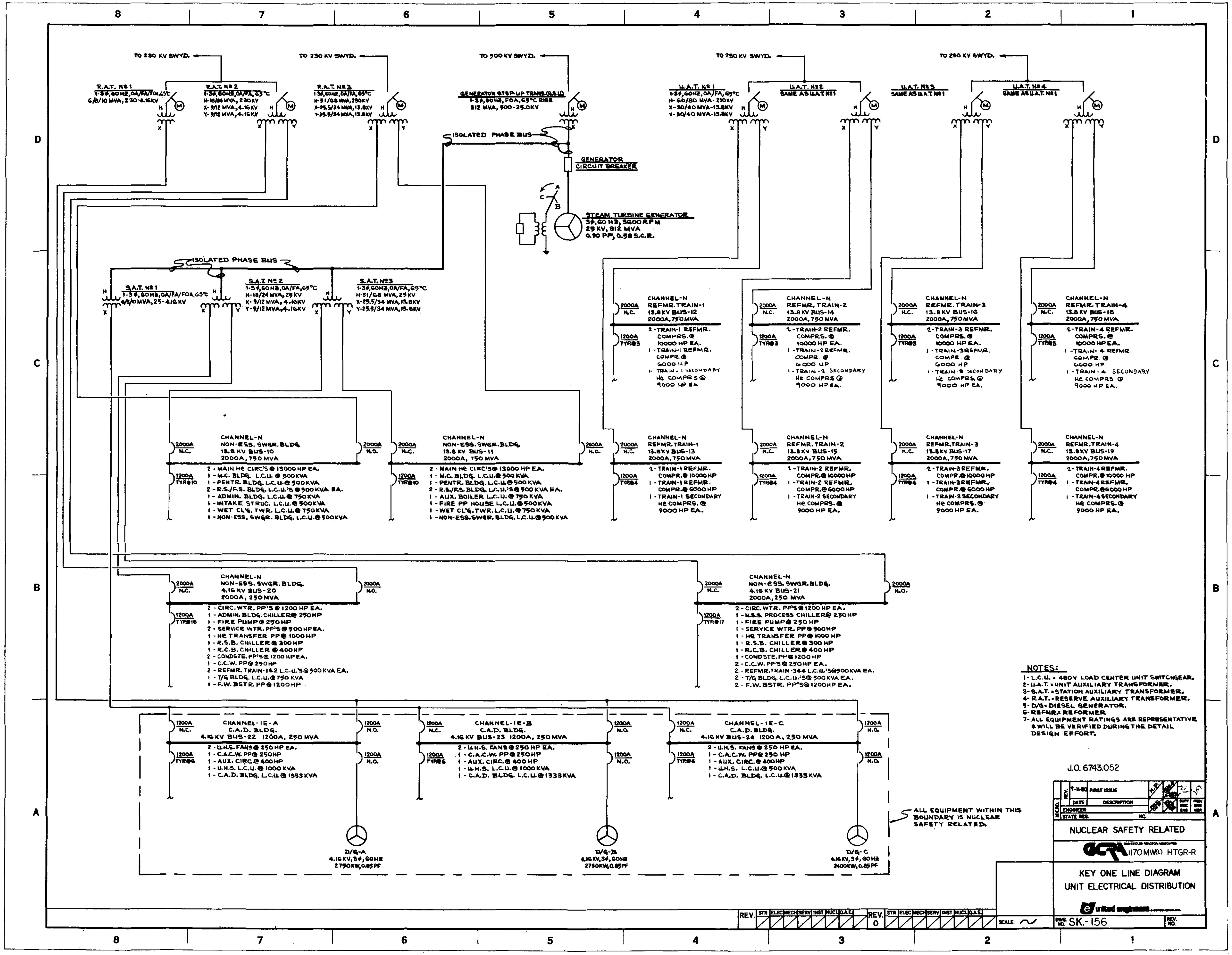
NOT NUCLEAR SAFETY RELATED

GCRA GAS-COOLED REACTOR ASSOCIATES 1170 MW(t) HTGR-R

HEAT BALANCE DIAGRAM

united engineers & constructors inc.

DWG. NO. SK-155 REV. NO.



TO 230 KV SWYD.
TO 500 KV SWYD.
R.A.T. NO 1
R.A.T. NO 2
R.A.T. NO 3
GENERATOR STEP-UP TRANS.(G.S.U.)
ISOLATED PHASE BUS
GENERATOR CIRCUIT BREAKER
STEAM TURBINE GENERATOR
U.A.T. NO 1
U.A.T. NO 2
U.A.T. NO 3
U.A.T. NO 4
SAME AS U.A.T. NO 1
S.A.T. NO 1
S.A.T. NO 2
S.A.T. NO 3
CHANNEL-N REFMR. TRAIN-1
CHANNEL-N REFMR. TRAIN-2
CHANNEL-N REFMR. TRAIN-3
CHANNEL-N REFMR. TRAIN-4
CHANNEL-N NON-ESS. SWGR. BLDG.
CHANNEL-1E-A C.A.D. BLDG.
CHANNEL-1E-B C.A.D. BLDG.
CHANNEL-1E-C C.A.D. BLDG.
D/G-A
D/G-B
D/G-C
ALL EQUIPMENT WITHIN THIS BOUNDARY IS NUCLEAR SAFETY RELATED.
NOTES:
1- L.C.U. = 480V LOAD CENTER UNIT SWITCHGEAR.
2- U.A.T. = UNIT AUXILIARY TRANSFORMER.
3- S.A.T. = STATION AUXILIARY TRANSFORMER.
4- R.A.T. = RESERVE AUXILIARY TRANSFORMER.
5- D/G = DIESEL GENERATOR.
6- REFMR. = REFORMER
7- ALL EQUIPMENT RATINGS ARE REPRESENTATIVE & WILL BE VERIFIED DURING THE DETAIL DESIGN EFFORT.
J.O. 6743.052
9-11-80 FIRST ISSUE
NUCLEAR SAFETY RELATED
GCRA 1170MW(t) HTGR-R
KEY ONE LINE DIAGRAM
UNIT ELECTRICAL DISTRIBUTION
united engineers
SCALE: ~
DWG. NO. SK.-156