**GA-A14915
UC-77**

GAS TURBINE HTGR PROGRAM

QUARTERLY PROGRESS REPORT FOR THE PERIOD ENDING DECEMBER 31, 1977

**by
PROJECT STAFF**

**Prepared under
Contract EY-76-C-03-0167
Project Agreement No. 46
for the San Francisco Operations Office
Department of Energy**

**GENERAL ATOMIC PROJECT 3227
DATE PUBLISHED: APRIL 1978**

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ABSTRACT

This report describes progress on the gas turbine high-temperature gas cooled reactor (GT-HTGR) program, Department of Energy Contract EY-76-C-03-0167, Project Agreement No. 46, performed during the period October 1, 1977, through December 31, 1977.

The report covers the demonstration plant design studies, core and fission product studies, shielding studies, heat exchanger design, and maintenance studies, including inservice inspection (ISI).

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1. INTRODUCTION AND SUMMARY

This report describes progress by General Atomic Company (GA) on the direct cycle gas turbine high-temperature gas cooled reactor (GT-HTGR) program, Department of Energy (DOE) Contract EY-76-C-03-0167, Project Agreement No. 46, for the period October 1, 1977, through December 31, 1977. The DOE Project Agreement No. 46 program is comprised of a conceptual design task and a materials R&D task. Main topics in the present report include demonstration plant design studies, core and fission product studies, shielding studies, heat exchanger design, and maintenance studies, including inservice inspection (ISI). Previous results from the DOE-sponsored GT-HTGR program are described in Refs. 1-1 through 1-7.

A summary of the demonstration plant conceptual design studies is given in Section 2. The demonstration plant is a 1500 MW(t) one loop intercooled plant which will be the basis for follow-on commercial plants in the U.S. and Europe. Demonstration plant conceptual design studies will continue through the remainder of FY-78.

Results of demonstration plant core studies, including fission product studies, are reported in Section 3. The core size established for the demonstration plant is 1530 MW(t) with a power density of 6.5 W/cc. The basis for selection of this core was its physical size, its basic geometry, and its minimum number of partial regions (regions with less than seven columns). Fission product studies were aimed at minimizing fission product release with its associated contamination of the turbomachinery.

Section 3 also reviews the establishment of a core task force. The purpose of the task force is to identify the objectives and milestones for higher temperature HTGR applications.

Section 4 reports the shielding studies conducted. These studies were aimed at establishing dose rate criteria, identifying and quantifying the significant sources of radiation from the standpoint of turbomachinery maintenance, determining the access and shielding requirements for the turbomachinery removal, and making preliminary estimates of decontamination requirements to allow contact maintenance of the turbomachinery.

Heat exchanger studies are reported in Section 5. The effort during this quarter was primarily a reassessment of the conceptual heat exchanger approaches and the ground rules covering them. The aim of this effort is to supplement information previously developed so that baseline conceptual heat exchanger approaches can be selected by April 1978.

Maintenance studies related to the 4000 MW(t), four loop reference plant design are reported in Section 6. The turbomachinery, turbomachinery control valves, recuperator, precooler, and core to turbine hot duct are addressed. The maintenance approach is formulated for conservatively high activity levels in order to provide equipment to deal economically with these high levels and, at the same time, permit the advantage of low activity levels to be exploited when they occur.

It should be noted that the information on fission product studies, shielding, and maintenance was developed with private funding and is presented to better define the total scope of work accomplished.

REFERENCES

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- 1-2. "Gas Turbine HTGR Program, Quarterly Progress and Task Closeout Report for the Period Ending September 30, 1976," ERDA Report GA-A14097, General Atomic, October 1976.

- 1-3. "Gas Turbine HTGR Program, Quarterly Progress Report for the Period Ending December 31, 1976," ERDA Report GA-A14243, General Atomic, January 1977.
- 1-4. "Gas Turbine HTGR Program, Quarterly Progress Report for the Period Ending March 31, 1977," ERDA Report GA-A14381, General Atomic, April 1977.
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- 1-6. "Gas Turbine HTGR Quarterly Progress Report for the Period Ending September 30, 1977," DOE Report GA-A14801, General Atomic, February 1977.
- 1-7. "Gas Turbine and Advanced HTGR Materials Screening Test Program, 10,000 Hour Results and Semiannual Progress Report for the Period Ending September 30, 1977," ERDA Report GA-A14407, General Atomic, 1977.

2. DEMONSTRATION PLANT STUDIES

2.1. DEMONSTRATION - COMMERCIAL PLANT RELATIONSHIPS

The nuclear demonstration plant is a key element in the direct cycle program. Not only will it provide the vehicle for demonstration of the plant performance and structural integrity of the major new power conversion system components, but also valuable operational experience. In addition, through the demonstration plant experience, all the necessary maintenance operations as well as the adequacy of repair equipment will have been verified before the first commercial plant is operational.

The relationship between the demonstration plant and the commercial variants which formulate the international cooperative direct-cycle program is shown in Fig. 2-1. From this array it can be seen that a coordinated effort in 1978 between GA and the HHT Project will lead to a more definitive description of a demonstration plant. An early decision in this joint effort has been the selection of a set of common cycle parameters. The demonstration plant concept will be based on a core thermal rating of 1500 MW(t) with an intercooled power conversion loop (approximately 600 MW(e)). The primary system will embody the warm liner principle, and the plant will be dry-cooled. The design concept being pursued by HHT will be based on a pebble bed core and will have a turbomachine designed for the European 50 Hz system. The design studies under way at GA are based on prismatic fuel elements and the turbomachine designed for the U.S. 60 Hz system. It should be emphasized that although two demonstration plant concepts are being studied, they are not rival design efforts but rather coordinated approaches made with the knowledge that the choice of core type influences the plant configuration. The final selection of core type is beyond the scope of this effort; this choice will be strongly influenced by international HTGR activities.

Fig. 2-1. Demonstration of commercial plant relationships

The relationships between the demonstration plant and commercial plant variants are shown in Fig. 2-1. The dashed lines on this figure delineate design-related activities. Currently the relationship between demonstration and commercial plants is better defined for the HHT program than for the GA program, as will be discussed later in this report. The HHT commercial plant, with a pebble bed core thermal rating of 3000 MW(t), is based on an intercooled cycle with the warm liner primary system feature. A single turbomachine of around 1200 MW(e) (50 Hz) has also been selected. A future variant of this plant (following successful operation of the demonstration plant) could embody two turbomachines of 600 MW(e) of the type run in the demonstration plant.

Studies are under way to select the best commercial plant variant for the U.S. market, and the two candidates are shown on Fig. 2-1. The follow-on reference plant is based on a non-intercooled cycle and has three Delta oriented power conversion loops with a loop rating of 1000 MW(t). In the primary system of this plant, the cavity liners are conventionally water-cooled. The alternative follow-on plant is based on an intercooled cycle and embodies two power conversion loops [1500 MW(t) rating]. A summary of the salient features of the demonstration and reference plant candidates is given in Table 2-1.

Engineering and economic studies are being made concerning selection of the reference plant for the U.S. market place. As mentioned earlier, the relationship between the demonstration plant and the reference plant for the GA program is less well defined than for the HHT program, because the selection of plant configuration has not yet been made. The role of the demonstration plant and the extent to which data from this plant would be applicable to the design of a U.S. commercial plant are important elements in the selection process. As shown in Fig. 2-1, there will certainly be substantial technology from the demonstration plant which could be factored into the design of a commercial plant. It is projected, however, that the function of the demonstration plant must offer, in addition to design information,

TABLE 2-1
GT-HTGR PLANT VARIANTS

Plant	1977 Reference Design	Follow-On Reference Plant	Alternative Follow-On Plant	GA Demonstration Plant	HHT Demonstration Plant ^(a)
Output power [MW(e)]	1584	1200 Nominal	1240 Nominal	620 Nominal	620 Nominal
Frequency (Hz)	60	60	60	60	50
Core power [MW(t)]	4000	3000	3000	1530	1500
Plant efficiency (%)	39.6	(b)	(b)	(b)	(b)
Intercooling	No	No	Yes	Yes	Yes
Plant configuration	Integrated	Integrated	Integrated	Integrated	Integrated
Core outlet temp [°C (°F)]	850 (1562)	850 (1562)	850 (1562)	850 (1562)	850 (1562)
Number of loops	4	3	2	1	1
Core type	Prismatic	Prismatic	Prismatic	Prismatic	Pebble
Fuel type	HEU	MEU ^(c)	MEU ^(c)	MEU ^(c)	HEU
Power density (W/cc)	7.15	7	6.8	6.5	5.5
Liner type	Conventional water-cooled	Conventional water-cooled	(b)	Warm liner	Warm liner
Plant cooling	Dry	Dry	Dry	Dry	Dry
Binary cycle option	Yes	Yes	No	No	No
Compressor ratio	2.50	(b)	(b)	(b)	(b)
Max system pressure [MPa (psia)]	1150 (7.93)	(b)	(b)	(b)	(b)
Turbomachine rating [MW(e)]	400	400 Nominal	620 Nominal	620 Nominal	620 Nominal
Recuperators/plant	4	3	2	1	2 or 1

TABLE 2-1 (Continued)

Plant	1977 Reference Design	Follow On Reference Plant	Alternative Follow On Plant	GA Demonstration Plant	HHT Demo Plant ^(a)
Precoolers/plant	4	3	2	1	2 or 1
Intercoolers/Plant	-	-	2	1	2 or 1
Plant design status	No further work planned	Delta design to be updated	New design	Coordinated effort GA/HHT in 1978	

(a) GA's estimate of HHT demonstration parameters and design features.

(b) Parameters and features to be determined from optimization and design studies.

(c) Fuel cycle design capable of changing between MEU and HEU.

integrity verification of major power conversion equipment items. Clearly the turbomachine is the most critical component from the development standpoint. For the alternative follow-on plant an obvious area that must be resolved involves the desirability/practicality of testing a U.S. designed 600 MW(e), 60 Hz gas turbine in the European based demonstration plant. For the follow-on reference plant an assessment must be made of the applicability of technology and test experience resulting from a European-designed 50 Hz, intercooled turbomachine to the design of a 400 MW(e), 60 Hz non-intercooled turbomachine for the U.S. plant. Thus the selection process for the most satisfactory commercial plant configuration for the U.S. market will involve an important assessment of the European sited nuclear demonstration plant's influence on development costs and risks.

2.2. DEMONSTRATION PLANT CYCLE PARAMETERS

An important task early in the demonstration plant studies will be an optimization study to select the major cycle parameters. In this systems-related effort the emphasis will be on selection of parameters (i.e. pressure ratio, system pressure, recuperator effectiveness, etc.) to give minimum total cost of power rather than maximum plant efficiency (although previous studies have in fact shown that the two are very close). All of the previous optimization studies have been done for a non-intercooled plant, so some modifications to the system's computer codes are necessary prior to initiation of the optimization study.

In early 1977 a set of cycle parameters (not optimized) were selected for an intercooled plant variant. To initiate design work on the demonstration plant, particularly for the reactor core, turbomachine, heat exchangers, and primary system, an initial decision to use the existing cycle data shown in Table 2-2 is considered valid. As new systems optimization study results (coordinated with the HHT project) are generated for the intercooled plant, the data will be factored into component and plant layout design efforts.

TABLE 2-2
INITIAL CYCLE PARAMETERS FOR GT-HTGR DEMONSTRATION PLANT

Parameters	Cycle Data
Reactor rating [MW(t)]	1500
Reactor outlet temp [°C (°F)]	850 (1562)
Reactor inlet temp [°C (°F)]	452.6 (846.6)
Ambient air temp [°C (°F)]	15 (59)
Compressor inlet temp [°C (°F)]	26.7 (80.1)
Compressor outlet temp [°C (°F)]	108.4 (227.1)
Compressor pressure ratio	3.0 (overall)
Max. cycle helium pressure [MPa (psia)]	7.93 (1150)
Recuperator effectiveness	0.898
Turbine isentropic efficiency (%)	92.2
Turbine disk cooling flow (%)	3.8
Compressor isentropic efficiency (%)	90.8 LP, 90.2 HP
Generator efficiency (%)	98.8
Plant auxiliary power [MW(e)]	11
Primary system pressure loss $\left[\frac{(\Delta P)}{(P)} \right] \%$	10.6
Plant efficiency (%)	41.23
Net electrical output [MW(e)]	618
System helium mass flow rate at compressor inlet [Kg/s (lb/hr)]	758.5 (6.02 x 10 ⁶)

2.3. DEMONSTRATION PLANT DESIGN STUDIES

For design studies initiated for the demonstration plant it has been recognized from the onset that because of the influence of introducing (1) the intercooler, (2) the warm liner, and (3) the goal to eliminate the thermal barrier to the greatest extent possible, the primary system is more complex than the early non-intercooled reference design. Some of the features and design options for the demonstration plant are given in Table 2-3. Current studies involve the orientation of the major components within the PCRV to establish satisfactory gas flow paths and also minimize the vessel diameter (an important economic factor). Scoping studies of different arrangements will lead to the selection of the demonstration plant concept. Some of the design features and design considerations to be weighed and evaluated in the concept selection process are given in Table 2-4. Following the selection of plant concept (in April 1978) the design studies will be directed towards establishing plant layout drawings.

TABLE 2-3
GT-HTGR DEMONSTRATION PLANT - FEATURES AND DESIGN OPTIONS

- Main Features

- One power conversion loop
- Core thermal rating - nominally 1500 MW(t)
- Turbomachine rating - nominally 620 MW(e), 60 Hz
- Intercooled cycle
- Prismatic core
- Warm liner
- Dry-cooled

- Design Options & Variables (To be coordinated with HHT)

- Core thermal power

- Layouts prepared for 1475, 1530, 1725, and 1920 MW(t);
1530 MW(t) favored because of turbomachine power limit
and adaptability to two loop 3000 MW(t) alternative
follow-on plant

- Gas flow paths

- Simplicity (low pressure loss)
- Simulation of commercial variant

- Turbomachine

- Two bearing design to be done by UTC
- Rating and performance
- Envelope
- Gas duct interfaces
- Drive end (compressor or turbine)
- Orientation in PCR
- Generator type and location

- Heat exchanger

- Features from HX study
- Exchanger(s) orientation in PCR
- Size as commercial plant
- Variable number of recuperators, precoolers, intercoolers

- Warm liner concept

- As HHT
- Engineered to meet U.S. criteria
- Liner and thermal shield details
- Gas source for HX cavity cooling

TABLE 2-3 (Continued)

CACS

- Commonality with steam cycle
- Number, capacity
- Position in PCRV

Special considerations

- Minimum PCRV size and cost
- Possible commercial application (impact on design)
- Test features
 - Helium storage for inventory control
 - Non-nuclear startup testing
 - Instrumentation
 - Simulation of multiloop operation/control
- Maintenance and ISI
- Man access capability
- BOP interfaces

TABLE 2-4
DEMONSTRATION PLANT DESIGN CONSIDERATIONS

Design Considerations

- Simulation of two loop plant
- PCRVR size
- Gas flow path complexity
- Performance
- Plant cost
- Commercial adaption of demonstration plant
- Risks and development cost to go to commercial plant
- Maintenance
- ISI
- Balance of plant considerations
- Commonality with HHT
- Commonality with steam cycle HTGR
- Applicability of generic program technology
- Safety and licensing
- Feasibility

Design Features to be Resolved

- Core position
- Heat exchanger(s) position
- Turbomachine orientation
- Number of heat exchangers
- Warm liner concept
- Major cavity thermal barrier
- Hot gas duct geometry
- CACS, capacity and type

3. CORE DESIGN

3.1. CORE DESIGN AND FISSION PRODUCT STUDIES

Studies of the GT-HTGR core design and fission product release have centered on generating candidate core arrangement layouts, establishing the expected core pressure drop, and defining the requirements for fission product release.

Four core arrangement layouts were prepared for evaluation at power levels of 1475, 1530, 1725, and 1920 MW(t). The basis for choosing these particular plants was to present, for a selected power level, a core which is physically compact and has a minimum number of partial regions (regions with less than seven columns) at the periphery. Attempts were made to avoid four column regions because of the problems associated with intra-region power tilts. Power density was less than 7.0 W/cm^3 in all cases. Table 3-1 summarizes the salient features of the alternative core designs. Based in part upon these analyses, the 1530 MW(t) core design was selected for the demonstration plant. The core arrangement layout selected is shown in Fig. 3-1. Preliminary pressure drop estimates made for the 1530 MW(t) core gave a maximum value of $5.9 \times 10^4 \text{ Pa}$ (8.54 psi) based on a maximum expected region power peaking factor of 1.6. This value is in close agreement with the estimated core pressure drop of $4.8 \times 10^4 \text{ Pa}$ (7 psi) for the lead plant steam cycle HTGR (SC-HTGR). Higher coolant operating temperatures as compared to the SC-HTGR, a modification of the coolant hole size [1.59-cm (0.625-in.) diam. for GT-HTGR versus 2.10-cm (0.826-in.) diam. for SC-HTGR], and the use of 10-row Fort St. Vrain (FSV) blocks account for the increase in the demonstration GT-HTGR core pressure drop. Preliminary cycle parameters were used in these calculations, and revised predictions will have to be performed when cycle parameters are updated.

TABLE 3-1
ALTERNATIVE CORE DESIGNS - GT-HTGR

	1475 MW(t)	1530 MW(t) ^(a)	1725 MW(t)	1920 MW(t)
Number of full regions	37	37	43	55
Number of five-column regions	6	12	12	6
Number of fixed orifice columns	30	12	12	0
Number of fuel columns	319	331	373	415
Power density (MW/m ³)	6.5	6.5	6.5	6.5
Number of refueling penetrations	55	55	61	61
Number of CRDs	49	49	55	61
Effective core diameter [m (ft)]	22.3	22.69	24.1	25.4

(a) Core size selected.

Fig. 3-1. Core arrangement for the 1530 MW(t) GT-HTGR

Scoping studies were carried out to assess what improvements in fuel design would be required to permit "hands-on" maintenance of GT-turbomachinery. Using previous GT core studies which included HEU fuel, TRISO fissile/BISO fertile fuel, three-year fuel cycles, ten-row fuel blocks (FSU type), and an 850°C (1562°F) outlet temperature as a basis, the following changes were proposed:

1. Utilization of an all-TRISO core. This would eliminate diffusive release from fertile particles of fission metals (in particular cesium) which in the past have been a significant source of long-lived γ -emitting plateout activity.
2. Modelling of diffusive release from oxide particles. In the past we have assumed total, instantaneous release of cesium inventories upon particle failure. With an improved data base and updated computer codes, more realistic assumptions of retention in oxide kernels can be made.
3. Five-fold reduction of heavy metal contamination in fuel rods. Heavy metal contamination results from uranium and thorium contamination in particle coatings and matrix material during manufacture. Potential improvements in particle fabrication and process development could bring about a five-fold reduction in contamination. Although the impact on the direct release of fission metals may not be dramatic, this improvement will bring about a substantial reduction in fission gas release. This, in turn, will reduce the level of metal plateout resulting from decay of gaseous precursors ($\text{Kr-90} \rightarrow \text{Sr-90}$; $\text{Xe-137} \rightarrow \text{Cs-137}$).
4. Reduction of TRISO fissile particle failure by a factor of two. With the improvements listed above, reduction by a factor of two in fissile failure will be necessary to permit "hands-on" maintenance (assuming a 100-fold decontamination factor). Because of

the low level of inservice failure expected with this core design, manufacturing defects in as-manufactured fuel will be the dominant source of failure (assumed to occur linearly with neutron fluence). Thus, to bring about a reduction in fissile particle coating failure, the defect fraction must be reduced from 0.2% to 0.1%.

The need for these recommended fuel improvements is strongly dependent on the proposed core design and operating history. The final fuel design selected will be reviewed when whole-core survey calculations have been completed later in the year. The predictions of fission product release will be based on an MEU rather than HEU core, as was done in the preliminary analyses.

3.2. CORE TASK FORCE

3.2.1. Background on Task Force

A core task force including representatives of the GT-HTGR and VHTR projects, fuel engineering, and plant engineering was established. The purpose of the task force was to identify the objectives and milestones for higher temperature reactor applications. A particularly significant issue for the GT and VHTR applications, which directly affects the core development, is the allowable level of primary coolant fission product activity. Because of the requirements for cleaner primary circuits, the core design must be changed from the SC-HTGR version.

3.2.2. Core Design and Performance Features

A table of core design features was agreed upon as necessary to meet the outlet gas temperatures and the fission product release requirements of the GT and VHTR applications. The task force considered only core design or

fuel quality changes from the SC design, with no changes to the generic fuel materials. Thus, the basic fuel materials have been assumed to be those currently planned for the later FSV reloads or the SC-HTGR.

The core design and performance features recommended by the task force are shown in Table 3-2.

3.2.3. Considerations for Circuit Radioactivity

During the course of this task force evaluation, two studies were conducted to define fission product release levels and allowable circuit activity for maintenance operations. The conclusions of these studies are that (1) reasonable design changes will allow a factor of 5 reduction in plateout activity and (2) this factor of reduction will allow direct disassembly of the turbomachinery after a decontamination step.

VHTR requirements for maintenance access and exposure limits more nearly parallel the SC-HTGR. However, because of the much higher VHTR core outlet temperature, fuel improvements and design parameters adopted for the GT-HTGR are expected to result in similar or lower dose levels for the VHTR than predicted for the SC-HTGR.

The fission product release predictions were made for the HEU fuel cycle. This analysis must be repeated for the MEU/Th cycle since particle performance, and therefore fission product release, may be significantly different with MEU fissile particle designs.

TABLE 3-2
CORE DESIGN AND PERFORMANCE FEATURES FOR GT-HTGR AND VHTR

Feature	Conditions
Power density	Nominal 7 W/cc
Core layouts	1530 MW(t) for demonstration and 3000 MW(t) for commercial GT; 842 MW(t) FSV for small VHTR.
Fuel cycle	Design capable of changing between MEU and HEU
Fuel lifetime	3 yr
Fuel element	10 row
Coated particle type	TRISO/TRISO
Region peaking factors (with tilts)	Choose same envelope as HTGR-SC
Core helium temperatures:	
In	453°C (847°F) intercooled GT, 490°C (914°F) non-intercooled GT; 475°C (887°F) for VHTR.
Out	850°C (1562°F) for GT; 950°C (1742°F) for VHTR
Core ΔP:	
Demonstration	5.9×10^4 Pa (8.5 psi)
Commercial	6.4×10^4 Pa (9.2 psi)
Fuel contamination factor	Lower by factor of 5 to 2×10^{-5}
Manufacturing defect fraction	Lower by a factor of 2 to 1×10^{-3}
Fission product basis for maintenance of GT	80% confidence on gases 50% confidence on metals (Level A)
Fission product release model from fuel particle	Model the diffusive release of Cs and Sr.

4. SHIELDING STUDIES

Radiation protection and shielding studies conducted during the first quarter of FY-78 were aimed at establishing dose rate criteria, identifying and quantifying the significant sources of radiation from the standpoint of turbomachinery maintenance, determining the access and shielding requirements for turbomachine removal, and making preliminary estimates of decontamination requirements to permit subsequent contact maintenance.

4.1. DEVELOPMENT OF GENERAL SHIELDING CRITERIA

A draft specification document for the 4000 MW(t) HEU version of the GT-HTGR entitled "Shielding and Source Strengths" was developed during the quarter (Ref. 4-1). The document sets forth shielding design requirements and recommendations, lists primary and secondary radiation source strengths, provides summary radiation level diagrams, and evaluates radiation damage to the PCRV liner.

Some of the significant dose rate criteria set forth in this document are: (1) 20 mrem/hr at the side of the turbomachine removal cask with the cask loaded, (2) 100 mrem/hr 3 ft from the heat exchanger removal cask (assuming 100-day decay), and (3) 20 man-rem total refueling exposure per unit. All dose rate criteria were established within the ground rules of Federal Regulation 10 CFR 20 and Regulatory Guide 8.8, and conform to ALARA (as low as reasonably achievable) guidelines.

The document also presents neutron and gamma fluxes and heating rates in the PCRV and internals. The shift in nil-ductility temperature in the PCRV liner resulting from 40-yr neutron exposure, calculated from damage functions published in the Nuclear Systems Materials Handbook, was found to be -8°C (17°F) in the top liner, 3°C (38°F) in the side liner, and nil in the bottom liner.

Secondary sources include fission-product plateout in the primary helium circuit. For this draft specification, plateout levels were derived from SC-HTGR data, corrected for the greater power level, outlet temperature, helium wetted surface area, and preferential plateout factors for the 4000-MW(t) GT-HTGR. As indicated in Table 4-1 (taken from the draft specification), these plateout distribution factors are very significant for turbomachine removal and maintenance. With respect to the absolute level of plateout, shielding design calculations are to be based on level B activities, whereas unshielded maintenance operations and containment access may use level A (expected) activities.

As new fission product information becomes available in CY-78 for the MEU fuel cycle, the draft shielding specification will be revised. It will also be extended to cover the one-, two-, and three-loop designs.

4.2. CALCULATIONS OF PLATEOUT AND INDUCED ACTIVATION LEVELS ON TURBOMACHINE

4.2.1. Induced Activation

A rough estimate was made of the neutron-induced radioactivity which might be expected in the GT-HTGR turbomachinery, in order to identify this as a possible problem area (Ref. 4-2). Considering that induced activity will remain even after all plateout activity is removed by decontamination, the feasibility of contact maintenance will depend upon the level of induced activity.

The geometry for the streaming calculation is shown in Fig. 4-1, where for illustrative purposes the streaming thermal neutrons are assumed to be absorbed in a thick metal disc of diameter equal to the inlet diameter. It was desired to compute the resulting gamma-ray dose rates near the disc surface for various alloys and exposure periods. For the alloys considered and with at least one day decay, the only important activation products are Cr-51, Fe-59, and Co-60. The results are shown in Table 4-2 for an operating thermal neutron flux level of 7×10^{10} n/cm²-sec in the lower reactor plenum on the centerline.

TABLE 4-1
PLATEOUT DISTRIBUTION FACTORS

	Duct to Turbine	Turbine	Recuperator (at inlet)
Cesium and Ba ^{137m}	1.0	2.6	53
Tellurium and I ¹³²	30	65	35
Other iodines	1.0	1.0	1.0
Silver	32	80	18
Other nuclides with noble gas precursors	1.0	1.0	1.0
Other nuclides without noble gas precursors and Sr, Y, La ¹⁴⁰	30	80	1.0

Fig. 4-1. Geometry for streaming calculation

TABLE 4-2
ACTIVATION DOSE RATES FROM EQUIVALENT DISC
(Dose Rates in mrem/hr, One Day Shutdown Time)

Alloy	Exposure Time (yr)				
	0.5	1	2	6	15
Carbon Steel	0.3	0.4	0.5	0.8	1.1
304 SS	0.7	1.2	2.2	4.9	7.6
Hastelloy-X	4.9	9.4	17	42	65
IN 100	23	44	83	200	310

In an actual situation, it would appear that neutrons either stream into the inside of the machine or are absorbed in the casing or missile rings, both of which are composed of low alloy steel. The neutrons reaching the interior of the machine will be absorbed in the IN 100 blades or in other hardware which should have a lower cobalt content. Activation gammas emitted from interior parts will be stopped to a great extent by the missile rings and casing. Therefore, the following tentative conclusions can be drawn:

1. The induced activation dose rate outside the turbomachinery casing will be less than 1 mrem/hr except at the helium opening, where gammas from activated blades may cause a somewhat higher dose rate.
2. The surface dose rate on some blades may be as high as 200 mrem/hr after a six-year exposure, but will drop off rapidly with separation distance or intervening layers of steel.
3. If the blades are replaced after six years, then the activation dose rate will be less 1 mrem/hr anywhere around the reassembled machine, even at the helium opening.
4. Induced activation will not prevent contact-maintenance of the GT-HTGR turbomachinery.

4.2.2. Plateout Activity

The basic approach to plateout has already been discussed in connection with the draft shielding specification. Several additional studies were performed (Ref. 4-3) to determine the effect of operating time, shutdown time, and circuit location on plateout levels. The nuclides that are important from the standpoint of turbomachine removal are Ag-110m, I-131, I-132, Cs-134, and Cs-137/Ba-137m, as listed in Table 4-3.

TABLE 4-3
SURFACE PLATEOUT ACTIVITY
LEVEL A, 6-YR ACCUMULATION, 10-DAY DECAY

Nuclide	Uniform ^(a) ($\mu\text{Ci}/\text{cm}^2$)	Preferential ($\mu\text{Ci}/\text{cm}^2$)				
		Turbine Inlet	Turbine	Turbine Outlet	Com- pressor	Contam- inated Casing
Ag-110m	0.14	8.8	10.8	9.5	0.14	0.14
I-131	2.2	2.2	2.2	2.2	2.2	2.2
I-132	0.7	42.0	45.0	42.0	0.7	0.7
Cs-134	5.4	5.4	14.0	17.3	5.4	5.4
Cs-137/Ba- 137m	2.9	2.9	7.6	9.4	2.9	2.9

(a) Based on a total plateout surface area of $2.94 \times 10^9 \text{ cm}^2$ in the 4000 MW(t) primary loop.

The potential for lower plateout inventories is being investigated by the Fuel Performance Branch. It appears that a reduction factor of 5 to 10 on cesium levels may be possible by reducing the fuel temperature, fuel particle failure fraction, and manufacturing contamination. Reduction of iodine levels can be achieved in a similar manner. Fortunately, all iodines have much shorter half-lives than cesiums. Decay times longer than 30 days can reduce iodine activities to insignificant levels as compared with cesium activities.

4.3. PERSONNEL ACCESS AND EXPOSURE TO PREPARE TURBOMACHINE FOR REMOVAL

Shutdown dose rates were calculated in the turbine and compressor bearing cavities for the conservative case of 40-yr operation and 3- or 10-day decay (Fig. 4-2). (For 6-yr operation the turbine cavity 3-day dose rate would be decreased by only 4% and the compressor cavity dose rate would decline by about 20%.) The calculated dose rates exceed the criteria set forth in the draft shielding specification. In order to reduce them to 100 mrem/hr, a modest additional thickness of steel must be added at both ends, as shown in Fig. 4-2.

When the reduced dose rates are combined with information from a time-and-motion study on turbomachine preparation and removal, it is determined that the integrated dose to all workers will be 1.7 man-rem. Despite uncertainties in the time and motion data and omission of other considerations such as internal doses, it seems reasonable to conclude that an integrated dose of less than 5 man-rem for the turbomachine removal may be achievable without violating any of the personnel exposure criteria.

4.4. SHIELDING DESIGN FOR TURBOMACHINE REMOVAL CASK

Shielding requirements for the turbomachine handling cask (Fig. 4-3) were determined (Ref. 4.4) based upon preferential Level B plateout for the case of 40-yr operation of a 4000 MW(t) GT-HTGR at 7 W/cc with 10-row fuel

Fig. 4-2. Shielding requirements for personnel access to turbine and compressor bearing cavities

Fig. 4-3. Steel shield thicknesses of turbomachine handling cask

blocks in the core followed by 10 days of decay. The required steel thickness for the cask is generally 12.7 cm (5 in.) except around the turbine inlet and at the turbine end, where the thickness is 20 cm (8 in.). The resulting dose rate at every point on the outside surface of the cask is 10 mrem/hr or less.

For the most part, Ag-110m, I-132, and Cs-134 are the major contributors to the external dose rate, so operation of the turbomachine for 6 yr instead of 40 yr would have a negligible effect on the shield design. The assumption of uniform Level B plateout on the casing may be highly conservative, since the region around the casing is not in the path of maximum helium flow and the casing is only contaminated with the primary helium from the compressor inlet.

4.5. ESTIMATES OF DECONTAMINATION REQUIREMENTS

The removal of a turbomachine, placement in the hot service facility, shipment to a central service facility, and decontamination and maintenance of the machine at the central facility have been addressed in Ref. 4-5. The importance of the degree of radioactive contamination is indicated by the following excerpt: "The unit being inspected must be completely stripped down to permit effective decontamination to a level where the parts can be handled directly. Those that cannot be decontaminated satisfactorily must, of course, be replaced." Thus, failure to achieve contact maintenance carries with it a large economic penalty, either for hardware replacement or for elaborate remote handling equipment.

The first step in the decontamination study was to establish the radiation criteria for contact maintenance. Consideration of recommendations

by GA, Batelle-Northwest, San Onofre, the British AERE, and the National Bureau of Standards led to the following guideline:

Maximum fixed surface contamination for contact maintenance (assumes removable contamination has already been eliminated by a decontamination process):

Most fission products	1000 CPM ^(a)
Sr-90, Ce-144, Eu-154	100 CPM

(a) CPM indicates counts per minute using a standard Geiger-Mueller beta-gamma detector.

Table 4-4 provides an overall breakdown of the fission-product activities on the turbine and the resulting CPMs as a function of a decontamination factor. An initial decontamination should take place in the hot service facility at the plant site to (1) eliminate removable surface contamination, (2) facilitate disassembly, and (3) simplify shipping of components to an off site decontamination and maintenance facility. Assuming this initial decontamination consists of a decontamination factor (DF) of 10, it can be calculated from Table 4-4 that the remaining CPM consists of 1300 from Sr-90/Ce-144/Eu-154 and 38,400 from the remaining nuclides. A further DF of 50 would result in 26 CPM from Sr-90/Ce-144/Eu-154 and 770 CPM from the remainder, resulting in the required equivalent 1000 CPM. Since an overall DF of $10 \times 50 = 500$ is of questionable feasibility, it may be necessary to seek a factor of 5 reduction in fission product release or a relaxation of the preferential plateout factors for the turbine.

A summary listing the necessary radioactivity reductions, the advantages of such reductions, and the means to achieve them is presented in Table 4-5.

TABLE 4-4
LEVEL A PLATEOUT SURFACE CONTAMINATION
AT MIDPOINT OF TURBINE: 6-YR OPERATION, 100 DAY DECAY

Nuclide	$\mu\text{Ci}/\text{cm}^2$	DPM per 100 cm^2	NO DF CPM	DF=10 CPM	DF=100 CPM	DF=1000 CPM
Cs ¹³⁴	14.0	5.2+7	1.2+5	1.24+4	1.23+3	120
Ag ^{110m}	8.5	3.1+7	7.2+4	7.2+3	720	72
Cs/Ba ^{137m}	7.6	2.8+7	6.5+4	6.5+3	650	65
Te ^{129m}	5.7	2.1+7	4.9+4	4.9+3	490	49
Sb ¹²⁵	4.0	1.5+7	3.5+4	3.5+3	350	35
Te ¹²⁹	3.6	1.3+7	3.0+4	3.0+3	300	30
Nb ⁹⁵	1.0	3.7+6	8.6+3	860	86	8.6
Sr ⁹⁰	0.63	2.3+6	5.4+3	540	54	5.4
Eu ¹⁵⁴	0.55	2.0+6	4.6+3	460	46	4.6
Zr ⁹⁵	0.4	1.5+6	3.5+3	350	35	3.5
Ce/Pr ¹⁴⁴	0.36	1.3+6	3.0+3	300	30	3.0

TABLE 4-5
REQUIRED DECONTAMINATION FACTORS OR
FISSION PRODUCT (FP) REDUCTIONS

Objective	Means to Achieve Objective
Unshielded shipping to decontamination/maintenance facility	DF of 30 to 110 at plant site or factor of 3 to 11 reduction in FP inventory and DF of 10 at plant site
Type A Curies packaging for shipping to decontamination/maintenance facility	DF of 50 at plant site or factor of 5 reduction in FP inventory and DR of 10 at plant site
Contact maintenance without possibility of airborne activity	DF or 10 at plant site followed by DF of 50 at decontamination/maintenance facility or factor of 5 reduction in FP inventory, DF of 10 at plant site, and DF of 10 at facility
Contact maintenance involving machining and possible airborne activity	DF of 10 at plant site followed by DF of at least 100 at facility plus virtually complete elimination of Sr^{90} , Ce^{144} , and Eu^{154} or substantial reductions in Sr^{90} , Co^{144} , and Eu^{154} release and DFs as above

REFERENCES

- 4-1. Engholm, B.A., "Draft Specification for GT-HTGR," General Atomic Company unpublished data, December 5, 1977.
- 4-2. Engholm, B.A., "Neutron Activation of GT-HTGR Turbomachine," General Atomic Company unpublished data, November 1, 1977.
- 4-3. Engholm, B.A., and S. Su, "GT-HTGR Turbomachinery Removal, Decontamination, and Maintenance," General Atomic Company unpublished data, December 12, 1977.
- 4-4. Su, S., "Shielding Requirements for Turbomachinery Handling Cask," General Atomic Company unpublished data, November 29, 1977.
- 4-5. Paget, J. A., "Maintenance Plan for the Gas Turbine HTGR Power Plant," USAEC Report GA-A13187, General Atomic Company, October 31, 1974.

5. HEAT EXCHANGER DESIGN STUDIES

5.1. INTRODUCTION

Instead of pursuing designs for specific plant options, the GT-HTGR heat exchanger engineering effort is currently dedicated to a reassessment of the conceptual heat exchanger approaches and the ground rules governing them. This activity is being undertaken at this time for the following reasons:

1. The recent reorganization of the GA engineering staff along functional lines has improved the availability of specialists in the various technical disciplines for direct involvement in GT-HTGR heat exchanger design.
2. Recent trends toward improved maintainability and inservice inspection capability in nuclear power plant heat exchangers could have a major impact on the selection of the heat exchanger design approaches.
3. The impact of intercooling and the PCRV warm liner GT-HTGR plant options on the heat exchanger design needs to be evaluated.

The aim of the effort under way is to supplement information previously developed (Ref. 5-1) so that baseline conceptual heat exchanger approaches can be selected by April 1978. In addition, this investigation includes the evaluation of conceptual approaches proposed to improve areas of technical uncertainty in the present straight-tube counterflow recuperator designs (e.g., more efficient tube support and wear protection and improved water-side heat transfer in the helium-to-water heat exchangers.

5.2. TECHNICAL APPROACH

The GT-HTGR heat exchanger concept selection study has been organized into three general phases, comprising (1) parametric thermal matrix studies, (2) mechanical design investigations, and (3) layout studies. At this writing phase 1 is under way, and some of the results generated thus far are discussed in the following sections. Table 5-1 presents a matrix showing how thermal matrix candidates were selected for parametric study and how these relate to previous studies (Ref. 5-1).

5.3. RECUPERATOR PARAMETRIC STUDIES

With the exception of thermal rating [MW(t) per loop], the operating conditions for the GT-HTGR recuperator are nearly the same for all plant options currently under consideration. Therefore, recuperator thermal matrix parametric comparisons based on operating conditions established for the non-intercooled GT-HTGR plant (summarized in Table 5-2) should produce results generally applicable to all plant options. The results developed in the reference study (Ref. 5-1) showing a plain tubular counterflow configuration to be the preferred recuperator thermal matrix approach are still considered valid. This was essentially reconfirmed in the evaluation of a finned tube concept which was a special case of longitudinal finned tubing not addressed previously. For this concept, the fins act as tube supports in addition to providing their normal heat transfer augmentation function when employed in a counterflow tube bundle. Compared to other fin tube approaches, this idea has the theoretical merit of inherently stiffening the tube bundle for lateral loads while eliminating the need for tube support grids and the significant parasitic pressure losses associated with them. The candidate approach considered here was based on a fin tube geometry comprising six rectangular-profile longitudinal fins located at equal circumferential spacings on the tube o.d. with the fin height equal to half the distance between tubes pitched in an equilateral triangular array. Tube support within this array is provided by so orienting the

TABLE 5-1
STATUS OF GT-HTGR HX PARAMETRIC STUDIES

Surface Geometries	Flow Configurations						Remarks
	Counterflow		Cross-counterflow				
			Disc/Donut		Helical		
	R	P	R	P	R	P	
Plain Tubular	1	1,3	1	1	1	1,2	Counterflow = lowest frontal area + large no. of tubes
Roughened Tubular							
o.d. rough	1	1	N	N	N	N	Low-profile circumferential ribs produce 2 x h at 3 x f
i.d. rough	1	1	N	N	N	N	
o.d./i.d. rough	1	1	N	N	N	N	
Fin tube							
Longitudinal							i.d. spiral fin tube for precoolers made and tested by Noranda Forge-Fin
o.d. plain	1,2	1,2	N	N	N	N	
o.d. offset	X	2	N	N	N	N	
i.d. plain	1	1	N	N	N	N	
i.d. spiral	X	2	N	N	N	N	
o.d./i.d. plain	1	1	N	N	N	N	
Transverse							
o.d. plain	N	N	X	X	X	2,3	Similar to HTGR steam generator fin tubes
o.d. offset	N	N	X	X	X	X	
Dimpled tube	Q	Q	Q	Q	Q	Q	Reliability, quality control, state-of-art
Non-tubular (plate-fin, etc.)	Q	Q	Q	Q	Q	Q	Non-inspectable, state-of-art

Legend:

R = Recuperator
P = Precooler
N = Not applicable
Q = Eliminated by qualitative arguments
X = Not investigated

1 = Previously reported (Ref. 5-1)
2 = Work in progress
3 = Investigated for both the non-intercooled and inter-cooled plant options

tubes that their fintips butt against one another to create a rigid structure. Several variations of this general mechanical approach are possible: e.g., (1) three fins per tube with the fin height equal to the tube spacing, (2) nontriangular tube pitching, and/or (3) twisting the fins. However, the concept described above was selected for study because it appeared to offer the best combination of high fin effectiveness, low frontal area requirement, and tube bundle compactness for favorable thermal size and performance results. The obvious practicality considerations associated with this approach were relegated to the mechanical design investigation phase of the concept selection study effort under the provision that analytical justification for this concept be demonstrated beforehand.

Using the operating conditions shown in Table 5-2, a parametric comparison of this concept with the plain tube approach was made, and the results are displayed in Fig. 5-1. The plain tube design used as the reference here was selected on the basis of minimum overall pressure loss for the frontal area and tube pitching range considered, in accordance with procedures used to establish previous plain tube recuperator designs. In addition, the same frontal area (HX diameter) and tube diameter were used for all cases, and rectangular fin profiles were assumed. Considering the economics and general complexity of finned tubing, this relative comparison reveals no particular incentive to adopt this concept unless subsequent mechanical design studies reveal a need for coarser tube pitching (or reduced number of tubes) and shorter effective tube lengths.

5.4. PRECOOLER PARAMETRIC STUDIES

Studies thus far have shown that straight tube, pure counterflow pre-cooler designs require minimal PCRV cavity diameters while facilitating headering and modularization. However, these designs are also characterized by large numbers of small diameter tubes and low water velocities, which may reduce their attractiveness as compared to other approaches from the standpoints of cost, maintainability/ISI, and technical uncertainty.

TABLE 5-2
RECUPERATOR DESIGN PARAMETERS
[Typical for 850°C. (1562°F) Reactor Outlet Temperature
(ROT) Non-Intercooled GT-HTGR Plant]

Fluid	HP Helium	LP Helium
Flow per HX [kg/s (lb/hr)]	545.6 (4.331x10 ⁶)	566.5 (4.497x10 ⁶)
Inlet temp [°C (°F)]	174.5 (346.1)	534.8 (994.7)
Outlet temp [°C (°F)]	498.1 (928.6)	223.3 (434.0)
Inlet pressure [MPa (psia)]	7.85 (1139)	3.25 (472)
Effectiveness	0.898	
LMTD [°C (°F)]	42.5 (76.5)	
UA Required per HX $\left[\frac{\text{MW}}{^{\circ}\text{C}} \left(\frac{\text{BTU}}{\text{hr}^{\circ}\text{F}} \right) \right]$	21.6 (40.96 x 10 ⁶)	
UA margin (%)	10	
Frontal area per HX $\left[\text{m}^2 (\text{ft}^2) \right]$	16.04 (172.7)	

Plain tube ref design statistics

Effective length [m (ft)]	11.9 (39)
Tube bundle weight [tonne (ton)]	258 (284)
Overall relative pressure loss (HP+LP)(%)	1.5
Tube o.d. x wall [mm(in.)]	11.1 x 1.1 (0.4375 x 0.045)
Tube p/d (triangular)	1.40

RECTANGULAR FINS AT $\ell = \left(\frac{p \cdot d}{2} \right)$
(SIX O.D. FINS PER TUBE)

Fig. 5-1. GT-HTGR recuperator fin tube parametric study

Parametric study efforts under way at this writing are aimed at investigating design alternatives that offer prospects for improvements in these areas.

The finned tube concept previously discussed for the recuperator was also investigated for applicability to the precooler. Here it was thought that adding surface area enhancement to the helium side, which is the controlling thermal resistance in plain tube designs, would produce more tangible benefits than in the recuperator application, where the inside and outside resistances in plain tube designs are nearly balanced. The results displayed in Fig. 5-2 were generated in the same manner as described in the preceding section for the recuperator (Fig. 5-1), using the design conditions summarized in Table 5-3 for all cases. The conditions chosen are representative of the precooler requirements for a non-intercooled GT-HTGR plant that does not include a bottoming power cycle option. From the standpoints of surface area requirements and water-side heat transfer, these conditions place more demands on the heat exchanger design than those of the precooler and intercooler in intercooled GT-HTGR plants. Although the curve map shows that this finned tube approach presents the opportunity for obtaining coarser tube pitching, shorter lengths, and lower tube material weights (considered to be a weak economic indicator), the benefits are obtainable only at the expense of higher helium-side pressure loss (plant efficiency penalty) and/or PCRV cavity diameter (plant cost penalty), both of which are considered unacceptable for plant design. Therefore, subsequent thermal design activities will not include this concept.

REFERENCE

- 5-1. "Gas Turbine HTGR Program Semiannual Progress Report for the Period July 1, 1975 through December 31, 1975," ERDA Report GA-A13740, General Atomic Company, January 1976.

COUNTERFLOW TUBULAR HX CONFIGURATION

RECTANGULAR O.D. FINS AT $\ell = \frac{1}{2} (p \cdot d)$

RESULTS ARE COMPARED TO
REFERENCE PLAIN TUBE SOLUTION
SELECTED ON BASIS OF $p/d = 1.40$

Fig. 5-2. GT-HTGR precooler fin tube parametric study

TABLE 5-3
PRECOOLER DESIGN PARAMETERS
[Typical for 850°C (1562°F) ROT Non-Intercooled GT-HTGR Plant]

Fluid	Helium	Water
Flow per HX [kg/s (lb/hr)]	570.7 (4.53x10 ⁶)	1234 (9.8x10 ⁶)
Inlet temp [°C (°F)]	221.1 (430)	20.6 (69)
Outlet temp [°C (°F)]	26.1 (79)	132.2 (270)
Inlet pressure [MPa (psia)]	3.21 (465)	- - -
Outlet pressure [MPa (psia)]	- - -	2.07 (300)
Effectiveness	0.972	
LMTD [°C (°F)]	30.1 (54.1)	
UA required per HX $\left[\frac{\text{MW}}{^{\circ}\text{C}} \left(\frac{\text{BTU}}{\text{hr}^{\circ}\text{F}} \right) \right]$	19.25 (36.5x10 ⁶)	
UA margin (%)	10	
Frontal area per HX [m ² (ft ²)]	9.29 (100)	
Plain tube ref design statistics		
Effective length [m (ft)]	(29)	
Tube bundle weight [tonne (ton)]	(152.4)	
Helium relative pressure loss (%)	1.2	
Tube o.d. x wall [mm (in.)]	9.5 x 1.24 (0.375 x .049)	
Tube p/d (triangular)	1.40	

6. SUMMARY OF MAINTENANCE AND ISI STUDIES [4000 MW(t), FOUR-LOOP, NON-INTERCOOLED REFERENCE DESIGN]

6.1. MAINTENANCE

In the integrated GT-HTGR plant, all of the helium turbine power conversion machinery operates on the reactor coolant helium and is installed inside the reactor vessel; thus it is important that maintenance aspects be considered during the plant conceptual design phase. The primary objective of the GT-HTGR plant maintenance plan is to provide facilities, equipment, and component designs of a reasonable cost that will yield good plant availability. It is therefore a matter of policy to design the reactor and the power conversion system installation as well as the related maintenance facilities for practical operation for both planned and unplanned maintenance.

All primary system equipment is designed for full plant lifetime, with no limited-life parts, and the influence that maintenance considerations have on the design of the major components and plant configuration has been studied. Turbomachinery change-out is regarded as a planned maintenance activity, and the sequence of events, together with the necessary equipment items for removal and re-installation of the helium gas turbine, is summarized below. The maintenance and repair of components such as the core-to-turbine hot gas duct, heat exchangers, and valves, which, although designed to last the plant lifetime may fail from unforeseen circumstances, are also briefly summarized.

Fission product contamination projections in various parts of the primary circuit have a strong influence on maintenance approach, and shielding

studies (particularly in the turbomachine bearing areas to permit man access) are discussed in Section 4. The maintenance plan is formulated for conservatively high activity levels, with the objective of providing equipment to deal economically with these high activity levels and, at the same time, permitting the advantage of low activity levels to be exploited when they occur.

The maintenance studies outlined below were done for the 4000 MW(t), four-loop reference plant configuration. Details of this plant layout are shown in Figs. 6-1, 6-2, and 6-3.

6.1.1. Turbomachine Remote Removability

While turbomachine removal and replacement is considered to be a scheduled maintenance operation, the current maintenance philosophy for a mature commercial plant is that the turbomachine will be replaced only if the diagnostic system indicates a problem, as outlined in Table 6-1. The turbomachine-vessel interfaces will be so engineered that the machine can be installed and removed remotely, but other methods will be provided in case the special apparatus should fail to function. A design constraint on the turbomachine assembly o.d. of 3.5 m (11.5 ft) permits transportation (by railroad) of a contaminated machine in a shielded cask to an off site maintenance facility for decontamination, disassembly, overhaul, and reassembly. After depressurization, the removal and replacement of the turbomachine will, in general, be performed by remote handling techniques with equipment that retains the helium atmosphere within the PCRV and also provides any needed radiation shielding.

The turbomachine maintenance procedures are summarized in Table 6-2. Details of the 400 MW(e) gas turbine are shown in Fig. 6-4, and a view of the turbomachine in the installed position is given in Fig. 6-5. The sequence of operations for machine removal is as follows. The generator is mounted on a pedestal with an integral air-bearing pallet in such a manner

Fig. 6-1. Elevation view of 4000 MW(t) GT-HTGR reference plant

SECTION 13-13

Fig. 6-3. Elevation view through power conversion loop of 4000 MW(t) GT-HTGR reference plant

TABLE 6-1
GT-HTGR TURBOMACHINE MAINTENANCE PHILOSOPHY

- Reliance on diagnostics to indicate operating problems

Temperatures
 Pressures
 Vibration Monitors
 Visual (fiber optics)
 Direct contact and gas-coupled emission transducers
 Clearances and eccentricities
 Magnetic metal chip detectors in lubrication system
 Oil spectrometric analysis
 Primary system chemistry

On-line
 computer
 system for
 condition
 monitoring

- Turbomachine installation and removal procedures established. Current design requires man access into the purged areas for connection/disconnection of drive coupling and servicing of system line flanges.
- For mature commercial plant the turbomachine will be replaced only if diagnostic system indicates problems.
- Stress levels in turbomachine are commensurate with the full 280,000-hr plant operating life (i.e., no life-limited subcomponents).
- Journal bearing areas accessible for inspection and maintenance (bearing pad replacement, etc.).
- In situ balancing of rotor considered possible by UTC.
- Alignment of turbomachine generator not considered a problem.
- Turbomachine envelope is within railroad limits permitting transportation to a central maintenance facility for decontamination and overhaul.
- For many maintenance operations experience from industrial gas turbines directly applicable.

TABLE 6-2
TURBOMACHINE MAINTENANCE PROCEDURES

Removal of a turbomachine from the PCRV after a period of service entails the following basic procedures:

- Remove the generator.
- Remove the high-speed shaft and containment closure.
- Remove the PCRV plug connections.
- Open the PCRV plug access port.
- Uncouple the PCRV plug to turbomachine connections and disconnect the shaft couplings.
- Install runways for removal equipment.
- Remove the PCRV plug.
- Remove the turbomachine thrust ring.
- Install the support bearing assembly to the compressor inlet bell.
- Install the isolation valve to the turbomachine cavity.
- Install temporary rails between the end bell and the isolation valve.
- Disconnect piping & connections in the access tunnel at the turbine end of the machine.
- Install an alignment turret in the access tunnel.
- Install the turbomachine cask.
- Withdraw the turbomachine into the cask.
- Close the isolation valve.
- Close the cask valve.
- Remove the cask.

In addition to the usual rigging equipment involved in handling heavy machinery, certain special accessories are required in order to perform the foregoing functions. These are:

- PCRV plug handling equipment
- Bearing support
- Alignment turret
- Removable rails

Fig. 6-4. HTGR helium gas turbine

Fig. 6-5. Turbomachine installed (reference design)

that when the hold-down studs have been removed the entire assembly can be moved to a parking location. Details of the runway system from the generator buildings are shown in Fig. 6-6. The flange couplings of the shaft between the turbomachine and the containment closure are disconnected and the shaft removed by means of an overhead monorail crane installed on the underside of the PCRV recess. Removal of the concrete-shield plug and thrust bearing assembly which closes the turbomachine cavity involves a number of steps, all of which are performed in air, "hands on." Using an air bearing pallet arrangement, the plug assembly is moved to a parking position.

After removal of the PCRV plug, the turbomachine is still retained by a thrust ring at the mounting flange and a torque pin at the turbine end of the machine. Before the thrust ring is removed, a maintenance seal in the compressor end bell is inflated. A set of temporary rails is then installed between the thrust ring and the end of the turbomachine cavity. Access is subsequently made to the bearing inspection tunnel at the compressor end of the machine, and the torque pin is removed. At the same time service system pipe flanges and instrumentation leads in the same area are disconnected.

A bearing support assembly is next installed, utilizing the monorail and the temporary rails, and is bolted to the compressor inlet bell. The assembly consists of two sets of roller bearings near the horizontal axis of the turbomachine which provide support for the compressor end of the machine as it is removed. Adjustment of these bearings on two 90 deg axes is provided by remotely operated screw jacks. An isolation valve is then moved into position on the monorail and bolted to the face of the turbomachine cavity. The valve is closed and sealed by inflatable seals to the turbomachine cavity, and the interspace atmosphere is changed from air to helium. Concurrently a remotely operated alignment turret is installed in the bearing access tunnel. Four extendible cylinders on the turret mounted at 90 deg to one another engage an internal groove on the end of the turbomachine. The purpose of the turret is to relieve the load from the locating

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Fig. 6-6. Twin 4000 MW(t) GT-HTGR plot plan

trunnions, guide the turbomachine off the circumferential seals, and lower the forward bearing pad into engagement with the centre rail. Lastly, the turbomachine handling cask is installed, evacuated, and backfilled with helium, the isolation valve is opened, and the cask rails are extended into the cavity. The turbomachine is now ready for removal as shown in Fig. 6-7.

After activation of the turret cylinders to relieve the load from the supporting trunnions, power is applied to the driving mechanism in the cask and the machine withdrawn to clear the supports and trunnions. Once the machine is installed in the cask the rails are withdrawn and the isolation valve closed. The atmosphere in the cask is then changed from helium to air, the anchors are released, air is supplied to the air bearings, and the cask is towed away on prepared runways (as shown in Fig. 6-6). The replacement machine is installed in approximately a reverse sequence to that described above. Various adjustments in alignment may have to be performed depending upon the degree of interchangeability achieved in manufacture of the units. These adjustments are facilitated by groups of television cameras located to provide a means of determining alignment of the various seals prior to engagement.

As mentioned earlier, the maintenance studies involved not only establishing sequences of operation but also included the conceptual design of special equipment items. Conceptual design work was done in sufficient depth to establish the feasibility of the bearing support assemblies and the alignment turret. The bearing support assembly concept shown in Fig. 6-8 provides adjustment of the turbomachine relative to the turbomachine cavity on two axes at 90 deg to one another. The assembly consists of a support plate which is bolted to the turbomachine end bell and carries electrically-activated screw jacks attached to a pair of horizontal beams. The support bearings on which the machine rides during removal and installation are attached to one of the beams. Movement of the beams is remotely controlled. The alignment turret concept shown in Fig. 6-9 has two main purposes, (1) to relieve the load from the turbomachine trunnions and (2) to

Fig. 6-7. Turbomachine prepared for removal (reference design)

Fig. 6-8. Bearing support assembly

Fig. 6-9. Alignment turret

provide axial movement sufficient to disengage both the trunnions and the circumferential seals. The machine consists of two telescoping cylinders driven by a ball screw to provide the axial motion. One large load-carrying screw jack and three smaller screw jacks engage a circumferential groove at the turbine end of the turbomachine. These jacks serve as a grappling device and also as a means of radially aligning the end of the turbomachine to ensure proper engagement of the circumferential seals.

The actual installation and removal of a large horizontal turbomachine by remote means, although novel, is considered to be a relatively straightforward maintenance operation. Technology in the areas of machine handling, support means, and alignment techniques from large open-cycle industrial gas turbines and steam turbines will be applicable. Maintenance-related technology from the SC-HTGR in the areas of remote removal/installation of contaminated components (e.g. circulator and steam generator) together with the design of special purpose mechanisms, handling machines, and casks will be applicable.

6.1.2. Core-To-Turbine Hot Gas Duct Maintenance

The metallic duct transporting high temperature helium from the core to the turbine is positioned in the vertical compressor discharge cavity in the PCR. Access to this duct assembly is from the top of the PCR. The plant maintenance requirements must provide for the replacement of the hot duct, which, although designed to last the plant lifetime, may fail from unforeseen circumstances. Although this duct transports the gas at the highest temperature in the plant (850°C), the vertical portion of the metallic assembly is immersed in the much cooler compressor discharge gas (175°C), and, with the appropriate insulation, metal temperatures considerably below the turbine inlet value are expected. This, together with the modest internal pressure differential of around 50 psi (compressive loading), results in an operating environment which is felt to be conducive to extended trouble free life.

To remove the turbomachine it is necessary to retract the hot gas duct a few inches from the machine casing using a remote mechanism. If during a routine ISI operation a flaw is observed in the duct assembly or the plant conditioning monitoring system indicates a failure, then provision must be made for removal and replacement of the assembly. Although detailed fission product plate-out calculations remain to be done, it is expected that the activity level on this duct inner face will be high after extended service. Should a faulted duct require removal it would be replaced by a new assembly, with no attempt made to repair the highly contaminated duct. To ensure the feasibility of such a duct removal/installation and to evaluate possible impact on the primary system design, a conceptual study has been performed to establish the maintenance procedure and identify the necessary special purpose equipment. A summary of maintenance considerations for the hot gas duct is given in Table 6-3.

The initial study for the sequence of operations for removal of the hot gas duct has been completed and details are given in Table 6-4. The basic geometry of the removable segment of the hot duct and its location within the PCRV are shown in Fig. 6-10. In the conceptual scheme established the hot duct will be supported on trunnions which will support both its weight and the maximum forces acting on the duct. The trunnions will also align a new assembly with sufficient accuracy to facilitate a remote coupling of the flanged joint. Insertion of the lower portion of the hot duct into the turbomachine will be accomplished automatically through mating chamfers on the two assemblies as the duct is lowered into position. Access for manual operation of the flange clamp mechanism and flange retraction mechanism will be provided in the PCRV. The approximate configuration of the assumed access cavity is shown in Fig. 6-10. A pair of guide rails is permanently installed in the vertical compressor discharge duct to guide a cable-suspended grapple from the cask to the top of the duct. The same rails mate with rollers attached to the duct segment to assure its orientation during removal and replacement. Although many of the mechanical design details remain to be resolved, a mechanism concept has been generated

TABLE 6-3
GT-HTGR TURBINE INLET DUCT MAINTENANCE PHILOSOPHY

- Duct assembly to be designed for full plant life (final geometries and materials selection yet to be made).
- Duct assembly to be replaceable without man access.
- Initial concept of telescoping bellows section (local retraction for machine installation/removal) felt to be a high risk design because of:
 - Thermal barrier problem
 - Complex assembly
 - Additional mechanism necessary
 - Materials problems
- Proposed approach for conceptual design involves disconnection of assembly in plane of horizontal reactor outlet duct, permitting retraction of complete assembly upwards about 458 mm (18 in.) for turbomachine installation/removal.
- Conventional remotely actuated mechanism for duct removal.
- Duct assembly offset in compressor discharge cavity to allow access to flanged joint.
- Guide rails provisioned in cavity for accuracy of duct assembly translation.
- TV camera coverage of duct insertion in turbomachine casing, together with sensors to ensure register proper fit, desirable.

TABLE 6-4
HOT DUCT MAINTENANCE PROCEDURES

Removal of the hot duct from the PCRV after a period of service entails the following basic procedures:

- Unbolt and remove the secondary closure flange.
- Unbolt the primary closure.
- Install the hot duct flange adapter and isolation valve.
- Install the hot duct handling cask.
- Evacuate air in cask, between valves, etc., and backfill with helium.
- Open valves and use cask hoist and grapple to remove the primary closure.
- Close valves to maintain helium atmosphere in PCRV and transfer closure to temporary storage.
- Reinstall cask in isolation valve; open valves and lower hoist until grapple engages lifting adapter on the hot duct.
- Remove the access flange in the hot duct maintenance cavity and start helium purge flow to minimize air ingress.
- Install the extension on the hot duct release tool and engage the tool with the mating spline on the hot duct assembly.
- Open the flange clamp assembly.
- Change the tool position and retract the flange.
- Move the tool again to lock the flange in the retracted position.
- Disengage the tool from the hot duct.
- Raise the duct assembly with the cask hoist. NOTE: 2 to 3 ft of motion will permit turbomachine removal.
- Close the cask and isolation valves.
- Transport the duct assembly to the service area.

Fig. 6-10. Hot duct handling study, GT-HTGR

for retracting the duct in a near horizontal plane (prior to vertical motivation), as shown in Fig. 6-11. Following decoupling the duct assembly is pulled up into the shielded cask.

Although the actual duct replacement operation is unique to the direct cycle plant, initial work has indicated that the techniques bear a resemblance to planned maintenance procedures for the SC-HTGR. The duct handling cask, isolation valves, and adapter are similar to the circulator handling equipment designed for the steam cycle lead plant.

6.1.3. Precooler Maintenance

The general maintenance philosophy developed for the helium-to-water precooler is summarized in Table 6-5. Because the major maintenance operations for this heat exchanger are heavily influenced by its configuration, the detailed repair and maintenance studies on the precooler have been deferred until comprehensive design evaluations in process are complete. Prior to these evaluations, a limited mechanical design effort was undertaken to address installation, ISI, and maintainability of the waterbox interfaces in the precooler design shown in Fig. 6-12.

In the reference plant design the location of the precooler cavity above the turbomachine precludes penetration of the water inlet and outlet pipes through the bottom head of the PCR. It was necessary, therefore, to route this piping obliquely through the PCR sidewall (as shown in Fig. 6-13) to provide a straight traverse path for the rectilinear-acting tube plugging machine.

The conceptual design of the tube plugging machine shown in Fig. 6-13 was pursued to the point at which the machine operating envelope was confirmed. The machine has been conceptually designed to accommodate replaceable heads that are capable of leak detection by sniffing and repair of the faulty lead tube by plug insertion. In addition, the pipe liner diameter

Fig. 6-11. Hot duct handling study, GT-HTGR

TABLE 6-5
GT-HTGR PRECOOLER MAINTENANCE PHILOSOPHY

- General characteristics

- Helium-to-water exchanger

- No phase-change or fouling

- Very low operating metal temperature

- Design to yield properties rather than creep

- Low fission product plateout predicted

- Low internal pressure differential (helium > water)

- Code approved materials

- Surface geometry and fabrication methods of proven type

- Designed for full life of plant (280,000 hr)

- Modular construction eases development, handling, assembly

- Complete exchanger assembly at factory

- Removal/replacement capability without man access, but operation eased if access at lower end (from compressor inlet duct).

- Maintenance and repair work to be performed remotely.

- Leak detection down to module level.

- External instrumentation (pressure and radiation monitors) to detect leaks (helium into water).

- Lead tube plugging from outside PCRV using special-purpose machine. Manual repair work in plane of tubesheet also possible.

- Exchanger replaceability

- In the event of significant loss of plant performance from excessive module plugging or an unlikely major structural failure, the complete precooler could be removed and replaced. This situation is, however, considered remote because of the conservative nature of the design and the non-hostile operating environment.

Note: Design work on the bottom-supported variant is in progress, and maintenance studies are in initial stage.

Fig. 6-12. Precooler-hex module, bottom supported, 270°F water outlet

Fig. 6-13. Precooler water piping mechanical study

is large enough to permit the use of controlled transporter mechanisms equipped with examination equipment for ISI of the pressure boundary weld and the outside surface of the water pipes. If subsequent calculations show that the environment in the vicinity of the tube sheets permits man access, the diameter of the water pipes will have to be increased slightly above the current 508 mm (20 in.) to facilitate direct manual plugging of the lead tubes.

For the purposes of precoolers removal and reinstallation (of a new unit) the waterbox (water pipe interface) can be cut, rewelded, and inspected by remote methods at the 10 deg angle plane indicated in Fig. 6-13. This particular parting line will provide a radial clearance of at least 25 mm (1 in.) between the waterbox and its cavity for vertical movement of the waterbox during precoolers removal.

6.1.4. Recuperator Maintenance

The general maintenance philosophy developed for the recuperator (summarized in Table 6-6 and Fig. 6-14) presents the recuperator design as it stood prior to the comprehensive heat exchanger design studies currently in process. As is the case with the precoolers, major maintenance operations on the recuperator will be sensitive to the particular heat exchanger configuration employed. Maintenance and ISI considerations are expected to have a significant influence on the final selection of the recuperator configuration.

Since the recuperator is a helium-to-helium heat exchanger, the effects of leaks that may occur in service manifest themselves in the area of performance and do not affect the operation and safety of the plant. If failures occur in the small individual tubes, the effect on plant performance will be minor. On the other hand, if failures occur in the lead tubes, the subsequent leakage (together with increased noise level in the primary system) will require remedial action, and the faulted module must

TABLE 6-6
GT-HTGR RECUPERATOR MAINTENANCE PHILOSOPHY

- General characteristics

Helium-to-helium exchanger (small leaks permissible)

No phase-change, fouling, corrosion, or erosion

Modest metal temperatures and pressure differentials

Code approved materials

Surface geometry and fabrication methods of proven type

Designed for full life of plant (280,000 hr)

Modular construction eases development, handling, assembly

Complete exchanger assembly at factory

- Removal/replacement capability without man access

- Maintenance and repair work to be performed remotely

- Leak detection down to module level (procedures to be resolved)

- Lead tube plugging capability (procedure to be developed)

- Repair of minor leaks could be delayed until plant refuel shutdown

- Man access would ease repair work

- Exchanger replaceability

In the event of significant loss of plant performance from excessive module plugging or an unlikely major structural failure, the complete recuperator could be removed and replaced. This situation is, however, considered remote because of the conservative nature of the design and the non-hostile operating environment.

Note: Design work on the IRT variant is in progress and maintenance studies are in initial stage.

Fig. 6-14. Recuperator with integral return tube modules

be plugged. Since the level of the dose rates in the recuperator cavity above the support plate is expected to preclude man access, the detection of the faulted module and the plugging operation must be done by remote means. The mechanical design of the recuperator top headering is thus strongly influenced by maintenance requirements. Design studies in this area are currently in progress and two early variants established for evaluation are shown in Fig. 6-15.

6.1.5. Helium Control Valve Maintenance

Each power conversion loop has four valves, (safety, bypass, attempera-tion, and trim valves) to satisfy plant control and safety functions. All four valves are integrated into the PCRV and are installed in separate cavities in the PCRV top head. All four valves present similar maintenance and handling problems, and so it was decided to base the study on the primary bypass valve. This particular valve is one of the larger units and presents the most severe actuator requirements. A typical arrangement of the bypass valve, actuator, and dump valves and reservoirs to accommodate the fast opening requirement is shown in Fig. 6-16. The generalized helium control valve maintenance philosophy is given in Table 6-7.

Each valve is bolted into its penetration at a flange several feet below the top of the PCRV. Preliminary shielding studies indicate that limited access to these bolts should be possible following 10-day shutdown. However, actual removal of a valve into a shielded cask will be performed remotely, both for personnel protection and for prevention of air ingress to the primary coolant.

It is proposed that the valve be designed to integrate the valve seat with the valve body, thus permitting the two to be preassembled and properly matched. This would also facilitate remote handling of the assembly as only one grappling operation is involved. It appears reasonable to assume that the valves can be handled by the hot duct removal cask. Although of a smaller diameter than the hot duct, the valves are approximately the same overall length.

Fig. 6-15. Recuperator top gallery mechanical study

Fig. 6-16. Bypass valve system

TABLE 6-7
GT-HTGR VALVE(S) MAINTENANCE PHILOSOPHY

● General characteristics

Conceptual designs done to identify major features
Configurations satisfy plant control and safety requirements
Conventional linearly actuated type (actuators not yet designed)
Integrated approach eliminates heavy walled structures
Clean operating environment
Code approved materials (not yet fully defined)
Valves designed for full plant operating life
State-of-the-art fabrication
Factory fabrication, assembly, and calibration

- Valve assembly to be replaceable without man access.
- Valves accessible from top of PCR.V.
- Conventional remotely actuated mechanisms for valve assembly (and lower seal unit) removal.
- Sequence of valve removal has been established.
- Maintenance will include period in situ testing of the valves.

The sequence of operations to remove the valve is given in Table 6-8. An isolation valve and cask installed over a valve penetration preparatory to removal of the bypass valve is shown in Fig. 6-17. Provision will be made in the cask to guide the grapple and valve in order to align the valve seat register and seals during installation. It is anticipated that the grapple and grapppling features will be similar to those used for control rod drives and other HTGR equipment.

6.2. INSERVICE INSPECTION (ISI) STUDY

For the direct cycle integrated plant the issue of ISI has added importance since all of the power conversion system components are installed inside the reactor vessel and are thus not accessible for direct visual inspection. An important distinction is made between ISI procedures necessary for regulatory and licensing commitments and those projected for the future to satisfy plant owners' requirements.

6.2.1. Importance of ISI

ISI requirements are continually becoming more demanding to satisfy the various code requirements, regulatory guides, and licensing commitments. Regulatory guides published to date which address aspects of ISI have been specifically directed toward LWRs. In some cases HTGR applicability can be clearly recognized for the vessel structures. Other sources of inspection requirements include 10 CFR 50 and commitments derived in the course of licensing discussions with NRC. Utility ISI practices which are not part of the normal mandatory program imposed by the regulations may reflect a variety of concerns associated with past operating experience. It is difficult to anticipate these supplemental ISI practices, which are principally directed towards the curtailing of unplanned outages.

An important aspect of ISI is the impact on component design, plant

TABLE 6-8
HELIUM CONTROL VALVE(S) MAINTENANCE

- Current valve is an integral unit with the internals, actuator, and seat element embodied in a single assembly with grapple engagement fittings for remote removal. The removal sequence has been conceptually established as follows:
 - Remove valve cavity closure
 - Remove hold down nuts and actuator connections
 - Install the reactor isolation valve
 - Position and secure the duct removal cask on the isolation valve
 - Engage the cask grapple with the top of the valve assembly
 - Change the atmosphere in the cask and actuator cavity to helium
 - Withdraw the valve into the cask
 - Close the cask valve
 - Remove the cask
- Dose rate calculations in progress on the valve assemblies to determine decontamination procedure necessary to permit "hands-on" maintenance.

Fig. 6-17. Installation of bypass valve system maintenance

layout, and access provisions. It is prudent therefore to consider ISI requirements during the plant conceptual design and to pursue a policy of providing inspectable designs.

6.2.2. Applicable ISI Technology

ASME Boiler and Pressure Vessel Code, Section X1, Division 2, Rules for InService Inspection and Testing of Components of Gas-Cooled Nuclear Power Plants, was initially issued as Winter 1975 Addenda to Section X1, Division 1 for LWRs. As a result of technical infractions associated with the publication procedures followed at that time, the code was subsequently withdrawn from issue during Summer 1976 and a separate Summer 1977 issue released for a one year "trial use and comment" period.

Section X1, Division 2 rules of the ASME Code address a single concept of the gas-cooled reactor, the SC-HTGR. Other concepts such as the direct cycle were of lesser priority at the time of initial code development and although the code in its present form is not strictly applicable, major design features characteristic of HTGR code applicability can be identified for the direct cycle plant. Components unique to the GT-HTGR, especially the turbomachine, hot duct, precooler, recuperator, and control valves, will require special attention in developing a program of ISI and testing.

6.2.3. Initial ISI Study for GT-HTGR

The rationale behind the selection of requirements for the above unique components (in the order of importance identified by utility inputs) must not only address safety related aspects, but should also consider the possibility of economic repercussions as a result of failures leading to prolonged shutdowns that may affect major component replacements and repairs. The initial studies of ISI requirements for the GT-HTGR plant have been directed toward the above five unique components. Although still

in a very early stage of development, the investigation has attempted to identify the type of inspection envisioned for key subelements in each of the major component items. Accordingly the method of examination has been tentatively identified and the impact on component design noted.

The initial study, although limited in scope, resulted in the formulation of tables identifying types of examination and methods for various regions of the major components. Tentative data for the turbomachine, hot-gas duct, precooler, recuperator, and helium control valves are shown in Tables 6-9, 6-10, 6-11, 6-12, and 6-13 respectively. It should be emphasized that the data presented in these tables are by no means final recommendations, but rather a set of initial parameters prepared as a part of the ISI scoping effort, which was essentially concerned with the impact of ISI on plant and component conceptual design.

TABLE 6-9
GT-HTGR TURBOMACHINE INSERVICE INSPECTION

Region or Subcomponent	Type of Examination	Method	Design Impact	Remarks
Bearings	Visual	From man access areas Borescopes Fiber optics	Access space environment	Development of specialized ISI methods necessary
	Direct contact	Acoustic emission transducers Oil analysis	Shielding	Experience from industrial gas turbine applicable
Blading	Visual	Remote	Viewing device accessibility	Proposed in-place exam by borescope for gross deformation
Rotor Discs Shaft Couplings Coupling bolts	Visual Surface and volumetric direct contact	Direct, manual gas coupled acoustic emission transducers	Removability of turbo-machine unit for shop examination	Existing equipment and techniques generally applicable
Structures and supports	Verification of continuing integrity	Visual in-place or design redundancy with monitoring	Accessibility for viewing in-place or instrumented to detect change	Existing equipment
Seals	No specific ISI requirements; visual examination desirable			
Welds	Volumetric of primary boundaries	Remote ultrasonic	Accessibility to locate transducer over welds	Adaptation of existing U/T equipment
Service systems piping and welds	Surface or volumetric	Manual	Access to welds	Adaptation of existing U/T equipment
Instrumentation and sensors	No specific ISI requirements; however, should be capable of detecting change in unit performance			
Mechanical closures	Volumetric for stud bolts > 2 in.	Remote ultrasonic	Accessibility to locate transducer over stud. Stud removability	Adaptation of existing equipment from LWRs
(Primary boundaries)	Visual for stud bolts < 2 in.	Remote		

TABLE 6-10
GT-HTGR TURBINE INLET DUCT INSERVICE INSPECTION

Region or Subcomponent	Type of Examination	ISI Method	Design Impact	Remarks
Materials Metals Insulation	Material surveillance	Tests covering: Creep rupture and fatigue interaction Tensile, dimensional stability, mechanical properties	Specimen placement and retrieval - lower plenum sidewall area. Accessibility provisions. Specimen handling mechanisms including costs	Specimen handling mechanisms, cask designs to be developed
Support structures Flanges Bellows Mechanisms Seals Welds Machine interface	Visual	Remote from lower plenum region via special penetrations located opposite cross duct entrances From top of PCRV via compressor outlet duct via special penetrations through primary closure	Number of penetrations in duct top head closure to be determined	Existing remote viewing devices to be adapted to suit refueling shutdown environment Special lighting effects critical to viewing resolution
Instrumentation and sensors	No specific ISI requirements; however, should be capable of detecting change indicative of breached barriers/loss of insulation.			

TABLE 6-11
GT-HTGR PRECOOLER (BOTTOM-SUPPORTED DESIGN) INSERVICE INSPECTION

Region or Subcomponent	Type of Examination	Examination Method	Design Impact	Remarks
Main support flange	ISI exempt	NA	Redundant load path required in design	Design currently under way
Module tubesheet	ISI exempt	NA	None	As part of support system these subcomponents are in compression
Module subheaders	ISI exempt	NA	None Allow 5D bends for sub-header inspectability	
Individual tubes	Unlike the HHT approach ISI of small tubes not provisioned in current design		ISI requirement would necessitate tube diameter increase and reduction in module number	Tube leaks (helium into water) necessitate isolation valve actuation and rapid draining of faulted section
Peripheral seals	ISI exempt	NA	Monitoring may be required if CACS operation sensitive to leaking seals	Effect of seal leakage reflected in performance loss
Support structure	ISI exempt	NA	Redundant load paths required in design	In situ structure failure monitor required
Water penetration pressure boundaries	Volumetric examination of welds	Remote/manual ultrasonic	Main water pipes should be increased to 22-in. diam. to permit man access	Development of traversing system required
Mechanical closures (primary boundary)	Volumetric for stud bolts > 2 in.	Remote ultrasonic	Accessibility to locate transducer over stud. Stud removability	Adaption of existing U/T equipment from LWRs
	Visual for stud bolts ≤ 2 in.	Remote		

TABLE 6-12
GR-HTGR RECUPERATOR (INTEGRAL RETURN TUBE DESIGN) INSERVICE INSPECTION

Region or Subcomponent	Type of Examination	Examination Method	Design Impact	Remarks
Main tubesheet	Volumetric exam of welds Not applicable if single forging	Remote ultrasonic	Access penetrations in cavity closure and upper seal plate	Development of traversing system and adaptation of transducer head required Upper gallery details not yet resolved
Module tubesheet	ISI exempt	NA	Back-up support system for modules required in the event failure of upper tubesheet or subheader occurs	Selection of module geometry and subheaders not yet finalized
Module subheaders	ISI exempt	Eddy current if failure of subheader unacceptable due to sonics or tube whip		
Individual tubes	No current requirements for ISI of small tubes			No safety implications
Return tubes/bellows	ISI exempt	NA	None	Effect of failure reflected in loss of performance
Peripheral seals	ISI exempt	NA	Monitoring may be required if CACS operation sensitive to leaking seals	
Support structure	ISI exempt	NA	Redundant load parts should be incorporated into design	
Mechanical closures (primary boundaries)	Volumetric for stud bolts > 2 in.	Remote ultrasonic	Accessibility to locate transducer over stud. Stud removability	Adaptation of existing U/T equipment from LWRs
	Visual for stud bolts ≤ 2 in.	Remote		

TABLE 6-13
GT-HTGR VALVE(S) INSERVICE INSPECTION

Region or Subcomponent	Type of Examination	Examination Method	Design Impact	Remarks
Support structures				
● External	Visual, surface, or volumetric	Remote	Avoid integrally welded support members (with primary boundaries)	Existing devices to be adapted to component configuration
● Internal	Surface or volumetric	When made accessible for other reasons		
Actuators	Valve full stroke exercised	If frequently exercised during operation no tests necessary; otherwise check during shut-down	Accessibility	Conceptual designs lack details in these areas
Plug and seat	Leak rate			
Stem guides	Visual on valve intervals	When made accessible for other reasons		
Thermal barriers	Performance check	Monitoring of cooling water temp	Visual examination of accessible surfaces normally required. Visual proposed when valve exposed for other reasons	
Welds	Volumetric of primary boundaries	Remote ultrasonic	Forgings preferred for valve bonnet. Liner support accessibility	Existing equipment to suit application
Seals	No specific ISI requirements			
Instrumentation	No specific ISI requirements, but should be capable of verifying valve operability			
Mechanical closure (primary boundary)	Volumetric for stud bolts > 2 in. Visual for stud bolts ≤ 2 in.	Remote ultrasonic Remote	Accessibility to locate transducer over stud. Stud removability	Adaptation of existing U/T equipment from LWRs