DOE-HTGR-86-069
Revision 1

HTGR

DO NOT MICROFILM
COVER

FORCED OUTAGE ASSESSMENT FOR THE MHTGR

AUTHORS/CONTRACTORS

GA TECHNOLOGIES INC.
BECHTEL NATIONAL INC.
STONE & WEBSTER ENGINEERING CORP.

ISSUED BY GA TECHNOLOGIES INC.
FOR THE DEPARTMENT OF ENERGY
CONTRACT DE-AC03-84SF11963

JULY 1987

DOE–HTGR–86–069
Revision 1
908960/1

DISCLAIMER

FORCED OUTAGE ASSESSMENT FOR THE MHTGR

Issued By:
GA Technologies Inc.
P.O. Box 85608
San Diego, California 92138–5608

DOE Contract No. DE–AC03–84SF11963

GA Project 6300

JULY 1987

Roll 2084 Roll 2166/1

GA 1485 (REV. 10/82)

GA Technologies Inc.

ISSUE SUMMARY

TITLE
FORCED OUTAGE ASSESSMENT FOR THE MHTGR

☐ R & D
☐ DV & S
☒ DESIGN

APPROVAL LEVEL 5

DISCIPLINE	SYSTEM	DOC. TYPE	PROJECT	DOCUMENT NO.	ISSUE NO./LTR.
0	0100	RGE	6300	908960	1

QUALITY ASSURANCE LEVEL	SAFETY CLASSIFICATION	SEISMIC CATEGORY	ELECTRICAL CLASSIFICATION
QAL II	NNS	NONCATI	NON1E

ISSUE	DATE	PREPARED BY	APPROVAL: ENGINEERING	APPROVAL: QA	APPROVAL: FUNDING PROJECT	APPROVAL: APPLICABLE PROJECT	ISSUE DESCRIPTION/ CWBS NO.
0	SEP 26 1986	C.G. Hoot T. D. Dunn J.L.Pickering	S.B.Inamati F.A.Silady for FAS Interface ~~Assurance~~ J.C.Flanagan	G.P.Connors		G.C.Bramblett	Initial issue 6351030201 HTGR-86-069/0

INFO 1

CONTINUE ON GA FORM 1485-1

*See list of effective pages (iv)

NEXT INDENTURED DOCUMENTS

908716
(HTGR-86-006)

~~APPLIED TECHNOLOGY~~

~~Any Further Distribution by any Holder of this Document or of Other Data Therein to Third Parties Representing Foreign Interests, Foreign Governments, Foreign Companies and Foreign Subsidiaries or Foreign Divisions of U.S. Companies Shall Be Approved by the Director, HTR Development Division, U.S. Department of Energy.~~

REV																												
SH																												
REV																												
SH	29	30	31	32	33	34	35	36	37	38	39	40	41	42	43	44	45	46	47	48	49	50	51	52	53	54	55	56
REV																												
SH	1	2	3	4	5	6	7	8	9	10	11	12	13	14	15	16	17	18	19	20	21	22	23	24	25	26	27	28

GLASNAPP WOR DOCs. 131, 130, 129, 86, 82, 91, 90

GA 1485-1 (REV. 10/82)

GA Technologies Inc.

ISSUE SUMMARY CONTINUATION SHEET

TITLE	DOCUMENT NO.	ISSUE NO./LTR.
FORCED OUTAGE ASSESSMENT FOR THE MHTGR	908960	1

ISSUE	DATE	PREPARED BY	APPROVAL: ENGINEERING	APPROVAL: QA	APPROVAL: FUNDING PROJECT	APPROVAL: APPLICABLE PROJECT	ISSUE DESCRIPTION/ CWBS NO.
1	JUL 31 1987	R.H.Hansen T.D. Dunn	S.Inamati F.Silady Interface Assurance	G.P.Connors		G.Bramblett	General Revision 6351 030 207 HTGR-86-069/1 Class II Change

LIST OF EFFECTIVE PAGES

Page Number	Page Count	Revision
i through ix	9	1
1-1 through 1-2	2	1
2-1 through 2-2	2	1
3-1 through 3-5	5	1
4-1 through 4-5	5	1
5-1 through 5-55	55	1
6-1	1	1
7-1	1	1
8-1	1	1
9-1 through 9-2	2	1
A-1 through A-2	2	1
B-1 through B-5	5	1
C-1 through C-3	3	1
Total pages	93	

CONTENTS

LIST OF ACRONYMS

AER	Applied Economic Research, Inc.
BSBSG	Building Structures and Building Services Groups
BWR	Boiling Water Reactor
CDF	Cumulative distribution function
ECA	Energy Conversion Area
EFOH	Average equivalent forced outage hours per year
EG	Electrical Group
EMTTR	Equivalent mean time to repair (equivalent full power hours lost)
FHSSS	Fuel Handling and Storage System
FOR	Forced outage rate
HPS	Helium Purification Subsystem
HRG	Heat Rejection Group
HTGR	High Temperature Gas-Cooled Reactor
HTS	Heat Transport System
HVAC	Heating, Ventilation, and Air Conditioning System
IEEE	Institute of Electrical and Electronic Engineers
LTA	Low temperature adsorber
LWR	Light-water reactor
MCIG	Miscellaneous Control and Instrumentation Group
MHTGR	Modular High Temperature Gas-Cooled Reactor
MTBF	Mean time between failure
MTTF	Mean time to failure
MTTR	Mean time to repair
NERC	North American Electric Reliability Council
NERC/GADS	NERC/Generating Availability Data System
NI	Nuclear Island

NPRDS	Nuclear Plant Reliability Data System
NSSS	Nuclear Steam Supply System
OPDS	Overall Plant Design Specification
PCB	Power Conversion Group
PCDIS	Plant Control, Data and Instrumentation System
PPIS	Plant Protection and Instrumentation System
PSG	Plant Service Group
RCCS	Reactor Cavity Cooling System
RS	Reactor System
RSG	Reactor Service Group
SCS	Shutdown Cooling System
SH	Service hours
TBCCWS	Turbine Building Closed Cooling Water System
VS	Vessel System
UPC	Utility Power Corporation

1. SUMMARY AND CONCLUSIONS

The plant forced outage assessment was performed as part of the conceptual design phase Modular HTGR (MHTGR). This assessment shows that the plant forced outage is predicted to be a total of 782 average equivalent hours per plant year, or 8.9%. Therefore, the MHTGR plant meets the forced outage goal of 876 h or 10% per plant year. The forced outage contribution from plant investment risk is not included in this assessment except for the outages from primary coolant leaks. Investment risk considers accident events which are typically less likely than those considered in unavailability assessments. Therefore, investment risk characteristics need to be quantified to assure that their contribution is less than 1% per plant year.

The plant forced outage assessment is based on the individual estimates of forced outage for each of the plant's subsystems (Refs. 1-1, 1-2).

The major contributors to forced outage are the turbine-generator, feedwater and condensate, heat transport, vessel systems. These systems alone contribute 5.8% per plant-year to the plant's unavailability (see Table 1-1).

2. INTRODUCTION

2.1. BACKGROUND

This forced outage assessment is based on the conceptual design information for the Standard MHTGR Plant as described in the Overall Plant Design Specification (Ref. 2-1). Furthermore, this assessment concentrated on subsystems identified as important in the major contributors to forced outage in the Standard MHTGR report (Ref. 2-2). These systems were the heat transport, reactor, turbine-generator, and feed-water and condensate. In addition the vessel system was determined to be a major contributor due to radioactive helium leakage.

2.2. OBJECTIVE

The objective of this forced outage assessment is to provide plant forced outage assessment results based on current design information for the Standard MHTGR and the latest reliability data, and compare the results to the top level goals.

2.3. REPORT CONTENT

This report contains the forced outage assessment for the 4 x 350 MW(t) MHTGR conceptual design.

Section 3 gives the unavailability methodology including assumptions used to perform this plant assessment.

Section 4 contains a summary level plant description.

TABLE 1-1
MAJOR CONTRIBUTORS TO FORCED OUTAGE

System or Subsystem	EFOH[a] (per yr)	Unavailability %/yr
Turbine-generator	220.1	2.51
Feedwater and condensate	114.3	1.30
Vessel	100.0	1.14
Heat transport	76.7	0.88
Total for major contributors	511.1	5.83

[a]Equivalent forced outage full power hours per plant year.

Section 5 contains the detailed NI and ECA forced outage assessment basis for each plant system and subsystem which have an impact on plant forced outage. This section also contains a summary level description of the system or subsystem.

Section 6 presents a brief description of the investment risk contributions to availability. (Primary coolant leaks only)

Section 7 presents the overall plant forced outage characteristics.

Section 8 presents a discussion of the results of this assessments.

Section 9 provides the references.

Appendix A contains the calculation of plant service hours (SH).

Appendix B contains summary tables of the plant forced outage data by system and subsystem.

Appendix C contains an explanation of the procedures used to convert availability data from different sources to a standard consistent form.

3. METHODOLOGY

3.1. BACKGROUND

The final forced outage assessment results are presented as average equivalent forced outage hours per plant year.

3.1.1. Average Equivalent Forced Outage

Average EFOH per plant year is conventional manner to express plant unavailability (Ref. 2-2). This approach is based on the Markov probability model (Ref. 3-1). This model works well when the system or subsystem MTBFs and MTTRs are constant with time and this is a standard assumption in unavailability work. The general unavailability equation as a function of time is given as:

$$UA(t) = 1 - A(t) = 1 - \left[\frac{\mu}{\lambda + \mu} + \frac{\lambda}{\lambda + \mu} e^{-(\lambda+\mu)t} \right]$$

where UA(t) = unavailability as a function of time

A(t) = availability as a function of time

t = time

μ = 1/MTTR = repair rate

λ = 1/MTTF = failure rate

MTTR = mean time to repair

MTTF = mean time to failure

The transient part of the unavailability function decays to zero rapidly. Steady-state system unavailability is given as:

$$UA(\infty) = 1 - \frac{\mu}{\lambda + \mu} = \frac{\lambda}{\lambda + \mu} = \frac{\frac{1}{\mu}}{\frac{1}{\lambda} + \frac{1}{\mu}} \quad .$$

Also,

$$UA(\infty) = \frac{MTTR}{MTTF + MTTR} = \frac{MTTR}{MTBF} = FOR$$

where FOR is the forced outage rate. The steady-state unavailability is used to calculate the plant's equivalent forced outage hours of the plant per year (EFOHs) caused by a system (or module) as follows:

$$EFOH_s = FOR \times [SH/(1 - FOR)] * NM * \%\ LOSS/100$$

where NM is the number of plant modules affected, % loss is the percent loss of plant full power cause by the loss of a system, and SH is the service hours for the plant system. Although each plant system has its own SH, the assessment of plant level availability necessitates that an equivalent or characteristic SH be calculated. In this assessment, two SH were calculated to describe the plant: one for systems/groups involved in residual heat removal and one for all other systems/groups. These calculations are described in Appendix A.

The plant $EFOH_p$ is the sum of all NI and ECA systems/subsystems and is calculated as follows:

$$EFOH_p = \sum_{NI} EFOH_s + \sum_{ECA} EFOH_s \quad .$$

The plant $EFOH_p$, as defined, gives the plant equivalent forced full power outage hours lost averaged over many years of plant operation.

3.1.2. Cumulative Distribution Function (CDF) of Forced Outage

The forced outage goal for the plant includes the cumulative equivalent unavailability from forced outages and investment risk events. The plant forced outage goals are taken from Ref. 2-1 and are listed below:

A. "It shall be a design goal that the calculated equivalent unavailability average over the lifetime of the plant owing to forced outages shall not exceed 10% (average of 876 h/yr)."

B. "Outages of six months or greater shall not contribute more than 10% of the total equivalent unavailability from forced outages, including those not expected to occur in an individual plant's lifetime."

Goal A covers all forced outages and Goal B covers the portion of forced outages which are investment risk events.

The horizontal goal line on Fig. 3-1 is the 10% equivalent unavailability of 876 h. A corner is shown on the goal line at one-tenth of 867 h, or 87.6 h equivalent unavailability, and 6 months or 4380 h individual outage time, labeled MTTR, is the investment risk portion of the overall goal.

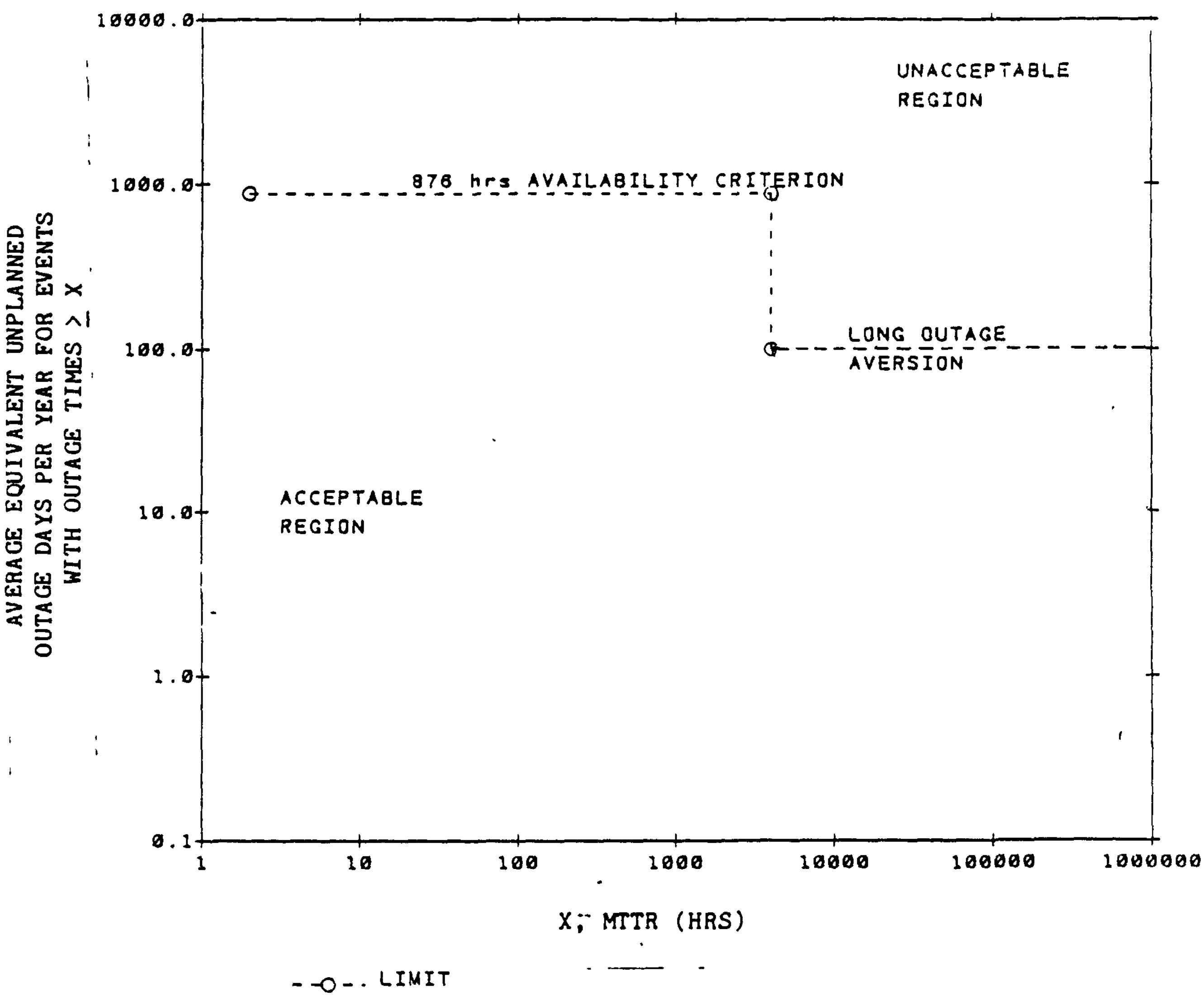

Fig. 3-1. Plant forced outage protection goals

3.2. ASSUMPTIONS

The major assumptions made in this forced outage assessment are listed in this section. Other assumptions made about specific system/ subsystem design features are given in Section 5.

1. The plant is assumed to be representative of a mature design free of early life or design errors. Only independent, random failures representative of the design characteristics of the plant are modeled.

2. On-line repair of components is assumed where possible.

3. Repair times include the time to start up and shut down a module and assume available spare parts, competent personnel, and appropriate tools and fixtures.

4. PLANT DESCRIPTION

The forced outage assessment is based on the standard MHTGR concept described in Ref. 2-1, which is summarized in the following subsections.

4.1. OVERALL PLANT GENERAL DESCRIPTION

The plant utilizes four high-temperature, gas-cooled reactor modules to produce high pressure high temperature steam to generate electricity using two turbine-generators. A common control room is used to operate all four reactors and the turbine plant. In addition, the plant includes systems and service facilities needed to support the reactors and electricity generating equipment.

A schematic flow diagram of the overall plant energy conversion process is shown in Fig. 4.1-1. The diagram shows two of the four reactor modules and one of the two turbine-generators.

A single module, which is the building block of the Standard HTGR, consists of an HTGR core, equipment for transporting heat from the core and for generating steam, and equipment for support and control. The equipment is housed in two welded steel vessels which are connected by a concentric cross-duct vessel. The module interfaces with the power conversion equipment at the inlet and outlet nozzles of the steam generator.

The power conversion equipment consists of two nonreheat power conversion trains as shown in Fig. 4.1-1. Each train includes a turbine-generator set, main steam, condensate, feedwater processing equipment, and interconnecting piping.

The plant is composed of the following systems and system groupings:

- Reactor System (RS)
- Heat Transport System (HTS)
- Shutdown Cooling System (SCS)
- Reactor Cavity Cooling System (RCCS)
- Vessel System (VS)
- Plant Protection and Instrumentation System (PPIS)
- Fuel Handling and Storage System (FHSS)
- Reactor Service Group (RSG)
- Power Conversion Group (PCG)
- Heat Rejection Group (HRG)
- Plant Control, Data, and Instrumentation System (PCDIS)
- Electrical Group (EG)
- Miscellaneous Control and Instrumentation Group (MCIG)
- Plant Service Group (PSG)
- Buildings, Structures and Building Service Group (BSBSG)

4.2. OVERALL PLANT ARRANGEMENT

The MHTGR plant is broken up into two major areas: a Nuclear Island (NI) containing the four reactor modules, and an Energy Conversion Area (ECA) containing the two turbine generators. The NI is separated from the ECA and surrounded by a double-fenced security boundary.

4.2.1. Nuclear Island

The plot plan (Ref. 4-1), consists of four reactor modules with common support facilities. The four modules, each of which produces a thermal output of 350 MW, are cross-headered to feed two turbine generators operating in parallel.

Systems containing radionuclides and "safety-related" systems are contained within the well-defined Nuclear Island. This allows the construction and operation of the other areas in accordance with conventional standards and practices.

Within the NI, each reactor module is housed in adjacent, but separate, reinforced concrete structures located below grade and enclosed by a common maintenance hall. This below-grade location provides significant design benefits by reducing the seismic amplifications typical of above-grade structures. With the exception of essential electrical power, safety systems for each module are independent of other modules and are localized within the individual concrete structures. These include "safety-related," protection and decay heat removal systems. Support functions which are not "safety-related," but which are located in part or entirely within the Nuclear Island include the main and shutdown cooling loops, normal cooling water, ventilation, helium processing, radwaste processing, and fuel handling.

The Reactor Module general arrangement is shown in Ref. 4-2. The reactor vessel is similar in size to a large boiling water reactor (BWR) vessel. The reactor vessel contains the reactor core, reflectors and associated neutron control systems, core support structures, and SCS heat exchanger and motor-driven circulator. The steam generator vessel houses a helically coiled steam generator bundle as well a motor-driven circulator. The pressure-retaining components are constructed of steel and designed using existing technology. The reactor vessel is uninsulated to provide for decay heat removal under accident conditions. The heat is transported to the passive RCCS which circulates outside air within enclosed panels surrounding the reactor vessel.

4.2.2. Energy Conversion Area

The Energy Conversion Area (ECA) envelopes all those facilities not in the Nuclear Island. As shown in Ref. 4-1, the principal structures in the ECA are the Operations Center (including the control room), Turbine Building, and main cooling tower.

Thermal energy from the four reactor modules is delivered to two steam turbine-generators to produce 588 MW(e) gross [538 MW(e) net] of electrical energy. The turbine-generator plant is similar to a modern fossil-fired plant, except that a nonreheat steam cycle is employed. A mechanical draft cooling tower rejects the condenser heat load to the atmosphere.

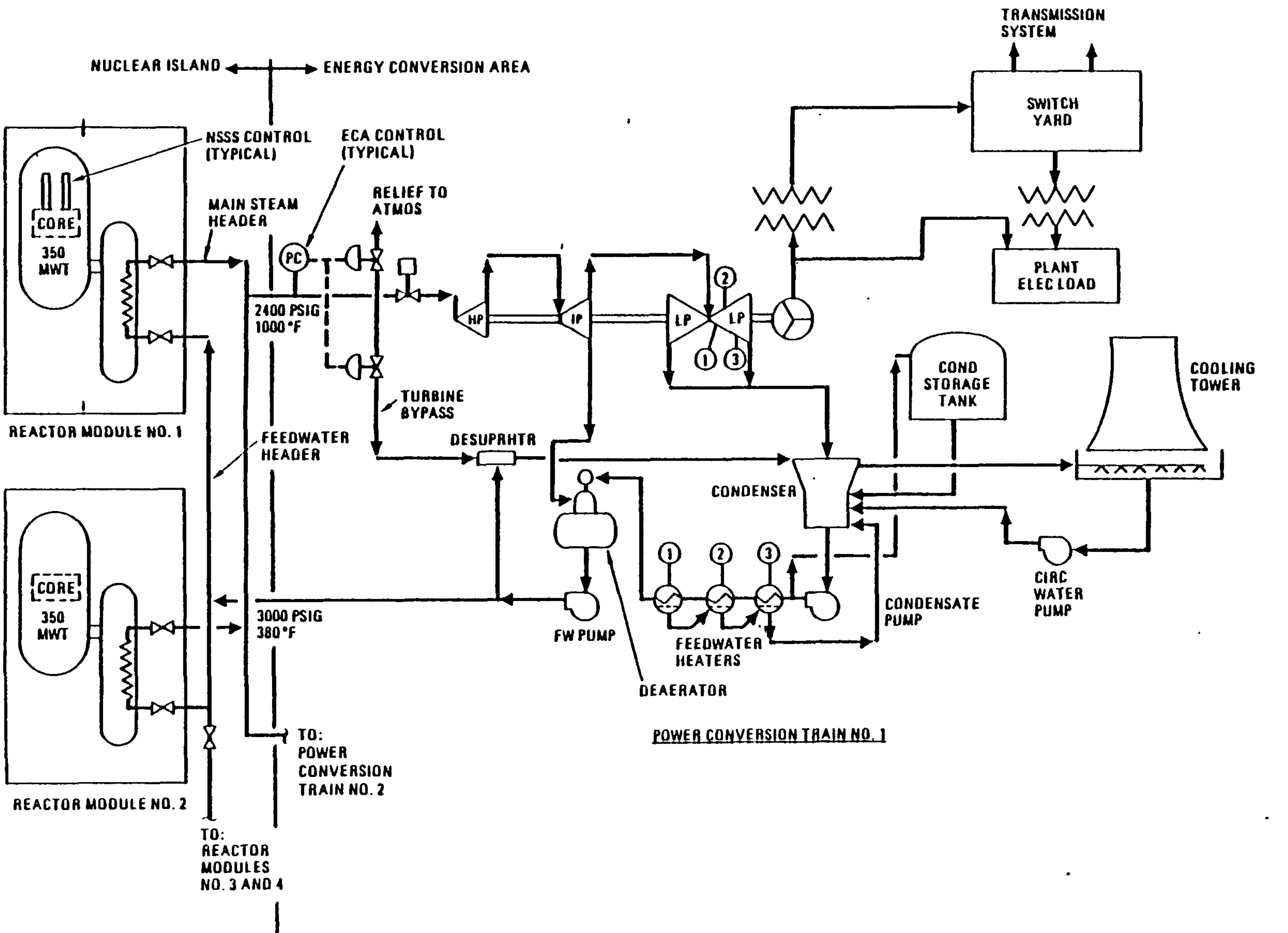

FIGURE 4.1-1
ENERGY CONVERSION FLOW DIAGRAM

4-5

908960/1

5. SYSTEM UNAVAILABILITY CHARACTERISTICS

5.1. REACTOR SERVICE GROUP (RSG)

5.1.1. Helium Purification System (HPS)

1. Summary System Description

This system is a circulating closed loop which removes a bleed stream of contaminated helium from the reactor vessel, and returns purified helium to it. Removal of chemical impurities is the principal purpose of the system. It also removes radioactive impurities and provides purified helium for purge uses (main circulators, shutdown circulators, pressure relief valves, and reactor system). The system consists of high-temperature filters and adsorbers, an oxidizer/cooler module, a dryer module, a low-temperature adsorber module, a compressor module, and two regeneration modules. The HPS operates whenever the reactor operates, as well as during reactor shutdowns to maintain primary coolant purity; however, it can be out of service for about 6 h without affecting module operation. If the HPS is out of service for longer, it may take the primary coolant system above technical specification limits for chemical impurities.

One HPS is provided for each module. Interconnections are provided between two HPS of adjacent modules.

2. Forced Outage Assessment Basis

Reliability block diagrams were prepared of the HPS, with components reliability repair time obtained from Refs. 5-1, 5-2, 5-3 and 5-4. The major contributor to unavailability is radioactive helium leakage through valves and welds.

3. Forced Outage Data

Effect of Failure	100% Loss of Module
MTBF	19,000 h
MTTR	42 h

5.1.2. Gaseous Radioactive Waste System

1. Summary System Description

The Gaseous Radioactive Waste system collects all radioactive and potentially radioactive gas waste generated in the reactor plant, excluding the reactor vessel system and other equipment leakage within the confinement. Normally, potentially radioactive gases are filtered, monitored, and released directly to the reactor building ventilation system. On a high radiation signal, these gases are diverted to the gas waste vacuum tanks for temporary storage, allowing for activity decay followed by release to the reactor building ventilation system.

The gas waste system consists of the following major equipment: two waste gas filters, two waste gas exhaust blowers in the normally nonradioactive loop plus one gas waste vacuum tank with a moisture drain tank, two multistage gas waste compressors (normal and high flow), and three gas waste surge tanks in the process loop.

2. Forced Outage Basis

The waste gas vacuum tanks, moisture drain tanks, diaphragm compressors, and waste gas surge tank are the essential components required for disposal of radioactive gas from the helium purification regeneration module. All components are either redundant, operate on an intermittent basis, or are sufficiently reliable that their contribution to plant forced outage will be small. The three-way valve which interfaces the potentially radioactive and radioactive portions of the system and waste gas vacuum tank are potential single point failures of the system. However, the source of potential radioactive gases are from nonessential systems which can be shut down until the valve or tank is repaired, in the event that high radiation is detected by the monitoring system. The Nuclear Plant Reliability Data System (NPRDS) Report (Ref. 5-5) section Cumulative Generic Gaseous Radioactive Waste Management System shows no failure affecting plant performance. NERC data, (Ref. 5-4) however, provides data for condenser off gas systems in nuclear plants, (component cause code 811 from the component summary report of all nuclear plants, all types, all sizes) and was used for the outage hours estimate. Component cause codes are a NERC numerical breakdown of systems/components to which outages are attributed to.

3. Forced Outage Data

Effect of Failure	100% Loss of Module
MTBF	58,300 h
MTTR	24.5 h

5.1.3. Reactor Service Equipment System

1. Summary System Description

   The Reactor Service Equipment System is comprised of components and structures which facilitate in-vessel and ex-vessel service and maintenance operations, as well as the handling and storage of a number of reactor components. The system contains the: main circulator service equipment, shutdown circulator service equipment, core service tools, auxiliary service cask and transporter assembly, service equipment, and the neutron detector service equipment.

   The circulator service equipment consists of shielded casks, isolation valves, and assorted adapters necessary to provide the capability for removal and replacement of the main and shutdown circulators as well as their respective loop shutoff valves.

2. Forced Outage Basis

   This system's equipment can be repaired while the reactor continues to operate.

3. Forced Outage Data

   No forced outage effect.

5.1.4. Liquid Nitrogen System

1. Summary System Description

   The Liquid Nitrogen System (LNS) provides liquid nitrogen for cooling the low-temperature adsorbers of the HPS and analytical instruments for the noble gas cold trap of the NSSS analytical instrumentation system, and gaseous nitrogen for other NSSS users.

The LNS operates continuously during normal operation and during refueling. It consists of two trains each servicing two reactor modules. Each train has nitrogen recondensers, two liquid nitrogen pumps and one phase separator storage tank. Each recondenser and pump has the capacity to provide 100% of the requirements of its associated train. In the event of a pump or recondenser failure, the faulty equipment can be isolated for maintenance and the nitrogen piped to the 100% back-up unit.

The loss of both pumps and both recondensers of a loop could lead to unavailability of the HPS for the associated two reactor modules. However, the HPS can be out of service for about 6 h without affecting plant operation.

2. Forced Outage Basis

The forced outage assessment is based on the NERC Component Cause Code Report, (Ref. 5-4) Nuclear Units, all types, 1 to 399 MW, Cause Code 226 (Chemical Addition System Reactor Coolant) and the above considerations of redundancy and lack of plant short-term sensitivity to this system.

3. Forced Outage Data

Effect of Failure	Loss of Two Modules
MTBF	372,200 h
MTTR	76 h

5.1.5. Reactor Plant Cooling Water System

1. Summary System Description

   The Reactor Plant Cooling Water System removes heat from the following reactor plant components: HPS coolers and compressors, HPS regeneration coolers and compressors, main helium circulator motor, moisture monitor compressor modules, neutron control assembly, and miscellaneous components.

   This system consists of a cooling water loop which serves these components of all four reactor modules. The system consists of two 100% (one backup) pumps, two 100% (one backup) heat exchangers, one surge tank, and one water chemistry package (filter-demineralizer train). During any of the above modes of operation, one pump and one heat exchanger will be operating while one pump and one heat exchanger serve as backup or are out of service for maintenance or repair. The loss of both pumps or both heat exchangers will shut the plant down.

2. Forced Outage Basis

   The forced outage assessment is based on the NERC Component Cause Code Report, (Ref. 5-6) Nuclear Units, all types, 1 to 399 MW, Cause Code 225 (Component Cooling System), with outages reduced to reflect the conservatism of this design compared to typical LWR's.

3. Forced Outage Data

Effect of Failure	Loss of Four Modules
MTBF	7.4×10^5 h
MTTR	21 h

5.1.6. Helium Storage and Transfer System

1. Summary System Description

The Helium Storage and Transfer System consists of two parts. The first consists of nine high-pressure storage tanks which provide makeup and purge helium to the various helium users, including circulator bearing seals, analysis packages, and cooling system surge tanks. The second, larger part of the system, provides for the lower pressure storage of primary coolant helium in 180 storage tanks. The system serves all four reactor modules.

The low-pressure storage part of the system receives helium from discharge of the HPS and is activated during shut-down and start-up operations only. It is not required to operate continuously. High pressure helium is supplied continuously to all the four reactor modules simultaneously from the high-pressure storage tanks. These tanks are replaced periodically with fresh tanks.

2. Forced Outage Basis

The principal failure mode of this system is the failure of one or both compressors. They are in the low-pressure portion of the system and would not cause shutdown. The high-pressure portion of the system is passive (except for control valves) and, although it supplies bearing seal and makeup helium, it could be out of commission and returned to operation before reactor operations would be affected.

3. Forced Outage Data

No outage effect.

5.1.7. Decontamination Services-System

1. Summary System Description

The Decontamination Services System provides services for general radioactive decontamination, drying, and vacuuming of equipment in the reactor equipment service facility and at the site fuel handling area in the reactor service building. This system consists of two independent decontamination equipment skids, each including the necessary pumps, tanks, vacuum units, and other equipment. The entire four-module plant is served by the two skid-mounted facilities.

2. Forced Outage Basis

The loss of the decontamination pump of a facility is the principal failure mode. The failure of one or both decontamination skids is not expected to have any impact on plant performance.

3. Forced Outage Data

No outage effect.

5.1.8. Hot Service Facility System

1. Summary System Description

   The Hot Service Facility System provides facilities for inspection, maintenance, care, and repair of reactor service equipment and tools. The facilities include a shielded vault with provisions for control of the vault environment; manipulators to perform inspection, maintenance, care, and repair services; and portable decontamination equipment.

   The primary power generation-related function is to provide inspection, maintenance, repair, etc., services to the reactor servicing equipment.

2. Forced Outage Basis

   This subsystem is not directly connected to a plant operating system. Therefore, failure will have no impact on forced outages.

3. Forced Outage Data

   No forced outage effect.

5.2. HEAT TRANSPORT SYSTEM (HTS)

5.2.1. Main Circulator Subsystem

1. Summary Subsystem Description

   The main helium circulator subsystem consists of a helium circulator (axial compressor) with variable speed electric motor drive with magnetic bearings, a main loop shutoff valve, and service systems. The main function of the circulator is to provide sufficient pressure rise to maintain the required helium flow rate. The main loop shutoff valve is designed to prevent the reverse flow of primary coolant through a shutdown circulator. Service systems include one to provide power and control of the magnetic bearings, a main loop shutoff valve service module and an electric motor control and power module.

2. Forced Outage Basis

   The failure rates for variable speed electric motors and their solid state power supplies are from a Westinghouse study (Ref. 5-7). Component reliability and repair time was obtained from Refs. 5-1, 5-2, and 5-3. The major contributors to unavailability are the magnetic bearings, compressor, and electric motor and its associated controller. Preliminary information on magnetic bearings was obtained from James Howden Ltd., an English company (Ref. 5-8).

3. Forced Outage Data

Effect of Failure	100% Loss of Module
MTBF	15,000 h
MTTR	85 h

5.2.2. Steam Generator Subsystem

1. Summary Subsystem Description

The primary function of the Steam Generator is to transfer heat from the primary coolant helium to the secondary coolant (water) and produce superheated steam during normal plant operation. The Steam Generator is also used to remove core heat after a reactor shutdown and during some other operating conditions. It also forms part of the primary coolant pressure boundary.

The steam generator is a once-through tubular unit with economizer, evaporator, and superheater sections. Helium flow is on the shell side and is countercurrent to the water/steam flow.

2. Forced Outage Assessment

The dominant failure modes (resulting in small leaks) of the heat exchangers according to a recent study of HTGR heat exchangers by C-E, are bimetallic weld failure, corrosion/erosion, similar metal weld failure, mechanical damage, fretting, and wear. The frequency of failure for these modes were estimated by C-E (Ref. 5-9). The frequency of large leaks (>1 lb/s) was assumed to be one-tenth that of small leaks (based on LWR experience). The time to restore the module of after a steam generator leak (shut down reactor, find and repair leak, remove moisture and start up) was estimated to be 288 h. This includes 24 h to shutdown a module, 192 h to find and repair the leak, 24 h to restart the module and 48 h to remove moisture from the helium.

3. Forced Outage Data

Effect of Failure	100% Loss of Module
MTBF	71,000 h
MTTR	288 h

5.3. REACTOR SYSTEM

5.3.1. Neutron Control Subsystem

1. Summary Subsystem Description

The Neutron Control Subsystem includes neutron control equipment and neutron monitoring instrumentation. The neutron control equipment includes the control rod drives and control equipment, and the reserve shutdown control equipment. The neutron monitoring instrumentation consists of six ex-core detectors, five in-vessel flux mapping units, and three start-up detector assemblies. The function of this system is to control and monitor the neutron generation rate in the reactor core.

2. Forced Outage Basis

The estimated MTBFs and MTTRs were based on data for similar equipment obtained from Refs. 5-1, 5-2, and 5-3. A module is assumed to operate with one control rod group out of service.

3. Forced Outage Data

Effect of Failure	100% Loss of Module
MTBF	130,000 h
MTTR	70 h

5.3.2. Reactor Core Subsystem

1. Summary Subsystem Description

The Reactor Core consists of fuel elements, graphite reflector elements, reactivity control material (control rods) and reserve shutdown balls and the start-up neutron sources. The Reactor Core generates heat from fissile energy and transfers this heat to the primary coolant. In addition, it provides the primary barrier to fission product release and maintains a geometry that assures adequate cooling and neutron flux control.

2. Forced Outage Basis

Failures of the reserve shutdown balls is not expected to cause forced outage. They are designed and tested to very high temperatures and radiation levels. Failures such as cracking or breakage of the balls will not affect their functional capability and so would not cause a forced outage.

Failures of the start-up neutron sources are also not expected to cause forced outage. They are redundant (3 installed, 2 needed) and after initial operations due to the fission products in the core which are neutron sources.

Failure of the control rods is a possible cause of forced outages. The metallic material in the control rods (i.e., control rod clodding) might possibly have to be replaced/repaired if exposed to excessive temperatures, higher then expected radiation levels or higher than expected coolant contaminants.

Failures of fuel or graphite reflector elements is also a possible cause of forced outage. A preliminary analysis of fuel block cracking during power operation (Ref. 5-10) showed that this could cause four hours per year unavailability. Other cracking analyses of the fuel block during transients (including earthquakes) and graphite reflector blocks during power operation and transients has not been done.

To account for these possible events, engineering pridgment was used to make an estimate of a single long outage (one month) in the life of a plant (40 years).

3. Forced Outage Data

Effect of Failure	100% Loss of Module
MTBF	263,000 h
MTTR	720 h

5.3.3. Reactor Internals Subsystem

1. Summary Subsystem Description

The Reactor Internals Subsystem consists of metallic and graphite components. The major metallic components are the upper plenum thermal protection structure, core support structure, core lateral restraint and hot duct. The major graphite components are the side reflector, lower plenum floor, and lower plenum support structure. The functions of this system are to restrain and support the core, to limit core bypass flow, to route helium flow, and to limit radiation exposure to the reactor vessel.

2. Forced Outage Basis

Since these are structural-type items, they are not expected to fail in the module life, except for the bellows in the hot duct. Failures will, however, take a long time to repair. The bellows failure rate was based on data obtained from English Magnox reactor experience. The restore time was estimated using engineering judgment.

3. Forced Outage Data

Effect of Failure	100% Loss of Module
MTBF	500,000 h
MTTR	1,000 h

5.4. VESSEL SYSTEM

1. Summary Subsystem (All) Description

The Vessel System consists of the Vessel and Duct Subsystem, the Vessel Support Subsystem and the Pressure Relief Subsystem.

The Vessel and Duct Subsystem includes the reactor and steam generator vessels and the connecting cross duct which function as the primary containment for the reactor core, the primary coolant system, and portions of the secondary cooling system. The Vessel Support Subsystem provides support for the reactor vessel (three flexible columns) and the steam generator vessel (two sliding bases and snubbers). The Pressure Relief Subsystem provides the vessels with two complete overpressure relief trains. Each train is sized to provide 100% relief capacity.

2. Forced Outage Basis

The unavailability of the Vessel System is due to radioactive helium leaks, penetration seal leakage, and pressure relief failures. The primary unavailability (~96 h/y) is due to radioactive helium leakage from small, frequent leaks, which was calculated in Ref. 6-1. The estimated MTBFs and MTTRs for penetration seal leakage and pressure relief failures are based on data from Refs. 5-1, 5-2, and Ref. 5-3.

3. Forced Outage Data

Effect of Failure	100% Loss of Module
MTBF	18,600 h
MTTR	250 h

5.5. SHUTDOWN COOLING SYSTEM (SCS)

5.5.1. Shutdown Cooling Water Subsystem

1. Summary Description

The Shutdown Cooling Water Subsystem provides cooling water for removing reactor stored and decay heat on loss of the HTS, removing reactor parasitic heat losses during reactor operations, and removing heat from the shutdown circulator motors during reactor operation. The system removes heat from the SCS via shutdown cooling heat exchangers and from shutdown circulator motors via cooling coils. It rejects the heat to the service water subsystem through the shutdown cooling water heat exchangers. This subsystem consists of a closed cooling loop with two 100% (one backup) pumps, one 15% jockey pump, and two 50% heat exchangers. The 100% capacity pumps remove heat during conduction cooldown conditions while the jockey pump removes a much smaller heat load during normal reactor power operation and during shutdown conditions. The system serves all four reactor modules. If, during normal operation, both 100% capacity pumps and heat exchangers fail, the jockey pump can carry the heat load under favorable SCWS cold leg water temperature conditions, (e.g., 70°F).

2. Forced Outage Basis

The forced outage assessment is based on the NERC Component Cause Code Report (Ref. 5-6) Nuclear Units, all types, 1 to 399 MW, Cause Code 225 (Component Cooling System), with outages per year reduced to reflect the conservatism of this design compared to typical LWR's.

3. Forced Outage Data

Effect of Failure	Loss of Four Modules
MTBF	740,000 h
MTTR	21 h

5.5.2. Other SCS Subsystem

1. Summary Subsystem Description

The other SCS subsystems are the:

Shutdown Circulator Subsystem, the Shutdown Cooling Heat Exchanger Subsystem, and the Shutdown Cooling Heat Removal Control Subsystem.

The SCS function is to provide decay heat removal cooling when the reactor is shut down. During pressurized or depressurized conditions, the SCS is required to operate when the HTS is unavailable. If the SCS does not operate during this condition the RCCS provides decay heat removal.

The Shutdown Circulator consists of the following components, (1) the shutdown circulator assembly, (2) the shutdown loop shutoff valve and ducting assembly, (3) the shutdown circulator service module, (4) the shutdown loop shutoff valve service module and (5) the electric motor control module. The shutdown circulator has a centrifugal compressor driven by an electric motor which is supported by magnetic bearings.

The Shutdown Heat Exchanger consists of a helium to water shell-and-tube, cross-counterflow heat exchanger using helical tubes.

The Shutdown Cooling Heat Removal Control is a load commutated variable frequency power supply with digital microprocessor-based controls.

2. Forced Outage Basis

The primary contributor to SCS unavailability is due to water leaks in the shutdown heat exchanger. An other source of unavailability is due to periodic testing of the SCS.

3. Forced Outage Data

Effect of Failure	100% Loss of Module
MTBF	105,000 h
MTTR	80 h

5.6. FUEL HANDLING AND STORAGE SYSTEM (FHSS)

1. Summary Subsystems (All) Description

The FHSS consists of the following subsystems: core refueling, site fuel handling, and spent fuel cooling.

The functions of the FHSS are to handle and store fresh or irradiated within the plant, receive fresh fuel, prepare irradiated fuel for shipment offsite and provide irradiated fuel cooling on-site.

2. Forced Outage Basis

The FHSS equipment can be repaired while the reactor continues to operate since it is not used until a refueling shutdown (except for the spent fuel cooling subsystems). Spent fuel cooling subsystem failures are not expected because of the redundancy provided.

3. Forced Outage Data

No forced outage effect.

5.7. PLANT PROTECTION AND INSTRUMENTATION SYSTEM (PPIS)

5.7.1. Safety Protection Subsystem

1. Summary Subsystem Description

The purpose of the Safety Protection Subsystem is to protect public health and safety. To do this, it is designed to meet IEEE Standard 603. The SPS includes the following functions: (1) reactor trip, which initiates automatic insertion of the outer control rods, (2) reactor trip and automatic insertion of the reserve shutdown balls (in case the control rods are not inserted), (3) steam generator isolation and (4) main circulator trip.

2. Forced Outage Basis

The MTBFs and MTTRs are based on information from Refs. 5-1, 5-2, and 5-3 and the current status of subsystem design.

3. Forced Outage Data

Effects of Failure	100% Loss of Module
MTBF	10,000 h
MTTR	12 h

5.7.2. Investment Protection Subsystem

1. Summary Subsystem Description

The purpose of the Investment Protection Subsystem is to measure plant process variables and to detect abnormal module conditions which will protect major plant equipment. This subsystem includes the following functions: (1) reactor trip using inner control rods, (2) HTS shutdown which limits the temperatures of the steam generator, (3) shutdown cooling system initiation, (4) steam generator dump, (5) primary coolant pressure pumpdown and (6) SCS heat exchanger isolation and drain. This subsystem includes all equipment from the process sensors to the actuation devices.

2. Forced Outage Basis

The MTBFs and MTTRs are based on information from Refs. 5-1, 5-2, and 5-3 and the current status of subsystem design.

Effects of Failure	100% Loss of Module
MTBF	36,000 h
MTTR	34 h

5.7.3. Special Nuclear Area Instrumentation Subsystem

1. Summary Subsystem Description

The Special Nuclear Area Instrumentation Subsystem provides preventative features, monitoring of safety systems status, and monitoring the modules. The SNAI includes the following functions: (1) reactor vessel pressure relief valve/ block valve closure interlock, (2) safety system information displays, (3) investment protection information displays, and (4) post-accident monitoring instrumentation.

2. Forced Outage Basis

A forced outage of 1 h per year was used based on engineering judgment and LWR experience.

3. Forced Outage Data

Effects of Failure	100% Loss of Module
MTBF	89,000 h
MTTR	12 h

5.8. PLANT CONTROL, DATA AND INSTRUMENTATION SYSTEM (PCDIS)

5.8.1. Supervisory Control/Data Management Subsystems

1. Summary Subsystem Description

The supervisory controls automatically coordinate the balancing of load levels. They automatically determine what contribution each module will make to the total steam flow. They also automatically determine the contribution of each turbine generator to the total plant electrical output.

The data management subsystem provides plant wide data communication and centralized data processing.

2. Forced Outage Basis

The MTBF and MTTR information was obtained from Ref. 5-11.

3. Forced Outage Date

Effects of Failure	100% Loss of Module
MTBF	6,500 h
MTTR	10 h

5.8.2. NSSS Control Subsystem

1. Summary Subsystem Description

The purpose of the NSSS Control Subsystem is to control reactor conditions and supply of steam for each reactor module. This subsystem includes process instrumentation and controls and interface equipment to communicate with the plant control data highways. The NSSS control includes regulation of reactor power to control helium temperature, loop steam temperature, and pressure delivered by each module during startup, normal operation, transients, and to regulate reactor conditions during shutdown.

2. Forced Outage Basis

The MTBF and MTTR are based on information from Refs. 5-1, 5-2, and 5-3 and the current subsystem design.

3. Forced Outage Data

Effects of Failure	100% Loss of Module
MTBF	7,500 h
MTTR	12 h

5.8.3. ECA Control (All Controlled ECA Systems)

1. Summary Description

The Energy Conversion Area (ECA) control is distributed in all the ECA systems/groups requiring control. This is not a subsystem, but is now a place to gather the downtime due to failures of these distributed ECA controls. The major portion of the ECA control is involved with controlling the two trains in the ECA (i.e., T-G sets, feedwater/condensate, circulating water, etc.). In the ECA, control is provided for main steam pressure, feedwater pump speed, balance of feedwater/steam flow between each train, and turbine load response. During startup/shutdown control is provided for the feedwater temperature and flow.

2. Forced Outage Data

The MTBF and MTTR information was provided by SWEC (Ref. 1-1).

3. Forced Outage Data

Effects of Failure	100% Loss of Module
MTBF	5,600 h
MTTR	12 h

5.9. MISCELLANEOUS CONTROL AND INSTRUMENTATION GROUP (MCIG)

5.9.1. NSSS Analytical Instrumentation System

1. Summary System Description

The NSSS analytical instrumentation system provides surveillance data to verify that the unit meets technical specifications. It measures chemical and radioactive impurities in the primary coolant, helium service, and gaseous waste systems.

2. Forced Outage Basis

Engineering judgment was used to estimate the MTBF and MTTR based on similar LWR experience (Ref. 5-4).

3. Forced Outage Data

Effects of Failure	100% Loss of Module
MTBF	30,000 h
MTTR	12 h

5.9.2. Radiation Monitoring System

1. Summary Description

   The Radiation Monitoring System consists of area monitors, airborne monitors, and process monitors located throughout the plant and at the site boundary. Certain monitors are located at specific areas in the plant, in plant effluents, and at the site boundary. A central radiation processor includes control and monitoring consoles and instrumentation cabinets located in the reactor service building. All displays and alarms are provided in the main control room and in the health physics/access control area in the personnel service building. This system has no power generation functions.

2. Forced Outage Basis

   Since it has no power generation function, failure will not cause a plant outage.

3. Forced Outage Data

   No forced outage effect.

5.10. OPERATOR AND MAINTENANCE ERRORS

1. Summary Description

   Operator and maintenance errors occur in plant operation. Recognizing this, an allowance for the historic amount is included (based on LWR experience).

2. Forced Outage Basis

These errors were estimated using data from "NERC Equipment Availability Reports (Ref. 5-4) - Nuclear Plants of All Sizes and Types, and Cause Codes 915 and 916".

3. Forced Outage Data

Effects of Failure	100% Loss of Module
MTBF	10,800 h
MTTR	78 h

5.11. POWER CONVERSION GROUP (PCG)

The PCG consists of the following systems and support systems:

- Turbine-generator and auxiliaries system.
- Main and bypass steam system.
- Feedwater and condensate system.
- Extraction and auxiliary steam system.
- Heater drains and condensate returns system.
- Condensate polishing system.
- Turbine building closed cooling water system.

Support Systems:

- Chemical feed system.
- Steam vents and drains system.
- Turbine plant sampling system.
- Startup and shutdown system.
- Steam and water dump system.

5.11.1. Turbine-Generator and Auxiliaries System

5.11.1.1. Turbine-Generator

1. Summary Description

The twin turbines for the 4 x 350 MW(t) plant are a 3600 rpm nonreheat design. Throttle conditions are 2400 psig/1000°F. Each tandem compound turbine contains one single flow high pressure one double flow intermediate pressure and one double flow low pressure turbine element.

Each of the two turbine-generators is nominally rated at 300 MW(e). Each generator has a 0.90 PF with 25,000 V, 3-phase, 60 Hz output and is totally enclosed and hydrogen cooled. The stator is wye-connected. Each neutral is high-resistance grounded through a distribution transformer. The excitation is furnished by shaft-driven alternators, the output of which are rectified by solid-state components and fed to the rotating fields of the generators.

2. Forced Outage Basis

The turbine-generator's forced outage rate was calculated from historical data specially compiled by AER from the Ref. 5-6, NERC/GADS Component Summary Report for fossil units. All the applicable turbine Component Codes 600 to 699 were used. All the applicable generator Component Codes 700 to 799 were used with the exception of Code 708 which represents generator "rotor windings."

The data presented for forced outages associated with the rotor windings for this survey grouping contained excessively high forced outage hours. This is not consistent with any of the data reporting for other generator size ranges for this NERC/CADS Component Cause Code. As was stated by Utility Power Corporation as part of its HTGR project trade study on availability of turbine-generators, "some of the overbearing causes of outage were random as well as others which will be precluded by modern designs." By judiciously eliminating Code 708 from this data set, the generator combined full forced plus derated hours becomes approximately 41 EFOH (in lieu of approximately 130 EFOH with Code 708 included). This result seems both reasonable and consistent with Utility Power Corporation reporting of approximately 35 EFOH for generators in the 200 to 299 MW unit size range.

3. Forced Outage Data

Effect of Failure	100% Loss of Load of One Turbine-Generator Set	10% Loss of Load of One Turbine-Generator Set
MTBF	3950 h	2200 h
EMTTR	97 h	11 h

5.11.2. Main and Bypass Steam System

1. Summary System Description

The main steam system provides a continuous steam supply to the turbine at 2400 psig, 1000°F. The main steam system conves steam from the steam generators to the HP turbines and provide a means of isolation and overpressure protection for each steam generator. The bypass steam system provides an alternate steam flow path from the steam generators during operating transient conditions by dumping steam to the condenser and/or to the atmosphere. Further, this system provides alternate flow paths for start-up operations, cold cleanup, hot cleanup and deaeration, and system warmup and cooldown operations. The turbine bypass system is equipped with pressure reducing and desuperheating stations to automatically condition main steam before dumping steam into the condenser in the event of a load rejection or during plant startup and shutdown. The main and bypass steam systems consist of steam line isolation valves, safety valves, and turbine bypass desuperheater stations.

2. Forced Outage Basis

The forced outage assessment is based on the normal power operations. Forced outage data is based on NERC Component Summary Report, Ref. 5-4 fossil units, all fuel types, 300 to 399 MW, Cause Codes 112 (desuperheaters and attemperators), 115 (safety valves), 116 (steam valves and piping), and 148 (start-up system superheater or turbine system).

3. Forced Outage Data

Effect of Failure	100% Loss of Load of One Turbine-Generator Set	25% Loss of Load of One Turbine-Generator Set
MTBF	31,300 h	15600 h
EMTTR	40 h	14 h

5.11.3. Feedwater and Condensate System

1. Summary System Description (Except Condenser)

The feedwater and condensate system transfers condensate from the condenser, preheats the feedwater, and delivers the required volume of feedwater to the inlet of the steam generator at 3000 psia and 380°F.

Two 100% capacity condensate pumps take suction from the condenser hotwell and pump the condensate through the polishing demineralizers and the low-pressure heaters to the deaerator.

Four stages of feedwater heating are provided to heat the feedwater being returned to the steam generators. Heaters are placed in series and operate under increasing pressure of various stages of turbine extraction steam. Heaters Number 2, 3, and 4 are 100% capacity and are of the closed-type, horizontal, U-tube arrangement. The Number 1 heater is a 100% capacity direct-contact, deaerating type. The fourth point feedwater heaters are installed in the condenser necks. Some additional heating is provided by a single 100% capacity gland exhaust condenser which is located between the condensate pumps and the fourth stage heaters and by a drain cooler which is located between the third and fourth point heaters.

Two 80% capacity, motor-driven, variable speed steam generator feedpumps are provided for each of the two plant feedwater heater strings. These take suction from the deaerator and deliver feedwater to the steam generators.

2. Forced Outage Basis (Except Condenser)

Forced outage estimates were based on historical data specially compiled by AER from Ref. 5-6 NERC/GADS Component Summary Report to reflect the HTGR BOP system. The Cause Codes used were 804 (condensate pumps), 901 (feedwater heaters - leaking), 902 (feedwater heaters - dirty), 918 (feedwater pump drive - electric), 922 (boiler feed pump), 929 (feedwater valves), 117 (valves and piping - feedwater and blowdown), and 817 (condensate cooler).

3. Forced Outage Data (Except Condenser)

Effect of Failure	100% Loss of Load of One Turbine-Generator Set	50% Loss of Load of One Turbine-Generator Set
MTBF	13,200 h	1,300 h
EMTTR	32 h	12 h

1. Summary Description - Condenser

Each of the two condensers for the 4 x 350 MW(t) plant are a single-shell, double-pass design that normally operate at 2.5 in. Hg abs. The condensers are designed to handle the normal heat rejection loads for the plant. The condensers receive steam from the main steam line through the flash tank during startup, shutdown, and while maintaining the plant at a minimum load during periods of very low demand. Two 100% capacity, mechanical, vacuum pumps are supplied for removing noncondensible gases from the condensers. The vacuum pumps are motor-driven, rotary, two-stage units.

2. Forced Outage Basis - Condenser

The FOR was calculated from historical data specially compiled by AER from Ref. 5-6 NERC/GADS Component Summary Report to reflect the HTGR BOP system. Condenser related codes in the 800 to 899 series were used. The following codes were considered inappropriate to the HTGR BOP system or were considered elsewhere in the report: 803 (circulating water pump); 804 (condensate pump); 811 (off-gas system); 812 (cooling tower); 813 (traveling screens); 816 (circulating water pipe); and 817 (condensate cooler).

The Cause Codes used were 800 (general), 801 (cleaning), 801 (tube failure), 805 (air removal pumps), 806 (shell), 807 (condenser control), 808 (air leakage), 809 (expansion joint), 810 (inspection), 814 (loss of vacuum), 815 (high condenser temperature), and 899 (miscellaneous).

Loss of the main condensers would lead to loss of main loop cooling and would challenge the SCS. A condenser tube leak could introduce contaminated water into the steam generators with significant risk of damage to the generators. Tube leaks represent the cause of 50% of both condenser full forced and forced derated outage hours each year.

3. Forced Outage Data Condenser

Effect of Failure	100% Loss of Load of One Turbine-Generator Set	20% Loss of Load of One Turbine-Generator Set
MTBF	26,000 h	1,200 h
MTTR	16 h	2.2 h

5.11.4. Extraction and Auxiliary Steam System

1. Summary System Description

For every 10°F rise in feedwater temperature, there is an approximate reduction of 1% in the heat that must be added to make steam. Low quality steam is therefore extracted from the turbines and used to heat the feedwater in the feedwater heaters. The differential pressure between the turbine extraction steam and the condensing steam in the feedwater heater provides the motive force to produce the required flow rate in the extraction steam system.

An auxiliary steam system is provided consisting of one 100% capacity electric boiler including feed pumps, deaerators, blowdown tank, and condensate transfer pumps. It supplies steam as required from either the auxiliary electric boiler or the main steam and bypass system low pressure flash tanks to the following loads:

- Heating of reactor plant cooling water and core auxiliary cooling water systems.
- Turbine gland steam, when main steam is unavailable.
- Deaerating feedwater heater during startup.
- Condenser hot well sparger.
- Central hot water heating system heat exchangers.

The auxiliary boiler is required for unit startup after any outage. The remainder of the system is required during normal power operations.

2. Forced Outage Basis

Failure rates for the system components are based on historical performance of similar components given in NERC Component Summary Report, Ref. 5-4 fossil units, all fuel types, 300 to 399 MW, for component Code 160 (auxiliary boiler). Component Codes 633 (gland steam controller) and 649 (water induction) were taken from the AER data compiled from Ref. 5-6 NERC/GADS Component Summary Report. The auxiliary steam piping and valving are included as part of the main steam and bypass system forced outage assessment. The data suggests almost no derated outage hours.

3. Forced Outage Data

Effect of Failure	100% Loss of Load of One Turbine-Generator Set
MTBF	212,000 h
EMTTR	78 h

5.11.5. Heater Drains and Condensate Returns System

1. Summary System Description

The heater drains and condensate returns system transports condensate from the condensate drain tanks and from feedwater heaters No. 2, 3, and 4. Condensate drain pumps are provided to transport auxiliary steam system condensate from the condensate drain tanks to the condensate system. Heater drains from the second and third point heaters contain control and bypass valves to control heater water level. It was assumed that a failed open heater drain valve would not reduce appreciably the power generating capability of the plant, provided operator action corrected heater level deficiencies before major damage occurred to the heaters. Heater levels are monitored and alarmed in the control room.

2. Forced Outage Basis

It was judged that operator error would be the major contributing factor to plant forced outages caused by the heater drain and condensate return system. Code 642 (piping - steam drain and gland) was used to assess forced outages due to operator error. The FOR for this cause code was calculated from the data compiled by AER from Ref. 5-6 NERC/GADS Component Summary Report. This data suggests no derated outage hours.

3. Forced Outage Data

Effect of Failure	100% Loss of Load of One Turbine-Generator Set
MTBF	323,500 h
EMTTR	50 h

5.11.6. Condensate Polishing System

1. Summary System Description

An externally regenerated, deep-bed type condensate polishing system is provided to remove ionic impurities from the condensate. Three polisher vessels, each designed to treat 50% of the total condensate flow, are provided. Equipment for external regeneration of resins includes a cation regeneration vessel; anion regeneration vessel, resin mix and storage tank; caustic dilution water heater; acid, caustic, and ammonia storage tanks and feed equipment; sample sink; and a control panel. To improve condensate polisher system performance and reduce waste volumes, ammoniation equipment is provided in addition to acid and caustic regeneration. This arrangement will reduce potential sodium leakage and allow potentially longer service runs between regenerations. The condensate polishing system is located in the turbine building. The system consists of a condensate discharge valve, three 50% capacity polishing demineralizers and discharge valves. A normally closed bypass valve is also provided. This bypass valve is considered essential to the plant operation. If the valve fails open, the polishing demineralizers are bypassed, subjecting the once-through steam generators to nontreated water. One polishing tank is off-line for regeneration during plant operation. The two on-line 50% capacity tanks are necessary for 100% power generation.

2. Forced Outage Basis

The polishing tanks were assumed to contribute little to plant forced outage. Failure of the demineralizer discharge valve will dominate the forced outage of the system. Data compiled from the NERC/GADS Component Summary Report, Ref. 5-4, PWR units, all size ranges, for cause code 908 (boiler water condition) was judged to be the most representative for calculating the forced outage data of the system. The Nuclear Plant Reliability Data Systems (NPRDS) (Ref. 5-5), Section A03-demineralizers was reviewed for data on demineralizer failure rates, and reported no failures in 114 operating years.

3. Forced Outage Data

Effect of Failure	100% Loss of Load of One Turbine-Generator Set
MTBF	38,300 h
EMTTR	13 h

5.11.7. Turbine Building Closed Cooling Water System (TBCCWS)

1. Summary System Description

The TBCCWS supplies cooling water for a diverse set of loads, including condensate pumps, generator cooling, and cooling for the station air compressors. It is a closed loop system with two 100% motor-driven water pumps, head tank and two 100% capacity heat exchangers providing cooling to various turbine plant components for each of the two units in the plant. This system dissipates heat to the cooling tower by way of the circulating water system.

2. Forced Outage Basis

Component forced outage data is based on data compiled by AER from Ref. 5-6 NERC/GADS Component Summary Report (cause Code 646) No full forced or derated outage hours are reported.

3. Forced Outage Data

No forced outage effect.

5.11.8. Power Conversion Group Support Systems

5.11.8.1. Chemical Feed System

1. Summary System Description

Chemical feed equipment is provided to adjust the pH of the condensate and remove oxygen from the feedwater. Ammonia feed equipment is provided for pH adjustment and includes a day tank and three 100% chemical metering pumps. Hydrazine feed equipment is provided for oxygen scavenging and includes a day tank and three 100% chemical metering pumps. Hydrazine and ammonia are stored in drums and transferred to the day tanks by hand pumps. Chemical feed equipment is located in the turbine building.

2. Forced Outage Basis

The chemical feed system controls the water quality of the condensate and feedwater system by controlling water pH and removing oxygen. Failures in the chemical feed system, if not immediately corrected, may result in forced outage of the condensate and feedwater system. The system forced outage calculations were based on NERC Component Summary Report

Ref. 5-4 fossil units, 300 to 399 MW, cause Code 907 (water treatment system).

3. Forced Outage Data

Effect of Failure	100% Loss of Load of One Turbine-Generator Set	10% Loss of Load of One Turbine-Generator Set
MTBF	313,200 h	27,200 h
EMTTR	41 h	3.2 h

5.11.8.2. Steam Vents and Drains System

1. Summary System Description

The steam vents and drains system consist of a number of individual piping runs with associated stop and control valve/traps. These lines direct steam or water to the condenser from components such as feedwater heater relief valves and main steam line low point drain points.

2. Forced Outage Basis

Forced outage estimates were based on AER data compiled from the Ref. 5-6 NERC/GADS Component Summary Report, cause Code 642 (piping, steam drain, and gland). It was assumed that failure of steam vents and drains by clogging could lead to a ruptured component in the turbine plant system causing a plant forced outage. This data shows no forced deratings.

3. Forced Outage Data

Effect of Failure	100% Loss of Load of One Turbine-Generator Set
MTBF	323,500 h
EMTTR	50 h

5.11.8.3. Turbine Plant Sampling System

1. Summary System Description

A turbine plant sampling system is provided to monitor condensate, feedwater, steam generator water, main steam, and HP turbine exhaust steam conditions. The sample piping directs and controls the flow of the steam or water to a central sample station.

2. Forced Outage Basis

No measurable impact on forced outage is expected.

3. Forced Outage Data

No forced outage effect.

5.12. HEAT REJECTION GROUP (HRG)

The HRG consists of the following system:

- Circulating water system.
- Service water system.
- Circulating water makeup and blowdown system

5.12.1. Circulating Water System

1. Summary System Description

The circulating water system provides cooling for the condensers and for the TBCCWS heat exchangers. The circulating water system is a closed loop system that consists of the cooling tower, circulating water pumps, condensers, and TBCCWS heat exchangers. Its main function is to convey the heat load rejected by the condenser to the cooling tower.

Two 50% circulating water pumps are provided for each condenser. The pumps are vertical mixed flow type pumps which deliver circulating water to the condenser and the TBCCWS heat exchangers.

A multi-cell type mechanical draft, wet cooling tower is provided. The tower is sized to provide the cooling requirements for the circulating water system and the service water system.

2. Forced Outage Basis

Component forced outage data is based on data compiled by AER from Ref. 5-6 NERC/GADS Component Summary Report, (cause Codes 803 (circulating water pump), 816 (circulating water system screens pipe, etc.), and 812 (cooling towers).

3. Forced Outage Data

Effect of Failure	100% Loss of Load of One Turbine-Generator Set
MTBF	10,100 h
EMTTR	16 h

5.12.2. Service Water System

1. Summary System Description

   The service water system is a closed-loop cooling system that picks up heat from the four NI cooling water systems. Two 100% capacity, vertical, wet pit pumps provide service water. The pumps take water from the cooling tower basin and circulate it through four 200% capacity shell and tube heat exchangers servicing the four NI cooling water systems.

2. Forced Outage Basis

   The forced outage assessment is based on NERC Component Summary Report, Ref. 5-4 nuclear units, all sizes and fuel types, cause Code 909 (service water). NERC data suggest almost no full forced outage hours. All the data was summed and treated as forced deratings and reported as an equivalent 30% loss of load.

3. Forced Outage Data

Effect of Failure	30% Loss of Load of One Turbine-Generator Set
MTBF	583,000 h
EMTTR	5 h

5.12.3. Circulating Water Makeup and Blowdown System

1. Summary System Description

The heat rejection group cooling water systems lose water through evaporation in the cooling tower, blowdown from the cooling tower and to a lesser extent through system leakage. Two 100% vertical pumps are provided for the circulating water makeup system. The pumps are located in the intake structure adjacent to the river. Two traveling screens are provided, each suitable for 50% of the flow requirements. Servicing the traveling screens are two 100% capacity screen wash pumps. The screens are protected by a bar rack and trash screen. Chlorination is provided at the makeup water intake structure to protect makeup water pipelines from biofouling. Two evaporators and two chlorinators, one operating and one spare, are provided. Also included are injectors, diffusers, and chlorine storage facilities. Chlorine will be injected in the makeup water intake structure pump bays.

Control of solids in the cooling tower water basin is provided by discharging a portion of the high solids concentration basin water and replacing it with low solids concentration makeup water. Flow control is provided by an overflow into the discharge pipe.

Chlorination is provided for the circulating water system to protect the main condenser and cooling tower fill from biofouling. Two evaporators (one operating and one spare) and two chlorinators (one operating and one spare) are provided. Also included are injectors, diffusers, and chlorine storage facilities. Chlorine will be injected in the pump bays immediately upstream of the circulating water pumps.

2. Forced Outage Basis

The heat rejection group subsystems are not expected to have a significant impact on forced outage hours.

3. Forced Outage Data

No forced outage effect.

5.13. BUILDING, STRUCTURES, AND BUILDING SERVICE GROUP (BSBSG)

This group consists of the following systems:

- Reactor building.
- Reactor service building.
- Reactor auxiliary building.
- Operations center.
- Standby power building.
- Radioactive waste management building.
- Turbine building.
- Fire pump house.
- Miscellaneous site building structures, personnel service, warehouse and maintenance, helium storage.
- Cooling tower basin and circulating water pump intake.
- Intake pump house and discharge structure.
- Makeup water treatment and auxiliary boiler building.

Building and structures are not considered to impact forced outage.

5.13.1. Operations Center (Control Room Area)

1. Summary System Description

The control room area HVAC systems serve the control room, the computer room, and the mechanical equipment room.

The control room area HVAC subsystem consists of two 100% air-conditioning units with filters and chilled water cooling coils, return/purge exhaust fan and electric heat coils.

2. Forced Outage Basis

Nuclear Plant Reliability Data System (Ref. 5-5) Reports A02 and A03 show no forced outages have been attributed to this system.

3. Forced Outage Data

No forced outage effect.

5.14. PLANT SERVICE GROUP (RSG)

This group consists of the following systems:

- Potable water.
- Storm drainage.
- Sanitary drainage and treatment.
- Plant fire protection.
- Waste water treatment.
- Auxiliary boiler.
- Raw water treatment.
- Instrument and service air.
- Central hot water heating.
- Plant drains.
- HVAC (ECA and NI).
- Demineralized water makeup.
- Station chilled water.

Only the auxiliary boiler system, instrument and service air system, and the Nuclear Island HVAC system are expected to have a measurable effect on forced outage.

5.14.1. Auxiliary Boiler System

This system has been included with the extraction and auxiliary steam system Section 5.11.4, in the power conversion group.

5.14.2. Instrument and Service Air System

1. Summary System Description

   The instrument and service air system supplies air for the entire plant. Two 100% compressors, receivers, and dryers are provided for each of the two turbine-generator units.

2. Forced Outage Basis

   The forced outage data was based on NERC Component Summary Report, Ref. 5-4 fossil units, all fuels, 600 to 799 MW, Cause Code 928 (station service air). Full forced outage data is reported. The data base contains insignificant forced derating outage hours.

3. Forced Outage Data

Effect of Failure	100% Loss of Plant Load
MTBF	313,250 h
EMTTR	9 h

5.14.3. Nuclear Island HVAC System

1. Summary System Description

The NI portion of the HVAC subsystem provides the following:

a. Once-through conditioned supply air and monitored/filtered exhaust, supplemented by local cooling, for the reactor auxiliary building and for each reactor building to maintain appropriate environmental conditions for equipment and personnel as follows:

(1) Reactor cavity - during shutdown.

(2) Steam generator cavity - continuous operation of unit cooler with once-through air during shutdown.

(3) Upper levels - continuous once-through air.

b. Once-through conditioned supply air to the reactor service building to ensure equipment operability and personnel comfort and monitored/filtered exhaust from the potentially radioactive areas.

c. Ventilation and heating for the reactor maintenance enclosure to maintain personnel comfort and protection during refueling and maintenance operations.

d. Once-through conditioned supply air to the personnel service building for personnel comfort, and monitoring/filtering of the air exhausted from the potentially radioactive rooms.

e. Backup, highly reliable recirculating air cooling for all rooms containing electrical equipment essential to protection of the public and continuous exhaust for all battery rooms subject to hydrogen generation.

The primary power generation function of the HVAC subsystem is to provide ambient environmental conditions required by the equipment and activities in the various NI buildings.

2. Forced Outage Basis

The NERC Component Cause Code Report, Ref. 5-6 Nuclear Units, PWR Type, Steam Generator Portion of Unit, 1 to 399 MW, Cause Code 242 (reactor containment cooling and ventilation) was used for a historical data base. The only record of shutdowns for HVAC failure was in the size range 800 MW and above, probably because of the larger number of units in that size range.

3. Forced Outage Data

Effect of Failure	Loss of One Module
MTBF	38,000 h
EMTTR	36 h

5.14.4. Raw Water Treatment System

1. Summary System Description

Chlorinated water from the circulating water makeup system is also provided to the raw water treatment system. Water is clarified, and filtered water is then pumped from the filter clear well to the makeup demineralizer system. Prior to being processed by the makeup demineralizer system, sodium sulfite is added for dechlorination. Sodium sulfite feed equipment consists of a day tank and two 100% chemical metering pumps. Granular or powdered sodium sulfite is stored in bags and manually added to the day tank. Clarified and filtered water is processed through two trains of demineralizers, each designed to treat 50% of the system design flow. Each train consists of a strong acid cation vessel, strong-base anion vessel, and mixed-bed vessel. A common degasifier is provided for both trains and removes oxygen and carbon dioxide. Demineralizer vessels are regenerated in place. Acid and caustic regeneration equipment, caustic dilution water heater, and acid and caustic storage tanks are provided. All of the raw water treatment equipment is housed in a water treatment building located adjacent to the machine shop and warehouse. Enough demineralized water tank storage is provided to ensure an uninterrupted supply to the condensate and feedwater system during expected transients.

2. Forced Outage Basis

The makeup water pumps and associated piping is addressed in the circulating water makeup and blowdown system, Section 5.12.3, as part of the heat rejection group. No measurable impact on forced outage is expected from the raw water treatment system because of the storage capacity of the tankage.

3. Forced Outage Data

No forced outage effect.

5.14.5. Other Mechanical Service Systems (General)

1. Summary System Description

NERC provides additional data reports for other mechanical systems and components as a general group.

2. Forced Outage Basis

To account for forced outages in BOP systems which may not have been included with outage hours reported in the other sections of this report, NERC cause codes in the 900 to 999 series were reviewed. The majority of these cause codes are either addressed elsewhere in this report or are inappropriate to the HTGR BOP. However, the following cause codes are judged to be appropriate for inclusion: Code 910 (computer), and Code 912 (cooling water limitations). Component outage data for these codes is based on the Ref. 5-4 NERC/GADS Component Summary Report, fossil fuel - all types, 300 to 399 MW.

3. Forced Outage Data

Effect of Failure	100% Loss of Load of One Turbine-Generator Set	10% Loss of Load of One Turbine-Generator Set
MTBF	23,200 h	9,100 h
EMTTR	8 h	1.3 h

5.15. ELECTRICAL GROUP (EG)

The consists of the following:

- Offsite and main generator transmission system.
- AC distribution system.
- Uninterruptible power supply system.
- Essential uninterruptible power supply system.
- Standby power generation system.
- IE DC power system.
- Essential DC power system.
- Grounding, lightning protection, heat tracing, and cathodic protection system.
- Communication system.
- Lighting and service power system.
- Plant security system.

1. Summary System Description

The plant electrical system transfers the electrical power generated by the two generators to the high-voltage switchyard through two unit transformers, controls and meters electrical energy, and protects the power-carrying components. It is the source of electrical power for the plant auxiliaries, through the unit auxiliary transformers including the plant control, protection, and surveillance equipment and the engineered safety features equipment.

2. Forced Outage Basis

Component failure rates and repair times are based on historical performance of similar components as given in NERC, Ref. 5-4 equipment outage data.

a. For the overall station electrical distribution, NERC Component Cause Code 904 (switchgear for fossil - all fuel types, 300 to 399 MW was used).

b. Two 3-phase unit transformers are used, each approximately 275 MVA. Outage data for fossil plants, all fuel types, 300 to 399 MW, cause Code 903 (main transformer) was used.

c. The dc power supply data was developed using Nuclear Power Reliability Data System NPRDS Ref. 5-5 Code (A02) (EXB) dc onsite power system and control.

d. The ac power supply data was developed using NPRDS3 Ref. 5-5 Code (A02) (EXB) ac onsite power system and controls.

e. For the two-unit auxiliary transformers NERC Component Cause Code 914 (auxiliary transformer) for fossil plants, all fuel types, 300 to 399 MW was used.

f. For the back-up power supply (generators), NERC nuclear, all types, 400 to 799 MW Component Cause Code 270 (emergency power system) was used.

g. Very little unit outage is expected to be caused by the medium voltage switchgear; low voltage switchgear; lighting and power services; plant security; plant communication; grounding, lightning, and cathodic protection, standby power generator and plant cabling subsystems.

3. Forced Outage Data

Overall Station Electrical Distribution

Effect of Failure	100% Loss of Load of One Turbine-Generator Set	20% Loss of Load of One Turbine-Generator Set
MTBF	52,200 h	104,400 h
EMTTR	14 h	3 h

Unit Transformers (Two)

Effect of Failure	100% Loss of Load of One Turbine-Generator Set	25% Loss of Load of One Turbine-Generator Set
MTBF	104,400 h	69,600 h
EMTTR	110 h	24 h

Medium Voltage Switchgear

No measurable impact on forced outage.

Low Voltage Switchgear

No measurable impact on forced outage.

DC Power Systems

Effect of Failure	100% Loss of Plant Load
MTBF	486,600 h
EMTTR	444 h

AC Power System

Effect of Failure	100% Loss of Plant Load
MTBF	16,700 h
EMTTR	15 h

Lighting and Power Services

No measurable impact on forced outage.

Plant Security

No measurable impact on forced outage.

Plant Communications

No measurable impact on forced outage.

Standby Power Generation.

No measurable impact on forced outage.

Unit Auxiliary Transformers (Two)

Effect of Failure	100% Loss of Load of One Turbine-Generator Set
MTBF	104,400 h
EMTTR	15 h

Grounding, Lightning, Heat Tracing, and Cathodic Protection

No measurable impact on forced outage.

Plant Cabling

No measurable impact on forced outage.

5.16. REACTOR CAVITY COOLING SYSTEM (RCCS)

1. Summary Description

The air-cooled RCCS removes heat from the reactor cavity by the natural convection of outside air through cooling panels located in the reactor cavity. The cavity cooling panels form a cylindrical wall surrounding the reactor vessel. The cooling panels collect the heat transferred from the vessel by radiation and convection. They protect the cavity walls from overheating during normal operation and provide in alternate means of decay heat removal in the event that the normal cooling systems (HTS and SCS) are lost. The system has no valves or active components. It has multiple inlet/outlet ports and interconnected parallel flow paths to assure continued cooling in the event of blockage of any single duct or opening. The principal failure mode of the system is sudden or progressive blockage. Progressive blockage (from causes such as rain, icing, and wind-blown debris) can be detected by flow and temperature monitoring and corrective measures taken. Sudden blockage (from causes such as sabotage and tornado missiles) is protected against by redundant design features and the protected location of panels (underground) and the inlet/outlet ports. Structures are tornado missile-hardened and seismically designed. A module outage will result if over 50% of the duct area is blocked, but shutdown for cleaning of progressive blockage would probably occur if 25% of the duct area were blocked.

2. Forced Outage Basis

The closest applicable Cause Code to the RCCS in the NERC Component Cause Code Report, Ref. 5-4, is Cause Code 999 (external cause), but the dissimilarity of the RCCS to any equivalent system in other nuclear power plants makes this data of questionable value. Also, this data indicates no forced outages were reported as caused by Cause Code 999 in the NI. Reference 5-9 assumes a per unit per demand failure probability of 10^{-6} for the RCCS based on engineering judgment. This extremely low failure probability is consistent with the system design and safety risk. However, even if it were some orders of magnitude higher, it would still be insignificant as a cause of forced outages.

3. Forced Outage Data

No forced outage effect.

6. INVESTMENT RISK CONTRIBUTION TO UNAVAILABILITY

The studies of investment risk concentrate on events less likely than those studied for unavailability. Investment risk studies (1) analyze outages which are less likely than normal forced outages, (2) include the impact of secondary failures that could lengthen an outage and (3) consider other criteria (e.g., dollar loss).

The current limited investment risk assessment (Ref. 6-1) considered outages due to primary coolant leaks in the primary coolant boundary and pressurized conduction cooldowns without the RCCS. The primary coolant leaks are included in the vessel system (Section 5.4) of this report.

A complete investment risk assessment is unavailable now but will be included when available, and the effect on forced outage will then be included in this report.

7. PLANT CHARACTERISTICS

7.1. AVERAGE FORCED OUTAGE UNAVAILABILITY

The plant average forced outage characteristics are summarized below:

	h/yr	%/yr
NI EFOH	276	3.1
ECA EFOH	506	5.8
	782	8.9
Plant Goal	876	10.0

8. DISCUSSION OF RESULTS

The average EFOH for the MHTGR plant is 782 h/yr. The forced outage goal is 876 h/yr; therefore, the forced unavailability meets the plant goal. Since this report is based on conceptual design information, changes in the forced outage assessment may be expected as the design evolves.

The major contributors to forced outage are given in Table 1-1 in order of importance. The systems or subsystems listed combine for a total average unavailability of 5.8% per year of the total forced unavailability.

The vessel is the largest NI contributor to forced outage with 100 h/yr. The majority of this (96 h/yr) is due to radioactive helium leaks.

The power conversion group (e.g., turbine-generator, feedwater, and condensate systems) is the largest contributor to ECA forced outage. SWEC has studied the turbine design and a 3600 rpm nonreheat design has been assumed for this assessment. Based on SWEC's study of the turbine-generator and condenser historical data, a reduction of several hundred hours from these contributors was achieved. The majority of this savings is due to elimination of double accounting when using the historical data from fossil units in the 400 to 599 MW(e) range and by shifting maintenance outages to planned outages.

9. REFERENCES

1-1. Craig, W. C., "Forced Outage Assessment Report, Balance of Plant Input," Stone and Webster Engineering Corporation J.O. No. 14884.21, SWEC/GA-0074, June 11, 1986.

1-2. Sweeney, T. M., " Forced Outage Assessment, Nuclear BOP" Bechtel National, Inc. GCRA/HTGR Project: Job 14508, BL-972, June 30, 1986.

2-1. "Overall Plant Design Specification Modular High Temperature Gas Cooled Reactor", HTGR-86-004, Rev. 5, (GA Document 908397/5), June 1987.

2-2. Dunn, T. D., "Major Contributors to Forced Outage in the 4 x 350 MW(t) HTGR," HTGR-86-006, Rev. 0, (GA Document 908716/0), January 1986.

3-1. Shooman, M. L., Probabilistic Reliability: An Engineering Approach, McGraw Hill Book Company, 1968.

4-1. "Plot Plan: 4 x 350 MW(t) MHTGR Plant; Side by Side Steel Vessel Concept," SWEC Drawing 14884-DM-201E-3, June 29, 1987.

4-2. "MHTGR - Reactor Module General Arrangement," GA Technologies Drawing 028047/9, July 13, 1987.

5-1. "Reactor Safety Study: Appendix III Failure Data," USNRC Report WASH 1400 (NUREG 75/104), October 1975.

5-2. "IEEE Guide to the Collection and Presentation of Electrical, Electronic Sensing Component, and Mechanical Equipment Reliability Data for Nuclear Power Generating Stations," IEEE Std. 500-1984.

5-3. "Nuclear Plant Reliability Data System (NPRDS) 1980 Annual Report of Cumulative System and Component Reliability," NUREG/CR-2232.

5-4. "North American Electric Reliability Council/Generating Availability Data System" - Report for the Years 1974-1983.

5-5. "Nuclear Plant Reliability Data System (NPRDS) 1982 Annual Report of Cumulative System and Component Reliability."

5-6. "North American Electric Reliability Council/Generating availability Data System (GADS)" - Report for the Years 1975-1984.

5-7. "HTGR Main Helium Circulator Electric Motor Drive Study," Westinghouse Electric Report LRA Report No. 22878, February 1978.

5-8. "Demonstration Testing of Components for Electric-Driven HTGR Helium Circulator Applications with Magnetic Bearings" Report EPRI-HTGR 86-04, August 1986.

5-9. "Probabilistic Risk Assessment for the Standard MHTGR," DOE-HTGR-86-011 Rev. 3, January 1987.

5-10. "Reliability Allocations for the Standard MHTGR," DOE-HTGR-87-008 Rev. 0, (GA Document 909279/0), February 1987.

6-1. Dunn, T. D., and Armbrust, G., "MHTGR Investment Risk Assessment," GA Report RGE 909037/0, September 1986.

APPENDIX A
SERVICE HOUR CALCULATION

A feature of a modular plant is that the loss of one module consitutes the loss of the fraction that module contributed to the plant's output. Therefore, the service hours (SH) for the various plant systems must be determined.

The NERC definition of service hours is "the total number of hours the unit was actually operated with breakers closed to the station bus." This definition is acceptable for a single unit plant producing electric power.

For the MHTGR, with four modules and 2 T-Gs the situation is different. The "service hours" we should be concerned for in forced outage evaluations are the hours that a particular system/group must operate to produce the goal of 80% equivalent availability.

Systems/group which are involved only with electric power production (not residual heat removal), must operate at a minimum at least 80% of the year. Because of failures in modules or T-Gs, they must actually operate more than this minimum time. A maximum time they must operate would be 90% of the time. This is all operation except scheduled outages, which occur 10% of the time. As a reasonable compromise, an operating time of 85%, or 0.85 x 8760 h/yr = 7446 h/yr will be used.

Systems/group which provide residual heat removal (RHR) must, in theory, operate 100% of the time. This would mean during all power production (80% of the year) plus forced outages (10% of the year) and scheduled outages (10% of the year). Since there is significant redundancy within the RHR systems/groups [e.g., one HTS, one SCS on the primary (helium) side and two energy conversion trains) year round operation of all of them is not required. As a reasonable compromise, an operating time of 90% of the year or 0.9 (8750 h/yr) = 7884 h/yr will be used.

In summary, the service (actual operating) hours are:

Systems/Groups not performing RHR = 7446 h/yr.
Systems/Groups performing RHR = 7884 h/yr.

APPENDIX B
SUMMARY TABLES OF PLANT SYSTEM
FORCED OUTAGE DATA

TABLE B-1
350 MW(t) MHTGR FORCED OUTAGE DATA

System/Group; System/Subsystem	MTTR	MTTF	% LOSS	NM	SH	EFOH	Reference Section
Reactor service; Helium purification	42	19,000	25	4	7,446	16.4	5.1.1
Reactor service; Gaseous radioactive waste	24.5	58,300	100	1	7,446	3.1	5.1.2
Reactor service; Liquid nitrogen	76	372,200	50	2	7,446	6.4	5.1.4
Reactor service Reactor plant cooling water	21	740,000	100	1	7,446	0.2	5.1.5
Heat transport system; Main circulator	85	15,000	25	4	7,884	44.7	5.2.1
Heat transport system; Steam generator	288	71,000	25	4	7,884	32	5.2.2
Reactor; Neutron control	70	130,000	25	4	7,884	4.2	5.3.1
Reactor; Reactor Core	720	263,000	100	1	7,884	24	5.3.2
Reactor; Reactor internals	1,000	500,000	25	4	7,884	15.7	5.3.3
Vessel; all	250	18,600	25	4	7,446	100	5.4.0
Shutdown cooling system; all except shutdown cooling water	85	105,000	25	4	7,884	6.4	5.5.2
Shutdown cooling; Shutdown cooling water	21	740,000	100	1	7,884	0.2	5.5.1

TABLE B-1 (Continued)

System/Group; System/Subsystem	MTTR	MTTF	% LOSS	NM	SH	EFOH	Reference Section
Plant protection, and instrumentation; Safety protection	12	10,000	25	4	7,884	9.5	5.7.1
Plant protection, and instrumentation; Investment protection	34	36,000	25	4	7,884	7.4	5.7.2
Plant protection, and instrumentation; Special nuclear area instrumentation	12	89,000	25	4	7,884	1.1	5.7.3
Plant control; Supervisory control/data management	10	6,500	100	1	7,446	11.5	5.8.1
Plant control; NSSS control	12	7,500	25	4	7,884	12.6	5.8.2
Plant control; ECA control	12	5,600	100	1	7,884	16.9	5.8.3
Miscellaneous cont. and inst.; NSSS analytical inst.	12	30,000	25	4	7,446	3.0	5.9.1
Operator and maintenance errors	78	10,800	25	4	7,446	53.8	5.10
Power conversion group; Turbine generator and auxiliaries	110	2,200	5	2	7,446	37.2	5.11.1
Power conversion group; Turbine generator and auxiliaries	97	3,950	50	2	7,446	182.9	5.11.1
Power conversion group; Main and bypass steam	56	15,600	12.5	2	7,884	7.1	5.11.2
Power conversion group; Main and bypass steam	40	31,300	50	2	7,884	10.1	5.11.2

TABLE B-1 (Continued)

System/Group; System/Subsystem	MTTR	MTTF	% LOSS	NM	SH	EFOH	Reference Section
Power conversion group; Feedwater and condensate (except condenser)	25	1,300	25	2	7,884	75.8	5.11.3
Power conversion group; Feedwater and condensate (except condenser)	32	13,200	50	2	7,884	19.1	5.11.3
Power conversion group; Condenser	11	1,200	10	2	7,884	14.5	5.11.3
Power conversion group; Condenser	16	26,000	50	2	7,884	4.9	5.11.3
Power conversion group; Extraction and auxiliary steam	78	212,000	50	2	7,446	2.7	5.11.4
Power conversion group; Heater drains and condensate returns	50	323,500	50	2	7,446	1.2	5.11.5
Power conversion group; Condensate polishing	13	38,300	50	2	7,446	2.5	5.11.6
Power conversion group; Power conversion group support-chemical feed	21	27,200	5	3	7,446	0.9	5.11.8.1
Power conversion group; Power conversion group support-chemical feed	41	313,200	50	3	7,446	1	5.11.8.1
Power conversion group; Power conversion group support-steam vents and drains	50	323,500	50	2	7,446	1.2	5.11.8.2
Heat rejection group; Circulating water (pumps and cooling towers)	16	10,100	50	2	7,884	12.5	5.12.1

TABLE B-1 (Continued)

System/Group; System/Subsystem	MTTR	MTTF	% LOSS	NM	SH	EFOH	Reference Section
Heat rejection group; service water	17	583,000	15	2	7,884	0.1	5.12.2
Plant services; NI HVAC	36	38,000	25	4	7,446	7	5.14.3
Plant services; instrument and serv. air	9	313,200	25	4	7,446	0.2	5.14.2
Plant services; other mechanical service	8	23,200	50	2	7,446	2.6	5.14.5
Plant services; other mechanical services	13	9,100	5	2	7,446	1.1	5.14.5
Electrical systems group overall station electrical distribution	14	52,200	50	2	7,884	2.1	5.15.1
Electrical systems group; overall station electrical distribution	15	104,000	10	2	7,884	0.2	5.15.1
Electrical systems group; unit transformer	96	69,600	12.5	2	7,884	2.7	5.15.2
Electrical systems group; unit transformer	110	104,400	50	2	7,884	8.3	5.15.2
Electrical systems group; DC power systems	444	486,600	100	1	7,884	7.2	5.15.5
Electrical systems group; AC power system	15	16,700	100	1	7,884	7.1	5.15.6
Electrical systems group; Unit aux. transformers	15	104,400	50	2	7,884	1.1	5.15.11

APPENDIX C
CONVERSION OF AVAILABILITY DATA TO STANDARD FORM

Some changes have been made to the way the forced outage hours per year are calculated. These changes are primarily the result of different service hours. This section will show what the new equations are, followed by an example of the changes made to the ECA data.

The first definition in this process is that of the MTBFs. The MTBF is the sum of the MTTF plus the mean time to repair a component.

$$MTBF = MTTF + MTTR$$

Based on Markov theory, the forced outage rate is computed as follows:

$$FOR = MTTR/MTBF = MTTR/(MTTF + MTTR)$$

Therefore, the calculation of the EFOHs of the plant per year caused by a component or system is computed as follows:

$$EFOH = FOR \times (SH/(1.-FOR)) \times NM \times \%\ LOSS/100$$

where NM is the number of identical components or systems (or modules), % LOSS is the percent loss of plant full power caused by the loss of the component, and SH is the service hours of the component.

Although each component in the plant has its own SH, the assessment of plant level availability necessitates that an equivalent or characteristic SH be calculated to describe the plant's available time. In this assessment, two SH were calculated to describe the MHTGR plant, one for systems/groups involved in residual heat removal and the other for all of the remaining systems/groups. These calculations are fully detailed in Appendix A.

ECA

The data reported by Ref. 1-2 for the ECA subsystems forced outage calculations were given in the form of MTBFs and EMTTRs. The EMTTRs need to be converted to MTTRs.

The MTTRs for 100% power loss remain the same. The given EMTTRs for partial power losses have to be changed into MTTRs. The derivation is as follows:

$$\text{MTTR} = \frac{\text{EMTTR}}{\%\text{Loss}/100}$$

EXAMPLE:

System Name: Power Conversion Group

Subsystem Name: Turbine Generator and Auxiliaries

PREVIOUS PARAMETERS:

Percent Loss: 10%

EMTTR = 11.0 h

MTBF = 2200 h

EFOH = 36.1 h/yr

SH = 7218 h/yr

NEW PARAMETERS:

Percent Loss: 10%

MTTR = 110 h

MTTF = 2200 h

EFOH = 37.2 h/yr

SH = 7446 h/yr