DOE-HTGR-86-087
Revision 1

UNCLASSIFIED

Received 1/1/87
MAY 06 1987

HTGR

METALS TECHNOLOGY DEVELOPMENT PLAN

DO NOT MICROFILM
COVER

AUTHORS/CONTRACTORS

GA TECHNOLOGIES INC.
OAK RIDGE NATIONAL LABORATORY
COMBUSTION ENGINEERING

DISTRIBUTION OF THIS DOCUMENT IS UNLIMITED

This document is
PUBLICLY RELEASABLE H.A.
6/17/02

PER DAVID STEINMAN, General
ATOMIC CLASSIFICATION OFFICER,
MEMO DATED 01/22/01.

ISSUED BY GA TECHNOLOGIES INC.
FOR THE DEPARTMENT OF ENERGY
CONTRACT DE-AC03-84SF11962

MARCH 1987

UNCLASSIFIED

DISCLAIMER

This report was prepared as an account of work sponsored by an agency of the United States Government. Neither the United States Government nor any agency Thereof, nor any of their employees, makes any warranty, express or implied, or assumes any legal liability or responsibility for the accuracy, completeness, or usefulness of any information, apparatus, product, or process disclosed, or represents that its use would not infringe privately owned rights. Reference herein to any specific commercial product, process, or service by trade name, trademark, manufacturer, or otherwise does not necessarily constitute or imply its endorsement, recommendation, or favoring by the United States Government or any agency thereof. The views and opinions of authors expressed herein do not necessarily state or reflect those of the United States Government or any agency thereof.

DISCLAIMER

Portions of this document may be illegible in electronic image products. Images are produced from the best available original document.

UNCLASSIFIED

DOE/HTGR--86-087-Rev.1

TI87 027680

NOTICE

This report contains information of a preliminary nature and was prepared primarily for internal use at the originating installation. It is subject to revision or correction and therefore does not represent a final report. It is passed to the recipient in confidence and should not be abstracted or further disclosed without the approval of the originating installation or USDOE Office of Scientific and Technical Information, Oak Ridge, TN 37830.

METALS TECHNOLOGY DEVELOPMENT PLAN

DISCLAIMER

~~This report was prepared as an account of work sponsored by an agency of the United States Government. Neither the United States Government nor any agency thereof, nor any of their employees, makes any warranty, express or implied, or assumes any legal liability or responsibility for the accuracy, completeness, or usefulness of any information, apparatus, product, or process disclosed, or represents that its use would not infringe privately owned rights. Reference herein to any specific commercial product, process, or service by trade name, trademark, manufacturer, or otherwise does not necessarily constitute or imply its endorsement, recommendation, or favoring by the United States Government or any agency thereof. The views and opinions of authors expressed herein do not necessarily state or reflect those of the United States Government or any agency thereof.~~

DISTRIBUTION OF THIS DOCUMENT IS UNLIMITED

This document is
PUBLICLY RELEASABLE

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

DOE Contract No. DE-AC03-84SF11962

GA Project 6300

MARCH 1987

UNCLASSIFIED

MASTER

JSW

GA Technologies Inc.**PROJECT CONTROL DOCUMENT APPROVAL SUMMARY**

TITLE METALS TECHNOLOGY DEVELOPMENT PLAN MHTGR					PROJECT 6300	CODE PPM	DOCUMENT NO. PC- 000210	REV. 4
C.M. STAMP	REV.	PREPARED BY	RESOURCE/ SUPPORT APPROVAL	FUNDING PROJECT APPROVAL	APPLICABLE PROJECT APPROVAL	DESCRIPTION/ CWBS NO.
0	W. S. Betts	C.F. McDonald	J-D. Wistrom		Initial Issue 6216030102
1	W. S. Betts	C.F. McDonald	J-D. Wistrom		Major update - add Section 4.3 6216030102
2	W. S. Betts	C.F. McDonald	J-D. Wistrom		Major update - add Section 4.4 delete ref to FAW 6216030102
3	W. S. Betts	C.F. McDonald R. F. Turner	J-D. Wistrom	G. Bramblett	Major update - add Section 5 6216030102 HTGR-86-087
4	W. S. Betts	R. E. Vollman	J. D. Wistrom		DOE-HTGR-86-087, Rev. 1 6216030101 - Update all sections. Delete Section 4.2, move 4.3 to 4.2 and 4.4 to 4.3.		

Cover	1 (Unnumbered)
Title Page	1 (Unnumbered)
Approval Summary	1
ii - xi	10
1-1 - 1-3	3
2-1 - 2-4	4
3-1 - 3-7	7
4-1 - 4-119	119
5-1 - 5-102	102
Total	248

APPROVED FOR RELEASE
 ON 08-11-2011 BY [REDACTED]
 AUTHORITY: [REDACTED]
 DATE: 08-11-2011
 BY: [REDACTED]
 FOR: [REDACTED]
 U.S. DEPARTMENT OF ENERGY

CONTENTS

NOMENCLATURE	vi
1. INTRODUCTION	1-1
2. SCOPE	2-1
3. CURRENT STATUS	3-1
3.1 Effects of Primary Coolant Chemistry and Temperature on Alloy 800H	3-1
3.2 Effects of Irradiation on Properties of Alloy 800H	3-3
3.3 Properties of 2-1/4 Cr-1 Mo Base Metal and Weldments	3-4
3.4 Irradiation Effects on the Reactor Vessel Material	3-5
3.5 Properties of SA533B at Elevated Temperature	3-6
4. REQUIREMENTS	4-1
4.1 Hot Duct	4-3
4.1.1 Design Description	4-3
4.1.2 Relevant Functions	4-5
4.1.3 Design Requirements/Assumptions	4-5
4.1.4 Technology Requirements	4-25
4.1.5 Schedule Requirements	4-27
4.1.6 References	4-27
4.2 Steam Generator	4-59
4.2.1 Design Description	4-59
4.2.2 Relevant Functions	4-61
4.2.3 Design Requirements/Assumptions	4-61
4.2.4 Technology Requirements	4-67
4.2.5 Schedule Requirements	4-72
4.2.6 References	4-72
4.3 Reactor Vessel	4-91
4.3.1 Design Description	4-91
4.3.2 Relevant Functions	4-92
4.3.3 Design Requirements/Assumptions	4-93
4.3.4 Technology Requirements	4-101
4.3.5 Schedule Requirements	4-104
4.3.6 References	4-107
5. BASE TECHNOLOGY DEVELOPMENT PROGRAM	5-1
5.1 Hot Duct	5-1
5.1.1 Data Needed	5-3
5.1.2 Approach and Rationale for Test Programs	5-3
5.1.3 Test Programs	5-8
5.1.4 Schedule	5-26
5.1.5 Costs	5-26

5.2	Steam Generator	5-46
5.2.1	Data Needed	5-46
5.2.2	Approach and Rationale for Test Programs	5-46
5.2.3	Test Programs	5-52
5.2.4	Schedule	5-52
5.2.5	Costs	5-75
5.3	Reactor Vessel	5-92
5.3.1	Data Needed	5-92
5.3.2	Approach and Rationale for Test Programs	5-92
5.3.3	Test Programs	5-95
5.3.4	Schedule	5-95
5.3.5	Costs	5-98
5.4	Overall Costs and Schedule Options	5-101

TABLES

4.1-1	Metallic materials - subcomponents of hot duct	4-31
4.1-2	Design temperatures for MHTGR	4-32
4.1-2a	The core outlet helium temperature during a reactor trip from 100% power AOO No. 1	4-33
4.1-2b	Primary coolant temperature during pressurized conduction cooldown - SRDC No. 2	4-34
4.1-3	Design duty cycle events	4-35
4.1-4	"Long term" expected and design primary coolant impurities	4-38
4.1-5	Summary of stresses induced by thermal and mechanical loads on hot duct components	4-39
4.1-6	Allowable strain range of Alloy 800H for selected cyclic conditions	4-40
4.1-7	Aging study on Alloy 800H weldments (Inconel 82 filler material, full tube specimens) aging time 3000 h	4-41
4.2-1	Principal materials for the steam generator components	4-75
4.2-1a	Principal welding processes and filler materials	4-76
4.2-2	Relevant functions of the steam generator	4-77
4.2-3	Primary and secondary side conditions	4-78
4.2-4	Steam generator subsystem operational points within the uncertainty envelope at 100% feedwater flow	4-79
4.2-5	Feedwater chemistry requirements	4-81
4.3-1	Principal materials for the reactor vessel	4-107
4.3-2	Relevant functions of the reactor vessel	4-108
4.3-3	Reactor vessel design temperatures for MHTGR	4-109
4.3-4	Equivalent fast fluence for MHTGR reactor vessel beltline and closure head	4-110
4.3-5	Projected transition temperature shifts for the MHTGR vessel beltline and head	4-111

FIGURES

4.1-1	Modular reactor hot duct evaluation view	4-42
4.1-1b	Hot duct guide rails and supports - cross section looking toward reactor vessel	4-43

4.1-2	MHTGR integrated approach functional analysis for safe, economical power	4-44
4.1-3	MHTGR integrated approach functional analysis for safe, economical power	4-45
4.1-4	Fatigue results of Alloy 800H at 850°C (1562°F)	4-46
4.1-5	Effect of an HTGR helium environment on the high cycle fatigue strength of Incoloy 800H	4-47
4.1-6	Effect of thermal aging in helium on the high cycle fatigue of Incoloy 800H at 760°C	4-48
4.1-7	Effect of pretest thermal aging in HTGR helium on the 10 ⁸ cycle fatigue strength for Incoloy 800H at 760°C (1400°F)	4-48
4.1-8	The effect of thermal aging on tensile strength of Alloy 800H.	4-49
4.1-9	Effect of 10,000 h thermal aging (in air) on the ductility of Alloy 800H (GA 409) at room temperature and at the aging temperature	4-50
4.1-10	Relation between the strain range and the macroscopic fatigue life	4-51
4.1-11	Room temperature toughness as a function of aging temperature for Alloy 800H	4-52
4.1-12	Comparison between weld metal Inconel 82 and Alloy 800H	4-53
4.1-13	Postirradiation ductility of experimental heats of Alloy 800 irradiated at 650°C and 700°C to about 0.8×10^{21} neutron/cm ² (thermal) and tested at 700°C	4-54
4.1-14	Creep-rupture results at 705°C of unirradiated and General Electric Test Reactor irradiated Alloy 800	4-55
4.1-15	Creep rupture of experimental Alloy 800 containing 0.13% C and (a) 0.17% or (b) 0.10% Ti	4-56
4.1-16	Irradiated and unirradiated fatigue behavior of Alloy 800 tested at 700°C (1292°F) compared with predictions based on tensile data	4-57
4.1-17	U.S. HTGR program reference schedule	4-58
4.2-1	Sectional view of steam generator subsystem	4-82
4.2-2	Section of SGSS taken at steam nozzle CL	4-83
4.2-3a	MHTGR integrated approach functional analysis	4-84
4.2-3b	MHTGR integrated approach functional analysis	4-85
4.2-4	Steam generator subsystem operational uncertainty envelope around MOL expected conditions at 100% feedwater flow	4-86
4.2-5	Carbon contents of 2-1/4 Cr-1 Mo steel exposed to high-temperature gas cooled reactor helium	4-87
4.2-6	Fatigue crack growth rate (da/dn) versus ΔK for annealed 2.25 Cr-1 Mo steel plate at 24°C and 260°C. A sinusoidal wave form at 10 Hz and an R ratio of 0.1 were used	4-88
4.2-7	Creep crack growth rate versus stress intensity, K, of annealed 2.25 Cr-1 Mo steel plate at 538°C compared to that for two other Cr-Mo steels (reproduced from Refs. 22 and 23).	4-89
4.2-8	Creep crack growth rate (CCGR) versus transient value of CCGR, C_t , of annealed 2.25 Cr-1 Mo steel plate at 538°C compared to a Cr-Mo-V steel (reproduced from Ref. 24)	4-89
4.2-9	Code case N47 creep-fatigue damage envelope	4-90
4.2-10	Proposed creep-fatigue damage envelope	4-90
4.3-1	Reactor vessel configuration	4-112
4.3-2	MHTGR integrated approach functional analysis	4-113
4.3-3	MHTGR integrated approach functional analysis	4-114

4.3-4a	Primary helium pressure during pressurized conduction cooldown event (SRDC 1)	4-115
4.3-4b	Peak reactor vessel midwall temperature during a pressurized conduction cooldown event (SRDC 1)	4-115
4.3-5a	Primary helium pressure during a pressurized conduction cooldown with rod withdrawal event (SRDC 4)	4-116
4.3-5b	Peak reactor vessel midwall temperature during a pressurized conduction cooldown with rod withdrawal event (SRDC 4)	4-116
4.3-6	Peak midwall temperatures of reactor vessel during a depressurized conduction cooldown event	4-117
4.3-7	Effect of irradiation temperature on transition temperature increase for an A302-B reference steel	4-118
4.3-8	Effect of irradiation on low copper A533B material at 300°F and 550°F	4-119

NOMENCLATURE

a	flaw (crack) size
AOO	Anticipated Operational Occurrence
BMW	bimetallic welds between Alloy 800H and 2-1/4 Cr-1 Mo
D	diameter
da/dN	fatigue crack growth rate
da/dt	creep crack growth rate
DDN	Design Data Needs
E	elastic modulus
h	thickness of cover plate
K_i	stress intensity for condition i
K_{IC}	critical stress intensity factor
L	length
LCF	low cycle fatigue
$LCF\Delta\epsilon$	low cycle fatigue strain range
N	number of cycles
OBE	operational basis earthquake
p	pressure
R	radius of curvature
RV	reactor vessel
Sa	high cycle fatigue strength
S_m	time-independent design stress intensity
S_R	Stress to cause rupture
S_t	time-dependent design stress intensity
S_u	ultimate tensile strength
S_y	yield strength
S_1	stress to produce 1% total strain
SRDC	Safety Related Design Condition
SSE	safe shutdown earthquake
T	temperature
σ	induced stress
σ_t	thermally induced stress

NOMENCLATURE (Continued)

ϕ	neutron fluence, n/cm^2
ϵ	strain
$\Delta\epsilon$	strain range
Δt	temperature difference
α	coefficient of thermal expansion

LIST OF DEFINITIONS

Anticipated Operational Occurrences: Off-normal, licensing basis events expected to occur once or more in the lifetime of a single plant; i.e., events with a frequency of occurrence >0.025 per plant year.

Correlation: An empirically derived mathematical relationship between two or more variables (e.g., the vapor pressure of cesium over graphite as a function of temperature and fluence).

Design Basis: That information which identifies the specific functions to be performed by a structure, system or component of the plant, and the specific values or ranges of values chosen for controlling parameters as reference bonds for design. These values may be:

1. Constraints derived from generally accepted state-of-the-art practices for achieving functions.
2. Requirements derived from analysis (based on theory and/or experiments) of postulated conditions for which a structure, system, or component must meet its functions.

Design Basis Event: Off-normal, licensing basis events not expected to occur in the lifetime of a single plant but which could occur among a population of plants; i.e., events with a frequency of occurrence between ≤ 0.025 and $>1 \times 10^{-4}$ per plant year.

Design Criterion: A basis for judging the acceptability of a particular action, procedure, or design solution.

Design Method: The overall methodology used to design a system, structure or component which may include one or more computer codes embodying one or more component models, the material property correlations and other input to the codes, and the manner in which these codes are applied in the design process (e.g., the fineness of the spatial mesh used with a finite-difference code).

Design Value: The nominal value of a plant or system operating parameter plus an appropriate margin.

Function: (Functional Analysis) A statement of what is to be achieved.

Functional Analysis: A systems engineering technique or method used to develop (1) functions required to meet established goals, (2) the relationships between functions and related requirements, and (3) the basis and justification for design selections through which the prescribed goals are met. (Functional Analysis is one part of the Integrated Approach.)

Functional Requirement: A bounding quantification that is derived from a functional need as opposed to being imposed by an institutional standard.

Functional Tree: A diagram which displays the interrelationships between functions. The MHTGR functional tree commences with the top-level goal, then continues with subordinate goals and plant states.

Goal: An endpoint or accomplishment towards which an effort is directed. The Integrated Approach has identified the following specific goals supporting the overall objective:

1. Safe, economical power (top level - Goal 0).
2. Maintain safe plant operation (Goal 1).

3. Maintain plant protection (Goal 2).
4. Maintain control of radionuclide release (Goal 3).
5. Maintain emergency preparedness (Goal 4).

Institution: A significant practice, relationship, or organization in a society. The general industry institutional activities are design, construction, operation, maintenance, and decommissioning.

Institutional Requirement: A bounding quantification that is imposed by an institution as opposed to a quantification derived from a functional need.

Integrated Approach: A system engineering technique for establishing and defending a well-developed nuclear plant design.

Licensing Basis Events: The off-normal or accident events used for demonstrating design compliance with NRC safety risk goals.

Nominal Value: The desired quantity/value of a parameter corresponding to a given plant condition.

Parameter: A specific measurable and quantifiable aspect of a physical item.

Requirement: The bounding quantification (limits) of a function.

Safety Related Design Conditions: Those conditions derived from design basis events in which the "safety related" systems, structures, and components (i.e., those needed for compliance with 10CFR100 dose criteria) alone accomplish the required functions of limiting radionuclide release.

Subfunction: One of a group of functions at the next lower level that supports the preceding (higher-level) function.

Technology: Experimental programs conducted to satisfy a technology development need.

Technology Development Need: The technology that must be developed to assure that the selected design meets the design requirements at the specified confidence level.

1. INTRODUCTION

This document presents the plan for the metals technology development required to support the design of the MHTGR within the U.S. National Gas-Cooled Reactor Program. Besides descriptions of the required technology development, cost estimates, and schedules, the plan also includes the associated design functions and design requirements.

The plan follows the principles of the Integrated Approach which is a systematic method for establishing and documenting the design basis for a nuclear power plant in a "top down" manner. An essential element of the Integrated Approach is the Functional Analysis which identifies the functions and associated design requirements, first for the power plant as a whole, then for the systems, subsystems, components, and subcomponents in descending order. Appropriate design selections are also part of the Functional Analysis.

As the Functional Analysis evolves, it is necessary to show that the design requirements for each function have been met (or, as an interim measure, can be met) before proceeding to the subordinate functions. This is done through design analyses and trade studies where different design concepts are evaluated and the best concept selected. The selected design will then be checked against the design requirements with three possible results:

1. The design requirements are met on the basis of existing technology. In those cases the Functional Analysis can proceed directly to the next level.
2. The design requirements are met (or, as an interim measure, are expected to be met) provided certain assumptions about nonexistent technology can be verified (examples of nonexistent technology are the actual values of material properties for the specific operating conditions and the validation of a design method). If this is the case, a need for technology is identified and included in the

Technology Program. When the needed technology has been provided through the Technology Program, the provisionally satisfied requirements will be reexamined.

3. The design requirements are not met, in which case another design selection must be made.

The process is illustrated in Fig. 1-1.

While the process of showing compliance with the design requirements is a part of the design program, the tasks of verifying the assumptions belong to the technology program. The assumptions are thus the bridge between design and technology within the Integrated Approach. By tying the technology development to the design in this clear and direct manner, two purposes are served. First, by limiting the technology program to tasks which are established in direct response to design needs, unnecessary or excessive technology development is avoided. Secondly, by introducing a systematic method for identifying all the missing design data, the risk of defining an incomplete technology program has been greatly reduced.

This plan covers those DDNs which specify a need for data on a metal. Section 4 covers the design process from the design function to the DDNs. Section 5 describes a plan for a technology program to generate the data specified in Section 4. GA Technologies Inc. (GA) is responsible for developing Section 4 covering the hot duct (i.e., Section 4.1). CE (Combustion Engineering) is responsible for developing Section 4 covering the steam generator subsystem (i.e., Section 4.2) and the reactor vessel (i.e., Section 4.3). The development of the Base Technology Development Program (i.e., Section 5) is the responsibility of ORNL (Oak Ridge National Laboratory).

Fig. 1-1. Relationship of design data needs to design process

2. SCOPE

This plan covers the metals material development and characterization needed for components located in key systems and subsystems in the nuclear island of the MHTGR plant. Specifically, this includes the components listed in Table 2-1 and shown in Fig. 2-1. Table 2-2 lists the DDNs* which resulted in these metals material data needs. These DDNs contain fall back positions available should the data needs not be generated.

*DDNs=Design Data Needs. A copy of DDNs is given in the following report: "Design Data Needs Modular High Temperature Gas-Cooled Reactor," HTGR-86-025, PC-000216, Rev. 3, March 1987.

TABLE 2-1
SUMMARY OF METAL COMPONENTS COVERED BY MTD
MODULAR HTGR PLANT

System/Subsystem	Component	Metal
Reactor System		
o Reactor Core Subsystem	None(a)	
o Neutron Control Subsystem	None	
o Reactor Internals Subsystem	o Hot Duct	Alloy 800H
Vessel System		
o Vessel and Ducts Subsystem	o Reactor Vessel (RV)	SB-533B
	o RV-to-SGV Cross Duct	SB-533B
o Pressure Relief Subsystem	None	
o Vessel Support Subsystem	None	
o Thermal Insulation Subsystem	None	
Heat Transport System		
o Steam Generator Subsystem	o Cooling Tubes	Alloy 800H, 2-1/4 Cr-1 Mo,
	o Tubesheets	Alloy 800H,
	o Tube Support Plates	2-1/4 Cr-1 Mo
o Main Circulator Subsystem	None	
Reactor Cavity Cooling System		
	None	
Shutdown Cooling System (SCS)		
o SC Heat Exchanger Subsystem	None	
o Shutdown Circulator Subsystem	None	
o SC Heat Removal Control Subsystem	None	
o SC Water Subsystem	None	
Plant Control Data and Instrumentation System		
	None	
Plant Protection and Instrumentation System		
	None	
Fuel Handling, Storage and Shipping System		
	None	

(a) None means subsystem contains no metal components which need to be included in this plan (i.e., there is no metal component with a materials related DDN).

TABLE 2-2
DESIGN DATA NEEDS (DDNs)

This table contains the DDNs as they relate to metallic materials.

<u>DDN Number</u>	<u>Title</u>	<u>Subsystem or Component</u>	<u>Priority^(a)</u>			
			<u>U</u>	<u>CB</u>	<u>Un</u>	<u>I</u>
M.10.17.26	Determine Effects of Irradiation on Properties of Alloy 800H	Hot Duct	1	M	M	M
M.10.17.28	Determine Effects of Primary Coolant Chemistry and Temperature on Alloy 800H	Hot Duct	1	M	M	M
M.21.02.02	Determine Properties of 2-1/4 Cr-1 Mo Base Metal and Weldments	Steam Generator (SG)	1	H	H	H
M.21.02.03	Determine Properties of Alloy 800H Base Metal and Weldments	SG	1	H	H	H
M.11.06.01	Nil-Ductility Transition Temperature Shift for Reactor Vessel Material Irradiated at Low Temperatures	Reactor Vessel	1	H	H	H
M.11.06.02	Determine Properties of SA533B (Mn-1/2 Mo-1/2 No) Base Metal and Weldment at Elevated Temperatures	Vessel and Duct Subsystem	1	H	H	H

(a) Definitions:

U = Urgency is a measure of the schedular date when the new data is required less the length of time to obtain the data (expressed in years). FY-87 is taken as the base year, i.e., urgency=1.

CB = Cost Benefit is a measure of a single plant cost impact relative to the development cost on a scale of high, medium, and low.

H = High: Plant Cost Benefit >10 x Development Cost

M = Medium: Plant Cost Benefit 5 to 10 x Development Cost

L = Low: Plant Cost Benefit <5 x Development Cost

Un = Uncertainty in Existing Data expresses the designer's lack of confidence in the available data on which the conceptual design is based, on a scale of high, medium, and low.

I = Importance of New Data expresses the significance or effect of the new data on the design including consideration of available back-up solutions, on a scale of high, medium, and low.

Fig. 2-1. MHTGR components showing primary coolant flow path during normal operation

3. CURRENT STATUS

Design data needs for metals have been identified in order to qualify selected components as summarized in Table 2-1. The data needed include:

1. The effects of primary coolant chemistry and temperature on Alloy 800H (DDNs M.10.17.28 and M.21.02.03).
2. The effects of irradiation on Alloy 800H (DDN M.10.17.26).
3. The effects of primary coolant chemistry and temperature on 2-1/4 Cr-1 Mo (DDN M.21.02.02).
4. The effect of irradiation on the nil-ductility transition temperature shift of SA533B (DDN M.11.06.01).
5. The strength of SA533B at elevated temperatures (DDN M.11.06.02).

This section discusses the status of the metals development program associated with each of the above five major areas.

3.1 Effects of Primary Coolant Chemistry and Temperature on Alloy 800H

The primary coolant helium contains impurities which can cause corrosion in the form of oxidation, decarburization, and carburization. At design temperatures above 538°C (1000°F) carbon transport in Alloy 800H via long-term interaction with the primary coolant helium has been shown to be the most potentially significant mode of corrosion with respect to influencing bulk mechanical properties such as tensile and creep properties. In addition, surface oxidation along with concurrent carbon transport may significantly affect surface sensitive properties such as fatigue, creep-fatigue, and crack growth.

Extensive data are available on the degree of carburization/decarburization and oxidation of Alloy 800H as a function of temperature, impurity levels, and exposure time in simulated HTGR primary coolant helium.

Data available on the effect of HTGR helium indicate that air is the more detrimental environment for

- o High cycle fatigue at temperatures from 538°C to 871°C.
- o Fatigue crack growth and creep crack growth at 677°C.

Data available on the effects of thermal aging of Alloy 800H for aging times up to 20,000 h indicate that:

- o At 816°C tensile strength not degraded significantly (i.e., by more than 20%).
- o At 816°C ductility reduced to as low as 20%.
- o At 760°C low cycle fatigue strength not degraded.
- o At 760°C high cycle fatigue strength reduced by up to 20% at 10^7 cycles depending on aging time.
- o At 593°C aging temperature the room temperature fracture toughness decreased. However, it is still sufficiently high to suggest that critical crack growth does not control the design.

At the design temperatures below 538°C (1000°F) corrosion and thermal aging are not expected to degrade the structural properties of Alloy 800H.

Data are needed on the effects of corrosion resulting from exposure to primary coolant helium with its design impurities and thermal aging at design temperatures from 538°C to 760°C (1000°F to 1400°F) on selected properties of Alloy 800H. Of interest is the effect of such exposure on subsequent service

capabilities of Alloy 800H components which operate at temperatures above 538°C (1000°F).

3.2 Effects of Irradiation on Properties of Alloy 800H

The irradiation data that exists are for Alloy 800 rather than Alloy 800H, but the response of the two materials should be similar. These irradiation data indicate Alloy 800 retains adequate ductility to an exposure of 10^{20} n/cm² if irradiated at temperatures below 593°C (1100°F).

The data available in the temperature range from 650°C to 760°C for Alloy 800 are at fluences from 1×10^{20} to 7×10^{22} n/cm², which is much higher than required. Nonetheless, the irradiation data show both the tensile and creep ductility are sensitive to titanium content, grain size, and to a lesser extent on carbon content.

The creep rupture strength of irradiated Alloy 800 is lower than that of unirradiated material in short-term tests, but the strengths appear to converge at longer test times of about 10^3 h, particularly at 705°C. The low-cycle fatigue strength of irradiated Alloy 800 is lower than that of unirradiated material at high (several percent) strain ranges, but appears similar at low (few tenths of a percent) strain ranges. No data are available on high-cycle fatigue behavior of Alloy 800.

Because the expected neutron exposure requirements are so low (i.e., $\leq 10^{18}$ n/cm²) by comparison to exposure of irradiation specimens for which data are available, data are needed on the effects of irradiation at design levels at temperatures from 593°C to 760°C (1100°F to 1400°F) on selected properties of Alloy 800H. Of interest is the effect of design irradiation levels on subsequent service capability of Alloy 800H in the hot duct.

3.3 Properties of 2-1/4 Cr-1 Mo Base Metal and Weldments

The primary coolant contains impurities which can cause corrosion in the form of oxidation, decarburization and carburization. At the design temperatures of the 2-1/4 Cr-1 Mo components, carbon transport has been shown to be the most potentially significant mode of corrosion with respect to bulk mechanical properties such as tensile and creep properties. In addition, surface oxidation along with concurrent carbon transport may significantly affect surface sensitive properties such as fatigue, creep-fatigue, and crack growth. Finally long-term thermal exposure may cause design-significant changes in material strength and toughness.

Data on 2-1/4 Cr-1 Mo exposed to dry HTGR helium showed that 2-1/4 Cr-1 Mo is carburized very slowly at 480°C, unchanged in carbon content at 540°C and decarburized at 595°C and 650°C. In wet HTGR-helium data showed that in 7000 h 2-1/4 Cr-1 Mo decarburized slightly at 403°C and 510°C, moderately at 580°C and severely at 650°C.

The 2-1/4 Cr-1 Mo exposed to dry helium for up to 20,500 h at temperatures up to 593°C showed rather small changed in carbon level but a change in microstructure which was felt due to aging at 593°C. In addition, the specimens showed a reduction in strength but not a significant change in ductility.

Available information on the creep-fatigue behavior indicates that from 490°C to 593°C a helium environment improves the fatigue life for all test wave forms in comparison with tests in air except for the tensile hold only tests.

Limited data to date has shown that for this material, the thermal aging while exposed to argon caused significant reduction in the tensile and creep properties.

Some limited test data obtained recently on 2-1/4 Cr-1 Mo weldments suggests that these weldments might be as strong as the base metal.

There are some fracture mechanics data for annealed 2-1/4 Cr-1 Mo steel, but not necessarily at the highest temperatures of interest for HTGR conditions. Fatigue crack growth rate and fracture toughness information, in particular, have been obtained for material aged at 510°C. There are very little creep crack growth rate data available.

Data are needed to determine the effects of exposure to primary coolant (He) with its design impurities and temperature over design life on selected properties of 2-1/4 Cr-1 Mo base metal and its weldments. These data need to be sufficient to quantify to a 95% confidence level.

3.4 Irradiation Effects on the Reactor Vessel Material

The pressure vessel integrity of operating commercial LWR power plants is based on a large data base of material property evaluations, fracture mechanics evaluations, and engineering experience in vessel design and fabrication. The reference temperature (RT_{NDT}) concept was introduced into the ASME Boiler and Pressure Vessel Code (Section III, NB-2300) in 1972. It provided a definitive and conservative method for specifying materials and for vessel operating limits. Criteria for reactor vessel material toughness (including RT_{NDT}) and design of surveillance (monitoring) program for vessel beltline materials were established in the Code of Federal Regulation in 1973 (10CFR50 Appendices G and H). In 1975, Regulatory Guide 1.99 was issued, providing the official NRC position on prediction of radiation effects for reactor vessel materials. Revision 02 of this Regulatory Guide is currently under review by NRC. Current research concentrates on the refinement of radiation damage mechanistic models for use in predicting radiation damage.

The bulk of the data is from and applicable to typical LWR conditions: irradiation temperatures in the range 277°C to 310°C (530°F to 590°F). These data have not identified an effect on the NDTT shift of varying irradiation

temperature within this temperature narrow band. However, limited data are available outside the range to indicate that lowering the irradiation temperature results in a significant increase in damage.

The current design approach is to use manganese-molybdenum, SA533 Grade B, Class 1 steel based on the proven LWR vessel and piping technology. The limited data available for low temperature operation are used to adjust the estimated NDTT shift calculated from Regulatory Guide 1.99, Ref. 2 based on the equivalent fast fluence. This is then used to demonstrate that the reactor vessel can indeed withstand the duty cycle.

Data are needed to characterize the neutron induced NDTT shift in the reactor vessel material at irradiation temperatures 177°C to 216°C (350°F to 420°F) and neutron fluence and spectrum expected for the MHTGR.

3.5 Properties of SA533B at Elevated Temperature

Section III of the ASME Code provides ultimate strength and yield data for SA533B to a temperature of 538°C (1000°F). Section III provides allowable stress intensity for Service Level A, B, C, and D loading up to a temperature of 700°F derived from these data. Since time-dependent effects (creep) must be considered as temperature increases, allowable stress intensities are not derived for the elevated temperature regime [$>371^{\circ}\text{C}$ (700°F)]. Therefore, the derivation and use of allowables for Service Level C and D loadings is not permitted in this elevated temperature range.

Limited creep strain data are available for this material for temperatures up to 649°C (1200°F) and for exposures of approximately 48 h. A constitutive equation has been fitted to the data. Elastic-plastic properties are available for temperatures up to 649°C (1200°F) and for strains up to 5%. Analytical relationships were developed for the stress versus plastic strain.

The evaluation of the reactor vessel indicates that the RV temperature exceeds the 371°C (700°F) temperature for which ASME Code Section III

allowable stress intensities are available only for limited Service Levels C and D events. Estimates of creep strains indicate that they are indeed negligibly small. Consequently, the methods and procedures for deriving allowable stress intensities for Service Level C and D conditions are considered valid and employed in judging the reactor vessel structural performance.

Data are needed to support a Section III Code inquiry to establish allowable stress intensities for SA533 Grade B, Class 1 and its weldment for special event (Service Levels C and D) loadings at elevated temperatures.

4. REQUIREMENTS

The contents of this section are as outlined below.

4.X Component Number X

This section applies to a specific component referenced in the nuclear island of the Modular HTGR Plant. Only components with identified design data needs for metallic materials properties are included.

4.X.1 Design Description

This section describes design features of the component. This description includes the materials specified for the component.

4.X.2 Relevant Functions

This section identifies the relevant functions (i.e., what is to be achieved) by the component described in Section 4.X.1. The relevant functions are those which result in technology requirements within the scope of this plan. This implies that the list of functions needs to be sufficient to cover all the requirements given in Section 4.X.3. This section also ties each function to a function number from the functional analysis tree and shows how component functions tie to higher level functions.

4.X.3 Design Requirements/Assumptions

This section identifies design requirements which must be satisfied to successfully perform each relevant function in Section 4.X.2 and which also result in technology data needs. Requirements are derived by specifying the bounding material quantities describing the function to be achieved. All requirements are quantified based on the latest available information and are assigned a requirement number (R No.) so they can be easily referenced in the text.

Next, this section shows how these requirements lead to reduced stresses (σ) and strains (ϵ). This is done by discussing the results of trade studies and design analyses. This discussion supports the technology requirements by comparing ϵ and σ with associated allowables and by identifying any assumptions made to develop these allowables.

4.X.4 Technology Requirements

Detail the data needs associated with each of the assumptions given in Section 4.X.3.

4.X.5 Schedule Requirements

This section gives the schedule on which the designer needs the data relative to the design and analysis tasks associated with component X. The overall schedule for the Modular HTGR calls for preliminary design to start in FY-88 with final design to start two years later in FY-90. The final design phase is a 4-yr duration.

4.X.6 References

This section gives the references called out in Section 4.X and its subsections.

4.1 Hot Duct

The hot duct is a component of the Reactor Internals Subsystem. This subsystem belongs to the Reactor System.

4.1.1 Design Description

The hot duct comprises that portion of the coaxial duct which channels core outlet primary coolant through its inside boundary and primary coolant from the main circulator along its outside boundary. The hot duct includes the following subcomponents: the pipe, bellows, thermal barrier and guide rails. A schematic of this assembly is shown in Fig. 4.1-1. The material used for each subcomponent is as specified in Table 4.1-1.

The pipe inside diameter is 1.19 m (47 in.). Its wall thickness is 13 mm (0.5 in.). The pipe thickness was determined by the requirement to sustain the pressure differential and to support itself under the imposed environmental conditions. The factors determining the i.d. of the pipe are the thermal barrier thickness, the requirement to limit the pressure drop in the duct, and the desire to use a commercially available bellows stock size (i.e., diameter and thickness).

The thermal barrier, including 13 mm (0.5 in.) thick coverplates, is approximately 76.2 mm (3 in.) thick. This limits the back face (i.e., pipe) temperature to 260°C (500°F), while the thermal barrier hot face temperature can reach to 760°C (1400°F) at rated power. Attachment fixtures secure the coverplates to the pipe as shown in Fig. 4.1-1b. Fibrous insulation is compressed between the coverplate and the pipe.

Bellows have been selected to seal the aft end of the pipe attached to the duct extension cylinder. A bellows design is thought to be the most reliable way to accommodate the axial thermal compression/expansion movement of the pipe coincident with the vertical movement of the steam generator outer shroud and possible distortion of the core exit plenum skirt. Seismic

movements are also accommodated efficiently with this design concept. The closest stock commercial bellows size that helps to meet the maximum velocity requirement (considering thermal barrier thickness) is 1.04 m (41 in.) diameter hence the selected diameter of the pipe. With the thermal barrier on the inside of the pipe, the free flow diameter is 1.04 m (41 in.) resulting in a rated power nominal velocity of 58 m/s (190 ft/s).

The flange at the hot duct inlet end is bolted to studs attached to a boss on the core outlet plenum skirt. Access to the fasteners is provided by means of three inspection ports located in the outer duct. Inspection of the attachments, bellows, hot duct exterior, cross duct interior and any pertinent vessel welds can be achieved through these hatches.

The bellows are welded to the steam generator end of the pipe and the bellows has a flange. The flanged bellows at the hot duct exit is bolted to a mating flange on the duct extension cylinder.

The duct extension cylinder provides directed flow to the steam generator and minimizes the effects of the vertical movement (due to thermal expansion) and vibration of the steam generator shrouds. The hot duct extension is attached to a flange located on the steam generator vessel. The flanges of the shroud and hot duct extension are flat thereby easing machining and sealing difficulties. A metallic O-ring is positioned at the flange interfaces.

Sealing of the hot duct extension and the inner and outer steam generator shrouds is accomplished by single ring seals at four locations.

Guide rails are employed to aid in the orientation, installation and removal of the cylindrical sections of the duct. The rails also serve as secondary support at the bottom of the duct and a vertical motion seismic restraint at the top (see Fig. 4.1-1b). The rails and the triangular cross section supports are designed to minimize flow interference in the cold gas annular region. The rails and supports are fabricated from ASME SB408

(Alloy 800H) with the supports welded to the outer duct. Chromium carbide is used on the surface between the hot duct pipe and the guide rails to limit friction and wear.

4.1.2 Relevant Functions

The design requirements discussed in Section 4.1.3 result from the need to satisfy the function to maintain integrity of hot duct [F1.1.2.2.1.1.1.1*]. How this function ties into in the functional analysis tree is shown in Figs. 4.1-2 and 4.1-3.

The hot duct provides a passage for the primary coolant to be transferred from the reactor core outlet plenum to the steam generator. As such it helps channel main primary coolant flow [F1.1.2.2.1.1.1]. To satisfy this function requires that the subfunction to maintain primary coolant flow path integrity [F1.1.2.2.1.1.1.1] and more specifically to maintain integrity of hot duct [F1.1.2.2.1.1.1.1.1] also be satisfied.

4.1.3 Design Requirements/Assumptions

This section presents a discussion on the requirements resulting in technology needs for the hot duct. These requirements are those which must be satisfied to adequately ensure that the hot duct maintains its structural integrity (i.e., function number 1.1.2.2.1.1.1.1). These requirements include design loads, environmental conditions, and standards for ensuring that the hot duct is performing satisfactorily. Also included are the assumptions which the designers judged to be prudent in order to show compliance with these requirements. Listed below are the requirements:

*This function applies to Goal 1 Maintain Plant Operation. The associated function for Goal 2 Maintain Plant Protection are also applicable. This function read "protect the capability to maintain integrity of hot duct."

<u>Reference</u>	<u>Requirement</u>
1	R1 Design as a permanent installation free of maintenance during the life of the plant.
1	R1.1 The operational life shall be 40 years.
1	R1.2 Availability allocation $\geq$ 7008 effective full power h/yr average operating time.
2	R11 Accommodate temperature, pressure and flow of the primary coolant.
	R11.1 Accommodate temperatures given in Table 4.1-2.
3, 4	R11.2 Accommodate a radial pressure differential (from the outer duct to inner duct) of: <ul style="list-style-type: none"> - 9.2 psi at 100% power - 3.4 psi at 50% steam flow - 1.1 psi at 25% steam flow - 0.09 psi at refueling
2, 4	R11.4 Accommodate a maximum primary coolant flow rate: <ul style="list-style-type: none"> kg/h (lb/h) - 566,100 (1,248,100) at rated 100% power - 330,400 (728,400) at 50% steam flow - 182,500 (402,400) at 25% steam flow - 131,100 (28,900) at refueling
6, 1	R12 Accommodate duty cycle events shown in Tables 4.1-2 and 4.1-3.

<u>Reference</u>	<u>Requirement</u>
1, 9	R12.1 Accommodate OBE loads with a maximum ground acceleration of 0.15 g horizontal and 0.15 g vertical (Event 22 in Table 4.1-3).
1	R12.2 Accommodate SSE loads with a maximum ground acceleration of 0.30 g horizontal and 0.30 g vertical (Event 34 in Table 4.1-3).
5	R12.3 Withstand a maximum pressure reduction rate of [0.075 MPa/s (11 psi/s)] during a rapid depressurization event (i.e., events 40 or 41 in Table 4.1-3).
7	R13 Compatible with primary coolant.
8	R13.1 The primary coolant is helium with chemical impurities as specified in Table 4.1-4.
	R16 Accommodate design acoustic loads.
10	R16.1 At 100% power, the maximum sound pressure level (SPL) in the annulus between hot duct and cross duct is <160 dB, while inside the hot duct, the SPL is <140 dB.
	R17 Accommodate design irradiation levels.

<u>Reference</u>	<u>Requirement</u>										
11	<p>R17.1 The design irradiation neutron fluences at the entrance to the hot duct (ϕ_0) are:</p> <p>Neutron energy level Fluence n/cm² x 10¹⁷.</p> <table> <tr> <td>E > 0.9 MeV</td><td>0.05</td></tr> <tr> <td>0.1 MeV < E < 0.9 MeV</td><td>0.07</td></tr> <tr> <td>3.0 eV < E < 0.1 MeV</td><td>0.6</td></tr> <tr> <td>E < 3.05 eV</td><td>2.0</td></tr> <tr> <td>Total</td><td>2.72</td></tr> </table>	E > 0.9 MeV	0.05	0.1 MeV < E < 0.9 MeV	0.07	3.0 eV < E < 0.1 MeV	0.6	E < 3.05 eV	2.0	Total	2.72
E > 0.9 MeV	0.05										
0.1 MeV < E < 0.9 MeV	0.07										
3.0 eV < E < 0.1 MeV	0.6										
E < 3.05 eV	2.0										
Total	2.72										
11	<p>17.2 The magnitude of the neutron fluence along the hot duct is:</p> $\phi = \frac{\phi_0}{8} \left(\frac{D}{L} \right)^2$ <p>where ϕ_0 is fluence at entrance (see R17.1) D is diameter of hot duct L is length</p>										
12	<p>R20 Differential thermal expansion between hot duct and outer duct shall be accommodated in the hot duct</p>										

Trade studies have been performed to ensure that the hot duct satisfies the above requirements. These studies show that the hot duct must withstand an operational life of at least 280,000 h (i.e., 40 yr times 7008 h/yr per R1). During this time it must accommodate temperatures from 32°C to 760°C (R11.1) while exposed to primary coolant (R13.1) and irradiation (R17.1).

Under these conditions the hot duct is stressed by thermal and mechanical induced loads as summarized in Table 4.1-5.

The hot duct thermal barrier, bellows, and elbow see primary coolant at core outlet temperatures. Therefore, these components see temperatures ranging from approximately 316°C to 760°C (R11.1). The other hot duct components are exposed to primary coolant at the cooler core inlet temperatures. This section is directed at those components exposed to the primary coolant at core outlet temperatures under the loading conditions summarized in Table 4.1-5.

Thermal stresses are induced by temperature gradients in the hot duct. These gradients are the largest in the attachment fixtures (see Fig. 4.1-1b) which connect the hot duct coverplates to the pipe. It is conservative to assume that during power operation these attachment fixtures see a temperature change (ΔT) equal to the difference between the temperature of the primary coolant exiting and entering the core. This is also realistic because of the high primary coolant flow rate at 25% steam flow to 100% power (R11.4). However, during refueling conditions ΔT will be significantly less due to the low flow rate (and the resulting low convective heat transfer coefficient). A rough estimate is that during refueling ΔT across the attachment fixture is one-half the difference between the temperatures of the primary coolant exiting and entering the core. Table 4.1-6 summarizes the resulting ΔT and $\Delta T/\Delta X$ across the attachment fixture for conditions from shutdown to 100% power.

Table 4.1-6 also gives the number of cycles (n) associated with changes in power (R12) and the code allowable LCF $\Delta \epsilon$ associated with n (per Ref. 14).

Based on the approximation that the induced strain range ($\Delta\epsilon$) is proportional to the induced $(\Delta T/\Delta x)^*$ together with the thermal stress analyses results given in Ref. 15, it follows that

$$\sigma_{t \max} = 46 \frac{\Delta T}{\Delta x} \text{ psi} , \quad 4.1-1$$

where $\sigma_{t \max}$ is the maximum induced thermal stress, psi ,

$\frac{\Delta T}{\Delta x}$ is the induced temperature gradient, °F/in .

Equation 4.1-1 is based on an attachment fixture with an outer diameter (D) of 2 cm (5 in.)* and a wall thickness (t) of 0.51 cm (0.20 in.)*. Using the linearly elastic relationship $\epsilon = \sigma/E$,

$$\Delta\epsilon_t = \frac{46}{E} \frac{\Delta T}{\Delta x} . \quad 4.1-2$$

The values of σ_t and $\Delta\epsilon$ are given in Table 4.1-6 based on $E = 1.7 \times 10^{11}$ Pa (25×10^6 psi) max (this was used to generate results in Eq. 4.1-1). A comparison of the induced $\Delta\epsilon$ to the allowable $\Delta\epsilon$ shows that the ~500 cycles associated with shutdown - 100% - shutdown (rounded up from 455 given in Table 4.1-3) is most controlling with the induced $\Delta\epsilon$ being 91% of the code (Ref. 14) allowable $\Delta\epsilon$. These 500 cycles have an average hold time at the elevated temperature condition of up to 760°C (1400°F) of approximately 600 h (i.e., 280,000 h divided by 500 cycles).

*Theory would indicate σ_t is proportional to

$$E\alpha \frac{\Delta T}{\Delta x} \sqrt{Dt}$$

for a cylinder with an axial temperature gradient.

It has also been shown (Ref. 15) that the thermal stress in the attachment fixtures resulting from temperature transients (R11.1) do not result in higher thermal stress than 100% steady state conditions.

During the conceptual design phase, the induced thermal strains are limited to 0.04% (which is approximately 80% of the code Ref. 14 allowable LCF $\Delta\epsilon$) to account for uncertainties in the effect of environmental conditions (i.e., primary coolant impurities and irradiation) on this property. It is also necessary to assume that exposure to design irradiation (R17), primary coolant (R13), and elevated temperatures (R11.1) over the design life (R1) will not significantly degrade da/dn , K_{IC} , E , S_R , S_1 , and da/dt of Alloy 800H.

Changes in the primary coolant temperature induce axial compression movements of approximately 2.2 in. and lateral shear movements of approximately 0.6 in. on the bellows. The majority of these movements is controlled by changes in gas at core inlet temperature conditions. It follows that the movement resulting from power changes between 25% and 100% power are small since, as summarized in Table 4.1-2, the temperature of the core inlet gas does not change significantly over this range. Therefore, the cycles associated with going from shutdown to 100% power are most controlling.

Ultimately the bellows will need to be tested to the design movements for the design number of cycles. However, this testing will be done on as-received material. It has been assumed that exposure to the primary coolant chemistry (R13) and temperature (R11.1) over the bellows design life (R1) will not significantly degrade the properties needed to structurally evaluate the bellows. These properties include the allowable LCF $\Delta\epsilon$, da/dN , K_{IC} , E , S_R , S_1 , da/dt of Alloy 800H.

The fracture mechanics properties da/dN , da/dt , and K_{IC} are necessary to establish the allowable initial flow size (a) that can exist in the hot duct

bellows. The following is a brief discussion on what values are considered acceptable.

$$\Delta K = K_1 - K_2$$

where K_i is stress intensity factor at condition i

$$K_i = \sigma \sqrt{\pi a} \quad 4.1-3$$

σ = induced stress

a = crack length or flaw size

It follows that for a given crack length

$$\Delta K = \Delta \sigma \sqrt{\pi a}$$

based on an elastic strain range ($\Delta \epsilon$) of 0.00063 (from Table 4.1-6) it follows that $\Delta \sigma \approx E \Delta \epsilon$. Over the temperature range of interest [i.e., 204°C to 760°C (400°F to 1400°F)] $E \leq 1.85 \times 10^{11}$ Pa (26.8×10^6 psi). It follows that $\Delta \sigma \leq 117$ MPa (17 ksi). Therefore, based on $a = 2.54$ cm (1 in). (which is conservatively large since a flaw this large would be easy to detect) $\Delta K \leq 33$ MPa $\sqrt{\text{m}}$ (30 ksi $\sqrt{\text{in.}}$). Note if under these conditions $da/dN < 2.5 \times 10^{-3}$ mm/cycle (1×10^{-4} in./cycle), then in 455 cycles a would increase only 1.17 mm (0.046 in.) which is sufficiently small to ensure that K_i never exceeds K_{IC} based on the assumption that $K_{IC} > 55$ MPa $\sqrt{\text{m}}$ (50 ksi $\sqrt{\text{in.}}$). Also, if $da/dt < 1.3 \times 10^{-9}$ mm/s (5×10^{-11} in./s) (1.8×10^{-7} in./h) then in 1×10^9 s (300,000 h), a would increase only 1.27 mm (0.05 in.) which is sufficiently small to ensure that K_i never exceeds 55 MPa $\sqrt{\text{m}}$ (50 ksi $\sqrt{\text{in.}}$).

Mechanically induced stresses are induced by sources 2 through 6 indicated in Table 4.1-5. Source 2 is the radial pressure drop across the hot duct pipe and bellows. A simple stress analysis can show that the induced stress by this loading is relatively low (less than 20% of the design allowable).

During a seismic event (R14 and R15) loads are induced into the hot duct elbow and bellows. The loading on the elbow is a result of the seismic induced accelerations which induce bending stresses at locations where the bellows are attached to the steam generator shrouds (see Fig. 4.1-1). The loading on the bellows is due to the relative movement of the hot duct pipe and elbow on both sides of the bellows. The axial movements are expected to be less than 6.35 mm (0.25 in.) which is small relative to thermal induced movements. The lateral shear movement while also less than 6.35 mm (0.25 in.) is relatively more significant.

This seismic loading will induce stresses and strains that need to be compared against allowables for low cycle fatigue, da/dN , and K_{IC} . Ultimately the bellows will need to be tested to the design movements for the design number of cycles. It is being assumed that the exposure to the primary coolant chemistry (R13) and temperatures (R11.1) over the bellows design life of 280,000 h (R1) will not significantly degrade the allowable low cycle fatigue strain range, K_{IC} , and da/dN of Alloy 800H.

The coverplates compress the fibrous insulation against the hot duct pipe (Source 5 in Table 4.1-5). This loading produces a uniform pressure load (p) of approximately 6895 Pa (1 psi) of the surface of each coverplate. This induced stresses (σ) in the coverplate and attachment fixture which are proportional to the dimensions of the coverplate and attachment fixture:

For a given D/L , R

$$\sigma \propto p \left(\frac{L}{h} \right)^2 \quad 4.1-4$$

where D is o.d. of attachment fixture
 L is length of coverplate
 h is thickness of coverplate
 R is radius of curvature

Based on the dimensions given in Fig. 4.1-1: $D/L = 0.2$, $R = 0.60$ m (23.5 in.), $L = 0.62$ m (24.6 in.), and $h = 12.7$ mm (0.5 in.), the maximum stress (σ_M) is (based on relationships given in Refs. 16 and 15):

$$\sigma_M = 1100 \text{ p} \quad 4.1-5$$

Therefore, for $p = 6895$ Pa (1 psi), $\sigma_M = 7.58$ MPa (1100 psi)

Per ASME Code, for a design to be acceptable under this loading

$$\sigma_M \leq K_t S_t$$

where

$$K_t = 1.5$$

S_t = allowable time-dependent design stress intensity

For as-received Alloy 800H at 760°C (1400°F) for 300,000 h, $S_t = 17.9$ MPa (2.6 ksi) (per Ref. 14).

Based on the above σ_M is approximately one-third the allowable at 760°C (1400°F). However, to allow for additions design modification, it is being assumed that the design would be okay if σ_M increased to 21 MPa (3 ksi) (~77%) of the allowable stress at 760°C (1400°F).

The stress intensity (S_t) is (per Ref. 17) the least of each of the three qualities:

1. Two-thirds minimum $\underline{S_R^*}$.

* Symbols are defined in Nomenclature section.

2. Eighty percent of the minimum stress to cause the onset of tertiary creep in time t .
3. The minimum S_1 .

Based on the above, it follows that at $T \leq 760^\circ\text{C}$ (1400°F) for times up to 300,000 h

- minimum S_R $> 21 \text{ MPa (3 ksi)}$
- minimum S_1 $> 13.8 \text{ MPa (2 ksi)}$

This type of loading also results in the need to know the values of da/dt and K_{IC} of Alloy 800H. This is necessary to help establish the allowable initial flaw size (a_I) in the hot duct coverplates and attachment fixtures.

During a rapid depressurization event (R11.3), gas within the fibrous insulation exerts a pressure load of up to 34 kPa (5 psi) on the surface of the coverplate. This load is of short duration (<1 min) and at the time of high pressure loads the temperature of the coverplates is between 538°C (1000°F) and 871°C (1600°F).* The resulting stresses (σ) in the coverplate and attachment fixture is $<48 \text{ MPa (7 ksi)}$ (based on Eq. 4.1-5). The ASME Code (Ref. 14) requires for this event that

$$\sigma \leq 1.2 K S_m$$

where

K is the section factor for the cross section being considered. $K = 1.5$ for rectangular cross section.

* It has been conservatively assumed that a rapid depressurization event could occur at time of maximum coverplate temperature [i.e., 871°C (1600°F) per footnote to Table 4.1-2b]. During normal operation, the maximum coverplate temperature is between 538°C and 760°C (1000°F and 1400°F).

S_m is the time-independent design stress intensity which is defined (per Ref. 17) as the lowest of the following values:

1. One-third the specified minimum tensile strength (S_u) at room temperature.
2. One-third S_u at operating temperature.
3. Two-thirds the specified minimum yield strength (S_y) at room temperature.
4. $0.9 S_y$ at operating temperature.

For Alloy 800H S_m is typically controlled by one-third S_u at operating temperature for temperatures above 427°C (800°F). For as-received Alloy 800H S_m equals 76 MPa (11 ksi) at 760°C (1400°F) and 37 MPa (5.3 ksi) at 871°C (1600°F). During the conceptual design phase, it is being assumed that the design is acceptable if $\sigma < 48$ MPa (7 ksi) which implies that $S_m \geq 28$ MPa (4 ksi). This leads to the requirement that the minimum S_u shall not be less than 83 MPa (12 ksi) and that the minimum S_y at the operating temperatures between 538°C and 871°C shall not be less than 31 MPa (4.5 ksi).

This type of loading also results in the need to know the critical stress intensity factor (K_{IC}). A crack in a structure is predicted to grow uncontrollably if

$$\sigma \sqrt{\pi a} > K_{IC}$$

where σ is the induced stress
 a is the crack length

For Alloy 800H at 760°C, the allowable service level A bending stress ($1.5 S_m$) equals 114 MPa (16.5 ksi). Assuming a is less than thickness of coverplate [i.e., 12.7 mm (0.5 in.)], it follows that $\sigma \sqrt{\pi a} < 23 \text{ MPa } \sqrt{\text{m}}$

(21 ksi $\sqrt{\text{in.}}$). In the trade studies K_{IC} has been assumed to be at least 55 MPa $\sqrt{\text{m}}$ (50 ksi $\sqrt{\text{in.}}$).

Acoustic loading (R16) can excite the hot duct coverplates, pipe, and bellows. The induced stresses in the pipe itself are small. Therefore, most attention needs to be directed toward the coverplates and bellows. The response of coverplates has been estimated using methods reported in Ref. 16. This estimate shows that Alloy 800H* coverplates with the dimensions shown in Fig. 4.1-1b and Table 4.1-1 respond most at a frequency of approximately 300 Hz. Over the design life of the hot duct (i.e., 280,000 h per R1) this will result in approximately 3×10^{11} cycles.

The acoustically induced stress can also be estimated using relationships given in Ref. 16. These relationships indicate

$$\sigma_M = C_1 \left(\frac{L}{h} \right)^2 \frac{P_p}{\eta}$$

where C_1 is a constant which depends on d/l , r , and mode shape
 P_p = peak acoustic load = $3.5 \times \text{rms sound pressure level } (P_{\text{rms}})$
 η = damping which ≈ 0.04 (per Ref. 16)

Based on $D/L \approx 0.2$, $R = 0.60 \text{ m}$ (23.5 in.), the largest value of C_1 is approximately 0.45. It follows that for $L = 0.62 \text{ m}$ (24.6 in.) and $h = 12.7 \text{ mm}$ (0.5 in.) that

$$\sigma_M = 1100 \frac{P_p}{\eta} \quad 4.1-7$$

This is the same as Equation 4.1-5 with p replaced by P_p/η .

* The plate frequency depends on the values of the elastic modulus (E), Poisson ratio (ν) and mass density (ρ) of the plate material.

The value of $P_{rms} < 200 \text{ Pa} = 0.029 \text{ psi}$ (based on an acoustic load of $< 140 \text{ dB}$ per R16.1). It follows that $\sigma_M < 19.3 \text{ MPa}$ (2800 psi), which is significantly less than the allowable high cycle fatigue strength of 41 MPa (6000 psi) assumed for Alloy 800H.

It is expected that the bellows will have its maximum response at a lower frequency than the coverplate. However, its induced stresses could be larger since at the location of the bellows the acoustic load on the outside of the bellows is approximately 160 dB (per R16.1) which equals 2 kPa (0.29 psi). While the bellows is still being sized, it is assumed that the design of the hot duct will require an allowable high cycle fatigue stress as high as 41 MPa (6 ksi) for 3×10^{11} cycles.

The trade studies discussed above rely on the following assumptions which need to be verified before the hot duct design can be qualified.

Reference Supporting	<u>Assumption of Trade Studies Assumption</u>
18 thru 28	1. Exposure to primary coolant chemistry and temperature over design life will not significantly degrade the design properties of metal used for hot duct.
29 and 30	2. In-service irradiation will not significantly degrade the design properties of metal used in hot duct.

The following subsection discusses why it is necessary to make each assumption and how each assumption affects the hot duct. Also, a discussion is given of the current data base.

4.1.3.1 Effects of Primary Coolant Chemistry and Temperature on Alloy 800H.

The primary coolant helium contains impurities [per Table 4.1-4] which can cause corrosion (Ref. 18) in the form of oxidation, decarburization and carburization. At design temperatures above 538°C (1000°F) (see R11.1) carbon transport in Alloy 800H via long-term (per R1) interaction with the primary coolant helium has been shown to be the most potentially significant mode of corrosion (Ref. 18) with respect to influencing bulk mechanical properties such as tensile and creep properties. In addition, surface oxidation along with concurrent carbon transport may significantly affect surface sensitive properties such as fatigue, creep-fatigue, and crack growth.

Extensive data are available on the degree of carburization/decarburization and oxidation of Alloy 800H (Ref. 18) as a function of temperature, impurity levels, and exposure time in simulated HTGR primary coolant helium. Specifically, the data reported in Ref. 18b summarizes the effect of exposures at 600°C to 950°C (1112°F to 1742°F) for up to 10,000 h in dry helium ($< 0.5 \mu\text{atm H}_2\text{O}$, $< 0.5 \mu\text{atm CO}_2$, $50 \mu\text{atm CO}$, $500 \mu\text{atm H}_2$ and $50 \mu\text{atm CH}_4$) and up to 15,000 h in wet helium ($10 \mu\text{atm H}_2\text{O}$, $10 \mu\text{atm CO}_2$, $50 \mu\text{atm CO}$, $150 \mu\text{atm H}_2$ and $10 \mu\text{atm CH}_4$). As a result of these exposures Alloy 800H experienced carburization (i.e., increases in bulk carbon content). This carburization was considered slight ($\leq 0.03 \text{ wt. \% C}$) in wet helium at all test temperatures and in dry helium at temperature up to 800°C (1472°F). Carburization increases with increasing exposure time and with increasing exposure temperature for a constant exposure time. Reference 18d reports that CO appears to be the dominant carburizing species at temperature $\leq 900^\circ\text{C}$ ($\leq 1652^\circ\text{F}$) while increases in H_2O concentration decrease carburization rate. Techniques have been developed (Ref. 18d) to predict the degree of corrosion as a function of impurity level temperature and time. Based on these techniques and the available data, it is concluded at the expected (and design) impurity levels shown in Table 4.1-4 carburization of Alloy 800H should be minimal at design temperatures of 760°C (1400°F) or less over the life of the plant (280,000 h). However, there is a potential for significant decarburization and surface oxidation in the environments indicated for expected and design conditions. In

addition, should the primary coolant become very dry (low moisture levels) during nominal plant operation or by virtue of moisture depletion in regions of components not fully swept by the coolant (stagnation regions such as crevices, for example), the potential for significant carburization will increase.

Data are available on how the corrosion phenomena affect the structural properties of Alloy 800 and Alloy 800H. Creep rupture testing has been done at GA in "wet" helium in the temperature range from 649°C to 815°C (1200°F to 1500°F). These results reported in Ref. 19 have rupture times from 1,000 to 25,000 h and fall well within the air data scatter bands. Similar results are reported in Ref. 18d. These data represent tests of fairly short duration when compared to 100,000 to 300,000 h design life.

Fatigue results at 850°C (1562°F) are reported in Ref. 20 on as-received and post-exposed (after 6000 h in reactor helium) Alloy 800H in both air and reactor helium. The results shown in Fig. 4.1-4 indicate no significant effect on the fatigue results of the 6000 h exposure. The results of high cycle fatigue testing at temperatures from 538°C to 871°C (1000°F to 1600°F) are reported in Ref. 21 on Alloy 800H tested in air and in very wet helium (2500 μ atm H₂O). The results shown in Fig. 4.1-5 show that air is the most detrimental environment except at 649°C (1200°F). The results shown in Figs. 4.1-6 and 4.1-7 show that aging reduces the fatigue strength by up to 20% at 10⁷ cycles depending on aging time. The most detrimental strength loss occurred after aging 1500 h.

Reference 18d concludes that high- and low-cycle fatigue life at temperatures between 650°C and 950°C (1202°F and 1742°F) is generally unchanged or enhanced in "wet" and "dry" helium environments compared to tests in air.

Some fracture mechanics data are available (Ref. 22) on a solution annealed and cold worked Alloy 800H at 677°C (1250°F) relative to fatigue crack growth (da/dN) and creep crack growth (da/dt).

Based on these data, it is concluded that:

1. da/dN of cold worked is higher than for as-received for a given ΔK at 677°C (1250°F).
2. da/dN is less dependent on material condition at 732°C (1350°F) than at 677°C (1250°F).
3. da/dN and da/dt both increases with increasing temperature.
4. da/dN is lower in HTGR helium than in air. Also oxidation lowers the fatigue life for a given applied strain range.
5. $da/dN = C (\Delta K)^M$

da/dN = fatigue crack growth rate, in/cycle.

ΔK = range of stress intensity factor, $\text{ksi}\cdot\text{in.}^{1/2}$

C and M are empirical constants. Based on tests in wet helium, ($10 \mu\text{atm H}_2\text{O}$) for 20% cold worked Alloy 800H at 732°C (1350°F), $C = 3.48 \times 10^{-11}$, $M = 3.95$. It follows, that for $\Delta K = 30 \text{ ksi}\cdot\text{in.}^{1/2}$, $da/dN = 2.4 \times 10^{-5} \text{ in./cycle}$.

6. $da/dt = 4.3 \times 10^{-4} C_t^{0.916}$

where

da/dt = creep crack growth in in./h

C_t = an energy rate parameter, in./psi·h

Based on data for solution-annealed Alloy 800H at 732°C (1350°) for a crack

length $a = 0.1 \text{ in.}$ $C_t \sim 1.9 \times 10^{-36} \sigma^9$,

Based on these data, it is concluded that:

1. da/dN of cold worked is higher than for as-received for a given ΔK at 677°C (1250°F).
2. da/dN is less dependent on material condition at 732°C (1350°F) than at 677°C (1250°F).
3. da/dN and da/dt both increases with increasing temperature.
4. da/dN is lower in HTGR helium than in air. Also oxidation lowers the fatigue life for a given applied strain range.
5. $da/dN = C (\Delta K)^M$

da/dN = fatigue crack growth rate, in/cycle.

ΔK = range of stress intensity factor, $\text{ksi}\cdot\text{in.}^{1/2}$

C and M are empirical constants. Based on tests in wet helium, ($10 \mu\text{atm H}_2\text{O}$) for 20% cold worked Alloy 800H at 732°C (1350°F), $C = 3.48 \times 10^{-11}$, $M = 3.95$. It follows, that for $\Delta K = 30 \text{ ksi}\cdot\text{in.}^{1/2}$, $da/dN = 2.4 \times 10^{-5} \text{ in./cycle}$.

6. $da/dt = 4.3 \times 10^{-4} C_t^{0.916}$

where

da/dt = creep crack growth in in./h

C_t = an energy rate parameter, in./psi·h

Based on data for solution-annealed Alloy 800H at 732°C (1350°) for a crack

length $a = 0.1 \text{ in.}$ $C_t \sim 1.9 \times 10^{-36} \sigma^9$,

with σ = creep stress in psi ,

based on $\sigma = 3 \text{ ksi} \rightarrow C_t = 3.8 \times 10^{-5}$,

and $da/dt = 3.8 \times 10^{-8} \text{ in./h.}$

Effects of Aging

Data are available on the effects of thermal aging (long-term exposure at elevated temperatures) of Alloy 800H in Refs. 23 and 24. These data are for aging times up to 20,000 h for multiple heats at temperature to 816°C (1500°F). The data in Ref. 23 indicates that thermal aging did not degrade the tensile strength significantly (i.e., more than 20%) as shown in Fig. 4.1-8. Thermal aging did reduce the ductility to values as low as 20% as shown in Fig. 4.1-9. The data in Refs. 24 indicates that low cycle fatigue strength is not degraded by aging (to 8000 h at 760°C) as shown in Fig. 4.1-10.

Figure 4.1-11 shows the effect of aging temperature on room-temperature fracture toughness J_{IC} for aging times of 5,000 and 10,000 h in air (Ref. 25). Both sets of data indicate a tendency toward room temperature embrittlement (toughness decrease) at an aging temperature of 593°C (1100°F). Up to an aging time of 10,000 h, it is apparent that the embrittlement rate is significantly greater at this temperature than at other temperatures where there appears to be no significant increase in embrittlement after 5000 h. For comparison with the results of Fig. 4.1-11, it is noted that tests at room temperature on solution-annealed Alloy 800H conducted as part of a series of round-robin tests yielded an approximate estimate of 1100 kJ/mm² (6300 in.-lb/in.²) for J_{IC} (Ref. 26). This suggests an early toughness decrease on thermal aging. The severity of the embrittling effect of a 593°C (1100°F) exposure will become more apparent from test results on material aged for longer times at this temperature.

The critical stress intensity factor K_{IC} may be calculated based on J_{IC} (the crack initiation resistance) from $K_{IC}^2 = J_{IC} E (1-\nu^2)$ where E = elastic

modulus and ν = Poisson's ratio. Based on the relationship, the lowest measured J_{IC} (i.e., 163 kJ/m²) yields a K_{IC} of approximately 192 MPa $\sqrt{m}$ (175 ksi $\sqrt{in.}$). This value is sufficiently high to suggest that critical crack growth does not control the design.

Weldments at elevated temperatures have been an area of concern because of the potential for limited ductility of weld metal and the potential for high strain concentrations (both metallurgical and geometric in the heat affected zone of weldments. Reference 27 states that the stress-to-rupture properties of the weld materials were not as good as the Alloy 800 base metal in the as-welded condition in the temperature range of 538°C to 760°C whether exposures were short or long. (See Fig. 4.1-12.) This is contrary to the results obtained at GA (Ref. 28) and may be due to variation in the heat of filler materials and parent materials. The limited data obtained at GA (Ref. 28) suggest that (1) weldments of Inconel 82 in Alloy 800H show tensile and creep rupture strengths that are approximately equal to that of the parent metal in either the as-welded or aged (300 h at 649°C) condition; (2) aging at temperatures between 593°C and 704°C for 3000 h tends to reduce the ultimate tensile strength (S_u) at 649°C and 760°C as the aging temperature increases. (See Table 4.1-7.) There was no significant effect of aging on S_u at 815°C and 900°C.

At the design temperatures below 538°C (1000°F) corrosion and thermal aging are not expected to degrade the structural properties of Alloy 800H.

Data are needed on the effects of corrosion resulting from exposure to primary coolant helium with its design impurities and thermal aging at design temperatures [from 538°C to 760°C (1000°F to 1400°F)] on selected properties of Alloy 800H. Of interest is the effect of such exposure on subsequent service capabilities of Alloy 800H component which operate at temperature above 538°C (1000°F). These data need to be sufficient to quantify to a 95% confidence that these properties meet or exceed design values.

4.1.3.2 Effects of Irradiation on Properties of Alloy 800H. The irradiation data that exists is for Alloy 800 rather than Alloy 800H, but the response of the two materials should be similar (Ref. 30) because irradiation data for Alloy 800 with carbon contents up to 0.06% indicates that irradiation response is not a strong function of carbon content. These irradiation data indicate Alloy 800 retains adequate ductility to an exposure of 10^{20} n/cm² if irradiated at temperatures below 593°C (1100°F) (Refs. 29 and 30).

The data available (Ref. 30) in the temperature range from 650°C to 760°C (1200°F to 1400°F) for Alloy 800 are at fluences from 1×10^{20} to 7×10^{22} n/cm², which is much higher than required (R17.1). Nonetheless, the irradiation data show both the tensile and creep ductility are sensitive to titanium content, grain size, and to a lesser extent on carbon content (Fig. 4.1-13).

The creep rupture strength of irradiated Alloy 800 is lower than that of unirradiated material in short term tests, but the strengths appear to converge at longer test times of about 10^3 h, particularly at 705°C (1301°F) (Figs. 4.1-14 and 4.1-15). The low-cycle fatigue strength of irradiated Alloy 800 is lower than that of unirradiated material at high (several percent) strain ranges, but appears similar at low (few tenths of a percent) strain ranges as shown in Fig. 4.1-16. No data is available on high-cycle fatigue behavior for Alloy 800.

Because the expected neutron exposure requirements (R17.1) are so low by comparison to exposure of irradiation specimens for which data are available, data are needed on the effects of irradiation at design levels (R17.1) at temperatures from 593°C to 760°C (1100°F to 1400°F) on selected properties of Alloy 800H. Of interest is the effect of design irradiation levels (R17) on subsequent service capability of Alloy 800H in the hot duct. These data need

to be sufficient to quantify to a 95% confidence* that these properties meet or exceed design values.

4.1.4 Technology Requirements

This section gives details of the data needs associated with each assumption discussed in the previous section. Quality Assurance must satisfy the requirements of 10CFR50 Appendix B.

4.1.4.1 Determine Effects of Primary Coolant Chemistry and Temperature on Alloy 800H (DDN M.10.17.28). Sufficient data are required to determine to a 95% confidence the effects of corrosion resulting from exposure to primary coolant with its design impurities (per Table 4.1-4) and thermal aging at temperatures from 1000°F to 1400°F for up to 300,000 h on the following properties of solution annealed Alloy 800H and its weldments as specified for use in hot duct thermal barrier, bellows and elbow in Table 4.1-1.

- a. Tensile properties in helium at temperatures from 538°C to 871°C (1000°F to 1600°F) with design values for:
 - o Minimum S_u of not less than 83 MPa (12 ksi).
 - o Average elastic modulus (E) within $\pm 20\%$ of average E for as received value.
 - o Minimum ultimate total elongation of at least 10%.
 - o Yield strength (S_y) of not less than 31 MPa (4.5 ksi).

*This reference to a 95% confidence is not meant to specify that the data must be subjected to a pure statistical treatment to give the designer and technologist the indicated confidence. Rather it is intended that the technologist relies on techniques previously used to determine design properties given in the ASME B&PV Code. These techniques are based on engineering principles together with statistics.

- b. Low cycle fatigue strain range ($\Delta\epsilon$) to 500 cycles in MHTGR-helium at temperatures from 204°C to 760°C (400°F to 1400°F) with a design value of at least 0.0004 based on hold times up to 600 h.
- c. Fracture toughness properties in MHTGR-helium at temperatures from 204°C to 760°C (400°F to 1400°F) with design values for:
 - o Critical stress intensity factor (K_{IC}) of greater than 55 MPa $\sqrt{m}$ (50 ksi $\sqrt{in.}$).
 - o Fatigue crack growth rate (da/dN) of less than $[2.5 \times 10^{-3} \text{ mm/cycle} (1 \times 10^{-4} \text{ in./cycle})]$ for $\Delta K \leq 33 \text{ MPa } \sqrt{m}$ (30 ksi $\sqrt{in.}$).
 - o Creep crack growth rate (da/dt) of less than $[1.3 \times 10^{-9} \text{ mm/s} (5 \times 10^{-11} \text{ in./s})]$ for creep stress $\leq 21 \text{ MPa}$ (3 ksi).
- d. Creep properties in MHTGR-helium at temperatures from 1200°F to 1400°F for duration up to 300,000 h with design values for:
 - o Minimum S_R of not less than 21 MPa (3 ksi).
 - o Minimum S_1 of not less than 14 MPa (2 ksi).
- e. High cycle fatigue strength (S_a) to 3×10^{11} cycles in MHTGR-helium at temperatures from 593°C to 760°C (1100°F to 1400°F) with a design value of at least $[41 \text{ MPa} (6.0 \text{ ksi})]$.

4.1.4.2 Determine Effects of Irradiation on Properties of Alloy 800H

(DDN M.10.17.26). Sufficient data are required to determine to a 95% confidence the effects of irradiation in helium at temperatures from 593°C to 760°C (1100°F to 1400°F) and at fluence levels from 10^{17} n/cm^2 thermal ($E \leq 2.4 \text{ eV}$) and 10^{17} n/cm^2 epithermal to 10^{18} n/cm^2 thermal and 10^{18} n/cm^2 epithermal on the following properties of solutional annealed Alloy 800H and its weldments as specified for use in hot duct thermal barrier components in Table 4.1-1.

- a. Tensile properties in helium at temperatures from 204°C to 871°C (400°F to 1600°F) with design values as indicated in Section 4.1.4.1.
- b. Low cycle fatigue strain range to 500 cycles in MHTGR-helium at temperatures from 204°C to 871°C (400°F to 1400°F) with a design value of at least 0.0004 based on hold times up to 600 h.
- c. Fracture toughness properties in MHTGR-helium at temperatures from 204°C to 871°C (400°F to 1400°F) with design values as indicated in Section 4.1.4.1.
- d. Creep rupture in MHTGR-helium at temperatures from 649°C to 760°C (1200°F to 1400°F) for duration up to 300,000 h with design values as indicated in Section 4.1.4.1.
- e. High cycle fatigue strength to 3×10^{11} cycles in MHTGR-helium at temperatures from 593°C to 760°C (1100°F to 1400°F) with a design value of at least [41 MPa (6.0 ksi)].

4.1.5 Schedule Requirements

All the data defined in Section 4.1.4 are needed at least three years before end of final design. This implies all data are needed by 10/90 based on schedule shown in Fig. 4.1-17.

4.1.6 References

1. GCRA 84-011, Rev. 1, "Utility/User Design Requirements for Small High Temperature Gas-Cooled Reactors," April 1985.
2. "Conceptual Design Data for Modular HTGR Plant," HTGR-86-023, GA Doc. 907807/7, April 1986.
3. Henderson, M., "Core Thermal/Hydraulics Design Requirements for 4 x 350 MW(t) SBSV Reactor Concept," GA Doc. 908431/0, Nov. 25, 1985.

4. Howard, W. W., "NSSS Thermal Performance Requirements for Modular High Temperature Gas-Cooled Reactor,": GA Doc. 908755/1, June 18, 1986.
5. Breher, W. E., "PSID Analysis of Licensing Basis Events for the Standard MHTGR," Doc. 908956/0, August 1986.
6. Lommers, L., "Modular HTGR Temperature Gas-Cooled Reactor Plant Design Duty Cycle," GA Doc. 908728/1, April 30, 1986.
7. Baccaglioni, G., R. Barker, and H. S. Chung, "HTGR SC/C Lead Plant Moisture Ingress Evaluation," GA Doc. 906519/1, September 30, 1983.
- 8a. Baccaglioni, G., "Plant Specification - Primary Coolant Chemistry - 2240 MW(t) HTGR-SC/C Plant," GA Doc. 906385/2, July 30, 1984.
- 8b. Chawla, O. P., "Long-Term Expected and Design Primary Coolant Impurity Levels for the MHTGR: A Lifetime Review" June 12, 1986.
- 9a. Olsen, B. E., and I. T. Almajan, "Soil Parameter Study for Seismic Analysis of MHTGR Plant," Doc. 908767/0, March 18, 1986.
- 9b. Sweeney, T. M., to L. D. Mears, "Transmittal of Seismic Floor Response Spectra, Linearly Varying Soil Site GCRA/HTGR Project: Job No. 14508 BL-927," Bechtel National, Inc. letter GCRA/BEC/81-202, April 29, 1986.
10. Baccaglioni, G., "Heat Transport System Design Report", Doc. 908759/0, HTGR-86-122, November 17, 1986.
11. Bucholz, J. A., and R. L. Childs, "Preconceptual Shielding Design Analyses for the Lower Portion of the MHTGR Reactor System Incorporating a 350 MW(th) Annular Prismatic Core," HTGR-86-060, ORNL/GCR-86/1, June 1986.
12. Mehta, D. D., "Reactor Vessel to Steam Generator Vessel Cross Duct Requirements and Design Options for the HTGR-MRS Side-by-Side Concept," memo CE-ADD-316/14001, October 1984.
13. Chan, T. W., and M. B. Richards, "MHTGR Plant Design Basis Transient Analysis," Doc. No. 908754/0, October 31, 1986.
14. Code Case N47 of ASME Boiler and Pressure Vessel Code.
15. Pickering, J. L., and W. S. Betts, "GT-HTGR Class B2 Integral Cast Coverplate/Attachment Fixture Temperature Limit Study," GA Doc. 904448/1, February 19, 1980.
16. Betts, W. S., "Thermal Barrier Acoustic Analysis," GA Doc. 904167/5, February 11, 1985.

17. "Criteria for Design of Elevated Temperature Class 1 Components in Section III, Division 1 of the ASME Boiler and Pressure Vessel Code," published by ASME in May 1976.
- 18a. Liang, S. S., and P. W. Trester, "Status Report on the Effects of Thermal Aging on Mechanical Properties of Alloy 800H," Doc. 908096/0, HTGR-85-44, June 18, 1985.
- 18b. Johnson, W. R., "Status Report on HTGR Helium Compatibility of Alloy 800H and Hastelloy X," Doc. 907514/B, September 25, 1984.
- 18c. Liang, S. S., et al., "Status Report on the Effects of Thermal Aging on Mechanical Properties of Hastelloy X and Alloy 800H," Doc. 907495/A, June 12, 1984.
- 18d. Trester, P. W., et al., "Assessment of Effects of Fort St. Vrain HTGR Primary Coolant on Alloy 800," EPRI NP-2548, August 1982.
19. "Status Report on Creep Rupture Tests of 2-1/4 Cr-1 Mo Steel, Alloy 800H, and Hastelloy X in Simulated HTGR-SC/C Helium," GA Doc. 908255/0, HTGR-85-131, August 1985.
20. Donnelly, M. E. (GE) letter to J. E. Fox (DOE), "Effect of Simulated Process Heat Helium on Fatigue Properties of HTGR Alloys," GE letter XL-895-830032, November 16, 1983.
21. Soo, P., and J. G. Y. Chow, "High Cycle Fatigue of HTGR Structural Alloys," Brookhaven National Laboratory Report, February 1980.
- 22a. Foulds, J. R., "Status Report on Alloy 800H Crack Growth," GA Doc. 908328/0, HTGR-85-153, September 1985.
- 22b. Foulds, J. R., "Status Report on Alloy 800H Crack Growth," GA Doc. 907608/1, August 31, 1984.
- 22c. Dahms, C. F., et al, "Properties of Recrystallized Alloy 800H and Associated HTGR Steam Generator Design Implementations," GA-C18396, February 1986.
- 22d. Foulds, J. R., "The effects of Recovery and Recrystallization on the High-Temperature Fatigue Crack Growth Behavior of Alloy 800H in Controlled-Impurity Helium", GA-A18321, December 1985.
- 23a. Liang, S. S., and P. W. Trester, "Status Report on the Effects of Thermal Aging on Mechanical Properties of Alloy 800H," GA Doc. 908096/0, HTGR-85-44, dated May 1985.
- 23b. Liang, S. S., et al., "Status Report of the Effects of Thermal Aging on Mechanical Properties of Hastelloy X and Alloy 800H," GA Doc. 907495/A, June 12, 1984.

- 23c. Lindgren, J. R., "Effects of Thermal Aging on Alloy 800H," GA Doc. 908980/0, September 1986.
- 24a. Kaae, J. L., and R. Villagrana, "Low Cycle Fatigue Data for Alloy 800H," GA Doc. 904446, March 11, 1980.
- 24b. Villagrana, R. E., et al., "The Effect of Aging and Cold Working on the High-Temperature Low-Cycle Fatigue Behavior of Alloy 800H: Part II Continuous Cycle Loading," GA Doc. GA-A15558, October 1979.
- 25. Foulds, J. R., "Status Report on Fracture Toughness Tests on Alloy 800H," Doc. 908811, March 1986.
- 26. Kasten, P. R., et al., "Gas-Cooled Reactor Programs HTGR Technology Development Program Annual Progress Report for Period Ending December 31, 1984," ORNL Report ORNL-6192, HTGR-85-062, July 1985, Section 4.4.
- 27. York, J. W., and R. L. Flury, "Assessment of Candidate Weld Metals for Joining Alloy 800," ORNL/Sub 4308-2, February 1976.
- 28. Lindgren, J. R., "Properties of Alloy 800H Weldments," GA Doc. 908232/0, HTGR-85-099, September 1985.
- 29. Lindgren, J. R., "Neutron Irradiation Effects on Properties of Candidate Metallics for Reactor Internals Components," memo MAB:104:JRL:85, October 31, 1985.
- 30. Lindgren, J. R., "Irradiation Effects on Candidate Reactor System Materials," GA Doc. 908822/0, May 30, 1986.

TABLE 4.1-1
METALLIC MATERIALS - SUBCOMPONENTS OF HOT DUCT

Subcomponent	Dimensions (Nominal)	Material
Pipe	47 in. i.d., 0.5 in. thick	ASME SB409 Condition X1.1.1.1 Solution annealed (Alloy 800H)
Elbow	41 in. i.d., 0.5 in. thick	ASME SB409 Condition X1.1.1.1 Solution annealed (Alloy 800H)
Thermal barrier		
o Coverplates	0.5 in. thick	ASME SB409 Condition X1.1.1.1 Solution annealed (Alloy 800H)
o Seal sheets	0.025 in. thick	ASME SB409 Condition X1.1.1.1 Solution annealed (Alloy 800H)
o Attachment fixture	5 in. diameter	ASME SB408, hot finished, semi-smooth machined (Alloy 800H)
Weldments ^(a)		Weld filler material - ASME SFA 5.14 ERNiCr-3 (Inconel 82)
o Alloy 800H to itself		
Bellows	0.031 in. to 0.063 in. thick	ASME SB409 Condition X1.1.1.1 Solution annealed (Alloy 800H)
Guide rail	0.5 in. thick plate; 3x4 in. bar	ASME SB408 and SB409 (Alloy 800H)
Upper steam generator shrouds	0.5 in. thick	ASME SB409 Condition X1.1.1.1 Solution annealed (Alloy 800H)

(a) Weldments exist in thermal barrier (between coverplates and attachment fixtures), in pipe, at joint between flange and bellows, at joint between bellows and elbow, and in upper steam generator shrouds.

TABLE 4.1-2
DESIGN TEMPERATURES FOR MHTGR

Service Level (1)	PSID Nomenclature	R. G. 1.48 Terminology (1)	Loading Condition	Number of Cycles N ⁽⁴⁾	Gas Temperature °C (°F)		Duration h	Reference
					Core Inlet	Average Core Outlet		
A		Normal	100% power	(3) 21,800 ⁽³⁾ 455	258 (497) ⁽⁷⁾	687 (1268) ⁽⁶⁾	280,000	2, 4
			50% steam flow		254 (490)	622 (1152)	(2)	
			25% steam flow		254 (489)	588 (1091)	(2)	2
			Refueling/shutdown		86 (187) ⁽⁵⁾	142 (288)		4
B	A00 1	Upset	Reactor trip from 100% power	131	284 (~490)	Table 4.1-2a	0.5/trip	5
C	SRDC 1 or	Emergency	Main loop trip and	1	Table 4.1-2b	See Table		5
D	SRDC 2	Faulted	failure of SCS	1		4.1-2b		
D	SRDC 11	Faulted	Depressurized conduction cooldown	1	538 (<1000)	<704 (<1300)		5

Notes:

- (1) The correlation between Service Level, and Regulatory Guide (RG) terminology is per Document No. 905842/2.
(2) The duration is included in 280,000 h at 100% power.
(3) The cycles indicated are from 100% power to indicated condition.
(4) Number of cycles is per and Table 4.1-3.
(5) Temperature quoted are lower expected design values for refueling condition. However, lower and higher temperatures are within design envelope. The lowest temperatures are 32°C (89°F) at core inlet and 83°C (181°F) at core outlet for refueling condition.
(6) Includes local hot streaks at hot duct entrance at temperatures up to 760°C (1400°F).
(7) 258°C is based on expected design value. However, temperature up to 269°C (516°F) are within design envelope.

TABLE 4.1-2a
THE CORE OUTLET HELIUM TEMPERATURE
DURING A REACTOR TRIP FROM 100% POWER
AOO NO. 1

Time Seconds	Temperature	
	°C	(°F)
0.0	704	(1300)
200	588	(1090)
1000	327	(620)
4000	310	(590)

TABLE 4.1-2b
PRIMARY COOLANT TEMPERATURE DURING
PRESSURIZED CONDUCTION COOLDOWN - SRDC NO. 2

(a) If there is a hot restart of the steam generator, the temperature in the upper plenum quickly cool down to (500°F). However, initially the temperatures in the Lower plenum increase. Based on calculations done in Ref. 13, it is estimated that the temperature in the lower plenum peaks at approximately 871°C (1600°F), approximately 1 h after initiation of hot restart if this hot restart occurs at time of peak temperatures (i.e., -100 h after initiation of pressurized conduction cooldown). Within approximately 1 h, temperatures will have decreased to below 704°C (1300°F). This is event 30b in Table 4.1-3.

TABLE 4.1-3
DESIGN DUTY CYCLE EVENTS

Event	Design No. of Occurrences (per Reactor Module)	Level of Service Limits
1. Startup from Refueling Conditions	143	A
2. Startup with Full Helium Inventory	312	A
3. Shutdown to Refueling Conditions	101	A
4. Shutdown with Full Helium Inventory	105	A
5. Rapid Load Increase (5% per min) (25%-100%)	1000	A
6. Normal Load Increase (0.5% per min) (25%-100%)	20800	A
7. Rapid Load Decrease (5% per min) (100%-25%)	1000	A
8. Normal Load Decrease (0.5% per min) (100%-25%)	17500	A
9. Step Load Increase (+15%)	1000	A
10. Step Load Decrease (-15%)	1000	A
11. Depressurized Decay Heat Removal, HTS to SCS Transition	80	A
12. Depressurized Decay Heat Removal, SCS to HTS Transition	122	A
13. Pressurized Decay Heat Removal, HTS to SCS Transition	61	A
14. Pressurized Decay Heat Removal, SCS to HTS Transition	86	A
15. Circulator Trip	30	B
16a. Reactor Trip from 100%	180(a)	B
16b. Reactor Trip from 25%		
17. Turbine Trip or Load Rejection	120	B

TABLE 4.1-3 (Continued)

Event	Design No. of Occurrences (per Reactor Module)	Level of Service Limits
18. Sudden Reduction of FW Flow	30	B
19. Steam Generator Tube Leak (Small)	9	B
20. Control Rod Insertion	5	B
21. Main Loop Overcooling	10	B
22. Operating Basis Earthquake (OBE)	1	B
23. Slow Primary System Depressurization	8	B
24a. Rod Withdrawal (normal rod speed) (P/F Trip)	1	C
24b. Rod Withdrawal (slow) (SGIT Trip)	1	C
25. Failure of Circulator Speed Control	1	C
26. Circulator Trip with He Shutoff Valve Failure	1	C
27. Steam Generator Tube Rupture	1	C
28. SCS Heat Exchanger Tube Leak	1	C
29. Total Loss of FW Flow	4	C
30a. Total Loss of SCS Cooling Water (HTS operating)	4	C
30b. Total Loss of SCS Cooling Water (SCS operating)	1	C
31. HTS Trip and Failure of SCS to Start (SRDC No. 1)	1	C
32. Loss of HTS Without Control Rod Insertion or SCS Operation (ATWS) (SRDC No. 2)	1	D(b)
33. Control Rod Withdrawal Followed by P/F Trip and RCCS Cooling (SRDC No. 3, 4)	1	D(b)

TABLE 4.1-3 (Continued)

Event	Design No. of Occurrences (per Reactor Module)	Level of Service Limits
34. Large Earthquake Followed by RCCS Cooling [Safe Shutdown Earthquake (SSE)] (SRDC No. 5)	1	D ^(b)
35. Steam Generator Tube Rupture Without Isolation (SRDC No. 6A)	1	D ^(b)
36. Steam Generator Tube Rupture Without Isolation, Followed by RCCS Cooling (SRDC No. 6B)	1	D ^(b)
37. Steam Generator Tube Leaks Without Isolation, Followed by RCCS Cooling (SRDC No. 7)	1	D ^(b)
38. Small Steam Generator Tube Leak Without Isolation, Followed by RCCS Cooling (SRDC No. 8)	1	D ^(b)
39. Small Steam Generator Tube Leak With Unterminated Dump Followed by RCCS Cooling (SRDC No. 9)	1	D ^(b)
40. Rapid Depressurization Followed by RCCS Cooling (SRDC No. 10)	1	D
41. Slow Primary System Depressurization With Loss of Forced Circulation (SRDC No. 11)	1	D ^(b)
42. Main Steam Pipe Rupture	1	D

(a) For components where reactor trip from 100% load is worse the breakdown should be 131 trips from 100% and 49 trips from 25%. For components where reactor trip from 25% load is worse, the breakdown should be 63 trips from 100% and 117 trips from 25%.

(b) In general, level D service limits are assigned to SRDCs for specified safety functions of safety related SSCs. However, level D limits are intended primarily for guidance. The plant level requirement is that 10CFR100 dose requirements not be exceeded. Event No. 31 (SRDC No. 1) and 40 (SRDC No. 10) are exceptions to this. Their minimum service limit levels are C and D, respectively.

TABLE 4.1-4
 "LONG TERM" EXPECTED AND DESIGN PRIMARY COOLANT IMPURITIES

Chemical Impurities	Expected ppmv	Design ppmv
H ₂ O	0.5	2
CO ₂	1	2
CO	2	5
H ₂	3	10
CH ₄	0.1	2
N ₂	2	10

(a) The ppmv signifies parts per million in volume. At rated power the design pressure is approximately 918 psia (62.5 atm) (per Ref. 4). Note that $0.5 \text{ ppmv} \times 62.5 \text{ atm} = 31.25 \mu \text{ atm}$.

TABLE 4.1-5
SUMMARY OF STRESSES INDUCED BY THERMAL AND
MECHANICAL LOADS ON HOT DUCT COMPONENTS^(b)

Source (Req. No.)	Type of Stress/ Component	Materials (a) Properties Needed
1 Temperature (R11, R12)		E, S_R , S_I , da/dt
1.1 ΔT between cover- plate and pipe	Low cycle fatigue (LCF) in attachment fixture	LCF $\Delta\epsilon$, da/dt, K_{IC}
1.2 ΔT of hot duct pipe (R20)	LCF in bellows	↓
2. Radial pressure drop (R11.2)	LCF of elbow and bellows	
3. Seismic: OBE (R12.1) and SSE (R12.2)	LCF of elbow and bellows	LCF $\Delta\epsilon$, da/dN, K_{IC}
4. Acoustic (R16)	High cycle fatigue (HCF) of coverplates/attachments	S_a , E
5. Compression of fibrous insulation	Steady state loading of coverplates and attachments	S_R , S_I , da/dt, K_{IC}
6. Depressurization (R12.3)	Short term loading of cover- plates and attachment fixtures	S_u , S_y , E, K_{IC}

(a) This section gives only those properties with identified design data needs.

(b) This table only addresses hot duct components which see elevated temperatures (>538°C) during full power operation.

TABLE 4.1-6
ALLOWABLE STRAIN RANGE OF ALLOY 800H FOR SELECTED CYCLIC CONDITIONS

i	Condition	°C	ΔT_i (°F) ^(a)	$\Delta T/\Delta x$ °C/cm (°F/in.) ^(b)	N ^(c)	Allowable $\Delta \epsilon$ ^(d)	$\sigma_{t \max}$ (ksi)	Induced $\Delta \epsilon_t$
1.	100% power	427	(769)	67 (308)			14	0.00057
2.	50% power	368	(662)	58 (265)			12	0.00049
3.	25% power	334	(601)	53 (240)			11	0.00044
4.	Shutdown for refueling	28	(50)	4.4 (20)			0.9	0.00004
5.	Shutdown-100%-shutdown	399	(719)	63 (288)	455 ^(e)	0.00058	13.1	0.00053
6.	25%-100%-25%	93	(167)	15 (67)	21,800	0.00023	3.1	0.00012
7.	50%-100%-50%	59	(106)	9.3 (42)			1.7	6.7 x 10 ⁻⁵

(a) For i=1 to 3 ΔT_i = temperature difference between core outlet (T_o) and core inlet (T_i) given in Table 4.1-2.
 $\Delta T_4 = 1/2 (T_o - T_i)$, $\Delta T_5 = \Delta T_1 - \Delta T_4$; $\Delta T_6 = \Delta T_1 - \Delta T_3$ and $\Delta T_7 = \Delta T_1 - \Delta T_2$.

(b) Based on $\Delta x = 2.5$ in. = 6.35 cm which is equal to thickness of fibrous insulation.

(c) N = number of cycles per Table 4.1-3; 455 is total of events 1 and 2 and 21,800 is from events 6 and 5.

(d) $\Delta \epsilon$ is the allowable strain range per Table T-1430-1C of Ref. 14 at 760°C (1400°F) for the number of cycles (N) indicated.

(e) Conservatively assume all shutdowns cycles given same low temperatures as shutdown for refueling.

TABLE 4.1-6
ALLOWABLE STRAIN RANGE OF ALLOY 800H FOR SELECTED CYCLIC CONDITIONS

i	Condition	°C	ΔT_i (°F) ^(a)	$\Delta T/\Delta x$ °C/cm (°F/in.) ^(b)	N ^(c)	Allowable $\Delta \epsilon$ ^(d)	$\sigma_{t \max}$ (ksi)	Induced $\Delta \epsilon_t$
1.	100% power	427	(769)	67 (308)			14	0.00057
2.	50% power	368	(662)	58 (265)			12	0.00049
3.	25% power	334	(601)	53 (240)			11	0.00044
4.	Shutdown for refueling	28	(50)	4.4 (20)			0.9	0.00004
5.	Shutdown-100%-shutdown	399	(719)	63 (288)	455 ^(e)	0.00058	13.1	0.00053
6.	25%-100%-25%	93	(167)	15 (67)	21,800	0.00023	3.1	0.00012
7.	50%-100%-50%	59	(106)	9.3 (42)			1.7	6.7×10^{-5}

(a) For i=1 to 3 ΔT_i = temperature difference between core outlet (T_o) and core inlet (T_I) given in Table 4.1-2.
 $\Delta T_4 = 1/2 (T_o - T_I)$, $\Delta T_5 = \Delta T_1 - \Delta T_4$; $\Delta T_6 = \Delta T_1 - \Delta T_3$ and $\Delta T_7 = \Delta T_1 - \Delta T_2$.

(b) Based on $\Delta x = 2.5$ in. = 6.35 cm which is equal to thickness of fibrous insulation.

(c) N = number of cycles per Table 4.1-3; 455 is total of events 1 and 2 and 21,800 is from events 6 and 5.

(d) $\Delta \epsilon$ is the allowable strain range per Table T-1430-1C of Ref. 14 at 760°C (1400°F) for the number of cycles (N) indicated.

(e) Conservatively assume all shutdowns cycles given same low temperatures as shutdown for refueling.

TABLE 4.1-7
AGING STUDY ON ALLOY 800H WELDMENTS (INCONEL 82 FILLER
MATERIAL, FULL TUBE SPECIMENS) AGING TIME 3000 h

Ultimate Tensile Strength					
Aging Temperature	Test Temperature				
	Room	649°C	760°C	815°C	900°C
As-welded	79.3 77.9			31.5 25.2	
593°C (1100°F)		73.8	42.5	27.6	17.9
649°C (1200°F)		60.2	39.9	29.9	21.0
704°C (1300°F)		51.2	31.3	30.3	20.9

NOTE: All failures occurred in parent metal.

GA Technologies

Fig 4.1-1

MODULAR REACTOR HOT DUCT
ELEVATION VIEW

Fig. 4.1-1b. Hot duct guide rails and supports - cross section looking toward reactor vessel

Fig. 4.1-2. MHTGR integrated approach functional analysis
for safe, economical power
Goal 1 -- Maintain plant operation
State 1.1.2.2.1.1. Channel primary coolant flow

Fig. 4.1-3. MHTGR integrated approach functional analysis for safe, economical power
 Goal 2 -- Maintain plant protection
 State 2.1.2.2.1.1. Protect the capability to channel primary coolant flow

Fig. 4.1-4. Fatigue results of Alloy 800H at 850°C (1562°F)

Fig. 4.1-5. Effect of an HTGR helium environment on the high cycle fatigue strength of Incoloy 800H

Fig. 4.1-6. Effect of thermal aging in helium on the high cycle fatigue of incoloy 800H at 760°C

Fig. 4.1-7. Effect of pretest thermal aging in HTGR Helium on the 10^8 cycle fatigue strength for Incoloy 800H at 760°C (1400°F)

b. Elevated-temperature 0.2% yield tensile strength

a. Elevated-temperature ultimate tensile strength

Fig. 4.1-8. The effect of thermal aging on tensile strength of Alloy 800H

Fig. 4.1-9. Effect of 10,000 h thermal aging (in air) on the ductility of Alloy 800H (GA 409) at room temperature and at the aging temperature

Fig. 4.1-10. Relation between the strain range and the macroscopic fatigue life

Fig. 4.1-11. Room temperature toughness as a function of aging temperature for Alloy 800H

Fig. 4.1-12. Comparison between weld metal Inconel 82 and Alloy 800H

UNIRRADIATED CONTROL SPECIMEN ELONGATIONS RANGED FROM 40% TO 70%.
NOTE GRAIN SIZE SPECIFIED FOR ALLOY 800H IS 30 μ TO 50 μ DIAMETER.

Fig. 4.1-13. Postirradiation ductility of experimental heats of Alloy 800 irradiated at 650°C and 700°C to about 0.8×10^{21} neutron/cm² (thermal) and tested at 700°C.

Fig. 4.1-14. Creep-rupture results at 705°C of unirradiated and General Electric Test Reactor irradiated Alloy 800

Fig. 4.1-15. Creep rupture of experimental Alloy 800 containing 0.13% C and (a) 0.17% or (b) 0.10% Ti. Specimens were irradiated or soaked (controls) at 760°C and tested in air at 760°C. Both tensile and creep data are plotted. Exposure of irradiated specimens was 2 to 3 x 10²⁰ n/cm² (rh).

Fig. 4.1-16. Irradiated and unirradiated fatigue behavior of Alloy 800 Tested at 700°C (1292°F) compared with predictions based on tensile data

Fig. 4.1-17. U. S. HTGR Program reference schedule

4.2 Steam Generator

The steam generator subsystem is part of the Heat Transport System.

4.2.1 Design Description

The steam generator is a vertically oriented, upflow boiling, cross counterflow, once-through shell and tube heat exchanger. It utilizes a multi-tube, helically wound tube bundle with primary coolant on the shell-side and secondary coolant in tubes.

The steam generator subsystem includes the following components: the cooling tubes, support plates, tubesheets, and shrouds. Figures 4.2-1 and 4.2-2 presents the integrated view of the steam generator subsystem components. The material used for each component is as specified in Table 4.2-1.

The tube material for the economizer, evaporator, and primary superheater (EES) sections of the steam generator is 2-1/4 Cr-1 Mo and the tube material for the finishing superheater (FSH) section of the steam generator is Alloy 800H. These two tube materials are joined together by bimetallic welds between the two tube bundles. The tube bundle was sized to transfer approximately 352 MW(t) at rated conditions from primary to secondary coolant for each module. The selected heat transfer area size is:

Number of tubes	= 439
Number of coils	= 28
Tube o.d.	= 0.875 in.
Tube i.d.	= 0.615 in.
Radial tube pitch	= 1.50 in.
Longitudinal tube pitch	= 1.391 in.
Heated tube length	= 431.0 ft
Tube bundle o.d.	= 148.5 in.
Tube bundle i.d.	= 67.5 in.

Support of the helical tube bundle is by means of six full, six partial equally-spaced drilled radial support plates, each approximately 3.81 cm (1-1/2 in.) thick, through which the tubes are wound. The finishing superheater (FSH) support plates are separately supported directly above and in-line with the economizer-evaporator-superheater (EES) support plates. The tube support plates are supported vertically at the lower end of the tube bundle by twelve gussets and horizontally by the inner and outer shrouds.

The primary coolant enters the inlet plenum above the steam generator bundle from the hot duct. Inlet plenum shroud is welded to the top of the tube bundle outer shroud and has a thickness of 2.54 cm (1.0 in.). The primary coolant flows downward across the helical tube bundle between the inner and outer shrouds and exits into a lower outlet plenum. The tube bundle outer shroud thickness is 3.81 cm (1.50 in.) and the inner shroud thickness is 3.81 cm (1.50 in.). The bottom of the inner shroud is enclosed to prevent flow bypass of the tube bundle. In the lower outlet plenum, the primary coolant is turned 180 ° into a return annulus defined by the steam generator vessel and the tube bundle outer shroud. After flowing upward through this annulus, the primary coolant enters the circulator inlet plenum above the cross duct elevation.

Secondary coolant is introduced through a penetration assembly welded to the bottom of the steam generator vessel. From the feedwater tubesheet that contains the tube orifices, the tubes are splayed out to form a lower-bundle expansion loop, the purpose of which is to accommodate differential expansion between the feedwater tubesheet and the lower extremity of the tube bundle, and routed to the lower end of the helical section of the tube bundle. After being spiraled upward through the EES section of the tube bundle, the tubes are splayed outward to the peripheral region of the tube bundle envelope to form a mid-bundle expansion loop. This mid-bundle expansion loop is to accommodate differential expansion between the EES tube bundle and the FSH tube bundle. Above the mid-bundle expansion loop, the tubes are routed inward to form the helical finishing superheater bundle. After having spiraled

through the finishing superheater bundle, the tubes are routed to form an upper-bundle expansion loop (to accommodate the gross axial expansion which will exist between the bundle and the steam generator vessel) and converged to their attachment points on the superheated steam outlet tubesheet through stagnant helium region below cross duct elevation. Each tube is continuous from tubesheet to tubesheet to allow tubeside in-service inspection.

4.2.2 Relevant Functions

The design requirements and the assumptions discussed in Section 4.2.3 result from the need to satisfy the functions listed in Table 4.2-2 as they relate to the steam generator. How these functions tie together in the functional analysis tree is shown in Figs. 4.2-3a through 4.2-3b.

The primary function of the steam generator is to transfer energy from primary coolant to secondary coolant (F1.1.2.3). To satisfy this function requires that the subfunction to transfer energy from primary coolant to heat exchanger (F1.1.2.3.1) and to transfer energy from heat exchanger to secondary coolant (F1.1.2.3.2) also be satisfied. To transfer energy from primary coolant to heat exchanger, means for the primary coolant to flow through heat exchanger (F1.1.2.3.1.1) and to support heat transfer surfaces (F1.1.2.3.1.2) have to be provided. Also, to transfer energy from heat exchanger to the secondary coolant, the means to flow secondary coolant through heat exchanger (F1.1.2.3.2.2) has to be provided.

4.2.3 Design Requirements/Assumptions

This section lists requirements which must be satisfied to adequately ensure that the steam generator transfers energy from the primary coolant to the secondary coolant (i.e., function number 1.1.2.3). These requirements include design loads, environmental conditions, and standards for ensuring that the steam generator is performing satisfactorily. Also included are the assumptions which the designers judged to be prudent in order to show compliance with these requirements.

<u>Reference</u>	<u>Requirement</u>
1	R1 Withstand 40-yr operating life
1	R1.1 Availability allocation for components designed to maintain energy transfer ≥ 7008 effective full power hours/year average operating
3	R11 Accommodate temperature, pressure, and flow of primary and secondary coolants
4, 6, 5	R11.1 Accommodate primary and secondary conditions given in Table 4.2-3
3, 4	R11.2 Total pressure losses in primary circuit less than 91 kPa (13.2) psi, at 100% rated reactor power
6, 14	R11.3 Total helium pressure loss from SG inlet to circulator inlet not to exceed 25.5 kPa (3.7) psi
7	R11.4 Accommodate a maximum helium hot/cold streak of $\pm 14^{\circ}\text{C}$ ($\pm 25^{\circ}\text{F}$) the SG tube bundle inlet
3, 4, 14	R11.5 Perform within the operational uncertainty envelopes specified for 100% and 25% power (see Fig. 4.2-4 and Table 4.2-4)
1, 8	R12 Accommodate duty cycle events shown in Table 4.1-3

- R12.1 Same as given in Section 4.1.3
 R12.2 Same as given in Section 4.1.3
 R13 Compatible with primary coolant

<u>Reference</u>	<u>Requirement</u>
9	R13.1 The primary coolant is helium with chemical impurities as specified in Table 4.1-4
6	R14 Secondary coolant will be in the tubes
5, 10	R14.1 Water shall be secondary coolant with chemical impurities as specified in Table 4.2-5
11	R16 Accommodate design acoustic loads
12 13, 26	R18 Approximately 350 MW(t) per reactor shall be transferred at rated conditions from primary to secondary coolant

Trade studies have been performed to ensure that the steam generator satisfies the above requirements. These studies show that the maximum normal operating temperature of the 2-1/4 Cr-1 Mo tubes is in the range of 454° to 538°C (850° to 1000°F). These trade studies rely on the following assumptions which need to be verified before the steam generator design can be qualified.

<u>Reference & Trade Studies</u>	<u>Assumption</u>
6	1. Exposure to primary coolant chemistry and temperature over design life will not significantly degrade the design properties of metals used in steam generator.

- 6 2. Proposed new creep-fatigue damage rules to Code
Case N-47 for 2-1/4 Cr-1 Mo and Alloy 800H are
conservative.

<u>Reference</u>	<u>Requirement</u>
6	3. Changes to Code Case N-47 concerning weldment properties of 2-1/4 Cr-1 Mo and Alloy 800H will not significantly affect the steam generator design.

The following subsection discusses why it is necessary to make each assumption and how each assumption affects the steam generator. Also, a discussion is given of the current data base.

4.2.3.1 Properties of 2-1/4 Cr-1 Mo Base Metal and Weldments. The primary coolant contains impurities (per Table 4.1-4) which can cause corrosion in the form of oxidation, decarburization and carburization. At the maximum design temperatures of the 2-1/4 Cr-1 Mo components (see R11), carbon transport has been shown to be the most potentially significant mode of corrosion with respect to bulk mechanical properties such as tensile and creep properties. In addition, surface oxidation along with concurrent carbon transport may significantly affect surface sensitive properties such as fatigue, creep-fatigue, and crack growth. Finally long-term thermal exposure may cause design-significant changes in material strength and toughness.

Data reported in Ref. 15 for 2-1/4 Cr-1 Mo exposed to dry HTGR-helium with chemical impurities in μatm of 2 H_2O , 19 CO , 350 H_2 , 32 CH_4 , and <0.5 N_2 showed that 2-1/4 Cr-1 Mo is carburized very slowly at 480°C , unchanged in carbon content at 540°C and decarburized at 595°C and 650°C (see Fig. 4.2-5).

In wet HTGR-He with chemical impurities in μatm of $15.2 \text{ H}_2\text{O}$, 50.2 CO , 15.2 CO_2 , 10.1 H_2 and 10.1 CH_4 data showed (Ref. 15) that in 7000 h

2-1/4 Cr-1 Mo decarburized slightly at 430°C and 510°C, moderately at 580°C and severely at 650°C.

In the tests performed to date, decarburization while exposed to sodium caused significant changes in the tensile and creep properties (Ref. 16). Under most conditions the property changes involved a reduction in strength. Also, from the creep-rupture tests in HTGR helium environment (Ref. 17) it was observed that 2-1/4 Cr-1 Mo steel tested at a temperature of 593°C (1100°F) is weaker than predicted from vendor air data.

Data are reported in Ref. 18 for 2-1/4 Cr-1 Mo exposed to helium with chemical impurities in μatm of 2 H₂O, 19 CO, 337 H₂, 32 CH₄, and <0.5 N₂ for up to 20,500 h at temperatures up to 593°C. These data indicated rather small changes in carbon level but a change in microstructure which was felt due to aging at 593°C. In addition, for specimens exposed at 593°C:

Based on tensile tests at 24°C (75°F)

Yield strength (S_y) $\approx$ 0.90 S_y of as-received

Ultimate strength (S_u) $\approx$ 0.95 S_u of as-received

No significant change in ductility

Based on creep tests at 593°C (1100°F) at 103 MPa (15 ksi)

Time to rupture (t_R) was reduced by factors of up to 6 from as-received material depending on heat tested

Available information on the creep-fatigue behavior of 2-1/4 Cr-1 Mo steel (Ref. 15) have indicated that this material tested over the temperature range 490°C (914°F) to 593°C (1100°F) in helium environment had improved fatigue life for all test wave forms in comparison with tests in air except for the tensile hold only tests.

Limited data to date has shown that for this material, the thermal aging while exposed to argon caused significant reduction in the tensile and creep properties (Ref. 16).

Recently, ASME Code Case N47 came out with a creep-fatigue design curve for 2-1/4 Cr-1 Mo that represents an approach of using total damage (D) values which intersect at values of 0.1 and 0.1 for all temperatures (see Fig. 4.2-9). There were other suggested approaches to the creep-fatigue design curves for this material; however, this approach was easy to use. It is assumed that this new rule is conservative (Assumption 5).

Some limited test data obtained recently on 2-1/4 Cr-1 Mo weldments (Ref. 19) suggests that these weldments might be as strong as the base metal. The current HTGR steam generator design is evaluated with the assumption that the changes to Code Case N-47 concerning weldment properties of 2-1/4 Cr-1 Mo will not significantly affect the steam generator design (Assumption 6).

There are some fracture mechanics data for annealed 2-1/4 Cr-1 Mo steel (Refs. 20 and 21). These included fracture toughness data at 24°C, 260°C and 538°C (75°F, 500°F and 1000°F). These data yield maximum K_{IC} values of 130 MPa $\sqrt{m}$ at 24°C, 88 MPa $\sqrt{m}$ at 260°C and 212 MPa $\sqrt{m}$ at 538°C.

Figure 4.2-6 shows the results of fatigue crack growth data. The results are not affected significantly by temperature from 24° to 260°C, but that is not surprising for this alloy. The test results at 538°C, however, do show an increase in fatigue crack growth relative to the lower test temperature results. There are not enough data to examine a threshold behavior; thus, linear curve fits were drawn only in the stable FCGR region. Although data smoothing has not been performed as outlined in ASTM E647, the data compare reasonably well with previous data obtained with annealed 2.25 Cr-1 Mo at room temperature.

There have been many parameters proposed to correlate creep crack growth rate data. Some of these are stress intensity factor (K), J-integral, transient value of creep crack growth rate (C_t), net section stress, and reference stress. For preliminary calculations, stress intensity, K and C_t have been used in Ref. 21. The da/dt data have been plotted against these

parameters in Figs. 4.2-7 and 4.2-8, respectively. Comparisons are made with other results (Refs. 22, 23, and 24) and are included on the plots.

Data are needed to determine the effects of exposure to primary coolant (He) with its design impurities and temperature over design life on selected properties of 2-1/4 Cr-1 Mo base metal and its weldments. These data need to be sufficient to quantify to a 95% confidence level.

4.2.3.2 Properties of Alloy 800H Base Metal and Weldments. The effects of primary coolant chemistry and temperature on Alloy 800H are discussed in Section 4.1.3.1. Some additional discussion is given below.

Current Code Case N-47 contains creep-fatigue damage allowables for Alloy 800H which are based on limited data and on oversimplified damage model (see Fig. 4.2-9). Furthermore, the test results obtained to date (Ref. 25) indicate that the linear damage design criterion in Code Case N-47, which shows total damage (D) values based on 1, is highly nonconservative for Alloy 800H at 650°C (1200°F) when applied to constant strain hold creep-fatigue loading cycles. It is also nonconservative, but to a lesser extent, for similar loading cycles on Alloy 800H at 760°C (1400°F). A new creep-fatigue damage rule has been proposed as a result (see Fig. 4.2-10). This change reflects a bilinear damage design criterion based on "D" values that intersect at 0.1 and 0.1 for all temperatures. It is assumed that these proposed new rules are conservative (Assumption 5).

Data are needed to determine the effects of exposure to primary coolant (He) with its design impurities and temperature over design life on selected properties of Alloy 800H base metal and its weldments. These data need to be sufficient to quantify to a 95% confidence level.

4.2.4 Technology Requirements

This section gives details of the data needs associated with each of the assumptions discussed in Section 4.2.3. Quality assurance must satisfy

effects of irradiation at 1100°F and 1400°F on fracture resistance. Tests will be performed in air (see 1 above).

K_{Ic} tests will be performed in air at room temperature and at the irradiation temperatures on all product forms.

Fatigue crack growth rates (da/dN) will be measured at the irradiation temperatures on all product forms. Tests will be conducted in the design environment at minimum stress to maximum stress ratios of 0.05 to 0.75 and a frequency of 1 Hz.

Creep crack growth (da/dt) rates will be measured at two stress levels for each irradiation temperature on all product forms. Testing will be done at the irradiation temperature in the design environment.

4. Creep tests will be conducted at the irradiation temperatures in the design environment. The tests on plate will be designed for 5000 h rupture and those on the weldments and weldmetal will be scheduled for rupture in 2000 h.
5. High-cycle fatigue - Tests will be run in the range 5×10^5 to 10^7 cycles at the irradiation temperatures to permit extrapolation to 3×10^{11} cycles. All product forms will be represented and a limited number of additional specimens will be irradiated to permit tests at other temperatures if required.

Specimen designs for all of the tests above (and those described in Sections to follow) will conform to those allowed by ASTM Standards.

5.1.2.2 Alloy 800H Primary Coolant Chemistry and Temperature Effects (DDN M.10.17.28). The effects of thermal aging (1000°F to 1400°F) and expected and design primary coolant compositions (see Table 4.1-4) on the properties of Alloy 800H will be determined to satisfy the technology require-

the requirements of 10CFR50 Appendix B. Section 4.2.3 specifies that sufficient data are required to determine to a 95% confidence that exposure to a given environment does not degrade selected properties more than a specified amount. This reference to a 95% confidence is not meant to specify that the data must be subjected to a pure statistical treatment to give the designer and technologist the indicated confidence. Rather it is intended that the technologist rely on techniques previously used to determine design properties given in the ASME B&PV Code. These techniques are based on engineering principles together with statistics.

4.2.4.1 Determine Properties of 2-1/4 Cr-1 Mo Base Metal and Weldments

(DDN M21.02.02). Sufficient data are required to determine to a 95% confidence the effects of thermal aging and corrosion at temperatures from 371°C (700°F) to 593°C (1100°F) for up to 300,000 h on the following properties of 2-1/4 Cr-1 Mo tubing and plate material and its weldments made using the gas tungsten arc process as specified for use in the steam generator in Table 4.2-1. The corrosion of the tubing and its weldments is that resulting from exposure to helium primary coolant with its impurities (per Table 4.1-4) on the outside and steam/water secondary coolant with impurities [per Table 4.2-5] on the inside. The corrosion of the plate material is that resulting from exposure to helium primary coolant with its impurities (per Table 4.1-4).

- a. Tensile properties in helium as a function of temperatures from 204°C (400°F) to 593°C (1100°F)* with design values for:
 - o Minimum ultimate tensile strength (S_u) of not less than 0.8 minimum S_u for as-received material.
 - o Minimum yield strength (S_y) of not less than 0.8 minimum S_y for as-received material.
 - o Average elastic modulus (E) within $\pm 20\%$ of average E for as-received material.

- o Stress-strain values up to 2% total strain.
- b. Cyclic stress/strain properties in MHTGR-helium at temperatures from 204°C (400°F) to 593°C (1100°F)* for at least first 10 cycles.
- c. Low cycle fatigue strain range ($\Delta\epsilon$) to 1000 cycles in MHTGR-helium at temperatures from 204°C (400°F) to 593°C (1100°F)* with a design value of at least 5×10^{-4} .
- d. Creep properties in MHTGR-helium at temperatures from 371°C (700°F) to 593°C (1100°F)* for duration up to 3×10^5 h with design values for:
 - o Minimum stress to rupture (S_R) of not less than S_R for as-received material.
 - o Minimum stress to cause 1% total strain (S_1) of not less than S_1 for as-received material.
 - o Minimum time to onset of tertiary creep.
- e. Creep-fatigue interaction properties in MHTGR-He at temperatures from 371°C (700°F) to 593°C (1100°F)* for 400 cycles with a strain range of at least 0.05% and hold times of (750) h to represent steam generator operating cycles.
- f. Fracture toughness properties for only the tubing material and its weldments in helium at temperatures from 204°C (400°F) to 538°C (1000°F) with design values for:
 - o Critical stress intensity factor (K_{IC}) of greater than $[49.5 \text{ MPa } \sqrt{\text{m}} (45 \text{ ksi } \sqrt{\text{in.}})]^*$.

* 538°C (1000°F) for tubing material.

- o Fatigue crack growth rate (da/dn) of less than $[8.25 \times 10^{-4} \text{ mm/cycle } (3.25 \times 10^{-5} \text{ in./cycle})]$ for $\Delta K \leq 45 \text{ MPa } \sqrt{\text{m}}$ ($41 \text{ ksi } \sqrt{\text{in.}}$) and $R = 0.0$ to 0.75 .
- o Creep crack growth rate (da/dt) of less than (TBD in./s) for K (TBD $\text{ksi } \sqrt{\text{in.}}$).
- o Effect of hold times on crack growth rates.

4.2.4.2 Determine Properties of Alloy 800H Base Metal and Weldments

(DDN M21.02.03). Sufficient data are required to determine to a 95% confidence that effects of corrosion resulting from exposure to primary coolant with its design impurities (per Table 4.1-4) and thermal aging at temperatures from 427°C (800°F) to 649°C (1200°F) for up to $3 \times 10^5 \text{ h}$ on the following properties of Alloy 800H and its weldments using Inconel 82 filler material as specified for use in the steam generator in Table 4.2-1.

- a. Tensile properties for tubing, plate, and forging material in helium as a function of temperatures from 316°C (600°F) to 649°C (1200°F) with design values for:
 - o Minimum ultimate tensile strength (S_u) of not less than 0.8 minimum S_u for as-received material.
 - o Minimum yield strength (S_y) of not less than 0.8 minimum S_y for as-received material.
 - o Average elastic modulus (E) within $\pm 20\%$ of average E for as-received material.
 - o Stress-strain values up to 2% total strain.

* These values are based on calculations at 100% SS and 482°C (900°F). The evaluation of transient conditions at lower temperatures (e.g., cooldown) could change these values significantly.

- b. Cyclic stress/strain properties in helium for tubing, plate, and forging material at temperatures from 316°C (600°F) to 649°C (1200°F) for at least first 10 cycles.
- c. Low cycle fatigue strain range ($\Delta\epsilon$) to 1000 cycles for tubing, plate, and forging material in MHTGR-helium at temperatures from 316°C (600°F) to 649°C (1200°F) with a design value of at least 5×10^{-4} in./in.
- d. High cycle fatigue strength to 1×10^6 cycles for tubing material in MHTGR-helium at temperatures from 471°C (880°F) to 642°C (1188°F) with a design value of at least 6.7 MPa (1.0 ksi).
- e. Creep properties for tubing, plate, and forging material in MHTGR-helium at temperatures from 427°C (800°F) to 649°C (1200°F) for duration up to 3×10^5 h with design values for:
 - o Minimum stress to rupture S_R of not less than S_R for as-received material.
 - o Minimum stress to cause 1% total strain S_1 of not less than S_1 for as-received material.
 - o Minimum time to onset of tertiary creep.
- f. Creep-fatigue interaction properties for tubing, plate, and forging material in MHTGR-helium at temperatures from 427°C (800°F) to 649°C (1200°F) for 400 cycles with a strain range of at least 0.05% and hold times of (750) h to represent steam generator operating cycles.
- g. Fracture toughness properties for the tubing and forging material in MHTGR-helium at temperatures from 316°C (600°F) to 649°C (1200°F) with design values for:

- o Critical stress intensity factor (K_{IC}) of greater than $[77 \text{ MPa } \sqrt{\text{m}}$
 $70 \text{ ksi } \sqrt{\text{in.}})]$.*
- o Fatigue crack growth rate (da/dn) of less than
 $[8.25 \times 10^{-4} \text{ mm/cycle } (3.25 \times 10^{-5} \text{ in./cycle})]$ for $\Delta K \leq 47 \text{ MPa } \sqrt{\text{m}}$
 $(43 \text{ ksi } \sqrt{\text{in.}})$ and $R = 0.0$ to 0.75 .
- o Creep crack growth rate (da/dt) of less than (TBD in./s) for K of
(TBD $\text{ksi } \sqrt{\text{in.}}$).
- o Effect of hold times on crack growth rates.

4.2.5 Schedule Requirements

Interim results, as they become available, on all data defined in Section 4.2.4 are needed throughout the conceptual and preliminary design phases (i.e., 10/89) based on schedule shown in Fig. 4.1-17. Final results on data are needed 12 months into the final design phase for incorporation into the final design and for ASME code action. This implies that all data are needed by 10/90 based on schedule shown in Fig. 4.1-17.

4.2.6 References

1. GCRA 86-002, Rev. 1, "Utility/User Design Requirements for the Modular High Temperature Gas-Cooled Reactor Plant," Doc. No. PC-000217/1, June 16, 1986.
2. Lindgren, J. R., "Updated Specification Alloy 800H Tubing," GA Doc. No. 908327/0, HTGR-85-132, September 22, 1985.
3. "Conceptual Design Data for Modular HTGR Plant," HTGR-86-023, GA Doc. No. 907807/7, April 1986.
4. Strausberg, S., to File, "Part Load Operating Condition," memo SD&PD:SS:121:85, December 12, 1985.

* These values are based on calculations at 100% SS and 1000°F to 1160°F. The evaluation of transient conditions at lower temperatures (e.g., cooldown) could change these values significantly.

5. "HTGR Steam Cycle Configuration Evaluation Report," GA Doc. No. GA-A14900, April 14, 1978.
6. Rao, R., "Power Level Assessment and Evaluation of a 3 x 350 MW(t) HTGR Concept," GA Doc. No. 908262/0, September 1985.
7. Strausberg, S., to File, "Hot Streak at Inlet to SG Bundle for the 4 x 350 MW(t) Modular HTGR," memo SD&PD:SS:147:85, November 11, 1985.
8. Lommers, L., "Modular HTGR Temperature Gas-Cooled Reactor Design Duty Cycle," GA Doc. 908728/1, April 30, 1986.
9. Baccaglini, G., "Plant Specification - Primary Coolant Chemistry - 2240 MW(t) HTGR-SC/C Plant," GA Doc. No. 906385/2, July 30, 1984.
10. Combustion Engineering, "CE Report Assessing the Water Chemistry Requirements and Materials of Construction for the HTGR-SC/C Lead Plant," CE Doc. ADD-85-540/14001, November 30, 1983.
11. Baccaglini, G., "Heat Transport System Design Report," GA Doc. No. 908759/0, HTGR-86-122 November 17, 1986.
12. Breher, W., "FY-84 Side-by-Side MRS Work Status Report," GA Doc. No. 907648/1, September 28, 1984.
13. Bellis, E. A., "Sensitivity of Temperature to Power and Power Density for MPBR Core Heatup Study," GA Doc. No. 907545/1, August 23, 1984.
14. Same as Ref. 11.
15. "Gas-Cooled Reactor Programs, HTGR Technology Development Program Annual Progress Report for Period Ending December 31, 1984," HTGR-85-062, ORNL-6192, July 1985, Section 7.4.
16. "Influence of Decarburization on the Tensile and Creep Properties of 2-1/4 Cr-1 Mo Steel," HTGR-85-157, ORNL/TM-9887, November 1985.
17. Excerpt from GA Technologies' Technology Program Semi-Annual Report for the Period October 84 - March 85, Section 7.3 - Heat Transport Materials.
18. McCoy, H. E., "Influence of Aging 2-1/4 Cr-1 Mo Steel to 20,500 h in HTGR-He," ORNL Report ORNL/GCR-85/6, HTGR-85-030, April 1985.
19. "Weld Property Tests on 2-1/4 Cr-1 Mo Steel Weldments," GA Document No. 908233, Issue 0, October 10, 1985.
20. "Materials Data for Fracture Mechanics Evaluation of HTGR Steam Generator," ORNL/GCR-85/13, September 1985.

21. Nonstad, R. K. and R. W. Swindeman, "Progress Report Materials Data for Fracture Mechanics Evaluation of HTGR Steam Generator," ORNL report dated September 1986.
22. Iwadate, T., J. Watanabe, and Y. Tanaka, Page 30 in "Residual Life Assessment, Nondestructive Examination, and Nuclear Heat Exchanger Materials," PVP-Vol. 98-1, American Society of Mechanical Engineers, New York, 1985.
23. Ohji, K., K. Ogura, S. Kubo, and Y. Katada, Page 9 in "Engineering Aspects of Creep," Vol. 2, Institute of Mechanical Engineers, London, 1980.
24. Saxena, A., and J. D. Landes, "Characterization of Creep Crack Growth of Metals," Paper 83-1D7-EVFLA-P1, Westinghouse Research and Development Center, Pittsburgh, June 1983.
25. "Creep-Fatigue Tests on Alloy 800H," GA Document No. 907114, Issue 4 September 19, 1985.
26. "Summary Report of FY85 HTGR Steam Generator Studies for GA Technologies Inc.," Report CENC 1707, Volume 1 through 2, September 1985.

TABLE 4.2-1
PRINCIPAL MATERIALS FOR THE STEAM GENERATOR COMPONENTS

Component	Dimensions (Nominal)	Material	Spec. No.	Heat Treatment
EES bundle helical tubes	0.875 in. o.d. 0.615 in. i.d.	2-1/4 Cr-1 Mo	SA-213, T22	Annealed
EES bundle tube support plates	1-1/2 in. thick	2-1/4 Cr-1 Mo	SA-387, Gr. 22, Cl. 1	Annealed
Inner and outer tube bundle shrouds	1.5 thick	2-1/4 Cr-1 Mo	SA-387, Gr. 22, Cl. 1	Annealed
FSH bundle helical tubes	0.875 in. o.d. 0.615 in. i.d.	Alloy 800H	SB-163, Gr. 1 (Note a)	Solution annealed
FSH bundle tube support plates	1-1/2 in. thick	Alloy 800H	SB-409, Gr. 1 (Note a)	Solution annealed
Superheater tubesheet	10.0 in. thick 38.30 in. o.d.	Alloy 800H forging	SB-564 (Note a)	Solution annealed
Feedwater tubesheet	8.0 in. thick 35.05 in. o.d.	Carbon/Moly steel	SA-508, Cl. II	Quenched & tempered
Weldments				
o Alloy 800H tubes to 2-1/4 Cr-1 Mo tubes		See Table 4.2-1a		
o Alloy 800H tubes to itself		See Table 4.2-1a		
o Alloy 2-1/4 Cr-1 Mo to itself		See Table 4.2-1a		

(a) The standard specification indicated for Alloy 800H is to be modified as specified in Ref. 2.

TABLE 4.2-1a
PRINCIPAL WELDING PROCESSES AND FILLER MATERIALS

Material	Spec. No.	Welding Process	Filler Material
2-1/4 Cr-1 Mo Plate ^(a)	SA 387, Gr. 22, Cl. 1	Submerged arc	Wire - SFA 5.23 EB3 Flux - Linde 124 Flux - Oerlikon OP76
		Shielded metal arc	Electrode - SFA 5.5 E9018-B3
		Gas tungsten arc	Wire - SFA 5.28 ER90S-B3
2-1/4 Cr-1 Mo Tube	SA 213, T22	Gas tungsten arc	Wire - SFA 5.28 ER90S-B3
Alloy 800H Plate	SB-409, Gr. 1	Submerged arc	Wire - SFA 5.14 ErNiCr-3 (Inconel 82) Flux - INCO Flux 4
		Shielded metal arc	Electrode - SFA 5.11 ENiCrFe2 (Incoweld A) Electrode - SFA 5.11 ENiCr Fe-3 (Inconel 182)
		Gas tungsten arc	Wire - SFA 5.14 ErNiCr-3 (Inconel 82)
Alloy 800H Tube	SB-163, Gr. 1	Gas tungsten arc	Wire - SFA 5.14 ErNiCr-3 (Inconel 82)
BMW Tube	Sb-163, Gr. 1/SA 213, T22	Gas tungsten arc	Wire - SFA 5.14 ErNiCr-3 (Inconel 82)

(a) 2-1/4 Cr-1 Mo Plate Material Weldment to Itself: The three basic welding procedures that would be utilized in joining this material are submerged arc welding, shielded metal arc welding, and gas tungsten arc welding. The submerged arc process would be used for the thicker weldments where a large volume of weld metal is required and where automation is possible such as the longitudinal and girth seams on shrouds. The shielded metal arc process would be used for thinner sections, and out of position welding on thicker sections. In general, the root pass on critical welds would be made with the gas tungsten arc process.

(b) Alloy 800H Plate Material Weldment to Itself: The three basic welding processes that would be utilized in joining this material are submerged arc welding, shielded metal arc welding, and gas tungsten arc welding. The submerged arc process would be used for the thicker weldments where a large volume of weld metal is required and where automation is possible. For out of position welds there are two basic choices of electrodes, Inconel 182 and Incoweld A. Based on past experience, the Inconel 182 performs better with respect to fabrication. The gas tungsten arc process would be used for making critical root passes and other critical out of position welds.

PC-000210/4

TABLE 4.2-2
RELEVANT FUNCTIONS OF THE STEAM GENERATOR

FN	Function ^(a)	Function Number
1	Transfer energy from primary coolant to secondary coolant	1.1.2.3
2	Transfer energy from primary coolant to heat exchanger	1.1.2.3.1
3	Transfer energy from heat exchanger to secondary coolant	1.1.2.3.2

(a) These functions are for Goal 1 - Maintain Plant Operation. The associated functions for Goal 2 - Maintain Plant Protection are also applicable. These functions read, "Protect the Capability to"

TABLE 4.2-3
PRIMARY AND SECONDARY SIDE CONDITIONS

	100% Power	25% Power
Reactor power per module	350 MW(t)	88.72 MW(t)
Helium maximum pressure, MPa (psia)	6.38 (925)	6.12 (887)
Steam generator helium inlet temperature, °C (°F)	686 (1266)	587 (1089)
Steam generator helium outlet temperature, °C (°F)	255 (491)	254 (489)
Feedwater inlet temperature, °C (°F)	193 (380)	194 (381)
Steam outlet temperature, °C (°F)	540 (1005)	538 (1000)
Steam outlet pressure, MPa (psia)	17.3 (2515)	16.7 (2421)
Helium inlet flow rate, kg/s (lb/h)	157.2 (1,248,100)	50.7 (402,940)
Feedwater inlet flow rate, kg/s (lb/h)	137.2 (1,089,736)	34.3 (272,425)

Table 4.2-4

Table 4.2-4

STEAM GENERATOR SUBSYSTEM OPERATIONAL
POINTS WITHIN THE UNCERTAINTY ENVELOPE AT 100% FEEDWATER FLOW (d)

	Point C (a)	Point 1
<u>Uncertainties Considered (c)</u>		
(b) Reactor Power,	0%	-2%
(b) SG heat transfer effectiveness	0%	+9%
(b) Total loop flow resistance	0%	+10.7%
<u>Heat Balance</u>		
(b) Reactor power, MW(t)	350	343
Heat added by circulators, MW(t)	3.13	3.48
Net power from SG, MW(t)	352.1	345.4
<u>Steam Generator (per loop)</u>		
Inlet helium flow, kg/s (lb/h)	157.2 (1,248,100)	169.8 (1,347,816)
Inlet helium temp., °C (°F)	685 (1266)	649 (1,201.1)
Inlet helium press., MPa (psia)	6.31 (915.8)	6.31 (915.7)
Outlet helium temp., °C (°F)	255 (490.6)	258 (496.7)
SG helium pressure drop, kPa (psi)	25.5 (3.7)	26.1 (3.78)
Inlet feedwater flow, kg/s (lb/h)	137.2 (1,089,736)	134.7 (1,069,023)
Inlet feedwater temp., °C (°F)	193 (380)	193 (380)
Inlet feedwater press., MPa (psia)	20.7 (3000)	20.6 (2,987.2)
Outlet steam temp., °C (°F)	540 (1005)	540 (1005)
Outlet steam press., MPa (psia)	17.3 (2515)	17.3 (2,511.3)

(a) Middle of Life Expected Performance.

(b) For information only. This is not applicable to the steam generator subsystem.

(c) These uncertainties are based on $\pm 2\sigma$ standard deviation estimates.

(d) Uncertainties at 25% Feedwater Flow are not presented here because they do not effect the material test data temperature range.

Table 4.2-4

Table 4.2-4 (Continued)

Point 2	Point 3	Point 4	Point 6
+2%	+2%	-2%	-2%
-9%	-9%	+9%	+9%
+10.7	-10.7%	-10.7%	+10.7%
357	357	343	343
3.48	2.6	2.25	3.48
359.4	358.6	344.3	345.5
167.8 (1,332,163)	158.1 (1,254,145)	146.9 (1,165,887)	158.4 (1,257,612)
677 (1,250.3)	697 (1,286.0)	697 (1,286)	671 (1,240.8)
6.31 (915.7)	6.31 (916.6)	6.33 (917.8)	6.31 (914.8)
265 (508.8)	260 (499.6)	246 (474.4)	252 (485.8)
26 (3.77)	23.1 (3.37)	19.8 (2.88)	28.4 (4.12)
140.1 (1,112,343)	139.8 (1,110,633)	134.3 (1,065,581)	134.7 (1,069,182)
193 (380)	193 (380)	193 (380)	193 (380)
21.0 (3,037.2)	20.9 (3,029.7)	20.4 (2,960.7)	20.5 (2,974.6)
540 (1,005)	540 (1,005)	540 (1,005)	540 (1,005)
17.4 (2,519)	17.4 (2,518)	17.3 (2,511)	17.3 (2,511)

TABLE 4.2-5
FEEDWATER CHEMISTRY REQUIREMENTS
(Maximum values at economizer inlet)

Steady-State	1	2	3
	Prior to Recirculating Water Through Steam Generators	Before Start of Boiling	Normal Operation
Total solids, ppb	250	100	65
Suspended solids, ppb	(a)	(a)	15
Silica (SiO ₂), ppb	30	30	15
Total iron, ppb	50	50	10
Total copper, ppb	20	20	3
Sodium, ppb	(a)	(a)	5
Chloride, ppb	20	10	5
Conductivity at 25°C µohm/cm	0.65	0.25	0.10
Dissolved oxygen, ppb	10	10	5
pH at 25°C	9.2 min(b)	9.2 min	9.1 min
Ammonia (NH ₃)	As required	As required	As required
Residual hydrazine (N ₂ H ₄), ppb	10-20(b)	10-20	10-20

(a) Values need not be determined at the indicated stage in the feedwater cleanup cycle.

(b) Hydrazine concentration raised to 200 ppm and pH to 10.0 minimum if wet layup of equipment is required.

Fig. 4.2-1. Sectional view of steam generator subsystem

Fig. 4.2-2. Section of SGSS taken at steam nozzle CL
(Section AA from Fig. 4.2-1)

Fig. 4.2-3a. MHTGR Integrated approach functional analysis
 Goal 1 -- Maintain plant operation
 Function 1.1.2 -- Maintain energy transfer

Fig. 4.2-3b. MHTGR Integrated approach functional analysis
 Goal 1 -- Maintain plant operation
 Function 2.1.2 -- Maintain energy transfer

Fig. 4.2-4. Steam generator subsystem operational uncertainty envelope around MOL expected conditions at 100% feedwater flow

Fig. 4.2-5. Carbon contents of 2-1/4 Cr-1 Mo steel exposed to high-temperature gas-cooled reactor helium

Fig. 4.2-6. Fatigue crack growth rate (da/dn) versus ΔK for annealed 2.25 Cr-1 Mo steel plate at 24° and 260°C. A sinusoidal wave form at 10 Hz and an R ratio of 0.1 were used.

Fig. 4.2-7. Creep crack growth rate versus stress intensity, K , of annealed 2.25 Cr-1 Mo steel plate at 538 °C compared to that for two other Cr-Mo steels (reproduced from Refs. 22 and 23).

Fig. 4.2-8. Creep crack growth rate (CCGR) versus transient value of CCGR, C_t , of annealed 2.25 Cr-1 Mo steel plate at 538 °C compared to a Cr-Mo-V steel (reproduced from Ref. 24).

Fig. 4.2-9. Code Case N47 creep-fatigue damage envelope

Fig. 4.2-10. Proposed creep-fatigue damage envelope

4.3 Reactor Vessel

The Reactor Vessel (RV) is a component of the Vessels and Duct Subsystem which belongs to the Vessel System (VS).

4.3.1 Design Description

The vessel system is composed of multiple, welded steel vessels to contain the primary coolant. The primary coolant pressure boundary is formed by separate reactor vessel and steam generator vessel connected by a concentric cross duct. The VS houses and supports the subsystems and components of the Reactor System, the Heat Transport System, and the Shutdown Cooling System. These components are arranged as shown in Fig. 2-1, so that the vessels and cross duct are operated at or dominated by the helium cold leg and isolated from the helium hot leg.

The reactor vessel supports the Reactor System at the bottom and provides radial support to the core barrel. A stagnant helium gap separates the core barrel from the reactor vessel. Cold leg helium is directed to the core entrance by channels welded to the core barrel outer surface. In this manner the reactor temperature is dominated by the cold leg helium. Furthermore, the reactor vessel is uninsulated in order to promote removal of core residual and decay heat in the event that both the main loop cooling and the Shutdown Cooling System are unavailable. The rejection of heat by the Reactor Vessel to the Reactor Cavity Cooling System during normal operation results in vessel temperature considerably below the helium cold leg.

In order to minimize development cost, the reactor vessel design has been developed using proven light-water reactor (LWR) vessel and pipe technology. The construction material is primarily manganese-molybdenum, SA533, Grade B, Class 1 steel. Figure 4.3-1 shows the reactor vessel design and its principal dimensions. The materials used for the principal parts of the RV are listed in Table 4.3-1.

The reactor vessel is composed of several shell courses welded together. Each course is composed of three or four plates bent and welded to form the cylindrical shell. The upper and lower heads are also formed from shaped plates welded together. Flanges are welded to one end of the cylindrical shell and to the upper head. The upper head is then bolted to the shell utilizing O-rings as seals to form the leak-tight pressure boundary. At the other end of the cylindrical shell, a forced core support ring is welded on to the shell. The lower head is then welded directly to the core support ring.

The upper head has nozzles and penetrations for control rod drives (24) and fuel handling and control rod drives (6), instrumentation, and the Helium Purification Subsystem (2). Extending above the upper head are 18 cylindrical housings for the control rod drives. Twelve of the housings contain two control rod drives each. Six of the housings each contain one additional control rod drive; however, these six housings also provide access to the core for the fuel handling equipment. Each of these housings is sealed with an end cap. There are three penetrations of each housing which provide access to the control rod drives for electrical power, instrumentation, and control. The lower portion of each housing is also cooled on the outside by cooling coils. These housings are provided with support for horizontal seismic loads by a restraining plate. This plate is supported by a large diameter cylinder which is attached directly to the vessel upper head.

On the interior of the reactor vessel, 48 lugs are provided for the core barrel seismic restraints. The lugs are welded to the inner surface of the vessel wall at four elevations. The lugs keep the core barrel in place during seismic events. The lugs by design do not interfere with the thermal expansion of the core barrel or the reactor vessel.

The lower head has penetrations for the shutdown cooling system, in-core flux monitoring units (5), and startup detectors (3). The reactor vessel connects to the cross duct by a nozzle.

In order to provide high reliability, in-service inspection of the steel pressure vessel is employed. All vessel and vessel penetration welds are periodically inspected in accordance with Section XI of the ASME Code.

Due to neutron irradiation the reactor vessel material will experience a rise in the temperature at which the steel changes from a ductile to a brittle material. A material surveillance program is provided to continuously monitor the vessel material, the weldment, and the near-affected zone.

4.3.2 Relevant Functions

The design requirements and assumptions discussed in Section 4.3.3 results from the need to satisfy the functions listed in Table 4.3-2 as they relate to the reactor vessel. The relationship of these functions with the functional analysis tree is shown in Figs. 4.3-2 and 4.3-3.

The reactor vessel part of the Vessel System supports the primary function to contain the primary coolant inventory (F1.1.2.2.2.1). This is accomplished largely by maintaining the primary coolant boundary integrity (F1.1.2.2.2.1.1). To satisfy this function requires that the subfunctions to maintain material strength (F1.1.2.2.2.1.1.1) and to control temperatures (F1.1.2.2.2.1.1.2) be satisfied. The maintenance of material strength is satisfied by controlling material degradation (F1.1.2.2.2.1.1.1.1) and monitoring material degradation (F1.1.2.2.2.1.1.1.2).

4.3.3 Design Requirements/Assumptions

This section lists requirements which must be satisfied to adequately insure that the reactor vessel contains the primary coolant inventory (F1.1.2.2.2.1). These requirements include design and operating conditions, environmental conditions, and standards for ensuring that the reactor vessel performs satisfactorily. Also included are the assumptions which the designer judged to be prudent in order to show compliance with these requirements.

<u>Reference</u>	<u>Requirement</u>
1	R1 Design as a permanent installation free of maintenance during life of the plant
1	R1.1 The operational life shall be 40 years
1	R1.2 Availability allocation for components designed to maintain ≥ 7008 effective power hr/yr average operating time
2	R11 Accommodate temperature, pressure, and flow of the primary coolant
	R11.1 Accommodate operating temperatures at rated power between: 193°C and 216°C (380° and 420°F) at vessel beltline; 177°C and 199°C (350° and 390°F) at closurehead.
	R11.2 Accommodate pressure differential of 6.27 MPa (910 psig) at rated power, 7.79 MPa (1130 psig) maximum vessel pressure during abnormal events.
1	R12 Accommodate plant duty cycles events shown in Table 4.1-3.
1, 6, 7	R12.1 Accommodate OBE loads with a maximum horizontal ground acceleration of 0.15 g
1	R12.2 Accommodate SSE loads with a maximum horizontal ground acceleration of 0.30 g

<u>Reference</u>	<u>Requirement</u>
3a, 3b	R12.3 Withstand pressure and temperatures predicted for reactor vessel during events shown in Tables 4.3-3.
4	R13 Compatible with primary coolant
5	R13.1 The primary coolant is helium with chemical impurities as specified in Table 4.1-4
	R16 Accommodate design acoustic loads
8	R16.1 At 100% power, the maximum sound pressure level (SPL) in the annulus between hot duct and cross duct is <160 dB
	R17 Accommodate design irradiation levels
9, 10	R17.1 The design neutron fluence to the reactor vessel is:

Neutron energy level	Fluence n/cm ² x 10 ¹⁷		
	Belt Line	Upper Closure	Head
E > 0.9 MeV	0.057	0.31	1.3*
0.1 MeV < E ≤ 0.9 MeV	0.40	1.7	2.7
3.05 eV < E ≤ 0.1 MeV	0.96	5.5	9.6
E ≤ 3.05 eV	0.027	2.2	19.1
Total	1.4	9.7	32.7*

* Fluence at center of control rod drive housing penetration of RV head; includes the effect of neutron streaming.

<u>Reference</u>	<u>Requirement</u>
11	R18 Limit primary coolant leakage to less than 10% per year of primary coolant inventory

Trade studies have been performed to ensure that the reactor vessel satisfies the above requirements. The trade studies rely on the following assumptions which need to be verified before the reactor vessel design can be qualified.

<u>References</u>	<u>Assumption</u>
12 thru 16	The LWR data base for NDTT shift is applicable to the MHTGR RV neutron fluence spectrum. The LWR data base for NDTT shift can be extrapolated to the lower operating temperatures of the MHTGR RV.
12, 17, 18, 19	Reactor vessel creep will be insignificant during exposure at elevator temperatures postulated for Service Level C and D loadings.

The following subsection discusses why it is necessary to make each assumption and how each assumption affects the reactor vessel. Also, a discussion is given of the current data base.

4.3.3.1 Irradiation Effects on the Reactor Vessel Material. This section documents the methodology used for projecting the nil-ductility transition temperature shifts for the MHTGR reactor vessel beltline and closure head. A review of reactor vessel irradiation test program reports was performed and data applicable to the HTGR irradiation conditions are discussed.

4.3.3.1.1. Basis for Shift Projections. The calculative procedures of the proposed revision (Ref. 21) to Regulatory Guide 1.99, "Radiation Damage to

Reactor Vessel Materials," are being used as the basis for predicting transition temperature shift in the MHTGR vessel beltline and closure head. This revision represents a combination, by P. N. Randall of the NRC, of the work performed by G. Guthrie of the Hanford Engineering Development Lab (Ref. 22) and G. Odette of the University of California at Santa Barbara (Ref. 23). In their work, Guthrie and Odette used only Light Water Reactor (LWR) surveillance capsule data. Guthrie accumulated his own data base as reported in Ref. 22, while Odette used the EPRI Surveillance Data Base (Ref. 24). The data bases contained virtually the same data and had the following characteristics: (1) irradiation temperatures in the range 277°C to 304°C (530° to 580°F), (2) fast fluences ($E > 1$ MeV) in the range 10^{17} to 10^{20} n/cm², (3) fast fluxes of 10^{10} to 10^{11} n/cm²/s, and (4) a spectrum for fast ($E > 1$ MeV), epithermal ($0.4 \text{ eV} < E < 1 \text{ MeV}$), and thermal ($E < 0.4 \text{ eV}$) of 10%, 25%, and 65%, respectively.

The irradiation environment of the MHTGR vessel beltline and closure head differ from the LWR irradiation data base in the operating temperature, neutron energy spectrum, and neutron flux levels. How these factors were accounted for in the shift projections is discussed below:

- A. Irradiation Temperature. The projected operating temperatures for the MHTGR vessel beltline and closure head are approximately 204°C and 177°C (400°F and 350°F), respectively. This is well below the LWR data base range of 277°C to 304°C (530° to 580°F). The effect is to increase the shift at a given fluence, because at a lower temperature, the mobility of the radiation-induced defects is reduced and annealing of the accumulated damage is lessened, thereby increasing shift. This is illustrated in Fig. 4.3-7 (Ref. 25) for A302B steel, which is equivalent to the recommended A533B steel except for nickel content. Because the damage tends to saturate at lower temperatures, the shifts at 177°C and 204°C (350° and 400°F) are approximately equivalent, but are about twice the shift at 288°C (550°F). An approximate factor of 2 is also seen in the Fig. 4.3-8 data, from Ref. 26, in the $5 - 10 \times 10^{19}$ n/cm² fluence-range from

irradiation of a low copper A533B plate at 149°C to 288°C (300° and 550°F). The larger ratios apparent at lower fluences in Fig. 4.3-8 are not representative and result from the different initial rates of embrittlement occurring in the very high fast flux field (3×10^{13} n/cm²/s, $E > 1$ MeV) of the test reactor (compared to 10^8 n/cm²/s, $E > 1$ MeV, in the MHTGR head). Based on this data, a conservative temperature correction factor of 2.25 was applied to the Regulatory Guide 1.99 shift predictions to correct for the irradiation temperature differences.

- B. Neutron Energy Spectrum. The neutron energy spectrum of the MHTGR differs from the LWR spectrum due to the roughly ten times more epithermal neutrons for every fast neutron hitting the MHTGR vessel than in a LWR vessel. Because an epithermal neutron also causes damage to the metal, a higher shift will result in a MHTGR vessel than in a LWR vessel for the same fast fluence. Data from Ref. 27, in which A302B plate material was irradiated in light water and graphite moderated test reactors at approximately 121°C (250°F), illustrate this point. At reported fast fluences of 0.5 and 0.75×10^{19} n/cm² ($E > 1$ MeV) the light water reactor shifts were 94°C and 114°C (170°F and 205°F) compared to 114°C and 133°C (205°F and 240°F) shifts for the graphite reactor. From the spectrum data given for the graphite test reactor, it appears the MHTGR intermediate to fast fluence ratio will be higher by a factor of 3, which should result in even higher shifts than in the graphite test reactor for the same fast fluence.

To account for the differences in neutron energy spectrum between the MHTGR and LWR reactors, the total neutron fluence of the MHTGR is converted into an "equivalent fast fluence" by the use of weighting factors, based on an average of those used in Ref. 20. The resultant fast fluence is then used in the Regulatory Guide 1.99 equation to calculate shift.

- c. Neutron Flux. The fast neutron flux in the MHTGR is in the range of 10^7 to 10^8 n/cm²/s compared to 10^{10} to 10^{11} for the LWR irradiated data base. There is no test data for fluxes in the HTGR range due to the prohibitive irradiation time necessary to accumulate significant fluences (e.g., 317 yr for 10^{18} n/cm², $E > 1$ MeV).

Due to the lack of data, it is unclear what effect, if any, the lower fluxes in the MHTGR will have on the damage mechanisms responsible for shift.

4.4.3.1.2. Projected Transition Temperature Shifts. The latest fluence numbers for the MHTGR vessel beltline and closure head from Refs. 9 and 10 are based on a 2-D analysis of the upper portion of the HTGR by J. Bucholz of the Oak Ridge national laboratory. The Ref. 10 analysis specifically accounted for neutron streaming up the RSS guide tubes into the head region. However, the interior of the RSS guide tube housings on the top of the head was modelled as a void; whereas, components will be present in the actual case. These components will scatter back to the head some of the neutrons currently modelled as streaming out of the top of the head through the penetrations; thereby, increasing the calculated fluence to the vessel head. To account for this, it is being assumed for conservatism that the maximum fluence in the head will be the peak neutron fluence streaming through the RSS penetration. These fluence numbers for the vessel beltline and head are converted into "equivalent fast fluence" numbers as discussed in Section 4.3.3.1.1 in Table 4.3-4.

The projected transition temperature shifts using the fluences from Table 4.3-4 and the Regulatory Guide 1.99 equation are shown in Table 4.3-5 for the beltline and head.

The Table 4.3-5 results indicate that due to the low fluence in the beltline region, the predicted transition temperature shift of 7°C (12°F) is rela-

tively small. In the head region, however, the neutron streaming can result in shifts of up to 43°C (77°F).

It must be recognized that the Table 4.3-5 numbers represent an extrapolation from the LWR data base to MHTGR irradiation conditions for which no prototypical irradiation data exist. Therefore, a confirmatory irradiation test duplicating anticipated conditions as closely as possible is considered necessary to establish a more accurate prediction of the transition temperature shifts for the MHTGR vessel.

4.4.3.2 Properties of SA533B at Elevated Temperature. Section III of the ASME Code provides ultimate strength and yield data for SA533B to a temperature of 1000°F. Section III provides allowable stress intensity for Service Level A, B, C, and D loading up to a temperature of 371°C (700°F) derived from these data. Since time-dependent effects (creep) must be considered as temperature increases, allowable stress intensities are not derived for the elevated temperature regime (>371°C). Therefore, the derivation and use of allowables for Service Level C and D loadings is not permitted in this elevated temperature range.

Limited creep strain data are available (Ref. 28) for this material for temperatures up to 649°C (1200°F) and for exposures of approximately 48 h. A constitutive equation has been fitted to the data. Elastic-plastic properties are available (Ref. 28) for temperatures up to 649°C (1200°F) and for strains up to 5%. Analytical relationships were developed for the stress versus plastic strain.

The evaluation of the reactor vessel indicates that the RV temperature exceeds the temperature for which ASME Code Section III allowable stress intensities are available [371°C (700°F)] only for limited Service Level C and D events. Estimates of creep strains (Ref. 17) indicate that they are indeed negligibly small. Consequently, the methods and procedures for deriving allowable stress intensities for Service Level C and D conditions is

considered valid and employed in judging the reactor vessel structural performance.

Data are needed to support a Section III Code inquiry to establish allowable stress intensities for SA533 Grade B, Class 1 and its weldment for special event (Service Levels C and D) loadings at elevated temperatures. Creep strain rate data are needed to demonstrate negligible creep for the anticipated service conditions and to permit the extension of the Section III rules for special event allowables to the 311°C to 538°C (700°F to 1000°F) temperature range. Creep tests on SA533B, Class 1 and its Weldment could be run with a stress level of 0.95 yield over the temperature range 371°C to 538°C (700°F to 1000°F) for periods up to 1000 h. Uniaxial tensile data would also be generated for this temperature range. If significant creep strains are encountered, then isochronous stress-strain curves would also be required for the anticipated service conditions and duration. Special event allowable stress intensities would then be established.

4.3.4 Technology Requirements

This section gives the details of the data need associated with each of the assumptions discussed in Section 4.3.3. Quality assurance must satisfy the requirements of 10CFR50, Appendix B. It is intended that the technologist relies on techniques previously used to determine design properties given in the ASME B&PV Code. These techniques are based on engineering principles together with statistics.

4.3.4.1 Irradiation Effects on the Reactor Vessel Material. Sufficient data are needed to provide the designer a means of conservatively calculating a NDTT shift of less than 44°C (80°F) through a correlation with equivalent fast fluence.

The specified service condition for the reactor vessel beltline are as follows:

Normal Operation

- o 6.38 MPa (925 psia)
- o 280,000 (40 yr at 80% availability)

RV Beltline

- o 193°C (380°F) to 216°C (420°F) metal temperatures
- o 1.4×10^{17} n/cm² (total fluence). See R17.1 for fluence distribution versus energy level.

RV Closure Head

- o 177°C (350°F) to 199°C (390°F) metal temperatures
- o 3.3×10^{18} n/cm² (total fluence). See R17.1 for fluence distribution versus energy level.

Off-Normal Operation

- o Service Level B metal temperature <371°C (700°F)
- o Service Level C
Pressurized Core Conduction Cooldown Event
Maximum metal temperature 399°C (750°F)
Time duration above 371°C (700°F) 400 h
- o Service Level D
Depressurized Core Conduction Cooldown Event
Maximum metal temperature 473°C (884°F)
Time duration above 371°C (700°F) 400 h

4.3.4.2 Properties of SA533B at Elevated Temperatures. Data are needed to establish allowable stress intensities for SA533 Grade B, Class 1 and its weldment for temperatures from 371°C to 482°C (700°F to 900°F) and load durations up to 400 h.

The specified service conditions for the reactor vessel are as follows:

Normal Operation

- o Metal temperature <371°C (700°F)
- o 280,000 h
- o Service Environment

Helium with 0.5 ppmv H₂O, 3 ppmv CO + CO₂, 3 ppmv H₂, 0.1 ppmv CH₄, and 2 ppmv N₂ on the internal surface.

Air in Reactor Building on external surface.

Off-Normal Operation

- o Service Level B metal temperature <371°C (700°F)
- o Service Level C
 - Pressurized Conduction Cooldown Event
 - Maximum metal temperature 407°C (764°F)
 - Time duration above 371°C (700°F) Ca. 250 h
 - Maximum hoop membrane stress 170 MPa (25 ksi)

o Service Level D

Pressurized Conduction Cooldown Event

Maximum metal temperature 411°C (771°F)

Time duration above 371°C (700°F) Ca. 250 h

Maximum hoop membrane stress 170 MPa (25 ksi)

Depressurized Conduction Cooldown Event

Maximum metal temperature 471°C (880°F)

Time duration above 371°C (700°F) Ca. 400 h

Shell axial stress <7 MPa (1 ksi)

4.3.5 Schedule Requirements

Interim results on all data defined in Section 4.3.4 are needed, as they become available, by the end of the Conceptual Design phase (i.e., 10/87 based on the schedule shown in Fig. 4.1-17). Final results on all the data are needed by the end of the Preliminary Design phase (i.e., 10/89 based on the schedule shown in Fig. 4.1-17).

4.3.6 References

1. GCRA 86-002, Rev. 1, "Utility/User Design Requirements for the Modular High Temperature Gas-Cooled Reactor Plant," Doc. No. PC-000217/1, June 16, 1986.
2. "Conceptual Design Data for MHTGR Plant," HTGR-86-023, GA Doc. 907807/7, April 1986.
- 3a. Letter, G. Baccaglini (GA) to S. A. Caspersson (C-E), "Transmittal of SRDC #1 and #11," GEN-GA/CE:072:GB:86, April 15, 1986.
- 3b. Letter, G. Baccaglini (GA) to F. G. Small (C-E), "Transmittal of SRDC No. 1 (Rev.), 2, 4, and 10; DBE No. 3, 7, and 8; and AOO No. 1 and 3," GEN-GA/CE:088:GB:86, April 30, 1986.
4. Baccaglini, G., R. Barker, and H. S. Chung, "HTGR SC/C Lead Plant Moisture Ingress Evaluation," GA Doc. 906519/1, September 30, 1983.
5. Baccaglini, G., "Plant Specification - Primary Coolant Chemistry - 2240 MW(t) HTGR-SC/C Plant," GA Doc. 906385/2, July 30, 1984.

6. Olsen, B. E., and I. T. Almajan, "Soil Parameter Study for Seismic Analysis of MHTGR Plant," Doc. 908767/0, March 18, 1986.
7. Sweeney, T. M., to L. D. Mears, "Transmittal of Seismic Floor Response Spectra, Linearly Varying Soil Site GCRA/HTGR Project: Job No. 14508 BL-927," Bechtel National, Inc. letter GCRA/BEC/81-202, April 29, 1986.
8. Baccaglioni, G., "Heat Transport System Design Report," GA Doc. No. 908759/0, HTGR-86-122, November 17, 1986.
9. Bucholz, J. a. and R. L. Childs, "Pre-conceptual Shielding Design Analysis for the Lower Portion of the MHTGR Reactor System Incorporation of 350 MW(t) Annular Prismatic Core," HTGR-86-060, ORNL/GCR-86/1, June 1986, Pages 19, 26, 152, C-10.
10. Bucholz, J. A. and R. L. Childs, "Pre-conceptual shielding Analyses for the Upper Portion of the MHTGR Reactor System Incorporating a 350 MW(t) Annular Prismatic Core," HTGR-86-112, ORNL/GCR-86/4, September 1986.
11. Baccaglioni, G., to A. Shenoy, "Loss of Primary Coolant Inventory Requirement - SBS 4 x 250 MW(t) Plant," memo SD&PD:GMB:026:85, March 13, 1985.
12. Letter, S. A. Caspersson (C-E) to A. J. Neylan (GA), transmitting: "Interim Report on the Applicability of LWR Steel Pressure Vessel Technology to HTGR-MRS Concepts," CE-ADD-84-311/14001, September 1984.
13. Letter, S. A. Caspersson (C-E) to A. J. Neylan (GA), transmitting: "Assessment of Neutron Embrittlement Concerns of Using a Steel Pressure Vessel in the HTGR Modular Reactor System," CE-ADD-84-348/14001, November 1984.
14. Letter, S. A. Caspersson (C-E) to A. J. Neylan (GA), "Applicability of LWR Steel Pressure Vessel Technology to the HTGR-MRS Concept," CE-ADD-85-409/14001, September 30, 1986.
15. Letter, S. A. Caspersson (C-E) to A. J. Neylan (GA), "Additional Information Regarding Temperature and Fluence Appropriate to Use of a Steel Reactor Vessel for the HTGR-MRS," CE-ADD-85-355/14001, November 1985.
16. Letter, F. G. Small (C-E) to A. J. Neylan (GA), transmitting: "Updated Projected Transition Temperature Shifts for the HTGR-MRS Reactor Vessel and Head," CE-ADD-86-595/14001, June 12, 1986.
17. Letter, S. A. Caspersson (C-E) to A. J. Neylan (GA), transmitting: "Stress Evaluation of Modular HTGR Vertical In-Line Steel Vessel Shell Based on Available GA Data," CE-ADD-84-347/14001, December 18, 1984.

18. Letter, M. Basol (C-E) to A. J. Neylan (GA), "250 MW(t) HTGR-MRS SBS Steel Vessel Concept-Metal Temperatures Exceeding Design Temperatures and 700°F for SA533 Grade B, Class 1, Material," CE-ASG-85-088, May 10, 1986.
19. Letter, S. A. Caspersson (C-E) to A. J. Neylan (GA), "Clarification and Expansion of C-E Recommendations Regarding HTGR-MRS Steel Reactor Vessel Temperatures Above 700°F," CE-ADD-85-312/14001, September 30, 1985.
20. "MRS Core Nuclear Design and Cost Summary Report," GA Document 906938, May 27, 1983.
21. "Proposed Revision 2 to Regulatory Guide 1.99, Radiation Damage to Reactor Vessel Materials," Office of Nuclear Regulatory Research, Task ME 305-4, February 1986.
22. "LWR Pressure Vessel Surveillance Dosimetry Improvement Program, Quarterly Progress Report, April 1983 to June 1983," NUREG/CR-3391 Vol. 2, April 1984.
23. Odette, G. R., et. al., "Physically Based Regression Correlations of Embrittlement Data from Reactor Pressure Vessel Surveillance Programs," EPRI NP-3319, January 1984.
24. Oldfield, W. R., et. al., "Irradiated Nuclear Pressure Vessel Steel Data Base," EPRI NP-2428, June 1982.
25. Steele, L. E., Neutron Irradiation Embrittlement of Reactor Pressure Vessel Steels, International Atomic Energy Agency Technical Report No. 163, 1975, Page 123.
26. Pachur, D., "Radiation Annealing Mechanisms of Low-Alloy Reactor Pressure Vessel Steels Dependent on Irradiation Temperature and Neutron Fluence," Nuclear Technology, 59, December 1982, pp. 463-475.
27. Serpan, C. Z., and Steel, L. E., "Damaging Neutron Exposure Criteria for Evaluating the Embrittlement of Reactor Pressure Vessel Steels in Different Neutron Spectra," NRL Report 6415, July 28, 1966.
28. Reddy, G. B. and Ayres, D. J., "High Temperature Elastic-Plastic and Creep Properties for SA533 Grade B, Class 1, and SA508 Materials," EPRI NP-2763, December 1982.
29. "Preliminary Safety Information Document for the Modular HTGR Plant," DOE Report HTGR-86-024, September 1986.

TABLE 4.3-1
PRINCIPAL MATERIALS FOR THE REACTOR VESSEL

Parts	Material
Shell and heads	SA533 Grade B, Class 1, Mn-Mo Quenched and tempered alloy steel plate Additional requirements for radiation affected areas in weight % are: Cu (MAX) - 0.10 P (MAX) - 0.012 S (MAX) - 0.022 V (MAX) - 0.010
Nozzles and flanges	SA508, Class 2 Quenched and tempered alloy steel forgings
O-rings	Inconel 718 silver plated
Weldments	
o Machine welding	Mn-Mo, TSAA and SAA-SMA 3/16 in. MIL-B-4 3/16 in. low Cu-P-B-4 for radiation Affected areas
o Manual welding	Mn-Mo, SMA, E8018G

TABLE 4.3-2
RELEVANT FUNCTIONS OF THE REACTOR VESSEL

Function Number	Function ^(a)
1.1.2.2.2.1	Contain Primary Coolant Inventory
1.1.2.2.2.1.1	Maintain Primary Coolant Boundary Integrity
1.1.2.2.2.1.1.1	Maintain Material Strength
1.1.2.2.2.1.1.2	Control Temperature

(a) These functions are for Goal 1 - Maintain Safe Plant Operation, State 1 - Maintain Safe Energy Production. These functions are also relevant for States 2 - Maintain Safe Plant Shutdown, State 3 - Maintain Safe Plant Refueling, and State 4 - Maintain Safe Plant Startup/Shutdown. The associated functions for Goal 2 - Maintain Plant Protection are also applicable. These functions read: "Protect the capability to ...".

TABLE 4.3-3
REACTOR VESSEL DESIGN TEMPERATURES FOR MHTGR^(a)

Loading Condition ^(b)	No. of Cycles (N)	Reactor Vessel Temperature [°C (°F)]	Duration (h)	Service Level
100% power		216 (420) ^(c)	280,000	A
50% steam flow		212 (413)	(e)	
25% steam flow	21,800 ^(d)	212 (413)	(e)	
Refueling shutdown	455	44 (111)		
Reactor trip from 100% power (A00-1)	131	216 (-420)	0.5/trip	B
Pressurized conduction cool-down (SRDC-1 and SRDC-2)	1	399 (750)	Peak	C
	1	372 (500)	150 (f)	
Pressurized conduction cool-down with rod withdrawal (SRDC-4)	1	399 (750)	Peak	D
		>372 (700)	200 (g)	
Depressurized conduction cooldown (SRDC-11)	1	460 (860)	Peak	D
		>427 (800)	200	
		>399 (750)	300 (h)	
		>372 (700)	400	

(a) Only the events resulting in boundary conditions are identified. Events not identified result in conditions within the range covered by the boundary events.

(b) Reference 29.

(c) Values given are maximum values. See requirement 11.1 for range of temperatures.

(d) The cycles indicated are from 100% power to indicated condition.

(e) The duration is included in 280,000 h at 100% power.

(f) See Fig. 4.3-4 for detailed time history.

(g) See Fig. 4.3-5 for detailed time history.

(h) See Fig. 4.3-6 for detailed time history.

TABLE 4.3-4
EQUIVALENT FAST FLUENCES FOR MHTGR REACTOR VESSEL BELTLINE AND CLOSURE HEAD

	Group ¹	Fluence ² (n/cm ²)	Damage ³ Factor	EFF ⁴
Beltline	A	5.7×10^{15}	1.0	5.7×10^{15}
	B	4.0×10^{16}	0.5	2.0×10^{16}
	C	9.6×10^{16}	0.0025	2.4×10^{14}
	D	2.7×10^{15}	0.07	1.9×10^{14}
				2.6×10^{16}
Head (118.95 cm axial reflector, and shield cans)	A	1.3×10^{17}	1.0	1.3×10^{17}
	B	2.7×10^{17}	0.5	1.35×10^{17}
	C	9.6×10^{17}	0.0025	2.4×10^{15}
	D	1.9×10^{18}	0.07	1.33×10^{17}
				4.0×10^{17}

¹Group A: ($E > 0.9$ MeV), Group B: ($0.1 \text{ MeV} < E \leq 0.9 \text{ MeV}$),
Group C: ($3.05 \text{ eV} < E \leq 0.1 \text{ MeV}$), Group D: ($E \leq 3.05 \text{ eV}$).

²From Ref. 10, Case 3, Sets 11 and 17.

³Based on average for groups from Ref. 20, Table 5.2.4.

⁴Equivalent Fast Fluence.

TABLE 4.3-5
PROJECTED TRANSITION TEMPERATURE SHIFTS FOR THE MHTGR VESSEL BELTLINE AND HEAD

	EFF (n/cm ²)	R.G. 1.99		Temp. Corr.	
		Rev. 2 Shift		°C (°F)	
		°C (°F)		(x 2.25)	
		Plate	Weld	Plate	Weld
Beltline	2.6×10^{16}	3 (5)	3 (5)	7 (12)	7 (12)
Head	4.0×10^{17}	19 (34)	19 (34)	43 (77)	43 (77)

Fig. 4.3-1. Reactor vessel configuration

Fig. 4.3-2. MHTGR integrated approach functional analysis.
 Goal 1: maintain plant operation.
 Function 1.1.2: maintain energy transfer.

Fig. 4.3-3. MHTGR integrated approach functional analysis.
 Goal 2: maintain plant protection.
 Function 2.1.2: protect the capability to maintain energy transfer.

Fig. 4.3.4a. Primary Helium Pressure During Pressurized Conduction Cooldown Event (SRDC 1)

Fig. 4.3-4b. Peak reactor vessel midwall temperature during a pressurized conduction cooldown event (SRDC 1)

Fig. 4-3.5a. Primary Helium Pressure During a Pressurized Conduction Cooldown with Rod Withdrawal Event (SRDC 4)

Fig. 4.3-5b. Peak reactor vessel midwall temperature during a pressurized conduction cooldown with rod withdrawal event (SRDC 4)

Fig. 4.3-6. Peak midwall temperatures of reactor vessel during a depressurized conduction cooldown event.

Fig. 4.3-7. Effect of irradiation temperature on transition temperature increase for an A302-B Reference steel.

Fig. 4.3-8. Effect of irradiation on low copper A533B material at 300°F and 550°F.

5. BASE TECHNOLOGY DEVELOPMENT PROGRAM

This section of the report describes the base technology programs which have been developed to provide the metals data and information necessary to satisfy the technology requirements specified in Sections 4.1.4 (hot duct), 4.2.4 (steam generator), and 4.3.4 (reactor vessel). A cross referencing of components with the DDNs derived from the technology requirements is as shown in Table 5-1. Also shown in this table are priorities specified by the designers.

Each of the reactor components mentioned above is covered in a separate subsection (Sections 5.1 through 5.3, respectively). Within these subsections, the base technology programs are indexed to and treated by the specific DDNs given in Table 5-1. Each subsection addresses data needed (Section 5.X.1), approach and rationale (Section 5.X.2), test programs (Section 5.X.3), schedule (Section 5.X.4), and cost (Section 5.X.5). Section 5.4 gives an overall summary of cost by fiscal year and discusses options relative to scheduling of the test programs.

The technology requirements for each component state that "sufficient data are required to determine to a 95% confidence . . ." that properties meet or exceed the design values specified. For a pure statistical treatment, this statement would need to be modified to call out a specific confidence level for a specific proportion of the data population (e.g., 95% confidence that 95% of the data . . .). To apply such a requirement to the MHTGR metals technology effort would result in a program with costs and schedule excessive in terms of real needs. Therefore, in the present case, the reference to "95% confidence" is not intended to prescribe that the data be subjected to a pure statistical treatment but, instead, to require that data analysis methods be consistent with those developed for ASME B&PV Code design properties determinations.

TABLE 5-1
CROSS REFERENCING OF MHTGR COMPONENTS
AND RELEVANT DDNS

Component	DDN	Priority
Hot duct	<u>M.10.17.26</u> - Determine Effects of Irradiation on Properties of Alloy 800H	Medium
	<u>M.10.17.28</u> - Determine Effects of Primary Coolant Chemistry and Temperature on Alloy 800H	Medium
Steam Generator	<u>M.21.02.02</u> - Determine Properties of 2.25 Cr-1Mo Base Metal and Weldments	High
	<u>M.21.02.03</u> - Determine Properties of Alloy 800H Base Metal and Weldments	High
Reactor Vessel	<u>M.11.06.01</u> - Nil-Ductility Transition Temperature Shift for Reactor Vessel Material Irradiated at Low Temperatures	High
	<u>M.11.06.02</u> - Determine Properties of SA533B Base Metal and Weldments at Elevated Temperatures	High

5.1 Hot Duct

The base technology programs for the hot duct are responsive to the technology requirements presented in Section 4.1.4 and its associated DDNs (M.10.17.26 and M.10.17.28). Some parts of these programs overlap programs needed for the steam generator (see Section 5.2 and DDN M.21.02.03) and are detailed and costed there because it is a higher priority program.

5.1.1 Data Needed

5.1.1.1 Determine Effects of Irradiation on Properties of Alloy 800H (DDN M.10.17.26). Data are needed to determine the effects of irradiation as detailed in Section 4.1.4.2.

5.1.1.2 Determine Effects of Primary Coolant Chemistry and Temperature on Alloy 800H (DDN M.10.17.28). Data are needed on the effects of exposure to the primary coolant chemistry and temperature (corrosion and thermal aging effects) as detailed in Section 4.1.4.1.

5.1.2 Approach and Rationale for Test Programs

The approach and rationale employed in generating the base technology programs in response to the hot duct technology requirements (Section 4.1.4) are described below in Sections 5.1.2.1 and 5.1.2.2 relating to irradiation effects and corrosion/aging effects, respectively. Test matrices, schedules, and costs are given in later sections (5.1.3 through 5.1.5).

5.1.2.1 Alloy 800H Irradiation Effects (DDN M.10.17.26). The effects of irradiation at temperatures from 1100°F to 1400°F at a total fluence level of 10^{18} n/cm² (equal thermal and epithermal) on the properties of Alloy 800H will be determined to satisfy the technology requirements specified in Section 4.1.4. Plate material (SB-409) representing the thermal barrier cover plates will be tested along with the appropriate weldments and weldmetal (Inconel 82).

Some data are available on the effects of irradiation at 1200°F to 1400°F on Alloy 800 (material with less restrictive composition than Alloy 800H) properties. However, fluences in these tests were high relative to the design fluence and produced damage more severe than is to be expected in the present case. Also, irradiation damage information does not exist for some properties (e.g., high-cycle fatigue).

Two irradiation experiments, one at 1100°F and one at 1400°F, are planned for the Bulk Shielding Reactor (BSR). Only a single irradiation capsule will be required for each temperature. The 10^{18} n/cm² fluence needed will be provided in only a few days even at the low BSR flux. Unirradiated control material testing is provided for in the test matrices associated with DDN M.10.17.28 (Section 5.1.3.2) and DDN M.21.02.03 (Section 5.2.3.2).

Considerations relative to the individual properties listed in Section 4.1.4.2 are given below.

1. Tensile - Tests of irradiated as-received plate, plate-to-plate weldments [containing base metal, heat affected zone (HAZ), and weld zone], and weldmetal will be conducted at room temperature and at 400°F through 1600°F. All tests will be done in air as no effect of the primary coolant environment is expected in these short-term experiments.
2. Low-cycle fatigue - Irradiated plate, weldment, and weldmetal specimens will be tested in the design primary coolant environment (See Table 4.1-4) at the irradiation temperatures. Tests will be conducted at strain ranges from 0.3 to 1.0%. A limited number of additional specimens will be irradiated for replication or for tests at other temperatures if found desirable.
3. Fracture toughness - Charpy V-notch (CVN) specimens of all product forms will be tested to provide a baseline understanding of the

effects of irradiation at 1100°F and 1400°F on fracture resistance. Tests will be performed in air (see 1 above).

K_{Ic} tests will be performed in air at room temperature and at the irradiation temperatures on all product forms.

Fatigue crack growth rates (da/dN) will be measured at the irradiation temperatures on all product forms. Tests will be conducted in the design environment at minimum stress to maximum stress ratios of 0.05 to 0.75 and a frequency of 1 Hz.

Creep crack growth (da/dt) rates will be measured at two stress levels for each irradiation temperature on all product forms. Testing will be done at the irradiation temperature in the design environment.

4. Creep tests will be conducted at the irradiation temperatures in the design environment. The tests on plate will be designed for 5000 h rupture and those on the weldments and weldmetal will be scheduled for rupture in 2000 h.
5. High-cycle fatigue - Tests will be run in the range 5×10^5 to 10^7 cycles at the irradiation temperatures to permit extrapolation to 3×10^{11} cycles. All product forms will be represented and a limited number of additional specimens will be irradiated to permit tests at other temperatures if required.

Specimen designs for all of the tests above (and those described in Sections to follow) will conform to those allowed by ASTM Standards.

5.1.2.2 Alloy 800H Primary Coolant Chemistry and Temperature Effects (DDN M.10.17.28). The effects of thermal aging (1000°F to 1400°F) and expected and design primary coolant compositions (see Table 4.1-4) on the properties of Alloy 800H will be determined to satisfy the technology require-

ments specified in Section 4.1.4. Plate material (SB-409), representing the thermal barrier coverplates, will be used in the testing program along with the appropriate weldments and weld metal (Inconel 82).

Section 4.1.3.1 describes the extensive data base which exists for Alloy 800H and formed the basis for its selection for use in the hot duct. However, data are lacking in a number of critical areas including the effects of exposure in high levels of water vapor (i.e., the design level specified in Table 4.1-4) and the effects of aging on fracture mechanics parameters.

Considerations relative to the individual properties listed in Section 4.1.4.1 are given below.

1. Tensile - Tests will be performed in air at 1000°F to 1600°F and at room temperature on annealed and aged plate material and on weldments and weldmetal. Aging will be done for periods of 10,000 and 20,000 h in the design environment only. (The expected environment should not produce results differing significantly from those obtained in earlier testing.) Aging temperatures will be 1100°F, 1200°F, and 1400°F. The lower two temperatures bound the region in which γ' formation may occur on aging; the highest temperature represents the condition under which the most significant decarburization could occur.
2. Low-cycle fatigue and creep-fatigue interactions - Testing will be conducted on annealed plate, the weldmetal from plate-to-plate weldments, and the heat affected zone (HAZ) of the weldments. Aging of each of these materials will be performed at 1000°F to 1400°F for 10,000 and 20,000 h prior to testing. All of these exposures and the actual testing will be done in the design environment. However, one series of tests will be done in the design environment. However, one series of test on annealed plate will be run in the expected environment since surface effects/conditions can influence fatigue behavior. Test temperatures will range from 400°F to 1400°F with the majority

of tests at 1000°F and above. Strain ranges of 0.3% to 2.0% will be involved. Cyclic stress/strain diagrams will be recorded for the first ten cycles of testing.

The creep-fatigue damage envelope at 1200°F will be determined in a test series involving strain ranges from 0.3% to 1.0%, tension and compression hold modes, and hold times from 0.01 to 0.5 h. Lower temperatures are not included because of the anticipated very low fraction of creep damage; a higher temperature was not selected because of an expected high fraction of creep. Additionally, 1200°F is the temperature of major interest for another MHTGR component (i.e., the steam generator, see later sections). Materials to be tested and aging environments and times are as described for low-cycle fatigue. However, aging will be done only at 1200°F and no tests will be done in the expected environment.

3. Fracture toughness - Charpy V-notch (CVN) specimens of plate, weldment, and the weldment heat affected zone (HAZ) will be tested to provide a baseline understanding of the effects of aging (in design environment at 1100°F through 1400°F) on fracture resistance.

K_{Ic} testing at room temperature and elevated temperatures (1100°F to 1400°F) will examine resistance to the initiation of crack extension in plate, plate weldment, and the HAZ. Tests will be performed on as-received/as-welded materials and on material aged 10,000 and 20,000 h in the design environment. Most of the testing will be done in air because the dimensions of the specimens and the short testing time (minutes) should preclude significant environmental effects. Elevated temperature testing is to be performed to ensure that low-energy tearing phenomenon are not occurring in these materials.

Fatigue crack growth rates (da/dN) will be determined at 600°F through 1400°F on as-received materials (plate, weldment, and HAZ) and on materials aged at 10,000 and 20,000 h in the design

environment. Most of the tests will be done in the design environment, but tests in air will also be done with the aim of understanding the effects of environment on crack growth. All testing will be done at a frequency of 1 Hz and a minimum-to-maximum stress ratio (R) of 0.05.

Creep crack growth (da/dt) tests will be run to describe crack extension resulting from creep mechanisms. As-received and aged (1100°F through 1400°F) materials will be tested in the design environment with some tests on as-received materials in air. Two nominal stress levels will be employed.

4. Creep - All tests will be conducted in the design environment at 1200°F or 1400°F. Plate, plate-to-plate transverse weldments, and weldmetal will be tested before and after aging in the design environment for 10,000 and 20,000 h at 1200°F and 1400°F.
5. High-cycle fatigue - Tests will be conducted from 5×10^5 to 10^8 cycles at temperatures from 1100°F to 1400°F to permit extrapolation to 3×10^{11} cycles. Plate, transverse weldment, and HAZ specimens will be tested before and after aging in the design environment for 10,000 and 20,000 h at 1100°F to 1400°F. Testing will be performed in the design environment except for a few tests on annealed plate in the expected environment.

5.1.3 Test Program

The base technology programs responding to the technology requirements for the hot duct materials (Section 4.1.4) are described below.

5.1.3.1 Alloy 800H Irradiation Effects (DDN M.10.17.26). Test matrices for post-irradiation mechanical property determinations on Alloy 800H are given in Tables 5.1.3.1-1 through 5.1.3.1-8. Unirradiated control specimens are

provided in the test matrices for Alloy 800H hot duct and steam generator materials (Sections 5.1.3.2 and 5.2.3.2, respectively).

5.1.3.2 Alloy 800H Primary Coolant Chemistry and Temperature Effects

(DDN M.10.17.28). Most of the tests required to satisfy this aspect of the technology requirements related to the hot duct are also needed to satisfy steam generator technology requirements. Where such overlap exists, the tests are detailed in Section 5.2 for the higher priority steam generator work. Only those tests unique to the hot duct are covered in the test matrices shown in Tables 5.1.3.2-1 through 5.1.3.2-8.

TABLE 5.1.3.1-1
TENSILE TEST MATRIX FOR IRRADIATED ALLOY 800H^c

Material Condition	Test Temperature, °F								
	75	400	600	800	1000	1100	1200	1400	1600
Irradiated at 1100°F									
Plate	x	x	x	x	x	x	x	x	x
Weldment ^a	x	x	x	x	x	x	x	x	x
Weldmetal ^b	x	x	x	x	x	x	x	x	x
Irradiated at 1400°F									
Plate	x	x	x	x	x	x	x	x	x
Weldment ^a	x	x	x	x	x	x	x	x	x
Weldmetal ^b	x	x	x	x	x	x	x	x	x

^aSpecimens from plate-to-plate weldment; gage section to contain base metal, HAZ, and weld zone.

^bSpecimens from all weldmetal zone of plate-to-plate weldment.

^cx = one test specimen. Therefore, this matrix calls for tensile testing of 54 specimens

TABLE 5.1.3.1-2
 LOW-CYCLE FATIGUE TEST MATRIX FOR IRRADIATED ALLOY 800H^{a,b,f}

Material Condition	Strain Range, %				
	1.0	0.7	0.5	0.4	0.3
Irradiated at 1100°C ^c					
Plate	x	x	x	x	
Weldment ^d	x		x	x	
Weldmetal ^e	x		x	x	
Irradiated at 1400°F ^c					
Plate	x	x	x	x	x
Weldment ^d	x		x		x
Weldmetal ^e	x		x		x

^aAll tests in design environment.

^bA limited number of additional specimens will be irradiated.
 Tests at other temperatures may then be conducted if required.

^cTest temperature = irradiation temperature.

^dSpecimens from plate-to-plate weldment.

^eSpecimens from all weldmetal zone of plate-to-plate weldment.

^fx = one test specimen. Therefore, this matrix calls for
 21 specimens.

TABLE 5.1.3.1-3
SPECIMENS FOR CHARPY V-NOTCH
IMPACT TESTS ON IRRADIATED ALLOY 800H

Material Condition	Number of Specimens ^a
Irradiated at 1100°F	
Plate	10
Weldment ^b	10
Weldmetal ^c	10
Irradiated at 1400°F	
Plate	10
Weldment ^b	10
Weldmetal ^c	10

^aSpecimens tested over temperature range such that fracture behavior transition is defined.

^bSpecimens from plate-to-plate weldment.

^cSpecimens from all weldment zone of plate-to-plate weldment.

TABLE 5.1.3.1-4
 TEST MATRIX FOR FRACTURE TOUGHNESS
 TESTS (K_{IC}) OF IRRADIATED ALLOY 800H^{a, d}

Material Condition	Test Temperature, °F		
	75	1100	1400
Irradiated at 1100°F			
Plate ^a	x	x	
Weldment ^b	x	x	
Weldmetal ^c	x	x	
Irradiated at 1400°F			
Plate	x		x
Weldment ^b	x		x
Weldmetal ^c	x		x

^aAll tests in air; replicate tests for each material/temperature combination.

^bSpecimens from plate-to-plate weldment.

^cSpecimens from weldment zone of plate weldment.

^d_x = two test specimens. Therefore, this matrix calls for 24 specimens.

TABLE 5.1.3.1-5
TEST MATRIX FOR FATIGUE CRACK GROWTH OF IRRADIATED ALLOY 800H^{a,d}

Material Condition	<u>Temperature, °F</u>		<u>R-ratio</u>		
	1100	1400	0.05	0.3	0.75
Irradiated at 1100°F					
Plate	x		x	x	x
Weldment ^b	x		x		x
Weldmetal ^c	x		x		x
Irradiated at 1400°F					
Plate		x	x	x	x
Weldment ^b		x	x		x
Weldmetal ^c		x	x		x

^aTests conducted in design environment; replicate tests at each material/temperature/R-ratio combination.

^bSpecimens from plate-to-plate weldment.

^cSpecimens from all weldmetal zone of plate-to-plate weldment.

^dx = two test specimens. Therefore, this matrix calls for 28 specimens.

TABLE 5.1.3.1-6
 TEST MATRIX FOR CREEP CRACK GROWTH TEST OF
 IRRADIATED ALLOY 800H^{a, f}

Material Condition	Test Temperature, °F	
	1100	1400
Irradiated at 1100°F		
Plate	x	
Weldment ^b	x	
Weldmetal ^c	x	
Irradiated at 1400°F		
Plate		x
Weldment ^b		x
Weldmetal ^c		x

^aTwo stress levels at each material/temperature combination; tests performed in design environment.

^bSpecimens from plate-to-plate weldment.

^cSpecimens from weldment zone of plate weldment.

^dx = two test specimens. Therefore, this matrix reflects 12 specimens.

TABLE 5.1.3.1-7
 CREEP TEST MATRIX FOR IRRADIATED ALLOY 800H^{a, f}

Material Condition	Test Temperature, °F	
	1100	1400
Irradiated at 1100°F		
Plate ^b	x	
Weldment ^{c, d}	x	
Weldmetal ^{c, e}	x	
Irradiated at 1400°F		
Plate ^b		x
Weldment ^{c, b}		x
Weldmetal ^{c, e}		x

^aAll tests in design environment.

^bFive specimens with maximum rupture time goal of 5000 h.

^cFour specimens with maximum rupture time goal of 2000 h.

^dSpecimens from plate-to-plate weldment.

^eSpecimens from all weldmetal zone of plate-to-plate weldment.

^fA total of 26 specimens is required.

TABLE 5.1.3.1-8
HIGH-CYCLE FATIGUE TEST MATRIX FOR IRRADIATED ALLOY 800H^{a,b,f}

Material Condition	Target Cyclic Life			
	5 x 10 ⁵	1 x 10 ⁶	5 x 10 ⁶	1 x 10 ⁷
Irradiated at 1100°F ^c				
Plate	x	x	x	x
Weldment ^d	x	x	x	x
Weldmetal ^e	x	x	x	x
Irradiated at 1400°F ^c				
Plate	x	x	x	x
Weldment ^d	x	x	x	x
Weldmetal ^e	x	x	x	x

^aAll tests in design environment.

^bA limited number of additional specimens will be irradiated. Tests at other temperatures may then be conducted if required.

^cTest temperature = irradiation temperature.

^dSpecimens from plate-to-plate weldment.

^eSpecimens from all weldmetal zone of plate-to-plate weldment.

^fx = one test specimen. Therefore, this matrix calls for 24 specimens.

TABLE 5.1.3.2-1
TENSILE TEST MATRIX FOR ALLOY 800H (HOT DUCT)^c

Material Condition	Test Temperature, °F					
	RT	1000	1100	1200	1400	1600
Annealed						
Plate					x	x
Plate-to-plate weld					x	x
Weldmetal ^a					x	x
Aged 10000 and 20000 h at 1100°F ^b						
Plate			x			x
Plate-to-plate weld			x			x
Weldmetal ^a			x			x
Aged 10000 and 20000 h at 1200°F ^b						
Plate						x
Plate-to-plate weld						x
Weldmetal ^a						x
Aged 10000 and 20000 h at 1400°F ^b						
Plate	x	x		x	x	x
Plate-to-plate weld	x	x		x	x	x
Weldmetal	x	x		x	x	x

^aSpecimens taken from weldmetal zone of plate-to-plate weldment.

^bAll specimens aged in design environment.

^cx = one or two test specimens as appropriate. Therefore, this matrix calls for 54 specimens.

TABLE 5.1.3.2-2

LOW-CYCLE FATIGUE AND CYCLIC STRESS-STRAIN MATRIX FOR ALLOY 800H (HOT DUCT)^f

Material Condition	Test Temperature °F	Stress Range, %							
		2.0	1.5	1.0	0.7	0.5	0.4	0.34	0.3
Annealed Plate	400	x	x	x	x	x	x		
	1300	x	x	x	x	x	x	x	x
	1400	x	x	x	x	x	x	x	x
Plate aged 10,000 h and 20,000 h ^b	1400		x	x		x	x		x
Weldmetal ^c	1400		x	x		x	x		x
Weldment aged 10,000 h and 20,000 h ^b	1400		x	x		x	x		x
HAZ ^d	1400		x	x		x	x		x
HAZ aged 10,000 h and 20,000 h ^{b,d}	1400		x	x		x	x		x
Annealed plate tested in expected environment	1400		x	x		x	x		x

^aAll aging in design environment; all testing in design environment except where noted.^bHalf of specimens aged 10,000 h, half 20,000 h; test temperature = aging temperature.^cSpecimen from all-weldment zone of plate-to-plate weldment.^dHeat affected zone of plate-to-plate weldment.^fx = one or two test specimens as appropriate; therefore this matrix calls for 67 specimens.

TABLE 5.1.3.2-3
CVN MATRIX FOR ALLOY 800H (HOT DUCT)^a

Material Condition	Material Form		
	Plate	Weldment ^b	HAZ ^c
Aged 10,000 h at 1400°F	x	x	x
Aged 20,000 h at 1400°F	x	x	x

^aTen specimens required for each condition/form combination; all aging in the design environment. This matrix calls for 60 specimens.

^bFrom plate-to-plate weldment.

^cHeat affected zone of weldment.

TABLE 5.1.3.2-4
 MATRIX FOR K_{IC} TESTING OF
 ALLOY 800H (HOT DUCT)^{a, f}

Test Temperature, °F	Number of Tests		
	Plate	Weldment ^b	HAZ ^c
<u>Annealed/As-Received</u>			
1400	5 ^d	3	3
<u>Aged 10,000 and 20,000 h at 1400°F^e</u>			
RT	4	4	4
1400	2	2	2

^aTested in air except where noted, 1T compact tension specimens.

^bFrom plate-to-plate weldment.

^cHeat-affected zone of weldment.

^dThree specimens tests in air, two in design environment.

^eAged in design environment, half for 10,000 h and half for 20,000 h.

^fThis matrix calls for 29 specimens.

TABLE 5.1.3.2-5
 FATIGUE CRACK GROWTH MATRIX FOR
 ALLOY 800H (HOT DUCT)^{a, f}

Test Temperature, °F	Number of Tests					
	Plate		Weldment ^b		HAZ ^c	
	Air	He ^d	Air	He ^d	Air	He ^d
<u>Annealed/As-Received</u>						
1400	2	2	2	2	2	2
<u>Aged 10,000 h at 1400°F^e</u>						
1400	1	1	1	1	1	1
<u>Aged 20,000 h at 1400°F^e</u>						
1400		1		1		1

^aAll tests on 0.5 in.-thick specimens with planar dimensions of 1T compact tension specimen; frequency of 1 Hz, R=0.05, no hold time.

^bFrom plate-to-plate weldment.

^cHeat-affected zone of weldment.

^dTested in design environment.

^fThis matrix calls for 21 specimens.

TABLE 5.1.3.2-6
 CREEP CRACK GROWTH MATRIX FOR ALLOY 800H (HOT DUCT)^a

Test Temperature, °F	Number of Tests		
	Plate	Weldment ^b	HAZ ^c
<u>Annealed/As-Received^d</u>			
1400	4	4	4
<u>Aged 10,000 h at 1400°F^e</u>			
1400	1	1	1
<u>Aged 20,000 h at 1400°F^e</u>			
1400	1	1	1

^aSpecimens shown are for design stress; an equal number will be tested at a second stress level. This matrix calls for 36 specimens.

^bFrom plate-to-plate weldment.

^cHeat-affected zone of weldment.

^dHalf of tests in air and half in the design environment.

^eTest temperature = aging temperature; aging and testing in design environment.

TABLE 5.1.3.2-7
CREEP MATRIX FOR ALLOY 800H (HOT DUCT)^a

Material Condition	Test Temperature, °F	
	1200	1400
Annealed/As-Received		
Plate		x
Plate-to-plate weldment		x
Weldmetal ^b		x
Aged 10,000 and 20,000 h at 1200°F		
Plate		x
Plate-to-plate weldment		x
Weldmetal ^b		x
Aged 10,000 and 20,000 h at 1400°F		
Plate	x	x
Plate-to-plate weldment	x	x
Weldmetal ^b	x	x

^aAll aging and testing in the design environment, six specimens per material-temperature combination.

Therefore, this matrix calls for 72 specimens.

^bAll weldmetal sample from plate-to-plate weldment.

TABLE 5.1.3.2-8

HIGH-CYCLE FATIGUE TEST MATRIX FOR ALLOY 800H (HOT DUCT)^a

Material Condition	Test Temperature °F	Target Cyclic Life							
		5x10 ⁵	1x10 ⁶	2x10 ⁶	5x10 ⁶	1x10 ⁷	2x10 ⁷	5x10 ⁷	1x10 ⁸
Annealed Plate	1400	x	x	x	x	x	x	x	x
Plate aged 10,000 h and 20,000 h ^b	1400	x	x	x	x	x	x	x	x
Plate-to-plate weldment	1400	x	x		x	x			
Weldment aged 10,000 h and 20,000 h ^b	1400	x	x		x	x			
HAZ ^c	1400	x	x		x	x			
HAZ aged 10,000 h and 20,000 h ^b	1400	x	x		x	x			
Annealed plate tested in expected environment	1400	x	x		x	x			

^aAll aging in design environment; all testing in design environment except where noted.

^bHalf of specimens aged 10,000 h, half 20,000 h; test temperature = aging temperature.

^cHeat-affected zone of plate-to-plate weldment.

^dx= one or two test specimens as appropriate. Therefore, this matrix calls for 52 specimens.

5.1.4 Schedule

The schedules for the base technology programs responding to the technology requirements for the hot duct materials (Section 4.1.4) are given in this section.

5.1.4.1 Alloy 800H Irradiation Effects (DDN M.10.17.26). Design and construction of the two irradiation capsules are scheduled in FY 1988. Irradiation should be complete and the capsules disassembled in the first quarter of FY 1989. The schedules for post-irradiation mechanical property testing are shown in Tables 5.1.4.1-1 through 5.1.4.1-5.

5.1.4.2 Alloy 800H Primary Coolant Chemistry and Temperature Effects (DDN M.10.17.28). The schedules for mechanical property tests on Alloy 800H base metals and weldments are given Tables 5.1.4.2-1 through 5.1.4.2-4.

5.1.5 Costs

The costs for the base technology programs responding to the technology requirements for the hot duct materials (Section 4.1.4) are given in tabular form in Sections 5.1.5.1 and 5.1.5.2 below.

5.1.5.1 Alloy 800H Irradiation Effects (DDN M.10.17.26). Costs for irradiating and testing specimens of Alloy 800H and its weldments are given by test type (or activity) and fiscal year in Tables 5.1.5.1-1 through 5.1.5.1-6. The total program cost of \$2305K is summarized in Table 5.1.5.1-7.

5.1.5.2 Alloy 800H Primary Coolant Chemistry and Temperature Effects (DDN M.10.17.28). Costs for testing programs on Alloy 800H plate and weldment materials for the hot duct are given by test type and fiscal year in Table 5.1.5.2-1 through 5.1.5.2-5. The total program cost of \$880K is summarized in Table 5.1.5.2-6.

TABLE 5.1.4.1-1

SCHEDULE FOR TENSILE TESTING OF IRRADIATED ALLOY 800H

Task	FY 88	FY 89	FY 90	FY 91
------	-------	-------	-------	-------

Tests on specimens irradiated
at 1100°F

Test on specimens irradiated
at 1400°F

TABLE 5.1.4.1-2
SCHEDULE FOR LOW-CYCLE FATIGUE TESTING
OF IRRADIATED ALLOY 800H^a

Task	FY 88	FY 89	FY 90	FY 91
------	-------	-------	-------	-------

Tests on specimens irradiated
at 1100°F

XXXXXXXXXX

Tests on specimens irradiated
at 1400°F

XXXXXXXXXX

TABLE 5.1.4.1-3
SCHEDULE FOR FRACTURE TOUGHNESS TESTING
ON IRRADIATED ALLOY 800H

Task	FY 88	FY 89	FY 90	FY 91
Charpy tests				
Fracture mechanics (K_{Ic}) tests				
Design, fabricate, and install hot cell test system for fatigue crack growth tests				
Fatigue crack growth tests				
Design, fabricate, and install hot cell test system for creep crack growth tests				
Creep crack growth tests				

TABLE 5.1.4.1-4
SCHEDULE FOR CREEP TESTING
OF IRRADIATED ALLOY 800H

Task	FY 88	FY 89	FY 90	FY 91
Design, fabricate, and install two creep test systems	*****			
Conduct tests on plate material		*****		
Conduct tests on weldment and			*****	

TABLE 5.1.4.1-5
SCHEDULE FOR HIGH-CYCLE FATIGUE TESTING
OF IRRADIATED ALLOY 800H

Task	FY 88	FY 89	FY 90	FY 91
------	-------	-------	-------	-------

Tests on specimens irradiated
at 1100°F

XXXXXX

Tests on specimens irradiated
at 1400°F

XXXXXX

TABLE 5.1.4.2-1
 SCHEDULE FOR TENSILE TESTING
 OF ALLOY 800H (HOT DUCT)

Task	FY 88	FY 89	FY 90	FY 91
Annealed/as-received (plate, weldments, weldmetal)	*****			
Aged 10,000 h at 1100°F to 1400°F (all product forms)		*****		
Aged 20,000 h at 1100°F to 1400°F				****

TABLE 5.1.4.1-2
 SCHEDULE FOR CONDUCTING LOW-CYCLE AND HIGH-CYCLE
 FATIGUE TESTING OF ALLOY 800H (HOT DUCT)

Material Condition	Test Type	FY 88	FY 89	FY 90	FY 91
Annealed Plate	lcf	*****			
	hcf		*****		
Plate aged 10,000 h	lcf		****		
	hcf				****
Plate aged 20,000 h	lcf			****	
	hcf				****
Weld Metal	lcf	***			
	hcf			***	
Weld metal aged 10,000 h	lcf		***		
	hcf				***
Weld metal aged 20,000 h	lcf			****	
	hcf				***
HAZ	lcf	****			
	hcf			****	
HAZ aged 10,000 h	lcf		***		
	hcf				**
HAZ aged 20,000 h	lcf			****	
	hcf				***

TABLE 5.1.4.2-3

SCHEDULE FOR FRACTURE TOUGHNESS TESTING ALLOY 800H (HOT DUCT)

Test Type/ Material Condition	FY 88	FY 89	FY 90	FY 91
Charpy V-notch				
Annealed/as-received		*****		
Aged 10,000 h		*****		
Aged 20,000 h			*****	
K _{Ic} tests				
Annealed/as-received	*****			
Aged 10,000 h		*****		
Aged 20,000 h			*****	
Creep crack growth				
Annealed/as-received	*****			
Aged 10,000 h		*****		
Aged 20,000 h			*****	
Fatigue crack growth				
Annealed/as-received	*****			
Aged 10,000 h		*****		
Aged 20,000 h			*****	

TABLE 5.1.4.2-4

SCHEDULE FOR CREEP TESTING OF ALLOY 800H (HOT DUCT)

Test Type/ Material Condition	FY 88	FY 89	FY 90	FY 91	FY 92
----------------------------------	-------	-------	-------	-------	-------

Annealed plate, weldment,
and weldmetal

Aged 10,000 h (all product
forms)

Aged 20,000 h (all product
forms)

TABLE 5.1.5.1-1
IRRADIATION COSTS

Activity	Cost, \$K	
	FY 1988	FY 1989
Design & construction of two irradiation capsules	330	20
Installation & irradiation in BSR & disassembly		30
	330	50

Total Irradiation Cost - \$380K

TABLE 5.1.5.1-2
 COSTS FOR HOT CELL TENSILE
 TESTING OF IRRADIATED ALLOY 800H

Activity	Cost, \$K		
	FY 1989	FY 1990	FY 1991
Tensile tests	50	50	60
Total tensile test cost - \$160K			

TABLE 5.1.5.1-3
 COSTS FOR HOT CELL LOW-CYCLE
 FATIGUE TESTING OF IRRADIATED ALLOY 800H

Activity	Cost, \$K		
	FY 1989	FY 1990	FY 1991
Equipment procurement and installation	30 ^a		
Testing		75	75
	30	75	75
Total low-cycle fatigue cost - \$180K			

^aCapital cost.

TABLE 5.1.5.1-4
 COSTS FOR HOT CELL FRACTURE TOUGHNESS
 TESTING OF IRRADIATED ALLOY 800H

Activity	Cost, \$K			
	FY 1988	FY 1989	FY 1990	FY 1991
Charpy V-notch testing		15		
Procure and install furnace for K_{Ic} tests ^a	15 ^a			
K_{Ic} testing		75		
Hot cell equipment for fatigue crack growth	50 ^a	50 ^a		
Fatigue crack growth tests		120	150	
Hot cell equipment for creep crack growth	20 ^a	25 ^a		
Creep crack growth tests		100	190	100
	85	385	340	100

Total fracture toughness testing cost - \$910K

^aCapital cost.

TABLE 5.1.5.1-5
COSTS FOR HOT CELL CREEP TESTING
OF IRRADIATED ALLOY 800H

Activity	Cost, \$K			
	FY 1988	FY 1989	FY 1990	FY 1991
Install test systems in hot cell	40 ^a	30 ^a		
Testing		125	200	125
	40	155	200	125

Total creep test cost - \$520K

^aCapital cost.

TABLE 5.1.5.1-6
COSTS FOR HOT CELL HIGH-CYCLE FATIGUE
TESTING OF IRRADIATED ALLOY 800H

Activity	Cost, \$K	
	FY 1990	FY 1991
Testing	75	80
Total high-cycle fatigue test cost - \$155K		

TABLE 5.1.5.1-7
TOTAL COST OF PROGRAMS ON IRRADIATED ALLOY 800H

Activity	Cost, \$K				Total
	FY 1988	FY 1989	FY 1990	FY 1991	
Irradiation	330	50			380
Tensile tests		50	50	60	160
Low-cycle fatigue tests	30	75	75		180
Fracture mechanics tests	85	385	340	100	910
Creep tests	40	155	200	125	520
High-cycle fatigue			75	80	155
	485	715	740	365	2305

TABLE 5.1.5.2-1

COSTS FOR TENSILE TESTING OF ALLOY 800H (HOT DUCT)

Activity	Cost, \$K			
	FY 1988	FY 1989	FY 1990	FY 1991
Tensile testing	25	15	15	15
Total tensile test cost - \$70K				

Table 5.1.5.2-2

COSTS FOR LOW-CYCLE FATIGUE TESTING
OF ALLOY 800H (HOT DUCT)

Activity	Cost, \$K			
	FY 1988	FY 1989	FY 1990	FY 1991
LCF testing	25	30	30	25
Total LCF test cost - \$110				

TABLE 5.1.5.2-3

COSTS FOR FRACTURE TOUGHNESS TESTING OF ALLOY 800H (HOT DUCT)

Activity	Cost, \$K			
	FY 1988	FY 1989	FY 1990	FY 1991
Charpy V-notch testing		5	5	
K_{Ic} tests	20	35	30	
Fatigue crack growth tests	20	30	30	25
Creep crack growth tests	15	25	25	15
	55	95	90	40

Total fracture toughness test cost - \$280K

TABLE 5.1.5.2-4

COSTS FOR CREEP TESTING OF ALLOY 800H (HOT DUCT)

Activity	Cost, \$K				
	FY 1988	FY 1989	FY 1990	FY 1991	FY 1992
Testing	70	60	70	50	50
Total creep test costs - \$300K					

TABLE 5.1.5.2-5

COSTS FOR HIGH-CYCLE FATIGUE TESTING OF ALLOY 800H (HOT DUCT)

Activity	Cost, \$K			
	FY 1988	FY 1989	FY 1990	FY 1991
Testing		40	40	40
Total HCF test costs - \$120K				

TABLE 5.1.5.2-6

TOTAL COST OF PROGRAMS ON ALLOY 800H MATERIALS FOR THE HOT DUCT

Activity	Cost, \$K					Total
	FY 1988	FY 1989	FY 1990	FY 1991	FY 1992	
Tensile tests	25	15	15	15		70
Low-cycle fatigue tests	25	30	30	25		110
Fracture toughness test	55	95	90	40		280
Creep tests	70	60	70	50	50	300
High-cycle fatigue tests		40	40	40		120
	<hr/> 175	<hr/> 240	<hr/> 245	<hr/> 170	<hr/> 50	<hr/> 880

5.2 Steam Generator

The base technology programs for the steam generator are responsive to the technology requirements presented in Section 4.2.4 and its associated DDNs (M.21.02.02 and M.21.02.03). As noted earlier, some parts of the program related to Alloy 800H (M.21.02.03) overlap programs needed for the hot duct (see Section 5.1).

5.2.1 Data Needed

5.2.1.1 Determine Properties of 2.25 Cr-1Mo Base Metal and Weldments (DDN M.21.02.02). Data are needed to determine the effects of thermal aging and environment as detailed in Section 4.2.4.1.

5.2.1.2 Determine Properties of Alloy 800H Base Metal and Weldments (DDN M.21.02.03). Data are needed to determine the effects of thermal aging and environment as detailed in Section 4.2.4.2.

5.2.2 Approach and Rationale for Test Programs

The approach and rationale employed in generating the base technology programs in response to the steam generator technology requirements (Section 4.2.4) are described below in Sections 5.2.2.1 and 5.2.2.2 for the 2.25 Cr-1Mo and Alloy 800H materials, respectively. Test matrices, schedules, and costs are given in later sections (5.2.3 through 5.2.5).

5.2.2.1 Base Metal and Weldment Properties of 2.25 Cr-1Mo (DDN M.21.02.02). The effects of thermal aging (900°F to 1100°F) and expected and design primary coolant helium compositions (see Table 4.1-4) on the properties of the 2.25 Cr-1Mo steel materials will be determined to satisfy the technology requirements specified in Section 4.2.4. Both plate (SA 387, Gr.22, C1.1) and tube (SA 213, T22) materials are utilized in the steam generator and will be included in the test program. Weldments manufactured by the gas tungsten arc (GTA) process will also be represented.

As noted in Section 4.2.3.1, considerable information relative to the properties of 2.25 Cr-1Mo and the effects of MHTGR conditions exists and forms the basis for the selection of this material as the reference for portions of the steam generator. In some instances, however, information on properties is very limited (e.g., creep crack growth). Also, studies involving MHTGR environmental effects have concentrated on impurity levels quite different than those now specified as expected and design. For example, most existing work was done at water vapor level $<5 \mu\text{atm}$ versus the expected ($31 \mu\text{atm}$) and design ($125 \mu\text{atm}$) water levels. In the former case, decarburization was experienced at temperatures $>1050^\circ\text{F}$. Limited experiments with $15 \mu\text{atm}$ water showed decarburization even at the lowest test temperature involved, 805°F . The effects of these higher prescribed water levels are, then, critical aspects of the test program.

Considerations relative to the individual properties listed in Section 4.2.4.1 are given below.

1. Tensile - Tests will be performed in air at 400°F to 1100°F and at room temperature on annealed and aged plate and tube materials (one heat of each) and on weldments. Aging will be done for periods of 10,000 and 20,000 h in both expected and design environments. Aging temperatures will be 900°F through 1100°F .
2. Cyclic stress/strain, low cycle fatigue, and creep-fatigue - All testing will be performed on plate base metal and weldments heat treated to simulate tubing microstructure. (Tubing dimensions are not suitable for manufacture of standard fatigue specimens.) Test temperatures will range from 700°F to 1100°F ; aging will be performed for 10,000 and 20,000 h in the design environment at both 1000°F and 1100°F . Testing will be done in the design environment except for a series of tests on annealed basemetal in the expected environment.

Determination of a damage envelope is a major objective of the creep-fatigue test program. Tension, compression, and tension plus compression hold modes will be employed with hold periods of 0.01 to 0.1 h and strain ranges of from 0.4% to 1.0%.

3. Creep - Tests will be conducted in both expected and design environments at 800°F through 1100°F. Plate, plate weldment, and tubing materials will be tested with a goal of 20,000 to 30,000 h of test time. Aging times and temperatures will mirror those employed for the tensile program. Additionally, ongoing creep tests on four heats of 2.25 Cr-1Mo materials will be continued to failure or to the end of the test program. About 25 tests are in progress in dry ($<5 \mu\text{atm H}_2\text{O}$) HTGR-helium at temperatures from 800°F to 1100°F. Test times on several specimens are in excess of 80,000 h.
4. Fracture toughness - Interest in fracture toughness properties is for the tubing material and its weldments. However, as discussed earlier relative to fatigue specimens, the tubing material dimensions do not permit the manufacture of conventional fracture toughness specimens. Plate material heat treated to simulate tubing microstructure will, therefore, be employed for almost all tests. A few tubing specimens of non-standard K_{IC} design will be tested for correlation. Charpy V-notch (CVN) specimens of plate, weldment, and the weldment heat affected zone (HAZ) will be tested to provide a baseline understanding of the effects of aging in design environment at 1000°F and 1100°F on fracture resistance.

K_{IC} testing at room temperature, 900°F and 1000°F will examine resistance to the initiation of crack extension in plate, plate weldments, and tubing materials in annealed and aged conditions. Most of the elevated temperature testing will be performed in air because the dimensions of the specimens and short testing time (minutes) should preclude significant environmental effects. Testing at the elevated temperatures was selected because, even though the

fracture mode should be expected to be fully ductile, metallurgical changes can accrue that result in a phenomenon called low-energy tearing.

Fatigue crack growth rates (da/dN) will be determined at 600°F through 1000°F on as-received material and after aging at 900°F and 1000°F. The lower temperature test will permit evaluation of the effects of strain aging. Most testing will be done in the design environment but some testing will be performed in air with the aim of understanding the effects of environment on crack growth. Check test will be done in the expected environment. Characterizing fatigue crack growth is difficult because of the large number of parametric variations (i.e., frequencies, R ratios, and hold times) which are possible. The basic test matrix to be described in Section 5.2.3.1 is based on a frequency of 1 Hz, $R=0.05$, and no hold time. However, the test plan also provides for a few tests to measure fatigue crack growth rates at frequencies to 100 Hz, R's up to 0.75, and hold times to 10 minutes on annealed material only. In addition, a number of contingency specimens will be aged to permit testing at various parametric conditions should the results above indicate that it is desirable.

Creep crack growth (da/dt) tests will be run at 900°F and 1000°F to describe crack extension as a consequence of creep mechanisms. As-received and aged materials will be tested in the design environment with some tests of annealed specimens in air. Two nominal stress levels will be employed.

5.2.2.2 Base Metal and Weldment Properties of Alloy 800H (DDN M.21.02.03).

The effects of thermal aging (800°F to 1200°F) and expected and design primary coolant compositions (see Table 4.1-4) on the properties of Alloy 800H will be determined to satisfy the technology requirements specified in Section 4.2.4. Plate (SB-409), tubing (SB-163), and forging (SB-564) materials are utilized in the steam generator and will be tested. Tube-to-tube, plate-to-plate, and

plate-to-forging weldments are also in the steam generator design and will be incorporated into the test program.

Considerations relative to the individual properties listed in Section 4.2.4.2 are given below.

1. Tensile - All of the product forms mentioned above (plus weldmetal from the plate-to-plate weldment) will be involved in the program. Aging will be done at 1000°F to 1200°F for 10,000 and 20,000 h in the design environment. Test temperatures will range from 800°F to 1200°F plus tests at room temperature.
2. Cyclic stress/strain, low-cycle fatigue, and creep-fatigue - Low-cycle fatigue and cyclic stress/strain testing will be done on plate, weld metal, and HAZ material at 600°F to 1200°F before and after aging at 1000°F and 1200°F for 10,000 and 20,000 h. Almost all aging and testing will be in the design environment. As-received and aged forging material will also be tested. Note that appropriately heat treated plate is to be used as a tubing substitute since tubing dimensions are not suitable for manufacture of fatigue specimens. Also, no testing is to be done on tube-to-tube or plate-to-forging weldment HAZs. Creep-fatigue tests will be run on plate (base metal, weld metal, and HAZ) forging material but only at 1200°F.
3. High-cycle fatigue - Tests involve only as-received plate, weldmetal, and HAZ specimens at 900°F through 1200°F. Aging is for 10,000 and 20,000 h at 1000°F and 1200°F. As for the low-cycle fatigue work described above, appropriately heat treated plate material substitutes for tubing.
4. Creep - Tests will be performed on all three product forms and their appropriate weldments. Considerations relative to aging and testing are as described above for tensile properties. All tests will be in the design environment at 800°F through 1200°F.

5. Fracture toughness - Charpy V-notch specimens of plate, plate weldmetal, plate-to-plate HAZ, forging, and forging-to-plate HAZ will be tested to provide a baseline understanding of the effects of aging (1100°F and 1200°F in the design environment) on fracture resistance. Characterization of the steam generator tubing will be provided by the testing of appropriately heat treated plate material. This applies for all fracture toughness properties except K_{Ic} .

K_{Ic} plate-to-plate HAZ, and plate-to-forging HAZ will be performed at room temperature, 1100°F and 1200°F. All tests will be performed in air since they are very short term. Aged (1100°F and 1200°F in design environment) materials will be tested at room temperature and the aging temperature. Specimens of the plate-to-forging HAZ will be aged but tested only if microstructural examination indicates differences relative to the plate-to-plate weldments.

Fatigue crack growth rates (da/dN) will be determined at 600°F through 1200°F for as-received forging, plate, plate weldmetal, plate-to-plate HAZ, and plate-to-forging HAZ specimens. Specimens aged 10,000 and 20,000 h at 1100°F and 1200°F will be tested at the aging temperature. The majority of testing will be done in the design environment. The effects of R ratio, hold time, and frequency will be determined using unaged materials. Forging, HAZ, and plate (heat treated to simulate tubing) samples will be used.

Creep crack growth (da/dt) tests will be run on as-received forging plate, plate weldmetal, plate-to-plate HAZ, and forging-to-plate HAZ specimens at 1100°F and 1200°F in the design environment and air. Material aged at 1100°F and 1200°F for 10,000 and 20,000 h will be tested at the aging temperature in the design environment.

5.2.3 Test Programs

The base technology programs responding to the technology requirements for the steam generator materials (Section 4.2.4) are described below.

5.2.3.1 Base Metal and Weldment Properties of 2.25 Cr-1Mo (DDN M.21.02.02).

Test matrices for the tensile, low-cycle fatigue, creep-fatigue and creep programs are given in Table 5.2.3.1-1 through 5.2.3.1-5. The various property measurement tasks under fracture toughness are described in Tables 5.2.3.1-6 through 5.2.3.1-9.

5.2.3.2 Base Metal and Weldment Properties of Alloy 800H (DDN M.21.02.03).

Test matrices for tensile, low-cycle fatigue, creep-fatigue, high-cycle fatigue, creep, and fracture toughness testing of Alloy 800H product forms for the steam generator are given in Tables 5.2.3.2-1 through 5.3.3.2-9.

5.2.4 Schedule

The schedules for the base technology programs responding to the technology requirements for the steam generator materials (Section 4.2.4) are given in this section.

TABLE 5.2.3.1-1
TENSILE MATRIX FOR 2.25 Cr-1Mo

Material Condition	Test Temperature, °F					
	RT	400	800	900	1000	1100
Annealed						
Plate	X	X	X	X	X	X
Tubing	X	X	X	X	X	
Weld metal ^a	X	X	X	X	X	X
Transverse Weld ^d	X	X	X	X	X	X
Aged 10,000 and 20,000 h ^b @ 900°F						
Tubing	X	X		X		
Weld metal ^a	X	X		X		
Transverse weld ^c	X	X		X		
Aged 10,000 and 20,000 h @ 1000°F ^b						
Plate	X	X			X	
Tubing	X	X			X	
Weld Metal ^a	X	X			X	
Transverse Weld ^d	X	X			X	
Aged 10,000 and 20,000 h @ 1100°F ^b						
Plate	X	X				X
Weld metal ^a	X	X				X
Transverse weld ^e	X	X				X

^aGage length made up of all weld metal from plate weldment.

^bAging in both expected and design environments.

^cTubing weldment.

^dPlate and tubing weldments.

^ePlate weldment.

^fx = 1 specimen for each material/aging time-temperature-environment combination; matrix requires 160 specimens.

TABLE 5.2.3.1-2
LOW-CYCLE FATIGUE MATRIX FOR 2.25 Cr-1Mo^a

Material Condition	Temp., °F	Strain Range, %						
		1.5	1.0	0.7	0.5	0.4	0.35	0.3
Annealed base metal	700	X	X	X	X	X	X	X
	800	X	X	X	X	X	X	X
	900	X	X	X	X	X	X	X
	1000	X	X	X	X	X	X	X
	1100	X	X	X	X	X	X	X
Base metal aged 10,000 and 20,000 h ^b	1000			X		X		X
	1100			X		X		X
Weld metal ^c	700			X		X		X
	800	X	X	X	X	X	X	X
	900			X		X		X
	1000	X	X	X	X	X	X	X
	1100			X		X		X
Weld metal aged 10,000 and 20,000 h ^b	1000			X		X		X
	1100			X		X		X
Heat affected zone ^c	700			X		X		X
	800	X	X	X	X	X	X	X
	900			X		X		X
	1000	X	X	X	X	X	X	X
	1100			X		X		X
Heat affected zone aged 10,000 and 20,000 h ^b	1000			X		X		X
	1100			X		X		X
Annealed base metal tested in expected environment	700			X		X		X
	800	X	X	X	X	X	X	X
	900			X		X		X
	1000	X	X	X	X	X	X	X
	1100			X		X		X

^a Tested in design environment unless otherwise noted; test matrix requires 140 specimens.

^b Aged in design environment, test temperature=aging temperature.

^c From plate weldment.

TABLE 5.2.3.1-3A
CREEP-FATIGUE MATRIX FOR 2.25CR-1MOa

		0.01 h hold time											
		Strain range, %											
Material Condition	Test Temp. °F	1.0 Mode ^b				0.5 Mode ^b				0.4 Mode ^b			
		T	C	T+C	T/C	T	C	T+C	T/C	T	C	T+C	T/C
Annealed base metal	700	X	X	X		X	X	X		X	X		
	1000	X	X	X		X	X	X		X	X		
	1100	X	X	X		X	X	X		X	X		
Base metal aged 10,000 and 20,000 h ^c	1000				X					X			
	1100				X					X			
Weld metal ^d	700				X					X			
	1000				X					X			
	1100				X					X			
Weld metal aged 10,000 and 20,000 h ^c	1000				X					X			
	1100				X					X			
HAZ ^d	700				X					X			
	1000				X					X			
	1100				X					X			
HAZ ^d aged 10,000 and 20,000 h ^c	1000				X					X			
	1100				X					X			
Annealed base metal in expected helium	1000				X					X			
	1100				X					X			

^aTested in design environment unless otherwise noted; test matrix requires 97 specimens.

^bHold mode: T = hold period at peak tension strain,
C = hold period at peak compression strain,
T+C = hold period at peak tension and compression strain,
T/C = hold mode demonstrated most detrimental to fatigue life.

^cAged in design environment before testing. Aging temperature = test temperature.

^dFrom plate weldment.

TABLE 5.2.3.1-3B
CREEP-FATIGUE MATRIX FOR 2.25CR-1Mo STEEL^a

		0.1 h hold time											
		Strain range, %											
Material Condition	Test Temp. °F	1.0 Mode ^b				0.5 Mode ^b				0.4 Mode ^b			
		T	C	T+C	T/C	T	C	T+C	T/C	T	C	T+C	T/C
Annealed base metal	700	X	X	X		X	X						X
	1000	X	X	X		X	X	X		X	X		
	1100	X	X	X		X	X	X		X	X		
Base metal aged 10,000 and 20,000 h ^c	1000												
	1100												
Weld metal ^d	700												
	1000												
	1100												
Weld metal aged 10,000 and 20,000 h ^c	1000												
	1100												
HAZ ^d	700												
	1000												
	1100												
HAZ ^d aged 10,000 and 20,000 h ^c	1000												
	1100												
Annealed base metal in expected helium	1000												
	1100												

^aTested in design environment unless otherwise noted.

^bHold mode: T = hold period at peak tension strain,
C = hold period at peak compression strain,
T+C = hold period at peak tension and compression strain,
T/C = hold mode demonstrated most detrimental to fatigue life.

^cAged in design environment before testing. Aging temperature = test temperature.

^dFrom plate weldment.

TABLE 5.2.3.1-3C
CREEP-FATIGUE MATRIX FOR 2.25CR-1Mo STEEL^a

		0.5 h hold time											
		Strain range, %											
Material Condition	Test Temp. °F	1.0 Mode ^b				0.5 Mode ^b				0.4 Mode ^b			
		T	C	T+C	T/C	T	C	T+C	T/C	T	C	T+C	T/C
Annealed base metal	700				X				X				X
	1000				X				X				X
	1100				X				X				X
Base metal aged	1000												
20,000 h ^c	1100												
Weld metal ^d	700												
	1000												
	1100												
Weld metal aged	100												
20,000 h ^c	1100												
HAZ ^d	700												
	1000												
	1100												
HAZ ^d aged	100												
20,000 h ^c	1100												
Annealed base metal	1000												
in expected helium	1100												

^aTested in design environment unless otherwise noted.

^bHold mode: T = hold period at peak tension strain,

C = hold period at peak compression strain,

T+C = hold period at peak tension and compression strain,

T/C = hold mode demonstrated most detrimental to fatigue life.

^cAged in design environment before testing. Aging temperature = test temperature.

^dFrom plate weldment.

TABLE 5.2.3.1-4
CREEP MATRIX FOR 2.25 Cr-1Mo^a

Material Condition	Testing Temperature, °F			
	800	900	1000	1100
Annealed				
Plate	X	X	X	X
Tubing	X	X	X	
Plate weldment	X	X	X	X
Tubing Weldment	X	X	X	
Weld metal ^b	X	X	X	X
Aged 10,000 and 20,000 h @ 900°F				
Tubing		X		
Tubing weldment		X		
Weld metal ^b		X		
Aged 10,000 and 20,000 h @ 1000°F				
Plate			X	
Tubing			X	
Plate weldment			X	
Tubing weldment			X	
Weld metal ^b			X	
Aged 10,000 and 20,000 h @ 1100°F				
Plate				X
Plate weldment				X
Weld metal ^b				X

^a

All tests conducted in both expected and design environments, six specimens per material-temperature combination; 480 total specimens.

^b

All weld metal sample from plate weldment.

TABLE 5.2.3.1-5
ONGOING CREEP TESTS ON 2.25 Cr-1Mo^a

Material Condition ^b	Temperature, °F	Current Test Time, h
Decarburized, 56448	900	2,000
Decarburized, 56448	900	4,500
Annealed, 56448	900	5,000
Annealed, 56448	900	26,000
Annealed, 56448	900	22,000
Annealed, 72768	1000	78,000
Decarburized, 56448	1000	500
Annealed, 56448	1000	4,000
Annealed, 72768	1000	69,000
Annealed, 72768	1000	75,000
Annealed, 56448	1000	15,000
Annealed, 56448	1000	15,000
Decarburized, 56448	1100	3,500
Annealed, 56448	1100	3,500

^aSpecimens on test in HTGR-helium with <5 μ atm H₂O; about one dozen additional tests ongoing in air.

^bNumerals indicate heat number.

TABLE 5.2.3.1-6
 CHARPY V-NOTCH MATRIX FOR 2.25 Cr-1Mo^a

Condition ^b	Plate	Plate Weldment	Heat Affected Zone
Annealed	X	X	X
Aged 10,000 h @ 1000°F	X	X	X
Aged 20,000 h @ 1000°F	X	X	X
Aged 10,000 h @ 1100°F	X	X	X
Aged 20,000 h @ 1100°F	X	X	X

^aTen specimens required for each product form-material condition combination to define Charpy curve; 150 total specimens.

^bAging performed in design environment.

TABLE 5.2.3.1-7
MATRIX FOR K_{Ic} TESTING OF 2.25 Cr-1Mo^a

Test Temperature, °F	Number of Tests			
	Plate ^b	Weldment ^c	HAZ ^c	Tubing
Annealed				
RT	5 ^d	3	3	3
900	5 ^d	3	3	3
1000	5 ^d	3	3	
Aged at 900°F ^e				
RT	4	4	4	2
900	2	2	2	2
Aged at 1000°F ^e				
RT	4	4	4	
1000	2	2	2	

^a Tested in air except where noted, 1T compact tension specimens; 79 total specimens.

^b Plate material heat treated to simulate tubing microstructure.

^c Weldment and HAZ specimens from plate weldment.

^d Three specimens tested in air, two specimens in design environment.

^e Aged in design environment, half of specimens for 10,000 h and half for 20,000 h.

TABLE 5.2.3.1-8
FATIGUE CRACK GROWTH MATRIX FOR 2.25 Cr-1Mo^a

Test Temperature °F	Number of Tests					
	Plate ^b		Weldment ^c		HAZ ^c	
	Air	He ^d	Air	He ^d	Air	He ^d
				Annealed		
600	2	2	2	2	2	2
800	2	2	2	2	2	2
900	2	3	2	3	2	3
1000	2	3	2	3	2	3
		Aged 10,000 h at 900°F ^e				
900	1	3	1	3	1	3
		Aged 20,000 h at 900°F ^e				
900		3		3		3
		Aged 10,000 h at 1000°F ^e				
1000	1	3	1	3	1	3
		Aged 20,000 h at 1000°F ^e				
1000		3		3		3

^a All tests on 0.5 in. thick specimens with planar dimensions of 1T compact tension specimen; frequency of 1 Hz, R=0.05, no hold time; 96 total specimens; see Note.

^b Plate material heat treated to simulate tubing microstructure.

^c Weldment and HAZ specimens from plate weldment.

^d Two specimens tested in design environment at all temperatures, one specimen tested in expected environment at 900 and 1000°F.

^e Aging in design environment.

Note: An additional 30 specimens of annealed plate material will be tested to determine the effects of R ratio (0.3 and 0.75), hold time (1, 5, and 10 min.), and frequency (10 and 100 Hz). All tests will be performed in the design environment.

TABLE 5.2.3.1-9
CREEP CRACK GROWTH MATRIX FOR 2.25 Cr-1Mo^a

Test Temperature °F	Number of Tests ^b		
	Plate ^c	Weldment ^d	HAZ ^d
Annealed			
900	4	4	4
1000	4	4	4
Aged 10,000 h at 900°F ^e			
900	2	2	2
Aged 20,000 h at 900°F ^e			
900	2	2	2
Aged 10,000 h at 1000°F ^e			
1000	2	2	2
Aged 20,000 h at 1000°F ^e			
1000	2	2	2

^aSpecimens shown are for design stress. An equal number will be tested at an off-normal stress; 96 total specimens.

^bHalf of tests performed in air and half in design environment.

^cPlate material heat treated to simulate tubing microstructures.

^dWeldment and HAZ specimens from plate weldment.

^eAging in design environment.

TABLE 5.2.3.2-1
TENSILE TEST MATRIX FOR ALLOY 800H (STEAM GENERATOR)^b

Material Condition	Testing Temperature, °F					
	RT	800	900	1000	1100	1200
Annealed/as-received						
Tubing	X	X	X	X	X	X
Tube-to-tube weld	X	X	X	X	X	X
Forging	X	X	X	X	X	X
Plate-to-forging weld	X	X	X	X	X	X
Plate	X	X	X	X	X	X
Plate-to-plate weldment	X	X	X	X	X	X
Weldmetal	X	X	X	X	X	X
Aged 10,000 and 20,000 h at 1000°F ^a						
Tubing	X	X		X		
Tube-to-tube weld	X	X		X		
Forging	X	X		X		
Plate-to-forging weld	X	X		X		
Plate						
Plate-to-plate weldment	X	X		X		
Weldmetal	X	X		X		
Aged 10,000 and 20,000 h at 1200°F ^a						
Tubing	X	X				X
Tube-to-tube weld	X	X				X
Forging	X	X				X
Plate-to-forging weld	X	X				X
Plate	X	X				X
Plate-to-plate weldment	X	X				X
Weldmetal	X	X				X

^aAll specimens aged in design environment.

^bx = one or two specimens as appropriate. Total number of specimens is 120.

TABLE 5.2.3.2-2
LOW-CYCLE FATIGUE AND CYCLIC STRESS-STRAIN MATRIX FOR ALLOY 800H (STEAM GENERATOR)^{a,b}

Material Condition	Test Temperature °F	Stress Range, %							
		2.0	1.5	1.0	0.7	0.5	0.4	0.34	0.3
Annealed Plate	600	x	x	x	x	x	x		
	800	x	x	x	x	x	x		
	900	x	x	x	x	x	x		
	1000	x	x	x	x	x	x	x	x
	1100	x	x	x	x	x	x	x	x
	1200	x	x	x	x	x	x	x	x
Plate aged 10,000 h and 20,000 h ^b	1000		x	x		x	x		x
	1200		x	x		x	x		x
Weldmetal ^c	800		x	x		x	x		x
	1000		x	x		x	x		x
	1200		x	x		x	x		x
Weldment aged 10,000 h and 20,000 h ^b	1000		x	x		x	x		x
	1200		x	x		x	x		x
HAZ ^d	1000		x	x		x	x		x
	1200		x	x		x	x		x
HAZ aged 10,000 h and 20,000 h ^{b,d}	1000		x	x		x	x		x
Annealed plate tested in expected environment	800		x	x		x	x		x
	1000		x	x		x	x		x
	1200		x	x		x	x		x
Forging	1000		x	x		x	x		x
	1200		x	x		x	x		x
Forging aged 10,000 h and 20,000 h ^c	1000		x	x		x	x		x
	1200		x	x		x	x		x

^aTest matrix requires 172 specimens.

^bAll aging and testing in the design environment except as noted.

^cTest temperatures = aging temperature.

^dHeat affected zone of plate-to-plate weldment.

TABLE 5.2.3.2-3A
CREEP FATIGUE TESTS MATRIX FOR ALLOY 800H (STEAM GENERATOR)^{a,b}

Material Condition	0.01 h hold time											
	Strain range, %											
	1.0 Mode ^b				0.4 Mode ^b				0.3 Mode ^b			
	T	C	T+C	T/C	T	C	T+C	T/C	T	C	T+C	T/C
Annealed plate	X	X	X		X	X	X		X	X	X	
Plate aged 10,000 h ^c	X	X			X	X						X
Plate aged 20,000 h ^c	X	X			X	X						X
Weld metal	X	X			X	X						X
Weld metal aged 10,000 h ^c	X	X			X	X						X
Weld metal aged 20,000 h ^c	X	X			X	X						X
HAZ ^d	X	X			X	X						X
HAZ ^d aged 10,000 h ^c	X	X			X	X						X
HAZ ^d aged 20,000 h ^c	X	X			X	X						X
Forging	X	X			X	X						X
Forging aged 10,000 h ^c	X	X			X	X						X
Forging aged 20,000 h ^c	X	X			X	X						X

^aTest temperature is 1200°F in design environment.

^bT = hold at peak tension strain amplitude,

C = hold at peak compression strain amplitude,

T+C = hold at peak tension and compression strain amplitudes,

T/C = hold out tension or compression strain amplitude, whichever results in larger reduction in fatigue life.

^cAged at 1200°F.

^dx = one specimen. Total test matrix = 109 specimens.

TABLE 5.2.3.2-3B
 CREEP FATIGUE TEST MATRIX FOR ALLOY 800H (STEAM GENERATOR)^{a,b}

Material Condition	0.1 h hold time											
	Strain range, %											
	1.0 Mode ^b				0.4 Mode ^b				0.3 Mode ^b			
	T	C	T+C	T/C	T	C	T+C	T/C	T	C	T+C	T/C
Annealed plate	X	X	X		X	X	X		X	X		
Plate aged 10,000 h ^c				X				X				X
Plate aged 20,000 h ^c				X				X				X
Weld metal				X				X				X
Weld metal aged 10,000 h ^c				X				X				X
Weld metal aged 20,000 ^c				X				X				X
HAZ ^d				X				X				X
HAZ ^d aged 10,000 h ^c				X				X				X
HAZ ^d aged 20,000 h ^c				X				X				X
Forging				X				X				X
Forging aged 10,000 h ^c				X				X				X
Forging aged 20,000 h ^c				X				X				X

^a

Test temperature is 1200°F in design environment.

^b

T = hold at peak tension strain amplitude,

C = hold at peak compression strain amplitude,

T+C = hold at peak tension and compression strain amplitudes,

T/C = hold out tension or compression strain amplitude, whichever results in larger reduction in fatigue life.

^c

Aged at 1200°F.

^d

Heat affected zone of weldment.

TABLE 5.2.3.2-3C
 CREEP FATIGUE TESTS MATRIX FOR ALLOY 800H (STEAM GENERATOR)^{a,b}

Material Condition	0.5 h hold time											
	Strain range, %											
	1.0 Mode ^b				0.4 Mode ^b				0.3 Mode ^b			
	T	C	T+C	T/C	T	C	T+C	T/C	T	C	T+C	T/C
Annealed plate	X	X			X	X						
Plate aged 10,000 h ^c												
Plate aged 20,000 h ^c												
Weld metal												
Weld metal aged 10,000 h ^c												
Weld metal aged 20,000 ^c												
HAZ ^d												
HAZ ^d aged 10,000 h ^c												
HAZ ^d aged 20,000 h ^c												
Forging												
Forging aged 10,000 h ^c												
Forging aged 20,000 h ^c												

^a Test temperature is 1200°F in design environment.

^b T = hold at peak tension strain amplitude,

C = hold at peak compression strain amplitude,

T+C = hold at peak tension and compression strain amplitudes,

T/C = hold out tension or compression strain amplitude, whichever results in larger reduction in fatigue life.

^c Aged at 1200°F.

^d Heat affected zone of weldment.

TABLE 5.2.3.2-4
HIGH-CYCLE FATIGUE TEST MATRIX FOR ALLOY 800H (STEAM GENERATOR)^{a,b}

Material Condition	Test Temperature °F	Target Cyclic Life							
		5x10 ⁵	1x10 ⁶	2x10 ⁶	5x10 ⁶	1x10 ⁷	2x10 ⁷	5x10 ⁷	1x10 ⁸
Annealed/plate	900	x	x						
	1000	x	x	x	x	x	x	x	x
	1100	x	x	x	x	x	x	x	x
	1200	x	x	x	x	x	x	x	x
Plate aged 10,000 and 20,000 h ^c	1000	x	x		x	x			
	1200	x	x		x	x			
Plate-to-plate weldment	900	x	x						
	1000	x	x		x	x			
	1100	x	x		x	x			
	1200	x	x		x	x			
Weldment aged 10,000 and 20,000 h ^c HAZ ^d	1000	x	x		x	x			
	1200	x	x		x	x	x	x	
	900	x	x						
	1000	x	x		x	x			
	1100	x	x		x	x			
	1200	x	x		x	x			
	1000	x	x		x	x			
	1200	x	x		x	x			
HAZ aged 10,000 and 20,000 h ^c	1000	x	x		x	x			
	1200	x	x		x	x			
Annealed plate tested in expected environment	1100	x	x		x	x			
	1200	x	x		x	x			

^aTest matrix requires 114 specimens.

^bAll aging and testing in design environment except as noted.

^cHalf of specimens aged 10,000 h, half 20,000 h; test temperature = aging temperature.

^dHeat affected zone of plate-to-plate weldment.

TABLE 5.2.3.2-5
 CREEP TEST MATRIX FOR ALLOY 800H (STEAM GENERATOR)^a

Material Condition	Test Temperature, °F				
	800	900	1000	1100	1200
Annealed/as-received					
Plate	x	x	x	x	x
Tubing	x	x	x	x	x
Forging	x	x	x	x	x
Tubing weldment	x	x	x	x	x
Plate-to-plate weldment	x	x	x	x	x
Plate-to-forging weldment	x	x	x	x	x
Weldmetal ^c	x	x	x	x	x
Aged 10,000 and 20,000 h at 1000°F					
Plate			x		
Tubing			x		
Forging			x		
Tubing weldment			x		
Plate-to-plate weldment			x		
Plate-to-forging weldment			x		
Weldmetal ^c			x		
Aged 10,000 and 20,000 h at 1100°F					
Plate				x	
Tubing				x	
Forging				x	
Tubing weldment				x	
Plate-to-plate weldment				x	
Plate-to-forging weldment				x	
Weldmetal ^c				x	
Aged 10,000 and 20,000 at 1200°F					
Plate					x
Tubing					x
Forging					x
Tubing weldment					x
Plate-to-plate weldment					x
Plate-to-forging weldment					x
Weldmetal ^c					x

^a All aging and testing in the design environment, six specimens per material-temperature combination for a total of 462.

^c From plate-to-plate weldment.

TABLE 5.2.3.2-6
CVN MATRIX FOR ALLOY 800H (STEAM GENERATOR)^{a,b}

Material Condition	Material Form				
	Plate	Weldment ^c	HAZ ^d	Forging	HAZ ^e
Annealed/as-received	x	x	x	x	x
Aged 10,000 h @ 1100°F	x	x	x	x	x
Aged 20,000 h @ 1100°F	x	x	x	x	x
Aged 10,000 h @ 1200°F	x	x	x	x	x
Aged 20,000 h @ 1200°F	x	x	x	x	x

^aTen specimens required for each condition/form combination; total specimens = 250.

^bAll aging in design environment.

^cFrom plate-to-plate weldment.

^dHeat affected zone of plate-to-plate weldment.

^eHeat affected zone of plate-to-forging weldment.

TABLE 5.2.3.2-7
MATRIX FOR K_{Ic} TESTING OF ALLOY 800H (STEAM GENERATOR)^{a,b}

Test Temperature, °F	Number of Tests					
	Plate	Weldment ^c	HAZ ^d	Tubes	Forging	HAZ ^e
Annealed/as-received						
RT				3	3	3
1100				3	3	3
1200				3	3	3
Aged 10,000 h @ 1100°F ^d						
RT					4	
1100					2	
Aged 20,000 h @ 1100°F ^d						
RT				2	4	
1100				2	2	
Aged 10,000 h @ 1200°F ^d						
RT					4	
1200					2	
Aged 20,000 h @ 1200°F ^d						
RT					4	
1200					2	
Annealed/As-received						
RT	3	3	3			
1100	5 ^d	3	3			
1200	5 ^d	3	3			
Aged 10,000 and 20,000 h @ 1100°F ^e						
RT	4	4	4			
1100	2	2	2			
Aged 10,000 and 20,000 h @ 1200°F ^e						
RT	4	4	4			
1200	2	2	2			
Aged 10,000 and 20,000 h @ 1400°F ^e						
RT	4	4	4			
1400	2	2	2			

^a All tests in air on 1T compact tension specimens; 140 total specimens.

^b Aging in design environment.

^c From plate-to-plate weldment.

^d Heat affected zone of plate-to-plate weldment.

^e Heat affected zone of plate-to-forging weldment.

TABLE 5.2.3.2-8
FATIGUE CRACK GROWTH MATRIX FOR ALLOY 800H (STEAM GENERATOR)^{a,b,c}

Test Temperature, °F	Number of Tests									
	Plate		Weldment ^d		HAZ ^e		Forging		HAZ ^f	
	Air	He ^g	Air	He ^g	Air	He ^g	Air	He ^g	Air	He ^g
Annealed/as-received										
600	2	2	2	2	2	2	2	2	2	2
1000	2	2	2	2	2	2	2	2	2	2
1100	2	2	2	2	2	2	2	2	2	2
1200	2	2	2	2	2	2	2	2	2	2
Aged 10,000 h @ 1100°F and 1200°F ^h										
1100							1	3	1	
1200							1	2	1	
Aged 20,000 h @ 1100°F and 1200°F ^h										
1100								3		
1200								2		
Aged 10,000 h @ 1400°F ^e										
1100	1	3	2	3	1	3				
1200	1	2	1	2	1	2				
Aged 20,000 h @ 1400°F ^e										
1100		3		3		3				
1200		2		2		2				

^aAll tests on 0.5 in. thick specimens with planar dimensions of 1T compact tension specimen; frequency of 1 Hz, R = 0.05, no hold time.

^bEffects of frequency (1, 10, and 100 Hz) will be determined with unaged specimens of plate and forging at two temperatures; hold time effects will be determined on plate (1, 5, and 10 min.); R ratios of 0.3 and 0.75 will be investigated on plate, forging, and appropriate weldments at two temperatures. Total of about 50 specimens will be required for these tests.

^cTotal test matrix = 144 specimens.

^dFrom plate-to-plate weldment.

^eHeat affected zone of plate-to-plate weldment.

^fHeat affected zone of plate-to-forging weldment.

^gTests in design environment.

^hAging in design environment.

TABLE 5.2.3.2-9
CREEP CRACK GROWTH MATRIX FOR ALLOY 800H (STEAM GENERATOR)^{a,b}

Test Temperature, °F	Number of Tests				
	Plate	Weldment ^c	HAZ ^d	Forging	HAZ ^e
Annealed/as-received					
1100	4	4	4	4	4
1200	4	4	4	4	4
Aged 10,000 h ^f					
1100				3	
1200				2	
Aged 20,000 h ^f					
1100				3	
1200				2	
Aged 10,000 h @ 1400°F ^e					
1100	3	3	3		
1200	2	2	2		
Aged 20,000 h @ 1400°F ^e					
1100	3	3	3		
1200	2	2	2		

^a Specimens shown are for design stress; an equal number will be tested at a second stress level. This matrix calls for 160 specimens.

^b Half of tests in air and half in the design environment.

^c From plate-to-plate weldment.

^d Heat-affected zone of weldment.

^e Heat-affected zone of plate-to-forging weldment.

^f Test temperature = aging temperature.

5.2.4.1 Base Metal and Weldment Properties of 2.25 Cr-1Mo (DDN M.21.02.02).

The schedules for mechanical property tests on 2.25 Cr-1Mo base metal and weldments are given in Tables 5.2.4.1-1 through 5.2.4.1-3.

5.2.4.2 Base Metal and Weldment Properties of Alloy 800H (DDN M.21.02.03).

The schedules for tests on Alloy 800H materials for the steam generator are given in Table 5.2.4.2-1 through 5.2.4.2-6.

5.2.5 Costs

The costs for the base technology programs responding to the technology requirements for the steam generator materials (Section 4.2.4) are given below in tabular form.

5.2.5.1 Base Metal and Weldment Properties of 2.25 Cr-1Mo (DDN M.21.02.02).

Costs for testing programs of 2.25 Cr-1Mo base metal and weldments are given by test type and fiscal year in Tables 5.2.5.1-1 through 5.2.5.1-3. The total program cost of \$2830K is summarized in Table 5.2.5.1-4.

5.2.5.2 Base Metal and Weldment Properties of Alloy 800H (DDN M.21.02.03).

Costs for testing programs on Alloy 800H materials for the steam generator are given by test type and fiscal year in Tables 5.2.5.2-1 through 5.2.5.2-6. The total program cost of \$3530K is summarized in Table 5.2.5.2-7.

TABLE 5.2.4.1-1
SCHEDULE FOR TENSILE AND CREEP TESTING OF 2.25CR-1Mo

[illegible]

TABLE 5.2.4.1-2
SCHEDULE FOR CONDUCTING LOW-CYCLE FATIGUE
AND CREEP FATIGUE TESTS OF 2.25CR-1Mo STEEL

Material Condition	Test Type ^a	FY 88	FY 89	FY 90	FY 91	FY 92
Annealed base metal	1cf	000				
Base metal aged 10,000 h	1cf		000			
Base metal aged 20,000 h	1cf			00		
Weld metal	1cf	000				
Weld metal aged 10,000 h	1cf		00			
Weld metal aged 20,000 h	1cf			00		
HAZ	1cf	00				
HAZ aged 10,000 h	1cf		00			
HAZ aged 20,000 h	1cf			00		
Annealed base metal	cf	oooooooooooooooooooooooooooooooooooo				
Base metal aged 20,000 h	cf				oooooooooooo	
Weldmetal	cf				oooooooooooo	
Weldmetal aged 20,000 h	cf					ooooooo
HAZ	cf				oooooooooo	
HAZ aged 20,000 h	cf					ooooooo

^alcf - low-cycle fatigue test; cf = creep-fatigue test.

TABLE 5.2.4.1-3
SCHEDULE OF FRACTURE TOUGHNESS TESTING
OF 2.25CR-1Mo STEEL

Activity	FY 87	FY 88	FY 89	FY 90	FY 91	FY 92
Charpy V-notch						
Unaged		□□□				
Aged 10,000 h			□□□			
Aged 20,000 h				□□□		
Creep crack growth						
Unaged		□□□□□□□□□□				
Aged 10,000 h			□□□□□□□□□□□□			
Aged 20,000 h				□□□□□□□□□□□□□□□□		
Fracture toughness						
Unaged		□□□□□□□□				
Aged 10,000 h			□□□□□□□□			
Aged 20,000 h				□□□□□□□□□□□□□□□□		
Fatigue crack growth						
Unaged		□□□□□□□□□□				
Aged 10,000 h			□□□□□□□□			
Aged 20,000 h				□□□□□□□□□□□□□□□□		

TABLE 5.2.4.2-1
SCHEDULE FOR TENSILE TESTING OF ALLOY 800H
(STEAM GENERATOR)

Activity	FY 88	FY 89	FY 90	FY 91
Annealed/as-received (tubing, forging, plate, and weldments)	□□□□□			
Aged 10,000 h @ 1000 and 1200°F (all product forms)		□□□□□□□□□□		
Aged 20,000 h @ 1000 and 1200°F (all product forms)				□□□□□□□□□□

TABLE 5.2.4.2-2
SCHEDULE FOR LOW-CYCLE FATIGUE AND CYCLIC STRESS-STRAIN
TESTING OF ALLOY 800H (STEAM GENERATOR)

Material	FY 88	FY 89	FY 90	FY 91	FY 92
All product forms	□□□□□□				
All product forms aged 10,000 h @ 1000 to 1200°F		□□□□□□□□□□			
All product forms aged 20,000 h @ 1000 to 1200°F				□□□□□□□□□□	

TABLE 5.2.4.2-3
 SCHEDULE FOR CREEP-FATIGUE TESTING OF ALLOY 800H
 (STEAM GENERATOR)

Material	FY 88	FY 89	FY 90	FY 91	FY 92
Annealed plate		□□□□□			
Plate aged 10,000 h			□□□□□□□		
Plate aged 20,000 h				□□□□□□□□	
Weld metal		□□□□□□□□			
Weld metal aged 10,000 h			□□□□□□□□		
Weld metal aged 20,000 h				□□□□□□□	
HAZ		□□□□□□□□			
HAZ aged 10,000 h			□□□□□□□□		
HAZ aged 20,000 h				□□□□□□□	
Forging		□□□□□□□□			
Forging aged 10,000 h			□□□□□□□□		
Forging aged 20,000 h					□□□□□□□□

TABLE 5.2.4.2-4
 SCHEDULE OF HIGH-CYCLE FATIGUE TESTING OF ALLOY 800H
 (STEAM GENERATOR)

Material	FY 88	FY 89	FY 90	FY 91	FY 92
Annealed/as-received (plate and weldments)	□□□□□□				
Aged 10,000 h @ 1000 and 1200°F (all product forms)			□□□□□□□□		
Aged 20,000 h @ 1000 and 1200°F (all product forms)				□□□□□□□□□□	

TABLE 5.2.4.2-5
SCHEDULE FOR CREEP TESTING OF ALLOY 800H (STEAM GENERATOR)

[illegible]

TABLE 5.2.4.2-6
SCHEDULE FOR FRACTURE TOUGHNESS TESTING OF ALLOY 800H
(STEAM GENERATOR)

[illegible]

TABLE 5.2.5.1-1
COSTS FOR CREEP AND TENSILE TESTING OF 2.25CR-1Mo

Activity	Cost, \$K					
	FY 1987	FY 1988	FY 1989	FY 1990	FY 1991	FY 1992
Creep testing of base metal and weldments in expected and design environments	85	150	150	100	75	50
Continuation of ongoing creep tests	25	25	25			
Tensile testing	10	20	10	10		
	<u>120</u>	<u>195</u>	<u>185</u>	<u>110</u>	<u>75</u>	<u>50</u>

Total Creep/Tensile Cost = \$735K

TABLE 5.2.5.1-2
COSTS FOR LOW-CYCLE FATIGUE AND CREEP-FATIGUE TESTING OF 2.25CR-1Mo

Activity	Cost, \$K					
	FY 1987	FY 1988	FY 1989	FY 1990	FY 1991	FY 1992
Equipment modification		50 ^a				
Cyclic stress/strain curves	5	50	50	50		
LCF life	5	50	50	25		
Creep-fatigue damage	5		100	125	125	125
	15	150	200	200	125	125

Total Fatigue Cost = \$815K

^aCapital cost.

TABLE 5.2.5.1-3
COST FOR FRACTURE TOUGHNESS TESTING OF 2.25CR-1Mo

Activity	Cost, \$K					
	FY 1987	FY 1988	FY 1989	FY 1990	FY 1991	FY 1992
Charpy V-notch testing	5	10	10	5		
Fracture toughness (K_{Ic})	5	15	60	60	65	40
Fatigue crack growth	30	30 135 ^a	125	125	80	60
Creep crack growth	25	25 95 ^a	80	80	60	55
	<u>65</u>	<u>310</u>	<u>275</u>	<u>270</u>	<u>205</u>	<u>155</u>

Total Fracture Toughness Cost = \$1280K

^aCapital cost

TABLE 5.2.5.1-4
TOTAL COST OF PROGRAMS OF 2.25CR-1Mo

Activity	Cost, \$K						TOTAL
	FY 1987	FY 1988	FY 1989	FY 1990	FY 1991	FY 1992	
Creep/tensile	120	195	185	110	75	50	735
Fatigue	15	150	200	200	125	125	815
Fracture toughness	65	310	275	270	205	155	1280
	<u>200</u>	<u>655</u>	<u>660</u>	<u>580</u>	<u>405</u>	<u>330</u>	<u>2830</u>

TABLE 5.2.5.2-1
COSTS FOR TENSILE TESTING OF ALLOY 800H (STEAM GENERATOR)

Activity	Cost, \$K			
	FY 1988	FY 1989	FY 1990	FY 1991
Testing	30	30	30	25
Total tensile test cost = \$115K				

TABLE 5.2.5.2-2
COSTS FOR LOW-CYCLE FATIGUE TESTING OF ALLOY 800H (STEAM GENERATOR)

Activity	Cost, \$K				
	FY 1988	FY 1989	FY 1990	FY 1991	FY 1992
Testing	30	60	70	50	60
Total LCF test cost = \$270K					

TABLE 5.2.5.2-3
COSTS FOR CREEP-FATIGUE TESTING OF ALLOY 800H (STEAM GENERATOR)

Activity	Cost, \$K			
	FY 1989	FY 1990	FY 1991	FY 1992
Testing	175	165	165	115

Total creep-fatigue test cost = \$620K

TABLE 5.2.5.2-4
COSTS FOR HIGH-CYCLE FATIGUE TESTING OF ALLOY 800H (STEAM GENERATOR)

Activity	Cost, \$K				
	FY 1988	FY 1989	FY 1990	FY 1991	FY 1992
Testing	20	60	80	80	60

Total HCF test cost = \$300K

TABLE 5.2.5.2-5
COSTS FOR CREEP TESTING OF ALLOY 800H (STEAM GENERATOR)

Activity	Cost, \$K				
	FY 1988	FY 1989	FY 1990	FY 1991	FY 1992
Equipment ^a	10	150	200	200	150
Testing	90				
Total Creep test cost = \$800K					

^aCapital cost.

TABLE 5.2.5.2-6
COST FOR FRACTURE TOUGHNESS TESTING OF ALLOY 800H
(STEAM GENERATOR)

Activity	Cost, \$K			
	FY 1989	FY 1990	FY 1991	FY 1992
Charpy V-notch testing	20	30		
Fracture toughness (K_{Ic})	150	150	40	20
Creep crack growth	150	150	125	30
Fatigue crack growth	100	335	90	35
	420	665	255	85

Total fracture toughness cost = \$1425K

TABLE 5.2.5.2-7
TOTAL COST OF PROGRAMS ON ALLOY 800H MATERIALS FOR THE STEAM GENERATOR

Activity (Tests)	Cost, \$K					TOTAL
	FY 1988	FY 1989	FY 1990	FY 1991	FY 1992	
Tensile	30	30	30	25		115
Low-cycle fatigue	30	60	70	50	60	270
Creep-fatigue		175	165	165	115	620
High-cycle fatigue	20	60	80	80	60	300
Creep	100	150	200	200	150	800
Fracture toughness	0	420	665	255	85	1425
	<u>180</u>	<u>895</u>	<u>1210</u>	<u>775</u>	<u>470</u>	<u>3530</u>

5.3 Reactor Vessel

The base technology programs for the reactor vessel are responsive to the technology requirements presented in Section 4.3.4 and its associated DDNs (M.11.06.01 and M.11.06.02).

5.3.1 Data Needed

5.3.1.1 Nil-Ductility Transition Temperature Shift for Reactor Vessel Material Irradiated at Low Temperatures (DDN M.11.06.01). Data are needed to provide a means of conservatively calculating a nil-ductility transition temperature (NDTT) shift of less than 80°F for SA533, Grade B, Class 1 pressure vessel steel and its weldments under MHTGR operating conditions. Maximum fluence at the reactor vessel beltline is 1.4×10^{17} n/cm² for metal temperatures of 400°F to 420°F; that at the closure head is 3.3×10^{18} n/m² at 350°F to 390°F. The neutron energy spectrum anticipated in these applications is described in Section 4.3.3.

5.3.1.2 Determine Properties of SA533B (Mn-1/2 Mo-1/2 Ni) Base Metal and Weldment at Elevated Temperatures (DDN M.11.06.02). Creep and tensile data are needed to establish allowable stress intensities under Section III of the ASME Code for SA533 Grade B, Class 1 pressure vessel steel and its weldments during Service Level C and D events. These events involve temperatures from 700°F to 884°F for times up to 400 h.

5.3.2 Approach and Rationale for Test Programs

The approach and rationale employed in developing the base technology programs in response to the reactor vessel technology requirements (Section 4.3.4) are described below in Sections 5.3.2.1 and 5.3.2.2 relating to the NDTT shift and elevated temperature stress allowables, respectively. Test matrices, schedules, and costs for these programs are given in Sections 5.3.3 through 5.3.5.

5.3.2.1 NDTT Shift in SA533B Steel (DDN M.11.06.01). The shell and heads of the MHTGR reactor vessel are to be constructed of SA533 Grade B, Class 1 quenched and tempered alloy steel plate. The vessel material, depending on position in the vessel, will be irradiated to end-of-life fluences on the order of 10^{17} to 10^{18} n/cm² (see Section 4.3.3) at temperatures from 350°F to 420°F. Irradiation is known to result in an NDTT shift in this material under light water reactor (LWR) conditions (approximately 550°F) but very little data exist at the lower temperatures characteristic of the MHTGR.

Section 4.3.3 also provides a breakdown of the fluences associated with different neutron energy groups while DDN M.11.06.01 gives only a total fluence which is a simple summation of the fluences for each energy group. It is also stated that an equivalent fast fluence (to correlate with LWR data) can be employed to account for the neutron spectrum. Although more detailed information is needed to accurately estimate equivalent fast fluences (and dpa values), approximate calculations result in values of 2.6×10^{16} n/cm² and 4.0×10^{17} n/cm² for the beltline and head, respectively. (Provided the neutron energy spectrum is known in detail, the exposure time in a test reactor can be adjusted to provide the total dpa desired.)

The test program consists of four capsules with each capsule containing twenty Charpy V-notch (CVN) and six tensile specimens. Irradiations will be conducted by Materials Engineering Associates (MEA) at the University of Buffalo Reactor (UBR). Three irradiation temperatures (350°F, 400°F and 550°F) are planned and correspond to the head, beltline, and a benchmark to LWR data, respectively. These should provide the basic information needed to determine the effects of neutron irradiation on the vessel material under MHTGR conditions.

Limited information in the literature for low alloy steels indicates a relatively constant shift in transition-temperature (Δ NDTT) with temperature up to a critical temperature of about 400°F, followed by a decreasing shift with increasing temperature. However, there is not enough information available to define the critical temperature, especially as it varies with

material composition. Therefore, it is considered necessary to include both 350°F and 400°F exposures for the plate material. Weldments will be irradiated only at 400°F. This assumes that there are no irradiation exposed weldments in the head area.

The specimens from the irradiation capsules should be sufficient to assess the shift in RT_{NDT} (Regulatory Guide 1.99 uses 30 ft-lb as a correlation with NDTT) as well as changes in tensile properties. The test program also includes unirradiated CVN and tensile specimens as well as drop weight specimens for determination of NDTT.

5.3.2.2 Elevated Temperature Stress Allowables for SA533B Steel

(DDN M.11.06.02). Section III of the ASME Code provides data on the ultimate tensile and yield strengths of SA533 Grade B, Class 1 steel to a maximum temperatures of 1000°F. However, allowable design stress intensities are not provided for temperatures above 700°F. The temperature of the reactor vessel material is expected to exceed 700°F only during Service Level C and D events and then only for short periods of time (maximum metal temperatures of 884°F with maximum time above 700°F of about 400 h associated with depressurized conduction cooldown). Despite these relatively undemanding conditions, ASME Code rules require that data be available to demonstrate negligible creep during the service events and to permit extension of the Section III rules for special event allowable above 700°F. Some data exist for temperatures to 1200°F (DPRI NP-2763, December 1982) but for "creep" periods of up to only 48 h.

Tensile and creep tests on one heat of SA533B plate material and its weldments are planned to provide the necessary data. The tensile tests of base metal, weld metal, and weldment specimens will be conducted in air at room temperature and at 700°F through 1000°F. Two strain rates will be employed. These data will provide a benchmark with the tensile data already contained in Section III.

The creep tests will be run on base metal, weld metal, and weldment specimens at a stress level of approximately 95% of yield strength at 700°F through 1000°F. Tests will be allowed to continue to failure or to 1000 h maximum duration. Testing will be done in MHTGR-helium (design levels of impurities) but more as a matter of convenience than necessity as little or no environmental effect is expected at such low temperatures and short times.

5.3.3 Test Programs

The base technology programs responding to the technology requirements for the reactor vessel material (Section 4.3.4) are described below.

5.3.3.1 NDTT Shift in SA533B Steel (DDN M.11.06.01). The four irradiation capsules required in the test program are described in Table 5.3.3.1-1. Mechanical property testing associated with each capsule is given in Table 5.3.3.1-2.

5.3.3.2 Elevated Temperature Stress Allowables for SA533B Steel (DDN M.11.06.02). The test matrices for tensile and creep property determinations on the reactor vessel material are shown in Table 5.3.3.2-1.

5.3.4 Schedule

The schedules for the base technology programs responding to the technology requirements for the reactor vessel material (Section 4.3.4) are given in this section.

TABLE 5.3.3.1-1
IRRADIATION CONDITIONS FOR SA533B STEEL

Capsule	Associated Component	Material Form	Irradiation Temp. (°F)	Approximate Equivalent Fast Fluence (n/cm) ^a
1	Head	Plate	350	1 x 10
2	Beltline	Plate	400	1 x 10
3	Beltline	Weldment	400	1 x 10
4	LWR benchmark	Plate	550	1 x 10

^aTo be further refined before irradiations.

TABLE 5.3.3.1-2
MECHANICAL PROPERTY TEST MATRIX TO DETERMINE NDTT SHIFT

Test Type	Test Conditions	Number of Specimens
<u>Irradiated^a</u>		
CVN	Temp. range necessary to define ductile-to-brittle transition	20
Tensile	Room temperature - 550°F	6
<u>Unirradiated^b</u>		
CVN	Temp. range necessary to define ductile-to-brittle transition	20
Tensile	Room temperature - 550°F	6
Drop-weight	Temp. range necessary to define NDTT	8

^aNumber of specimens shown is for each capsule

^bNumber of specimens shown is for each material form.

TABLE 5.3.3.2-1
TEST MATRIX FOR TENSILE AND CREEP-TESTING OF SA533B STEEL

Material Condition	Test temperature, °F				
	25	70	800	900	1000
<u>Tensile tests^a</u>					
Base metal	x	x	x	x	x
Weld metal	x	x	x	x	x
Transverse weldment	x	x	x	x	x
<u>Creep tests^b</u>					
Base metal		x	x	x	x
Weld metal		x	x	x	x
Transverse weldment		x	x	x	x

^aTwo strain rates. Two tests at each strain rate. Stress strain curves will be recorded for each test. Total number of specimens = 60.

^bThree tests at each temperature. Tests to be conducted at a stress level of 95% of the yield strength at the test temperature. Tests will be allowed to accumulate up to 1000 hours. Total number of specimens = 36.

5.3.4.1 NDTT Shift in SA533B Steel (DDN M.11.06.01). The schedule for irradiation and testing of SA533B reactor vessel steel is given to Table 5.3.4.1-1.

5.3.4.2 Elevated Temperature Stress Allowables for SA533B Steel (DDN M.11.06.02). The schedule for tensile and creep testing of the reactor vessel material is given in Table 5.3.4.2-1. All testing should be completed by early in FY 1988.

5.3.5 Costs

The costs for the base technology programs responding to the technology requirements for the reactor vessel material (Section 4.3.4) are given below.

5.3.5.1 NDTT Shift in SA533B Steel (DDN M.11.06.01). The costs for irradiation and testing of the reactor vessel steel are given in Table 5.3.5.1-1.

5.3.5.2 Elevated Temperature Stress Allowables for SA533B Steel (DDN M.11.06.02). The total cost for tensile and creep tests of the reactor vessel material is estimated as \$150K.

TABLE 5.3.4.1-1
SCHEDULE FOR IRRADIATION TESTING OF SA533B STEEL

	FY 1987	FY 1988	FY 1989
Planning	□□□□□□		
Specimen fabrication		□□□	
Irradiation		□□□□	
Testing		□□□□□□□□□□	
Analysis and reporting		□□□□□□□□□□	

TABLE 5.3.4.2-1
SCHEDULE FOR TENSILE AND CREEP TESTING OF SA533B

	FY 1987	FY 1988
Procurement of A533B material	□□□□□□□□□□	
Tensile testing of base metal	□□□□□	
Fabrication of weldment	□□□	
Creep testing of base metal	□□□□□□	
Tensile testing of weldment	□□□□□□	
Creep testing of weldment	□□□□□□□□□□	

TABLE 5.3.5.1-1
COSTS FOR IRRADIATION AND TESTING OF SA533B STEEL

Task	FY 1987	FY 1988	FY 1989
Planning	25		
Specimen fabrication		20	
Irradiation		175	
Testing		125	
Analysis and reporting		20	25
	<hr/>	<hr/>	<hr/>
	25	340	25

Total program cost = \$390K

5.4 Overall Program Costs

In developing the schedules for the metals base technology programs responding to technology requirements specified in Sections 4.1.4 through 4.3.4, it was assumed that completion of most of the work by the end of FY 1991 and all work before the end of FY 1992 was necessary to provide proper input to final design, licensing submittals, and materials orders. Budget constraints were not considered in developing the schedules except for FY 1987 and 1988 but reasonably anticipated manpower and facility availabilities were taken into account. Operating and capital cost requirements to meet these schedules are presented in Tables 5.4-1 and 5.4-2 for the major elements of the program. Note that a total of \$6900K is required for high priority programs on steam generator and reactor vessel materials. An additional \$3185K would be needed to complete the medium priority work related to the hot duct. Taking both into account gives a total of \$10085K. Approximately 40% of this total is for activities related to fracture mechanics properties.

FY 1988 is heavily loaded with capital equipment funds (\$445); it would be possible to divide the total (\$550K) more equally between FYs 1988 and 1989. Operating costs for high priority programs are more than \$1000K in each of FYs 1988 through 1991, peaking at almost \$1800K in FY 1990, and falling to \$800K in FY 1992. Some, but by no means all, of the front end needs (e.g., FY 1988) could be pushed into later years and still allow a FY 1992 completion. Funding required for the medium priority work peaks in FY 1990 at \$985K and falls off to only \$50K in FY 1992.

TABLE 5.4-1
COSTS BY FISCAL YEAR FOR HIGH PRIORITY PROGRAMS

Fiscal Year	Steam Generators ^a		Reactor Vessel		Total
	Operating	Capital	Operating	Capital	
1987	200		150		350
1988	545	290	365		1200
1989	1555		25		1580
1990	1790				1790
1991	1180				1180
1992	800				800
	<hr/> 6070	<hr/> 290	<hr/> 540	<hr/> 0	<hr/> 6900

^a A significant fraction of the high priority work for the steam generator will provide data also needed for the hot duct (medium priority).

TABLE 5.4-2
COSTS BY FISCAL YEAR FOR
MEDIUM PRIORITY PROGRAMS

Fiscal Year	Hot Duct		Total
	Operating	Capital	
1988	505	155	660
1989	850	105	955
1990	985		985
1991	535		535
1992	50		50
	<hr/> 2925	<hr/> 260	<hr/> 3185