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MODULAR HTGR TEST PROJECT DEFINITION STUDY FINAL REPORT

APPLIED TECHNOLOGY

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The Study has drawn to the maximum extent possible upon this design development. In particular, experience has been drawn from Bechtel National, Inc., Combustion Engineering, Inc., GA Technologies Inc., General Electric Company, and Stone & Webster Engineering Corporation, the vendor/supplier/architect engineer (AE) participants in the U.S. HTGR Program.

For the purposes of obtaining site-specific cost and schedule data, the study was carried out for two representative sites:

- Site No. 1 is Tennessee Valley Authority's Widows Creek Steam Plant in northeastern Alabama. At this site, the nuclear island of the one-reactor module would be constructed and coupled to the existing turbine generator on Unit 1 of that coal-fired station.

- Site No. 2 is the site previously designated for the New Production Reactor (NPR) on the Idaho National Engineering Laboratory (INEL). At this site, a complete one-reactor module plant would be constructed, including a turbine generator, and electrical transmission and distribution systems.

SECTION 1

SUMMARY AND ASSESSMENT

1.1 INTRODUCTION

This report documents a study of the scope, schedule, costs, benefits and limitations associated with a Modular High Temperature Gas-Cooled Reactor (MHTGR) Test Project. Such a Project is viewed as the culmination of the MHTGR development effort after which the subsequent commercial MHTGRs would be deployed as a high temperature steam supply for electric generation or process steam cogeneration. The report serves as a necessary input to the ongoing Project strategy development effort within the U.S. HTGR Program. In particular, the report serves as the basis for GCRA's Project Strategy Plan that is under development for the MHTGR.

The Study to produce this report reflects an initiative begun by the Tennessee Valley Authority (TVA) and expanded through Gas-Cooled Reactor Associates (GCRA) to its present scope. Study contractors, who are also participants in the U.S. HTGR Program, contributed in the overall effort through cost sharing of their respective workscopes. EG&G, the operating contractor at the Idaho National Engineering Laboratory (INEL), has supported the Study by making available site specific information on the Idaho National Engineering Laboratory (INEL) reservation.

1.2 BACKGROUND

From its inception, GCRA has sought to establish a National HTGR Project that could substantiate the HTGR as an attractive commercial energy supply alternative for the U.S. utility/user industry. In the earlier efforts to establish a commitment to a Project for the 2240 MWt Lead Plant, a Project Strategy Plan (Reference 1) evolved and provided a generally accepted framework for the overall HTGR Program.

With the adoption of the MHTGR as the reference design concept, and recognizing the changes and constraints within the U.S. HTGR Program, a fresh approach to HTGR Project development was necessary. A brief study (Reference 2) was made of two possible development strategies, a one-reactor test project, and a two-reactor commercial demonstration approach.

In early 1985, the GCRA Project Strategy Subcommittee was formed from its utility members to address the key issues associated with the development and execution of the first strategy, an MHTGR Test Project. The outcome of the Subcommittee deliberations were Utility Perspectives and Guidelines for Modular HTGR Development (Reference 3). This provided an initial statement of the Project objectives and established guidelines for the planning and execution of the Project.

In summary, it appeared feasible with the MHTGR to construct a single module plant for a realizable funding that could be jointly raised by the private sector and the federal government. Furthermore, the design inherent safety margins permit a new dimension of performance testing that would demonstrate the MHTGR's passive investment protection and safety features. This demonstration evidence is deemed to be invaluable for utility/investor and public acceptance as well as providing potential support for the design certification of subsequent commercial plants.

In mid-1985, the need for the Test Project Definition Study, reported herein, was recognized within GCRA. When EPRI declined a proposal to conduct such a study, TVA took the initiative to support it. After joint development of the workscope, the current study was established by GCRA and TVA. In addition, complementary resources have been provided by Arizona Public Service Company, Public Service Electric and Gas Company, Public Service Company of Colorado, and The New York Power Authority.

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1.3 THE NEED FOR TESTING

In planning the development of the MHTGR Project definition effort, the question of the relative need for and benefits of a prototype test unit has been a central issue. This section explains why GCRA and the Study participants believe that such testing is invaluable to the commercialization of the HTGR.

1.3.1 Prototype Testing

Field testing of a new product is a generally accepted prerequisite to commercial deployment whenever it is practical. Even though the automobile industry has manufactured millions of vehicles, and model changes are often minimal, road testing of new models is the general rule. This continues to be the case even though design analysis methods have reached unprecedented levels of sophistication. Even with this combination of analysis and testing, generic problems with basic systems such as brakes and transmissions continue to surface in the commercial products. Similarly, a number of costly examples can be gleaned from past experience with pressurized water reactors (PWRs), boiling water reactors (BWRs), heavy water reactors (HWRs), and gas cooled reactors (GCRs), where generic problems have been identified after deployment of significant numbers of units.

For prototype testing, as the cost of a product increases, and the anticipated number to be deployed decreases, the cost/benefit ratio for testing becomes less favorable. Clearly, in the case of a nuclear power plant, it is not practical to consider a series of destructive tests of prototypes to prove out a design. While prototype testing can never remove all risk from subsequent deployment, experience shows that the possibility and optimum extent of prototype testing of advanced nuclear power reactors should be given serious consideration.

In the past, nuclear plant design uncertainties have been accommodated through the use of conservative margins and analytical methods. The validity of this approach from the standpoint of public safety has been verified by the fact that although significant differences from predicted performance have been experienced in the operation of light water reactors (LWRs), actual injury to the general public has been maintained at unprecedented low levels.

Unfortunately, even for large LWRs the capital and operating costs and complexities arising from the implementation of these design conservatisms are challenging their economic viability. In addition, from an overall standpoint, what appears to be a conservative design feature for one concern may impose significant additional risks in other areas. In general, added system complexity can greatly increase the difficulty of operation and maintenance and detract from activities directed toward stable normal operation. Future LWR design will address these difficulties by incorporating the large body of operating experience being generated by currently operating plants. The amount and applicability of operating experience for the MHTGR is at a much lower level.

There is no doubt that a MHTGR could be designed and licensed without prototype testing to demonstrate reliability or response characteristics for low probability events. Where uncertainties exist or are perceived to exist, additional design conservatism and engineered safety systems can be added to address specific issues under consideration. However, the smaller MHTGR faces economic challenges. Added capital and operating costs as a result of dealing with uncertainties can destroy its economic viability, and its simplicity.

The primary feature that could allow the MHTGR to overcome the economic penalties associated with its size and one of its major assets is its inherent response characteristics that limit the consequences of even low probability events. If these characteristics can be established with certainty, the savings in construction and operating costs associated with minimizing the need for engineered safety systems and other safety related support systems may substantially offset the disadvantages associated with economies of scale. From the results of the design effort to date, it would appear that prototype testing may be quite practical. Depending on the resolution of future design and licensing issues, it may be essential if the concept is to maintain the degree of design simplicity required to make it economically viable.

1.3.2 Test Objectives

For testing in the normal mode, an important objective of a well conceived program is to effectively compress the time scale, so that the equivalent of a much longer period of operation is obtained. As a result, the confidence in the long-term performance of subsequent commercial units is substantially increased. This objective is addressed by formulating a program to evaluate normal operation, anticipated operating transients, and expected maintenance activities.

It is reasonable to expect that design improvements will be identified as a result of the experience gained during the construction and operation of the prototype. To the extent that these improvements may modify the results and conclusions of the testing, it may be necessary to incorporate the modifications into the prototype and repeat some of the testing. While this process will extend the test program, it should be considered a significant benefit since the potential for backfit requirements in subsequent commercial units should be reduced.

Testing involving the simulation of low probability events has a different objective. Nuclear power generation is a capital intensive technology, and recent history indicates a potentially high risk of having to shut the plant down for an extended period before reaching its design life. As a result, the perceived financial risk is a major issue and a potential barrier to a successful commercial market for the MHTGR. The majority of the existing nuclear power plants which have been shut down were not shut down as the direct result of consequences of an event at that plant, but rather due to regulatory actions stemming from a perception of unacceptable risk to the public.

For the reasons discussed above, the issues of investment protection and licensing are considered too interrelated to address separately. Public perceptions that a nuclear power plant is too dangerous, fueled by activities such as annual emergency drills, are reflected in the actions of the regulatory agencies. The political benefits of a successful test program that addresses events associated with safety and licensing could be quite effective in reducing the financial risk perceived by utilities.

1.3.3 Test Benefits

The benefits of a successful prototype testing program are discussed below. In general, the benefits are interrelated and contribute to the overall benefit of an economically competitive, broadly acceptable option for electricity generation (and later process heat). These benefits are:

- Confirm Analysis Margins - The uncertainties in the predicted response characteristics will be better understood, and there will be a reduced tendency to add more margins or features as the design evolves.

- Support Design Optimization - By compressing operating experience through a well conceived testing program, a more refined, optimized design for the commercial units can be expected.

- Support Design Certification - Recognizing the limited HTGR experience that can support the design certification process by the Nuclear Regulatory Commission, a well conceived test program, in concert with analysis and separate effects testing, may be a significant support to design certification. This support will take two forms: First, the demonstration of reliably operating hardware that will support the design predictions for accident probability calculations. Second, the ability to demonstrate, if necessary, the inherent response to certain accidents, thus reducing uncertainties in the design predictions for accident consequences.

- Increase Public Confidence - By demonstrating industry confidence in the concept through the conduct of a prototype test program, increased confidence of the general public can be expected.

- Minimize Owner's Risk for Utilities and the Financial Community - Increased confidence of the industry and the general public along with licensing simplification will reduce the utility's financial risk of purchasing a MHTGR. A better understanding of the plant's inherent response to accidents will help to quantify the investment risk and to reduce the cost of insuring the plant.

- Improve Cost Effectiveness - All of the benefits discussed above will contribute to the reduction of capital and operating costs for a commercial MHTGR.

For the above reasons, GCRA and the Study participants believe that the prpoosed testing is invaluable to the commercialization of the MHTGR.

1.4 MODULAR HTGR DESIGN DESCRIPTION

In the reference plant design, developed by the U.S. HTGR Program, four reactor modules and two turbine generator sets are used to achieve the 558 MWe plant rating. Each reactor module is housed in a vertical cylindrical concrete enclosure or silo that is fully embedded in the earth. The nuclear island (NI) portion of the plant, consisting of four reactor silos and adjacent structures that house fuel handling, helium processing, and other reactor service systems, has been kept to a minimum size and complexity. A conventional secondary containment is not required. A common nonsafety-related control room is used to operate all four reactors and the turbine plant.

The Project comprises one of the four reactor modules in the reference plant, with the required reactor service systems, and the turbine generator in the case of the full plant remote government site.

In the reference MHTGR configuration, each reactor module consists of separate reactor and steam generator steel vessels connected by a horizontal coaxial crossduct (Figure 1-1). The core, graphite reflector and metallic core support structure are installed in the reactor vessel. The reactor is graphite-moderated, helium-cooled, and uses prismatic ceramic fuel in the form of hexagonal blocks. The active core occupies an annular region surrounded by inner and outer graphite reflector elements. Gravity-assisted control rod mechanisms operate control rods in the inner and outer reflector elements. A reserve shutdown system is provided in the innermost region of the active core.

The ceramic fuel comprises low enriched uranium (LEU) and thorium fuel particles with silicon carbide coatings distributed in a graphite matrix in rod form within graphite fuel blocks. The fuel particle coatings provide the primary, and most effective, barrier to fission product release from the plant. It is these coatings and the high temperature, high heat capacity characteristics of the ceramic fuel which gives the HTGR its inherent safety characteristics. Fission products are retained within the coated particles up to and above temperatures experienced in pressurized and depressurized cooldown transients following loss of the main and shutdown cooling systems.

A vertical helical coil steam generator is located in a separate vessel with the main circulator mounted vertically on top of the steam generator vessel. The top of the steam generator vessel is connected to the reactor vessel below the core by a coaxial crossduct.

A shutdown cooling system (SCS) is located at the bottom of the reactor vessel. It has two functions: to provide backup cooling to ensure investment protection for the components of the primary system, and to permit primary circuit maintenance after reactor shutdown when the steam generator and/or main circulator is out of service. The SCS circulator provides helium flow for the shutdown cooling mode.

A significant feature of the MHTGR design is its passive decay heat rejection in the unlikely event that both the primary and shutdown cooling systems are unavailable. The decay heat rejection is inherently passive because heat from the reactor core is conducted through the mostly uninsulated steel reactor vessel wall and transferred by radiation, conduction and natural convection to reactor cavity cooling panels in the silo. Heat removal from these panels is by passive convective air cooling for all modes of operation.

The fuel temperatures that occur during this passive heat rejection mode are below the temperature that would cause significant fuel particle damage and attendant release of fission products into the primary coolant system. Potential radionuclide releases, including the release that could come from a simultaneous primary circuit break and core heatup in the passive decay heat rejection mode, result in doses to the public that are an order of magnitude below 10CFR100 limits. Consistent with utility/user requirements, analyses of the resultant doses indicate that a public evacuation plan will not be required for a MHTGR.

One of the fundamental advantages of the MHTGR is the small number of systems and components required to ensure the safety of the reactor. The fuel particle coatings are the first and fundamental safety system which form a pressure vessel to retain fission products. The primary circuit

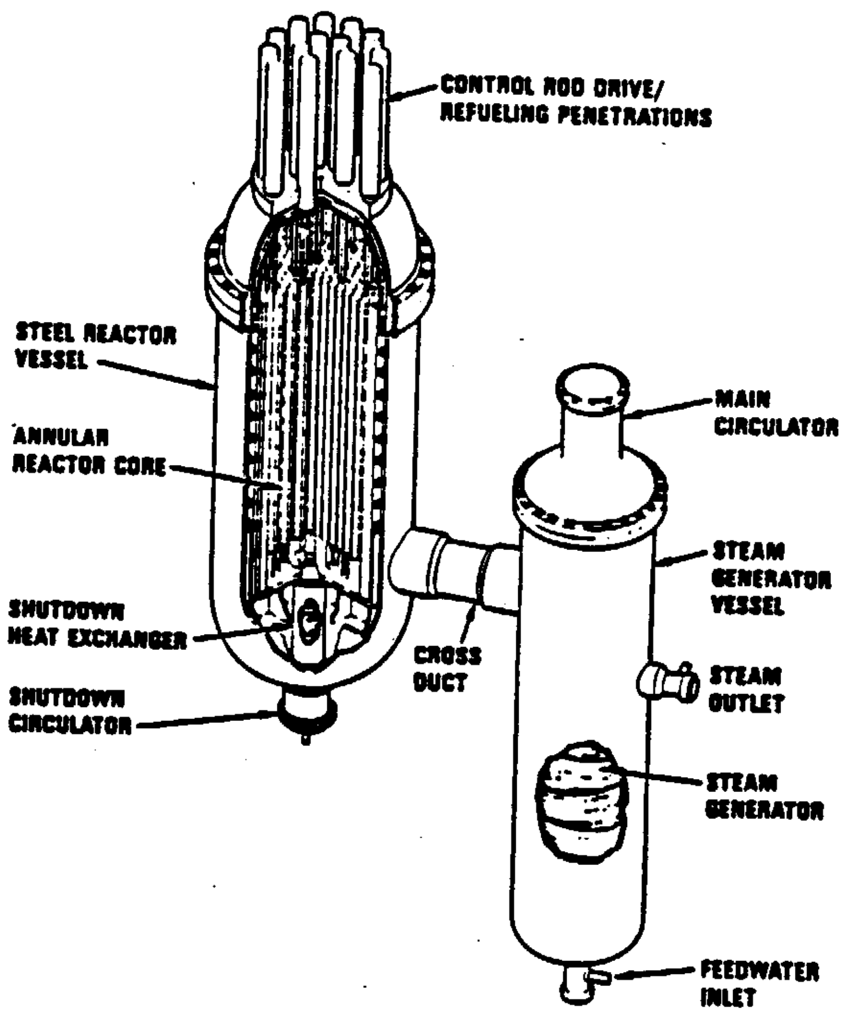

FIGURE 1-1

REACTOR VESSEL AND STEAM GENERATOR

pressure vessel boundary is the next, working outwards from the fission product source. This involves the two main pressure vessels, the cross duct, the stream generator steam piping up to the first isolation valve, and various helium penetrations up to the first isolation valve. With the exception of the control rod drives and reserve shutdown system, no active systems in the primary circuit are required to function to keep the reactor in a safe condition. No buildings are required to retain fission products as part of their design specification. Exterior to the reactor vessel, the passaive reactor cavity cooling system is required to remove decay heat in the event of a conduction cooldown.

As one can see, the number of safety-related systems of the type which have caused delays and cost overruns in previous nuclear power plants is significantly reduced in the MHTGR design. Moreover, the systems that are required to perform as safety systems to protect the public are for the most part passive. Thus, the amount of construction, operation, and inspection that has to be approved by the NRC is greatly reduced. In particular:

- The fuel particle requires quality control in production and, after that, no further direct operating or maintenance action is required.

- The pressure vessel boundary is, for the most part, factory fabricated. There is a small amount of site welding and, after that, the only significant operating and maintenance activity is required inservice inspection.

- The control rod drives are factory-fabricated and are very amenable to complete removal from the reactor for inspection and maintenance.

- The reactor cavity cooling system is a passive system and amenable to easy visual inspection and is tolerant of minor damage.

If the construction, operating, and maintenance requirements for safety-related systems are compared to most currently operating reactors, there is a significant reduction in the number of structures, pumps, valves, motors and other similar equipment required to maintain the reactor in a safe condition, or to mitigate the consequences of an accident, and which, consequently, have to be constructed, operated, and maintained to the NRC requirements. The ability to concentrate on a few key and mostly passive systems should improve plant reliability.

Accordingly, the ability of the Project to hold the line against the addition of back-up safety-related systems is considered essential.

1.5 TEST PROJECT OBJECTIVES

The scope of the Project would be to design, license, construct, operate and thereby demonstrate a full scale, prototypical standard module to ensure that the product is economic, licensable and affords a very high degree of investment protection. The five main technical objectives of the Project are:

1. <u>Demonstrate HTGR-Unique Licensing Process and Support Design Certification of Standard Module and Nuclear Island</u> - Through the reference plant development effort, a disciplined "requirements" based approach to licensing is being developed that should fully capitalize on the HTGR's unique safety characteristics. This process will be practiced and demonstrated through such a Project. In addition, the MHTGR has the ability to test the reactor module and plant response to key licensing basis events as a potentially efficient and convincing supplement to analysis and separate effects testing. In the event licensing issues are identified which are not amenable to ready resolution via calculation or separate effects testing, this facility could be used to resolve such gray issues via demonstration testing, rather than by added conservatism or engineered safeguards.

2. Demonstrate Investment Protection Capability - Independent of potential tests to support the design certification process, the Project will demonstrate the capability of the reference plant design to withstand key design basis events without impairing future economic power production capability.

3. Demonstrate Power Production Capability - The scheduling of operational testing will be arranged to allow demonstration of normal plant maneuvering and sufficient steady state full power operation to establish a demonstrated basis for reliable power operation. A reliability improvement program will be an integral part of the testing.

4. Demonstrate Plant Costs and Schedule - The Project will provide the first-of-a-kind (FOAK) experience for a construction approach based on separation of the nuclear island from the turbine island. The cost and schedule experience in the overall fabrication and construction program will serve as an invaluable data base for prospective vendors and customers.

5. Demonstrate Operating and Maintenance (O&M) Activities - The Project will provide for a demonstration of key O&M activities. The results of such activities will provide feedback to the design of the reference plant that will simplify and reduce O&M procedures and operations.

In addition, there are secondary objectives, not addressed in this Study, which would be achieved when the plant is licensed for long-term power production at the conclusion of the test program:

- Provide an operator training facility for follow-on commercial plants.

- Provide long-term component and material surveillance, combined with the development of repair capabilities, where necessary, for the life limiting components.

- Improve the long term reliability of systems and components.

Further, and perhaps most challenging, the Project must demonstrate whether and how the prospective participants can establish and implement a Project plan. The cost/risk/benefit-sharing arrangements, the resultant management roles of the participants, and the ability of the participants to work together effectively to accomplish the technical objectives is a formidable objective of the Project. At GCRA, these aspects are being pursued in parallel to this Study through the development of a proposed MHTGR Project Strategy Plan, that if successful, will establish the framework for the detailed Project plan.

1.6 PROJECT DEFINITION STUDY SCOPE

This Study covers all aspects of the Project from the beginning of preliminary design in FY1987 to the end of the demonstration/test period but not including its use for operator training, long-term component and material surveillance, and long-term reliability improvement. All activities required to support the Project are addressed, including licensing, plant operation, and maintenance during the test period, the new fuel supply and spent fuel disposal. Further, an estimate is made of the electricity generation capacity factor during the test period and the resulting revenues produced by the sale of the electricity. Technology support costs have been estimated based on the ongoing efforts within the U.S. HTGR Program to establish a detailed technology development plan.

The Study has been managed by GCRA and the responsibilities assigned for the Study are shown below:

Area	Responsibility
Site Characterization	Stone & Webster Engineering Corp.
Nuclear Island Design & Cost	Combustion Engineering, Inc. GA Technologies, Inc. General Electric Co. Bechtel National, Inc.

Licensing & Test Plan	General Electric Co.
Turbine Island Design & Cost	Bechtel National, Inc.
Plant Schedule	Bechtel National, Inc.
Plant Cost Estimate	Stone & Webster Engineering Corp.
Owner's & Site Costs	Gas Cooled Reactor Associates Tennessee Valley Authority
Fuel Cycle Requirements and Cost	GA Technologies, Inc.
Technology Support and Cost	GA Technologies, Inc.
Report Preparation	Stone & Webster Engineering Corp. Gas Cooled Reactor Associates Tennessee Valley Authority

The Study report will serve as an initial baseline for ongoing Project strategy development efforts within GCRA and other prospective Project participants.

1.7 TEST PLAN

The overall test program is estimated to take two years and has been subdivided into a number of phases for ease of definition. The duration of each phase was estimated based on an analysis of each test. For the purposes of estimating power generating revenue, a plant capacity factor of 50% was assumed for the two years.

Figure 1-2 shows the sequence of test phases planned for the Project based on a target schedule with no funding constraints. The first five phases, up to and including the startup test, are representative of the startup program for a mature plant and should provide a basis for the startup planning for subsequent commercial plants. Phase 6 is a

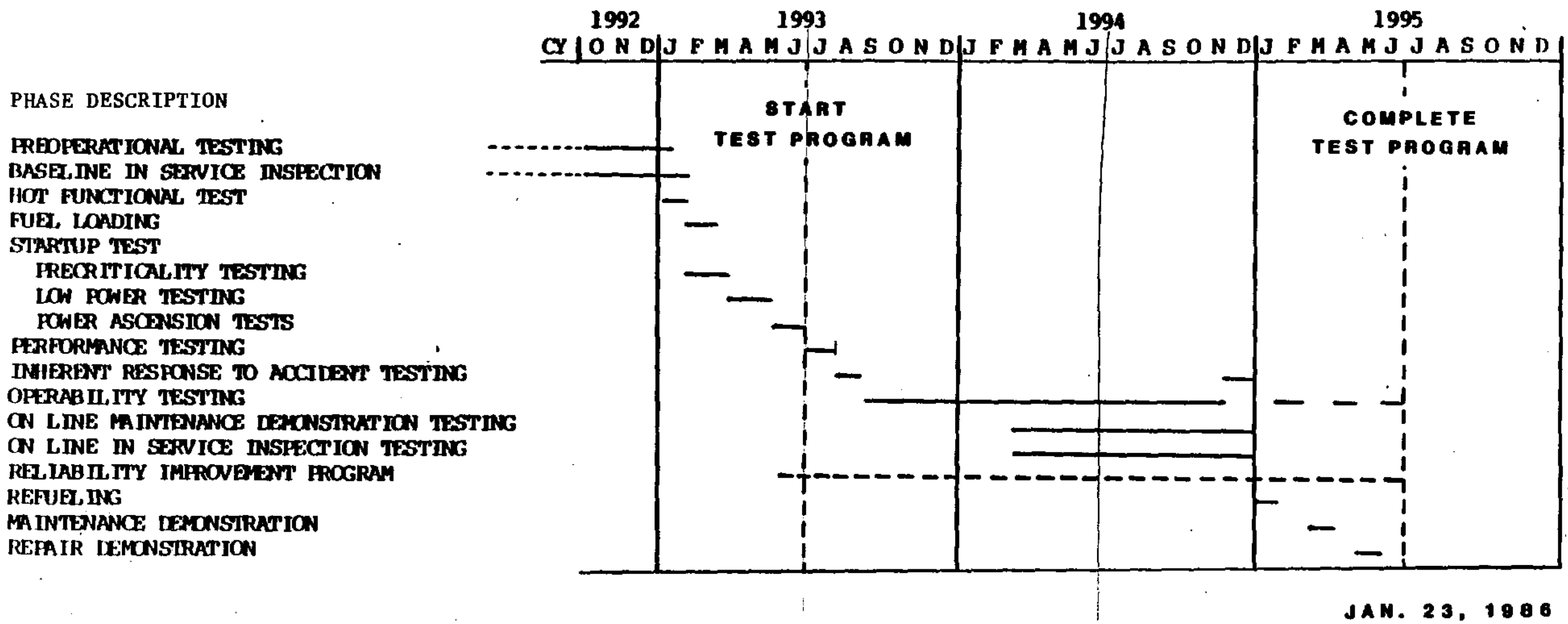

FIGURE 1-2

MHTGR PROJECT TESTING PLAN

performance test phase, where infrequent transients that are expected to occur during the plant life are simulated. Phase 7 is the simulation of low probability events within the design basis. This is the only phase where testing of events that are not expected to occur during the life of a single commercial plant is planned. Phase 8 and onward involve operation, maintenance, inservice inspection and refuelling activities compressed in time to obtain the maximum advantage of the Project. Each of the test phases is described in turn in the following sections.

1.7.1 Preoperational Tests

The objective of these tests is to demonstrate the capability of safety-related and other selected structures, systems and components to meet performance requirements, including safety-related requirements, in all operating modes and throughout the full design operating range to the extent that they can be tested outside full plant service operations. These tests are used to demonstrate that individual system performance is acceptable and that the plant is ready for hot functional testing and initial fuel loading. The testing will commence during the construction phase, as the systems and their supporting subsystems become available, and as all related construction acceptance testing has been completed. Where there has been modular fabrication of systems and to the extent possible, preoperational tests will commence in the fabrication shop.

1.7.2 Baseline Inservice Inspection

The objective of this inspection phase is to provide a preservice baseline inspection against which all future inservice inspections can be compared to determine the extent of any degradation. These tests will be performed no later than the scheduled period, that is, immediately before the hot functional tests, but preferably during the construction phase.

1.7.3 Hot Functional Test

The objective of this test is to run as much of the plant as possible as close to its normal operating conditions as possible prior to fuel loading. In particular, an objective is to run the primary coolant system at full reactor gas inlet temperature and design pressure. Heat will be supplied by the helium circulator. Selected systems will be operated, to the extent possible, in both normal and abnormal modes. After completion of the hot flow tests, selected components will be inspected for damage or wear. This test will also give a first check on the vessel heat losses and the operation of the reactor cavity cooling system.

1.7.4 Fuel Loading

The objective of this phase is to load the first core into the reactor and to make all of the initial preparations for taking the reactor critical. It is expected that fuel loading will be done in air to provide easy access for personnel and equipment. The fuel will be loaded in layers after the reflectors and control rods have been installed. Special absorber elements may also be needed to assure shutdown margins during loading. As fuel loading progresses, neutron flux monitoring will be performed, and the results will be analyzed and compared with predictions before continuing.

1.7.5 Startup Tests

The objective of this phase is to take the reactor critical and bring it up to 100% power in stages, loading the turbine and generating electricity as the power level warrants.

The three phases to startup testing are precritical testing, low power testing and power ascension testing. The precritical testing will begin with fuel loading, and continue throughout the loading of fuel and replacement of any special absorber elements.

After criticality, low power testing will be performed to (1) confirm the design and, to the extent practical, validate the analytical models and verify the correctness or conservatism of assumptions used in the plant analysis, and (2) confirm the operability of plant systems and design features that could not be completely tested during the preoperational test phase due to lack of adequate heat and a representative core. As the fuel and reflector elements are heated above reactor gas inlet temperature, there will be outgassing of the impurities in the graphite. This will provide a further demonstration of the helium purification system.

After low power testing is complete, the plant will be brought to 100% power in a series of discrete power level stages. Major testing will be performed at each power level before proceeding to the next higher power level.

Startup testing will be completed when the reactor is at 100% power and the following duty cycles have been run through at least once in the course of reaching full power:

Startup from depressurized conditions
Startup with full helium inventory
Normal load increase and decrease
Rapid load increase and decrease
Pressurized decay heat removal with SCS
Control rod insertion
Reactor trip from 25% power
Reactor trip from full power

At this point in the test program, it can be verified that the plant operating parameters are within design limits.

1.7.6 Performance Testing

The objective of this phase is to subject the plant to the majority of the less frequent duty cycles within the range of cycles encountered during normal operation.

The following tests were developed from the plant duty cycle and have been included in the test plan for the performance tests:

Step load increase and decrease ("±"15%)
Excess feedwater flow
Sudden feedwater flow reduction
Turbine trip
Circulator trip
Forced cooldown on SCS

If no problems are encountered, it is expected that the performance tests will be completed within one month. There will be no attempt to run any transients for a representative number of cycles for full plant life.

1.7.7 Inherent Response to Accident Tests

The objectives for this phase are:

- To support the contention that the reference plant investment risk goals can be met and, hence, that the estimated plant insurance costs are reasonable.
- To demonstrate to the public, the investment community, and the utility/user industry, that the MHTGR is a forgiving plant concept with low consequences to nuclear incidents.
- To support the design certification by NRC of a standard nuclear island design.

In all three cases, the ideal requirement is to demonstrate that the probability of occurrence of an accident sequence, the sequence of events that follows, and the resulting consequences have been accurately predicted.

In the case of the probability of an accident occurrence the accidents can be divided into external events such as missiles and earthquakes and internal events caused by plant failures or operator error. The test program does not assist in the former. In the case of accidents caused by internal plant faults or sequences of faults, the test program will assist through the reliability improvement program (Section 1.7.11) and the general plant operations during the test program. The probabilities can be judged to be optimistic or pessimistic by gathering data on component reliability and comparing it to the data bases used to establish accident sequence probability.

In the case of the sequence and consequences of an accident, the inherent response to the accident test phase can provide answers to several questions:

- Is the sequence of events correctly predicted?

- Are the ultimate parameters such as pressure vessel temperature and fission product release correctly predicted?

- Did the control and plant protection instrumentation correctly and clearly inform the operator of the true sequence of events?

- Were the optimum automatic or manual actions taken during the course of the test, or should the control system or procedures be revised?

To assess the worth of simulating potential accident sequences, each is reviewed against the ability to provide the above information.

Test sequences, where there is uncertainty in any one or more of the four areas, could clearly support the design certification process. The same is true of the investment risk assessment. For the reasons stated in Section 1.3, the need for convincing demonstrations of the inherent response of the plant for investment risk purposes and for the support of design certification are too interrelated to be discussed separately.

The key factor in the demonstration of plant characteristics to the public is to remove uncertainty in the prediction of the final plant state: that is, to answer the question, "Are all ultimate plant parameters correctly predicted"?

For the test project, the following criteria are proposed for the acceptable consequences of a planned test.

- Site boundary doses must be acceptable under the license obtained for the Project.

- No plant damage should result in a reduction of the 40-year life of the plant, assuming that no repairs that would take longer than 6 months are required following testing.

- Incremental fuel damage should be limited so that the long-term maintenance dose rates and fission product releases in the event of a depressurization accident following the test phase are not compromised.

The list of candidate tests for this phase of testing was developed from the plant duty cycle, the list of licensing basis events, and the probabilistic risk assessment.

The review resulted in the selection of four events that have been investigated. These are:

- Reactivity transients

- Pressurized cooldown with RCCS

- Steam/water ingress

- Depressurized cooldown with RCCS

The selection of the four inherent response to accident tests covers the performance of the key systems which provide safety and investment protection.

For the fuel, the normal operation performance and accident performance will be demonstrated in the conduction cooldown tests. Temperatures above the design basis will have been previously demonstrated in separate capsule tests.

For the control rod drive and the plant protection system that calls for rod insertion, the reactivity transient tests and the steam/water ingress tests will demonstrate the correct sequence and timing of their reactions.

For the reactor cavity cooling system, the conduction cooldown tests will demonstrate its performance.

The selection covers the key design basis events and envelopes the outcome of many initiating events. For example, a pressurized cooldown could be initiated by a feedwater system or component failure, a local electrical failure or a station blackout. The selection is not sensitive to changes in event tree probabilities for the above reason. As the design and beyond design basis events are established, this list will be reviewed for completeness and the test plans reassessed. Since there has not been much detailed transient analysis on the reference or demonstration plant, conclusions that can be drawn in this report are tentative.

Reactivity Transients

Two rod withdrawal tests were identified: (1) The peak rod bank withdrawal with power-to-flow trip and (1) the peak rod bank withdrawal with helium temperature trip. These tests would demonstrate correct functioning of the trip sequence and timing and provide confirmation of the trip set points and the extent of the kinetics of the reactivity transient. The tests would allow any necessary resetting of trip points based on observed transients and would demonstrate reactivity feedback to the core at temperatures above normal.

A further reactivity transient test was identified, the anticipated transient without SCRAM (ATWAS), where there is a failure of the plant protection system to insert the control rods immediately following a circulator trip. This transient would demonstrate the inherent ability of the negative temperature coefficient to control the core temperature for a significant time following the loss of coolant flow.

Pressurized Cooldown

This is a primary transient in terms of demonstrating the inherent response of the plant, i.e., with failure of both the main and shutdown cooling systems, to demonstrate a cooldown via the reactor cavity cooling system.

This test demonstrates that both the main and shutdown cooling systems do not need to be safety-related and, hence, it provides a clear demonstration of the inherent safety of the plant. It also provides the utility/user with a demonstration of the investment risk protection of the plant upon failure of the main and shutdown cooling systems.

At least two tests are envisaged. One test would be performed from less than full power, the other from full power. The key information obtained would be the temperature-time history for the primary circuit components and fuel damage inferences.

There will not be any breach of the primary circuit barrier and, hence, no radionuclide release. The plant is not expected to sustain damage and incremental fuel particle failure is expected to be insignificant. There should be no impact on plant O&M activities.

to project whether such a simulation could be taken to completion or whether the transient would have to be curtailed to prevent unacceptable fuel or plant damage in the context of the criteria previously discussed in this section. The current estimate of fuel temperatures is shown in Figure 1-3. These are the most severe temperatures that are seen in the test program, and comparing them to Figure 1-4, which shows expected fuel damage as a function of temperature, it can be seen that no significant damage is expected. The maximum depressurization rate associated with the conduction cooldown, as a design basis event, is low enough that the transient can be simulated without a blowdown to the atmosphere.

Moreover, the blowdown rate is also low enough that nothing significant could be learned with respect to liftoff of plated-out fission products. Hence, there is no incentive to perform such a blowdown to atmosphere.

More difficult to assess is the extent to which a design basis depressurization, which would result in measurable liftoff of plated-out fission products, should be demonstrated.

Such a determination must be considered in the context of an overall strategy for confirming fission product liftoff rates during such postulated events and is a subject for follow-on work. As such, the full rapid design basis depressurization event has not been included in the test series under investigation.

1.7.8 Operability Testing

The objective of this phase is to subject the plant to continuous opera tion over a continually changing set of operating conditions for a significant period so as to search out any areas of unreliability or noncompliance with expected response and to judge whether the number of plant operators is sufficient. During this phase, delinquent testing activities, or test reruns, if necessary, may be completed without compromising this objective.

There would be a preplanned sequence of operations during this phase that would include switching to and from back-up service systems, component

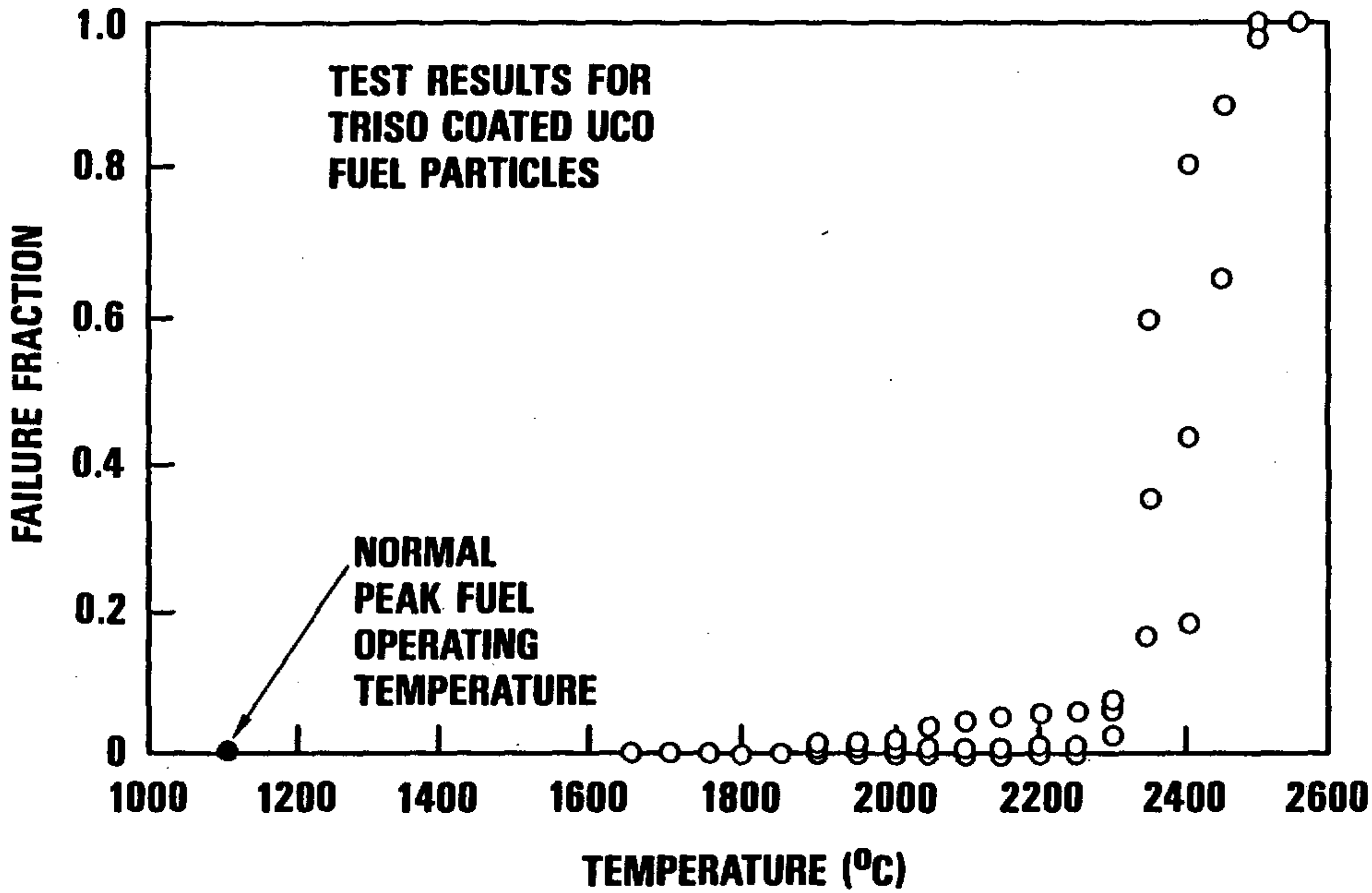

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FIGURE 1-4

FUEL FAILURE RATES WITH TEMPERATURE

shutdown and restart after a short soaking period, and other combinations of events that would look for failures due to unusual operational sequences. An objective would be to duplicate a wide range of plant operations that would normally be accumulated over a much longer time period.

1.7.9 On-Line Maintenance Demonstration

The objective of this phase would be to demonstrate that all maintenance planned for on-line performance can be done within the existing environment of radiation, temperature and space. This phase would be carried out in parallel with the operability testing and would consist of at least one demonstration of each maintenance procedure scheduled to be carried out with the plant online.

1.7.10 On-Line Inservice Inspection

The objective of this phase would be to demonstrate that all the inservice inspections that are planned to be carried out online can be done within the existing environment of radiation, temperature and space. This phase, again, would be carried out in parallel with the operability testing and would consist of at least one demonstration of each inservice inspection procedure that is scheduled to be carried out with the plant online.

1.7.11 Reliability Improvement

This test phase will involve collecting data and evaluating the performance of the operating systems and equipment as well as monitoring corrective actions. The objective is to supplement or substantiate the HTGR reliability data base and to establish trends in achieving plant availability goals.

The reliability test phase will run through the test program and will start with the beginning of prototype or demonstration testing. A small team will be assigned to keep records on each component and system on a systematic basis from the initiation of testing through the end of the Project test phase.

At the end of the Project, a revision will be made to the design data base on availability to assess the basis from which the investment risk calculations were made.

1.7.12 First Fuel Reloading

The five objectives of this phase are: (1) to demonstrate the ability to shut down to depressurized conditions (if this has not already been done during an earlier test phase), (2) to demonstrate the core refueling opera tion, (3) to demonstrate inservice inspection procedures, (4) scheduled mainte nance procedures, and (5) unscheduled maintenance procedures that cannot be performed online.

The activities included in the last test sequence of unscheduled maintenance testing are:

Steam generator tube leak repair
Circulator repair
Inner cross duct bellows repair
Shutdown cooling system repair

Following this first fuel reloading sequence and the maintenance and inservice inspection demonstration, the plant would be returned to service and the planned project testing would be complete. This test phase is an important part of the overall demonstration of plant availability.

1.8. PROJECT LICENSING

1.8.1 Selected Licensing Approach

The licensing approach adopted for planning purposes for both the utility and INEL sites was to obtain an NRC 104 license with both a construction permit and an operating license. This was done following due consideration of the possibility of not having to formally license a test facility that was installed on the INEL site. The formal licensing approach was selected because it provides future commercial plant owners the maximum assurance that the licensing of commercial plants would encounter no obstacles as a result of using special procedures that may be available on the INEL site.

An NRC Class 104 license applies to demonstration plants and plants that are used in research and development when production of products or electricity is not the primary purpose. To qualify for a Class 104 license it would have to be anticipated that not more than 50 percent of the annual cost of operating the test facility is devoted to energy generation for sale and distribution. This is likely to be the case until the test program is completed.

The proposed request for a Class 104 license for the test phase is based on the belief that NRC would initially license the facility with some operating restrictions which would be lifted as the results of the test program were received. Ultimately, at the completion of the testing phase, the license would be upgraded to a commercial Class 103 license.

NRC support for the Project's role in the demonstration of licensability and for the design certification process is anticipated. In the NRC's original policy statement on advanced reactors (Reference 5), one question that was included was:

> "What degree of proof would be sufficient for the NRC to find that a new design is based on technology which is either proven or can be demonstrated by a satisfactory technology development program? For example, is it necessary or advisable to require a prototypical demonstration of an advanced reactor concept prior to final licensing of a commercial facility?"

In its response (Reference 6), following review of the comments received, the NRC stated that:

> "...The Commission does, however, consider the use of a prototypical demonstration facility as having a high potential

for being an acceptable way of resolving many safety-related issues."

The DOE HTGR program is proceeding in FY-1986 to develop licensing criteria unique to the MHTGR design. This Study assumes that the process will be successful and that acceptable criteria for the licensing of a standard nuclear island design would have been approved by the NRC via a licensability statement that would be available early in the Project licensing activities. This is one of the activities in the licensing plan for the standard HTGR (Reference 7).

1.8.2 Test Project Licensing Plan

Based on the selected licensing approach, the Project licensing plan would consist of the following steps:

- A preliminary safety analysis report and preliminary probabilistic risk assessment will be prepared as a basis for obtaining a construction permit.

- An environmental report will be prepared which will include any necessary information on the programmatic aspects of the HTGR as a reactor type which are incremental to the LWR fuel cycle. It is not expected that there will be any major impacts outside the scope covered by the environmental impact statement for the LWR.

A final safety analysis report will be prepared as a basis for obtaining an operating license. A detailed test plan will also be submitted with the application for an operating license. In addition, a final probabilistic risk assessment report, as identified in the NRC's policy statement on severe reactor accidents (Reference 8), and an emergency planning report with evacuation or sheltering requirements, will be submitted.

Figure 1-5 shows the licensing schedule for the Project and the major Project activities of design, construction and testing. It also shows a

FY	1987	1988	1989	1990	1991	1992	1993	1994	1995	1996	1997	1998
STANDARD NUCLEAR ISLAND LICENSING SCHEDULE (REFERENCE)												
NRC LICENSABILITY STATEMENT	X											
PREPARE/ISSUE PSSAR	XXXX	XXXX										
PRELIM. DESIGN APPROVAL (PDA REVIEW)			XXXX	XXXX								
PDA				X								
PREPARE/ISSUE FSSAR						XXXX	XXXX					
FINAL DESIGN APPROVAL (FDA REVIEW)								XXXX	XXXX			
NRC ISSUE FDA/PETITION FOR RULE MAKING									X			
SUPPORT CERTIFICATION OF STANDARD DESIGN										XXXX	XXXX	
CERTIFICATION OF STANDARD NUCLEAR ISLAND DESIGN											X	
DEMONSTRATION PROJECT SCHEDULE												
PROJECT DEFINITION	XXXX											
DESIGN/PREORDER	XXXX	XXXX	XXXX	XXXX								
SITE CONSTRUCTION					XXXX	XXXX	XXX					
TEST PROGRAM							X	XXXX	XXX			
DEMONSTRATION PROJECT LICENSING SCHEDULE												
PREPARE/ISSUE LEGAL DOCUMENT ESTABL'G PROJECT ORG	XXXX											
OBTAIN STATE CERTIFICATION OF SITE	XXXX											
PREPARE OUTLINE TEST PLAN AND REVIEW WITH NRC	XXXX											
PREPARE/ISSUE PSAR/APPLICATION FOR CONSTR. PERMIT	XXXX	XXXX										
ISSUE PRELIM. PROBABILISTIC RISK ASSESSMENT (PRA)	XXXX	XXXX										
OBTAIN SITE METEROLOGICAL DATA	XXXX											
PREPARE/ISSUE ENVIRONMENTAL REPORT (ER)		XXXX										
SUPPORT ENVIRONMENTAL PUBLIC HEARINGS			XXXX	XXXX								
PREPARE/ISSUE PRELIM. DEMO. TEST PLAN		XXXX										
SUPPORT CONSTRUCTION PERMIT HEARINGS			XXXX	XXXX								
NRC ISSUE SER/EIS				XXXX								
CONSTRUCTION PERMIT (CP)				X								
PREPARE/ISSUE FSAR AND FINAL PRA					XXXX	X						
ISSUE FINAL DEMONSTRATION TEST PLAN						XXXX						
ISSUE FINAL ER AND EMERGENCY PLANNING REPORT						XXXX						
NRC ISSUE FINAL ENVIRONMENTAL STATEMENT						XXXX						
SUPPORT FINAL PUBLIC HEARINGS						XXX	X					
NRC ISSUE FINAL SER						XXX	X					
SUPPORT PUBLIC HEARING ON OPERATING LICENSE						XXX	X					
NRC ISSUE OPERATING LICENSE FOR TEST OPERATIONS							X					
PREPARE/ISSUE DEMONSTRATION TEST REPORT								X	XXX			
NRC ISSUE AMENDMENT TO FINAL SER								X	XXX			

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FIGURE 1-5 MHTGR LICENSING SCHEDULE

If it were planned to simulate either the above licensing basis event or the less restricting design basis rapid depressurization event without conduction cooldown, then additional measures would likely have to be taken, especially in the case of any nonremote site. Such measures would involve the use of data from previous, less severe, tests, testing in favorable weather conditions, the use of hold-up tanks and/or filters, or a test sequence which is not fully prototypical.

1.9 COST AND SCHEDULE

1.9.1 Cost

The cost estimate covers all aspects of the Project from preliminary design to completion of the test period. It includes all Project support activities such as licensing, plant operation during the test period, new fuel supply and spent fuel disposal. It also includes revenues produced by the sale of the electricity generated, assuming a 50% capacity factor over the two-year test period, but excludes use of the facility for operator training, long-term component and material surveillance, and long-term reliability improvement.

The Project costs are estimated in 1985 dollars and include 20% contingency in the nuclear island costs and 10% in the power conversion area (turbine island) costs. Contingencies for the technology support, fuel and operating costs have not yet been added. More complete contingency analyses will be developed in the next phase of Project development.

Direct Costs

The direct cost portion of the estimate is comprised of factory material, site material and site labor costs for both the nuclear island and the turbine island. Factory material includes all factory-fabricated equip ment costs. These costs were estimated at the three digit account level and summarized to the following two-digit direct cost accounts:

20. Land and Land Rights
21. Structure and Improvements
22. Reactor Plant Equipment
23. Turbine Plant Equipment
24. Electric Plant Equipment
25. Miscellaneous Plant Equipment
26. Main Condenser Heat Rejection System

The direct costs for the main vessels, steam generator and cross duct were estimated by Combustion Engineering Inc., based on fabrication in their Chattanooga facility. Reactor internal components, fuel handling equip ment, control rod drives and the main circulators were estimated by GA Technologies based to a large extent on experience with very similar equipment for the Fort St. Vrain plant. General Electric Co. estimated the costs for plant control equipment and Bechtel National, Inc., estimated the buildings and service equipment in the nuclear island and the total costs for the turbine island.

All equipment and component costs were based on an assumption that there were no further plant orders during the fabrication of the equipment, and hence all plant costs were developed on a one-of-a-kind basis.

In addition, transportation costs were determined for the reactor vessel, core barrel, steam generator, circulator cross duct and turbine generator (for the INEL site only). These costs were included in the indirect costs. All other transportation costs are included in appropriate direct cost accounts.

For the utility site, the repowering option resulted in the inclusion of costs for items such as retubing of the condenser and the feedwater heaters, new booster pumps and a new demineralizer plant. Such items were included in the turbine plant accounts.

approach and amounts to approximately $14 million per year. The demonstration/ test costs are comprised of two main components, the plant operation and maintenance costs together with any additional staffing requirements for specific tests, and the test equipment costs. In general, there is very little extra staffing required for this Project over and above the planned operating staff as estimated above. The main exception is the specialized support required in the inherent response to accident test phase. In the demonstration/test program, the initial fuel load will not be used to its 22-month capacity and so no reload fuel costs were included in the estimate.

For both sites, an estimate was made of the revenues that could be expect ed from the sale of power generated. Approximately 20 mills/kWh was used on both sites as the cost of power on a non-firm basis. These revenues were deducted from the costs in account 94.

The indirect costs for the two sites are very similar, $431 million for Widows Creek and $465 million for INEL, and by far the largest portion is the nuclear island first-of-a-kind engineering costs, of about $289 million. These costs will be developed and scrutinized in greater detail during the next phase of Project development.

The other components, construction services, field engineering and the owner's costs are relatively small in total, $132 million for Widows Creek and $161 million for INEL. The biggest single item in that total is the first core cost of $41 million.

Cost Estimate Summary

Table 1-1 provides the cost estimate summary for the Project. Table 1-2 provides the resultant cash flow requirements, begining October 1986. Two cash flows are shown, one for an unconstrained budget schedule and one for a schedule with a near-term budget constraint which slips the manufacturing and construction by two years. Schedules are discussed in the following section.

Table 1-1

PROJECT COSTS (1985 $M)

	Utility Repowering Site	INEL Site
Structures & Improvements	41	60
Reactor Plant Equipment	102	104
Turbine Plant Equipment	16	37
Electrical Plant Equipment	8	17
Miscellaneous Plant Equipment	3	6
Main Condenser Heat Rejection Equipment	-	6
Total Direct Costs	170	230
Construction Services	26	47
Home Office Design & Engineering	299	304
Field Engineering	10	14
Owners Costs Including Operation & Revenue	96	100
Total Indirect Costs	431	465
Contingency	102	116
Technology Support Costs	108	108
Total Project Costs	811	919

Table 1-2

CASH FLOW (1985 $M)

Unconstrained Schedule for Utility Site	1987	1988	1989	1990	1991	1992	1993	1994	1995	1996	1997	1998	Total
Direct Costs	-	-	7	42	64	35	19	3	-	-	-	-	170
Indirect Costs	47	66	68	73	71	53	16	15	18	4	-	-	431
Contingency	8	12	13	19	22	15	6	3	3	1	-	-	102
Technology Support	29	35	22	15	6	1	-	-	-	-	-	-	108
Total	84	113	110	149	163	104	41	21	21	5	-	-	811
Unconstrained Schedule for INEL Site													
Direct Costs	-	-	8	47	83	54	30	8	-	-	-	-	230
Indirect Costs	48	67	70	77	77	67	23	13	18	5	-	-	465
Contingency	8	12	13	20	26	20	9	4	3	1	-	-	116
Technology Support	29	35	22	15	6	1	-	-	-	-	-	-	108
Total	85	114	113	159	192	142	62	25	21	6	-	-	919
Constrained Schedule for Utility Site													
Direct Costs	-	-	-	-	7	42	64	35	19	3	-	-	170
Indirect Costs	20	28	37	43	58	68	71	53	16	15	18	4	431
Contingency	4	5	7	8	12	16	22	15	6	3	3	1	102
Technology Support	15	17	18	18	18	15	6	1	-	-	-	-	108
Total	39	50	62	69	95	141	163	104	41	21	21	5	811
Constrained Schedule for INEL Site													
Direct Costs	-	-	-	-	8	47	83	54	30	8	-	-	230
Indirect Costs	20	28	37	45	60	72	77	67	23	13	18	5	465
Contingency	4	5	7	9	12	16	26	20	9	4	3	1	116
Technology Support	15	17	18	18	18	15	6	1	-	-	-	-	108
Total	39	50	62	72	98	150	192	142	62	25	21	6	919

Note: Operating costs and revenues beyond test period are not shown.

JAN. 21, 1986

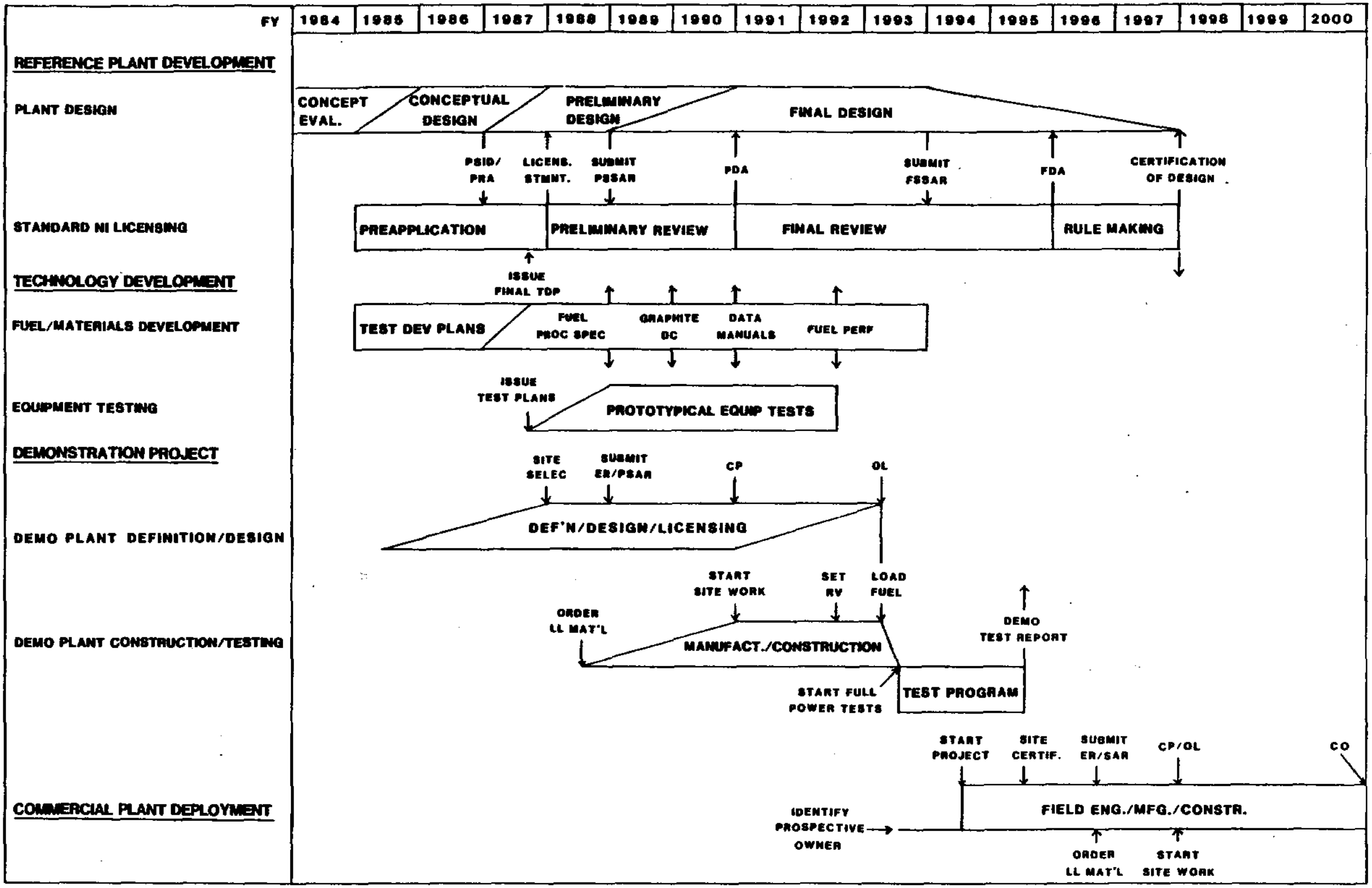


FIGURE 1-6 MHTGR PROGRAM/PROJECT SCHEDULE - NO FUNDING CONSTRAINTS

initial order date of September 1988 for the target schedule. Thus, for the unconstrained target schedule, material ordering is required approxi mately two years after the project start. All other major vessel equip ment can be procured within this time frame.

The circulators and fuel handling equipment which are needed on site a little later than the vessels can be procured and tested within the four year time span available.

Detailed activities for manufacturing, construction, and startup are shown on the fabrication and construction schedule (Figure 1-7).

Startup and Testing

The schedule for the the test program discussed previously in Section 1.7, is shown on Figure 1-2. Acceptance/systems startup tests and the hot functional test culminating in fuel load are also noted on the fabrication and construction schedule, Figure 1-7.

The performance test is scheduled immediately after the startup testing and is estimated to take about a month. This is followed by the first phase of the inherent response to accident tests. The last phase of these tests is after the operability tests. The prime objective of the operability tests is to keep the plant operat ing to prove its availability over an extended period. Thus, any delays encountered on earlier tests or retesting requirements can be absorbed in this period, with no adverse impact on the overall two-year test program.

Fuel Supply

The initial fuel load for the unconstrained target schedule is required on site in February 1993. By this time, a representative production sample produced in the same equipment as the production fuel must have undergone capsule irradiation testing as a basis for NRC to issue an operating license for the Project.

To meet this schedule, GA Technologies, Inc.'s fabrication plant must be in production by December 1990, in order to allow two years for fabrication of the first core. Expansion modification to the facility will take 30 months and so fuel contracts must be placed by June 1988.

Constrained Schedule

Recognizing budget constraints in the federal government in the near term, a delayed schedule is shown in Figure 1-8, which is the basis for the constrained cash flows given in Section 1.9.1, Table 1-2. This schedule, which is the expected basis for ongoing strategy development, adds two years to the design phase of the project.

1.10. FUEL REQUIREMENTS

One of the fundamental characteristics of the MHTGR which makes it inherently safe is the ability of the coated particle fuel to retain fission products up to the maximum fuel temperatures seen in pressurized or depressurized cooldowns with loss of the main and shutdown cooling systems. This retention of fission products is dependent on the contami nation and defect fractions of the new fuel and on the inservice fuel particle coating failure rates. The fuel specification calls for the following:

Low Enriched (19.9%) Uranium Thorium Coated Particles
Surface Contamination: $<2\times10^{-5}$ gHM/gHM in core
Defective SiC Fraction: $<4\times10^{-5}$
Inservice Failure Rate: $<5\times10^{-5}$

The specification is for a fuel that will allow a licensing basis depres surization accident without exceeding the Protective Action Guidelines (PAG) limits for no requirements for evacuation and sheltering planning.

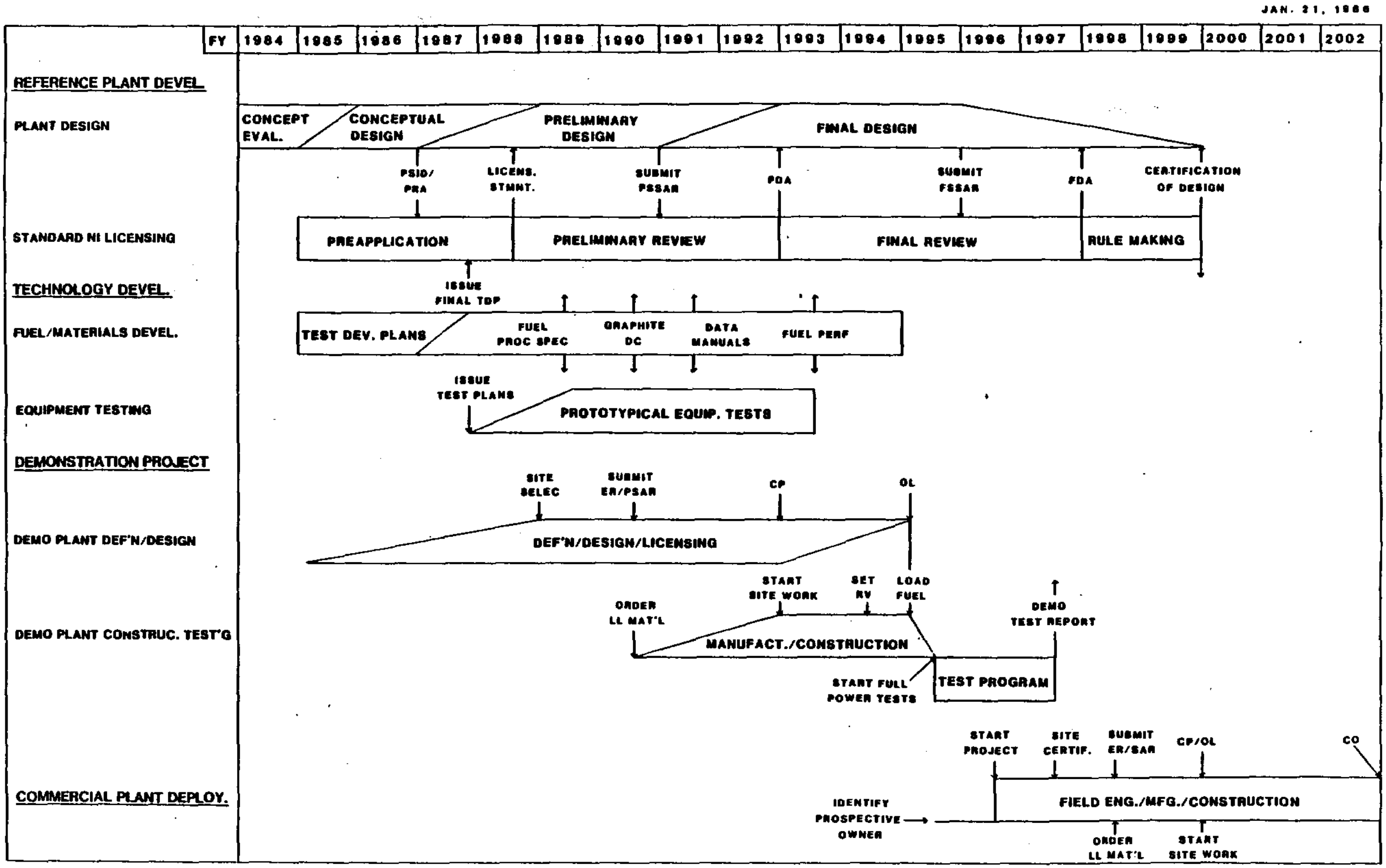
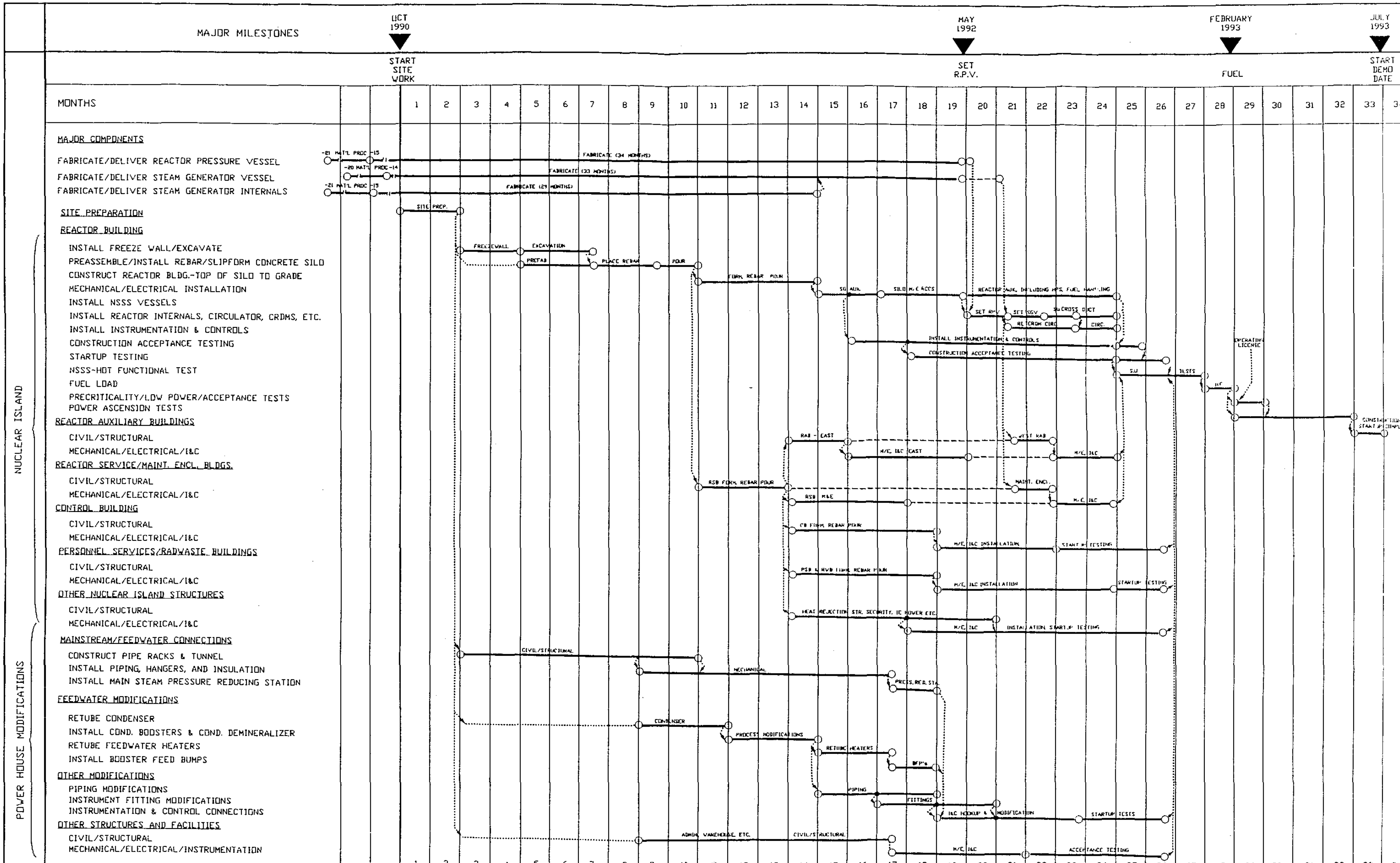


FIGURE 1-8 MHTGR PROGRAM/PROJECT SCHEDULE - DELAYED TWO YEARS BY FUNDING CONSTRAINTS



DESCRIPTION	REV	DATE
ISSUED FOR REPORT	0	12/12/85
ISSUED FOR FINAL REPORT	1	1/10/86

PROJECT: GCRA/TVA MODULAR HTGR TEST PROJECT DEFINITION STUDY

TITLE: FABRICATION & CONSTRUCTION

Figure 1-7

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The above levels imply a high quality fuel. This level has been demonstrated at laboratory scale in the U.S. and in production scale in the FRG. During the design and fabrication phase of the Project, there will have to be a parallel effort to develop the existing fuel process line to produce the required quality, and a proof test program to irradiate the fuel at normal operating temperatures and to perform core heat-up simulation tests to verify acceptable inservice performance. As a comparative reference, the fuel currently being manufactured for FSV has achieved defect fractions of 8×10^{-4} and inservice failure has not been significant enough to be measurable.

The graphite fuel blocks in which the fuel rods are contained are essentially the same as those in the currently operating Fort St. Vrain reactor which operates with a longer fuel cycle (6 years) and at higher gas temperatures than the MHTGR design.

The fuel costs developed for this project are based on an initial core design which is identical to the first core of a commercial plant. This core has a 22-month life at 80% capacity factor, and will thus have considerable life left at the end of a two-year test program.

1.11. TECHNOLOGY SUPPORT

The technology support program required for this Project is based on the ongoing efforts within the DOE HTGR Program to prepare a detailed Technology Development Plan. The technology support program is estimated to require about $108 M over about 6 years assuming it is run in parallel with the Project. Summary elements within this estimate are:

	$ Million
Fuel Development and Qualification	6
Fuel Process Development	12
Fission Product/Source Term Development	12
Graphite Development	13
Materials Development	11
Component Testing	
Circulator	10
Control Rod	3
Fuel Handling	2
Steam Generator	5
Core	11
Reactor Internals	4
Other	19
Total	$108

Very preliminary studies indicate that the program cost could be reduced by about $23 million if a less extensive technology support program were undertaken, and instead, confirmatory test information was provided by the test project. A further $4 million could be saved by increased cooperation with the ongoing HTGR Program in the Federal Republic of Germany. Due to the very preliminary nature of this review, a firm recommendation for the minimum technology program required to support the test project is not available for this report but is planned to be developed during 1986.

The technology support program has a number of activities that must be accomplished on a tight schedule in order to meet the overall Project schedule. One is the need to produce and test production quality fuel prior to fabrication of the first core. Another is the need to complete the fabrication and testing of the circulator, together with any necessary rework for the production unit to meet the Project's construction schedule.

1.12 ASSESSMENT

1.12.1 Project Requirements

Table 1-3 shows a series of requirements that were developed for the Project in an early review of the need for such an undertaking. The utility/user requirements are stated first, based on the groundrules and objective set by the GCRA Test Project Strategy Subcommittee. The incremental requirements of the vendors, the NRC and the public, as represented by DOE, are shown. For each requirement, a statement of the ideal way of meeting the requirement is given and the capability of the project to meet the requirement is stated.

In summary, the proposed Project, as defined in the Study, can meet the majority of the requirements, namely:

- Demonstrate commercial plant performance, and provide an availability trend curve indicating whether an 80% availability is achievable in commercial plants.

- Demonstrate the HTGR unique licensing process and provide support for the design certification of a standard nuclear island.

- Provide a demonstration of the insurability and investment risk protection of the plant by simulation of key design basis events.

- Provide a basis from which capital costs, O&M costs, and fuel costs for commercial plants can be reliably predicted based on first-of-a-kind experience.

- Provide a demonstration of constructability and commercial plant schedule projections.

- Provide a demonstration of a licensed fuel fabrication facility and component manufacturability as a basis for commercial pricing.

Based on all of the above, it will demonstrate whether the MHTGR is a low cost, reliable, safe energy source.

However, it does not meet the ideal requirements in two areas. It does not demonstrate the interactions between two or more reactors and turbine plants, but it is felt that this can be adequately handled by computer simulation. Also, the duration of the test program is such that full fuel burnup will not be achieved within the initial demonstration/test period. However, the planned fuel irradiation capsule tests will provide adequate assurance of the performance under full irradiation conditions prior to receipt of the operating license.

1.12.2 Comparison of Sites

Within this Study, two sites were considered: (1) the Widows Creek site in which an existing turbine would be repowered, and (2) a complete stand-alone full plant at a remote INEL site. The representative sites chosen for this Study serve to delineate the site-dependent impacts.

Specific site-related benefits and limitations based on the two representative sites selected are summarized as follows.

For utility sites in general, the advantages of an established nuclear utility host, the possible cost savings due to repowering, the use of existing power distribution and transmission system, and other site facilities, are clear. Against this must be balanced the potential for

TABLE 1-3

TEST PROJECT REQUIREMENTS AND CAPABILITY TO MEET THEM

Requirement	Ideal for 100% Completion	Project Capability
Utility/User Requirement		
Demonstration of low cost, reliable, safe energy from HTGR's	A demonstration project which achieves its objective within the planned schedule and cost Utility/Users ready to buy commercial plants under normal free-market arrangements Vendor and fuel supplier team(s) prepared to offer competitive firm price contracts for plant and fuel Design certification for standard nuclear island	Achievable
Demonstration of the HTGR-Unique Licensing approach and support for design certification of the standard nuclear island for a broad range of plants	Fully licensed plant with the ability to simulate key design basis events	Achievable
Demonstration of commercial plant performance; capability to produce rated output in compliance with duty cycle	12 Months of operation covering the range from shutdown to full power, including 3 months of continuous operation at full power and demonstration of response to required duty cycle transients	Achievable
Demonstration of the availability of a commercial plant	Long term operation at rated power with an established availability in excess of 80%	Two years operation during demonstration/test period with an established availability trend curve indicating 80% availability is achievable for commercial plants

Table 1-3 (Cont)

Requirement	Ideal for 100% Completion	Project Capability
Basis for estimating the capital cost of commercial plant	Established vendor team(s) prepared to offer firm price contract to build subsequent plants or nuclear island	Achievable
Basis for estimating the O&M costs of commercial plant	Two years operating & maintenance cost data within design and staffing requirements and planned maintenance costs	Two years O&M experience during demonstration/test period with trend curve indicating commercial plant requirements can be met
Basis for estimating the fuel costs of commercial plant	Established fuel supplier(s) prepared to offer firm price contract for long-term fuel fabrication services	Achievable except that full fuel burnup data will be obtained in separate effects tests
	Full fuel burnup performance	
	Contract with government for spent fuel treatment/management	
Demonstration of the constructability and startup of commercial plants	Completion of construction and startup according to the planned schedule	Achievable on a first-of-a-kind basis
Demonstration of insurability of commercial plants	Ability to perform full scale demonstration of compliance with investment risk criteria	Achievable for most, possibly all, design basis events
Demonstration of commercial plant expendability	With one unit in full power operation, construct and start up a second unit	Not planned

Table 1-3 (Cont)

Requirement	Ideal for 100% Completion	Project Capability
Incremental Vendor Requirements		
Demonstration of component manufacturability	Manufacture of components/modules in commercial based facility	Achievable
Basis for pricing commercial plants, components, systems, and fuel supply	Complete cost data base plus 12 months of operation covering the range from shutdown to full power, including three months of continuous full power operation	Achievable
Incremental NRC Requirements		
Basis for licensing fuel facility	Completed commercial facility with process and security procedures approved	Achievable
Incremental Public/DOE Requirements		
Capitalise on HTGR's very high temperature capability and broad process heat applications	Extend project objectives for applications/tests in higher temperature applications	Not planned
Share in international market within non-proliferation policy	Vendor and fuel supplier team(s) prepared to offer competitive firm price bid to foreign buyers for plant and fuel (based on once-through, low enriched fuel cycle)	Achievable

lower availability of the refurbished turbine plant as a result of mismatched operating conditions and age. For the specific site used in this study with a 425 M exclusion area boundary, there are potential restrictions in performing a severe accident test such as a design basis depressurization. There are also potential undesireable dynamic effects in the long steam and feed lines required for the nuclear island which was about 2000 feet from the turbine plant.

For the remote INEL site, the advantages are the enhanced ability to perform the severe accident tests, the availability of experienced reactor test personnel, and the ease of long-term storage of spent fuel. The disad vantages are the lack of an established utility host and high overland transportation costs, higher site labor rates and lower site labor productivity than on the Widows Creek site.

1.13 FUTURE PROJECT DEVELOPMENT ACTIVITIES

The results reported herein constitute the initial efforts to define a MHTGR Demonstration/Test Project. The purpose of this effort has been to establish whether sufficient bases exist to pursue a private-sector Project initiative. During the course of this Study, the overall budget environment within the federal government and DOE's advanced reactor development programs in particular has necessitated that any near-term, advanced reactor demonstration project be based on a private-sector initiative. Accordingly, follow-on activities during 1986 must be considered in that context.

Three categories of Project development activities during 1986 have been identified:

1. Expansion of scope from initial Study, namely,

   a. Perform a detailed analysis of each of the planned inherent response to accident test sequences to be able to better plan the tests, and to more accurately predict the plant state throughout the tests. This would also allow a better understanding of the test equipment and test instrumentation requirements.

b. Develop more detailed cost estimates with particular emphasis on the front-end design development and technology support activities.

2. Identify ideal utility site requirements for both "green-field" and repowering applications. In addition, identify candidate sites and determine and envelope of site-related requirements on the Project's design, license, and schedule.

3. Develop a project strategy plan as a proposed framework for Project implementation. This plan will address the proposed cost/risk sharing arrangements and the roles of the prospective participants. From this effort and the Project definition results, a Project solicitation package will be developed to solicit private-sector support for 1986 and beyond.