ornl

ORNL/TM-13498

RECEIVED

AUG 06 1998

OSTI

**OAK RIDGE
NATIONAL
LABORATORY**

LOCKHEED MARTIN

High Flux Isotope Reactor Cold Neutron Source Reference Design Concept

D. L. Selby
A. T. Lucas
C. R. Hyman
J. D. Freels
P. T. Williams
S. J. Chang
W. E. Hill
C. T. Ramsey
J. A. Bucholz
D. H. Cook
R. H. Nelson
M. A. Woody
R. M. Moon
K. A. Morgan

May 1998

JAT **MASTER**

DISTRIBUTION OF THIS DOCUMENT IS UNLIMITED

MANAGED AND OPERATED BY
LOCKHEED MARTIN ENERGY RESEARCH CORPORATION
FOR THE UNITED STATES
DEPARTMENT OF ENERGY

ORNL-27 (3-96)

This report has been reproduced directly from the best available copy.

Available to DOE and DOE contractors from the Office of Scientific and Technical Information, P.O. Box 62, Oak Ridge, TN 37831; prices available from (615) 576-8401, FTS 626-8401.

Available to the public from the National Technical Information Service, U.S. Department of Commerce, 5285 Port Royal Rd., Springfield, VA 22161.

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.

**HIGH FLUX ISOTOPE REACTOR COLD NEUTRON SOURCE
REFERENCE DESIGN CONCEPT**

D. L. Selby, A. T. Lucas, C. R. Hyman, J. D. Freels, P. T. Williams,
S. J. Chang, W. E. Hill, C. T. Ramsey, J. A. Bucholz, D. H. Cook,
R. H. Nelson, M. A. Woody, R. M. Moon, K. A. Morgan

May 1998

Prepared by
Oak Ridge National Laboratory
Oak Ridge, Tennessee 37831-6285
managed by
Lockheed Martin Energy Research Corp.
for the
U.S. Department of Energy
under Contract No. DE-AC05-96OR22464

CONTENTS

LIST OF FIGURES	vii
LIST OF TABLES	xi
ACRONYMS	xv
1. INTRODUCTION	1-1
1.1 PROJECT PURPOSE AND SCOPE	1-1
1.2 PROJECT GOALS AND REQUIREMENTS	1-1
1.2.1 User Requirements and Goals	1-1
1.2.2 Regulatory Compliance Requirements	1-2
1.2.3 Safety Requirements and Goals	1-2
1.2.4 Operational Requirements and Goals	1-4
1.2.5 Maintenance Requirements	1-6
1.3 PROJECT SCHEDULE	1-6
1.4 REFERENCES FOR CHAPTER ONE	1-7
2. REFERENCE CONCEPT DESIGN	2-1
2.1 GENERAL SYSTEM DESIGN	2-1
2.1.1 Operating Parameters	2-1
2.1.2 System Configuration	2-1
2.2 SYSTEM OPERATING CONDITIONS	2-4
2.2.1 Normal Operation	2-4
2.2.2 Standby Operation	2-5
2.3 INDIVIDUAL MODULE DESCRIPTIONS	2-6
2.3.1 Moderator Plug Assembly	2-6
2.3.2 Pump Module	2-9
2.3.3 Heat Exchanger Module	2-11
2.3.4 Gas-Handling System	2-12
2.3.5 Refrigerator Module	2-14
2.3.6 Transfer Lines Module	2-14
2.3.7 Insulating Vacuum Systems Module	2-15
2.3.8 Inert Blanket	2-16
2.3.9 Control and Instrumentation	2-16
2.3.10 Equipment Enclosures and Safe Room	2-18
2.3.11 Vent System	2-19
2.4 PHYSICS ANALYSIS	2-21
2.4.1 Physics Codes, Model Description, and Cross-Section Data Sets	2-21
2.4.2 Brightness, Gain Factors, and Comparisons with Existing Facilities	2-26
2.4.3 Nominal Heat Load Calculations	2-27
2.4.4 Design Optimization Studies	2-28
2.5 THERMAL ANALYSIS OF KEY COLD SOURCE COMPONENTS	2-36
2.5.1 Description of CFD Code CFX-4	2-36
2.5.2 Moderator Vessel Analysis	2-42
2.5.3 Vacuum Tube Analysis	2-49
2.5.4 Conclusions	2-56

2.6 MODERATOR VESSEL AND VACUUM STRUCTURAL ANALYSIS	2-56
2.6.1 HFIR Cold Neutron Source Moderator Vessel Structural Design Analysis	2-56
2.6.2 Vacuum Tube Structural Analysis	2-74
2.7 COLD SOURCE TESTING	2-84
2.7.1 Full-Scale Hydrogen Loop Tests	2-84
2.7.2 Mark I Moderator Vessel Pressure Tests	2-89
2.8 REFERENCES FOR CHAPTER TWO	2-89
3. HFIR COLD SOURCE SAFETY ANALYSIS	3-1
3.1 SAFETY DESIGN PHILOSOPHY AND PRINCIPAL SAFETY FEATURES	3-1
3.1.1 Safety Design Philosophy	3-1
3.1.2 Summary of Principal Safety Features	3-2
3.2 HYDROGEN-RELATED HAZARDS ANALYSIS	3-4
3.2.1 In-Confinement Safe Room Response to H ₂ /O ₂ Burns	3-4
3.2.2 Preliminary Beam Tube H ₂ Detonation Analysis	3-8
3.2.3 Vacuum Tube Cryopumping Analysis	3-12
3.2.4 Future H ₂ Hazards Analyses	3-14
3.3 ACCIDENT IDENTIFICATION AND PROBABILISTIC FREQUENCY ASSESSMENT	3-15
3.3.1 Safety Goal	3-15
3.3.2 System Design Evaluated	3-15
3.3.3 Methodology	3-16
3.3.4 Examination of Event Scenarios	3-23
3.3.5 Recommendation of Events for Further Safety Analysis	3-30
3.4 PRELIMINARY SYSTEMS TRANSIENT ANALYSIS	3-31
3.4.1 Description of the ATHENA Code	3-31
3.4.2 Description of the ATHENA HFIR Cold Source Model	3-33
3.4.3 Analysis Results	3-45
3.4.4 Conclusions and Recommendations	3-72
3.5 EXPECTED RESPONSES FOR OFF-NORMAL EVENTS ASSOCIATED WITH THE HB-4 COLD SOURCE	3-74
3.5.1 HB-4 Beam Tube Wall	3-79
3.5.2 Vacuum Tube	3-81
3.5.3 Refrigeration System	3-83
3.5.4 Hydrogen Circulators	3-84
3.5.5 Vacuum Systems	3-85
3.5.6 Electrical Power Supply	3-87
3.5.7 Hydrogen Barriers	3-88
3.5.8 Seismic and Wind/Tornado Events	3-92
3.5.9 Nonoperational Events	3-93
3.6 CLASSIFICATION OF COLD SOURCE EQUIPMENT: SAFETY, QUALITY, AND SEISMIC	3-94
3.6.1 Safety Classification Guidelines	3-95
3.6.2 Quality Level Guidelines	3-96
3.6.3 Seismic Criteria	3-98
3.6.4 HB-4 Cold Source Equipment And Component Classifications	3-98
3.5 REFERENCES FOR CHAPTER THREE	3-107

4. PROJECT PLANS	4-1
4.1 HFIR COLD SOURCE CONFIGURATION MANAGEMENT PLAN	4-1
4.1.1 Project Organization	4-1
4.1.2 Configuration Management Objectives	4-1
4.1.3 Configuration Management Plan Scope	4-1
4.1.4 Configuration Control Implementation	4-1
4.2 PROCUREMENT PLAN	4-2
4.3 CONSTRUCTION AND INSTALLATION PLAN	4-3
4.3.1 Building Construction	4-4
4.3.2 Equipment Installation	4-6
4.4 OPERATIONS PLAN	4-7
4.4.1 Operations Organization	4-8
4.4.2 Operations Staffing	4-8
4.4.3 Operational Budget	4-9
4.4.4 Operations and Maintenance Policy	4-9
4.4.5 Normal Modes of Operation	4-10
4.4.6 Annealing of the Cold Source Moderator Vessel	4-13
4.4.7 Abnormal and Emergency Operating Conditions	4-13
4.4.8 Waste Management	4-15
4.5 QUALITY ASSURANCE PLAN	4-15
4.6 RELIABILITY, AVAILABILITY, AND MAINTAINABILITY PLAN	4-17
4.6.1 Goals and Requirements	4-17
4.6.2 Cold Source Systematic Approach to RAM	4-17
4.6.3 Maintaining RAM Program Integrity During Final Design and Construction	4-19
4.6.4 Implementing A Long-Term RAM Program	4-19
4.7 COLD SOURCE TESTING PLAN	4-19
4.7.1 Component Development Tests	4-20
4.7.2 Integral Hydrogen Loop Tests	4-20
4.7.3 Acceptance Tests	4-20
4.7.4 Installed Loop Tests With Moderator Plug Assembly Mounted Outside the HFIR Building	4-21
4.7.5 Tests With the Cold Source Fully Installed With and Without Reactor Power	4-22
4.8 REFERENCES FOR CHAPTER FOUR	4-22
APPENDIX A: ORTHO/PARA-HYDROGEN PARAMETRIC PHYSICS STUDY	A-1
APPENDIX B: CALCULATION STEPS FOR LH ₂ STRUCTURES/AIR EQUILIBRATION ANALYSIS	B-1

LIST OF FIGURES

1.1	HFIR cold source project schedule	1-7
2.1	HFIR cold source system flow diagram	2-2
2.2	HFIR cold source temperature vs entropy curves	2-3
2.3	Piping and valve connections to the high density circulators	2-5
2.4	Cold source facility layout	2-7
2.5	HFIR cold source moderator vessel	2-8
2.6	Pump module top view	2-9
2.7	Pump module side view	2-10
2.8	Pump module piping diagram	2-11
2.9	Hydrogen gas-handling module	2-13
2.10	Hydrogen transfer line	2-15
2.11	Temperature control system	2-18
2.12	Layout of hydrogen safe room and refrigerator equipment rooms	2-20
2.13	Vent system diagram	2-21
2.14	HFIR reactor core midplane layout	2-23
2.15	Calculated gain factors compared with facility gain factors	2-28
2.16	Ortho hydrogen impact on flux spectrum in NIST cold source geometry	2-30
2.17	Ortho hydrogen impact on flux spectrum in HFIR cold source geometry	2-31
2.18	Nuclear heating of cold source components	2-42
2.19	Moderator vessel assembly	2-43
2.20	Thermal conductivity of aluminum and alloys	2-45
2.21	Pressure vs enthalpy for hydrogen	2-46
2.22	Temperature vs entropy for hydrogen	2-46
2.23	Specific heat and density of parahydrogen	2-47

2.24	Thermal conductivity and viscosity of hydrogen	2-47
2.25	Moderator wall temperature distribution and flow field for 1 L/s of H ₂ at 15 bar	2-50
2.26	Vacuum tube geometry for model No. 6	2-51
2.27	Vacuum tube geometry for model No. 7	2-52
2.28	Water flow path for forced-convection cooling of vacuum tube	2-55
2.29	Temperature distribution in vacuum tube	2-55
2.30	ABAQUS output of Tresca stresses for section point 1, the outer surface	2-60
2.31	ABAQUS output of Tresca stresses for section point 2, the middle surface	2-61
2.32	ABAQUS output of Tresca stresses for section point 3, the inner surface	2-62
2.33	Aluminum 6061-T6 toughness and 0.2% flow stress curves	2-63
2.34	Impact of specimen thickness on toughness	2-65
2.35	Probability of fracture based on maximum membrane stress	2-67
2.36	Probability of fracture based on maximum bending stress	2-68
2.37	Path-to-failure curves for the unirradiated and fully irradiated moderator vessel	2-74
2.38	Vacuum tube cap stress distribution	2-76
2.39	Vacuum tube short cylinder stress distribution	2-77
2.40	Vacuum tube long cylinder stress distribution	2-77
2.41	Schematic of AEDC hydrogen loop experiment	2-85
2.42	Top view of hydrogen test loop inside the AEDC Mark-I vacuum chamber	2-86
3.1	MELCOR nodalization used to examine the viability of an in-confinement safe room	3-5
3.2	MELCOR calculated temperature response to H ₂ /O ₂ reaction in the in-building safe room concept	3-6
3.3	MELCOR calculated pressure response to H ₂ /O ₂ reaction in the in-building safe room concept	3-7
3.4	Hydrogen containment and pressure relief concept	3-16
3.5	Master logic diagram of initiating events for the HFIR cold source	3-22

3.6	Overall noding diagram of the HFIR cold source ATHENA model	3-34
3.7	ATHENA noding diagram of the moderator vessel for the HFIR cold source	3-38
3.8	Pressurizer vessel noding diagram of the HFIR cold source ATHENA model	3-38
3.9	Noding diagram of the circulator station subsystem of the HFIR cold source ATHENA model	3-40
3.10	ATHENA noding diagram of the HFIR cold source heat exchanger	3-41
3.11	Noding diagram of the concentric transfer lines subsystem of the HFIR cold source ATHENA model	3-42
3.12	The noding diagram of the ATHENA model of the feed and bleed subsystem for the HFIR cold source	3-43
3.13	Pressure instrument edits for the transition from normal to standby simulation at 4 bar using the HFIR cold source ATHENA model	3-49
3.14	Temperature instrument edits for the transition from normal to standby simulation at 4 bar using the HFIR cold source ATHENA model	3-50
3.15	Valve position and T10 edits for the transition from normal to standby simulation at 4 bar using the HFIR cold source ATHENA model	3-51
3.16	Valve mass flow rate edits for the transition from normal to standby simulation at 4 bar using the HFIR cold source ATHENA model	3-52
3.17	Miscellaneous variable edits for the transition from normal to standby simulation at 4 bar using the HFIR cold source ATHENA model	3-53
3.18	Pressure instrument edits for the transition from normal to standby simulation at 15 bar using the HFIR cold source ATHENA model	3-59
3.19	Temperature instrument edits for the transition from normal to standby simulation at 15 bar using the HFIR cold source ATHENA model	3-60
3.20	Valve position and T10 edits for the transition from normal to standby simulation at 15 bar using the HFIR cold source ATHENA model	3-61
3.21	Valve mass flow rate edits for the transition from normal to standby simulation at 15 bar using the HFIR cold source ATHENA model	3-62
3.22	Miscellaneous variable edits for the transition from normal to standby simulation at 15 bar using the HFIR cold source ATHENA model	3-63

3.23	Pressure instrument edits for the high-density circulator trip and decrease in secondary heat removal simulation at 15 bar using the HFIR cold source ATHENA model	3-67
3.24	Temperature instrument edits for the high-density circulator trip and decrease in secondary heat removal simulation at 15 bar using the HFIR cold source ATHENA model	3-68
3.25	Valve position and T10 edits for the high-density circulator trip and decrease in secondary heat removal simulation at 15 bar using the HFIR cold source ATHENA model	3-69
3.26	Valve mass flow rate edits for the high-density circulator trip and decrease in secondary heat removal simulation at 15 bar using the HFIR cold source ATHENA model	3-70
3.27	Miscellaneous variable edits for the high-density circulator trip and decrease in secondary heat removal simulation at 15 bar using the HFIR cold source ATHENA model	3-71
4.1	Cold source buildings	4-4

LIST OF TABLES

2.1	Supercritical hydrogen gas at 15 bar (217.6 psia)	2-3
2.2	Liquid hydrogen at 4 bar (58 psia)	2-3
2.3	Normal system operating parameters	2-4
2.4	Standby operating parameters	2-6
2.5	Key HFIR physics model parameters	2-22
2.6	Energy bands of interest	2-24
2.7	Proposed cold source instruments	2-25
2.8	Neutron brightness comparison between the HFIR cold source and the ILL horizontal cold source	2-26
2.9	Gain factors as calculated for the proposed HFIR liquid hydrogen cold source	2-27
2.10	Nominal cryogenic heat loads for the reference HFIR cold source	2-29
2.11	Relative brightness as a function of the cold source thickness(t), percent ortho-H (x), and wavelength (w), where the relative brightness is defined as $RB(t,x,w) = B(t,x,w)/B(5,0,w)$	2-32
2.12	Brightness ratio	2-33
2.13	Comparison of relative fluxes in HB-1 with and without the HB-4 LH ₂ cold source	2-34
2.14	Comparison of RB-7A neutron fluxes with and without HB-4 LH ₂ cold source	2-35
2.15	Comparison of RB-8 neutron fluxes with and without HB-4 LH ₂ cold source	2-35
2.16	Comparison of VXF-22 neutron fluxes with and without HB-4 LH ₂ cold source	2-35
2.17	Summary of moderator vessel cases: hydrogen results	2-48
2.18	Summary of moderator vessel cases: vessel wall results	2-49
2.19	Results of vacuum tube analysis	2-54
2.20	Preliminary moderator vessel thicknesses	2-58
2.21	Strip sizes used to simulate moderator vessel caps	2-59

2.22	Probability of fracture (in log scale) at maximum membrane stress for the moderator vessel under irradiation and internal pressure varied from 10 bars to 25 bars	2-68
2.23	Probability of fracture (in log scale) at maximum bending stress for moderator vessel under irradiation and internal pressure varied from 10 bars to 25 bars	2-68
2.24	Stress intensity factors at $\phi = 0$ and at $\phi = 90$	2-69
2.25	Comparison of stress intensity shape factors obtained by Chang using the parametric expression with numerical results obtained by Neuman and Raju	2-72
2.26	Effect of radiation on F_0 and S_0	2-73
2.27	Path-to-failure curve data	2-73
2.28	Preliminary vacuum tube design dimensions	2-75
2.29	Summary of static stress solutions for the vacuum tube	2-78
2.30	Buckling load factors for the vacuum tube	2-78
2.31	Allowable buckling loads for vacuum tube	2-79
2.32	Impact of vacuum tube out-of-roundness	2-83
3.1	Selected triple-point data	3-13
3.2	FMEA for postulated cold source abnormal events	3-19
3.3	Insufficient boundary failure event frequency estimates	3-24
3.4	Pressure increase scenario frequency estimates	3-25
3.5	Boundary failure frequency estimates for events inside of Building 7900	3-28
3.6	Boundary failure frequency estimates for events outside of Building 7900	3-29
3.7	Recommended events for further analysis	3-31
3.8	Relationship between AEDC experiment tags and ATHENA cold source model node numbers	3-48
3.9	Sequence of events for transition to standby mode for a 4-bar liquid hydrogen operating condition	3-54
3.10	Sequence of events for transition to standby for supercritical hydrogen operating mode	3-58

3.11	Sequence of events for HD circulator trip transient for supercritical hydrogen operating mode	3-66
3.12	HB-4 cold source equipment classification results	3-99
4.1	Equipment procurement strategy	4-3
4.2	Breakdown of normal cold source operation costs	4-9
4.3	Procedural outline for system cool-down	4-11

ACRONYMS

1-D	one-dimensional
2-D	two-dimensional
3-D	three-dimensional
AEDC	Arnold Engineering Development Center
AIX	Advanced Interactive eXecutive
ALARA	as low as reasonably achievable
AN	annulus volume
ANL	Argonne National Laboratory
ANS	Advanced Neutron Source/American Nuclear Society
AOP	abnormal operating procedures
API	American Petroleum Institute
ASME	American Society of Mechanical Engineers
ATHENA	Advanced Thermal-Hydraulic Energy Network Analyzer
BJ	branch junction
BNL	Brookhaven National Laboratory
BOC	beginning of cycle
BWR	boiling water reactor
CCB	configuration control board
CEGB	Central Electricity Generation Board
CERN	European Organization for Nuclear Research
CFD	computational fluid dynamics
CMP	configuration management plan
C_p	constant pressure
CPU	central processing unit
CR	control rod
CS	cold source
DBE	design basis earthquake
DOE	Department of Energy
EOC	end of cycle
EOP	emergency operating procedures
FAB	feed and bleed
FAD	flaw assessment diagram
FEDC	Fusion Engineering Design Center
FMEA	failure modes and effects assessment
FSAR	final safety analysis report
FTE	full-time equivalent
HD	high-density
HEX	heat exchanger
HFBR	High Flux Breeder Reactor
HFIR	High Flux Isotope Reactor

HOG	hot off-gas
HP	health physicist
HPST	high-pressure storage tank
HS	heat structure
I&C	instrumentation and controls
IBC	insufficient boundary cooling
ID	identification/inner diameter
IKE	Institut Fur Kernenergetik Und Energiesysteme
ILL	Institut Laue-Langevin
INEEL	Idaho National Engineering and Environmental Laboratory
JV	valve junction
LANL	Los Alamos National Laboratory
LD	low-density
LH ₂	liquid hydrogen
LM	Lockheed Martin
LMES	Lockheed Martin Energy Systems, Inc.
LOCA	loss-of-coolant accident
LOSP	loss of off-site power
LOVA	loss of vacuum
MOC	middle of cycle
NASA	National Aeronautics and Space Administration
NIST	National Institute of Standards and Technology
NNS	nonnuclear safety
NOEC	no-observed-effect concentration
NRC	National Regulatory Commission
OHSP	Occupational Health and Safety Plan
Ops	operations
ORNL	Oak Ridge National Laboratory
PID	pipng and instrument diagram
PM	preventive maintenance
PORC	Plant Operations Review Committee
PRA	probabilistic risk assessment
PWR	pressurized water reactor
QA	quality assurance
QL	quality level
RB	removable beryllium
RELAP5	Reactor Linearized Analysis Program 5
RGA	residual gas analyzer
rpm	revolutions per minute
RRC	Research Reactor Committee
RRD	Research Reactor Division

SAR	safety analysis report
SBHE	special building hot exhaust
SBLOCA	small-break loss-of-coolant accident
SC	safety classification
SCDAP	Severe Core Damage Analysis Program
SS	steady-state
SSE	safe shutdown earthquake
STP	standard temperature and pressure
SV	single volume
TDJ	time-dependent junction
TDV	time-dependent volume
TSR	technical safety requirements
UPS	uninterruptable power supply
USQD	unresolved safety question determination
VFX	vertical experiment facilities
VJ	valve junction

1. INTRODUCTION

1.1 PROJECT PURPOSE AND SCOPE

In February 1995, Oak Ridge National Laboratory's (ORNL's) deputy director formed a group to examine the need for upgrades to the High Flux Isotope Reactor (HFIR) system in light of the cancellation of the Advanced Neutron Source Project. One of the major findings of this study was that there was an immediate need for the installation of a cold neutron source facility in the HFIR complex. In May 1995, a team was formed to examine the feasibility of retrofitting a liquid hydrogen (LH₂) cold source facility into an existing HFIR beam tube. The results of this feasibility study indicated that the most practical location for such a cold source was the HB-4 beam tube. This location provides a potential flux environment higher than the Institut Laue-Langevin (ILL) vertical cold source and maximizes the space available for a future cold neutron guide hall expansion. It was determined that this cold neutron beam would be comparable, in cold neutron brightness, to the best facilities in the world, and a decision was made to complete a preconceptual design study with the intention of proceeding with an activity to install a working LH₂ cold source in the HFIR HB-4 beam tube. During the development of the reference design the liquid hydrogen concept was changed to a supercritical hydrogen system for a number of reasons. This report documents the reference supercritical hydrogen design and its performance.

The scope of this project includes the development, design, procurement/fabrication, testing, and installation of all of the components necessary to produce a working cold source within an existing HFIR beam tube hole in the pressure vessel. All aspects of the cold source design will be based on demonstrated technology adapted to the HFIR design and operating conditions. The development of operating and maintenance procedures will also be covered by this project. The project will be considered complete when the facility is declared ready for normal operation.

The cold source project has been divided into four phases: (1) preconceptual, (2) conceptual design and testing, (3) detailed design and procurement, and (4) installation and operation. Although there is some time overlap between the phases, in general, they are sequential. The preconceptual design study was completed in early FY 1996 and the Preconceptual Design Report was issued in December 1995.¹ This report marks the conclusion of the conceptual design phase and establishes the baseline reference concept. Although testing of the concept will continue, this report provides the basis by which the detailed design of the HFIR cold source will proceed.

1.2 PROJECT GOALS AND REQUIREMENTS

Early in the development of the HFIR cold source concept a series of design goals and requirements were identified to establish a basis for measuring the adequacy of the design. These goals and requirements were developed based on input from the user community, HFIR operations, HFIR maintenance, and from other groups within the Research Reactor Division (RRD) responsible for issues involving detailed design, safety, regulatory compliance, and quality control. The remainder of this section provides a compilation of these goals and requirements.

1.2.1 User Requirements and Goals

Requirement 1: The purpose of the cold source is to increase the available neutron flux delivered to instruments at wavelengths from 4 to 12 Å. Optimization is to be based on the neutron brightness (/s/cm²/steradian/Å). The gain factor on brightness, as measured on HB-4, for these wavelengths should be comparable to existing cold sources of similar geometry (~10 to 20 at 7Å).

Requirement 2: Because there is an established experiment station that uses the HB-1 beam tube, which is close to the proposed cold source cryostat location, there should be no significant degradation of the thermal spectrum at HB-1 as measured after the cold source installation.

Requirement 3: Installation and testing of the cold source shall be designed and scheduled so as to cause minimum disruption to the beam-user program.

Goal 1: Delivery of the beam from the cold source to the reactor face should not perturb the source spectrum by more than 10% and should not restrict the available divergence.

1.2.2 Regulatory Compliance Requirements

Requirement 4: Installation of the cold source cryostat shall be performed per procedures written to the requirements of ORNL/RRD/INT-89, *The HFIR Operations Procedure Writer's Guide*,² and approved per ADM-0100, *Operations Procedures*,³ prior to the installation of the cold source. All installation and operations activities shall also be subject to the applicable requirements of the ORNL Directives and Guidance System of policies, procedures, and guidelines located on the electronic ORNL internal web page.

Requirement 5: Operating, abnormal, and emergency procedures written to the requirements of ORNL/RRD/INT-89 shall be approved per ADM-0100 prior to the installation of the cold source.

Requirement 6: Technical safety requirement modifications needed to support the operation of the cold source shall be approved and issued prior to installation.

Requirement 7: Expected maximum releases of tritium from the cold source when added to routine HFIR gaseous releases shall not exceed the requirements of Technical Specification 3.19.⁴ In addition, radiation protection issues will be evaluated according to the as low as reasonably achievable (ALARA) principle and will meet the applicable requirements of 10 CFR 835.

Requirement 8: The requirements of ORNL Procedure EPM-2.1, *Environmental Protection and NEPA Compliance*,⁵ shall be completed prior to the detailed design of the cold source. In addition, general environmental protection issues will be subject to the applicable environmental protection sections of the electronically based ORNL Directives and Guidance System.

Requirement 9: *The HFIR Safety Analysis Report (SAR)*⁶ shall be revised to reflect inclusion of the cold source at the SAR update following installation of the cold source.

Requirement 10: *The HFIR Probabilistic Risk Assessment (PRA)*⁷ shall be revised to reflect inclusion of the cold source at the PRA update following installation of the cold source.

1.2.3 Safety Requirements and Goals

The HFIR cold neutron source facility will be designed such that there is a low probability (less than 1×10^{-6} per year best estimate frequency) that neither the reactor nor the public will be endangered by accidents that could occur within the cold source or as a result of the cold source facility interacting with the reactor or its safety systems. In addition, the design and operation of the HFIR cold source will follow National Aeronautics and Space Administration (NASA) guidelines for dealing with hydrogen. Thus, the risk to cold source system workers at the HFIR site is expected to be consistent with normal industrial

hazards associated with cryogenic systems. To achieve this overall safety goal, a number of requirements and goals are proposed, as described below.

Requirement 11: Under normal operating conditions, the hydrogen shall be contained by a combination of barriers so that no single failure would result in air or water contacting the hydrogen. For the portion of the system inside the reactor vessel, this should be accomplished by a combination of a high-integrity hydrogen circulating system surrounded by a vacuum and a vacuum vessel. Under abnormal operating conditions (including conditions of degraded hydrogen system cooling, loss of vacuum barrier, loss of inert gas barrier, loss of off-site power, or hydrogen leakage), at least one containment barrier shall exist.

Requirement 12: Materials selected for use in the hydrogen system shall be compatible with the low temperature environment and be resistant to hydrogen embrittlement. Quality requirements for the hydrogen system materials shall be comparable to those currently in use for the reactor primary coolant pressure boundary material.

Requirement 13: Very high purity hydrogen and inert gas shall be used in the cold source to preclude oxygen contamination of either the primary hydrogen system or the inert gas blanket system. The gases shall be rigorously tested prior to use in the facility. Administrative controls shall be in place during gas transfers to preclude oxygen intrusion and unmonitored/uncontrolled heating of cryogenic fluids.

Requirement 14: The vent system shall be designed to prevent the formation of an explosive cloud or the back-flow of air into the vent system. In the event of an ignition of the hydrogen in the vent system, no reactor safety system will be affected to the extent that its safety function is not performed. A vent from the He- refrigeration system shall be independent of the main vent from the liquid hydrogen loop or be shown by analysis to be no threat to successful hydrogen venting in all cases.

Requirement 15: Vessels that comprise the primary hydrogen barrier shall be designed and fabricated using the guidance of the *ASME Boiler and Pressure Vessel Code*.⁸ Piping that comprises the primary hydrogen barrier shall be designed and fabricated following the guidance of ASME B31.3, *Chemical and Petroleum Plant Piping*.⁹

Requirement 16: Components that form any of the multiple hydrogen boundaries inside the reactor building shall be seismically qualified or shall be shown by analysis to maintain hydrogen containment integrity following a design basis earthquake (DBE). The hydrogen vent system handling hydrogen leaks from equipment inside the reactor building shall be designed to withstand seismic and wind/tornado events. The vacuum tube helium flooding system shall be seismically qualified. Support structures and systems located within the reactor building that ensure the normal functions of the above cold source systems shall also be qualified seismically. Active components of the hydrogen system and the helium refrigeration system shall not be required to function during and following a DBE and thus do not require seismic qualification. All cold source instrumentation and operation controllers shall be designed to "fail safe;" [i.e., depressurization and H₂ inventory expulsion into the storage tank should result automatically upon any controller failure (including failure due to a DBE)]. In addition, any cold source loop instrumentation interfacing with the HFIR safety system shall be designed to fail (seismic or nonseismic) in a manner such that the resulting signal ensures reactor shutdown.

Requirement 17: The vacuum tube shall be designed to withstand the external pressure produced by a leaking beam tube. Proposed modifications of Technical Specification 3.8.3(b)(1)¹⁰ specify a value of 690 psig as the bounding external pressure. During the HFIR hydrotest, special measures shall be taken to ensure the integrity of the vacuum tube. These measures may include purging the system of hydrogen, pressurizing the vacuum tube, and post-hydrotest leak testing of the vacuum tube.

Requirement 18: The consequences of events that involve beam tube failure shall be analyzed and shown to be less than the HFIR SAR acceptance criteria. For limiting events, which include low probability sequences in the frequency range of 10^{-4} to 10^{-6} per year, the consequence must be lower than the 10 CFR 100 off-site dose limits. For beam tube-failure-initiated events, proposed modifications to Technical Specification 3.8.3(b)(2)¹¹ limit the consequences to a small-break loss-of-coolant accident (SBLOCA) of no more than 100 gpm at 690 psig.

Requirement 19: The cold source facility shall be designed in conjunction with the beam tube to withstand a hypothetical hydrogen-oxygen reaction that could occur in the event air were introduced into the vacuum tube concurrent with a release of hydrogen. For the in-vessel portions of the cold source, the combination of the beam tube and vacuum tube shall be sufficient to withstand the reaction. A credible energy release is determined based on design basis failure modes. For the ex-vessel portions of the cold source, the beam tube alone shall be designed to withstand the effects of a reaction within the vacuum tube.

Requirement 20: If the helium refrigerator system pressures are set lower than the hydrogen system pressure, adequate means for detecting hydrogen in the heat exchanger exhaust stream shall be provided, in the event of a heat exchanger leak. In addition, adequate pressure relief shall be provided on the helium side to deal with the higher pressure source from the hydrogen side, in the event of a heat exchanger tube break.

Goal 2: Operational interfaces between the cold source and the reactor should be minimized. In particular, the classes of cold source system events that require a reactor shutdown are limited to ones that may threaten (directly or indirectly) primary and secondary reactor coolant pressure boundaries. Although a reactor scram is a controlled transient, it is still a transient and therefore places demands on system equipment. A significant increase in the number of reactor scrams per year would clearly have some safety implications. When appropriate, automatic reactor power setbacks should be considered as an alternative to a reactor scram.

1.2.4 Operational Requirements and Goals

The HFIR is a fully functional facility that has operated for more than 30 years. As a result, there are policies, procedures, and guidelines that have been long established for the day-to-day business activities. The following requirements and goals have been established to assure that the cold source facility is compatible with these practices.

Requirement 21: The cold source and its ancillary equipment shall be designed, procured, and installed per the requirements of RRAP-3.2, *Configuration Control of Plant Design Modifications*.¹² Per RRAP-3.2, procurement requirements are specified in RRAP-1.9, *Procurement of Items and Services*,¹³ and fabrication requirements are specified in RRAP-3.3.1, *Procedure for Fabrication Control Governing Modification of Reactor Systems and/or Replacement Hardware*.¹⁴

Requirement 22: The cold source shall be designed assuming reactor operation at 100 MW.

Requirement 23: The cold source shall be operated by Research Reactor Division (RRD)/HFIR operations personnel.

Requirement 24: Instrumentation shall be designed so that abnormal operating procedures (AOPs) and emergency operating procedures (EOPs) associated with the protection of the cold source that require immediate actions are capable of being performed from (1) the cold source control room independent of

any other location or (2) the HFIR control room independent of another location. Follow-up operator actions may require activities at the location of the cold source equipment.

Requirement 25: Instrumentation readouts and controls shall be calibrated in engineering units mutually agreed upon by the cold source design team, the cold source safety analysis team, and an RRD-designated operations representative.

Requirement 26: The cold source support equipment layout shall be approved by the HFIR operations manager to ensure that access to existing equipment required for operations is not blocked or impeded.

Requirement 27: All cold source operators shall be trained on the operations of the cold source prior to its initial operation.

Requirement 28: The installation of the cold source cryostat shall be performed under an approved plan consistent with the ALARA concept embedded in Requirement 7 of Sect. 1.2.2. Tools shall be designed and procedures shall be written to ensure acceptable doses are not exceeded as a result of this job, based on the ALARA assessment.

Requirement 29: Cold source operators and craftsmen involved in the installation of the cold source cryostat shall be trained on the installation procedures prior to installation.

Requirement 30: All installation and testing activities at HFIR shall be scheduled through the plan-of-the-day process and the outage planning process.

Requirement 31: The results of all preoperational testing shall be reviewed and approved by the HFIR operations manager and the HFIR section head prior to initial start-up with the cold source operational.

Requirement 32: Operational testing of the cold source, including initial reactor start-up and ascension to power, shall be performed per procedures written to the requirements of ORNL/RRD/INT-89 and approved per ADM-0100. Procedures shall be reviewed and approved prior to performance of activities requiring their use.

Requirement 33: Cryostat removal procedures shall be developed, approved, and issued prior to the initial operation of the cold source equipment.

Goal 3: Radiological conditions in the beam room and experiment room should be maintained at current levels or improved.

Goal 4: The cold source should be designed to minimize administrative reactor shutdowns to protect cold source equipment (the goal shall be <1 reactor shutdown per year).

Goal 5: The need for a backup power supply (e.g., diesel generator) for the cold source should be evaluated to assure reliable operation.

Goal 6: The cold source refrigeration system design should consider the HFIR secondary coolant system as a heat sink, provided the tower is capable of handling the additional load.

Goal 7: The design of the cold source control and automated safety systems should be such as to eliminate the need for dedicated cold source staff during off-shifts.

Goal 8: Instrumentation and controls, valves, and switches shall be located for easy access by operators on routine shift checks and operators manipulating the controls of the cold source as determined by the HFIR operations manager.

1.2.5 Maintenance Requirements

Requirement 34: Installation of all ex-core cold source components shall be administered through the RRD work package system per RRD-M-1.6, *Preparing and Processing a Maintenance Work Package*,¹⁵ and RRD-M-1.14, *Maintenance Work Control*.¹⁶

Requirement 35: Breakdown maintenance procedures shall be developed, approved, and issued prior to initial reactor operation with the cold source cryostat installed.

Requirement 36: Maintenance personnel shall be trained on the routine and breakdown maintenance procedures prior to the initial reactor operation with the cold source cryostat installed.

Goal 9: An appropriate spare parts inventory (part, maximum stock, reorder point, and storage requirements) of cold source equipment should be developed and stocked to ensure timely maintenance action in the event of equipment failure.

Goal 10: A preventive/reliability-centered maintenance program for cold source equipment should be developed and implemented prior to initial operation.

1.3 PROJECT SCHEDULE

A project schedule has been developed as presented in Fig. 1.1. This schedule is based on the assumption that the cold source components to be located inside the HB-4 beam tube will be installed during the six-month reactor outage currently scheduled to begin in February 2000 to replace the reactor beryllium reflector. This will minimize the impact on existing user programs.

A number of factors will affect the ability to meet this schedule: the timing of the funding, the number of problems encountered in the testing program, the efficiency of the procurement activities, and the quality and timing of the review process. Special emphasis will be made in the planning of these activities to minimize the risk in the schedule.

It should be noted that the status bars on the schedule represent the state of the project as of August 1, 1997. At this time, the project is on schedule in all activities with the exception of the Research Reactor Committee (RRC) approval in principle task. This activity has been delayed because mechanical problems with the reactor (unrelated to the cold source project) that have monopolized key HFIR staff members needed to initiate this activity. It is recognized that this is a key activity that will quickly lead to a hold on key procurements if substantial additional delays in this task materialize.

In addition to the internal laboratory review groups and the U.S. Department of Energy (DOE) regulatory review group, a group external to ORNL and the HFIR cold source project was formed to provide important technical review of the project. This group is composed of subject matter experts with significant experience with hydrogen cold sources in reactors and/or the safety requirements for dealing with low temperature hydrogen. Although not shown at the level of schedule presented in Fig. 1.1, this group is scheduled to meet at key times in the progression of the cold source project to provide important review (prior to the start of conceptual design, prior to the start of detailed design, prior to the start of installation, and prior to the start of operation). These reviews are, in general, considered hold points in the project with the exception that holds are not placed on perceived low-risk long-lead procurement items that must proceed on a different schedule to support the project.

Fig. 1.1. HFIR cold source project schedule.

As noted in Fig 1.1, installation activities that can be performed while the reactor is operating are scheduled to be completed well before the reactor shutdown for the beryllium changeout. This provides extra schedule slack time to assure that these components will be installed, tested, and ready for use when they are needed.

As stated at the beginning of this section, this schedule is based on the planned scheduled outage for the reactor beryllium reflector replacement. If for any reason the beryllium changeout schedule is modified, the schedule for this project will be revised so that the schedules are consistent.

1.4 REFERENCES FOR CHAPTER ONE

1. D. L. Selby et al., *High Flux Isotope Reactor Cold Source Preconceptual Design Study Report*, ORNL/TM-13136, Lockheed Martin Energy Systems, Inc., Oak Ridge Natl. Lab., December 1995.
2. R. A. Crone, comp., *The High Flux Isotope Reactor Operations Procedure Writer's Guide*, ORNL/RRD/INT-89, Martin Marietta Energy Systems, Inc., Oak Ridge Natl. Lab., March 1993.
3. *Research Reactors Division HFIR Operations Procedures*, ADM-0100, Operations Procedures, ORNL/RRD/INT-99, Lockheed Martin Energy Systems, Inc., Oak Ridge Natl. Lab., October 1995.
4. *High Flux Isotope Reactor Technical Specifications*, 6.13, Radiological Effluent Monitoring, ORNL/TM-5711/R2, Martin Marietta Energy Systems, Inc., Oak Ridge Natl. Lab.

5. *ORNL Environmental Protection Manual*, EPM-2.1, Rev. 0, Environmental Protection and NEPA Compliance, Martin Marietta Energy Systems, Inc., Oak Ridge Natl. Lab., May 1993.
6. *High Flux Isotope Reactor Safety Analysis Report*, ORNL/M-2344, Vols. 1-5, Martin Marietta Energy Systems, Inc., Oak Ridge Natl. Lab.
7. D. H. Johnson et al., *The High Flux Isotope Reactor Probabilistic Risk Assessment*, ORNL/RRD/INT-36/R1, Martin Marietta Energy Systems, Inc., Oak Ridge Natl. Lab., August 1991.
8. "Code Case N-519—Use of 6061-T6 and 6061-T651 Aluminum for Class 1 Nuclear Components," *American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code*, Section III, Division 1, January 1994.
9. *Chemical Plant and Petroleum Refinery Piping*, ASME b31.3, American Society of Mechanical Engineers, 1993.
10. High Flux Isotope Reactor Technical Specifications, 3.8, Limitations on Experiments, 3. Specifications, part b, paragraph (1), ORNL/TM-5711/R2, Martin Marietta Energy Systems, Inc., Oak Ridge Natl. Lab.
11. High Flux Isotope Reactor Technical Specifications, 3.8, Limitations on Experiments, 3. Specifications, part b, paragraph (2), ORNL/TM-5711/R2, Martin Marietta Energy Systems, Inc., Oak Ridge Natl. Lab.
12. *Research Reactors Division Administrative Policies and Procedures Manual*, RRAP3.2, Configuration Control System, ORNL/RRD/INT-12, Rev. 6, Martin Marietta Energy Systems, Inc., Oak Ridge Natl. Lab.
13. *Research Reactors Division Administrative Policies and Procedures Manual*, RRAP-1.9, Procurement of Items and Services, ORNL/RRD/INT-12, Rev. 6, Martin Marietta Energy Systems, Inc., Oak Ridge Natl. Lab.
14. *Research Reactors Division Administrative Policies and Procedures Manual*, RRAP-3.3.1, Procedure for Fabrication Control Governing Modification of Reactor Systems and/or Replacement of Hardware, ORNL/RRD/INT-12, Rev. 6, Martin Marietta Energy Systems, Inc., Oak Ridge Natl. Lab.
15. *RRD Maintenance Manual, Volume 1, Maintenance Program*, MMP-0801, Preparing and Processing a Maintenance Work Package, Rev. 2, ORNL/RRD/INT-42, Lockheed Martin Energy Systems, Inc., Oak Ridge Natl. Lab.
16. *RRD Maintenance Manual, Volume 1, Maintenance Program*, MMP-0800, Maintenance Work Control, Rev. 1, ORNL/RRD/INT-42, Lockheed Martin Energy Systems, Inc., Oak Ridge Natl. Lab.

2. REFERENCE CONCEPT DESIGN

2.1 GENERAL SYSTEM DESIGN

The reference design (see system diagram Fig. 2.1) uses hydrogen gas at a supercritical pressure of 15 bar (217.6 psia) that enters the cold source moderator vessel at 18 K and leaves at approximately 21 K (see temperature and entropy curves in Fig. 2.2). A flow rate of 1 L/s is maintained by a mechanical circulator that is backed up by an installed standby that can be brought on-line remotely from the control center. This redundancy increases the cold source availability and provides an option to replace a failed circulator while the system remains in operation.

The thermophysical properties of supercritical hydrogen at 15 bar (217.6 psia) and liquid hydrogen at 4 bar (58 psia) are reproduced in Tables 2.1 and 2.2 for comparison. The major advantage of supercritical hydrogen is total freedom from problems associated with two-phase fluid during cool-down, warm-up, transitioning to a standby state or possible off-normal transients. It also eliminates any possibility of local boiling in the moderator vessel and permits more flexibility in its operating temperature. This more than compensates for the increased pressure and the resulting increased heat load from thicker structure walls. Although the supercritical option represents the reference design, the system could revert to a liquid hydrogen system with little or no modification to the design (note that some additional testing and system qualification would be required), if the need were to arise.

2.1.1 Operating Parameters

The total heat load has been calculated to be slightly more than 2 kW, of which about 1.3 kW represents heat generated in the aluminum moderator vessel walls, with the majority of the remaining heat load being generated in the hydrogen itself. A contingency of 200 W is allowed for intrinsic heat loads and nuclear heating of the feed and return lines close to the core. The fluid flow rate of 1 L/s was chosen in order to generate velocities within the moderator vessel that provide adequate heat transfer between the fluid and the vessel walls. The circulator speed is adjustable by about $\pm 20\%$, allowing the system to be finally trimmed as necessary. A standby cooling system protects the vessel from overheating, even at maximum reactor power, in the event of a loss of primary refrigerator cooling or circulators.

2.1.2 System Configuration

A vacuum envelope that is split into two sections insulates all cold components of the loop. One section is dedicated to the moderator plug assembly and the other to the remainder of the loop. This would limit the spread of contamination in the event of a system failure in which hydrogen was released into the vacuum space. The entire hydrogen system, including the vacuum chambers and their pumping systems, is further enclosed by a continuous inert blanket. This blanket is continuous but is segmented at different areas of the system. The inert medium depends on the location within the system, but is either gaseous helium, gaseous nitrogen gas, or, where appropriate, a secondary vacuum. The blanket surrounding the moderator vacuum vessel is an exception and is composed of water entrained with helium gas, the water acting as a heat transfer medium while the helium maintains a leak detection capability. The inert blanket provides double isolation of hydrogen from atmospheric air and also prevents air from entering insulating vacuum spaces where it could cryopump on to cold surfaces, making detection difficult or even impossible. Also, in areas of high radiation, such frozen air could break down to form hazardous unstable free radicals. The main loop is protected from pressures above 19 bar (275.6 psia) by a rupture disc that exhausts into a vent stack that is dedicated to the cold source system. Each vacuum chamber and sector of the inert blanket are independently protected against internal

Fig. 2.1. HFIR cold source system flow diagram.

Table 2.1. Supercritical hydrogen gas at 15 bar (217.6 psia)

Temperature (K)	Density (gms/L)	Cp (J/gm.K)	Viscosity (μ Pa/s)
17	75.62	7.407	19.62
18	74.75	7.875	17.83
19	73.83	8.393	16.32
20	72.86	8.934	15.01
100	3.643	13.89	4.298
120	3.014	15.29	4.847

Table 2.2. Liquid hydrogen at 4 bar (58 psia)

Temperature (K)	Density (gms/L)	Cp (J/gm.K)	Viscosity (μ Pa/s)
17	74.56	7.632	18.26
18	73.61	8.152	16.58
19	72.6	8.729	15.15
100	0.9705	13.53	4.211
120	0.8701	15.05	4.781

Fig. 2.2. HFIR cold source temperature vs entropy curves.

pressures exceeding 4 bar (58 psia). All rupture discs, relief valves, and vacuum pump exhausts are connected to the dedicated vent system (see Sect. 2.3.11.).

2.2 SYSTEM OPERATING CONDITIONS

2.2.1 Normal Operation

The hydrogen system operates in a closed loop that remains at a nominally constant pressure at all temperatures. A summary of the general operating parameters is provided in Table 2.3. At ambient temperature, the majority of the hydrogen inventory resides in the large storage vessel, which has a capacity of 557 ft³ and is connected to the loop by a control valve. The overall pressure of the system, including the storage vessel, in this state is 4 bar (58 psia). At the start of a cool-down, the loop pressure is increased to its operational 15 bar (217.6 psia). The fluid density within the cold loop increases with falling temperature, but pressure is maintained by gas from the feed vessel, which is also connected to the loop by a second control valve. The pressure in the feed vessel is maintained at 1.6 bar (23.2 psia) by a transfer pump that draws gas from the storage vessel. During cool-down, pressure in the storage vessel is reduced accordingly until it reaches partial vacuum when the loop attains normal operating conditions. Pressure in the loop is controlled in response to a pressure sensor in the loop. The feed and vent valves operate on either side of a dead band (within which both are closed). The operating circulator is constantly monitored for abnormal current, excess vibration, reduction in speed, or high motor temperature. Any of these could indicate the development of a fault and, under certain circumstances, will initiate a start-up of the spare circulator and the shutdown and isolation of the operating one. Once off-line, a faulty circulator can be warmed to ambient temperature, evacuated, and filled with helium gas ready for replacement. All operations to this point are carried out remotely from the control station, but if it is decided to replace the circulator while the system remains in operation, this is possible using the following operating procedures (as applied to the configuration as shown in Fig. 2.3).

Table 2.3. Normal system operating parameters

Parameter	Value
Total hydrogen mass	5.04 kg
Equivalent gas volume at STP	2094 ft ³
Storage vessel volume	557 ft ³
Nominal pressure	15 bar (217.6 psia)
Freezing temperature of hydrogen at nominal pressure	14.3 K
Main rupture disc pressure setting	19 bar (275.6 psia)
Vacuum and inert blanket rupture disc setting	4 bar (58 psia)
Nuclear heat load	2.01 kW (1.42 kW in structure and 0.59 kW in H ₂)
Nominal hydrogen inlet temperature	18 K
Nominal hydrogen outlet temperature	21 K
Normal operational flow	73.83 g/s at 15 bar
Overall system pressure drop	0.1 MPa

1. Lock both outer isolation valves in the closed position to prevent them being remotely operated.
2. Close both inner valves (which is only possible locally).
3. Replace the circulator drive unit assembly.
4. Evacuate the new unit and backfill it with helium gas.
5. Open the inner valves and unlock the outer valves.
6. Purge again (from the control station) with helium gas.
7. Evacuate and fill the circulator with hydrogen gas.
8. Open the cooling loop and precool the circulator to operating temperature.

Fig. 2.3. Piping and valve connections to the high density circulators.

2.2.2 Standby Operation

If a problem necessitates shutting down the cold source, it can be put into a standby mode that controls the moderator vessel temperature at an acceptable level, even with the reactor at full power. This is achieved by circulating hydrogen gas at 4 L/s and 100 K with a low density circulator. At this temperature, the hydrogen gas density is only about 20% of that at 20 K. Cooling of the hydrogen to 100 K is effected using the passive liquid nitrogen precooling stage of the refrigerator. With the active section of the refrigerator bypassed, it can be shut down and serviced, leaving the system operating in standby. The objective of the standby state is to allow the reactor to continue in full power operation in the event of the more common cold source failure modes, such as refrigerator or circulator problems, though other applicable failure modes might be identified when the system becomes operational. This

reduces the impact of cold source failure on other user activities at HFIR. A summary of the standby condition general operating parameters is provided in Table 2.4.

The low density circulator also serves two other purposes. During the initial cool-down, it assists cool-down of the main loop components. Switchover to the high density circulator is made when current drawn by the low density circulator motor rises to a preset upper limit. Also, at the end of each running cycle, annealing of the moderator vessel at ambient temperature is required to decrease the effects of radiation damage to the vessel. This uses residual reactor heat, following the shutdown, to heat up the vessel. The temperature rise of the vessel is controlled by gas flow provided by the low density circulator in a manner similar to that of a standby state, but at a higher set point.

Table 2.4. Standby operating parameters

Parameter	Value
Total heat load (vessel walls only)	1.42 kW
Nominal hydrogen gas inlet temperature	100 K
Nominal hydrogen gas outlet temperature	120 K
Volumetric flow rate	4 L/s
Mass flow rate at nominal pressure	14.57 g/s
Overall system pressure drop	0.21 bar (3 psi)

2.3 INDIVIDUAL MODULE DESCRIPTIONS

The overall cold source system is shown in Fig. 2.4. The system is designed on a modular basis using eleven basic units described below.

2.3.1 Moderator Plug Assembly

2.3.1.1 Functional Requirements

The moderator plug assembly is a cylindrical vacuum chamber that is bolted and sealed into the forward end of the reactor beam tube. Its function is to contain and accurately retain the cold source vessel in its correct position and to provide vacuum insulation for the moderator and hydrogen service pipes. The outer vessel plug serves as a secondary hydrogen barrier in the event of a moderator vessel failure and a secondary reactor water barrier in the event of beam tube failure. All components must meet safety, seismic, and quality level requirements as defined in Sect. 3.6 of this report.

2.3.1.2 General Description

A diametral clearance is required between the vacuum tube and the beam tube walls to ensure the removal capability of the moderator plug assembly. Heat generated in the vacuum vessel has to be transferred to the beam tube cooling system, and a helium/water mixture is circulated through the gap to facilitate this. The water provides conduction across the gap, and the helium provides vacuum leakage monitoring for the vacuum tube. The validation of the saturated helium/water system performance is the subject of a development program at the University of Tennessee, Knoxville.

Fig. 2.4. Cold source facility layout.

2.3.1.3 Components of the Moderator Plug Assembly Module

Vacuum vessel

The vacuum tube is hemispherical at its forward end and has a window at its outer end to allow cold neutrons to exit to the beam line. The wall thickness and the sealing arrangements will be designed to withstand full reactor coolant water pressure in the annulus, should the beam tube fail.

Moderator vessel

The moderator vessel (see Fig. 2.5) is rigidly mounted to prevent contact with the inside of the vacuum tube. It is sized based on physics analysis to provide optimal modification of the neutron spectrum when filled with supercritical hydrogen at a pressure of 14 bar (203.1 psia). The two cryogenic pipes provide support for the vessel, but allowance is made for differential shrinkage between them. In the proposed configuration, the upper return pipe constitutes the main support member. The lower feed pipe incorporates a loop with two short bellows (that are restrained to allow bending only) at each end of the loop to allow axial flexibility of the pipe. Both tubes are anchored to the inside of the beam tube by thin-walled concentric tubes that act as a thermal break. The anchor point is distanced from the reactor core to limit neutronic heating in the material, and the cantilevered pipe allows the moderator vessel to move axially about 1 mm as the pipes change dimension with temperature. Design of the vessel is optimized to ensure adequate heat transfer over its entire inner surface. It must have minimum mass to limit the heat load and must also reflect manufacturing limitations.

A three-dimensional (3-D) physics analysis has been performed to determine the optimal geometry of the moderator vessel and to supply heat loads for thermal analysis. These evaluations are summarized in Sect. 2.4 of this report.

Flow analysis using the CFX-4¹ computational fluid dynamics code (formerly known as FLOW3D) has been carried out to map fluid velocities and temperatures over the entire inner surface of the moderator vessel. This work is described in Sect. 2.5.

The moderator vessel has been designed to the guidance of the ASME Boiler and Pressure Vessel Code Section III, Division I, Class I, Code Case N-519 for aluminum 6061-T6 for all operating

Fig. 2.5. HFIR cold source moderator vessel.

conditions, although its small size excludes it from mandatory compliance. A finite element stress analysis has been carried out for this vessel and is described in Sect. 2.6.

Industrial advice on machining and fabrication techniques has also been solicited in an attempt to produce a vessel that represents the best possible compromise of the above requirements.

Feed lines

From the feed pipe anchor point, both pipes continue back to pass through the vacuum vessel flange. They then transition to a dual concentric configuration, which includes vacuum insulation and inert blanket. This multiple pipe assembly forms an integral part of the plug assembly, which will require special tools and support jigs for installation. However, since the moderator plug assembly is replaceable only as a complete unit, the concentric pipe will be cut off close to the flange for removal. This will make the removal and disposition of the more active main body of the assembly easier.

Neutron window

The outer end of the vacuum tube is closed by a double-walled aluminum window with helium gas between. This window must have as little impact on the cold neutron current as possible, but also be able to withstand a positive pressure of 4 bar (58 psia) if the primary hydrogen containment failed.

2.3.2 Pump Module

2.3.2.1 Functional Requirements

The pump module provides a supercritical hydrogen flow of 1 L/s under normal operating conditions and a hydrogen gas flow of 4 L/s in a standby state. It also provides the main temperature and pressure control and monitoring functions for the system. The composite vacuum and inert blanket vessel must have inlet and outlet cryogenic connectors to match the main transfer lines that connect it to the moderator plug assembly and the heat exchanger module (see Figs. 2.6, 2.7, and 2.8). All components must meet safety, seismic, and quality level requirements as defined in Sect. 3.6 of this report.

Fig. 2.6. Pump module top view.

Fig. 2.7. Pump module side view (dimensions in inches).

2.3.2.2 Components of the Pump Module

High density circulators

Two high density circulators are installed, one of which is a redundant unit. During normal operation, the circulator in use is monitored for excessive vibration, abnormally high motor current, falling rotational speed, and abnormally high winding temperature. As previously mentioned, the redundant unit will be started and the operating unit shut down, if readings extend beyond the established operating band.

Circulator isolation valves

Two valves are fitted in both inlet and outlet ports of each high density circulator. This doubly isolates the circulator from the hydrogen loop allowing its on-line replacement while the system remains in operation. The outer valves incorporate a pressure relief valve to vent, in a safe manner, hydrogen trapped between the inner and outer valves when both are closed.

Low density circulator

A single low density circulator is mounted into an appendage to the main vessel that shares the same vacuum insulation. Single isolation valves allow it to be replaced when the system is not operational. The intermittent duty of this circulator makes on-line replacement an additional expense that is not justified.

Fig. 2.8. Pump module piping diagram.

Cold loop temperature sensors

One sensor is situated just downstream of the heat exchanger and provides temperature control of the hydrogen during normal operation. Further sensors provide temperature indication at various points of the loop. All sensor leads pass through seals in a vessel flange, making them replaceable without opening up the vacuum system.

Cold loop pressure sensors

One sensor, located just upstream of the circulators, is used to control the loop pressure. Further sensors provide pressure indication at various parts of the loop. Small bore tubes transmit pressure from the take-off point to transmitters, all of which are grouped together on a flange mounted to the pump module vacuum vessel. This allows individual transmitters to be replaced without a need to open the vacuum system.

2.3.3 Heat Exchanger Module

2.3.3.1 Functional Requirements

The heat exchanger comprises a vacuum insulated interface between the hydrogen loop and the refrigerator. Temperature sensors within the module provide temperature control during cool-down to prevent possible freezing of the hydrogen.

The module must fulfill two distinct operational functions:

1. Cool hydrogen at 15 bar (217.6 psia) from ambient temperature to operating temperature over a period of about 12 h. (This is considered a reasonable time period to cool the components of the loop.)
2. Maintain the supercritical gas within its normal operating temperature set points with a flow of 1 L/s or maintain gas at standby temperatures.

All components must meet safety, seismic, and quality level requirements as defined in Sect. 3.6 of this report.

2.3.3.2 Components of the Heat Exchanger Module

Heat exchanger

The heat exchanger is of the aluminum core type used in the construction of cryogenic refrigerators. This is a very compact type of heat exchanger with a very small fluid inventory of about 1.6 L. In normal operation, the cooling channel (helium gas) operates at 14–20 K at a pressure of between 1–2 bar and a fluid mass flow rate of about 80–90 g/s. The cooled channel (hydrogen) operates at about 18–22 K at a pressure of 15 bar (217.6 psia) and a volumetric flow rate of 1 L/s. In standby, the cooling helium is about 80–90 K with a volumetric flow rate of 13 L/s, and the hydrogen is 100–120 K and 1.5 MPa with a volumetric flow rate of 4 L/s. A temperature sensor very close to, or even within, the heat exchanger itself will provide temperature control during the cool-down to eliminate any possibility of hydrogen freezing while the gas density and specific heat are low.

Containment vessel

The containment vessel is double-walled. The inner wall provides vacuum insulation for the heat exchanger, and the outer wall provides space for an inert blanket. Both vacuum and inert blanket walls are designed to withstand a positive pressure of 4 bar (58 psia) (relief pressure of the rupture disc) in the event of a primary containment failure. The vessel must have bayonet connectors to match the transfer lines that connect it to the refrigerator cold box and to the pump module. An additional outer flange will facilitate continuity of the inert blanket over the transfer line to the cold box.

2.3.4 Gas-Handling System

2.3.4.1 Functional Requirements

The gas-handling module provides storage of the hydrogen inventory at ambient temperature and controls its movement and pressure as required. All components must meet safety, seismic, and quality level requirements as defined in Sect. 3.6 of this report.

2.3.4.2 General Description

A schematic of the gas-handling module is shown in Fig. 2.9. The hydrogen inventory is a fixed mass of gas that, once charged into the system, is double-isolated from the atmosphere. All hydrogen gas bottles are removed once the system has been charged, and the only reasons for replacement of the gas are either service requirements or a failed rupture disc. The storage vessel adds sufficient volume to the loop to allow the hydrogen inventory to expand at ambient temperature to a pressure of 4 bar (58 psia).

Fig. 2.9. Hydrogen gas-handling module.

At the start of a cool-down the gas-handling system raises the loop pressure to 15 bar (217.6 psia) by compressing gas from the storage vessel. As cooling proceeds and its density increases, further gas is transferred in from the storage vessel to maintain the loop pressure. The vessel is sized such that under normal operating conditions, the storage vessel is reduced to a partial vacuum. The gas-handling system thereafter admits, or vents, gas back to the storage vessel to keep the loop pressure within ± 0.1 bar of its set point. The main storage vessel is designed for a maximum pressure of 19 bar (275.6 psia) (primary system rupture disc pressure), but it must also be double-walled to incorporate the inert blanket. The valves and feed vessel that comprise the remainder of the gas-handling system are housed in a vessel that constitutes inert blanketing. The gas transfer compressor/pump, which is also inert-blanketed, resides outside and close to the storage tank. All interconnecting pipework is double-walled to provide continuity of the inert blanket. The inert blanket medium for the gas-handling system is a secondary vacuum.

2.3.5 Refrigerator Module

2.3.5.1 Functional Requirements

The refrigerator provides the cooling power required by the loop. It has a maximum rating of 3.5 kW with a refrigerator return temperature of 20 K, and the refrigerant helium gas is interfaced with the loop by the heat exchanger module (see Sect. 2.3.3). All components must meet safety, seismic, and quality level requirements as defined in Sect. 3.6 of this report.

2.3.5.2 General Description

The refrigerator system comprises five major equipment items: a bank of five helium screw compressors, a vacuum-insulated cold box that contains all interstage heat exchangers, a second coldbox containing four two-cylinder piston expanders, a motor control center, and an inventory control system. Refrigeration power depends on the number of compressors and expanders used, but fine operational control is provided by in-line heaters in the helium refrigerant circuit. The complete refrigerator system is located in a dedicated separate structure outside of the reactor building.

2.3.6 Transfer Lines Module

2.3.6.1 Functional Requirements

Transfer lines are required to transport cryogen between discrete components of the system. All must be vacuum-insulated and those carrying hydrogen must also incorporate an outer helium blanket. The vacuum will be constantly pumped by the respective pump station. All components must meet safety, seismic, and quality level requirements as defined in Sect. 3.6 of this report.

2.3.6.2 General Description

Transfer lines can be configured in many ways, using vacuum-insulated single-pipe runs or with flow and return pipes within a common vacuum tube. These lines can be made from rigid or flexible tube and can be coupled using proprietary cryogenic fittings. Conversely, special couplings, based on an acclaimed design known as the "Johnston" coupling, are frequently used. This design is able to effect a single cryogenic pipe joint using an ambient temperature seal. A classic development in transfer line design was to nest flexible spirally convoluted flow and return tubes one inside of the other with vacuum insulation between them and insulating spacers to prevent the cold lines from contacting the outer vacuum tubes (see Fig. 2.10). This allows a combined flow and return coupling to be configured. Although more difficult to build, this design offers some notable benefits:

1. It has the smallest overall cross-sectional area, allowing both flow and return ways to be accommodated in a diameter a little larger than that of a single flow line.
2. Combined coupling of both flow and return ways is possible using a single ambient temperature seal and an internal thermal seal, which is activated at low temperature. This technique, which was developed for the ISIS moderator systems and published at the International Cryogenic Engineering Conference (1985),² allows easier coupling and decoupling with much improved reliability.
3. Since the innermost tube is surrounded by the returning cold hydrogen annulus, radiation heat leakage to that line is extremely small. Using this as the flow line allows a cold source to be operated at the lowest temperature possible without the risk of hydrogen freezing in the heat

Fig. 2.10. Hydrogen transfer line.

exchanger. It also allows the control sensor to be situated back in the pump module (30 m from the moderator) and still be representative of the temperature of the hydrogen entering the moderator vessel.

4. The design is better suited than most to be constructed from semi-flexible tubing that allows a complete transfer line to be transported and installed in a single length. This makes installation much easier and completely avoids any on-site fabrication.
5. The design principles are well documented and proven, being used widely at the European Organization for Nuclear Research (CERN), the Joint European Fusion facility, the ISIS facility, and some facilities in the United States.

Three main transfer lines are proposed in the HFIR design concept:

1. One transfer line is required to transport helium refrigerant between the refrigerator and the main heat exchanger. This is a relatively short line that is not required to be within an inert blanket. It could be configured from two individual lines for flow and return using Johnstons couplings, or it could be the concentric design described above.
2. The second transfer line provides transport of cryogenic hydrogen between the main heat exchanger and the pump module. This short line must be included within the inert blanket and thus would require an additional outer tube. It could use either of the designs suggested for the previous line.
3. Finally, the third transfer line provides transport of cryogenic hydrogen between the pump module and the moderator plug module. This must be included within the inert blanket and would require an additional outer tube. It is most suited to the concentric tube design, and such has been chosen for the reference design.

2.3.7 Insulating Vacuum Systems Module

2.3.7.1 Functional Requirements

The insulating vacuum systems (pressure of about 10^{-6} torr) provide insulation for all cryogenic components of the system. The vacuum systems modules provide constant leak monitoring and also serve

as pressure containment in the event of a primary containment failure. All components must meet safety, seismic, and quality level requirements as defined in Sect. 3.6 of this report.

2.3.7.2 General Description

The vacuum envelope is divided into two sections; one is dedicated to the moderator plug assembly and the transfer line out to the reactor building outer wall (to limit the spread of contamination in the event of a primary containment failure). The other is dedicated to the remainder of the cold loop, including the pump and heat exchanger modules. Each system is serviced by a vacuum pump station, which comprises a turbo pump, a roughing pump, a fast-acting isolation valve, and a gas analyzer sensor, all housed in a vessel that is integrated into the inert blanket. If necessary, a complete vacuum pump station is replaced as a unit, which is made possible while the cold source remains in operation. The integrity of neither the vacuum system nor the inert blanket is challenged when a vacuum pump station is replaced. The vacuum is first isolated by a valve with an actuator that extends through a seal in the inert blanket wall. The coupling flange has seals to isolate the vacuum and the inert blanket independently.

2.3.8 Inert Blanket

2.3.8.1 Functional Requirements

The inert blanket provides a second barrier between the hydrogen and atmospheric air (a third barrier in the case of vacuum-insulated components). The barrier also allows constant first-order leak monitoring. All components must meet safety, seismic, and quality level requirements as defined in Sect. 3.6 of this report.

2.3.8.2 General Description

The inert blanket is divided into sectors that together constitute an unbroken barrier over the entire hydrogen-bearing system, including those sections that are vacuum-jacketed. This is necessary because air leaked into an insulating vacuum space would cryopump onto cold surfaces, making its detection very difficult. Also, in areas of high radiation, such frozen air could break down to form hazardous unstable free radicals. Some of the larger components are double-walled to incorporate the inert blanket, and smaller modules (such as the gas-handling system) are assembled into smaller vessels that provide an inert blanket. All individual components are interconnected with pipework that is further contained by flexible pipework to provide continuity of the inert blanket. The inert blanket medium is slightly pressurized helium or nitrogen gas or, in some cases, a secondary vacuum (approx 10^{-2} torr). In the case of the large containment vessel, an outer vacuum negates the possibility of inleakage when it is at sub-atmospheric pressure during normal operation.

2.3.9 Control and Instrumentation

2.3.9.1 Functional Requirements

The function of the control and instrumentation system is four-fold:

1. to monitor and control the hydrogen temperatures and pressures;
2. to monitor vacuum systems, water supply systems and "other gasses" used for monitoring or inert blanketing;
3. to provide interlocks between components of the cold source to ensure its safe operation at all times; and

4. to interface with the HFIR data acquisition systems to warn of cold source malfunctions and, in the event of a major fault signal, to initiate reactor shutdown.

All components must meet safety, seismic, and quality level requirements as defined in Sect. 3.6 of this report.

2.3.9.2 General Description

The main control system is PC-based, using windows-interfaced software. The primary control unit is located in the refrigerator room, but the main monitoring station will be in the cold source support building. A further computer drop will be provided in the main reactor control room.

2.3.9.3 Pressure Control

The hydrogen loop pressure is controlled by two control valves, one of which admits gas to the loop while the other vents gas from the loop. Both are controlled by a single pressure sensor located just upstream of the circulators. Vented gas returns to the main storage vessel, and a transfer pump moves gas from this vessel to maintain a smaller feed vessel at a pressure above that of the loop. The system is consequently totally closed, and under normal operating conditions the storage vessel is under vacuum. To prevent both feed and vent valves being open together (which could lead to instability), there is a dead band centered on the average operating pressure, over which both valves are fully closed. Gas entering or leaving the loop passes through the pressurizer, which is a vessel connected into the loop at its bottom end, its top end being attached to the top plate of the pump module vessel. Since the vessel is an appendage to the liquid flow line, it holds a static pocket of gas that stratifies in temperature: gas is added and vented at the top of the pressurizer, which remains nominally at ambient temperature.

2.3.9.4 Temperature Control

The temperature control system is illustrated in Fig. 2.11. Temperature control is provided by a heater in the refrigerant line upstream of the heat exchanger. This is powered in response to one of three sensors, depending on the mode of operation:

1. The refrigerant temperature is limited to a minimum of 14 K during the cool-down phase to prevent freezing of the hydrogen. Freezing is most likely when the gas is warm, and its density and specific heat are low. The low density circulator is operated during the cool-down, and the controlling temperature element (sensor TS1) is at the heat exchanger. The same sensor is used when the system is operating in standby mode at 100 K. At about 50 K, the high density circulator takes over automatically, and temperature control is switched to sensor TS2.
2. Temperature sensor TS2 is located in the feed line just downstream of the heat exchanger. This location is representative of the temperature at the entrance to the moderator vessel and allows the moderator to run as cold as possible.
3. If it is necessary to operate the refrigerator while the hydrogen loop is warmed up, a bypass valve and isolation valve allow the refrigerator to operate in a closed loop. The normal heater will be turned off and the calorimeter heater, which is part of the refrigerator system, maintains the refrigerator return temperature at 20 K.

The refrigerator is configured to use only sufficient compressor capacity to meet the prevailing cooling conditions. Helium gas is returned to the refrigeration cycle at a nominal temperature of 20 K by

Fig. 2.11. Temperature control system.

the internal calorimeter, which operates in response to a sensor in the return pipe. If the calorimeter power rises above 900 W, compressors are shut down until the flow capacity is commensurate with the current heat load. After cool-down to normal operational temperatures, before starting the reactor, the hydrogen loop control system activates a second heater in the refrigerant flow line to 2.3 kW. The calorimeter then stabilizes the refrigerator at the required compressor flow rate. On starting the reactor, power is automatically removed from the control heater as required, while the calorimeter continues to return the gas at 20 K. Furthermore, in the event of a reactor scram, the control heater again compensates, allowing the refrigerator to function at a nominally steady state, while allowing the overall system to respond quickly and smoothly to normally anticipated heat load changes. The calorimeter is capable of sufficient power to equal the full refrigerator capacity and is used to measure refrigeration performance when necessary.

2.3.10 Equipment Enclosures and Safe Room

2.3.10.1 Functional Requirements

The main functions of the equipment enclosure are to provide protection from the weather and to provide space for maintenance activities. The enclosures must meet uniform building codes and include adequate environment for workers. In addition, the enclosures must have doors and ceiling panels required for access to equipment.

The safe room is a special area for isolation of the hydrogen-bearing equipment. Its main function is to provide an environment that minimizes the potential for a hydrogen burn or detonation. All electrical equipment within the safe room must either meet intrinsically safe requirements (maximum possible energy release of 0.019 mJ) using dedicated electronic barrier systems or be inert gas-blanketed. The safe room must be appropriately vented and must include blowout panels consistent with the volume of space inside the safe room.

All components must meet safety, seismic, and quality level requirements as defined in Sect. 3.6 of this report.

2.3.10.2 General Description

Several structures are currently anticipated to house the cold source equipment external to the reactor building. These structures will house the hydrogen safe room, the refrigerator equipment, the cold source control room, and the cold source operations support needs.

The refrigerator equipment is housed in two separate rooms in a structure outside of the reactor building. One room contains the refrigerator cold box heat exchanger and expander while the other room contains the refrigerator compressors as shown in Fig. 2.12. Large gas vessels and gas bottles are located outside of the main refrigerator building.

The pump module and heat exchanger modules are located in a frangibly-constructed safe room adjacent to the main refrigerator building (also shown in Fig. 2.12). The safe room is constantly ventilated by a hydrogen-qualified fan at a rate sufficient to ensure that a detector at the outlet is able to detect hydrogen at concentrations well below the lower combustible limit. Such a signal causes the extraction rate to be increased to the equivalent of one air change per minute. Personal grounding devices and antistatic over garments will be used by personnel entering the safe room whenever the cold source is operational.

Another structure houses the cold source control room and space for the cold source operations staff. This building is located approximately 150 feet from the hydrogen safe room.

2.3.11 Vent System

2.3.11.1 Functional Requirements

The vent system is required to transport gas safely from any of the hydrogen pressure safety systems, or the vacuum pumps, to a safe height for release to the atmosphere. It must also transport air (or H_2 /air) from the safe room at a rate of up to one air change per minute for release to the atmosphere. During operation of the cold source, the vent is required to be inerted with N_2 gas up to its outlet nonreturn valve (which will be located just before the safe room connection point).

All components must meet safety, seismic, and quality level requirements as defined in Sect. 3.6 of this report.

2.3.11.2 General Description

A schematic of the vent system flow diagram is shown in Fig. 2.13. Individual rupture discs protect the main loop, each vacuum system, and each inert gas blanket sector. All are connected into a common manifold that is piped to a dedicated vent stack. The exit of the stack is closed by a low pressure relief valve, and nitrogen gas is passed through the vent system, continuously maintaining a positive pressure of about 1.5 psi, effectively making the vent system also double-isolated from the atmosphere. All vacuum and purge pumps vent into the stack, together with any other relief systems. The stack itself will be situated in a safe area relative to prevailing winds and other site equipment.

Fig. 2.12. Layout of hydrogen safe room and refrigerator equipment rooms.

Fig. 2.13. Vent system diagram.

2.4 PHYSICS ANALYSES

Physics analyses have been performed to characterize the cold source performance. This section provides a description of the physics models developed and the resulting analyses that have been performed to support the design of the cold source. These analyses have provided the performance information, such as cold neutron brightness down the beam tube and important design data regarding component heat loads, that has been used to develop the reference cold source concept.

2.4.1 Physics Codes, Model Description, and Cross-Section Data Sets

2.4.1.1 HFIR Description

The HFIR cold source physics analyses are, of course, greatly dependent on the model used to represent the HFIR reactor, which is the source of neutrons reaching the cold source. The HFIR reactor core model used in these studies is believed to be the most detailed physics model of the HFIR that has

been developed. This model has been and will continue to be reviewed by HFIR staff to ensure an accurate representation of the HFIR system. Validation of the reactor model has been performed by comparisons with measured data. The key reactor parameters used in this study are provided in Table 2.5. It should be noted that although at present the HFIR operates at 85 MW, all calculations are based on a 100-MW operating condition (because an upgrade to this power level is being considered, the cold source must be designed to accommodate this higher power level).

Table 2.5. Key HFIR physics model parameters

Parameter	Value
Power level	100 MW
Maximum fuel cycle length	23 d
Active fuel height	20 in.
Active fuel region volume	1.79 ft ³ (50.6 L)
U ²³⁵ loading	9.4 kg
Total B ¹⁰ burnable poison loading	2.8 g
Fuel plate thickness	0.050 in.
Cladding thickness	0.010 in.
Coolant channel thickness	0.050 in.

Figure 2.14 shows a cross-section of the HFIR core and reflector at the reactor's midplane, as modeled in the current 3-D Monte Carlo analysis. The proposed cold source will be placed adjacent to the beryllium block at the end of the HB-4 beam tube. It should also be noted that two aluminum windows in the cold neutron beam path have been incorporated into the design and are included in the analysis model.

2.4.1.2 Codes and Cross-Section Data

Because of the inherent 3-D nature of the HFIR core, the various experimental and user facilities in the reflector, and the location of the proposed cold source in the HB-4 beam tube, the 3-D MCNP Monte Carlo code (version 4a)³ was used in virtually all of the analyses performed to date. In addition to being able to model all aspects of the 3-D geometry, it also has the advantage of being able to represent all of the cross-section data on a point wise energy basis. Various point-wise cross-section libraries are typically supplied with the MCNP code. Each is the result of different historical evaluations of the cross-section data for the various nuclides. Except for U-235, Al-27, and parahydrogen and orthohydrogen, the data used here corresponds to the Los Alamos National Laboratory (LANL) recommended data set for each nuclide. In the case of the U-235, the recommended data set from LANL was modified so as to include the energy-dependent delayed fission product gammas as regular secondary gammas. (Normally delayed fission product gammas are not included in the regular MCNP gamma production cross-sections, and one needs to determine these sources off-line with ORIGEN and then run a second MCNP calculation with only these source terms present.) By now treating them as regular secondary gammas,

Fig. 2.14. HFIR reactor core midplane layout.

one can now track those gammas without approximation (just like the prompt gammas) and determine where in the system that energy is deposited without the persistent need for a second MCNP calculation. In the case of the Al-27, the recommended data set from LANL was modified so as to include, as regular secondary gammas, those short-lived decay gammas coming from the decay of Al-28 produced by thermal neutron capture in Al-27. By accounting for these gammas in this fashion, we can automatically track those gammas without any approximations and determine where that energy is deposited rather than where it is first created. In the case of the parahydrogen and orthohydrogen in liquid LH₂ at 20 K (72 kg/L), it was determined that the new (as yet unreleased) LANL evaluation of the Institut Fur Kernenergetik Und Energiesysteme (IKE) data from Germany⁴ was superior to previous data files based on measurements and analyses performed at the National Institute of Standards and Technology (NIST). (Revised calculations in FY 1998 will be performed to assess the potential design impact of now having the hydrogen in a gaseous state at this same temperature and density.) With no artificial adjustments, we normally calculate a k_{eff} between 1.006 and 1.008 (± 0.0006) at beginning-of-cycle (BOC), middle-of-cycle (MOC), and end-of-cycle (EOC) when the correct fuel composition and known control rod positions are used in the MCNP model.

2.4.1.3 Source Terms Used in the Analyses

Neutron-only k_{eff} calculations were used to determine the neutron source distribution in the core. The core model used in most analyses assumed EOC fuel concentrations. This was done in order to obtain a realistic upper limit on the cryogenic heat loads that peak at EOC.

Acceptable statistical uncertainties (less than 2–3%) at the various point detectors of interest generally required 288 batches of 10,000 source neutrons, for a total of 2,880,000 histories, although the results from the first 20 batches (300,000 histories) were typically discarded as the source distribution within the core was being converged. Final reference results were performed with 1224 batches of 10,000 source neutrons, for a total of 12,240,000 histories. Reportable results were, therefore, based on the remaining histories after the source was deemed converged. These calculations were typically performed on the IBM SP-2 multinode cluster of workstations.

2.4.1.4 Method Used for Calculating the Brightness

The cold neutron beam magnitude is typically expressed in terms of its “brightness,” which is defined as the “number of neutrons per second, per square centimeter, per angstrom, per steradian” at a point detector 477 cm from the center of the former HB-1/HB-4 thoughtube, or 466 cm from the cold source capsule. Throughout this study, the calculated brightness is tabulated in six energy bands as defined in Table 2.6. These energy bands provide data over the range of interest as defined by the proposed cold source instruments presented in Table 2.7.

Table 2.6. Energy bands of interest

Energy range (eV)	Corresponding neutron wavelength range (angstroms)
0.01000–0.10400	0.89–2.86
0.00325–0.01000	2.86–5.02
0.00145–0.00325	5.02–7.51
0.00085–0.00145	7.51–9.81
0.00058–0.00085	9.81–11.93
0.00033–0.00058	11.93–15.86
0.00010–0.00033	15.86–28.60

To determine the brightness in the direction of interest (down the HB-4 beam tube), the scalar flux was tabulated at a point 188.1 in. away from the midpoint of the HB-4 beam tube, along its centerline. More specifically, the model only tabulated a select portion of the scalar flux at that point detector. Looking back from that point, the liquid hydrogen portion of the cold source (having a diameter of 3.9 in.) would subtend an angle of ± 0.60 degrees about the beam tube centerline, or a solid angle of 3.5×10^{-4} steradians. To ensure that the MCNP tally counter associated with that point detector counted only neutrons traveling within that solid angle, next event estimates of the flux at that point were made only from collision sites in the cold source and those material regions directly behind the cold source.

Table 2.7. Proposed cold source instruments

Instrument type	Wavelength (angstroms)
SANS 1 (D11)	7-15
SANS 2	6
SANS 3 (short ORNL)	4.7
Triple-axis	1.8-6
Reflectometer	5
Biology single-crystal (DB 21)	8
Cold neutron test station	6-40

Specifically, this included only the following regions: the liquid hydrogen and aluminum capsule forming the cryostat, and those portions of the vacuum vessel, beam tube, water space, and beryllium block directly behind a 3.875-in.-diam disk corresponding to the diameter of the cold source capsule. Next event estimates from all other collision sites were ignored by setting the probability of a contribution from these other cells to zero. This not only had the desired effect of selectively tallying only those neutrons traveling within the desired range of solid angular directions, but it also allowed the calculations to proceed about five times faster than they would have if next event estimates were made from all collision sites in the problem.

2.4.1.5 Methods Used for Calculating Neutron and Gamma Heating

Estimates of the neutron and gamma heating in the liquid hydrogen and in the various aluminum components around the cold source were originally obtained by two independent methods. Both methods rely on region-averaged estimates of the total neutron or gamma track length in the various cells comprising the system. In the first method, the average energy-dependent neutron or photon heating number (MeV/collision) available in the MCNP cross-section libraries is folded with the total (point-wise energy-dependent) neutron or gamma cross-section, which is then folded with the (point-wise energy-dependent) neutron or gamma track length each time a neutron or photon passes through a cell of interest. These values are then integrated over all energy and spatially averaged over the volume of the respective cells to obtain the specific neutron and gamma heating rates in terms of watts/gram. In the second method, the cell-averaged neutron and gamma fluxes in the aluminum-filled cells were first tallied in multigroup energy bins corresponding to the 39n/44g group structures used in the analysis of the Advanced Neutron Source (ANS) reactor. These multigroup neutron and gamma fluxes were then folded with the multigroup neutron and gamma kerma factors described in Appendix D of ORNL/TM-12672⁵ to obtain the average specific heating rate (watts/gram) for the various cells. (These multigroup kerma factors were based on the KAOS-V kerma factor data library available from Argonne National Laboratory (ANL) and/or the Radiation Shielding Information Center in Oak Ridge.)⁶ After several such calculations showed the directly-calculated MCNP heating rates to be in good agreement with those calculated indirectly using the multigroup data, the latter procedure was dropped.

Lastly, nuclear heating in the aluminum due directly or indirectly to neutron radiation has three separate components: that directly because of the kinetic interactions between the neutrons and the aluminum, that because of the subsequent gamma decay of Al-28 atoms that became activated by thermal neutron absorption in the Al-27, and that because of the subsequent beta decay of Al-28 atoms that become activated by thermal neutron absorption in the Al-27. The cell-averaged neutron heating rate tallies used here for cells containing aluminum account for all three processes. Basically, the point-wise energy-dependent flux is folded with the point-wise total cross-section and the energy-dependent neutron

heating number (MeV/collision) to obtain the direct component. In addition, the point-wise flux is also folded with the point-wise Al-27 absorption cross-section to obtain the Al-28 production rate, which is then folded with the average beta decay energy (1.2423 MeV). This approach is based on the assumption that the Al-28 decay beta deposits its energy at the point where it is created. The Al-28 decay gammas, on the other hand, do not in general deposit their energy where they are created. Using the newly modified Al-27 cross sections described above, these Al-28 decay gammas are now treated as additional secondary gammas (produced by neutrons interacting with Al-27). With this scheme, these gammas are then tracked by MCNP (along with all other secondary gammas), depositing their energy in various components along the way. This approach inherently models all points of origin exactly, and tracks the gammas until all of the energy has been deposited in various structures. The only minor disadvantage is that, in a given component, these contributions are now indistinguishable from all other secondary gammas. Still, the added convenience and improved accuracy of this approach far outweigh this one small disadvantage.

2.4.2 Brightness, Gain Factors, and Comparisons with Existing Facilities

Table 2.8 shows the brightness of the 5-cm-thick HFIR cold source in the six energy bands of interest, as well as the comparable values for the horizontal cold source in the ILL reactor. These comparisons suggest that the proposed HFIR cold source will be comparable to the ILL cold source.

Table 2.8. Neutron brightness comparison between the HFIR cold source and the ILL horizontal cold source

Wavelength (Å)	Neutron energy range (meV)	HFIR cold source brightness ^a (s ⁻¹ · cm ⁻² · Å ⁻¹ · steradian ⁻¹)	ILL horizontal cold source brightness ^b (s ⁻¹ · cm ⁻² · Å ⁻¹ · steradian ⁻¹)	Ratio HFIR/ILL
0.887–2.86	10.00–104.0	1.09×10^{13}	7.72×10^{12}	1.41
2.86–5.02	3.25–10.0	7.54×10^{12}	4.05×10^{12}	1.86
5.02–7.51	1.45–3.25	1.55×10^{12}	1.39×10^{12}	1.12
7.51–9.81	0.85–1.45	3.99×10^{11}	4.00×10^{11}	1.00
9.81–11.93	0.58–0.85	1.56×10^{11}	1.49×10^{11}	1.05
11.93–15.86	0.33–0.58	5.44×10^{10}	5.64×10^{10}	0.96
15.86–28.60	0.10–0.33	7.54×10^9	Not available	–

^aHFIR data based on MCNP calculations at 100 MW with statistical standard deviations below 2% with the exception of the lowest energy group where the statistical standard deviation is 2.6%.

^bAverage values over energy range as estimated from Fig. 5 of a paper presented by Ageron at the 1990 International Workshop on Cold Neutron Sources.⁷

Although a full validation of the HFIR calculations has not yet been undertaken, some evaluations were performed to determine if they were consistent with existing data. Two questions were addressed:

1. Are the calculated thermal neutron fluxes at the cold source location (without cold source in place) consistent with expected values?
2. Are the gains in longer wave neutron fluxes consistent with gains measured at existing hydrogen cold source facilities?

The first question addresses the ability of the model to calculate the physics of the reactor core and to transport the neutrons to the cold source location. In order to answer this question, a comparison was made with previously reported HFIR fluxes. Although comparisons with identical geometry could not be made, the fluxes were roughly comparable, and the differences can in general be explained by the geometry differences. The gain factor issue was addressed by comparing the calculated gain factors associated with a HFIR cold source (as shown in Table 2.9) with gain factors measured at other hydrogen cold source facilities as reported in ref. 7. This comparison is summarized by Fig. 2.15 where the calculated gain factors for the HFIR hydrogen cold source are plotted as a function of neutron wavelength along with reported values from existing hydrogen cold source facilities.

Table 2.9. Gain factors as calculated for the proposed HFIR liquid hydrogen cold source

Wavelength (Å)	Neutron energy range (meV)	HFIR brightness with LH ₂ cold source (s ⁻¹ · cm ⁻² · Å ⁻¹ · steradian ⁻¹)	HFIR brightness without LH ₂ cold source (s ⁻¹ · cm ⁻² · Å ⁻¹ · steradian ⁻¹)	Gain ratio
0.887–2.86	10.0–104.0	1.09×10^{13}	3.79×10^{13}	0.29
2.86–5.02	3.25–10.0	7.54×10^{12}	1.61×10^{12}	4.68
5.02–7.51	1.45–3.25	1.55×10^{12}	1.16×10^{11}	13.36
7.51–9.81	0.85–1.45	3.99×10^{11}	1.95×10^{10}	20.46
9.81–11.93	0.58–0.85	1.56×10^{11}	Not available	—
11.98–15.86	0.33–0.85	5.44×10^{10}	Not available	—
11.98–15.86	0.33–0.85	7.54×10^9	Not available	—

2.4.3 Nominal Heat Load Calculations

Accurate determination of the overall heat load associated with the cold source is quite important since this will define the cooling requirements for the cryogenic cooling system. Especially important are the heat loads associated with the liquid hydrogen, the aluminum capsule containing the liquid hydrogen, and the inlet and outlet tubes carrying the liquid to and from the cold source. In addition, one must also consider the intrinsic heat load associated with pumping the liquid hydrogen through the system. This aggregate heat load must all be removed by the cryogenic cooling system. In addition, there is concern with the heat loads associated with the cold source vacuum vessel wall. Even though this heat load will not have to be removed cryogenically, the design must still supply adequate cooling.

As previously stated, one must consider energy deposited by prompt fission gammas from the core, delayed fission gammas from the core, and secondary gammas produced in these and other components, as well as energy deposited directly by neutrons from the core and energy deposited by Al-28 decay gammas and Al-28 decay betas to estimate the total nuclear heat load on the various components. Nonnuclear intrinsic heat loads such as frictional heating and conventional thermal radiation must also be considered, although the latter can be shown to be negligible. Once these specific heat loads (watts/g) were determined using the standard neutronic model, the volumes and masses of the corresponding cells in the design were used to estimate the actual heat loads. Table 2.10 shows the origin and distribution of the nominal cryogenic heat loads (in watts) associated with the cold source at a HFIR operating power of 100 MW. As indicated in this table, the preliminary estimate of the heat load is approximately 2.2 kW.

Fig. 2.15. Calculated gain factors compared with facility gain factors.

The uncertainty to be applied to the nominal heat load calculation is a separate issue. In the past, large uncertainty factors (40% and higher) have been applied to the calculated heat loads. It is believed that today the calculational techniques and models are better than this and, thus, that it represents an overly conservative approach that translates to additional costs. Therefore, a heat load measurement will be conducted in HFIR close to the proposed cryostat location and used to develop a defensible uncertainty factor. A 40% uncertainty factor, however, was used to define the refrigerator requirement, which was a long lead procurement item. This did not add substantially to the cost of the refrigerator and should give us additional margin if the efficiency of the refrigerator performance declines with age.

2.4.4 Design Optimization Studies

A number of physics studies have been performed in arriving at the present reference design concept. Several of the studies reported in this section were documented in the preconceptual design report⁸ and are included here for completeness while the early hydrogen thickness sensitivity studies have been significantly supplemented and revised as noted in Sect. 2.4.4.1.

2.4.4.1 Impact of Variations in the Orthohydrogen/Parahydrogen Ratio in Liquid LH₂

A parametric evaluation was performed to determine the sensitivity of the cold source performance to the orthohydrogen/parahydrogen ratio. Previous analyses¹ had indicated that the orthohydrogen/parahydrogen ratio had little impact on the optimal hydrogen thickness for the proposed geometric shape of the HFIR cold source. This was found to be inconsistent with evaluations performed at NIST, and,

Table 2.10. Nominal cryogenic heat loads for the reference HFIR cold source

Source of heating	Liquid hydrogen (W)	Aluminum structure (W)	Total (W)	Percent of system total
Gamma (primary, secondary, and delayed fission)	244.8	1025.2	1270.0	57.4
Direct neutron	316.7	8.9	325.6	14.7
Aluminum decay beta	0.0	227.8	227.8	10.3
Nuclear heating of inlet/outlet tubes ^a	31.5	157.6	189.1	8.6
Total nuclear heating	593.0	1419.5	2012.5	91.0
Intrinsic system heating			200.0	9.0
Total cryogenic heat load			2212.5	100.0

^aValue is based on a 0.375-in. inside diameter (ID) inlet tube and a 0.5 in. ID outlet tube, both with a 0.039 in. thick aluminum wall.

therefore, a decision was made to model the NIST cold source geometry and examine the spectrum changes when the ortho/para ratio was significantly changed. The results are illustrated in Fig. 2.16. As seen from this figure, the 65% ortho case and the all-para case have significantly different spectral shapes in the 1–8 meV region. These results were found to be very consistent with those previously reported by NIST.⁹ The cold source geometry was then modified in ten evolutionary steps to arrive at the basic HFIR cold source geometry. These modifications included changing the reflector from D₂O to beryllium (with trace amounts of ³He at equilibrium levels of ⁶Li); this was followed by several size-reduction steps, and a general shape change of the hydrogen from a spherical shell to a dented sphere. A more detailed explanation of the geometry changes and the results from each change are presented in Appendix A.

The results of this study were somewhat surprising but reassuring. It was found that just changing the reflector material from D₂O to the beryllium described above resulted in significant spectrum changes, reducing the cold neutron population in the 65% ortho-case by more than 10%. Other changes had similar effects so that, when compared with NIST, the relationship between the 65% ortho case and the all para case was found to be substantially different for the smaller HFIR geometry (restricted size because of beam tube) as shown in Fig. 2.17. The implication of this analysis is that, for the proposed HFIR geometry, the cold neutron brightness was, in general, insensitive to the ortho/para ratio over the 7 to 10 Å wavelength range of interest. This relationship was found to change somewhat as the hydrogen thickness was made smaller, but the neutron brightness observed for these cases was always lower than the reference hydrogen thickness (Table 2.11). Additional studies indicated that some increase in brightness could be obtained by increasing the hydrogen thickness, but the increased hydrogen inventory resulted in undesirable increases in heat load. This was good news for the development of the HFIR cold source because the relative ortho/para content in the high radiation field is not well understood, and could have resulted in a large uncertainty in the optimization of the cold source geometry, if sensitivity to this ratio were high.

Fig. 2.16. Ortho hydrogen impact on flux spectrum in NIST cold source geometry.

2.4.4.2 Optimization of Viewing Window Orientation

The cold moderator capsule, when viewed from the beam tube axis, has a circular outer periphery of diameter 3.875 in. There is a central reentrant cavity, roughly elliptical in shape, with the inlet and outlet tubes displaced outward along the short axis of the cavity and located near the outer circular boundary. The rectangle inscribed in this cavity has dimensions of 2.649×1.5 in. The question of the best orientation of this rectangle relative to rotation about the beam tube axis was evaluated to determine the preferred orientation of the inlet and outlet hydrogen lines (i.e., lines must fit outside of the inscribed viewing rectangle). A horizontal orientation means that the long axis of the elliptical cavity is horizontal, and the inlet and outlet tubes are aligned in the vertical direction. A rotation through 90 degrees about the beam tube axis takes the horizontal orientation into the vertical orientation.

If the cold source were to be operated with liquid phase in the moderator, the horizontal orientation with the inlet tube below the outlet would be preferred from a mechanical design perspective because of the phase change issue. However, if the operation is to be restricted to the supercritical phase, the orientation is less important with respect to the cold source mechanical design.

Fig. 2.17. Ortho hydrogen impact on flux spectrum in HFIR cold source geometry.

From the neutron scattering point of view, the orientation question hinges on the issue of source brightness uniformity. If the source were uniformly bright over the full circular area (3.875 in. diam), the orientation would be immaterial, and a horizontal orientation would be chosen because the mechanical support of the cryostat is somewhat easier in this orientation. However, if the brightness in the central elliptical cavity were significantly greater than that of the remainder of the source area, a vertical orientation would be desired. For experiments employing a crystal monochromator diffracting in the horizontal plane, this orientation would give better resolution for equal signal strengths because of the smaller horizontal divergence in the beam incident on the monochromator.

To resolve this question, Monte Carlo calculations were performed to determine the average brightness over the full circular area (76.09 cm²) and over the inscribed rectangle in the central elliptical cavity (25.64 cm²). The results are summarized in Table 2.12, which gives the ratio of the average brightness over the inscribed rectangle to that of the full circle.

Table 2.11. Relative brightness as a function of the cold source thickness(t), percent ortho-H (x), and wavelength (w), where the relative brightness is defined as $RB(t,x,w) = B(t,x,w)/B(5,0,w)$

Thk. cm	Wavelength (angstroms)	0% orthoh		20% orthoh		40% orthoh		60% orthoh	
		ratio	fsd	ratio	fsd	ratio	fsd	ratio	fsd
2	11.93-15.86	0.7332	0.035	0.6644	0.038	0.6698	0.037	0.6670	0.037
2	9.81-11.93	0.7114	0.030	0.7369	0.031	0.7232	0.032	0.7474	0.032
2	7.51-9.81	0.7387	0.020	0.7651	0.021	0.8322	0.022	0.8392	0.021
2	5.02-7.51	0.7412	0.016	0.7672	0.017	0.8268	0.017	0.8700	0.017
2	2.86-5.02	0.8027	0.010	0.7291	0.010	0.7300	0.011	0.7178	0.011
2	0.89-2.86	1.2263	0.009	1.1507	0.010	1.1224	0.010	1.1197	0.010
3	11.93-15.86	0.8342	0.033	0.7469	0.036	0.6872	0.036	0.7069	0.036
3	9.81-11.93	0.8234	0.029	0.7948	0.030	0.7708	0.029	0.8047	0.030
3	7.51-9.81	0.8442	0.019	0.8650	0.020	0.8978	0.021	0.9103	0.021
3	5.02-7.51	0.8480	0.016	0.8723	0.017	0.9259	0.017	0.9570	0.017
3	2.86-5.02	0.8898	0.010	0.8014	0.010	0.7801	0.010	0.7780	0.011
3	0.89-2.86	1.1255	0.009	1.0361	0.010	0.9974	0.010	0.9906	0.010
4	11.93-15.86	0.9541	0.033	0.7598	0.035	0.7476	0.037	0.7070	0.037
4	9.81-11.93	0.9161	0.027	0.7930	0.029	0.8200	0.029	0.8005	0.030
4	7.51-9.81	0.9735	0.023	0.8911	0.020	0.9242	0.021	0.8960	0.021
4	5.02-7.51	0.9232	0.016	0.9241	0.016	0.9467	0.017	0.9471	0.018
4	2.86-5.02	0.9463	0.010	0.8333	0.011	0.7933	0.010	0.7782	0.011
4	0.89-2.86	1.0440	0.009	0.9426	0.010	0.9106	0.010	0.9050	0.010
5	11.93-15.86	1.0000	0.026	0.7635	0.036	0.6894	0.036	0.6732	0.038
5	9.81-11.93	1.0000	0.023	0.8103	0.028	0.7853	0.031	0.7483	0.030
5	7.51-9.81	1.0000	0.015	0.9128	0.020	0.8933	0.023	0.8708	0.021
5	5.02-7.51	1.0000	0.018	0.9302	0.016	0.9430	0.017	0.9369	0.017
5	2.86-5.02	1.0000	0.008	0.8338	0.010	0.7771	0.010	0.7472	0.011
5	0.89-2.86	1.0000	0.008	0.8777	0.010	0.8430	0.010	0.8308	0.010
6	11.93-15.86	1.0071	0.031	0.6921	0.035	0.6747	0.037	0.6436	0.039
6	9.81-11.93	1.0448	0.027	0.8269	0.029	0.7654	0.030	0.7339	0.031
6	7.51-9.81	1.0495	0.017	0.9104	0.020	0.8682	0.021	0.8544	0.021
6	5.02-7.51	1.0224	0.015	0.9354	0.017	0.9348	0.017	0.8873	0.017
6	2.86-5.02	1.0360	0.009	0.8241	0.010	0.7729	0.010	0.7207	0.011
6	0.89-2.86	0.9608	0.009	0.8243	0.010	0.7896	0.010	0.7808	0.010
7	11.93-15.86	1.0613	0.029	0.6860	0.036	0.6714	0.038	0.6060	0.037
7	9.81-11.93	1.0905	0.026	0.7518	0.029	0.7515	0.031	0.7101	0.031
7	7.51-9.81	1.1096	0.017	0.8530	0.020	0.8611	0.020	0.8045	0.021
7	5.02-7.51	1.0655	0.015	0.9007	0.016	0.8956	0.017	0.8625	0.017
7	2.86-5.02	1.0646	0.009	0.8028	0.010	0.7325	0.010	0.6953	0.011
7	0.89-2.86	0.9259	0.009	0.7679	0.010	0.7448	0.010	0.7503	0.011
8	11.93-15.86	1.1188	0.029	0.7075	0.036	0.6085	0.037	0.6142	0.39
8	9.81-11.93	1.1386	0.025	0.7659	0.031	0.7396	0.040	0.6619	0.031
8	7.51-9.81	1.1122	0.017	0.8429	0.020	0.7915	0.021	0.7680	0.021
8	5.02-7.51	1.0938	0.015	0.8823	0.016	0.8434	0.017	0.7248	0.018
8	2.86-5.02	1.0737	0.009	0.7838	0.010	0.7043	0.011	0.6610	0.011
8	0.89-2.86	0.8979	0.009	0.7343	0.010	0.7006	0.010	0.7068	0.011

Table 2.12. Brightness ratio (inscribed rectangle/full circle)

Wavelength (Å)	Brightness ratio
2.86 to 5.02	0.936 ± 0.009
5.02 to 7.51	0.998 ± 0.016
7.51 to 9.81	1.046 ± 0.020
9.81 to 11.93	1.048 ± 0.030
11.93 to 15.86	1.086 ± 0.036
15.86 to 28.60	1.178 ± 0.047

Somewhat surprisingly, the two brightnesses are nearly equal for all wavelengths of interest, indicating a nearly uniform brightness over the full circular area. The gradual increase of the ratio as the wavelength increases indicates a slightly cooler spectrum for the inscribed rectangle compared to the full circle. While this study gives no compelling reason to favor a particular orientation from the neutron scattering point of view, the collective judgment of the ORNL scattering group favored a vertical orientation. Although as mentioned previously, this orientation does make the mechanical support concept a little more difficult, the decision was made to go with the vertical orientation.

2.4.4.3 Effect of the cold source (in HB-4) on experiments that use HB-1

The HB-1 and HB-4 beam tubes are oriented 180 degrees from each other. The two tubes are separated by a small beryllium block. There was concern within the user community that the presence of the cold source in the HB-4 beam tube might cause some undesirable perturbations for the user station located at the end of the HB-1 beam tube. While it is true that the presence of the liquid hydrogen in the cold source will modify the spectrum of the neutrons in and around the HB-4 beam tube, it was noted that very little beryllium and water at normal reflector temperature is required to return the spectrum to that normally expected in the reflector. Moreover, because the neutrons immediately surrounding the cold source are not expected to contribute to the source at HB-1, it was believed that the presence of the cold source should have very little (if any) effect on flux levels at the end of the HB-1 beam tube. However, an analysis was performed to verify this belief. At first glance (see Table 2.13), some spectrum changes are visible. However, upon closer evaluation one finds that, for all energy groups, a ratio of 1.00 is within the statistically derived uncertainty factor. Based on these findings, the fluxes at the instrumented end of HB-1 will not be significantly affected by the presence or absence of LH_2 in the cold source capsule located in HB-4.

2.4.4.4 Extent of Accelerated Silicon Production in the Throughtube

With a much more intense source of cold neutrons now located in the HB-4 beam tube, there was some concern over the extent to which this might accelerate silicon production in the aluminum beam tube. This process of transmuting the structural aluminum into silicon occurs when Al-27 absorbs a neutron, becomes Al-28, and very quickly decays via beta emission to become silicon. With many more neutrons present at very low energies, this transmutation process will be accelerated. The question is, "to what extent?" Moreover, since the HB-4 beam tube is both a semipermanent component and a pressure boundary, the question is worth investigating.

Table 2.13. Comparison of relative fluxes^a in HB-1 with and without the HB-4 LH₂ cold source

Energy range (eV)	With LH ₂ cold source	Without LH ₂ cold source	Ratio
$1 \times 10^5 - 2 \times 10^6$	0.050 ± 0.002	0.052 ± 0.002	0.96
$1 \times 10^2 - 1 \times 10^5$	0.111 ± 0.003	0.107 ± 0.003	1.04
$0.625 - 1 \times 10^2$	0.099 ± 0.003	0.097 ± 0.003	1.02
0.104 - 0.625	0.115 ± 0.003	0.114 ± 0.003	1.01
0.01 - 0.104	0.597 ± 0.010	0.596 ± 0.009	1.00
0.00325 - 0.01	0.031 ± 0.001	0.029 ± 0.001	1.05
0.00145 - 0.00325	0.0034 ± 0.0002	0.0034 ± 0.0002	0.99
$8.5 \times 10^{-4} - 0.00145$	0.0006 ± 0.00007	0.0005 ± 0.00006	1.13
$< 8.5 \times 10^{-4}$	0.00020 ± 0.00003	0.00019 ± 0.00003	1.09
Total	1.01 ± 0.01	1.00 ± 0.01	1.01

^aAll fluxes are normalized relative to the total neutron flux in HB-1 without the HB-4 LH₂ cold source. The ratio represents the expected nominal change in the HB-1 neutron flux when the LH₂ cold source is added to the system.

First, it must be recognized that cold neutrons are readily absorbed and cannot travel very far in most room temperature materials without being absorbed or upscattered into higher energy levels. The cold neutron flux levels in the HB-4 beam tube will be less than at the tip of the cold source vacuum vessel and/or capsule. At most, the cold neutron flux levels in the hemispherical tip of the HB-4 beam tube will be comparable to those in the hemispherical tip of the aluminum cold source capsule immediately adjacent to the liquid hydrogen in the present neutronic model.

Estimates of the silicon production rate at any position can be readily and reliably estimated from the Al-28 decay-beta heating rates, which were based on the actual steady-state Al-27 absorption rate and an average beta-decay energy of 1.2423 MeV. Using this approach, it was estimated that 0.7% of all the aluminum atoms in the cold source capsule will be converted to silicon every "full power year" (assuming the HFIR is operating at 100-MW, 365.25 full-power days per year). The amount of Si produced in the wall of the HB-4 beam tube close to the cold source vacuum vessel will be less than the 0.7% rate, but not substantially less. This rate of transmutation is believed to be slightly higher than the present transmutation rate and consistent with increases observed for a similar analysis performed for the ANS reactor cold source. The implication of this increase has been factored into component lifetime estimates.

2.4.4.5 Impact of Cold Source on Other HFIR Facilities

The HFIR is a multifunction research reactor that has facilities for isotope production, materials irradiation, and beam experiments. In its current configuration, the HFIR has 12 irradiation positions in the removable beryllium reflector (RBs), 8 irradiation positions in the control rod access plugs (CRs), 22 vertical experiment facilities (VXF) in the permanent reflector, and 4 beam tubes. There are also many irradiation and production positions located in the central hole of the annular fuel element. The proposed cold source represents a change in the configuration of the reflector system, and therefore may have an impact on the neutron fluxes at these facilities. Calculations were performed with and without the cold

source to determine its impact on the reactor in terms of the neutron fluxes at these experimental facilities as well as on the core reactivity.

The MCNP calculations for the base case (no cold source) and the perturbed case (with cold source) were performed with 5 million neutron histories. The ratio of the effective multiplication factors (k_{eff} with cold source/ k_{eff} without cold source) was determined to be 0.9992. This indicates essentially no reactivity impact on the core.

In addition to the multiplication factors, the neutron fluxes in the RB and VXF positions were also calculated. The results of the calculations for the nearest large RB position (RB-7A), the nearest small RB position (RB-8), and the nearest VXF position (VXF-22) are given in Tables 2.14, 2.15, and 2.16, respectively. The results indicate that the changes in the neutron fluxes in these regions are small and are generally within the MCNP calculational statistics.

Table 2.14. Comparison of RB-7A neutron fluxes with and without HB-4 LH₂ cold source

Energy range (eV)	RB-7A neutron fluxes without HB-4 cold source (s ⁻¹ · cm ⁻²)	RB-7A neutron fluxes with HB-4 cold source (s ⁻¹ · cm ⁻²)	Nominal ratio with/without
>1.83 × 10 ⁵	4.85 ± 0.09 × 10 ¹⁴	4.78 ± 0.09 × 10 ¹⁴	0.99
101–1.83 × 10 ⁵	6.85 ± 0.12 × 10 ¹⁴	6.90 ± 0.12 × 10 ¹⁴	1.01
0.414–101	4.10 ± 0.07 × 10 ¹⁴	4.11 ± 0.07 × 10 ¹⁴	1.00
<0.414	1.35 ± 0.02 × 10 ¹⁵	1.35 ± 0.02 × 10 ¹⁵	1.00

Table 2.15. Comparison of RB-8 neutron fluxes with and without HB-4 LH₂ cold source

Energy range (eV)	RB-8 neutron fluxes without HB-4 cold source (s ⁻¹ · cm ⁻²)	RB-8 neutron fluxes with HB-4 cold source (s ⁻¹ · cm ⁻²)	Nominal ratio with/without
>1.83 × 10 ⁵	5.17 ± 0.21 × 10 ¹⁴	5.08 ± 0.21 × 10 ¹⁴	0.98
101–1.83 × 10 ⁵	6.20 ± 0.26 × 10 ¹⁴	6.04 ± 0.25 × 10 ¹⁴	0.97
0.414–101	4.32 ± 0.19 × 10 ¹⁴	4.06 ± 0.19 × 10 ¹⁴	0.94
<0.414	1.11 ± 0.03 × 10 ¹⁵	1.11 ± 0.03 × 10 ¹⁵	0.99

Table 2.16. Comparison of VXF-22 neutron fluxes with and without HB-4 LH₂ cold source

Energy range (eV)	VXF-22 neutron fluxes without HB-4 cold source (s ⁻¹ · cm ⁻²)	VXF-22 neutron fluxes with HB-4 cold source (s ⁻¹ · cm ⁻²)	Nominal ratio with/without
>1.83 × 10 ⁵	4.22 ± 0.27 × 10 ¹³	4.34 ± 0.28 × 10 ¹³	1.03
101–1.83 × 10 ⁵	1.03 ± 0.05 × 10 ¹⁴	1.12 ± 0.05 × 10 ¹⁴	1.09
0.414–101	9.92 ± 0.41 × 10 ¹³	1.11 ± 0.04 × 10 ¹⁴	1.12
<0.414	8.39 ± 0.15 × 10 ¹⁴	8.50 ± 0.15 × 10 ¹⁴	1.01

2.5 THERMAL ANALYSIS OF KEY COLD SOURCE COMPONENTS

This section describes the detailed multidimensional thermal analyses performed on two key cold source components: the moderator vessel and the vacuum tube.

2.5.1 Description of CFD Code CFX-4

The numerical studies were carried out using the commercial computational fluid dynamics (CFD) computer code CFX-4 (see ref. 1), running on the IBM Risc/6000 workstation (Model 580) under the control of the AIX operating system, Version 3.2.5. CFX-4 is a suite of computer programs for the prediction of laminar and turbulent fluid flow and heat transfer. In addition to convective heat transfer within fluids, CFX-4 can also be used to simulate conjugate heat conduction within solid structures bounding and/or within the flow field. Internal heat generation can be accommodated through user-supplied subroutines. The governing system of partial differential equations for conservation of mass, momentum, and energy are discretized using generalized coordinates and solved with a modified finite-volume based SIMPLE algorithm.¹⁰

Originally written at Harwell Laboratories in the United Kingdom, CFX-4 is currently being developed by AEA Technology, plc, at Didcot, Oxfordshire, UK, and marketed in the United States by AEA Technology Engineering Software, Inc, Pittsburgh, PA. The code has been certified by Lloyd's Register Quality Assurance Company with ISO 9001 approval of its quality management program for software development, validation, and support.

2.5.1.1 Conservation Law System

The turbulent flow of a liquid cryogen with internal heat generation is governed by the physical principles of conservation of mass, momentum, and energy. An appropriate mathematical form for this conservation law system is the coupled set of nonlinear partial differential equations called the Reynolds-averaged Navier-Stokes equations for an incompressible (constant density) fluid. Using index notation, steady-state forms for the conservation of mass, momentum, and energy are the continuity equation

$$\frac{\partial U_j}{\partial x_j} = 0 \quad , \quad (1)$$

the momentum equation,

$$\frac{\partial}{\partial x_j} \left[\rho U_i U_j - \mu \left(\frac{\partial U_i}{\partial x_j} + \frac{\partial U_j}{\partial x_i} \right) + \rho \overline{u_i' u_j'} \right] + \nabla P = 0 \quad , \quad (2)$$

and the energy equation,

$$\frac{\partial}{\partial x_j} \left(\rho U_j H - \rho \alpha \frac{\partial H}{\partial x_j} + \rho \overline{u_j' h'} \right) - S_H = 0 \quad , \quad (3)$$

where ρ , μ , and α are the density, molecular dynamic viscosity, and molecular thermal diffusivity, respectively, of the fluid; U_i is the time-averaged velocity vector resolution of the mean flow field; H is the time-averaged static enthalpy; P is the *motion* pressure (equal to the thermostatic pressure minus the hydrostatic pressure), and S_H represents any distributed volumetric heat sources. Repeated indices imply summation over the computational domain, $\mathbb{R}^3$. In CFX-4, Eqs. (1)–(3) are discretized using generalized

coordinates with a nonstaggered finite volume mesh. Steady-state solutions for the discrete equation set are calculated using the SIMPLEC algorithm.¹¹

2.5.1.2 Turbulence Modeling

Arising from the time-averaging of the convection terms in the original instantaneous conservation law system, statistical double correlations appear in Eqs. (2) and (3) that characterize the effects of turbulence on the transport of momentum (Reynolds stresses, $\rho \overline{u_i' u_j'}$), and thermal energy (turbulent heat fluxes, $\rho \overline{u_i' h'}$) in the mean flow field. Transport equations for each of these double correlations can be derived, but these new equations unfortunately contain additional unknown higher-order correlations; therefore, Eqs. (1)–(3) do not represent a closed set, and an exact closure for these correlations does not exist. The result is the classic *turbulence closure problem*, and approximate closure is the task of turbulence models.

For engineering calculations, the most commonly used turbulence models are based upon the Boussinesq eddy-viscosity approximation, where the Reynolds stresses are modeled by a constitutive relation having a form similar to the Stokes viscosity law for Newtonian fluids,

$$-\rho \overline{u_i' u_j'} = -\frac{2}{3} \rho k \delta_{ij} + \mu_T \left(\frac{\partial U_i}{\partial x_j} + \frac{\partial U_j}{\partial x_i} \right), \quad (4)$$

where k is the turbulent kinetic energy per unit mass ($k \equiv \frac{1}{2} \overline{u_i' u_i'}$), δ_{ij} is the Kronecker delta, and μ_T is the *eddy viscosity*. Similar *eddy diffusivity* hypotheses, coupled with Reynolds-analogy approximations, result in constitutive relations for the turbulent heat flux

$$-\rho \overline{u_i' h'} = \Gamma_H \frac{\partial H}{\partial x_i} = \frac{\mu_T}{\sigma_H} \frac{\partial H}{\partial x_i}, \quad (5)$$

where Γ_H is the thermal eddy diffusivity, and σ_H is the turbulent Prandtl number (with a standard value of 0.9). Equations (4)–(5) represent defining relations for the eddy viscosity and eddy diffusivities. A method for calculating these quantities remains to be identified to obtain closure.

From dimensional arguments, turbulence models based on the Boussinesq approximation, Eq. (4), assume that the eddy viscosity is proportional to the product of length and velocity scales that are characteristic of the turbulence. The two-equation k - ϵ (ϵ is the isotropic turbulence dissipation rate) turbulence model uses the Prandtl-Kolmogorov equation,

$$\mu_T = C_\mu \rho \frac{k^2}{\epsilon}, \quad (6)$$

to relate k and ϵ to the eddy viscosity, μ_T . The term C_μ is an empirically-derived constant with the standard value of $C_\mu = 0.09$. For this model, the turbulent velocity scale is $\sqrt{k}$, and the characteristic length scale is $k^{3/2}/\epsilon$. Transport equations for k and ϵ are solved to complete the approximate turbulence closure, specifically,

$$\frac{\partial}{\partial x_j} \left[\rho U_j k - \left(\mu + \frac{\mu_T}{\sigma_k} \right) \frac{\partial k}{\partial x_j} \right] -$$

$$(\mu + \mu_T) \frac{\partial U_i}{\partial x_j} \left(\frac{\partial U_i}{\partial x_j} + \frac{\partial U_j}{\partial x_i} \right) + \rho \epsilon = 0 , \quad (7)$$

$$\frac{\partial}{\partial x_j} \left[\rho U_j \epsilon - \left(\mu + \frac{\mu_T}{\sigma_\epsilon} \right) \frac{\partial \epsilon}{\partial x_j} \right] -$$

$$C_1 \frac{\epsilon}{k} (\mu + \mu_T) \frac{\partial U_i}{\partial x_j} \left(\frac{\partial U_i}{\partial x_j} + \frac{\partial U_j}{\partial x_i} \right) + C_2 \rho \frac{\epsilon^2}{k} = 0 , \quad (8)$$

where σ_k , σ_ϵ , C_1 , and C_2 are empirically-derived constants. Their default values in CFX-4 are $\sigma_k = 1.0$, $\sigma_\epsilon = 1.217$, $C_1 = 1.44$, and $C_2 = 1.92$. In the standard high-Reynolds number k - ϵ model, the computational domain does not extend all the way to solid walls. So-called *wall functions*, derived from the logarithmic "law-of-the-wall" velocity profile, are employed to simulate the effect of the no-slip wall boundary condition on the flow field adjacent to the wall. Related *heat flux wall functions* are also available for the energy equation to calculate the heat transfer between the flow domain and solid boundaries.¹¹

2.5.1.3 Wall function boundary conditions

For the standard high-Reynolds number k - ϵ turbulence model, wall functions are employed to impose Neumann boundary conditions on the momentum and energy equations.

Definitions

The boundary of the mesh (at point p) is offset from a no-slip wall by a distance δ . The dimensionless distance related to this offset is expressed in terms of a Reynolds number type modulus, y^+ ,

$$y^+ \equiv \frac{\rho u^* \delta}{\mu} , \quad (9)$$

where u^* is called the *friction velocity*, μ is the molecular dynamic viscosity, and ρ is the density of the fluid. The friction velocity is further defined by

$$u^* \equiv \left(\frac{\tau_w}{\rho} \right)^{1/2} , \quad (10)$$

where τ_w is the shear stress at the wall. A dimensionless velocity can be defined by

$$u^+ \equiv \frac{u_p}{u^*} , \quad (11)$$

where it is assumed that the flow at point p is tangent to the wall.

Assumptions

The following assumptions provide the technical bases for the use of wall functions.

1. It is assumed that the boundary node at point p is in a region close to the wall (but not too close, $y^+ > 30$) where the rate of production of turbulent kinetic energy, $k \equiv \overline{u'_i u'_i} / 2$, is in balance with its dissipation rate. Through this region, the shear stress is constant and related to the turbulent kinetic energy by (p. 324, ref. 10)

$$\tau_w = \tau_p = \rho C_\mu^{1/2} k_p \quad , \quad (12)$$

where C_μ is an empirically-derived constant. Using Eqs. (9) and (10), the friction velocity and dimensionless distance can now be related to the turbulent kinetic energy in this region by

$$u^+ = C_\mu^{1/4} k_p^{1/2} \quad , \quad (13)$$

$$y^+ = \frac{\rho C_\mu^{1/4} k_p^{1/2} \delta}{\mu} \quad . \quad (14)$$

2. It is assumed that at point p , the logarithmic *law-of-the-wall* velocity profile¹⁰ holds,

$$u^+ = \begin{cases} y^+ , & \text{for } y^+ < y_0^+ \\ \frac{1}{\kappa} \ln(E y^+) , & \text{for } y^+ > y_0^+ \end{cases} \quad , \quad (15)$$

where y_0^+ is the larger root of the equation

$$y_0^+ = \frac{1}{\kappa} \ln(E y_0^+) \quad . \quad (16)$$

Equations (15) and (16) include a wall-roughness constant, E ($= 9.0$ for a smooth wall) and the von Karman constant, κ ($= 0.419$).

3. It is assumed that a similar logarithmic *law-of-the-wall* profile can be applied to describe the distribution of enthalpy near the wall. This profile has the form

$$H^+ = \begin{cases} \sigma_H y^+ , & \text{for } y^+ < y_H^+ \\ \frac{\sigma_H}{\kappa} \ln(E_H y^+) , & \text{for } y^+ > y_H^+ \end{cases} \quad , \quad (17)$$

where y_H^+ is the larger root of the equation,

$$y_H^+ = \frac{\sigma_H}{\kappa} \ln(E_H y_H^+) \quad . \quad (18)$$

The constant is calculated from a formula by Jayatillaka¹²

$$E_H = E \exp \left[9.0 \kappa \left(\left(\frac{\text{Pr}}{\sigma_H} \right)^{0.75} - 1 \right) \left(1 + 0.28 \exp \left(\frac{-0.007 \text{Pr}}{\sigma_H} \right) \right) \right], \quad (19)$$

where Pr is the molecular Prandtl number for the fluid. A dimensionless enthalpy, H^+ , can be defined by

$$\begin{aligned} H^+ &= \frac{(\rho \tau_w)^{1/2}}{q_w} (H_w - H_p) \\ &= \frac{(\rho \tau_w)^{1/2}}{q_w} c_p (T_w - T_p). \end{aligned} \quad (20)$$

where H is the mass-specific enthalpy, q_w is the heat flux at the wall (assumed to be constant between the wall and point p), c_p is the specific heat at constant pressure, and T_w and T_p are the temperatures at the wall and at point p , respectively.

Wall functions

Wall functions are employed in imposing Neumann boundary conditions on the momentum and energy equations. Solving Eq. (15) [with Eqs. (10)–(14)] for the shear stress at the mesh boundary, results in

$$\tau_p = \tau_w = \frac{\rho \kappa u_p C_\mu^{1/4} k_p^{1/2}}{\ln \left(E \frac{\rho C_\mu^{1/4} k_p^{1/2} \delta}{\mu} \right)}, \quad (21)$$

or if δ is unknown,

$$\tau_p = \tau_w = \frac{\rho \kappa u_p C_\mu^{1/4} k_p^{1/2}}{\ln(E y_0^+)}. \quad (22)$$

Therefore, given current estimates for u_p and k_p , the shear stress at point p can be calculated from either Eq. (21) or Eq. (22) and then applied as a Neumann boundary condition coupled with a tangency condition on the momentum equations. Iterative cycling between the momentum equations (constrained by the conservation of mass) and the k - ϵ turbulence model should produce convergence to the final wall shear stress.

For the energy equation boundary condition, Eq. (17), with Eq. (20), can be solved for the heat flux at the wall, obtaining the energy equation wall function

$$q_p = q_w = \frac{\rho \kappa C_\mu^{1/4} k_p^{1/2} c_p (T_w - T_p)}{\sigma_H \ln \left(E_H \frac{\rho C_\mu^{1/4} k_p^{1/2} \delta}{\mu} \right)}. \quad (23)$$

Given a fixed-temperature boundary condition at the wall, T_w , along with current estimates for T_p and k_p , the heat flux at point p can be calculated and applied as a Neumann boundary condition on the energy equation. Conversely, if the heat flux at the wall is given as problem data, then this heat flux can be applied directly at point p as a Neumann boundary condition. For this latter case, Eq. (23) can be solved for T_w , and the effective wall temperature can then be calculated as a post-processing operation upon convergence of iterative cycling between the energy and momentum equations and the turbulence model.

2.5.1.4 Application of Nuclear Heating

Nuclear heat, resulting from neutron and gamma bombardment of the aluminum walls of the cold source moderator vessel and vacuum tube, the hydrogen flowing through the cold source, and the water in the space between the vacuum tube and the beam tube, is simulated in the CFX-4 code as internally distributed heat sources that are added to the energy equation, Eq. (3). Look-up tables of nuclear heating values in terms of W/g of material were developed as reported in Sect. 2.4.1.5. These data are presented as a function of position along the HB-4 beam tube as measured from the midpoint of the HB-1/4 beam tubes. Near the reactor, the data are also divided into reactor-side and far-side heating values. A schematic drawing of the moderator vessel, along with representative nuclear heating data for the reactor-side of the moderator vessel as a function of location, is shown in Fig. 2.18. The hydrogen cryogen has a maximum nuclear heating rate of 20.2 W/g with a rapid decline moving away from the reactor core. Both the aluminum moderator vessel and vacuum tube have a maximum nuclear heating value of 5.75 W/g at positions nearest the core with a more gradual rate of decrease of heating (relative to the hydrogen) as one moves down the beam tube.

The following code fragment demonstrates how nuclear heating data is applied to the energy equation source terms in the user-supplied subroutine USRSRC.

```
C---- EXAMPLE OF INTERNAL HEAT SOURCE: ADD 20.0 W/g IN 'BLOCK-NUMBER-2'
C
C---- STATEMENT FUNCTION FOR ADDRESSING
      IP(I,J,K)=IPT((K-1)*ILEN*JLEN+(J-1)*ILEN+I)
      :
      DENSH2 = 74.750E+3 !(g/m^3) density of hydrogen at 15 bar,18 K
      :
      CALL GETVAR('USRSRC','H      ',IVAR) !FIND VARIABLE NUMBER FOR ENTHALPY
      IF (IVAR .EQ. IEQN) THEN
C USE IPREC TO FIND ADDRESSES
      CALL IPREC('BLOCK-NUMBER-2','BLOCK','CENTRES',IPT,ILEN,JLEN,KLEN,
      &          CWORK,IWORK)
C
C LOOP OVER BLOCK
      DO 103 K = 1, KLEN
        DO 102 J = 1, JLEN
          DO 101 I = 1, ILEN
            INODE = IP(I,J,K) ! USE STATEMENT FUNCTION IP TO GET ADDRESSES
            SU(INODE,1)=SU(INODE,1)+20.0*DENSH2*VOL(INODE)!ADD HEAT SOURCE
101      CONTINUE
102      CONTINUE
103      CONTINUE
      ENDIF
```

Ref.: J. A. Bucholz, "Discussion of New Calculations for the July 1996 Conceptual Design: MNCP Results from Run HF072 ...," Internal Correspondence to D. L. Selby, Oak Ridge National Laboratory, Oak Ridge, TN, Aug. 16, 1996.

Fig. 2.18. Nuclear heating of cold source components.

The mass-specific heat load in W/g is multiplied by the density of the material and the volume of the node to obtain a total internal heat source for the node in watts. This heating rate is then added to any preexisting source terms that arise from the SIMPLEX algorithm. The model geometry is divided into connecting blocks, and in each block there is a single material. Blocks containing aluminum would, therefore, use the density of aluminum to calculate the heat source term.

2.5.2 Moderator Vessel Analysis

A 3-D thermal-hydraulic analysis was carried out on the moderator vessel for a range of operating conditions. The nominal steady-state operating condition is based on a supercritical hydrogen pressure of 15 bar absolute (217.6 psia) with a supply temperature from the refrigeration system of 17 K and an average flow rate through the vessel of 1 L/s. It was also desirable to obtain performance information for the lower subcritical pressure of 4 bar absolute (58.0 psia). A standby operating mode is also envisioned that requires the moderator to run at a steady-state condition of 15 bar absolute (217.6 psia) with a supply temperature of 100 K and a flow rate of 4 L/s.

2.5.2.1 Model Geometry

An assembly drawing depicting the geometry of the moderator vessel is shown in Fig. 2.19. The moderator vessel is constructed of Al 6061-T6 with wall thicknesses varying from 0.039 in. to 0.158 in. There were competing design requirements between maintaining the structural integrity of the vessel under pressure and thermal stress loads and minimizing the mass of aluminum to keep the total nuclear

Fig. 2.19. Moderator vessel assembly.

heating rate as low as possible. These design requirements produced the varying wall thicknesses. The overall length of the vessel is 6.031 in. with an inner radius at the hemispherical endcap of 1.86 in. The hydrogen cryogen enters the vessel at the lower nozzle, and after one pass exits at the upper nozzle, both nozzles having an inner diameter of 0.375 in.

The CFD model, constructed with 94,880 finite volumes, consisted of the aluminum walls (with varying thickness) of the moderator vessel and the volume displaced by the hydrogen moderator. In all cases, the hydrogen was assumed to be incompressible, and a standard two-equation $k-\epsilon$ turbulence model was employed to determine the effects of turbulence on the flow field and heat transfer because of turbulent mixing within the hydrogen.

2.5.2.2 Nuclear Heating and Boundary Conditions

Applied as spatially-distributed internal heat generation sources, nuclear heating of both the aluminum and hydrogen was included in the model. Heat transfer between the hydrogen and the moderator vessel was calculated by turbulent “law-of-the-wall” logarithmic wall functions. A fixed uniform velocity boundary condition was applied at the inlet that, when integrated over the inlet cross-sectional area, would give the required total volumetric flow rate. Since for an incompressible fluid the absolute pressure is not used explicitly in the calculations (only the pressure gradient), a relative reference pressure of 0.0 can be applied to the outflow plane. This boundary condition results in a “continuative” or “vanishing normal velocity derivative” outflow boundary. All external aluminum surfaces had radiative boundary conditions applied, assuming the moderator vessel had a complete view of the vacuum tube (set at a constant 373 K) with blackbody radiative properties. This assumption is conservative in that it overstates the amount of heat gain by radiation from the surrounding vacuum tube.

2.5.2.3 Thermophysical Properties

The density of aluminum is 2.7 g/mL. Thermal conductivity data^{13,14,15} for the aluminum alloy Al 6061-T6 is presented in Fig. 2.20 over a range of temperatures and compared to the thermal conductivities of pure aluminum and two other alloys, Al 5052 and Al 2024-T4, obtained from the literature.¹⁶ Pure aluminum at cryogenic temperatures has a very high thermal conductivity that increases with decreasing temperature. The presence of even a small amount of alloying material or contaminant, however, reverses this trend with a decreasing conductivity for a decreasing temperature. In addition, some reduction in the thermal conductivity is expected because of radiation-induced effects; however, the degree of change for Al 6061 is not known.¹⁷ Since both electrical and thermal conduction are determined primarily by the free electrons, an increase in electrical resistivity implies a decrease in thermal conductivity. Resistivity measurements in irradiated copper have shown increases as large as about 30% for doses above 1 dpa. Calculations, to be reported in the next section, were carried out to determine the sensitivity of the results to uncertainties in the assumed aluminum thermal conductivity. At 20 K, Al 2024-T4 has a thermal conductivity that is approximately 45% below that of Al 6061-T6.

At room temperature, the equilibrium mixture of “normal” hydrogen is fixed at 25% para and 75% ortho; however, for the temperature range of 20 to 250 K, the percentages of parahydrogen and orthohydrogen vary appreciably at equilibrium.¹⁸ Below 20 K, hydrogen is essentially all parahydrogen in the absence of irradiation. Under irradiation conditions, some amount of orthohydrogen is produced even at 20 K and below. Except for small differences in specific heat, however, the thermodynamic and transport properties of para and normal hydrogen are effectively the same. The property data for parahydrogen used in these calculations were obtained from the NIST Standard Reference Database for Pure Fluids.¹⁹

Figure 2.21 shows the relationship between the reference supercritical hydrogen operating mode and a liquid hydrogen system on a plot of pressure vs enthalpy (note that the low pressure liquid conditions are also shown for information). For the high pressure and standby operating modes between 14 and

Fig. 2.20. Thermal conductivity of aluminum and alloys.

15 bar absolute, the hydrogen is a supercritical fluid. The low-temperature states (17–22 K) are shown to lie close to each other on a temperature vs entropy plot even in a liquid phase condition (see Fig. 2.22). Figure 2.22 also depicts a zone of thermodynamic states called the “near-critical” region¹⁸ that for hydrogen is approximately bounded by the reduced pressures and the temperatures of at the reduced pressure and the transposed critical temperature, T^* , which is defined as the temperature where the specific heat attains a maximum for a given supercritical pressure. For hydrogen at 15 bar absolute, as shown in Fig. 2.23 with specific heat and density plotted vs temperature. The difficulties with operating within this near-critical region are because of the uncertainties associated with accurately predicting transport properties and convective heat transfer.¹⁸ The operating states proposed for the cold source lie well outside of the near critical region; however, certain operational transients (such as high-pressure cool down) will pass through this region. The transport properties of thermal conductivity and viscosity are shown in Fig. 2.24. properties of para and normal hydrogen are effectively the same. The property data for parahydrogen used in these calculations were obtained from the NIST Standard Reference Database for Pure Fluids.¹⁹

Fig. 2.21. Pressure vs enthalpy for hydrogen.

Fig. 2.22. Temperature vs entropy for hydrogen.

Fig. 2.23. Specific heat and density of parahydrogen.

Fig. 2.24. Thermal conductivity and viscosity of hydrogen.

2.5.2.4 Model Results

Steady-state results for 15 cases are summarized in Tables 2.17 and 2.18. The nominal low-temperature/high-pressure steady-state operating condition requires a supercritical hydrogen pressure of 15 bar (217.6 psia) with a supply temperature of 18 K at the inlet to the moderator vessel and a flow rate of 1.0 L/s. The hydrogen flow field is characterized by an expanding jet, formed at the inlet, that flows up the wall of the hemispherical endcap (see Fig. 2.25). The jet then bifurcates as it rolls over, forming a recirculating cell that provides a downwash across the face of the ellipsoidal interior wall. Flow reversals are observed in both the inlet and outlet transition regions. The results for the hydrogen moderator/cryogen are presented in Table 2.17. For all cases, the integrated nuclear heat load on the vacuum vessel is 1.49 kW. An additional 0.04 kW has been added to the vessel nuclear heating to allow for the net heat transfer because of thermal radiation interchange between the vacuum vessel and the moderator vessel. Nuclear heating of the cryogen within the moderator varies with the density of the hydrogen. For the two low pressure cases at 8 bar absolute (116 psia) and 4 bar absolute (58 psia), the hydrogen bulk temperature remains subcooled throughout the moderator vessel. For all cases at the design flow rate of 1 L/s, the hydrogen pressure drop from inlet to outlet ranges from 0.105 bar (1.52 psi) to 0.135 bar (1.96 psi).

Table 2.17. Summary of moderator vessel cases: hydrogen results

H ₂ operating conditions			Nuclear heating ^{a,b}		H ₂ results	
Pressure [bar (psia)]	Inlet temp. (K)	Flow rate (L/s)	Vessel (kW)	H ₂ (kW)	∇P [bar (psi)]	Outlet temp. (K)
15 (217.6)	18	1	1.53	0.63	0.135 (1.96)	21.7
8 (116.0)	18	1	1.53	0.62	0.133 (1.93)	21.6
4 (58.0)	18	1	1.53	0.60	0.129 (1.87)	21.2
15 (217.6)	100	4	1.53	0.03	0.105 (1.52)	109.2
15 (217.6)	100	3	1.53	0.03	0.059 (0.85)	112.3
15 (217.6)	18	1.2	1.53	0.63	0.194 (2.82)	21.2
15 (217.6)	18	0.8	1.53	0.63	0.086 (1.25)	22.6

^aAssumes HFIR operating at 100 MW_{th}.

^bIncludes 0.04 kW of thermal radiative heating from the vacuum vessel at 373 K.

Table 2.18 presents maximum vessel wall temperatures. The first three cases investigate the sensitivity of the results to $\pm 20\%$ variations in flow rate at a hydrogen pressure of 15 bar absolute (217.6 psia). Subsequent cases check the sensitivity of the maximum wall temperature results to uncertainties in the thermal conductivity of the walls and changes in operating pressure. Using thermal conductivity data for Al 2024-T4 results in a decrease in wall heat conduction of approximately 45%. As an upper bound on the most optimistic estimate for wall conduction, property data for pure aluminum was applied. For the low temperature cases, the wall temperatures remain below 47 K. A 20% reduction in flow rate raises the peak wall temperature by 3 K; however, increasing the flow rate by 20% only lowers the wall temperature by 0.9 K. The most optimistic prediction for peak wall temperature is 28.7 K for pure aluminum.

Table 2.18. Summary of moderator vessel cases: vessel wall results

H ₂ operating conditions			Aluminum type	k/k ₆₀₆₁ ^a	Max. vessel temp. (K)	Max. wall super heat ^b (K)
Pressure [bar (psia)]	Inlet temp. (K)	Flow rate (L/s)				
15 (217.6)	18	1.0	6061-T6	1.0	43.2	NA
15 (217.6)	18	1.2	6061-T6	1.0	42.3	NA
15 (217.6)	18	0.8	6061-T6	1.0	46.2	NA
15 (217.6)	18	1.0	2024-T4	0.548	46.6	NA
15 (217.6)	18	1.0	pure Al	428	28.7	NA
8 (116.0)	18	1.0	6061-T6	1.0	37.6	+7.8
8 (116.0)	18	1.0	2024-T4	0.548	44.6	+14.8
8 (116.0)	18	1.0	pure Al	428	28.6	-1.2
4 (58.0)	18	1.0	6061-T6	1.0	37.8	+11.8
4 (58.0)	18	1.0	2024-T4	0.548	45.2	+19.2
4 (58.0)	18	1.0	pure Al	428	27.7	+1.7
15 (217.6)	100	4.0	6061-T6	1.0	139.9	NA
15 (217.6)	100	4.0	2024-T4	0.51	144.5	NA
15 (217.6)	100	4.0	pure Al	3.1	135.4	NA
15 (217.6)	100	3.0	6061-T6	1.0	146.8	NA

^aRelative conductivity = thermal conductivity/conductivity of 6061-T6 at 20 K or 100 K for standby cases.

^bSuperheat = (T_{wall} - T_{sat}); for H₂: T_{sat} = 29.84 K at 8 bar (116.0 psia); T_{sat} = 25.96 K at 4 bar (58.0 psia).

The results for Al 2024-T4 provide conservative estimates of wall superheat and maximum wall temperatures. At a subcritical pressure of 8 bar (116.0 psia), the maximum vessel wall temperature is 44.6 K or 14.8 K above the saturation temperature of 29.84 K. Upon lowering the pressure to 4 bar (58.0 psia), a portion of the vessel wall near the outlet reaches a temperature of 45.2 K or 19.2 K above the saturation temperature of 25.96 K. Because of this superheating of the walls, there is a potential for some subcooled nucleate boiling at 4 bar (58.0 psia). In its standby mode of operation at 15 bar (217.6 psia), the hydrogen temperature could rise to a supply inlet temperature of 100 K with an increased flow rate of 4.0 L/s. Corresponding to a reduced hydrogen density at this temperature, the nuclear heating of the hydrogen is also reduced from a nominal 0.63 kW at 18 K to 0.03 kW at 100 K. The maximum wall temperature for this case is 144.5 K.

2.5.3 Vacuum Tube Analysis

The external environment for the moderator vessel is a hard vacuum, contained by an aluminum tube that sits inside the beam tube. Since this vacuum tube will be exposed to neutron and gamma radiation, some means of removing its nuclear heating load must be provided. The beam tube is cooled with forced convection "by water flowing through 24 parallel milled grooves cut longitudinally into, and equally

Fig. 2.25. Moderator wall temperature distribution and flow field for 1 L/s of H_2 at 15 bar.

spaced around, the periphery of the tube. Defined flow passages are formed by covering the beam tube with a close-fitting jacket. Reactor primary cooling water is forced through the coolant jacket by the pressure drop across the reactor core.²⁰ The gap between the vacuum tube and the beam tube will be filled with water flowing along the length of the tubes. This water-filled gap provides a thermal bridge between the vacuum tube and beam tube with the ultimate heat sink being the cooling water flowing along the outside of the beam tube.

2.5.3.1 Model Geometry

Two vacuum tube geometries were investigated for this study, designated as Model No. 6 in Fig. 2.26 and Model No. 7 in Fig. 2.27. Earlier models examined heat transfer across varying gap widths and different gap fluids. In each of these figures, the relative position of the moderator vessel is shown in outline; however, the moderator itself was not explicitly a part of the vacuum tube models. Vacuum tube wall thicknesses in Model Nos. 6 and 7 vary from a minimum of 0.1 in. at the hemispherical endcap to a maximum of 0.27 in. beyond the transition region. The minimum gap width is 0.042 in. along the 15.35-in. straight section of Model No. 6, near the reactor core. Model No. 6 also has an 8-in. transition piece in which the diameter of the vacuum tube/beam tube assembly increases as well as an increase in gap width to 0.079 in. Model No. 7 has a shorter transition region than Model No. 6, along with a thinner vacuum tube wall and a larger gap width in the straight section between the endcap and the transition piece. The overall length of both models extended over a distance of approximately 78.7 in.

Geometry for CFD Model #6
BEAM TUBE/VACUUM TUBE ASSEMBLY

P.T. WILLIAMS 10/6/96

all dimensions in inches

Fig. 2.26. Vacuum tube geometry for model No. 6.

Technical drawing of a beam tube assembly. The drawing includes a cross-section at the bottom and a side view above it. The cross-section shows a semi-circular beam tube with a central gap. Dimensions for the cross-section include: wall thickness of 0.1250, gap of 0.0250, and radii R2.3125, R2.6875, R2.2350, and R2.1100. The side view shows the beam tube (0.3750 thick) and a vacuum tube (0.2700 thick) with a gap of 0.0790. The total length is 14.0, with segments of 3.0, 2.7130, 2.9830, 3.0620, and 3.4370. A coordinate system (x, y, z) is shown at the bottom left. The text 'all dimensions in inches' is present.

P.T. WILLIAMS 10/31/96

all dimensions in inches

Fig. 2.27. Vacuum tube geometry for model No. 7.

2.5.3.2 Nuclear Heating and Boundary Conditions

As in the moderator vessel analysis, nuclear heating of the aluminum beam and vacuum tube as well as the water in the gap is included in the model by adding source terms to the energy equation. The mass-specific nuclear heating rates were obtained as described in Sect. 2.4.1.5, which assumes that the HFIR is operating at 100 MW_{th}. As shown in Table 2.19, Model No. 6 produces integrated heat loads (over a length of 78.7 in.) of 15.13 kW for the beam tube, 6.12 kW for the vacuum tube, and 0.91 kW in the water gap. In Model No. 7, with a slightly thinner vacuum tube and larger water gap, the integrated heat loads are 15.5 kW for the beam tube, 5.35 kW for the vacuum tube, and 1.43 kW for the water gap. Except for Case 5b in Table 2.19, the inner surface of the vacuum tube was assumed adiabatic. For Case 5b, the first 3 in. nearest the reactor are assumed to have a perfect view of the moderator vessel at a fixed temperature of 20 K with a blackbody thermal radiation boundary condition. The outer surface of the beam tube is fixed at a temperature of 48.89 C. For the forced-convection cases, water is introduced into the gap at the far end of the beam tube and flows along the bottom of the vacuum tube in a channel formed by two railings that support the vacuum tube inside the beam tube. At the hemispherical endcap, the flow sweeps up the wall and turns to return along the length of the beam tube/vacuum tube, where it is ultimately withdrawn (see Fig. 2.28).

2.5.3.3 Thermophysical Properties

The aluminum beam tube and vacuum tube have a density of 2.7 g/mL and a thermal conductivity of 210 W/m-K. Water properties at a temperature of 48.89 C were obtained from the "Standard Fluid" properties library included with the CFX-4 code. The density of water is 988.2 g; the thermal conductivity is 0.6423 W/m-K; the viscosity is 5.417×10^{-4} Pa-s; and the isobaric thermal expansion coefficient is 4.347×10^{-4} K⁻¹.

2.5.3.4 Model Results

A summary of the results for eight cases are presented in Table 2.19. A typical temperature distribution is shown in Fig. 2.29 for the forced-convection case of Case No. 8. As the flow sweeps up along the face of the hemispherical endcap of the vacuum tube, it turns to flow back down the beam tube away from the reactor. This turning produces a flow separation on the sides of the vacuum tube. It is within this separation region that the maximum vacuum tube temperature of 67.6 C occurs. The total volumetric water flow rate is 0.4 L/s.

Case No. 1 assumes a stagnant water gap in which no natural convection currents are allowed to develop. This case is not physically real but does represent a conservative upper bound on the heat transfer resistance of the water-filled gap. The maximum vacuum tube temperature is an excessive 237°C. In Cases 2 and 5-7, buoyancy-induced natural convection currents are calculated by CFX-4 resulting in a significant drop in peak vacuum tube temperature, located on the side of the hemispherical endcap nearest the reactor. For forced convection, Cases 3 and 4 show the sensitivity of the results to the assumed water supply temperature. Dropping the supply temperature by 22.2°C results in a decrease in peak vacuum tube temperature of only 1.7°C. This result highlights the fact that the water flowing through the gap is not intended to remove the nuclear heat load of the vacuum tube, but rather it serves to enhance the thermal bridging effect between the vacuum tube and beam tube. Cases 5a and 5b check the sensitivity of the analysis to the presence of the moderator vessel as a cryogenic heat sink at 20 K. In Case 5b, the inner surface of the first 3 in. of the vacuum tube is allowed to see the moderator vessel and transfer heat by thermal radiation. This additional heat sink only drops the predicted peak vacuum tube temperature by 1.4°C. Cases 6 and 7 check the sensitivity of the natural convection calculations to the

Table 2.19. Results of vacuum tube analysis

Case No.	Model No.	Offset from HFIR core (in.)	Nuclear heating			Water flow rate (L/s)	Water supply temp. (C)	Maximum vacuum tube temp. (C)
			Beam tube (kW)	Vacuum tube (kW)	Water gap (kW)			
1	7	0	15.50	5.35	1.43	CO ^a	CO ^a	237.0
2	6	0	15.13	6.12	0.91	NC ^b	NC ^b	114.2
3	6	0	15.13	6.12	0.91	0.4	48.9	68.7
4	6	0	15.13	6.12	0.91	0.4	26.7	67.0
5a ^c	7	0	15.50	5.35	1.43	NC ^b	NC ^b	117.4
5b ^c	7	0	15.50	5.35	1.43	NC ^b	NC ^b	116.0
6	7	4	10.13	3.46	0.98	NC ^b	NC ^b	107.1
7	7	6	7.98	2.70	0.75	NC ^b	NC ^b	95.4
8	7	0	15.50	5.35	1.43	0.4	48.9	67.6

^aCO-conduction only in water gap.^bNC-natural convection in water gap.^cCase 5b is identical to Case 5a except that the cold source is running and acts as a radiative heat sink at 20 K.

Fig. 2.28. Water flow path for forced-convection cooling of vacuum tube.

Fig. 2.29. Temperature distribution in vacuum tube.

uncertainty in the position of the vacuum tube relative to the HFIR core. Offsetting the beam tube/vacuum tube assembly away from the reactor by 4 in. and 6 in. reduces the predicted peak vacuum tube temperature from 117.4°C to 107.1°C and 95.4°C, respectively.

2.5.4 Conclusions

Multidimensional thermal-hydraulic analyses have been carried out for two key cold source components: the moderator vessel and the vacuum tube.

The hydrogen flow field in the moderator vessel consists of a number of complicated 3-D flow structures, characterized by an expanding jet, formed at the inlet, that flows up the wall of the hemispherical endcap. The jet then bifurcates as it rolls over, forming a recirculating cell that provides a downwash across the face of the ellipsoidal interior wall. Flow reversals are observed in both the inlet and outlet transition regions. It is within these separation regions that the peak wall temperatures occur. For the supercritical low-temperature and standby operating modes, the temperature gradients are acceptable from a thermal-hydraulic point of view. Additional thermal stress calculations will address any issues concerning the mechanical design and structural integrity of the vessel. At the two subcritical pressures investigated, 8 bar (116.0 psia) and 4 bar absolute (58.0 psia), significant wall superheat was predicted within the transition regions of the moderator vessel. At these levels of superheat, there is a potential for localized subcooled nucleate boiling. This is one of the reasons why the liquid operational mode at lower pressure was dropped from the reference concept. If interest in the liquid operational mode is revived, the effects of such boiling on cold source operation and moderator performance cannot be addressed by the current CFD model, since multiphase flow with boiling is not simulated by CFX-4. These issues would have to be investigated by both 1-D transient systems analyses (e.g., ATHENA²¹ modeling) and experimental testing.

Two design options were investigated for the vacuum tube. For both designs, forced convection cooling of the vacuum tube with a water flow rate of 0.4 L/s produced acceptable temperature gradients with a maximum vacuum tube temperature of 68.7°C. Again, mechanical design issues will be addressed by appropriate thermal stress calculations. If the water flow is lost, then the peak vacuum tube temperature rises to 117.4°C. If the water pressure is below approximately 1.83 bar absolute (26.5 psia), then boiling will occur.

2.6 MODERATOR VESSEL AND VACUUM TUBE STRUCTURAL ANALYSIS

Because of their small size, the moderator vessel and vacuum tube are exempted from the full requirements of a stamped ASME coded pressure vessel. Nevertheless, they will be designed to the intent of that code for all operating conditions. A finite element stress analysis has been used to evaluate stresses in both the moderator vessel and vacuum tube. In addition, a probabilistic fracture analysis has been performed for the moderator vessel. It is also planned to test an actual test vessel and measure dimensional changes at pressures up to 1.5 times its design pressure. Sections 2.6.1 and 2.6.2 provide a summary of the work that has been performed.

2.6.1 HFIR Cold Neutron Source Moderator Vessel Structural Design Analysis

The moderator vessel is designed by the computer-aided design code I-DEAS,²² and the stress distribution of the model is obtained by the finite element code ABAQUS.²³ The moderator vessel is made of 6061-T6 aluminum alloy and is modeled approximately by linear shell elements. The design pressure is 19 bars that corresponds to the rupture disk pressure of the moderator structure. The normal operating pressure is 15 bars. The design follows the ASME BPV Code Section 3, Division 1 guideline that permits the use of design by analysis. The allowable primary stress is 13.5 ksi as recommended by

Code Case N-519 with low temperature provision down to as low as 4 K. More elaborate description on the application of the ASME Code is stated in the following section.

2.6.1.1 ASME Code Acceptance Criteria Adopted in Moderator Vessel Design Analysis

As stated in Sect. 2.3.1.3 of this report, the intention is to follow ASME Boiler and Pressure Vessel Code guidelines for the moderator vessel. After some intensive sessions of discussion with ASME Code experts, it was agreed that the design allowable stresses, stress interpretation (using maximum shear criterion), and analysis procedure through finite element analysis will essentially follow Boiler and Pressure Vessel Code Section III, Division I, Class I, Code Case N-519 for aluminum 6061-T6 guidelines. The only exception is that the Code Case N-519 item (k) requires an in-service surveillance program that, for practical reasons, will not be performed. The basis for the exception is threefold for the moderator vessel: (1) the area is not accessible for either placing or removing specimens, (2) if specimens could be placed at the moderator vessel location and kept at the cold temperature of the moderator vessel during irradiation, the cold temperature could not be maintained during the removal and testing activity, and (3) the cold source moderator vessel is contained within the vacuum vessel so that the failure of the moderator vessel is not a catastrophic event.

The moderator vessel is classified in Sect. 3.6 of this report as a Safety Class 3 component (SC-3). The ASME Code Section III, NCA-2134 (d) allows the analysis of SC-3 components by following higher QA class (e.g., SC-1 or SC-2) rules if needed. A list of ASME Boiler and Pressure Vessel Code guidelines that are relevant to this design analysis is shown in the following:

1. The analysis of the moderator vessel essentially follows SC-1 design and analysis rules. SC-1 components may apply the Code Case N-519 for providing allowable aluminum tensile stresses, may use the maximum calculated shear stress in comparison with the allowable shear stress, and may apply NC-3200 for the method of design by analysis.
2. According to Code Case N-519, the allowable primary stress (membrane stress) for 6061-T6 aluminum alloy is 13.5 ksi, and the secondary stress (bending stress) is $13.5 \times 1.5 = 20.25$ ksi. Provisions on low temperature conditions have been outlined in this code case, and these values are good down to about 4 K.
3. In Section III, there is no minimum thickness requirements for the structural components.
4. In Section III, for structures of dimensions less than certain specified length or size, the Code does not require the design to follow the Code requirement. The moderator vessel is a structure of small enough dimension that this would allow the vessel to be exempt from the Code. Nevertheless, the project has taken the position that the vessel will follow the Code guidelines, and the design satisfies all of the above items.

2.6.1.2 Shape, thickness, and dimension of the moderator vessel

The outer surface of the model consists of a hemisphere that is continued by a circular cylindrical surface. Their dimensions are designed to fit the inner surface of a vacuum tube that is dimensioned and enclosed within the HB-4 HFIR beam tube in order to isolate the moderator unit. The inner surface of the moderator consists of a half ellipsoidal shell surface that is continued by a cylindrical surface of elliptical cross section. The outer circular cylindrical surface and the inner cylindrical surface are then cut with a slant angle at top part and at the lower part. The two cylindrical surfaces are joined and welded together. The moderator shape is shown in Fig. 2.5 of this report.

Hydrogen under supercritical condition is designed to flow through the moderator volume where it is used to lower the energy level of neutrons entering the moderator. Hydrogen under supercritical condition requires approximately 15 bars pressure at 20 K. Safety relief for overpressure events is provided by rupture disk set at 19 bar (275.5 psi). Therefore, the moderator needs to be designed for a static pressure of at least 19 bars. Stress distribution as a result of transient pressure surge and uneven temperature distribution should also be considered when setting the maximum design stress of the moderator.

The thicknesses of the moderator at different parts of the surface are different. It is determined iteratively by the finite element solutions so that the maximum membrane and secondary stresses follow approximately within the allowable stresses guidelines that are recommended by the ASME Code.

The preliminary thickness distribution is tabulated in Table 2.20. The thickness of the outer cap is 1 mm and that of the outer cylinder is 2 mm. The thickness ratio is $\frac{1}{2}$. As a result of the basic principles of the shell theory, this thickness ratio will not produce the incompatible deformation and, therefore, the discontinuous stress across the junction of these two surfaces.

Table 2.20. Preliminary moderator vessel thicknesses

Surface	Thickness (in.)
Outer spherical cap	0.0393 (1 mm)
Outer cylindrical body	0.0787 (2 mm)
Inner elliptical cap	0.0394 (1 mm)
Inner cylindrical body	0.1181 (3 mm)
Edges	0.1181 (3 mm)
Patch thickness	0.0787 (2 mm)

The thickness distribution in the actual fabricated model is a slight modification of the values that are listed in the above table. In order to gain more stiffness for both the inner and the outer cap that are only 1 mm thick, the actual fabrication of the moderator vessel allows both caps to be joined smoothly to attach the cylindrical surfaces that are 3 mm and 2 mm thick, respectively.

The thickness variations for the two caps follow the following analytical expressions:

(1) The inner cap thickness t in mm follows:

$$t = 3 - 2 \left(\frac{z - a}{b_2} \right)^2 \quad (24)$$

$$b_2 = 1.051 \text{ in.}$$

$$a = 1.171 \text{ in.}$$

(2) The outer cap thickness t in mm follows:

$$t = 2 - 1 \left(\frac{z}{b_1} \right)^2 \quad (25)$$

$$b_1 = 1.899 \text{ in.}$$

The variable z is expressed in inches and is the distance measured from the center of the outer hemispherical cap along the longitudinal direction of the moderator.

Based on the above formulas, the finite element model for either of the two caps is designed approximately as a series of circular strips. Each strip is assumed to have constant thickness. The thickness for these strips are shown in Table 2.21. The numerical values presented in this table are the realization of the analytical expressions shown earlier.

Table 2.21. Strip sizes used to simulate moderator vessel caps

Strip	Inner cap thickness (in.)	Outer cap thickness (in.)
1	0.1178	0.07858
2	0.1156	0.07720
3	0.1111	0.07449
4	0.1044	0.07039
5	0.0955	0.06492
6	0.0844	0.05811
7	0.0709	0.03937
8	0.0472	N/A

The material used for the moderator vessel is 6061-T6 aluminum alloy. The material constants used are:

- Young's modulus = 10×10^6 psi
- Poisson's ratio = 0.33
- Coefficient of thermal expansion = $13.1 \times 10^{-6}/^\circ\text{F}$.

2.6.1.3 Pressure induced stress and thermal stress

The ABAQUS finite element is used to model the actual fabricated moderator shape. The model has a total of 10,999 nodes and 10,984 elements. Shell element is used in the model. There are three stress output points on each cross-section plane of the shell element. The three ABAQUS output points for the shell structure are top shell surface, middle shell surface, and bottom shell surface. The stress distributions for all three shell surfaces were obtained. With respect to the moderator structure, section point 1 represents the outer surface, section point 2 the middle surface, and section point 3 the inner surface. The operating internal pressure is 15 bars, but the rupture disk pressure of 19 bars is used as the

design pressure. ABAQUS output of Tresca stresses for section points 1, 2, and 3 are plotted in Figs. 2.30, 2.31, and 2.32, respectively. The maximum stress at section point 1 is 22.6 ksi at the patch area and the maximum stress at section point 2 is 9.1 ksi at the lower part of the outer surface.

According to the ASME Code, the membrane stress along the middle surface of the shell should be classified as the "primary stress." The Code Case N-519 gives the aluminum 6061-T6 an allowable primary stress of 13.5 ksi. For the moderator vessel subjected to internal pressure, the primary stress is produced as a result of the sustained internal pressure. The output of the ABAQUS calculation shows that the maximum primary stress is 9.1 ksi, which is less than the allowable of 13.5 ksi with substantial margin.

Both the top surface stress and the bottom surface stress are generated partly by bending deformation of the shell. These stresses, according to ASME Code, are classified as the secondary stress. The allowable secondary stress requirement is $1.5 s_m$ where s_m is the allowable primary stress of 13.5 ksi for the N-519 Code Case. Therefore, the secondary stress has the allowable value of 20 ksi.

Two circular patches of 2 mm thick and 2 in. in diameter each are attached to the inner cylindrical surface at the high bending stress region. These patches produce stiffening effects that not only lower the local bending stress but also reduce bending stress that is away from the patches but is influenced because of the patches. The bending stress at the patch is 22.6 ksi, which is slightly over the ASME Code requirement for the allowable secondary stress of 20 ksi. Owing to our CAD analysis, the stress calculation is believed to be fairly accurate. Also, it is located at the interior of the inner cylindrical

Fig. 2.30. ABAQUS output of Tresca stresses for section point 1, the outer surface.

Fig. 2.31. ABAQUS output of Tresca stresses for section point 2, the middle surface.

surface and not located within the weldment. Furthermore, the handbook value of the yield stress is approximately 35 ksi. Therefore, the higher secondary stress is believed to still have sufficient margin of safety, but efforts will be made to reduce this stress in the detailed design.

As shown in Fig. 2.32, the maximum bending stress along the edge weldment is much higher than the allowable secondary stress of 20 ksi. This junction will be designed and fabricated differently than that represented in the model and will not contain the sharp corners where the high stress loads appear. It is believed that this will substantially lower the calculated stress values, and the stress load in this region is not expected to be a problem. This will be verified with additional analysis once the revised geometry has been established.

The maximum temperature calculated is approximately 40 K versus the average temperature of 20 K. The corresponding thermal stress calculated from ABAQUS is approximately 3 ksi. Therefore, the thermal stress is small, although this value has to be added to the pressure-induced stress. It should also be noted that the thermal stress, according to the ASME Code, is also classified as secondary stress.

In conclusion, the primary membrane stress (maximum) generated by the internal pressure of 19 bars is less than the allowable stress of 13.5 ksi required by the ASME Code. The secondary stress at locations other than the weld region is slightly greater than that required by ASME Code. However, it is considered acceptable as compared with the yield stress of 32 ksi. The thermal stress, that is also

Fig. 2.32. ABAQUS output of Tresca stresses for section point 3, the inner surface.

classified as secondary stress, is approximately 3 ksi. Along the edge of the weldment, the secondary stress is high and the primary membrane stress is low, but with the planned redesign of the weldment region, this is not expected to be a problem.

2.6.1.4 Fracture Probability and Leak-Before-Break Analysis for the Cold Neutron Source Moderator Vessel

Fracture mechanics calculations are made in order to ensure the safety of the moderator vessel against failure by catastrophic fracture. The 6061-T6 aluminum alloy material property is based on the experimental results of K. Farrell and others.^{24,25,26} The fracture toughness value to be used in the fracture analysis is based on an earlier communication with D. Alexander (ORNL). The fracture analysis of the moderator vessel consists of (1) the probability of fracture calculations at the locations of the moderator where either the primary stress or the secondary stress assumes the highest value, (2) the vessel wall leak-before-break analysis by applying an edge crack solution, and (3) the crack penetration calculation as a result of radiation embrittlement by applying the flaw assessment diagram (FAD). The probability of fracture for the vessel is calculated by using a direct probability integration method²⁷ instead of the Monte Carlo simulation method used by the PRAISE Code or by the FAVOR Code developed at ORNL.²⁸ It provides a quantitative estimate on how the vessel is structurally weakened as a result of the

embrittlement of the moderator material after an extended period of irradiation. The leak-before-break analysis leads to the conclusion that the moderator vessel will satisfy this condition, and no catastrophic fracture is possible before leaking first appears. A mass spectrometer will be installed to monitor the leak of hydrogen. The crack penetration calculation is carried out by using the method of FAD. The FAD method is a well established fracture mechanics procedure.²⁹ After a period of neutron irradiation, the result shows that the critical crack depth of the penetrating crack does not change significantly as a result of radiation. The effect of irradiation is mainly manifested by the mode of fracture that shifts gradually from failure by rupture to failure by brittle fracture.

Aluminum alloy 6061-T6 embrittlement data

The 6061-T6 Al embrittlement data was reported earlier by K. Farrell and R. T. King.²⁴ New 6061-T6 Al embrittlement data of 20 years irradiation in the High Flux Beam Reactor (HFBR) was obtained at Brookhaven National Laboratory (BNL). The BNL notch-impact tests are the first such tests for heavily irradiated 6061-T6 alloy. The 6061-T6 aluminum embrittlement data reported by Farrell and King indicate that the effect of neutron irradiation creates damage in 6061-T6 aluminum, which tends to increase the yield stress and to reduce the ductility. These effects were plotted against a range of neutron fluence in that report. It is possible that the reduction in ductility is correlated to the fracture toughness of the material. More often, the loss of ductility implies a decrease in toughness. For the convenience of the present calculation, a modified plot is made and shown in Fig. 2.33 that shows the effect of irradiation to

Fig. 2.33. Aluminum 6061-T6 toughness and 0.2% flow stress curves.

the yield strength increase and the toughness decrease. This plot is the assumed material property for the aluminum alloy to be used in this fracture mechanics analysis.

In Fig. 2.33, for an increase in neutron fluence up to $1.8 \times 10^{27} \text{ n/m}^2$, the yield stress was increased from 40 ksi to 67 ksi. The loss of ductility as reported by Farrell et al. reached a saturated value for an extended period of neutron fluence. Therefore, the fracture toughness is also assumed to reach a saturated value in Fig. 2.33. For the unirradiated 6061-T6 Al, the fracture toughness of $25 \text{ ksi}\sqrt{\text{in.}}$ was obtained by D. Alexander from the Metals and Ceramics Division, ORNL.³⁰ At present, Alexander recommends an estimated value of $15 \text{ ksi}\sqrt{\text{in.}}$ at saturation. The toughness is believed to reach the saturated value as the fluence reaches 10^{26} n/m^2 and remains constant up to $2.0 \times 10^{27} \text{ n/cm}^2$. The reduced toughness of $15 \text{ ksi}\sqrt{\text{in.}}$ is believed to be a rough but conservative estimate. The reduction in toughness may also be reflected by the recent impact energy data^{25,26} from BNL if a correlation between the toughness and the impact energy at fracture can be established.

Acceptance criteria adopted in fracture mechanics analysis

For an acceptable fracture mechanics analysis of the moderator vessel structure, an appropriate safety margin is needed. A consistent and up-to-date recommendation was advanced by the American Petroleum Institute (API) Design Code recently on the method of "Fitness for Service" in which the margin of safety for fracture mechanics calculation was outlined. The cold source project shall adopt the API recommendation and use the following factors of safety:

1. Factor of 1.6 for the design pressure.
2. Factor of 1.4 for the choice of the critical flaw size.
3. Factor of 1.4 for the toughness values.

The 60% increase in the design pressure will raise the stress distribution along the moderator vessel wall uniformly by the same factor, because the finite element result is linear and elastic. A 40% reduction of fracture toughness will be made to meet the API recommendation. However, the nominal recommended toughness value is based on the plane strain toughness value. For the present analysis, the moderator vessel wall has thickness values ranging between 1 mm to 5 mm. It is well known that the plane stress toughness value will be substantially higher than the plane strain toughness value.³¹ As illustrated in Fig. 2.34 obtained from the literature,³¹ the energy release rate G_c for 7075-T6 aluminum alloy increases from 20 to 60 as the specimen thickness decreases from 15 mm to 5 mm for a factor of 3. Since G_c is proportional to the square of K_{Ic} , the K_{Ic} increases a factor of $\sqrt{3} = 1.7$ as the thickness decreases from 15 mm or greater to 5 mm. A combination of the API recommended factor of 1.4 reduction and the increase of plane stress factor of 1.7 leads to an overall increase in toughness given by

$$1.7 K_{Ic} \times \frac{1}{1.4} = 1.2 K_{Ic} \quad (26)$$

Therefore, we will adopt a 20% increase in K_{Ic} value from the recommended nominal value of $15 \text{ ksi}\sqrt{\text{in.}}$ to give an analysis toughness value of $18 \text{ ksi}\sqrt{\text{in.}}$

In applying these factors of safety, it needs to be cautioned that a factor of 60% increase in design load, combined with a decrease of 40% in fracture toughness may provide a condition of double use of the safety factor. A similar case may be found in the ASME Code as a result of the use of both a load factor and a conservative allowable stress. Even though it may lead to an excessive conservatism, in the probability of fracture calculation performed in this study, we still employ all factors of safety as recommended by API Code.

Fig. 2.34. Impact of specimen thickness on toughness (taken from ref. 31).

Both in the Marshall Report³² and in HFIR pressure vessel radiation embrittlement study,²⁸ a flaw density of 0.03 flaws/cubic feet is assumed as a reasonable estimate for pressurized water reactor (PWR) vessels. We shall adopt this density even though in the present study, the moderator vessel is made of aluminum alloy rather than pressure vessel steel. It is the project's opinion that with the available inspection techniques (radiography, dye penetrant, and eddy current) to be applied to the fabrication of the small moderator vessel, the final vessel will have a flaw density that will be at least as good.

The highest surface flaw density for the moderator vessel of 5 mm thickness at the patch area is estimated as

$$0.03 \frac{5 \text{ mm} \times \text{in.}}{12 \text{ in.} \times 25.4 \text{ mm}} = 0.0005 \text{ flaws/ft}^2. \quad (27)$$

In the HFIR study, the surface flaw density is determined to be 0.007, or 14 times larger than that used in the moderator study, because the HFIR vessel wall is 3 in. thick. A total area of square feet underneath HB-3 was assumed in a HFIR study that led to a total of 0.007 flaws for the probability calculation. Likewise, for the present study, one square foot area is also assumed to represent the total vessel surface in the following probability calculation.

Probability of fracture analysis method

The method of probability of fracture analysis developed earlier²⁷ is adapted for the moderator vessel fracture probability calculations. The following information is required for the calculations: (1) the distribution function of the possible cracks in the sample, (2) the fracture toughness of the material at different levels of neutron irradiation, and (3) the regions of maximum stress distribution of the moderator vessel. The crack size is assumed to follow an exponential distribution function (Marshall distribution), and the toughness is assumed to be that shown in Fig. 2.33. The fracture toughness is further assumed to vary according to a Gaussian distribution. The maximum membrane stress is 9.1 ksi located at the lower part of the outer surface, and the maximum bending stress located at the patch area of the inner cylindrical surface is 22.6 ksi for moderator internal pressure of 19 bars. Along the edges of the vessel, the finite element results show much higher stresses, but these edges are being eliminated from the design. Therefore at the edge location the finite element results are not representative of the realistic values.

The Marshall distribution function of crack size x is assumed to have the following form:

$$f(x) = 4.064 \times \exp (-4.064 x) , \quad (28)$$

where x is expressed in inches. The Marshall distribution is developed for the pressure vessel steel. It is not known whether this distribution can appropriately represent the aluminum alloy. However, this is an exponential distribution. A variation of the numerical coefficients in the above distribution is believed to fit the aluminum alloy. It should also be noted that this is only a preliminary calculation and further improvement of the distribution function may be made for a better representation of the aluminum alloy in future analyses. A function $B(x)$ is often assumed as the probability that a crack of size x cannot be detected,

$$B(x) = 0.05 + 0.995 \times \exp (-2.870 x) . \quad (29)$$

The probability of fracture at a deterministic fluence value is an integration of the joint probability of the crack size and the fracture toughness,

$$P(F) = \int_0^{\infty} f(a) B(a) \int_{-\infty}^{K_I} \frac{1}{0.15 M(K_{lc}) \sqrt{2\pi}} \exp \left\{ - \frac{[x - M(K_{lc})]^2}{2[0.15 M(K_{lc})]^2} \right\} dx da , \quad (30)$$

where K_I is the stress intensity factor at the condition of fracture $K_I = \sigma \sqrt{\pi a}$.

The actual probabilistic analysis is performed using the analysis procedure described above with the application of the various factors of safety as described in the preceding section to ensure an appropriate margin of safety.

Probability of fracture calculations for the moderator vessel

The probability of fracture calculation of the moderator vessel is based on the API Code guidelines of the fitness for service. The distribution function for crack density is multiplied by a factor 1.4 to account for the crack size. The fracture toughness is adjusted for the plane stress condition to increase by a factor of 1.7 and decrease by a factor of 1.4, based on the API Code recommendation.

Both maximum membrane stress and maximum bending stress are considered in performing the probabilistic fracture analysis.

Maximum membrane stress is selected by using the contour stress plot for the shell midplane section. The maximum membrane stress is 9.1 ksi located at the outer cylindrical body near pipe inlet for the moderator under 19-bars internal pressure. This location of maximum stress will be used as a reference point for the calculation of the probability of fracture due to membrane stress. A variation of internal pressures from 10 to 25 bars is used to obtain the membrane stresses at this point and an API Code load factor of 1.6 is adopted for the membrane stresses. The results are shown in Fig. 2.35.

A similar probability of fracture calculation is performed using the maximum bending stress of 22.6 ksi at 19-bars internal pressure. It is located on the moderator outer surface at patch area. A variation of internal pressures from 10 bars to 25 bars is used in the probability of fracture calculations. An API Code load factor of 1.6 is also adopted for the applied stress. The results are shown in Fig. 2.36. It should be noted that the bending stress used in the plots is the secondary stress. Although the probability of fracture by secondary stress is higher, it is not as critical as the probability generated by the primary stress.

The numerical values for the probabilities of fracture for the maximum membrane stress and for the maximum bending stress are calculated by using the probability integral. The results are tabulated in Table 2.22 and Table 2.23.

Fig. 2.35. Probability of fracture based on maximum membrane stress.

Fig. 2.36. Probability of fracture based on maximum bending stress.

Table 2.22. Probability of fracture (in log scale) at maximum membrane stress for the moderator vessel under irradiation and internal pressure varied from 10 bars to 25 bars

I.P. (bars)	Radiation embrittlement (10^{25} n/m^2 fluence)									
	2	4	6	8	10	12	14	16	18	20
10	-10.0	-9.7	-9.2	-8.9	-8.5	-8.3	-8.0	-7.9	-7.9	-
15	-8.4	-7.7	-7.2	-6.9	-6.6	-6.4	-6.2	-6.1	-6.1	-
20	-7.0	-6.4	-6.0	-5.7	-5.5	-5.3	-5.2	-5.1	-5.1	-
25	-6.1	-5.5	-5.2	-5.0	-4.9	-4.7	-4.6	-4.5	-4.6	-

Table 2.23. Probability of fracture (in log scale) at maximum bending stress for moderator vessel under irradiation and internal pressure varied from 10 bars to 25 bars

I.P. (bars)	Radiation embrittlement (10^{25} n/m^2 fluence)									
	2	4	6	8	10	12	14	16	18	20
10	-6.1	-5.6	-5.3	-5.0	-4.9	-4.7	-4.6	-4.6	-4.6	-
15	-4.8	-4.5	-4.3	-4.2	-4.1	-4.0	-4.0	-4.0	-4.0	-
20	-4.2	-4.0	-3.9	-3.9	-3.8	-3.8	-3.7	-3.7	-3.7	-
25	-4.0	-3.8	-3.7	-3.7	-3.7	-3.6	-3.6	-3.6	-3.6	-

Penetration fracture of the moderator wall leak-before-break analysis

In order to study the leak-before-break condition for the moderator vessel, a part through crack of variable aspect ratios will be assumed. Cracks of elliptic shape are used for the study. This problem has been analyzed by Neuman and Raju³³ for variable aspect ratios, and numerical results were obtained. The condition that the part through crack will either extend its length or will propagate toward the penetration of the wall relies on the relative magnitudes of the stress intensities generated at the crack-tip region or at the crack-front region.

The plate contains an existing part through-flaw of crack length $2c$ with the aspect ratio $= a/c$ where a is the depth of the crack. The plate has thickness t and ϕ is the angle measured from the major axis of the crack. Uniform stress is applied along the direction perpendicular to the crack plane.

From numerical results obtained by Neuman and Raju, normalized stress intensity factors are provided in Table 2.24.

Table 2.24. Stress intensity factors at $\phi = 0$ and at $\phi = 90$ (leak or break)

a/c	$2\phi/\pi$	a/t			
		0.2	0.4	0.6	0.8
0.2	0	0.617	0.724	0.899	0.190
	1	1.173	1.359	1.642	1.851
0.4	0	0.767	0.896	1.080	1.318
	1	1.138	1.225	1.370	1.447
0.5	0	0.841	0.955	1.126	1.335
	1	1.124	1.185	1.300	1.354

The normalized stress intensity factor is defined by the following equation:

$$K_n = \frac{K_I}{\sqrt{\pi c}} \quad (31)$$

The above numerical values for K_n at crack front ($\phi = 90$) and at crack tip ($\phi = 0$) indicate that for $a/c = 0.2$ through 0.5 and $a/t = 0.2$ through 0.8 ,

$$K_I (\phi = 90) > K_I (\phi = 0) \quad (32)$$

This implies that part-through cracks of the above geometrical dimensions will propagate toward breaking the moderator wall rather than the extension of the crack size along its major axis.

A straight-forward critical crack length calculation is shown in the following. A crack of length $2c$ is assumed to be located along the wall of the moderator vessel. The crack is assumed to be small and the vessel wall is assumed approximately to be a flat plate. Under these conditions, the stress intensity factor K is

$$K = \sigma \sqrt{\pi c} , \quad (33)$$

where σ is the applied stress along the direction perpendicular to the crack plane.

For the moderator vessel design pressure of 19 bars, the maximum membrane stress, section point 2 shown in Fig. 2.30 of the previous section, is approximately 9.1 ksi of primary stress. A safety factor of 60% as described on p. 2-64 is applied to give a value of 14.6 ksi. Recall that a plane stress toughness of $18 \text{ ksi}\sqrt{\text{in}}$ has been regarded as a reasonable toughness value in compliance with the API Code recommendation. It follows, therefore, the critical crack length for a catastrophic extension of the crack can be solved from

$$18.0 = 14.6 \sqrt{\pi c} , \quad (34)$$

for the critical crack length

$$c_b = \left(\frac{18.0}{14.6} \right)^2 \frac{1}{\pi} = 0.48 \text{ in.} \quad (35)$$

By using a very conservative assumption that the maximum primary stress is located at the patch area of the thickest vessel wall, the leak-before-break analysis shown in the previous section indicates that an a/c value of 0.5 still satisfies the leak condition rather than the break condition from $a/t = 0.2$ through as much as 0.8 of the wall thickness. For a/c value of 0.5 with 5-mm vessel wall thickness, the corresponding half-crack length is

$$c_e = \frac{5 \text{ mm} \times \text{in.}}{25.4 \text{ mm} \times 0.5} = 0.39 \text{ in.} \quad (36)$$

The above crack length of 0.39 in. is still a subcritical crack length for breaking, whereas, the vessel already leaks. This is also consistent with the result that $c_b = 0.48 \text{ in.}$ and $c_b > c_e$. As a result, leak of the vessel will begin before break.

Failure assessment diagram analysis of crack penetration fracture

A fracture mechanics analysis procedure, FAD, was developed by the Central Electricity Generation Board (CEGB) in the United Kingdom and by the U.S. Nuclear Regulatory Commission (NRC) for predicting structure fracture.²⁹ The method requires a set of material constants including fracture toughness, yield stress, and elastic constants.

The method is based on the theoretical elastic-plastic crack model of Dugdale³⁴ and its extended version. This model can be expressed by a path to failure curve in a two-dimensional plane with x -axis representing the applied force and y -axis the material toughness. Any point that is enclosed within the curve represents the condition of no fracture. Both CEGB R/6 procedure and the NRC procedure are based on this theory but adapted to realistic conditions. This procedure is used in this analysis to determine the critical crack depth of a part through-crack that will penetrate the moderator vessel wall under the present moderator internal working pressure of 19 bars. The main emphasis is on the effect of radiation embrittlement. The result shows that after a sustained period of irradiation, the critical crack length remains at 0.4 of the wall thickness, but the mode of fracture is gradually shifted from failure by rupture to failure by brittle fracture.

Newman and Raju³³ made an extensive calculation to determine the stress intensity factor K_I for a given surface crack on the surface of a rectangular plate. Their numerical solution will be used here to analyze the elliptic crack front stress intensity. If the crack front stress intensity reaches the K_{Ic} value, then the part through crack will extend and propagate toward the direction to penetrate the moderator wall.

In the Newman and Raju formulation, the K_I value is represented in the following form,

$$K_I = S \sqrt{\frac{\pi a}{Q}} F. \quad (37)$$

It is readily recognized that the quantities Q and F are the scaling factors with respect to a simple crack depth $2a$ under the plane stress condition subject to a tensile stress of S ,

$$K_I = S \sqrt{\pi a}. \quad (38)$$

The factor Q is recommended in the paper as

$$Q = 1 + 1.464 \left(\frac{a}{c} \right)^{1.65} \quad (39)$$

that is a function of the crack depth to crack length ratio a/c . The crack length is $2c$.

For a large crack length, it is reduced to a two-dimensional problem, and, therefore, $Q = 1$ as it should be. The factor F represents the magnification of the stress intensity with respect to the simple crack case $S\sqrt{\pi a}$. An interpolation of the data obtained by Newman and Raju for $a/c = 0.2$ leads to a parametric expression

$$F = 0.144 \left(\frac{a}{t} \right)^2 + 1.015 \left(\frac{a}{t} \right) + 0.956, \quad (40)$$

where t is the thickness of the plate.

The use of this expression has been compared to the numerical results obtained in their paper and the results are shown in Table 2.25.

The parametric representation for the path to failure curve in FAD is

$$K_r = \frac{S\sqrt{\pi t}}{K_{Ic}\sqrt{Q}} \left[\sqrt{\frac{a}{t}} F\left(\frac{a}{t}\right) \right] = K_o \left[\sqrt{\frac{a}{t}} F\left(\frac{a}{t}\right) \right], \quad (41)$$

$$S_r = \frac{S}{\sigma_y} \frac{1}{1 - a/t} = S_o \frac{1}{1 - a/t}, \quad (42)$$

where σ_y is the yield stress of the material.

Table 2.25. Comparison of stress intensity shape factors obtained by Chang using the parametric expression with numerical results obtained by Neuman and Raju

a/t	F (Neuman and Raju)	F (Chang)
0.1		1.059
0.2	1.173	1.165
0.3		1.273
0.4	1.359	1.385
0.5		1.500
0.6	1.642	1.617
0.7		1.737
0.8	1.851	1.860

The two constants K_o and S_o are determined by the specific problem to be analyzed,

$$K_o = \frac{S\sqrt{\pi t}}{K_{Ic}\sqrt{Q}}, \quad (43)$$

$$S_o = \frac{S}{\sigma_y}. \quad (44)$$

It is observed that the path-to-failure curve is proportional to the applied tensile stress S but inversely proportional to K_{Ic} and σ_y . For 6061-T6 aluminum alloy, the embrittlement by radiation tends to decrease the value of the fracture toughness K_{Ic} but increase the value of the flaw stress (or yield stress) σ_y . This implies that the path-to-failure curve is gradually shifted toward a smaller value of S_o . It means that if a surface crack is present on the moderator surface, then an extended period of radiation makes it easier to fracture at the crack front and penetrate the moderator vessel wall by fracture rather by rupture failure.

For the moderator vessel subjected to 19-bars internal pressure, the maximum primary stress of 9.1 ksi is located at the lower part of the outer cylindrical surface. This maximum stress is obtained from the ABAQUS output described in an earlier section.

Other material constants and the moderator vessel dimensions required to determine the two constants K_o and S_o are

$$\begin{aligned} t &= 0.197 \text{ in.} && (5 \text{ mm}) \\ K_{Ic} &= 25 \text{ ksi } \sqrt{\text{in.}} && (\text{unirradiated at room temperature}) \\ K_{Ic} &= 15 \text{ ksi } \sqrt{\text{in.}} && (\text{irradiated at room temperature}) \\ \sigma_y &= 40 \text{ ksi} && (\text{unirradiated at 373 K}) \\ \sigma_y &= 70 \text{ ksi} && (\text{irradiated to } 2.0 \times 10^{27} \text{ n/m}^2 \text{ at room temperature}) \\ Q &= 1 + 1.464 (0.2)^{1.65} = 1.10286 \end{aligned}$$

The thickness of the vessel is assumed to be the value of 5 mm at the patch area although at the point of maximum primary stress, the thickness is only 2 mm. This assumption provides a conservative estimate

because the following calculation is intended to calculate the shift of the failure mode from ductile failure to brittle fracture as a result of irradiation. Higher value of the thickness t gives more embrittlement shift of the failure mode toward brittle fracture and gives less critical crack length. Also at the patch area the primary stress is fairly high with a value of approximately 6 ksi. Therefore, the following calculation provides an upper bound critical crack depth for either the lower part of the outer body or the patch area.

From the above constants, the two constants K_o and S_o are listed in Table 2.26. These values have been derived with the safety factors according to the API guidelines. The applied stress of 9.1 ksi has been multiplied by a factor of 1.6. The toughnesses have been reduced by a factor of 1.4 and the yield stresses by a factor of 2/3.

Table 2.26. Effect of irradiation on F_o and S_o

	Unirradiated	Irradiated
K_o	0.606	1.015
S_o	0.543	0.310

Also, for a/t from 0.1 to 0.8, the parameter values for the path-to-failure curve are shown in Table 2.27.

Table 2.27. Path-to-failure curve data

a/t	$\sqrt{a/t} F(a/t)$	$1/(a - a/t)$
0.1	0.316	1.111
0.2	0.521	1.250
0.3	0.697	1.429
0.4	0.876	1.667
0.5	1.061	2.000
0.6	1.253	2.500
0.7	1.453	3.333
0.8	1.664	5.000

The path-to-failure curves for the unirradiated and fully irradiated moderator vessel at the lower part of the outer body are plotted in Fig. 2.37. The curve of the irradiated moderator is at the left side from that of the unirradiated moderator. For both the unirradiated and the irradiated case, the moderator will fail at approximately 0.4 of the wall thickness. The failure mode shifts gradually from rupture to brittle fracture.

Fig. 2.37. Path-to-failure curves for the unirradiated and fully irradiated moderator vessel.

Conclusion

The probability of fracture is obtained as a measure of the embrittlement condition for the moderator vessel. A direct probability integration method is used in this calculation. The same method, but using Monte Carlo simulation, is often employed in the nuclear industry. The leak-before-break is satisfied by the moderator design structure. This implies that no catastrophic failure may be initiated before leaking begins. A mass spectrometer will be installed to detect the leaked hydrogen. The path-to-failure calculation indicates that, after a period of irradiation, the failure mode of the moderator vessel wall will change from failure by rupture to failure by brittle fracture. The critical crack length remains the same.

2.6.2 Vacuum Tube Structural Analysis

The vacuum tube is designed to resist external pressure of 690 psi. The pressure is assumed to be caused as a result of the failure of the HFIR beam tube. Both static stress and buckling stress are calculated in order to determine the required thickness distribution of the vacuum tube. Out of roundness is also analyzed to show its buckling sensitivity to imperfection of the shape. Several methods of buckling analysis are used, and several choices of the tube thicknesses are assumed. These results are described in the following sections. From these results, the design dimensions have been determined.

2.6.2.1 Shape and Dimension of the Vacuum Tube

A vacuum tube is designed to enclose the moderator capsule, which is located within beam tube HB-4. It extends the entire length of the beam tube. The function of the vacuum tube is to isolate and

protect the moderator capsule from possible hydrogen leakage and accidental collapse of the beam tube. Similar to the design requirement for the moderator, the vacuum tube needs to be designed to contain the minimum possible amount of metal in order to avoid excessive neutron heating. Heat generated within the vacuum tube is to be removed by cooling water that circulates in the space between the beam tube and the vacuum tube.

The end of the vacuum tube that encloses the moderator capsule consists of a hemispherical cap that is extended a short distance up to the edge of the beryllium reflector. It is then enlarged and further extended to a longer distance and is ended at the exterior of the reactor vessel. This end is closed with a screen. The tube remains in vacuum condition.

The vacuum tube design is mainly dictated by the need to provide an appropriate buckling strength when it is subjected to an external pressure of 690 psi. The pressure is chosen as a result of possible beam tube collapse. After several iterative calculations, the preliminary design dimensions have been determined and are listed in the Table 2.28.

Table 2.28. Preliminary vacuum tube design dimensions

	Cap	Short cylinder	Long cylinder
Diameter, in.	4.51	4.51	5.84
Length, in.		14.26	60.00
Thickness, in.	0.10	0.16	0.27
Stress, ksi	7.50	8.10	7.00

The stress values shown in this table have been obtained by the finite element method using the ABAQUS code. The static Von Mises stress distributions of the vacuum tube for the thickness distribution of 0.10, 0.16, and 0.27 are shown in Figs. 2.38, 2.39, and 2.40, respectively. The Tresca stress can be obtained approximately by multiplying the Von Mises stress by a factor of 1.15. The buckling load factor of 2.88 is also obtained for an applied surface pressure of 690 psi.

2.6.2.2 Finite Element Method with Linear Shell Elements

The tube structure consists of a spherical cap that is continued by a short tube of length up to the edge of the reflector. It is then enlarged and extended up to the exterior of the reactor vessel and closed by a screen.

The vacuum tube structure is modeled by 26822 linear shell elements. Uniform pressure of 690 psi is applied on the external surface of the structure. The finite element model does not include the vacuum tube end diaphragm that will be designed separately.

The finite element results are again obtained by applying the ABAQUS code. Three different sets of tube thicknesses along the three sectors of the tube are assumed. Both static stress distributions and buckling load factors (multiple of 690 psi) are calculated. The prescribed boundary condition in the calculations is the fixed constraint at the end of the long tube.

A summary of the static solutions in terms of the Tresca stress at the midsurface of the shell is provided by Table 2.29, and the buckling load factors are provided in Table 2.30.

TIME COMPLETED IN THIS STEP 1.00 TOTAL ACCUMULATED TIME 1.00
 ABAQUS VERSION: 5.5-1 DATE: 16-APR-97 TIME: 14:23:12
 STEP 1 INCREMENT 1

Fig. 2.38. Vacuum tube cap stress distribution.

2.6.2.3 Shell Buckling Formula

It has been recommended in the text book *Cryogenic Systems*, by R. Barron, that the buckling of a long tube and a short tube can be calculated by analytical procedures. A factor of safety equal to four is recommended in buckling design.

Long cylindrical tube, the enlarged portion

The buckling pressure p_c for a long circular cylindrical tube is obtained by applying the formula

$$p_c = \frac{2E (t/D_o)^3}{1 - \nu^2} \quad (45)$$

where

- D_o = outer diameter,
- t = thickness of the tube,
- E = Young's modulus = 10×10^6 psi,
- ν = Poisson's ratio = 0.33.

Fig. 2.39. Vacuum tube short cylinder stress distribution.

Fig. 2.40. Vacuum tube long cylinder stress distribution.

Table 2.29. Summary of static stress solutions for the vacuum tube

Tube thickness (in.)	Cap stress (ksi)	Short cylinder (ksi)	Long cylinder (ksi)
0.10, 0.20, 0.30	8.7	8.1	6.9
0.10, 0.15, 0.25	9.2	9.9	7.7
0.10, 0.16, 0.27	8.6	9.3	8.1

Table 2.30. Buckling load factors for the vacuum tube

Tube thickness (in.)	Assumed applied load (ksi)	Buckling load factor
0.10, 0.20, 0.30	0.69	4.72
0.10, 0.15, 0.25	0.69	2.49
0.10, 0.16, 0.27	0.69	2.88

The condition for a long tube is defined by

$$\frac{L}{D_o} > 1.537 \frac{(1 - \nu^2)^{1/4}}{(t/D_o)^{1/2}}, \quad (46)$$

where

$$\begin{aligned} L &= \text{tube length} = 60.00 \text{ in.} \\ D_o &= \text{outer diameter} = 5.84 \text{ in.} \end{aligned}$$

The factor of safety used for design is equal to four.

Short cylindrical tube

The short tube buckling pressure p_c is obtained by applying the formula

$$P_c = \frac{2.42 E (t/D_o)^{5/2}}{(1 - \nu^2)^{3/4} [L/D_o - 0.45 (t/D_o)^{1/2}]} \quad (47)$$

The short tube is defined by

$$\frac{L}{D_o} < 1.537 \frac{(1 - \nu^2)^{1/4}}{(t/D_o)^{1/2}}, \quad (48)$$

where

$$\begin{aligned} L &= 14.26 \text{ in.} \\ D_o &= 4.51 \text{ in.} \end{aligned}$$

Hemispherical cap buckling

The cap buckling pressure p_c is obtained by applying the formula

$$p_c = \frac{0.50 E (t/R_o)^2}{\sqrt{3} (1 - \nu^2)^{1/2}}, \quad (49)$$

where R_o is the outer radius of the cap and for this problem

$$R_o = 2.236 \text{ in.} \quad (50)$$

Summary of results

The results for several tube thicknesses are summarized in Table 2.31. The results show that with a safety load factor of four, the allowable load is lower than the calculated value, and, thus, some additional refinement of the design will be required.

Table 2.31. Allowable buckling loads for vacuum tube

Buckling for tube	Load factor	Applied allowable load (ksi)
0.125, 0.189, 0.296	4.0	0.690
0.100, 0.160, 0.270	4.0	0.517 (1.33 × 0.517 = 0.640)

For the thickness distribution shown in the second line, the allowable load (0.640 ksi) is still lower than 0.690 ksi, but it is greater than 0.517 ksi. The 0.640 ksi is allowed by the ASME Code because the vacuum tube buckling loads can be classified as an infrequent loading condition.

2.6.2.4 Vacuum Tube Buckling Load Based on ASME B & PV Code

A design procedure for the cylindrical tube is recommended in Section 8, Division 1, ASME BPV Code. Based on this procedure, two sets of vacuum tube thicknesses are assumed and then checked for the maximum allowable buckling pressure p_a .

Long cylindrical tube buckling

The enlarged portion of the vacuum tube is assumed to have the following dimensions:

$$\begin{aligned} L &= 60 \text{ in.} \\ D_o &= 5.842 \text{ in.} \\ t &= 0.296 \text{ in.} \end{aligned}$$

Then,

$$\begin{aligned} \frac{L}{D_o} &= \frac{60}{5.842} = 10.27 \text{ ,} \\ \frac{D_o}{t} &= \frac{5.842}{0.296} = 19.74 \text{ .} \end{aligned} \tag{51}$$

From ASME Code, Section 8, Division 1,

$$\begin{aligned} A &= 0.03 \text{ (Fig. 5-UGO-28.0, ASME, Sect. 8, Div. 1) ,} \\ B &= 9000 \text{ psi (Fig. 5-UNF-28.30, ASME, Sect. 8, Div. 1) ,} \\ p_a &= \frac{4}{3} \frac{9000}{D_o/t} = 607 \text{ psi .} \end{aligned}$$

According to ASME BPV Code, for the infrequent event, the allowable pressure can be increased one-third of its value; that is,

$$p_a = 607 \times \frac{4}{3} = 809 \text{ psi} > 690 \text{ psi} \text{ .} \tag{52}$$

Short cylindrical tube buckling

The vacuum tube dimension in the range from the cap to the enlarged section is

$$\begin{aligned} L &= 14.256 \text{ in.} \\ D_o &= 4.508 \text{ in.} \\ t &= 0.189 \text{ in.} \end{aligned}$$

Then,

$$\begin{aligned} \frac{L}{D_o} &= \frac{14.256}{4.508} = 3.16 \text{ ,} \\ \frac{D_o}{t} &= \frac{4.508}{0.189} = 23.85 \text{ .} \end{aligned} \tag{53}$$

$$\begin{aligned} A &= 0.004 \text{ (Fig. 5-UGO-28.0, ASME, Sect. 8, Div. 1).} \\ B &= 9500 \text{ psi (Fig. 5-UNF-28.30, ASME, Sect. 8, Div. 1).} \end{aligned}$$

From the ASME Code, Section 8, Division 1, Section UF-28(c), the maximum allowable pressure p_a is 531 psi:

$$p_a = \frac{4}{3} \frac{9500}{D_o/t} = 531 \text{ psi} \quad (54)$$

For the infrequent loading case, the allowable can be raised 30% higher; that is,

$$p_a = 531 \times \frac{4}{3} = 707.8 \text{ psi} > 690 \text{ psi} \quad (55)$$

Long tube buckling with an alternative thickness

For this section, the tube thickness is assumed to be 0.270 in. Hence,

$$\begin{aligned} L &= 60 \text{ in.} \\ D_o &= 5.842 \text{ in.} \\ t &= 0.270 \text{ in.} \end{aligned}$$

Then,

$$\begin{aligned} \frac{L}{D_o} &= \frac{60}{5.842} = 10.27 \quad , \\ \frac{D_o}{t} &= \frac{5.842}{0.270} = 21.64 \quad . \end{aligned} \quad (56)$$

From ASME, Section 8, Division 1,

$$\begin{aligned} A &= 0.003 \\ B &= 9000 \\ p_a &= \frac{4}{3} \frac{9000}{D_o/t} = 554.5 \text{ psi.} \end{aligned}$$

For the infrequent loading case, this pressure can be raised to 30% more. Therefore,

$$p_a = 554 \times \frac{4}{3} = 739 \text{ psi} > 690 \text{ psi} \quad (57)$$

Short tube buckling with an alternative thickness

For this section, the tube thickness is 0.16 in. Hence,

$$\begin{aligned} L &= 14.256 \text{ in.} \\ D_o &= 4.508 \text{ in.} \\ t &= 0.160 \text{ in.} \end{aligned}$$

Then, from ASME, Section 8, Division 1,

$$\frac{L}{D_o} = \frac{14.256}{4.508} = 3.16 ,$$

$$\frac{D_o}{t} = \frac{4.508}{0.160} = 28.18 .$$
(58)

By applying these parameters, the following coefficients are obtained:

$$A = 0.0025$$

$$B = 900 \text{ psi}$$

The allowable stress p_a is

$$p_a = \frac{4}{3} \frac{9000}{D_o/t} = 426 \text{ psi} .$$
(59)

For infrequent loading conditions, a 30% increase of the allowable pressure is permitted; that is,

$$p_a = 426 \times \frac{4}{3} = 568 \text{ psi} < 690 \text{ psi} .$$
(60)

The allowable buckling pressure is 568 psi; that is less than the calculated load value of 690 psi. However, this short section is constrained by the cap and by the long tube section of 0.27 in. thickness. Also the finite element solution shows a buckling factor of 2.88. However, this section will be reexamined as part of the detailed design effort.

2.6.2.5 Cylindrical tube out-of-roundness effect

The effect of out-of-roundness for the cylindrical tube has been discussed in *Theory of Elastic Stability*, by S. Timoshenko.³⁵ Out-of-roundness has a detrimental effect on the buckling strength of the vacuum tube. With the surface imperfection w to the radius r ratio of 3%, the buckling strength can be reduced to as much as 50%. For a circular cylinder of radius r and the maximum imperfection w , the maximum compressive stress is

$$\sigma_{\max} = \frac{pr}{t} + 6 \frac{r^2}{t^2} \frac{w}{r} \frac{1}{1 - \frac{p}{p_{cr}}} ,$$
(61)

where p is the external pressure and p_{cr} is the critical pressure for a tube that does not have imperfection,

$$\begin{aligned}
p_{cr} &= \frac{2E}{1 - \nu^2} \left(\frac{t}{2r} \right)^3 \\
&= \frac{2 \times 10^7}{1 - 0.33^2} \left(\frac{t}{2r} \right)^3 \\
&= 2.8 \times 10^6 \left(\frac{t}{r} \right)^3 \text{ psi}
\end{aligned} \tag{62}$$

If we assume σ_{max} to be the yield stress σ_y of 40,000 psi for 6061-T6 aluminum alloy, then a rearrangement of the maximum compressive stress equation gives

$$p^2 - \left[\frac{\sigma_y t}{r} + \left(1 + 6 \frac{w}{r} \frac{r}{t} \right) p_{cr} \right] p + \frac{\sigma_y t}{r} p_{cr} = 0, \tag{63}$$

from which the critical pressure p for the out-of-roundness tube can be solved.

The vacuum tubes with two sets of thicknesses for short tube and long tube will be analyzed for their effects of out-of-roundness. The results are based on the above formula and summarized in the Table 2.32.

Table 2.32. Impact of vacuum tube out-of-roundness

Thickness (in.)	t/r	w/r	p_{cr} (ksi)	p (ksi)
0.25	0.25/2.92 (0.0856)	0.01	1.756	1.14
		0.02		0.89
		0.03		0.74
0.15	0.15/2.25 (0.0667)	0.01	0.831	0.61
		0.02		0.49
		0.03		0.42
0.27	0.27/2.92 (0.0925)	0.01	2.216	1.370
		0.02		0.060
		0.03		0.876
0.16	0.16/2.25 (0.0711)	0.01	1.006	0.719
		0.02		0.575
		0.03		0.484

2.6.2.6 Conclusion

A short tube has a stronger resistance to buckling failure than longer tubes. The hemisphere cap has a stronger resistance to buckling than cylindrical surfaces. The out-of-roundness effect is also critical to buckling failure of the vacuum tube. Out-of-roundness, for example, with the radial deformation of 3% of its radius, may reduce the buckling strength to half. But the vacuum tube is supported by the outer beam

tube that will prevent the vacuum tube from buckling in its worst buckling mode. From these factors, the vacuum tube dimension with thickness distribution 0.1 in., 0.16 in, and 0.27 in. is recommended. This dimension provides a balanced static stress distribution and buckling strength.

2.7 COLD SOURCE TESTING

A number of tests were included in the initial planning of the cold source project (see Test Plan, Sect. 4.7). Some of these tests have already been performed while others are scheduled over the next two years. Two particular planned testing activities have been completed and are documented below.

2.7.1 Full-Scale Hydrogen Loop Tests

A decision to test a representative hydrogen loop was made early in the project. These tests represent a major step in the validation of the HFIR cold source design concept. Arnold Engineering Development Center (AEDC) was selected as the site for these tests for a number of reasons:

1. A large vacuum chamber (42 ft in diameter and 82 ft high designed for space development use) was available capable of housing a simple but full-scale test loop made up of pipework and of realistic size and length and active components identical to (or representative of) those to be used in the final system.
2. A refrigerator system, similar to that ordered for the HFIR cold source, was already piped into the chamber.
3. The chamber was previously approved for the use of hydrogen; thus, operating procedures and safety requirements were already in place.
4. AEDC is situated within three hours of travel from ORNL, making it convenient for project staff.

2.7.1.2 Hydrogen Loop Test Objectives

A hydrogen loop test plan was developed to address three major test objectives:

1. to validate expected operation of the system design and demonstrate its stability under all normal circumstances and anticipated faulted situations that could be practically replicated,
2. to provide benchmarking data for computer models that will be used to evaluate accident scenarios that are impractical to reproduce in real tests, and
3. to evaluate the conceptual control and instrumentation system and provide feedback on the philosophy of operational interlocks and safety shutdown systems.

2.7.1.3 Test Loop Design

The loop was designed by ORNL and built by AEDC staff. Active components were provided by ORNL, who also oversaw the loop construction and built the software-based computer control and instrumentation systems. Some compromises had to be accepted, and the main heat exchanger between the refrigerator and the hydrogen loop was physically larger and heavier than the heat exchanger planned for the HFIR cold source system. Valves and circulators had to be modified to operate entirely within a

vacuum, which added more complexity to the test program. A schematic of the loop configuration is provided in Fig. 2.41, and a photograph of the loop taken from within the chamber is presented in Fig. 2.42. The wall shown in the center of Fig. 2.41 is the Mark I vacuum chamber wall. Those items shown to the left of the wall are located inside the vacuum chamber.

The chamber takes about 8–12 h to evacuate sufficiently to operate a cryogenic system, and added to that is the required cooling time. Accordingly, some on-the-spot decisions had to be made when minor instrumentation or equipment failed to operate correctly, before warming up and making any changes or adjustments. A critical safety strategy set point required for chamber operation was that if the chamber vacuum rose above 0.2 torr, it was assumed to be air, and all power would be switched off. The system therefore was designed to be fail-safe to dump the loop pressure into the internal tanks in the event of power loss. Some of these constraints restricted the choice of some equipment, such as valves, and the final HFIR cold source system is anticipated to function even better than the test loop.

2.7.1.4 Test Description

Four tests were carried out within this program over a period of seven months. Each of the four tests were performed over approximately a three-to-four day period with 24-h testing each day. The time between tests was required to study the data obtained, to identify modifications of the loop for the next test, and to plan the conditions of the next series of tests.

Test 1

The first was carried out in February 1997 when the loop and refrigerator were cooled down together. It proved impossible to reach loop temperatures below about 100 K, which was subsequently attributed to helium refrigerant flow restriction through the long transfer lines and interface heat

Fig. 2.41. Schematic of AEDC hydrogen loop experiment.

Fig. 2.42. Top view of hydrogen test loop inside the AEDC (42' diam) Mark-I vacuum chamber.

exchanger. This had the net effect of reducing available cooling power. Nevertheless, this test demonstrated the test loop capable of functioning as designed, with complete control and freedom from excessive pressure or temperature excursions.

Test 2

Before the next test, the heat exchanger flow-ways were reversed to reduce the helium pressure drop and to raise the cooling capacity of the refrigerator. A remotely-operated bypass valve was also installed across the heat exchanger together with a stop valve to isolate the heat exchanger. The hydrogen loop was evacuated prior to cooling the refrigerator with the objective of cooling the heat exchanger before attempting to cool the hydrogen loop. It proved impossible to reach the operating temperature and even with the heat exchanger isolated and bypassed, the refrigerator was still unable to generate the required helium gas flow because of restrictions in the feed lines. The test was terminated at this point as no new data would have been generated.

Test 3

Prior to the next test, the helium refrigerant pipelines inside the Mark I vacuum chamber were increased in diameter. The bypass and stop valves were removed, and the pipes were re-routed and shortened to reduce the helium restriction as much as possible. The refrigerator and heat exchanger were again cooled with the hydrogen loop under vacuum and reached temperatures in the lower teens Kelvin. The loop was then filled with hydrogen and successfully cooled to 22 K. A shortage of hydrogen gas

prevented lower temperatures from being reached, but some data were obtained by applying an appropriate heat load at higher temperatures.

The final planned test of this series was to transition into a standby mode while maintaining the loop pressure, which proved difficult. At this point, a rupture disc in a remote part of the gas feedline burst and released gas into the Mark I chamber, terminating operations. The true reason for the disc failure has not been resolved. The system was undergoing a planned transient, but there were no indications of a pressure surge anywhere else in the system. At the time, we did not have any pressure sensors in this remote feedline, and, therefore, we did not have local pressure data where the rupture disk burst. It should be noted that because of constraints of the loop configuration, this feedline setup was not prototypic of the planned HFIR system.

Test 4

Following a number of further operational improvements, the system was set up for another series of tests. Additional instrumentation was installed, particularly in the area where the burst of the rupture disk occurred in test series 3, and the gas supply system was improved. The refrigerator and heat exchanger were again cooled, while the hydrogen loop was under vacuum, and testing was started on August 13, 1997. Since this test represents the penultimate of this series, it is described in greater detail in this report.

Procedure. The loop was filled with hydrogen to 1.4 MPa with the low density circulator operating to assist the cooling process. At 50 K, the high density circulator was started with its isolation valve closed. The isolation valve was then opened and the low density circulator isolated and shut down, all of which were smooth and fully controlled operations. The hydrogen feed temperature was reduced to 22 K, and the system was left to stabilize prior to starting the program of specific experiments described in the remainder of this section.

Experiment 1. The objective of the first experiment was to create perturbations in the hydrogen loop to check its ability to retain stability by making appropriate adjustments to the system parameters. The circulator was increased in speed by 10%, and the system made appropriate correction of the pressure while maintaining the temperature. However, the increased voltage demand caused the power supply safety system to shut down the circulator. Though not an anticipated operation, the subsequent flow loss of about 10 sec duration demonstrated that the system was able to sustain and maintain the loop within operating limits under such circumstances. The circulator speed was then reduced by 10% of normal, and a larger than anticipated fall in loop temperature was observed. In view of this, the flow rate was further reduced, and the hydrogen operating temperature fell to 17.3 K. The flow rate at this point was estimated to be about $\frac{1}{2}$ L/s. It is worth noting that this was the original choice of flow rate, but an increase to 1 L/s became necessary to meet the heat transfer requirements of the moderator vessel. It was decided to maintain this flow rate for most of the remaining experiments as it allowed a higher test load to be applied to the loop.

Experiment 2. The objective of the second experiment was to demonstrate the ability of the system to recover from the application of a heat load at the moderator, such as would occur when the reactor is started up. Prior to a HFIR start-up, the cold source system will be cooled to operating temperature and a heater in the helium circuit (controlled by a sensor in the hydrogen loop) will load the refrigerator slightly in excess of the anticipated reactor heat load. When the reactor starts up, the temperature control system compensates for the added heat by removing power from the control heater. In the test loop, the heater is in the hydrogen loop to remove the thermal inertia of the heavy heat exchanger (the final heat exchanger will be light enough to allow the heater to be in the helium loop, for added safety and design convenience). The control heater was accordingly powered manually to 500 W, which was the maximum possible, raising the loop temperature to about 24 K. A load of 200 W was applied to the dummy moderator heater (equal to the true 10% of reactor power applied initially during start-up). The system correctly responded by reducing power to the control heater. The lower flow rate could be expected to

increase the response time, but the loop regained stability within 5 min. The loop is about 60-m long, and the final loop will probably include feedback signals to allow the system to anticipate changes. However, the test demonstrated that full control could be maintained using basic piping and instrument diagram (PID) control algorithms.

Experiment 3. The objective of experiment 3 was to observe the loop response to a further increase in heat load. The remaining 300 W was applied to the dummy moderator and all pressures and temperatures monitored. All remained within the set points, and the system stabilized.

Experiment 4. The objective of experiment 4 was to remove heat from the dummy moderator to simulate a reactor scram. The dummy heat load was instantaneously shut down. Note that in reality this would be represented by step functions (i.e., down to 40% in 1 s, 25% in 1 min, 16% in 3 min, and 7% in 10 min), making a sudden heat removal clearly pessimistic. The system responded by applying heat to the control heater and attaining stability. There was a cycling function that was improved by changing the proportional band function, but the system remained within operational limits throughout the test. More sophisticated control, as indicated earlier, would improve the stability times still further.

Experiment 5. The objective of experiment 5 was to demonstrate an ability to transition from a normal operational state to standby. This is the experiment that involved the bursting of the rupture disk in test series 4. This operation is intended to be carried out with a full heat load of about 2 kW, but since only 500 W applied to the dummy moderator was possible, the effects of the higher load can only be modeled until final testing with the true loop can be carried out. Also, the refrigerator would be placed in standby in the final system, which was not practical for the test. Instead, power was manually applied to a heater remaining in the helium loop, to effectively cancel out the refrigerator. This heater would then be reduced in power by hand to sustain the loop at 100 K. The loop warmed up as required, and the high density circulator remained in operation. At about 50 K the low density circulator was started at half speed, with its isolation valve closed. It was then brought on-line, and the high density circulator was isolated and stopped. The intent was to avoid overloading of the circulator by operating it at too high a density and to increase its speed as the temperature reached 100 K (standby speed). Unfortunately, materials that had cryopumped on to the moderately cold surfaces boiled off, spoiling the vacuum and causing an automatic shutdown of all power supply. Under such circumstances, the control system kicked in as designed and dumped all of the hydrogen, which effectively terminated the test series. The likeliest source of cryopumped material was water from a small but persistent leak into the chamber from the cooling jacket of the low density circulator, which was not designed to operate in a vacuum (note that this pump is not required to operate in a vacuum in the planned HFIR system). However, at the time of the failure, the system was very close to the planned 100 K standby condition. The high density circulator line had been isolated, and the low density circulator line had been opened, resulting in all flow being controlled by the low density circulator. The system was on a steady temperature rise profile and had reached a temperature at some points in the system of 70 K. Data taken at 1 s intervals throughout the test were recorded and will be used to compare with model analysis.

2.7.1.5 Conclusions

Each test series performed provided new information and insights into the operation of the hydrogen loop. Although there were some disappointments in not fulfilling all the goals of the test plan, the objectives of the tests, as identified in Sect. 2.7.1.2, were accomplished. The viability of the system was clearly demonstrated; a substantial data base for validation of the analytical models was obtained; and a good evaluation of the control system was performed. It is believed that unless the model evaluations of the tests uncover unexpected results, the usefulness of this test loop is now exhausted, and the next full loop tests will be conducted at the HFIR site using the prototype system before its installation into the reactor. Most of the equipment will be in its final position, but the transfer line will be rerouted so as not to enter the reactor building. A true moderator vessel will be added together with a dummy heat load to allow realistic tests to be carried out together with personnel training. Some additional tests are, however,

expected to be performed at AEDC with a smaller chamber. The objective of these tests will be to flow supercritical hydrogen at 18 K through a heated true moderator vessel in order to confirm the heat transfer calculation. Vessel surfaces will be monitored for any hot spots indicated by the model and for simulated normal and standby operating conditions.

2.7.2 Mark I Moderator Vessel Pressure Tests

As part of the development of the fabrication of the moderator vessel, a full-scale vessel was fabricated and tested to 1.5 times the operating pressure. The vessel initially performed very poorly under the test conditions. Subsequent to the initial test an aluminum welding expert from the laboratory met with the fabricator, and a means of improving the welds was identified. Following rewelding of the vessel by the fabricator, it passed the pressure test, but failed a later helium leak test. The leak occurred at one of the edge welds on the vessel. The test performance of this early vessel was actually better than expected and provided high confidence that, with some modifications, the fabrication of the vessel will not be a problem.

As a result of the data obtained from these tests, the fabricator has already proposed an alternative fabrication process that would eliminate the need for edge welds and could be assembled with only one circular weld. The revised drawings will be electronically transferred to ORNL for design appraisal and any changes returned to the vendor until an optimal design is agreed upon. This Mark II (and hopefully final) moderator vessel is expected to be fabricated and tested next year.

2.8 REFERENCES FOR CHAPTER TWO

1. *CFX 4.1 Flow Solver User Guide*, Computational Fluid Dynamics Services, Harwell Laboratory, Oxfordshire, United Kingdom, October 1995.
2. "The Development of Special Transfer Lines for the ISIS Project," International Cryogenic Engineering Conference, Massachusetts Institute of Technology, Boston, September 1985.
3. J. F. Breisemeister, *MCNP - A General Monte Carlo N-Particle Transport, version 4a*, LA-12625-M, Los Alamos National Laboratory, Los Alamos, New Mexico, November 1993.
4. R. E. MacFarlane, *New Thermal Neutron Scattering Files for ENDF/V-VI Release 2*, LA-12639-MS, Los Alamos National Laboratory, Los Alamos, New Mexico, March 1994.
5. J. A. Bucholz, *Initial Global 2-D Shielding Analysis For the Advanced Neutron Source Core and Reflector*, ORNL/TM-12672, Lockheed Martin Energy Systems, Inc., Oak Ridge Natl. Lab., August 1995.
6. Y. Farawila, U. Gohar, and C. Maynard, *KAOS/LIV-V: A Library of Nuclear Response Functions Generated by KAOS-V Code from ENDF/B-V and Other Data Files*, ANL/FPP/TM-241, Argonne National Laboratory, Argonne, Illinois, April 1989.
7. P. Ageron, "Neutronic Design of the ILL Cold Sources: An Historical Perspective," *Proceedings International Workshop on Cold Neutron Sources*, 1-18, LA-12146-C, cosponsored by Los Alamos National Laboratory Manuel Lujan Jr. Neutron Scattering Center (LANSCE) and Oak Ridge National Laboratory Advanced Neutron Source (ANS), August 1991.

8. D. L. Selby et al., *High Flux Isotope Reactor Cold Source Preconceptual Design Study Report*, ORNL/TM-13136, Lockheed Martin Energy Systems, Inc., Oak Ridge Natl. Lab., December 1995.
9. R. E. Williams, J. M. Rowe, and M. Blau, "Benchmark of the Nuclear Heat Deposition in the NIST Liquid Hydrogen Cold Source," Ninth International Symposium on Reactor Dosimetry, Prague, Czech Republic, September 1996.
10. J. P. van Doormaal and G. D. Raithby, "Enhancements of the SIMPLE Method for Predicting Incompressible Fluid Flows," *Numerical Heat Transfer* 7, 147-163 (1984).
11. B. E. Launder and D. B. Spaulding, "The Numerical Computation of Turbulent Flows," *Computer Methods in Applied Mechanics and Engineering* 3, 269-289 (1974).
12. C. L. V. Jayatillaka, "The Influence of Prandtl Number and Surface Roughness on the Resistance of the Laminar Sublayer to Momentum and Heat Transfer," *Progress in Heat and Mass Transfer* 1, 193 (1969).
13. M. F. Marchbanks, *Aluminum Alloys Properties Databook*, Savannah River Laboratory, May 1988.
14. U.S. Dept. of Defense, *Metallic Materials and Elements for Aerospace Vehicle Structures*, MIL-HDBK-5E, June 1, 1987.
15. *Structural Alloys Handbook*, Vol. 3, Battelle's Columbus Division, Metals and Ceramics Information Center, 1986 edition.
16. Y. S. Touloukian et al., *Thermophysical Properties of Matter, Vol. I: Thermal Conductivity of Metallic Elements*, IFI/Plenum, New York, 1970.
17. K. Farrell, Oak Ridge Natl. Lab, personal communication to D. L. Selby, Oak Ridge Natl. Lab., July 30, 1997.
18. W. Frost, *Heat Transfer at Low Temperatures*, Plenum Press, New York, 1975.
19. D. G. Friend, *NIST Standard Reference Database 12, NIST Thermophysical Properties of Pure Fluids, Version 3.1*, National Institute of Standards and Technology, Gaithersburg, Maryland, 1992.
20. *Operating Manual for the High Flux Isotope Reactor*, 5-15, ORNL/TM-1138/R1, Martin Marietta Energy Systems, Inc., Oak Ridge Natl. Lab., September 1982.
21. ATHENA [see RELAP5/Mod3 Code Manual, Volumes 1-7 Inclusive, The RELAP5 Code Development Team, NUREG/CR-5535, INEL-95/0174 (formerly EGG-2596), June 1995.]
22. I-DEAS Master Series, Release 4.0, Structure Dynamics Research Corporation, Milford, Ohio, 1997.
23. ABAQUS 5.6, Hibbit, Karlsson, and Sorensen, Inc., Pawtucket, Rhode Island, 1997.

24. K. Farrell and R. T. King, "Tensile Properties of Neutron-Irradiated 6061-T6 Aluminum Alloy," *Effects of Radiation on Structural Materials*, 440-449, ASTM STP 683, eds. J. A. Sprague and D. Krammer, American Society for Testing and Materials, 1979.
25. J. R. Weeks, C. J. Czajkowski, and P. R. Tichler, "Effects of High Thermal and High Fast Fluences on the Mechanical Properties of Type 6061 Aluminum on the HFBR," *Effects of Radiation on Materials*, 441-452, ASTM STP 1046, eds. N. H. Packan, R. E. Stoller, and A. S. Kumar, American Society for Testing and Materials, Philadelphia, 1990.
26. C. J. Czajkowski et al., *Tensile and Impact Testing of an HFBR Control Rod Follower*, BNL Report, Nuclear Waste and Material Technology Division, Brookhaven National Laboratory, Upton, New York, August 1989.
27. S. J. Chang, "Probability of Fracture and Life Extension Estimate of the High Flux Isotope Reactor Vessel," *ASME J. Pressure Vessel Technology* **120**, 1998.
28. R. D. Cheverton, J. G. Merkle, and R. K. Nanstad, *Evaluation of HFIR Pressure Vessel Integrity Considering Radiation Embrittlement*, ORNL/TM-1044, Martin Marietta Energy Systems, Inc., Oak Ridge Natl. Lab., 1988.
29. M. F. Kanninen and C. H. Popelar, *Advanced Fracture Mechanics*, Oxford University Press, New York, 1985.
30. D. Alexander, Oak Ridge Natl. Lab., personal communication to D. L. Selby, Oak Ridge Natl. Lab., April 10, 1992.
31. J. F. Knott, *Fundamentals of Fracture Mechanics*, p. 116, John Wiley and Sons, New York, 1973.
32. W. Marshall, "An Assessment of the Integrity of PWR Pressure Vessels," Second Report, *United Kingdom Atomic Energy Authority*, March 1982.
33. J. C. Newman and I. S. Raju, "Analysis of Surface Cracks in Finite Plates Under Tension or Bending Loads," NASA Technical Paper 1578, National Aeronautics and Space Administration, 1979.
34. D. S. Dugdale, "Yield of Steel Sheets Containing Slits," *J. of the Mechanics and Physics of Solids* **8**, 100, 1960.
35. S. Timoshenko and S. Gere, *Theory of Elastic Stability*, McGraw-Hill, New York, 1960.

3. HFIR COLD SOURCE SAFETY ANALYSIS

3.1. SAFETY DESIGN PHILOSOPHY AND PRINCIPAL SAFETY FEATURES

The primary purpose of the cold source safety analysis is to ensure that the reactor safety systems and the overall safety level of the reactor are not impacted by the cold source or any of its systems. A secondary purpose of the cold source safety analysis is to ensure that on-site personnel and equipment also are not adversely impacted by the cold source or any of its systems. The safety goal for effects on the reactor is to have a low probability ($<1 \times 10^{-6}$ per year best estimate frequency) that fuel will be damaged or a nuclear safety function lost because of the cold source or any of its equipment. The safety goal for the cold source facility is that a major release of hydrogen into the reactor building should be a low probability limiting event, with no loss of the confinement safety function.

The cold source project design work is proceeding in parallel with development of the cold source safety analysis. At each stage of the design and testing, the safety analysis is providing input to the decision-making process and developing input for the final safety analysis for the facility. The final safety analysis of the facility will be used as an authorization basis document to support the safety review by ORNL safety committees and by DOE. The cold source final safety analysis report (FSAR) is expected to be an attachment to an unreviewed safety question determination document that is submitted to add the cold source facility to the HFIR authorization basis. The appropriate amount of summary descriptive and safety information is expected to be inserted into Chap. 10 of the HFIR SAR following initial operation of the cold source. The cold source safety analysis will then be maintained over the life of the facility.

The safety of the cold source is being ensured by multiple approaches. Adequate conservatism is being given to the design of the liquid hydrogen boundaries to ensure a high integrity boundary. A quality assurance (QA) program is in place to control design, analysis, procurement, fabrication, installation, and operation of the facility. Development testing is being carried out as part of the design, and final testing will be performed as part of the start-up of the facility. An accident analysis is being performed to examine the bounding aspects of the design, as well as the probability and consequences of various event sequences.

3.1.1 Safety Design Philosophy

The cold source safety design philosophy is a defense-in-depth approach that provides several means to avoid the accidental contact between hydrogen and air and also provides the means to mitigate a hydrogen release, given that a release is assumed to occur. The principles of conservatism, simplicity, redundancy, fail-safe design, and passive safety features are included in the design as much as possible.

3.1.1.1 Containment by Multiple Barriers

Under normal operating conditions, the hydrogen will be contained by at least two barriers, with a monitored vacuum surrounding the inner barrier. For cryogenic portions of the hydrogen loop, an inert gas layer (helium) blankets the vacuum boundaries.

3.1.1.2 Use of High Quality Materials

Hydrogen pressure boundary materials and the gases used in the cold source system will be of a very high quality level, consistent with the American Society of Mechanical Engineers (ASME) Code and piping standards that govern the pressure boundary design.

3.1.1.3 Redundancy of Key Process Safety Components

Cold source moderator volume cooling, hydrogen venting, and reactor safety system interfaces will be achieved using redundant components. The single failure criterion will be applied to the design and analysis of these three cold source safety functions.

3.1.1.4 Fail-Safe Design Features

Components of the cold source system will be designed to fail safe with respect to their process control or safety function upon loss of support system power or gas supply.

3.1.1.5 Passive Safety Features

The cold source design will use passive safety features where possible to achieve a safety objective. The strength requirement on the vacuum tube to withstand beam tube failure, the use of rupture disks for over-pressure protection, and conduction/radiation cooling of the moderator volume upon loss of all active cooling are three examples of the use of passive safety features.

3.1.1.6 Provision for Mitigation of Hydrogen Release Events

The cold source design will provide for the mitigation of hydrogen release events. Within the reactor building, a vent system will route hydrogen released from the primary hydrogen pressure boundary out of the reactor building. A safe room will surround the key cold source active components.

3.1.2 Summary of Principal Safety Features

This section provides a brief description of the key safety features of the cold source design. The impact of these features and their interactions under normal and abnormal conditions are discussed in greater detail in the remaining sections of this chapter.

3.1.2.1 Containment

The hydrogen is contained within a high integrity boundary that is vacuum insulated. The vacuum level and residual gas concentration are monitored. For cryogenic portions of the system, the vacuum is surrounded by another boundary, which is inert gas (helium) blanketed. The inert gas (helium) pressure is monitored.

3.1.2.2 Moderator Volume Cooling Systems

Three modes of cooling the moderator volume are provided by the design. Under normal operating conditions with high-density (HD) supercritical fluid, one of two high-density circulators provides cooling. Under standby operation with low-density (LD) supercritical fluid, a single low-density circulator is provided. Under emergency (loss of all forced cooling) conditions, the capability to flood the inside of the vacuum tube with helium and conductively/radiatively cool the moderator volume is provided. Preliminary plans are to drive the three hydrogen circulators with normal ac power with backup electrical power provided by batteries to any one of the three.

The loop heat sink is the normally-refrigerated, helium-cooled heat exchanger for normal and standby operation. As a backup heat sink, only the liquid-nitrogen-cooled portion of the refrigerator will be used to cool the helium used in the heat exchanger. The emergency cooling heat sink is the water-cooled vacuum tube.

3.1.2.3 Over-Pressure Protection

Cold source loop over-pressure protection is provided by rupture disks and relief valves.

3.1.2.4 Hydrogen Expansion Tank

For standby operation, a large tank is provided for expansion of the approximately 5 kg of supercritical hydrogen to ambient temperature, 4-bar conditions.

3.1.2.5 Hydrogen Vent Systems

Hydrogen vent systems are provided to route controlled quantities of released hydrogen safely to a central stack. The vent will be designed with a continuous, inert gas purge.

3.1.2.6 Hydrogen Safe Room

A hydrogen safe room is provided outside the reactor building for housing the active components of the cold source system. The safe room is designed to minimize sources of ignition, provide proper ventilation during normal operation, ensure system cleanliness during maintenance, and provide controlled venting during hydrogen release events.

3.1.2.7 Process Control System

A computer-integrated process control system will control the cold source loop temperature, pressure, pump speed, and various valve positions. A set of operating limits and possibly technical specifications will be developed based on the design and safety analysis work.

3.1.2.8 Reactor Scram or Power Setback

A number of criteria have been identified at this stage of the safety analysis as necessary cold-source-related reactor scram criteria. This list of scram criteria is based on the importance of cold source boundary integrity and the forced-flow safety function. Three categories of cold-source-related scrams have been identified as described below:

1. Any detected failure of beam tube wall or vacuum tube wall.
 - A. Beam tube failure as detected by water-helium gap pressure. The gap is protected by a relief valve, which will open and relieve up to 100 gpm to the pool. A gap pressure caused by a small leak that is sufficient to open the relief valve should indicate a beam tube leak and be subject to a *manual scram*. A sustained gap pressure above the relief valve set point indicates a large leak and should cause an *automatic scram*.
 - B. Vacuum tube failure detected by a pressure switch (vacuum level) and residual gas analyzer (RGA) instrumentation. Degraded vacuum conditions should generate either a switch to standby conditions and/or *manual scram*, depending on the level of vacuum loss. Complete loss of vacuum as indicated by a pressure switch should cause an *automatic scram*.

2. Loss of forced-flow cooling of the moderator volume.
 - A. Complete loss of circulation (loss of both high-density circulators and the low-density circulator) measured by loop delta-pressure, should cause an *automatic scram*.
 - B. Complete loss of heat sink (loss of both normal and nitrogen precool refrigeration) should generate a *manual scram*, based on hydrogen temperature.
 - C. Loss of inventory, as indicated by a pressure switch (vacuum level), should cause an *automatic scram*.
3. Cold source fire alarm should generate a *manual scram*.

3.2 HYDROGEN-RELATED HAZARDS ANALYSIS

Much of the HFIR HB-4 cold source design has been motivated by the desire to prevent hydrogen from reacting with oxygen either in the HFIR confinement atmosphere or inside the beam tube. Features of the cold source design addressing this goal have been previously discussed in Chap. 2 of this report. Since the publication of the *HFIR Cold Source Preconceptual Design Study Report* (ORNL/TM-13136),¹ several preliminary analyses have been performed to address hydrogen hazards associated with the cold source. These evaluations are the subject of this section.

Section 3.2.1 investigates the pressure/temperature atmospheric response to a H_2/O_2 reaction within the safe room located inside the HFIR confinement. Section 3.2.2 presents an analysis to determine the peak detonation pressure pulses inside the HB-4 beam tube employing a bounding (maximum) estimate of the quantity of O_2 that can be envisioned within the beam tube. Section 3.2.3 presents the results of an evaluation of cryopumping of air onto the exposed LH_2 transfer lines inside the cold source vacuum tube. Section 3.2.4 identifies potential areas of future work involving analysis of H_2 hazards.

3.2.1 In-Confinement Safe Room Response To H_2/O_2 Burns

The H_2 safe room of the cold source was initially planned to be housed inside the HFIR confinement building in the southeast side of the experiment room. The safe room houses the pump module (which in turn contains the two high-density circulators, the low-density circulator, the pressurizer, and most of the parameter sensors), the heat exchanger module, parts of the gas-handling system, and one of the vacuum pumping stations. All electrical apparatus within the safe room is either rendered intrinsically safe, by electronic safety barriers that limit possible power discharges to a maximum of 0.019 mJ, or is contained within the inert blanket. Personnel will wear grounding devices and nonstatic overgarments within the room. The room is constantly force-ventilated to the vent stack at a rate of 1/4 air change/min, which is raised to 1 air change/min if 1% hydrogen is detected. The refrigeration system is located outside the HFIR confinement, adjacent to the safe room, and is coupled by a transfer line to the secondary side of the cold source heat exchanger.

Despite the precautions evident in the design of the safe room, early analysis efforts were directed toward investigation of the potential consequences of significant hydrogen leaks into the safe room atmosphere and subsequent hydrogen/air reactions.

3.2.1.1 MELCOR Input And Modeling Description

The pressure/temperature response of an in-building safe room to a hydrogen leak was calculated by the MELCOR² computer code. A three-node model (see Fig. 3.1) was developed for this small room

Fig. 3.1. MELCOR nodalization used to examine the viability of an in-confinement safe room.

(2400 ft³ = 68 m³) and incorporated into the overall MELCOR model of the HFIR confinement. The two nodes surrounding the third node represent the suction and exhaust regions of the safe room that interface with the exhaust fans. These two nodes have a combined volume of 34 m³. The third node also has 34 m³ of volume and represents the atmospheric volume immediately surrounding the equipment in the safe room housing the H₂. It is this node that receives the H₂ when the H₂ leak is assumed to occur. A flow path connection between the safe room and the remainder of the experiment room was added to model a blowout panel. Flow paths connected to the outside atmosphere were added to represent normal and emergency fan suction and exhaust paths from/to the outside. The entire hydrogen inventory of this preconceptual design of the cold source system (350 g) was added to the atmosphere over a time span of 120 s to simulate a leak in the H₂ bearing equipment.

The H₂/O₂ reaction (deflagration) in air is modeled by the BURN package of MELCOR as a thermodynamic equilibrium process. The duration of the reaction for each node is controlled by the size (volume) of the node and the flame speed. Flame speed is calculated by a correlation that depends on hydrogen concentration and diluent (H₂O) concentration. Flames are permitted to propagate from one node to another either horizontally or vertically through the flow paths connecting the nodes. Assuming continuous availability of ignition sources, H₂/O₂ reactions are initiated when the H₂ concentration in an O₂ bearing atmosphere reaches a threshold of 4% by volume and are considered complete when the maximum amount of H₂ has been consumed, given the available inventory of O₂. Inerting effects are accounted for by allowing the reaction to occur only when the mole fraction of diluent (H₂O) is less than 55%.

The pressure mitigation effect of a blowout panel between the proposed in-confinement safe room and the experiment room was modeled. This flow path (area = 1.0 m^2) was assumed to be normally shut and was allowed to be opened only when the pressure of the safe room exceeded $1.37 \times 10^5 \text{ Pa}$ (5 psig). In addition, normal and emergency condition ventilation fans were represented. The suction of these fans was assumed to be from the reactor building outside atmosphere and not the building atmosphere. This precaution maintained the important function provided by the special building hot exhaust (SBHE) system of fission product filtering of all of the confinement atmosphere before release to the outside atmosphere through the plant stack. The function of the safe room emergency fan was to rapidly dilute and remove leaked H_2 and transport it to the outside atmosphere.

The normal safe room ventilation fan was assumed to produce a gas flow of 1/4 room change/min [e.g., $0.28 \text{ m}^3/\text{s}$ (600 CFM)]. The emergency fan was assumed to start when the H_2 concentration of the safe room exceeded 1% mol fraction of H_2 and assumed a flow of 1 room change/min [e.g., $1.13 \text{ m}^3/\text{s}$ (2400 CFM)]. If during the H_2/O_2 reaction, the blowout panel opened, the emergency (and normal operation) fan was shut down in order to preclude drawing HFIR confinement atmosphere through the safe room exhaust system and bypassing the fission product filters of the SBHE system.

3.2.1.2 Pressure And Temperature Results

Figure 3.2 presents the calculated atmosphere temperatures of the safe room, as well as the experiment room and the SBHE inlet duct. The calculated pressure responses in these regions are presented in Fig. 3.3. Injection of hydrogen gas at 21 K at a rate of 2.9 g/s was begun coincident with the hydrogen leak at 60 s. A slight cooling effect of the safe room atmosphere is seen in Fig. 3.2 as the cold H_2 gas mixes with the much warmer ambient temperature gas.

Fig. 3.2. MELCOR calculated temperature response to H_2/O_2 reaction in the in-building safe room concept.

Fig. 3.3. MELCOR calculated pressure response to H_2/O_2 reaction in the in-building safe room concept.

By about 80 s, the hydrogen concentration has reached the 1% level, and the emergency fan begins operation. The effect of this fan is to increase the air flow through the safe room, to dilute the H_2 concentration in the safe room atmosphere, and to remove a portion of the extant hydrogen prior to it reacting with oxygen. It also results in a drop in the safe room pressure (see Fig. 3.3) as the atmosphere in the safe room is exhausted to the outside.

By about 135 s, the hydrogen concentration has reached 4%, and a burn begins in the safe room (see Figs. 3.2 and 3.3). It is clear that the emergency fan is insufficient to maintain the hydrogen concentration below the ignition threshold. The safe room pressure increases to about 19 psi, the blowout panel opens, and the safe room fans shut down. (Again, the purpose for deactivation of the safe room fans is to preclude bypass of the SBHE filters for the reactor building atmosphere.) The opening of the blowout panel allows hot safe room atmosphere to enter the experiment room, to mix with the ambient temperature gas, and to flow into the SBHE system. The increase in the average experiment room temperature (Fig. 3.2) is estimated to be about 11 K (20°F).

Atmosphere temperatures (Fig. 3.2) remain above 500 K (440°F) for much of the safe room after 135 s because of continuous H_2/O_2 reaction. Burning is prolonged because of the continuous H_2 injection from the leak source and the relatively large amount of O_2 in the room. Indeed, for the innermost portion of the safe room (CV-430), the atmosphere temperature surpasses 1000 K (1340°F).

By about 170 s, the SBHE system has established flow through the safe room, and outside air is permitted to enter continuously, pass through the safe room, and exit the safe room through the relieved blowout panel. The H_2/O_2 reaction ceases at this time because of completion of the burn started at 135 s. Between 170 and 180 s, H_2 continues to leak into the safe room, but a burn does not start because of the fact that the H_2 mole fraction is once again below 4%. The cold gaseous H_2 combines with the very hot safe room atmosphere, and the atmosphere begins to cool. After 180 s, the H_2 leak stops, and the hot safe room atmosphere continues to cool by mixing with the cool air entering the safe room from the outside.

environment and by thermal radiation and convection from the hot gas to the cold structures inside the safe room.

3.2.1.3 Conclusions Of In-Confinement Safe Room Analysis

The results of the MELCOR analysis indicates two very important features of the in-confinement concept. First, an H_2/O_2 reaction was not prevented despite operation of the emergency safe room exhaust fan. The current study employed a fan removing one volume change per minute. Either a much larger fan or a much smaller H_2 leak must exist in order to preclude H_2/O_2 reactions inside the safe room via dilution and mixing with extant air. Second, an H_2/O_2 reaction inside the in-building safe room is very likely to generate high temperatures (>1000 K) inside the small in-building room and to generate sufficient pressure that hot safe room atmosphere is expelled into the reactor confinement atmosphere.

These results combined with the very limited space available inside the reactor building resulted in the decision to relocate the cold source safe room outside the reactor building.

3.2.2 Preliminary Beam Tube H_2 Detonation Analysis

The HFIR cold source moderator vessel is doubly contained within the vacuum tube and the HB-4 beam tube very close to the reactor core. In addition, the cold source moderator vessel walls are the weakest structures in the cold source system. The safety margin between the wall failure stresses and the normal operating stresses within the moderator vessel walls is relatively small. Therefore, these walls are perhaps vulnerable to failure because of a combination of internal pressurization and loss of strength at elevated temperatures, such as would occur during loss of moderator vessel wall cooling.

The beam tube wall represents a portion of the primary pressure boundary of the reactor coolant system and thus serves to protect and isolate the reactor coolant and core structures from mishaps inside the beam tube. If a failure of the moderator vessel wall occurs inside the vacuum tube, and if the vacuum tube were to fail, there is a potential for subsequent internal pressurization loading of the beam tube wall. Therefore, an initial preliminary analysis of the peak detonation pressure loading of the beam tube wall has been performed to assess the over-pressurization potential.

3.2.2.1 Worst Case Beam Tube Detonation Event Selection and Description

In order for an H_2/O_2 reaction to occur inside the vacuum tube vacuum region, three separate structural failures must occur in the cold source system. First the neutron window of the beam tube extension must fail and permit beam room air at 1 atm pressure to flow into the rough vacuum portion of the beam tube extension adjacent to the reactor pool. Such an occurrence would be detected by the degradation of vacuum and would result in the transition of loop operation into standby (100 K) mode, which would greatly degrade the intensity of the cold neutron beam. Based on a previous study,³ the frequency of failure of this neutron window is estimated at 0.0084/year.

A second failure (failure of the double-walled helium-filled neutron window of the vacuum tube) is required in order to allow air to contact H_2 inside the innermost portion of the beam tube. The failure frequency of this double window is estimated at less than the 0.0084/year value from ref. 3 because of the use of support structures between the two windows. Such a failure would be detected by an RGA system sampling the vacuum tube vacuum and might result in a transition in loop operation to a low-pressure, high-temperature (100 K) mode.

The result of these two gross neutron window failures (single-wall beam tube neutron window and double-wall vacuum tube window) places beam room air at 1 atm in the vacuum space of the vacuum tube in direct contact with the cold exposed liquid hydrogen (LH_2) structures. Such structures consist of the moderator vessel walls and the bare LH_2 transfer tubing transporting LH_2 into and out of the moderator vessel. These structures are very cold (17–20 K) at normal operating conditions and would act

to cool the entering beam room air. They are so cold that they would initially condense and freeze gaseous O₂ and N₂ entering from the beam room.

The third simultaneous structural failure required in order for H₂ and O₂ to potentially mix inside the cold source vacuum tube is a failure of the moderator vessel wall. As mentioned before, this structure has a relatively low margin of safety with respect to wall stresses. The estimated frequency of failure of such a thin-walled structure is 1.3×10^{-4} /year from Sect. 3.3. The result of such a wall failure would be to allow H₂ to flow directly into the vacuum tube vacuum space. Such a failure would be detected by the cold source inventory/pressure control system as well as by the detection of H₂ in the vacuum tube vacuum space and would result in the transition of the loop into a low-pressure high-temperature (100 K) mode of operation. As currently envisioned, such an occurrence would also result in a reactor shutdown to preclude a potential threat to the vacuum tube and to the HB-4 beam tube.

Mechanical features of the cold source system design would act to mitigate the effects of such an H₂ leak from the moderator vessel. The first feature is the continued removal of gas from the vacuum tube vacuum space by the associated vacuum pump and the exhaust of such gas into the hardened H₂ ventilation duct. If the moderator vessel failure were small such that the vacuum pump could remove the hydrogen as it leaked into the vacuum space, hydrogen concentrations inside the vacuum tube might be limited such that detonable mixtures might not occur. Second, a relief valve (or rupture disk) is planned for the vacuum tube vacuum space so that any high-pressure gas extant inside the vacuum tube would be vented directly into the hardened H₂ ventilation duct. Administrative actions to shut down the reactor and to operate the cold source in a low-pressure high-temperature (>100 K) mode would additionally act to minimize the inventory of hydrogen lost to the vacuum space and to preclude the cryopumping of any air in the vacuum tube vacuum space onto the cold structures.

For the H₂/O₂ mixing scenario considered in this section, the overall occurrence frequency is estimated at less than $0.0084 \times 0.0084 \times 1.3 \times 10^{-4} = 9.2 \times 10^{-9}$ /year. From the perspective of the expected frequency of occurrence, this scenario is beyond the design basis and need not be considered. However, to ascertain the margin of safety of the HB-4 beam tube wall to internal H₂/air detonations, this "incredible frequency" event has been chosen for further analysis. As stated in Sect. 1.2.3, it is a safety requirement that the beam tube wall survive an internal hydrogen/air detonation event.

That the chosen detonation event is a worst-case event is because of the inventory of oxygen available to react. (The only other event that can result in a higher inventory of air residing in the vacuum tube is one involving sustained cryopumping of O₂ and N₂ onto the cold LH₂ structures throughout a HFIR operating cycle. This event is discussed in Sect. 3.2.3 but can be prevented with proper design features and leak detection methods.) The chosen detonation event occurs only because it is assumed that both the beam tube neutron window and the vacuum tube neutron window fail catastrophically and result in atmospheric pressure air coming in direct contact with the cold exposed walls of the moderator vessel and the exposed LH₂ transfer lines inside the vacuum tube. The path followed by the entering atmosphere is assumed initially evacuated, and the end-state inventory inside the vacuum tube is maximized by equilibrating the entering air directly with the cold aluminum structures. No thermal interaction between the entering air and the vacuum tube wall is considered. Thus maximum cooling of the entering air permits a maximum inventory of air to enter the vacuum tube. Such enhancement of the air inventory inside the vacuum tube results in about 2–3 times the air content inside the vacuum tube than would exist if no equilibration occurred. As discussed earlier, mechanical and administrative actions will be in place to mitigate such an event if it were to occur.

3.2.2.2 Thermal Equilibration Analysis Technique For Determining Maximum Vacuum Tube O₂ and N₂ Inventories

The purpose of this section is to describe in general terms the equilibration process used in the determination of the maximum oxygen inventory extant in the vacuum tube. The process has been computer programmed because of the iterative nature of several steps of the calculation. It is emphasized

that the equilibration procedure followed does not represent the dynamics, but only the end state of the process. It is also noted that the process is assumed to occur at 1 atm of total pressure inside the vacuum tube. Ideal gas law behavior is assumed for the gases. The following discussion presents overview information, while details of the actual calculation are presented in Appendix B.

The temperature of the LH₂ aluminum structures is assumed to be at 15 K initially. This temperature compares to the 17–20 K expected in the operating cold source moderator vessel and was chosen to maximize the heat exchange involved in the equilibration process, and thus the resulting O₂ inventory in the vacuum tube. This temperature is below the condensation temperature of O₂ (90 K) and N₂ (77 K) and freezing temperatures of these substances (O₂ freezing temp = 54.4 K; N₂ freezing temp = 63.1 K), and so initial heat transfer effect will be to condense and freeze O₂ and N₂ onto the cold LH₂ structures.

However, freezing is not considered in the equilibration process as it represents an intermediate state of the O₂ and N₂ that effectively promotes additional ambient temperature gas inflow into the vacuum tube and leads to heating above the freezing temperatures. What is important to the goal of this calculation is the final O₂ and N₂ inventory inside the vacuum tube such that the gas is at 1 atm and is in thermal equilibrium with the aluminum structures. This end state may consist of a two-phase condition of liquid condensate on the surface of the structure in thermal equilibrium with gas in the vacuum space. Because the condensation temperature of O₂ exceeds N₂, the final state gas composition may be rich in N₂ and deficient in O₂ while the final end-state equilibrium condensate composition (if any) may be rich in O₂ and deficient in N₂.

Furthermore, for all equilibration calculations performed to date, only O₂ condensate is necessary to be modeled for the end-state condition. This is because of the small thermal mass of the cold LH₂ structures inside the vacuum tube and the correspondingly small condensate inventories calculated. For all calculations performed to date, the final N₂ inventory has always been determined as completely gaseous.

The equilibration analysis assumes that the cold exposed LH₂ walls cool and initially condense the entering air. The cooling and condensation of the entering air reduces the pressure of the air and allows more air to enter. This cooling effect of the atmosphere leads to heating of the cold aluminum walls. The determination of the O₂ and N₂ inventories extant in the vacuum tube at the time of assumed moderator vessel failure is established by the full thermal equilibration of O₂ and N₂ inside the vacuum tube with the LH₂ structures inside the vacuum tube. The details of the equilibration analysis are presented in a five-step procedure described in Appendix A.

3.2.2.3 Inventory Calculation Results

O₂ and N₂ inventory results have been recently calculated for the cold source design. Important inputs into this calculation include:

M1P = LH₂ structures aluminum mass. This mass includes the mass of the moderator vessel walls and the exposed walls of the LH₂ transfer lines inside the vacuum tube = 1536.4 g.

VLH2 = volume of LH₂ housed inside the LH₂ structures = 1520.2 cm³.

VTUBE = vacuum space volume inside the vacuum tube = 22.92 L.

Results of the analysis indicate maximum condensed masses of O₂ = 22.2 g and N₂ = 30.0 g during the equilibration process. The end-state would be two-phase. The liquid O₂ remaining on the hydrogen bearing structures is 14.7 g, and the O₂ in the vacuum tube atmosphere is 9.2 g, for a total O₂ inventory of 23.9 g. The N₂ extant in the vacuum tube is calculated to be 78.8 g and is all gas. The end-state equilibrated temperature is 90 K and the final pressure is 1 atm.

As a check of the algorithm, the mass fraction of O_2 is $23.9/(23.9 + 78.8) = 0.233$, while the mass fraction of N_2 is $78.8/(23.9 + 78.8) = 0.767$. These are precisely the mass fractions of O_2 and N_2 in air and reflect the fact that the equilibration procedure described previously involves only the in-flow of ambient temperature air into the vacuum tube. The molar content of the vacuum tube atmosphere is $O_2 = 9.21/32 = 0.29$ g-mol and $N_2 = 78.8/28 = 2.8$ g-mol for a total of 3.1 g-mol. This result matches exactly the number of moles of gas expected (from the ideal gas law) to fill the vacuum tube vacuum space at a temperature of 90 K at a pressure of 1 atm.

It is also noted that the vacuum tube gas mixture is concentrated in N_2 at $2.82/3.10 = 0.91$, or 91% mol fraction. Likewise, it is deficient in O_2 at $0.29/3.1 = 0.093$ or 9.3% mol fraction. For comparison, normal dry air is about 79% N_2 and 21% O_2 .

3.2.2.4 Gas Inventories Considered In The Preliminary Detonation Analysis

For purposes of the detonation analysis pursued in the following section, it is assumed that the entire inventory of *condensed* O_2 is available for reaction with H_2 leaking into the vacuum tube from the failed moderator vessel. In addition, a portion of the 9.3% O_2 gas mixture residing in the vacuum tube at the end of the equilibration process is also assumed to be available for reaction.

The purpose of this section is to explain the reasoning for selecting actual quantities of H_2 , O_2 , and N_2 that are considered in the gas detonation analysis. It is again emphasized that the goal of the current detonation analysis is to determine the peak pressures associated with the worst case detonation event inside the confines of the beam tube. It is clear that such an event would be limited by the quantity of oxygen available for reaction and not by the quantity of hydrogen available. As a point of reference, the quantity of high-density supercritical hydrogen residing inside the moderator vessel and the vacuum tube portions of the transfer lines during normal operation of the cold source loop is approximately 111 g.

It is assumed that H_2 gas initially leaves the failed moderator vessel at 90 K and displaces a portion of the existing 9.3% O_2 gas mixture residing inside the vacuum tube at the end of the equilibration process. Furthermore, it is assumed that the inventory of this initial H_2 gas is stoichiometric with respect to the remaining O_2 gas not displaced from the vacuum tube. Finally, it is assumed that the gas displacement process occurs at atmospheric pressure. The gas inventories extant in the vacuum tube after the displacement process consistent with these assumptions are as follows: $O_2 = 7.8$ g, $N_2 = 66.5$ g, and $H_2 = 0.97$ g.

To make the remaining condensed O_2 inventory available for reaction with H_2 , it is assumed that the entire 14.7 g of condensed O_2 boils off very rapidly (at 90 K) and enters the vacuum tube atmosphere along with an additional quantity of H_2 gas at 90 K. The mass of H_2 added is 1.8 g and is stoichiometric with respect to the boiled-off O_2 . This boil-off process is assumed to occur rapidly so that the pressure does not remain constant inside the vacuum tube. The gas pressure inside the vacuum tube after the boil-off process is 1.44 atm (21.3 psia). This pressure is deduced by the ideal gas law for the conditions presented below.

Vacuum tube conditions at time of detonation:

Gas temperature = 90 K

O_2 mass = $14.731 + 7.773 = 22.504$ g = 0.70325 g-mol

N_2 mass = 66.472 g = 2.374 g-mol

H_2 mass = $0.9716 + 1.8412 = 2.8128$ g = 1.4064 g-mol

gas volume = 22.92 L

These inventories are considered in the following section in the estimate of peak detonation pressures.

3.2.2.5 Beam Tube Detonation Analysis And Results

The method used to deduce maximum peak pressures during a H₂/air detonation was taken from Sect. A15.0 of the BNL cold source safety analysis report.⁴ The method determines two components of peak detonation pressures: (1) static or side-on pressure of the burned gas and (2) impact or reflected pressure. Static peak pressures would be measured by a moving observer immediately behind the supersonic detonation wave, while impact pressure would occur if a stationary plane surface were placed normal to the direction of the motion of the wave.

The method is based upon experimental data for static and reflected peak pressures in H₂/O₂ mixtures initially at atmospheric pressure and 18°C (291 K). These experiments show approximately 18- and 37-fold rises for static and reflected pressures, respectively, over the initial pressure.

Corrections are made to estimate the inerting effect of the presence of N₂ in the reacting mixture. The following mixtures and their peak static pressures are taken from ref. 4.

Unburned mixture	Peak static pressure (atm)
2H ₂ + O ₂	18.05
2H ₂ + O ₂ + 3N ₂	15.63
2H ₂ + O ₂ + 5N ₂	14.39

It is observed that the higher the N₂ content of the unburned mixture, the lower the peak static detonation pressure.

Using the gas inventories presented in Sect. 3.2.2.4, the following detonation reaction is considered for the HB-4 beam/vacuum tube.

The dilution effect of the 3.376 N₂ concentration is determined by simple interpolation from the experimental results. The expected peak static detonation pressure for this mixture occurring from an initial temperature of 291 K and 1 atm of pressure is deduced as 15.4 atm. This effect is represented as a multiplier of the pure H₂/O₂ detonation (e.g., 15.40/18.05 = 0.8532). Thus, in general terms: Ps/Pi = 0.8532 * 18 = 15.3576, and Pr/Pi = 0.8532 * 37 = 31.5684.

The above expressions are assumed to hold for any initial pressure as long as it occurs at the same initial temperature as the experiments performed for the pure H₂/O₂ detonations. Again, the initial temperatures for the experiments were 18°C (291 K). For the HFIR HB-4 vacuum tube, the initial mixture pressure for the given gas composition at 291 K is determined from the ideal gas law at Pi = 4.671 atm (68.66 psia).

Substituting Pi into the previous expressions given for Ps and Pr gives the final result of Ps = 71.731 atm (1054.4 psia) and Pr = 147.45 atm (2167.5 psia) for the worst case hydrogen/air detonation event in the HB-4 beam tube.

3.2.3 Vacuum Tube Cryopumping Analysis

An important aspect of ensuring the safety of the HFIR HB-4 cold source is an evaluation of the importance of the phenomenon of cryopumping in regards to defining the maximum O₂ inventory that might be available to react with hydrogen inside the beam tube. Cryopumping occurs when gases such as O₂ and N₂ come into direct contact with cold surfaces and condense. In addition to condensation from gas into liquid, freezing may occur if the surface is cold enough.

The cryopumping phenomenon is largely undetectable as the pumped gas species are removed from the atmosphere and are in equilibrium at very low vapor pressures. Induced heat loads into the hosting structures are very small ($\ll 1$ W) compared to other heat loads and are undetectable by heat balances performed for the structures. High-level vacuums may be maintained through the cryopumping process even though small leaks may exist in the vacuum boundaries. Thus the normal leak detection techniques of monitoring the level of vacuum and of monitoring the composition of the residual gases inside the evacuated space may be of little use. The only reliable way to detect leaks is to ensure that the gas species flowing through the leaking vacuum boundary are not susceptible to cryopumping and to examine the residual vacuum space gases for the unaffected species. The current cold source design makes maximum use of this technique by provision of a layer of helium between the outside air and the vacuum space housing cold source structures likely to produce cryopumping.

The condensation temperatures for O_2 and N_2 at atmospheric pressure are 90 K and 77 K, respectively. Furthermore, the triple-point pressure/temperature combinations for selected species of interest are given in Table 3.1.

Table 3.1. Selected triple-point data⁵

Species	Temperature (K)	Pressure (torr)
H_2 (para)	13.8	52.4
N_2	63.1	93.73
O_2	54.4	1.12
He	none	none

The purpose of this section is to quantify the potential for cryopumping inside the vacuum tube. Inspection of this data establishes the likelihood of cryopumping of both O_2 and N_2 onto the cold 17–20 K LH_2 structures housed inside the vacuum tube. Furthermore, the expected normal operating vacuum level inside the vacuum tube is estimated at $1e-6$ torr, and it is concluded that both O_2 and N_2 would sublime onto the surfaces directly as solid without passing through a condensed liquid phase. It is also concluded that helium will remain a gas for the vacuum levels and temperatures of interest.

The goal of this analysis is to establish (during the HFIR operating cycle) the upper limit of O_2 (and N_2) that could be accumulated inside the vacuum tube vacuum space if no mitigative precautions are taken. Therefore, it is assumed that the portion of the beam tube lying beneath the reactor pool outside the reactor vessel is at a high-grade vacuum along with the interior region of the double-wall neutron window at the end of the vacuum tube. It is assumed that this double-wall window and the neutron window at the end of the beam tube develop small leaks allowing beam room atmosphere to enter the vacuum space of the vacuum tube. The vacuum pumping system for the vacuum tube vacuum space is assumed to maintain the required operating vacuum despite the leaks, specifically at the vacuum level of $1e-6$ torr. Simultaneous failure of both neutron windows is estimated to have an occurrence frequency of less than $0.0084 * 0.0084 = 7e-5$ /year from ref. 3 and would be classified as a “limiting” event.

3.2.3.1 Method And Results

The calculation of the rate of cryopumping consists of the simple evaluation of the molecular strike rate of O_2 and N_2 gases impacting the cold LH_2 structures inside the vacuum tube. From gas theory,⁶ the rate at which molecules strike the surface is given by:

$$n_{flux-i} = \frac{P_i}{\sqrt{2\pi mkT_g}} \quad (64)$$

where:

- n_{flux-i} = strike rate for species I (molecules/m²-s),
- P_i = gas species I pressure (kg/m-s²),
- m = mass of one molecule of species I, (gram/molecule),
- k = Boltzmann const (= 1.3805e-23 j/K),
- T_g = gas temperature (K).

Assuming that the residual gas mixture of the vacuum tube is at 1e-6 torr, the gas composition is 21% O₂ and 79% N₂ (mole fraction), and that the capture coefficient is unity (e.g., every molecule striking the cold surface sticks), the strike rates are estimated as 7.52964e17 molecules/m²-s and 3.02832e18 molecules/m²-s for O₂ and N₂, respectively. The total surface area of the moderator vessel and the LH₂ transfer lines inside the vacuum tube is estimated at 3512.19 cm². These results correspond to vacuum tube atmosphere mass removal rates of 1.406e-5 g-O₂/s and 5.654e-5 g-N₂/s. Assuming uniform removal rates throughout the normal HFIR operating cycle of 23 d, the cumulative masses of condensate on the LH₂ structures inside the vacuum tube are O₂ = 27.94 g and N₂ = 112.36 g.

3.2.3.2 Conclusions Of Cryopumping Analysis

The 27.9 g of O₂ and the 112.4 g of N₂ exceed the corresponding inventories of the equilibration analysis of Sect. 3.2.2.4. The current results represent upper limits for the cryopumping phenomenon as it is assumed that the capture coefficient is unity for both species. It also assumes that the evaporation rate for the gases is zero, so that the net rate of mass transfer is the condensation rate. Finally, the cryopumping analysis is likely conservative because of the assumption that the gas leaked into the vacuum tube vacuum space is entirely O₂ and N₂. Such an assumption is consistent only with a configuration whereby the region between the vacuum tube neutron window and the beam tube neutron window is completely evacuated.

It is not yet decided if the beam tube extension in the final HB-4 cold source design is to have the space adjacent to the reactor pool maintained under vacuum. Any gas mixture in this region of the beam tube must have a minimal impact on the nature of the cold neutron beam passing through. It is clear, however, that incorporation of a helium atmosphere in this region is highly desirable. Simultaneous leaks through the neutron windows would be detectable as the helium carrier gas would not condense onto the cold LH₂ surfaces and would be monitored by residual gas analysis of the vacuum tube vacuum space. It is estimated that to produce the gas inventories (22.504 g of O₂ and 66.472 g of N₂) considered in the detonation analysis of Sect. 3.2.2.4, the helium atmosphere of the beam tube extension would require contamination at a level of 16.9% O₂ and 53.4% N₂ and continuous in-leakage of these gases throughout the entire HFIR operating cycle (23 d). Such large levels of helium contamination should be easily detectable prior to significant buildup of O₂ onto the LH₂ structures and should provide adequate time to respond.

3.2.4 Future H₂ Hazards Analyses

Several of the events recommended for selection as design basis events for the HFIR cold source (see Sect. 3.3) involve hazards associated with hydrogen. These events include loss of hydrogen circulation, loss of refrigeration, loss of vacuum tube vacuum, break in the moderator vessel boundary inside the

1. What could cause a direct release of hydrogen to the atmosphere?

This question is answered by postulating what event(s) or type of events can create a challenge to the containment features discussed below to the degree that hydrogen containment can no longer be assured. Obviously a breach induced in the hydrogen containment boundary applies, but also any events that substantially alter the operating conditions of the hydrogen circulating loop must be considered. Natural phenomena (seismic or tornadic events), material failure, rapid system operating degradation, and human error are the types of events that can necessitate relieving hydrogen pressure or cause a release of hydrogen through a failed boundary.

2. What helps prevent the escape of hydrogen to the surroundings? List of preventive and mitigative safety features:

- H₂ loop pressure control system,
- H₂ relief valve to expansion tank (volume),
- H₂ overpressure relief valve rupture disk set to H₂ vent line/stack,
- expansion vessel rupture disk to H₂ vent line/stack,
- vacuum jacket (main loop) overpressure relief to H₂ vent line/stack,
- vacuum thimble rupture disk to H₂ vent line/stack,
- helium jacket overpressure relief to H₂ vent line/stack, and
- thimble cooling water relief to reactor pool (assumed).

3. What exists to detect/indicate the leakage of hydrogen out from the transfer line if it should occur?

- hydrogen boundary failure detection mechanisms,
- H₂ pressure sensors,
- H₂ temperature sensors,
- helium refrigerant temperature sensors,
- vacuum jacket gas analyzer (main transfer loop),
- helium jacket gas analyzer (main transfer loop), and
- vacuum jacket gas analyzer (thimble region).

The transport system uses multiple, concentric (coaxial) layers of convoluted, stainless-steel transfer line and a vacuum insulation blanket (10^{-5} torr) to contain the hydrogen and maintain operating set points. The vacuum layer is also monitored by gas analyzers to detect hydrogen or inleaking helium, if a breach developed in one of the inner walls of the transfer line. An outer blanket of helium insulation above atmospheric pressure prevents oxygen from freezing on the outside surface of the vacuum layer or being pulled in through a small hole to freeze on the hydrogen-bearing line. This helium is also monitored for inleaking air (outleaking hydrogen would also be detected) to indicate a loss of outer transfer line integrity. When a coupling is necessary to join sections of transfer line, special seals are used that constrict when cold to form a leak-tight barrier. These seals are also designed to maintain separation and continuity across the designated pipe channels (hydrogen, vacuum, or helium).

The multiple containment philosophy is also maintained in the moderator thimble assembly situated within horizontal beam tube HB-4. Prior to entering the thimble assembly, the coaxial hydrogen transfer line separates into individual inlet and return tubes that run the length of the moderator thimble to the moderator vessel. The space inside the thimble is maintained at 10^{-5} torr by a separate vacuum system, is monitored by a separate gas analyzer, and has a dedicated rupture disk. Because of the heat load near the core, helium enriched cooling water is circulated between the inner surface of HB-4 and the outer surface of the moderator thimble. The purpose of this cooling water is primarily to cool the moderator thimble. However, in doing so, it also helps to minimize the radiative heat transferred from the thimble to the

moderator vessel during operation (assuming a constant reactor heat load). If a leak develops in the thimble, cooling water leaking into the vacuum region will freeze on the surface of the hydrogen lines, and the entrained helium will be liberated into the vacuum space and detected via the gas analyzer(s). The vacuum thimble has been designed to withstand the maximum primary pressure of 690 psig external, and the new HB-4 is qualified to 1500 psig external pressure, though HFIR technical specifications presently require less.

Should it be necessary to vent hydrogen from an intact system, the primary option for pressure relief is the expansion vessel via a solenoid valve and relief valve. Further relief is available via a separate relief valve and/or rupture disk that relieves directly to the vent line and a stack that exhausts to atmosphere outside the HFIR Building. All vent lines are vacuum-jacketed, double-walled lines. If hydrogen is leaking into the vacuum space, the vacuum pump will draw it out into the vent line or, if pressure is increasing at too great a rate, a 1 bar (g) rupture disk sized to handle the expanding gas is provided to accomplish the same task. A rupture disk (2 bar g) is also provided in the helium blanket should a double-breach occur (inner- and secondary-line layers). Rupture disks will be sized based on the maximum amount of gas expansion that can occur. The above information answers questions 2 and 3 and illustrates the defense-in-depth approach to preventing release of hydrogen (i.e., shows what must fail in order for hydrogen to escape through other-than-intended pathways).

3.3.3.2 Qualitative System Modeling

With the current system design state and existing safety features, determining what could happen began with postulating general failure modes and accident types (loss-of-cooling function, line break, increase in heat load, etc.) and qualitatively estimating the consequences of each event. This is not the classic failure modes and effects assessment (FMEA) that details each component in a system or systems from the bottom up. However, it provides useful information in assessing a developing design where all components and features are not fully accounted for. This qualitative assessment is shown below in Table 3.2.

3.3.3.3 Event Scenario Categorization

The accident categories and frequency groups in the HFIR SAR⁷ are based on those recommended by NRC Reg. Guide 1.70 for reactor transients. A similar set of event categories⁸ that apply to the continued containment of hydrogen by the cold source systems are as follows:

- insufficient H₂ boundary cooling,
- increase in H₂ system pressure,
- release during system filling/exhaust, and
- failure of boundary integrity.

The event sequences that fall into each of these categories are further grouped by their frequency of occurrence. **Moderate** frequency event sequences are defined as sequences with a frequency greater than or equal to 10⁻¹ per year. **Infrequent** event sequences are those estimated to occur in the frequency range between 10⁻¹ and 10⁻⁴ per year. **Limiting** events are determined to have frequencies of occurrence less than 10⁻⁴ but greater than 10⁻⁶ per year. Based on ANSI/ANS-51.1-1983,⁹ DOE Order 6430.1A,¹⁰ and DOE/OR-901,¹¹ accident sequences occurring at a frequency of less than 10⁻⁶ per year are considered beyond the design basis of the HFIR cold source and are generally not included for further analysis except to provide validation.

To facilitate the process of categorizing the events presented in Table 3.2 below, a master logic diagram (see Fig. 3.5) was constructed to align those events that result in similar repercussions to the system under one of the four event categories listed.

Table 3.2. FMEA for postulated cold source abnormal events

Failure mode/event	Result to system	Recovery actions	Potential consequences
Loss of H ₂ cooling due to: ☒ Loss of electric power ☒ Loss of refrigeration ☒ Loss of H ₂ main circulation	Loss of refrigeration H ₂ warms under back-up-powered circulator Warming of H ₂ Warming of H ₂ All: increased heat retention in moderator plug assembly from gamma heating	Vent Start gas circulator for controlled warming Start gas circulator for controlled warming	All of the following apply to these events: Expected: controlled venting to expansion vessel Designed for ①: normal venting fails. Relief valve to expansion vessel opens or rupture disks blow at 19 bar to exhaust header Designed for ②: hot moderator vessel breached due to rapid pressure increase from boiling/expanding hydrogen. H ₂ enters the moderator thimble vacuum region where rupture disks vent to the exhaust header Designed for ③: normal venting and rupture disks fail. Inner transfer line breach into vacuum blanket where a rupture disk vents to the exhaust header Designed for ④: all above fail and breach into helium blanket occurs. Rupture disk vents to the exhaust header Worst case: all the above fail. Breach of outer H ₂ containment into beam cavity, beam room, or into safe room outside 7900
Spontaneous break of H ₂ transfer line	1. Release H ₂ to vacuum jacket/thimble region 2. H ₂ released to beam room 3. H ₂ released outside 7900	Manual emergency venting	See ②, ③, ④, and worst case, above, and extended cold source unavailability
Spontaneous moderator vessel breach due to incipient failure	Release H ₂ to vessel plug vacuum/thimble	System shutdown; reactor shutdown to reduce heat load	See ② above. Extended cold source unavailability Neutron window failure if pressure relief is insufficient Worst case: Vessel plug assembly breach. H ₂ expands into He/H ₂ O-filled cavity

Table 3.2 (continued)

Failure mode/event	Result to system	Recovery actions	Potential consequences
<p>Oxygen intrusion</p> <p>▣ when system is opened for maintenance</p> <p>▣ due to chemical breakdown of other intruding gasses</p>	<p>Possible exothermic reaction and physical system damage</p>	<p>Evacuate research personnel</p>	<p>Possible fire scenario Reactor shutdown Extended cold source unavailability</p> <p>(Relief capacity and pressure wave effects for this scenario unknown at present)</p>
<p>Increase in moderator heat load</p> <p>▣ Reactor power excursion</p> <p>▣ Normal start-up</p>	<p>Rapid increase in heat load Consequences unclear at present</p> <p>Increase in heat load</p>	<p>Reactor scram at 110% power Further actions not presently clear</p> <p>Expected: none Abnormal: undefined</p>	<p>Depending on further analysis, hydrogen may suddenly boil. Further analysis of physical effects of this event on the moderator plug and vessel are required to better define the consequences and recovery actions</p> <p>Expected: normal programmed temperature control Abnormal: system warms above operating set points. Further actions undefined</p>
<p>Seismic Event</p> <p>▣ 0.005 g–0.12 g</p> <p>▣ > 0.12 g–2.04 g</p>	<p>System intact</p> <p>Undetermined at this time</p>	<p>Check cold source control room instruments and walkdown system</p> <p>Same</p>	<p>That portion of the system inside 7900 is designed for these levels. Thus, there should be no adverse effects in the building. However, components outside 7900 may suffer damage and release hydrogen</p> <p>Undetermined at this time. Potential for multiple breaches in H₂ containment</p>

Table 3.2 (continued)

Failure mode/event	Result to system	Recovery actions	Potential consequences
H ₂ flow blockage or freezing due to failed refrigeration control	Circulator cavitation and failure Rapid heating of moderator vessel inventory	Adjust temperature control or stop refrigerator Open ventilation system to expansion vessel	Same as for loss of circulation including worst case. Most likely location for freeze plug blockage is between main heat exchanger (inclusive) and moderator vessel
I&C failure ☐ Central computer crashes ☐ individual sensors fail	Depends on control logic and failure mode. If failure causes an action to occur, system instability could lead to hydrogen heat-up	Shut down system and vent to expansion vessel automatically, manually, or via relief valve	Expected: system unavailability Worst case (no action taken) same as for loss of circulation
Failure of safety window integrity in vacuum thimble	Loss of moderator thimble plug vacuum	For small leaks, possible operation in standby at 100 K, else shutdown reactor and CS and vent H ₂ to expansion vessel	When standby operation is maintained: Extended outage at end of fuel cycle to isolate, repair, and test. Whether or not beam tube extension is inerted, evacuated, or air-filled will play a part in determining the consequences. When standby can not be maintained: System is shutdown to protect from cryopumping of air onto H ₂ carrying components if inleakage path is available. Reactor runback or shutdown to protect moderator vessel. Long-term cold source unavailability for repair and testing
Loss of vacuum	Loss of gas analyzer function and insulative property of vacuum barrier	Start of redundant vacuum skid	Increased heat load on inner piping, and thus H ₂ , from heat transfer from outside the system
Seal failure ☐ Transfer line seal ☐ Component seal (e.g., valve, etc.)	Depends on which layer is affected (H ₂ , Vac, or He): 1. slow leak to vacuum detected by gas analyzer 2. large leak and H ₂ depress 3. loss of vac. (He picked up by gas analyzer) 4. loss of He pressure and inventory	For hydrogen leak, warm system to standby or shutdown For vacuum leak or helium leak, monitor until end of cycle or shut down for replacement	System unavailability for seal replacement or loss of use of cold source while running in standby For large leakages, same consequences as line break.

Fig. 3.5. Master logic diagram of initiating events for the HFIR cold source.

3.3.4 Examination of Event Scenarios

Events that are a high risk for releasing hydrogen outside the system (note that the term system here refers to those portions of the cold source that form part of the hydrogen containment) are identified by considering the event's frequency of occurrence, its subsequent effect on the system, and the probability of failure of equipment designed to mitigate the event. Initiating event frequency and component failure data were taken from reports on cryogenic equipment performance, where possible, and the HFIR PRA as documented in ref. 3. The quantitative analysis assumes that there are 10 fuel cycles per year at 23 d per cycle (approximately 5500 operating hours per year). Assuming that plant availability remains constant, the cold source systems will have to exceed plant availability because they must be at operating temperature prior to reactor start-up. Thus, an annual operating demand of 6000 h has been used in the calculations to estimate the event frequencies. Based on the total event sequence frequency, events are classified as moderate, infrequent, or limiting as described above in Sect. 3.3.3.3. Sequences that have consequences that are clearly bounded by events of similar frequency classification within each initiator category or that have a frequency that is considered beyond design basis (mean annual probability $<10^{-6}$) are eliminated from further discussion. The remaining events are recommended for further analysis.

The consequences resulting from seismic or tornadic activity are encompassed by the events included in this frequency assessment. Also, the frequency of occurrence of earthquakes and tornados in excess of the plant design basis for these events is generally less than the internal events discussed here. Therefore, these external events presently are not given further consideration.

3.3.4.1 Insufficient Boundary Cooling Events

Because of the extremely cold temperatures required for continuous cold source operation, minimizing or eliminating unnecessary heat flux from the surrounding environment is very important, whether operating at 4 bar or 15 bar pressure. Approximately 1.6 kW must be removed from the cold source moderator to maintain proper conditions during reactor operation. In an effort to ensure continuity of operation, redundancy has been designed into the refrigeration system (three out of five screw compressors and three of four expansion engines required for normal operation), the hydrogen circulation system (redundant circulators), and the vacuum pumping stations. Loss of a cooling system or increase in the heat load without recovery actions (standby equipment start, automatic relief, or temperature control adjustment) will set into motion the warming and expansion of the hydrogen at a rapid rate. (Hydrogen has a gas-to-liquid volumetric expansion ratio of 800:1.) Standby/emergency mode operating equipment, such as the low-density hydrogen circulator or refrigeration operation on liquid nitrogen, are not considered in any of the initiating event frequency estimates because their unavailability would not initiate an abnormal event during normal operation. The events evaluated as producing insufficient boundary cooling or increased heat load are summarized in the following paragraphs. Their estimated annual frequency of occurrences are found in Table 3.3.

Loss or degradation of refrigeration can result from compressor failure, flow obstruction, loss of helium through a break or seal failure, failure of the vacuum system, cold box failure, loss of support system, or failure of one or more expanders. In the worst case, the effect on the hydrogen loop is immediate warming and gas expansion. In the simple case of degradation of refrigeration, a gradual hydrogen loop warming to standby operation is expected to occur.

The refrigeration system also has an emergency mode of operation that requires only one compressor, the liquid nitrogen precooler, and an open path to the main heat exchanger. Availability of this mode of operation is expected to be sufficient for operating the hydrogen loop in standby operation at 100 K. It is not clear if one compressor will be designated as the emergency compressor or if any available compressor can be used for this mode. Redundancy of these units should contribute to a low likelihood of emergency mode unavailability, but the likelihood of compressor failures during normal operation will likely increase due to common cause failures.

Table 3.3. Insufficient boundary failure event frequency estimates

Event	Frequency	Main contributor	Frequency category
Loss of refrigeration	0.78/year	Expander unavailability	Moderate
Degraded hydrogen circulation	1.8/year	Running circulator fails during operation	Moderate
Loss of normal hydrogen circulation	0.23/year	Running and standby circulators fail	Moderate
Loss of vacuum—major (thimble or safe room)	1.3E-4/year	Pump module and isolation failure	Infrequent
Loss of vacuum—minor (thimble or safe room)	2.5E-2/year	Seal and casing leakage	Infrequent
Loss of site power	4.4E-1/year	NA	Moderate
Loss of preferred feeder	4.1E-1/year	NA	Moderate
Reactor start-up	10/year	NA	Moderate
Power excursion	1.6E-1/year	NA	Moderate
Loss of thimble cooling	Not evaluated at this point	See discussion above	NA

Loss of normal hydrogen circulation may be caused by high-density circulator unavailability, flow-path blockage due to spurious valve closure, or a loss of dedicated power sources. Expected system response to loss of both high-density circulators is a controlled transition to standby with an associated response from the refrigeration system and start of the low-density circulator. As the hydrogen warms, an associated response is normal venting to the large storage tank via solenoid valve. However, relief valves and/or rupture disks are present to allow excess venting during rapid pressure/temperature increases as a backup if the transition to standby fails. Insufficient relief may result in breaching the loop boundary at a weak point in the circuit, forcing hydrogen into the vacuum boundary. An additional option is to scram the reactor to preclude overheating the aluminum cold source equipment in the reactor vessel.

Loss of vacuum, either in the moderator region, in the transfer lines, or in the pump module will lead to a gradually increasing heat load on the hydrogen, its expansion, and possibly a transition to standby cold source operation. A large loss of vacuum (e.g., blown seal, boundary failure) would likely require system shutdown to protect against cryopumping of air onto the cold surfaces that could lead to other hazards upon warming. Otherwise, the loss of vacuum is considered to be a rather benign event with regard to system stability, unless it is accompanied by additional failures that could affect the hydrogen boundary.

Loss of off-site power will result in the unavailability of the refrigeration system as it is currently designed. The moderator heat load will decrease as the reactor scrams on high flux-to-flow, but delayed gamma heating will still supply a substantial heat load to the hydrogen. While the high density circulators are designed to swap over to an uninterruptible power supply (UPS), the vacuum pumps that serve the hydrogen and refrigeration systems will coast to a stop, and the vacuum isolation valves should then close to minimize vacuum loss. The low-density circulator is not served by UPS. Nitrogen gas supply to pneumatic equipment should remain available and will be used with the battery-powered instruments to continue the pressure control function. If a liquid-nitrogen-cooled backup refrigerator is incorporated into

the design, the cold source may be able to maintain standby conditions at 100 K, reducing the time necessary to cool the system back down to normal operating temperature prior to a reactor restart. If no such provision is made, the hydrogen will warm, expand, and vent to the expansion vessel.

Increase in hydrogen heat load because of normal reactor start-up or reactor power excursion places demands on (1) the temperature control system to balance the refrigeration output and (2) the inventory control system to relieve any increase in pressure from expanding hydrogen (or add inventory if the temperature control system overcompensates and overcools the hydrogen). Normal start-up is a planned occurrence, and, thus, the design incorporates temperature and inventory control features to cope with the transition to normal hydrogen loop operation. Reactivity transient events will be short in nature because of the safety scram set point at 110% reactor power. The added heat load during the time it takes to reach the reactor scram set point could increase the hydrogen temperature in the loop.

Moderator thimble cooling is provided to assist in transporting heat from the moderator thimble out to the beam tube. This feature is currently understood to be passive (i.e., no forced circulation) and accomplished with demineralized water that has helium entrained. It is not clear how, without some forced mixing, the helium will be prevented from evolving out of the water and forming a bubble at the top of the vacuum thimble. However, unless changes are made to convert this into an active system, no further probabilistic evaluation will be performed on the thimble cooling function. Failures that would result in losing this design feature are covered in other events.

3.3.4.2 System Pressure Increase Events

These events refer to incidents where the hydrogen or hydrogen boundary is subjected to additional external pressure. This increased pressure and the effects of the precipitating failure on other components are the phenomena that the mitigative features (pressure relief, cooling, etc.) must contend with. Pressure increases as defined here are severe enough to warrant a required system or reactor shutdown. They go beyond simple loss of vacuum scenarios to physically affect the cold source system. Pressure increase scenario frequency estimates are given in Table 3.4

Table 3.4. Pressure increase scenario frequency estimates

Event	Frequency	Main contributor	Frequency category
HB-4 integrity failure with failure to relieve pressure	$<<10E-7/\text{year}$	HB-4 failure	Beyond design basis
Moderator thimble integrity failure	$5.7E-5/\text{year}$	Spurious thimble failure	Limiting
Air intrusion scenario	See discussion	No longer considered separate event	NA

Failure of HB-4 beam tube, whether large or small, will allow reactor coolant water to enter the moderator thimble cooling space, increasing the pressure in this region. If HB-4 did fail, a rupture disk (assumed at this point) should actuate to prevent reaching the normal reactor pressure of 32.3 bar (468 psi) or the moderator vacuum thimble external design pressure of 47.7 bar (690 psi). Thimble failure or deformity brought on by this scenario could result in contact with the cold-hydrogen bearing components held within, leading to further undesirable events. The relief mechanism for the final system has not been determined at this point in the design. The present HFIR configuration utilizes a rupture disk as described above.

If leakage of reactor coolant into the thimble annulus were to occur slowly over time, the reactor coolant pressure detectors or makeup systems would not likely indicate such an event. Dedicated pressure measuring devices should be incorporated to signal a small leakage through HB-4.

Moderator vacuum thimble failure because of a flaw in this component inside the reactor vessel would allow the helium-enriched water surrounding it to leak into the vacuum region insulating the moderator vessel. Water would immediately freeze onto the hydrogen lines and/or vessel while the liberated helium should be detected by the gas analyzers that precede the vacuum pumping stations. The effect on the stability of the hydrogen system or on the integrity of the transfer lines and moderator vessel is unknown at this time. Operation could not continue in this state, however, and the assembly would require repair or replacement. (Note: failures in the rear flange area of the thimble assembly where the neutron window and hydrogen lines penetrate are excluded from this scenario.) A failure (pinhole leak or larger) in the rear flange would draw helium gas in from the beam tube extension region. This gas should also be detected by the residual gas analyzers. The potential consequences of this event are down time to locate and possibly repair the leak and should pose no threat to the hydrogen-bearing components within the thimble region.

Reference 3 reported **air intrusion events** as a separate class of events. Further study has led to the conclusion that air intrusion is the consequence of a specific failure mode such as an integrity-related event or a maintenance error. The potential consequences of an air intrusion event are, to an extent, the reason for many of the safety features in the cold source design (vacuum and helium blankets to guard against cryo-pumping and the residual gas analyzers). There is virtually no way for air to enter loop during operation because of the higher hydrogen system pressure. Air leaking into vacuum space is indicative of a different failure mode.

Thus, air intrusion is a consequence of other failures and its likelihood is already considered in the frequencies of these other events. It is recommended that strict adherence to procedures and attention to detail during and after maintenance be followed, especially purge and evacuation procedures, and that frequent leak inspections be performed.

3.3.4.3 Gas Transfer Events

Transportation events, loss of containment during gas transfer and inadvertent hydrogen venting events have the potential to release hydrogen gas directly to the atmosphere in and around the safe room building. The present system design prevents release within Building 7900 from gas transfer accidents unless other serious failures have occurred simultaneously. Transportation accidents involve damage or destruction of gas-bearing containers during transport or vehicular mishap during gas bottle delivery. (Barriers should be constructed of sufficient strength to prevent vehicular contact with the safe room building.)

Equipment failure or human error during system filling could create a scenario whereby a small quantity of air could enter the expansion volume with the warm hydrogen. Presently, no gas analysis instrumentation exists in this portion of the system to detect this occurrence (a possible risk management option). The exact consequences of this event can be speculated to involve a reaction in the transfer line or expansion volume, depending on the amount of air that is involved. A preexisting seal or valve failure could be the cause of this. However, the fill lines will be evacuated prior to hydrogen intake to ensure no air is present during the transfer. Any existing deficiency in the containment boundary should be evidenced at that time. These conditions and the relatively low pressures involved during gas handling make a loss of gas containment remote.

An inadvertent venting of hydrogen is possible during system filling if erroneous actions are taken by the operators/technicians that violate such existing safety features as low-pressure activated isolation valves on the gas bottles and fill lines. These would be classified as a Type II error of commission¹² with

an estimated likelihood of 4.7×10^{-3} per operation. Thus, the likelihood of a gas-handling event is governed by human errors of commission at a conservatively estimated frequency of

$$\Phi_{\text{GHE}} = 4.7 \times 10^{-3} \text{ per gas-handling operation} \quad (65)$$

Were any of these events to occur, there should not be any reactor safety implications because (1) no such work should be performed during reactor operation and (2) the bottle station is outside the reactor and safe room buildings. Safety features on the gas bottles and in the system fill lines should also limit the amount of gas released from component or seal failure to negligible amounts that quickly dissipate or burn off. However, the inventory fill system should be designed to ensure that no single human error will be sufficient to result in a gross release of hydrogen or to open a direct path that allows air into the system.

3.3.4.4 Hydrogen Boundary Failure Events

Because of the design emphasis on hydrogen containment, there are no single boundary failures that result in a release of hydrogen into the building. The events described below are related to spurious hydrogen boundary integrity failure at normal system operating conditions, not to leaks or breaches caused by another event (e.g., rising system pressure/temperature from a loss-of-boundary cooling). The system response to boundary failures will be in accordance with pressure conditions. The expansion vessel-to-high-pressure storage tank (HPST) transfer pump will continue to inject hydrogen into the HPST in an effort to restore system pressure. The effect will be to force more hydrogen out through the breach. At this time the positive and negative consequences of a shutdown of the inventory control system on detection of abnormal pressure drops on hydrogen in the vacuum system have not been fully evaluated. Therefore, for transient analysis and subsequent discussions later in this chapter, the conservative assumption is made that the inventory control system will continue to try and make up the pressure drop caused by the hydrogen boundary failure. This class of scenarios will be reevaluated when hydrogen inventory control procedures are developed. There should be a pressure set point where hydrogen remaining in the system is vented out through the exhaust header to prevent overpressurization of other barriers.

Breach events are separated into two broad categories—those inside Building 7900 and those occurring outside HFIR. The consequences of a hydrogen release in these two locations differ in the potential damage from fire or detonation and nuclear safety ramifications.

Events inside Building 7900

Events inside the HFIR Building concern the release of hydrogen into the beam room area (G-11), an area usually occupied by several experiment personnel and frequented by operators. Depending on the severity of the event and the mechanism of release, the hazards can include projectiles, fires, explosions, oxygen deficiency resulting in asphyxiation, and freezing. In the current design, the transfer line that passes through the electrical equipment room (G-12) and the primary pony motor battery room (G-13) will be enclosed in a conduit to minimize or eliminate the possibility of releasing hydrogen into these adjoining areas where nuclear safety-related equipment is located. The only portion of the transfer line exposed to the building environment is in the beam room in the vicinity of HB-4. The remainder is in the beam tube extension, a sealed and evacuated/inerted space. The boundary failure frequency estimates for events inside Building 7900 are given in Table 3.5. The unique aspects of each break or leak scenario are summarized below.

Table 3.5. Boundary failure frequency estimates for events inside of Building 7900

Event	Frequency	Main contributor	Frequency category
Single boundary failure	2.0E-2/year	Weld failure	Infrequent
Moderator vessel breach	1.3E-4/year	Weld failure	Infrequent
Catastrophic line break in beam room	3.0E-6/year	Common material flaw	Limiting

Single boundary failure in the transfer line could result from a faulty weld, poor seal between two sections of transfer line, a fitting (curve, u-bend, etc.), or a material failure in the transfer line material. The anticipated system response is that the escaping hydrogen is detected by the gas analyzer(s) in the vacuum system, alerting operators to the event. As the hydrogen expands into this space, the vacuum pumps will draw the gas out into the exhaust header. If the increasing pressure exceeds the capacity of the vacuum pumps, rupture disks upstream of the pump skid should burst to relieve the overpressure to the header. Multiple failures would be required for the event to progress further beyond the vacuum boundary.

Moderator vessel breach concerns all of the hydrogen bearing lines and the vessel contained within the moderator vacuum thimble. Failure could result from an incipient material flaw, butt or circumferential weld failure, or a break in the inlet or outlet transfer line. The immediate consequences of this event are uncertain at best. Ideally, the outcome is similar to the single boundary failure event described above where the vacuum system relief mechanisms mitigate the consequences. Another scenario is that a thimble full of very cold hydrogen gas results in freezing the thimble cooling water jacket, even though the hydrogen pressure is relieved. As the water expands, it could damage the beam tube or moderator thimble sufficiently for hydrogen to escape these boundaries.

If pressure relief efforts fail altogether, pressure will relieve through the path of least resistance, be it back through the transfer line or by blowing out the beam tube window. The worst case, an incredible scenario, would involve thermally induced failure of the moderator thimble, pressurizing the cooling water space and an ejection of the thimble and transfer line out to the beam room. Again this is considered to be an incredible scenario because of the defense-in-depth features of the system, the design of the thimble flange required to support this structure, and the assumed beam shutter arrangement at the beam port in G-11.

A transfer line break in the beam room may occur from an incipient common material flaw in all layers of the co-axial transfer line that is open to the building atmosphere or from an external influence. Breaks resulting from external events such as seismic activity or heavy equipment movement in the beam room are difficult to quantify with existing design information. Overhead crane operation in this area should have physical barriers put in place to eliminate any risk to damaging the transfer line from human error while moving equipment. The transfer line should be protected from potential impact as well. The consequence of this event would be oxygen deficiency, fire, and possibly detonation. A definite air-hydrogen reaction will occur speedily, but the degree of this reaction is left to separate calculations.

Events outside Building 7900

Hydrogen leakage events that occur outside the HFIR Building are of lesser consequence from a nuclear safety standpoint than those that could happen inside the building. Any cold hydrogen outside the HFIR Building has the same triple containment as that inside the building; this includes the transfer lines, pumps, valves, and pressurizer vessel. The safe room is equipped with blow-out panels to relieve the pressure of expanding gas or hydrogen-air reaction. Warm hydrogen gas in the inventory control system is maintained under vacuum containment, including the lines, valves, and the large expansion

tank/storage vessel. Of concern in these scenarios are the effects of a large hydrogen release near a ventilation intake or the force from an explosion of a large amount of the gas next to the building. The estimated frequencies for the events outside of Building 7900 are provided in Table 3.6. The scenarios are similar to those above in type and are as follows.

Table 3.6. Boundary failure frequency estimates for events outside of Building 7900

Event	Frequency	Main contributor	Frequency category
Transfer line—single boundary failure	7.7E-2/year	Seal leakage	Infrequent
Transfer line—complete break (catastrophic)	1.5E-6/year	Common material flaw	Limiting
Heat exchanger leakage (minor failure)	1.3E-2/year	Plate or weld failure in the heat exchanger	Infrequent
Heat exchanger breach (major failure)	7.9E-3/year	Gross material failure in plates or body	Infrequent
Expansion vessel (E.V.) leakage (< 5 g/s)	6.0E-3/year	Flange or valve stem leakage	Infrequent
E.V. catastrophic leakage or rupture (> 800 g/s)	6.0E-6/year	Material failure	Limiting

Single boundary failure could occur in the transfer line or in the inventory fill line, across the main heat exchanger, through a component body rupture, seal leak, or weld failure. Small leaks into the vacuum region will likely be swept into the exhaust header and safely mitigated by warming up the system. Regardless, additional failures are required for hydrogen, whether warm or cold, to escape into the air.

Transfer line break (catastrophic) where all layers of the co-axial tubing are breached could occur inside or outside of the safe room. Current plans are for the transfer line and surrounding pipe to be buried between the reactor building wall and the safe room building. This will provide maximum protection for the transfer line. If the line exits the building above ground, any grounds-keeping done in the area should be by hand or noncorrosive chemicals to ensure that projectiles are not a risk to damaging the line. Otherwise, the contributors to this event are the same as above for inside Building 7900. If the break issues hydrogen directly to the air, the hydrogen should quickly burn off or dissipate. If there is a hydrogen line break into the safe room, ventilation fans and blowout panels exist to promote dilution of the hydrogen cloud and deal with any pressure wave. Fire could also result from the hydrogen-air reaction.

Expansion vessel/storage tank failure is included here because of the relatively large inventory of hydrogen that could be stored close to Building 7900. This large vessel (approximately 15 m³) is a double-walled tank with the interstitial layer either inerted or evacuated. A single barrier leakage should only result in having to purge the system to repair the flaw. The inner vessel and surrounding outer blanket will be equipped with relief devices to deal with a single barrier leak. Failure of those devices is required for the event to progress further. Should a catastrophic rupture of this vessel occur, venting all hydrogen inventory to the immediate area next to Building 7900, the force of the resulting pressure wave from the H₂-air reaction could result in damage to the building structure, secondary coolant line, or safe

room building. Leakages outside the tank are divided into small (< 5 g/s) and catastrophic (approximately 800 g/s) based on available data.

3.3.5 Recommendation of Events for Further Safety Analysis

The events that are recommended for further safety analysis in forming the design basis of the cold source were chosen based on their frequency of occurrence and likely consequence(s) using the methodology presented in Sect. 3.3.4. Maintenance-related events and human error events are not explicitly included in the events recommended for further safety analysis because the actual activities are not clearly defined yet, and the consequences of these events are likely bounded by the events that are included here. The possibility for these events must be considered in formulating post maintenance and preoperational testing procedures, as well as maintenance procedure task lists. As stated earlier, no emergency or standby equipment is factored into the probability that any of these events will occur. The effectiveness of this auxiliary equipment is better evaluated in a consequence analysis or, if appropriate, a deterministic analysis.

In the insufficient boundary cooling (IBC) category, all events are categorized as moderate with the exception of the loss-of-vacuum (LOVA) event. LOVAs, outside of the moderator thimble, are considered to be rather benign events unless accompanied by additional failures in other systems/components. The consequence of all IBC events is bounded by the loss of normal hydrogen circulation scenario ($\Phi = 0.23/\text{year}$), because of the fact that the reactor is still at full power when forced flow stops. The possibility of having UPS backup power to the circulators precludes the loss of off-site power (LOSP) event ($\Phi = 0.44/\text{year}$) from being the bounding moderate event, despite the higher frequency of occurrence, because the presence of circulation will allow a more controlled warming of the hydrogen. It is also recommended that the loss of refrigeration scenario ($\Phi = 0.78/\text{year}$) be further analyzed to determine the consequences to the hydrogen loop from a sudden loss of heat removal. A LOVA in the moderator thimble ($\Phi = 2.5\text{E}-2/\text{year}$) not related to material failure could have system stability implications because of increased heat transfer and is also recommended for further evaluation. Because of the present passive system design, the loss of moderator plug cooling scenario is not presently suggested for further analysis. However, if it is determined that an active system is required to fulfill this function, further consequence analysis will likely be required in the future. No additional events are recommended for analysis from the IBC category.

Pressure increase scenarios involve sudden changes in pressure in excess of normal operating parameters internal or external to the system. Of the two events in this category, one is recommended for further analysis based on the results of this study: loss of moderator thimble integrity ($\Phi = 5.7\text{E}-5/\text{year}$), a limiting event. Consequence evaluation should focus on the ability to contain hydrogen within the loop or to vent it safely outside of the building.

No events from the gas transfer event category are recommended for further evaluation (see earlier discussion in Sect. 3.3.5.3).

Different boundary failures can require different methods for mitigation, from gas/pressure relief in the vacuum layer to high-speed exhaust fan operation in the safe room, in order to prevent a hydrogen-oxygen reaction. Three boundary failure events were estimated to occur with sufficient potential consequences to warrant further analysis: moderator vessel breach ($\Phi = 1.3\text{E}-4/\text{year}$), complete transfer line rupture in the inside Building 7900 ($\Phi = 3.0\text{E}-6/\text{year}$) and rupture of the large expansion vessel ($\Phi = 6.0\text{E}-6/\text{year}$). The transfer line rupture and expansion vessel catastrophic failure are both limiting events, but their potential consequences differ greatly. Also suggested for further evaluation is the single inner boundary failure in the transfer line located in Building 7900 ($\Phi = 2.0\text{E}-2/\text{year}$) to assess the ability of the vacuum, gas analyzer, and venting systems to handle this event.

Based on the methods in Sect. 3.3.4, the estimated event frequencies and their potential consequences, the events discussed above and shown in Table 3.7 below are recommended for further analysis to determine their impact on the safety and stability of the cold source systems and any impact they may impose on reactor operation.

Table 3.7. Recommended events for further analysis

Category	Event	Frequency category
Insufficient boundary cooling	Loss of hydrogen circulation	Moderate
	Loss of refrigeration	Moderate
	Loss of moderator thimble vacuum	Infrequent
Pressure increase	Loss of moderator thimble integrity	Limiting
Gas-transfer events	None recommended, bounded by boundary failure events	
Boundary failure	Moderator vessel breach	Infrequent
	Transfer line rupture in G-11	Limiting
	Expansion vessel rupture	Limiting
	Inner transfer line boundary failure in 7900	Infrequent

Any changes in the design of the cold source systems or components or the acquisition of more recent or applicable reliability data may necessitate or encourage a reevaluation of one or more of the event frequencies listed above. This could result in either a reduction of recommended scenario analyses or an increase, depending on the new information. The conclusions above reflect the latest available design and industry information as of May 15, 1997.

3.4 PRELIMINARY SYSTEMS TRANSIENT ANALYSIS

A preliminary systems transient analysis capability has been developed to support the safety analysis of the HFIR cold source. The analysis capability is based on the Advanced Thermal Hydraulic Energy Network Analyzer (ATHENA)¹³ code (described below) and the model(s) is (are) up-to-date with the current conceptual design of the cold source as of approximately May 1997. As the cold source design evolves, the model(s) will need to be updated periodically in order to properly analyze the scenario of interest. This section describes the ATHENA code, the HFIR cold source model developed for ATHENA, some analysis results, conclusions, and recommendations.

3.4.1 Description of the ATHENA Code

ATHENA is a computer program for simulating thermal-hydraulic systems. ATHENA is essentially the same code as the Reactor Linearized Analysis Program 5 (RELAP5),¹⁴ developed by the Idaho National Engineering and Environmental Laboratory (INEEL) for detailed thermal-hydraulic safety

analysis in the nuclear industry. RELAP5 is likely to be delivered by INEEL bundled with the Severe Core Damage Analysis Program (SCDAP), in which case it is compiled separately to become RELAP5. RELAP5 is limited to light water as the primary fluid type being analyzed, whereas ATHENA allows for additional fluids. The version of ATHENA utilized herein is a subset of SCDAP (version 3.1-release D) and allows for and includes light water, heavy water, hydrogen, lithium, potassium, helium, nitrogen, sodium, sodium-potassium, lithium-lead, and ammonia (fluid types 1 through 11, respectively), as possible simultaneous working fluids to be analyzed. In the case of the HFIR cold source, independent flow paths of hydrogen, helium, and light water are simultaneously modeled in the ATHENA model (discussed below). Furthermore, the capability exists to include helium (or other gases as necessary) as a noncondensable gas mixture with the other primary fluids (i.e., light water, etc.) should that become a desired subsystem to analyze.

ATHENA essentially provides a one-dimensional (1-D), time-dependent solution to the coupled nonequilibrium thermodynamic two-fluid (phase) conservation law equation system of fluid mass, momentum, and energy, coupled with the energy equation applied to a 1-D volume of metal (i.e., heat conduction equation applied to "heat structures." Some multidimensional extensions are available where a single dimensional representation is not adequate. For example, a "reflood" option is available for the heat structures that allows for some two-dimensional (2-D) simulation capability. A pseudo 3-D fluid model is available for certain limited applications. Pseudo 2- and 3-D is obtained via the pipe component and cylindrical heat structures for the case of a stacked set of fluid/structure nodes as is often required (e.g., a nuclear reactor core or pressurizer). The present HFIR cold source ATHENA model utilizes the reflood option within a portion of the model to simulate a required 2-D heat transfer component. For all the partial multidimensional capability built into the code, it must be emphasized that, fundamentally, ATHENA is a 1-D code. Any extensions thereof are, at best, incomplete approximations to a true multidimensional problem. Nevertheless, as a thermal-hydraulic systems simulator, including control system integration, ATHENA is very good and is considered the best available. The RELAP5 companion code has been developed by the U.S. Nuclear Regulatory Commission (NRC) and others such as DOE and is widely used worldwide for nuclear safety analysis. The validation database against light water is extensive. In contrast, the validation data for the extension ATHENA-fluids is essentially nil, even though a large amount of data are presented in the code documentation for nonwater fluids. As will be discussed elsewhere in this report, the current plans are to validate the ATHENA HFIR cold source model to the extent possible against data taken from the cold source experiments performed at the AEDC.

The unique feature within both ATHENA and RELAP5, that allows for multiple fluids to be simulated in ATHENA, is the external generation of the "root" fluid properties of pressure, temperature, specific volume (density), internal energy, thermal expansion, isothermal compression, specific heat, and entropy. These properties are generated externally via a user-input grid of pressure-temperature value pairs by a stand-alone computer code designed for the purpose of fluid property table generation. The tables (or table in the case of RELAP5) are then read in by ATHENA one time during code initialization. Thereafter, during a system simulation, all other fluid properties, including "nonroot" properties such as flow losses, heat transfer coefficients, and other constitutive relationships, are generated ultimately via interpolations from this table. The accuracy of the interpolation is determined by the resolution of the input grid of pressure-temperature pairs, the accuracy of the fluid-property governing equations of thermodynamic state, and the machine accuracy. This approach has proven to be successful on this project except in regions of fluid-property singularity such as the thermodynamic critical point of hydrogen. A code change is required in order to remove the singularity point (discussed further below).

In all other respects, ATHENA operates identically to RELAP5. Provided the user is familiar with generic fluid flow and heat transfer issues (especially those issues associated with the cryogenic fluids of the HFIR cold source), an experienced RELAP5 user will have no trouble adapting to the slightly-different ATHENA code. The ATHENA input stream is nearly identical to the RELAP5 input stream.

The HFIR cold source ATHENA input stream is discussed in the next section. Some familiarity with RELAP5 nomenclature is necessary to understand fully all aspects of the input.

3.4.2 Description of the ATHENA HFIR Cold Source Model

An ATHENA or RELAP5 model is made up of four basic components: (1) volume, (2) junction, (3) heat structure, and (4) control system. These four basic components are necessary to describe "state" variables solved by the code at each time-step of the solution: (1) from the six-equation hydrodynamics equations, six independent variables are defined (or chosen): total fluid pressure, vapor void fraction, vapor internal energy, liquid internal energy, vapor velocity, and liquid velocity; (2) from the heat conduction equation, the metal temperature distribution; and (3) the set of control variables are solved from their equations. Several contributing variables are also solved that can feed directly into the state variables that optionally include noncondensable gas mass quality, reactor power from point kinetics, pump dynamics equations, valve dynamics, and other "process" models. Further, several input options such as general data tables, heat structure material property tables, etc., are available in addition to the aforementioned externally-generated fluid property tables. The code includes the ability to internally generate complex constitutive relationships for friction losses and heat transfer coefficients as well as include many of these quantities by input. This allows for a model to be built of very complex systems, such as the HFIR cold source, by defining individual "blocks" of volume, junction, heat structure, and control-system components and connecting them together in a logical manner. This logical connection of individual components is described next.

3.4.2.1 Introduction and Overview of ATHENA Model

A nodding diagram of the total HFIR cold source ATHENA model is shown in Fig. 3.6. The model depicted by the nodding diagram, as in the real system, is best described by the individual components making up the total system. Since the overall nodding diagram shown by Fig. 3.6 is too detailed to view accurately, each component description below will include a magnified view of that component nodding.

Starting at the beam tube transfer lines return leg coming into the reactor vessel, the HD hydrogen flows into the lower-elevation entrance to the moderator vessel. The flow within the moderator vessel is quite complex and is being modeled in detail separately utilizing computational fluid dynamics (CFD) technology. This complex flow geometry is further approximated by the 1-D representation of ATHENA, but conserving the overall mass, momentum, and energy balances of the moderator.

The flow leaves the moderator vessel through the beam tube transfer line flow leg and combines (preserving separation boundaries) with the return leg into the concentric transfer lines. The concentric transfer lines then carry and return the hydrogen (preserving vacuum separation boundaries) from the reactor vessel walls to the safe room building outside the HFIR confinement building.

The flow lines then split off separately, forming a secondary loop within the safe room. The normally warm (entrance) side of the loop first interfaces with the pressurizer vessel. The primary purpose of the pressurizer vessel is to provide a stratified temperature and fluid density interface between the cold loop and the ambient-temperature hydrogen gas feed and bleed tanks. This type of interface arrangement will enhance the pressure stability and allow for pressure control.

After passing the pressurizer vessel, the flow continues on to the circulator or pump station. The circulator station contains three parallel flow paths, each with isolation valves set to be properly configured for the desired operation. Two HD circulators are available for "normal" full-power operation. Only a single HD circulator is operated at a time, with the second increasing the availability of the cold source system during normal operation. A single LD circulator is available for standby operation when either the cold source is not required or the reactor is in a shutdown mode. The circulator station is shown with isolation valves on both sides of the circulators. The hydrogen loop experiments at AEDC

HFR Cold Neutron Source
Thermal-Hydraulic Systems Modeling
Hydrogen-Loop Noding Diagram
ATHENA-RELAP5

nomenclature:
PV pipe volume
HS heat structure
SJ single junction
AN annulus volume
BV branch volume
SV separator volume
TDV tank dead volume
SV separator volume
SV separator volume

Fig. 3.6. Overall nodding diagram of the HFR cold source ATHENA model.

include only a single isolation valve downstream of each circulator, whereas the present HFIR design includes isolation valves on both the upstream and downstream sides of the circulators. This design feature enhances maintenance of the circulators. Furthermore, the ATHENA simulations have shown that an isolation valve on both sides of the circulator will enhance system stability (not yet clear if it is a numerical or a real stability enhancement).

After completing the flow through the circulator station, the hydrogen fluid in normal operation continues on through the heat exchanger. The hydrogen is cooled by a gaseous helium refrigeration loop on the secondary side of the heat exchanger. It is assumed that appropriately placed inlet and outlet boundary conditions on the helium are sufficient for the refrigerator loop simulation. More detail can be added to the secondary helium side at a later time should it be necessary.

The cooler hydrogen fluid continues through the loop and returns into the inner portion of the concentric tubes transfer line. The transfer lines provide the final return flow path for the hydrogen back into the beam tube of the reactor vessel.

Finally, a "feed-bleed" subsystem exists that provides for pressure control of the loop and fill/purging of the hydrogen inventory. This subsystem is made up of a feed tank, bleed tank, connecting transfer lines, and several valves for control and isolation.

The modeling of each individual component group will now be described in more detail.

3.4.2.2 Component and Subsystem Descriptions

This section describes the details of each component group or subsystem within the total HFIR cold source ATHENA model. Several generic "rules of thumb" were applied as modeling techniques or input options across all the models and are described here.

Geometry

A nominal length of 1 m was arbitrarily chosen as the default volume length to be used anywhere in the model. The flow length of some volumes of the model, such as the long transfer lines from the reactor to the outside safe room (volumes 10 and 12), were increased to as high as 8 m to reduce the number of volumes while maintaining accuracy. As a result, length to diameter ratios as high as 79 to 500 exist in parts of the model.

Another criterion imposed is to restrict the ratio of adjacent volumes to no greater than a factor of 2. Some analysts in the RELAP5 community allow this ratio to be as high as 10. However, based on the experience here, a factor of 2 on this ratio has been found to reduce the likelihood of code inaccuracies.

Three regions of the model require sufficient detail (number of volumes) in order to achieve the desired accuracy in these regions: (1) the moderator vessel, (2) the HD and LD circulators, and (3) the pressurizer. The volume noding in these regions, combined with the adjacent volume ratio and nominal volume length yield the noding density shown by Fig. 3.6.

Wall roughness

At the time of the current model development, the actual wall roughness of all the transfer lines and other major components is not known. The nominal design volumetric flow rate is 1.0 L/s at the rated hydrogen fluid conditions of 15 bar pressure and 20 K temperature. Using the head-flow performance curves of the HD circulator manufacturer, an average wall roughness of 15 μm was chosen throughout the model to yield the desired design values of 1.0 L/s at about 1.0 bar of circulator head (or total loop pressure drop). This assumed quantity of wall roughness will be modified as more information is known about the exact transfer line losses, form losses, and elevation changes (i.e., all details of the piping layout). No additional form losses from elbows, tees, etc., are known or included in the model at this time.

Volume control flags

Unless stated differently in the individual component descriptions below, the volume control flags, *tlpvbfe*, are set to their default values (*tlpvbfe*=0000000). This has the following effects, respectively: (1) thermal front tracking is not to be used; (2) mixture level tracking is not to be used; (3) water packing scheme is to be used; (4) vertical stratification model is to be used; (5) pipe interphase friction is to be applied; (6) wall friction effects are to be computed; and (7) nonequilibrium thermodynamic effects are to be used. Nonequilibrium thermodynamic conditions can exist even in this supercritical loop under the right conditions (e.g., the pressurizer). Further, the ATHENA code is generalized such that should nonequilibrium effects disappear, the fluid is properly treated automatically. In addition, the ATHENA code developers do not recommend running the code without this flag enabled (legacy).

Junction control flags

Unless stated differently in the individual component descriptions below, the junction control flags, *efvcahs*, are set to their default values (*efvcahs*=0000000). This has the following effects, respectively: (1) modified pipe volume (PV) term in the energy equations will not be used; (2) CCFL model will not be applied; (3) horizontal stratification will not be used; (4) choking model will be applied; (5) a smooth area change or no area change is used; and (6) two-equation (nonhomogeneous) momentum equation is used; and (7) momentum flux is used in both directions (to and from) of adjacent volumes.

Hydrodynamic initial conditions

ATHENA utilizes a time-relaxation solution technique to achieve a steady-state initial condition. In addition, because of the complexities of all the variable possibilities that the user can specify on input, it has been found from experience that precise specification of initial conditions is not necessary. Provided the initial conditions are specified reasonably close to the steady-state condition desired, and the chosen time-step is sufficiently small for the solution technique used, it is best to allow the code to compute the precise steady-state and then restart from that point. Therefore, unless otherwise stated in the individual component descriptions below, all the volume initial conditions are set to 14 bar pressure and 17 K temperature for the 15 bar nominal (supercritical) operating point. (As will be noted in Sect. 3.4.3, some early analyses were performed for a backup liquid hydrogen system operating at 4 bar pressure.) For the 4 bar nominal (subcooled liquid) condition, the volume initial conditions are 3 bar pressure and 17 K temperature. Furthermore, unless otherwise stated, the junction initial conditions are input as 1.0 L/s for normally flowing paths or zero for nonflowing paths (such as the pressurizer) at the fluid volume initial thermodynamic conditions.

Generic heat structure parameters

Unless otherwise stated below, all the heat structures of the model use the following parameters:

1. To keep the run time to a minimum, the number of grid points within a heat structure is three. This implicitly assumes that very small temperature gradients exist across the 1-D heat structures.
2. Accurate initial temperatures are not necessary since a separate steady-state calculation is specified for all the heat structures.
3. Except for the moderator and beam tube return and flow leg heat structures, all internal power generation is zero.
4. Except for the copper rod within the pressurizer component, all heat structure materials are assumed aluminum. The specific aluminum alloy thermal properties are not precisely known at this writing; therefore, the data for 2024-T4 alloy was assumed¹⁵ at low temperatures, and pure aluminum was

assumed at higher temperatures. This is a conservative approach with respect to maximum steady-state surface temperatures adjacent to the hydrogen fluid. This conservative assumption will likely be relaxed during the detailed design stage when properties for aluminum alloys such as 6061-T6 will most likely be used.

5. At all convection heat transfer boundaries not explicitly specified, the heat transfer coefficient is computed by the internal ATHENA heat transfer package.
6. Except in the copper rod heat structure of the pressurizer component, the "reflood" option is not utilized. Among other things, the reflood option allows for a full 2-D representation of the heat structure.
7. The companion boundary volume hydraulic diameter is used for the heat transfer hydraulic diameter.
8. The heated length option (both forward and reverse) is ignored by a large input value (important for critical heat flux).
9. The grid spacer length and grid loss coefficients (forward and reverse) are not used (important for critical heat flux).
10. The local heat flux to average heat flux ratio from the start of boiling (local boiling factor) is set to unity. This is a good assumption for zero or flat power profiles or regions with continuous subcooled liquid.

Moderator vessel

The moderator vessel fluid is modeled by ATHENA pipe volume #3 as shown by Fig. 3.7. The first five and last five volumes represent the normal entrance and exit regions, respectively. The middle eight volumes represent the "dome" of the spherical-shaped region of most importance for slowing of neutrons (i.e., the cold source). The outside and inside metal surfaces are represented by two layers of heat structures (#32-1..5 for the outside entrance surface; #32-6..10 for the outside exit surface; #31-1..8 for the outside dome surface; #133-1..5 for the inside entrance surface; #133-6..10 for the inside exit surface; and #131-1..8 for the inside dome surface).

An independent modeling subproject, utilizing the CFD code CFDX-FLOW3D,¹⁶ has been performed and is described elsewhere in this report. An interface system has been written for the CFD model that generates an "equivalent" (to the extent possible) ATHENA model of the moderator vessel. The interface system is made up of a set of user-supplied FORTRAN subroutines within CFDX-FLOW3D and an external MATH-CAD script. The present ATHENA model is consistent with the CFD interface produced by the Bucholz nuclear heating data as of 8/16/96¹⁷ and the Williams' CFD data as presented in Sect. 2.5 of this report. Any design changes beyond these time stamps will require an update of this interface output.

Following the nodding diagram as presented by Fig. 3.7, the CFD interface system produces consistent values for the moderator vessel hydrodynamic components (following the pipe volume nomenclature from the ATHENA input requirements) of: (1) volume flow area, (2) junction flow area, (3) volume, (4) vertical orientation angles, (5) elevation changes, and (6) hydraulic diameters for all 18 volumes shown. Furthermore, the CFD interface produces consistent values for the moderator vessel heat structures of: (1) thickness, (2) surface area, (3) nuclear heating fraction, and (4) heated hydraulic diameter. With this CFD interface system, the overall fluid volume, pressure drop, and energy transfer is conserved. However, because of limitations within the 1-D framework of ATHENA, which causes the velocity distribution to be averaged, the heat transfer coefficient is not estimated correctly as will be discussed later.

Pressurizer vessel

The pressurizer vessel fluid is modeled by PV #60 as shown by Fig. 3.8. Attached to the left side of PV-60 is heat structure (HS) #160, which represents the copper rod level control device. The top heat

Fig. 3.7. ATHENA noding diagram of the moderator vessel for the HFIR cold source.

Fig. 3.8. Pressurizer vessel noding diagram of the HFIR cold source ATHENA model.

structure node (-10) is coupled to single volume (SV) #64 and further coupled to HS-64. The HS-64 and SV-64 combination is a reflood-option workaround to allow for a fixed temperature (Dirichlet constraint) representing the ambient temperature outside the vacuum boundary (fixed-temperature boundaries are not allowed for the reflood option). Time-dependent junction (TDJ) #63 is only present as a dummy zero-flow junction to satisfy ATHENA input requirements that all volumes must have at least one junction. Thus, the combination of HS-64, SV-64, TDJ-63, and HS-160 provide the 2-D (via reflood option) representation for the copper rod with the ambient temperature boundary applied at the top node. The coupling at the bottom node (HS-160-1) exists at the second fluid node (PV-60-2), thus defining the liquid/vapor interface for a 4-bar operating point or the HD/LD interface for the 15-bar operating point.

Attached on the right side of PV-60 is HS-60, which represents the pressurizer side wall. The aluminum side wall is not an active system like the copper rod but is necessary to model in order to simulate its energy storage capacity. It is not clear at this writing how important axial conduction along the side wall will be. If necessary, the side wall could also utilize the reflood option to allow for a 2-D solution and achieve the axial conduction.

Valve junction (VJ) #623 is also shown in Fig. 3.8, which is part of the pressure control system included in the feed/bleed subsystem. At this writing, all details regarding the pressure control system design are not complete; hence, a "perfect" pressure control is provided by the combination of time-dependent volume (TDV) #62 and junction #61.

The interface between the pressurizer and main loop is a small pipe and is represented in the model by SV-58 and junctions SJ-59 and BJ-22-3 (branch junction).

The final design of the pressurizer geometry is not known at this writing. In lieu of a design drawing, the pressurizer is assumed to be 40 cm high and a total volume of 15 L. The initial temperature is assumed to be 17 K for the bottom node and 300 K for the remaining nodes. The initial flow is assumed to be zero throughout. The copper rod is assumed to be 1 cm in radius, while length and surface areas are defined by the pressurizer pipe volume lengths, all nodes initially being at a temperature of 300 K. The aluminum pressurizer side wall is also assumed to be 1 cm thick with the length and surface areas defined by the pressurizer fluid volume geometry.

The steady-state solution within the pressurizer portion of the model produces a stratified temperature (and hence, density) along the axial direction with pressure changes only due to static head. The dynamic solution is highly dependent on the final control system design (which is not included at this time) and is anticipated to be very important for correct loop dynamic simulation.

Circulator station

The circulator station is depicted as an ATHENA-model nodding diagram shown by Fig. 3.9. Currently, two HD circulators (named in the model as pump-a and pump-b) are modeled (main and standby) for normal operation and a single LD circulator (named pump-c) will be used during standby operation. Each circulator is placed along a separate parallel path that combines at the entrance and exit of the circulator station. The design drawings of the circulators, showing actual geometry, have not been provided. Therefore a small volume has been assumed with a length of 0.2 m and a flow area of 0.0002 m^2 . The adjacent volumes are also assumed small, with flow area the same as the majority of the system piping [5/8 in. inside diameter (ID)] and a length of 0.1 m. These small volumes are necessary for circulator simulation accuracy and influence the model nodding leading up to and away from the circulator station.

All the isolation valves are assumed to be 5/8 in. ID nominal with an opening/closing time of 2.0 s. These valves are modeled as an ATHENA motor valve type with an abrupt area change option (as required by the code).

Fig. 3.9. Noding diagram of the circulator station subsystem of the HFIR cold source ATHENA model.

Perhaps the most important input requirement for the circulator station is the performance characteristics of the HD and LD circulators. Most, but not all, of this information was provided by the manufacturer. Some of this information could be updated later as better data is obtained. A discussion of this important data follows.

The main and standby HD circulator rated velocity (rotational speed) is 7080 rpm (741.42 rad/s). The rated volumetric flow rate is 1.0 L/s (.001 m³/s) yielding a rated head from the head-flow curve equal to 150.82 m. The rated fluid density is 74.464 kg/m³. The rated torque was assumed equal to the power taken at product testing with air (voltage-current product) divided by the rotational speed yielding 0.427 N-m. From the rated torque, the moment of inertia is assumed equal to the product of pump coastdown time, and rated torque is divided by the rotational speed, yielding 0.00864 kg/m². The coastdown time was given by the manufacturer as 15 s linear. A set of homologous pump curves were derived from the manufacturer head-flow curves in the standard manner. The details of this derivation are not included here for the sake of brevity.

The LD circulator rated velocity (rotational speed) is 8575 rpm. The rated volumetric flow rate is 4.0 L/s (.004 m³/s), yielding a rated head from the head-flow curve equal to 521.82 m. The rated fluid density is 74.464 kg/m³. The rated torque was assumed equal to the rated horsepower-efficient product divided by the rotational speed, yielding 0.274 N-m. The moment of inertia was computed based on assumed sum of individual component masses quoted verbally by the manufacturer yielding 0.00864 kg/m². The coastdown time was given by the manufacturer as 15 s linear. A set of homologous pump curves were derived from the manufacturer head-flow curves in the standard manner. The details of this derivation are not included here for the sake of brevity.

The steady-state performance of the circulator station yielded expected results (i.e., the head-flow curve in the first quadrant of the homologous curves yielded the correct pressure drop around the loop at 1.0 L/s). The dynamic performance, particularly during periods of reverse flow through the circulators, may need some further investigation. The lack of confidence during reverse flow is due to the lack of reliable head-flow data from the manufacturer under these conditions. This could become important, depending on the results of the safety analyses.

Heat exchanger

The heat exchanger (HEX) is depicted as an ATHENA-model noding diagram shown by Fig. 3.10. The hydrogen side of the HEX is shown as PV-70, while the helium is shown as PV-72. The HEX tubes are modeled as HS-70. The counter-flow HEX is modeled by attaching the hydrogen flow on the left side of the HS with minimum node numbering on the inlet PV and HS, and attaching the helium flow on the right side of the HS with PV maximum noding coupled with the minimum HS node numbering. Currently, the helium secondary-side refrigeration loop is limited to the HEX with TDV pressure-pressure boundary conditions. The hydrogen primary-side heated loop is bounded by SJ-65 on the inlet coming from the circulator station header PV-50 and by SJ-75 on the HEX outlet feeding into the return transfer line PV-80.

Currently, the HEX design has not been finalized. In lieu of the final design, a "perfect" heat exchanger was assumed that essentially approximates the HEX geometry and holds the secondary-side helium temperature constant at the set point. The assumed geometry is based on verbal communication with the lead design engineer. This resulted in a facial front of 8 in. $\times$ 3 in. with 10 metal plates of 1/32 in. thickness each. The flow area and surface area is assumed equally split between the "shell" and "tube" sides. The total length is assumed to be approximately 3 ft.

In addition, as a proof-of-principal, several stand-alone HEX models were tested and showed the ability of the ATHENA code to include both hydrogen and helium simultaneously. Further, these stand-alone models were utilized to help in evaluating HEX alternatives for the AEDC experiments.

Concentric transfer lines

The concentric transfer lines are depicted by the noding diagram shown in Fig. 3.11. The normally-warmer hydrogen feeds from the moderator flow leg into the inside tube represented by PV-10 and into the single-tube split of PV-20 just before the pressurizer. The normally colder hydrogen feeds from the heat exchanger return represented by PV-80 into the outside tube represented by annulus volume (AN) #12. The thermal lag associated with the inside concentric tube is modeled by HS-10. Similarly, the important effect of heat capacity associated with the thermal lag of the intermediate and outside concentric tube is modeled by HS-12. Convection boundary conditions are utilized on these heat structures on one side facing the corresponding hydrogen fluid. On the other side of these heat structures, an adiabatic boundary condition is assumed representing the vacuum region physically present between the concentric tubes.

Fig. 3.10. ATHENA noding diagram of the HFIR cold source heat exchanger.

Fig. 3.11. Noding diagram of the concentric transfer lines subsystem of the HFIR cold source ATHENA model.

The inside tube (PV-10) is assumed to be a 5/8 in. ID tube. The first three subvolumes (PV-10-1, PV-10-2, PV-10-3) are vertically oriented with lengths of 0.436 m, 0.5 m, and 0.436 m, respectively. The lengths of the remaining horizontally-oriented subvolumes 4–15 are 0.299, 0.5, 1.0, 1.0, 2.0, 4.0, 8.0, 4.0, 2.0, 1.0, 1.0, and 0.5 m, respectively. The outside tube (AN-12) is assumed to have an outside diameter of the annulus of 3/4 in. The lengths and orientation are similar to PV-10, but the numbering is ordered in reverse since the flow is in the opposite direction.

At the time of this writing, the design and specification of the concentric tubes is not finalized. Hence, the heat structures #10 and #12 are not present in the model (i.e., the thermal lag is assumed zero). This part of the model will be added later as necessary. Very little heat generation is found in the transfer lines away from the reactor. Hence, omission of the heat generation in these heat structures is a minor effect. Further, the heat capacity of the transfer lines is less important for short-term transients, but should long-term transients become important for analysis (such as a start-up or long-term decay heat removal), it will be necessary to include these heat structures (#10 and #12). In addition, the model fluid volumes will require a rework to match the actual transfer line geometry.

Feed and bleed subsystem

An important subsystem of the HFIR cold source system is the “feed and bleed” (FAB) subsystem depicted as an ATHENA noding diagram in Fig. 3.12. The purpose of this subsystem is twofold: (1) to

Fig. 3.12. The noding diagram of the ATHENA model of the feed and bleed subsystem for the HFIR cold source.

enable the hydrogen pressure in the loop to be controlled and (2) to purge the loop of high-density fluid with low-density fluid in the event this becomes necessary. The bleed tank (SV-620) is designed with enough capacity to receive the entire loop inventory of hydrogen at the time of peak mass inventory (supercritical, high-density conditions of normal operation). The feed tank (SV-610) is available to replenish the hydrogen inventory in a controlled manner. It is anticipated that additional valving and tanks may be needed on the inlet side of the feed vessel, in which case the feed vessel will become an intermediary inventory system. For example, additional model components may be required to include a hydrogen gas recycling loop that also includes a charging pump and associated valves, tanks, and a control system.

The control valves shown as VJ-611, VJ-613, VJ-621, VJ-630, and VJ-623 (see also Fig. 3.6), along with connecting transfer lines (depicted by BV-612 and BV-622), are combined with the feed and bleed tanks and a control system to complete the system. During normal operation, the purge valve (VJ-613) is fully closed and isolation valves VJ-630 and VJ-623 are fully open. The control system would then open and close valves VJ-621 (bleed) and VJ-611 (feed) in such a manner as to keep the loop pressure essentially constant at the operating point. The system will be designed such that the pressure control system will respond sufficiently to any anticipated disturbances (i.e., smoothly return the pressure back to the set point).

In the event that the pressure cannot be controlled normally as a result of a loop disturbance, or it is desired to purge the HD fluid for some other reason, then the FAB subsystem can enter into the purge

mode of operation. In this mode, isolation valve VJ-630 is fully closed, and VJ-613 and VJ-623 are fully open. Control valves VJ-611 and VJ-621 open in a controlled manner so that essentially the entire inventory of HD hydrogen is dumped into the bleed tank (SV-620) and is replaced by LD hydrogen from the feed tank.

At this writing, the control logic design has not been finalized for the FAB subsystem. A proof-of-principal calculation is included in this section of the report (see transient results: transition from normal to standby mode of operation at 4 bar). However, during the time it is enabled, pressure control is modeled as a perfect control system via TDV-62 as discussed earlier (Fig. 3.8). As the design evolves, it is expected that the model of the FAB subsystem will also change significantly.

Control system

The ATHENA HFIR cold source model control system variables are rather simple at this stage in the model development. Several control system variables are defined purely for output purposes and have no impact on the calculated results, including the volumetric flow rate into the moderator (L/s), the enthalpy of the inlet and outlet fluid flow of the moderator (J/kg), the flow energy in and out of the moderator (W), the moderator inlet and outlet flow energy difference (W), the power input from the copper rod of the pressurizer (W), and several control system set points needed by control systems (constants). In addition, an attempt was made to model the pressure control system, but this is not complete at this time. As the system design and model evolves, this portion of the model will correspondingly increase in size and complexity.

3.4.2.3 Problem Areas

The ATHENA code and HFIR cold source model are the best known available engineering tools for the intended purpose of performing safety and design analysis. The history of the RELAP5 code has consistently demonstrated that tremendous insight can be gained by performing the thermal-hydraulic systems safety analysis on plant operations and accident conditions. However, the ATHENA form of RELAP5 does not have an equivalent history. In addition to the lack of validation data, there have been a few other problems that have arisen thus far into the project.

The steady-state wall surface temperature predicted by ATHENA within the moderator vessel is greater than the more detailed CFD predictions by the CFDX-FLOW3D code. A separate effects study has shown that the difference can be attributed to the heat transfer coefficient. Although a heat transfer coefficient is not required by the CFD code since the wall conditions are more precisely approximated, an equivalent heat transfer coefficient was computed as an output variable for comparison purposes with ATHENA. The difference in heat transfer coefficient can be attributed to the importance of multidimensional (3-D) flow and heat transfer effects of the moderator vessel. In particular, the CFD code predicts a region of recirculation flow in the moderator vessel that in turn leads to relatively high velocities near the inner and outer wall (yet still above the boundary layer adjacent to the wall) of the cap region of the moderator vessel. The average net flow through the moderator is consistent between the two codes (ATHENA and CFDX-FLOW3D), but because ATHENA is 1-D, only the average free stream velocity is computed in the cap region of the moderator vessel. This in turn causes ATHENA to predict a smaller-than-reality heat transfer coefficient, and hence, greater surface temperature. Therefore, despite consistent geometry and global conservation laws, the 3-D effects are quite important for the moderator vessel. Currently, the ATHENA predictions are conservative with respect to maximum surface temperatures. This problem will be aggravated during transient analyses and may predict boiling in the liquid hydrogen scenario when it does not actually occur. Indeed, the current steady-state ATHENA predictions show that about 3% void fraction is generated in the moderator for a 4-bar operational set point due to subcooled nucleate boiling. No action is currently planned to modify the ATHENA predictions because of this problem.

The fluid properties of hydrogen and helium are computed externally from the ATHENA code and then read in later as a permanent database during the calculation. This process works quite well, and very precise estimates of the fluid properties are generated by refining the input of pressure-temperature value pairs. However, hydrogen experiences some severe fluid property spatial gradients (even at steady-state conditions) in some portions of the HFIR cold source (in particular, the pressurizer). Many local optima exist for all the properties, but the critical point is the most severe. In particular, the heat capacity at constant pressure (C_p) is infinite at the critical point. Other properties, such as the speed of sound, are computed from local derivatives of the root properties. There have been many instances during the ATHENA simulations where the code would reach an out-of-bounds point on a fluid property (the speed of sound in particular) as regions of the model approach the critical point. It is anticipated at this writing that a "code fix" will be necessary to correct the problem since reliable simulations are not obtainable.

The pressurizer contains a copper rod that stratifies the hydrogen temperature across the height of the vessel. This component requires a 2-D representation, via the reflood option of the heat structure component, in order to simulate the heat transfer accurately. The reflood option of RELAP5, and thereby ATHENA, has a history of troublesome application, primarily from the numerical instabilities and increased computational times due to required small time steps. In addition, it is possible that the side walls of the pressurizer may also require the reflood option to be properly simulated. An increased instability of the HFIR cold source model has been directly attributed to this portion of the pressurizer (heat structures using the reflood option). At this writing, efforts are under way to obtain an updated version of ATHENA that may contain an improved reflood model. It is not anticipated that this would be a "show-stopper."

The circulators are modeled using the pump component of ATHENA. Currently, no pump performance data is available for reverse-flow situations through the circulators. The model currently predicts reverse flow in many circumstances. Since the circulator flow lines, as well as the remainder of the system, do not physically include check valves, this is probably a very real situation (reverse flow can exist).

The secondary side of the heat exchanger is cooled by helium. The thermal and transport properties of gaseous helium, even at low temperatures, result in less stable numerical calculations. Hence, a smaller time-step than would otherwise be needed for an equivalent hydrogen flow channel is required. It may be necessary to exclude the pure helium flow path in order to produce quicker simulations.

The thermal and transport fronts, brought on by purging of the low-temperature hydrogen loop with room temperature hydrogen, causes severe numerical instabilities during the simulation. The only solution to this problem is to use very small time-steps; hence, long computational times are required for the solution.

3.4.3 Analysis Results

Although the present reference design has only one normal operating mode (15-bar supercritical hydrogen mode), a liquid hydrogen mode at 4 bar was considered earlier in the design as a second normal operating mode. Therefore two separate modes of operation have been analyzed. For 4-bar operation, the majority of the loop is subcooled, and the hydrogen will be stratified in the pressurizer such that two-phase fluid exists. For 15-bar operation, the entire loop and pressurizer system is supercritical; hence, no interaction between phases exists. However, as will be shown below, significant fluid property variations do exist in the pressurizer at 15-bar operation.

3.4.3.1 Results of the Two-Phase System at 4-Bar Operation

The analysis results of subcritical (two-phase in the pressurizer) operation at 4 bar are subdivided into steady-state and transient sections. The only transient analyzed at 4 bar is the transition from normal

to standby mode. As previously noted, there are no longer any plans to operate in this state. However, these results are presented for the sake of documentation.

Steady-state results

The steady-state calculation for the nominal 4-bar model requires a rather small time-step throughout the simulation primarily because of the two-phase interaction in the pressurizer. The calculation was executed for the first second of simulation time with a time-step of 1.0×10^{-5} s, and extended to 10 s with a larger time-step of 5×10^{-5} s, then 20 s with a time-step of 1×10^{-4} s, and finally to steady-state (in this case 300.0 s) with a maximum time-step of 2×10^{-4} s. Judging from the minor-edit outputs, by this time it appeared that the system was sufficiently steady to conclude a steady-state had been reached.

The results show a maximum steady-state pressure in the system just downstream of the HD circulator of 4.002 bar. The corresponding minimum pressure at the HD circulator inlet is 2.967 bar for a total differential pressure around the loop of 1.035 bar at a volumetric flow into the moderator vessel of 0.958 L/s (recall the loop losses have been tuned to yield approximately 1-bar pressure drop at 1 L/s flow).

The minimum temperature in the hydrogen loop is just downstream of the heat exchanger outlet at exactly 17 K (recall the perfect heat exchanger). The maximum temperature in the loop, at the heat exchanger inlet, is 20.95 K. The fluid temperatures at the moderator inlet and outlet are 17.08 K and 20.47 K, respectively. The additional temperature rise between the heat exchanger outlet and the moderator inlet (0.08 K) and the moderator outlet and heat exchanger inlet (0.48 K) can be attributed to the frictional losses in the transfer lines, the work done by the HD circulator (0.35 K by the circulator alone), and the energy deposited into the loop from the copper rod of the pressurizer. The total power deposited into the moderator heat structures from nuclear heating is about 1524.5 W. The corresponding total power deposited into the hydrogen fluid adjacent to the moderator heat structures from both direct nuclear heating and convection heat transfer is about 2155.3 W (hence, about 630.8 W is deposited directly into the fluid by nuclear heating). This agrees well with the control variable computed flow energy (CV-28) of 2098.5 W. It should be noted that a new physics analysis (see Sect. 2.4.3) has indicated nuclear heat loads that are about 7% lower than those reported here and used in the transient analyses discussed in this section.

Perhaps the most challenging part of the model development has been the pressurizer. The steady-state results reveal why this is the case. The copper rod (HS-160) shows an axial temperature profile starting at 243.9 K at the top node (HS-160-10) and dropping to about 119.5 K at the bottom node, which interfaces with the second fluid node up within the pressurizer (PV-60-2). The top node is external to the vessel and interfaces with a heat structure and single volume combination that allows a Dirichlet constraint (constant temperature) boundary condition of 300 K. The temperature profile in the radial direction of the copper rod is essentially flat throughout at steady-state since very little energy is deposited into the fluid except during level and/or pressure disruptions. The fluid temperatures of the hydrogen volumes adjacent to the copper rod (PV-60-2 through PV-60-10) are considerably less than the surface temperatures of the copper rod. The volume just below the copper rod is at a temperature of 25.9 K, which is several degrees above the nominal loop temperature at that point (20.6 K) and significantly less than the fluid nodes above (~ 35, 57, 85, 111, 136, 163, 190, 218, and 246 K for PV-60-2 through PV-60-10, respectively). The fluid is in two-phase thermal nonequilibrium. Most of the pressurizer fluid is superheated since the saturation temperature is about 25.96 K at 4 bar. The fluid density distribution across the pressurizer is quite large (66.8 kg/m³ below the copper rod and 0.39 kg/m³ adjacent to the top node of the copper rod). These dramatic changes in fluid properties, coupled with the complex phenomena associated with the reflood option of the code, cause the calculation to be much more sensitive and less stable numerically. Further, because the pressurizer outside-wall temperature is essentially the same as the adjacent fluid volumes, it may be important to also include axial conduction in

this heat structure. Currently, the side wall axial conduction is not simulated, and no heat from the copper rod is allowed to conduct into the loop via the pressurizer side wall.

Perhaps the most important steady-state result is the presence of void generated in the moderator vessel (on the 4-bar case). Because of the high surface temperature (about 60 K) of the moderator wall in the dome region, subcooled nucleate boiling is predicted. This void (about 3% hydrogen vapor) is then swept through the loop and does not collapse until just downstream of the heat exchanger outlet well before returning to the moderator vessel. The prediction of this void is inconsistent with the more detailed CFD predictions for reasons that have already been discussed. The persistence of hydrogen vapor in the presence of subcooled liquid along the extended flow path between the moderator vessel outlet and the heat exchanger inlet is a nonequilibrium thermodynamic effect that is not completely understood at this writing and that requires further study.

Transient results: Transition from normal to standby mode

The transient results are shown by plots of the variables of interest vs time. The variables chosen for display are labeled and physically located similar to the instrumented variables from the AEDC experiments, plus additional variables have been chosen to understand more fully the phenomena. The relationship between the ATHENA simulation variables and the similar AEDC experiment tags is given in Table 3.8. The displayed plots are shown in groups of up to nine variables per page. The first group of plotted variables, given by Fig. 3.13, shows the computer CPU time utilized to perform the transient calculation, followed by the pressure distribution around the cold source system tagged by instruments P2 through P9, respectively. The second group of plotted variables, given by Fig. 3.14, show the total mass of the modeled loop (including extraneous TDVs), the computed mass error, and the liquid hydrogen temperature distribution around the cold source system tagged by instruments T3 through T9, respectively (superheated vapor temperature if not indicated by these plots). The third group of plotted variables, given by Fig. 3.15, show a single temperature instrument output, shown as T10, along with seven valve position plots. The next group of plots, given by Fig. 3.16, shows the corresponding mass flow rates through the same respective seven valves. Finally, the last group of plots, Fig. 3.17, shows several miscellaneous variables of interest to better interpret the results. The essential sequence of events calculated for this transient is provided in Table 3.9. A detailed discussion of each plot follows.

The computer CPU time is shown as the first plot of Fig. 3.13. The figure shows that a significant amount of computer time is required to perform these calculations on a modern workstation. The sudden reset in time indicates a restart of the calculation. Therefore, the total time required is the sum of the end points of the curves. The most significant part of this figure is nearly identical in shape with the computed mass error plot to be shown later.

The pressure at the inlet and outlet tube side (hydrogen) of the heat exchanger (P6 and P2, respectively) indicates three distinct phases. Because the HD circulator was assumed (for purposes of simulation completion) to trip and coast down in 0.1 s (it actually will coast down linearly in 15 s), the initial drop in pressure at time zero is expected. Then the pressure remains essentially constant until the LD circulator is started at 14.0 s. The P2 and P6 pressure oscillates thereafter but remains near the feed vessel pressure. The pressure at the feed vessel exit (P7) is very similar to that shown by P2 and P6 for this case.

The pressure at the inlet and outlet of the moderator vessel (P3 and P4, respectively) is quite complex, with both curves being essentially the same. After an initial drop because of the HD circulator trip, a slight increase takes place at about 4 s. Then, starting at about 6 s, a general heat-up of the fluid in the vessel starts to take place with the saturation temperature/pressure being reached near the 8-s mark. From this point up until the 14-s mark, a mixture of boiling and fluid displacement by purging from the feed vessel takes place with oscillatory behavior of the pressure (peak-to-peak values of pressure swings being rather low). Then at 14 s, the LD circulator trips on, and the pressure remains near the feed vessel value for the duration of the simulation.

Table 3.8. Relationship between AEDC experiment tags and ATHENA cold source model node numbers

ATHENA		AEDC tag	Description
Name	Node		
p	72010000	P1 (PT-1)	Inlet to shell side (helium) of heat exchanger
p	70080000	P2 (PT-2)	Outlet of tube side (hydrogen) of heat exchanger
p	1090000	P3 (PT-3)	Inlet of moderator
p	5010000	P4 (PT-4)	Outlet of moderator
p	30060000	P5 (PT-5)	Inlet of circulator station
p	70010000	P6 (PT-6)	Inlet of tube side (hydrogen) of heat exchanger
p	612010000	P7 (PT-7)	Feed vessel exit
p	610010000	P8 (PT-8)	Feed vessel
p	620010000	P9 (PT-9)	Bleed vessel
n/a	n/a	P10 (PT-10)	H ₂ supply pressure
tempf	72010000	T1 (TE-1)	Inlet to shell side (helium) of heat exchanger
tempf	72080000	T2 (TE-2)	Outlet to shell side (helium) of heat exchanger
tempf	70080000	T3 (TE-3)	Outlet of tube side (hydrogen) of heat exchanger
tempf	1090000	T4 (TE-4)	Inlet of moderator
tempf	5010000	T5 (TE-5)	Outlet of moderator
tempf	20030000	T6 (TE-6)	Just before inlet to pressurizer
tempf	30060000	T7 (TE-7)	Inlet of circulator station
tempf	70010000	T8 (TE-8)	Inlet of tube side (hydrogen) of heat exchanger
tempf	612010000	T9 (TE-9)	Feed vessel exit
tempf	610010000	T10 (n/a)	Feed-bleed recycle
vlvstem	57	vlv1 (BV-1)	Low-density circulator (pump c) exit isolation valve stem position
vlvstem	42	vlv2a (BV-2)	High-density circulator #1 (pump a) exit isolation valve stem position
vlvstem	44	vlv2b (n/a)	High-density circulator #2 (pump b) exit isolation valve stem position
vlvstem	613	vlv3 (BV-3)	Liquid purge isolation valve
vlvstem	630	vlv4 (BV-4)	Feed-bleed crossover isolation valve
vlvstem	611	vlv5 (CV-5)	Feed vessel control valve
vlvstem	621	vlv6 (CV-6A/6B)	Bleed vessel control valve
pmpvel	40	SIC-01	Pump velocity high-density circulator #1 (pump a)
pmpvel	45	n/a	Pump velocity high-density circulator #2 (pump b)
pmpvel	55	SIC-02	Pump velocity low-density circulator (pump c)

Fig. 3.13. Pressure instrument edits for the transition from normal to standby simulation at 4 bar using the HFIR cold source ATHENA model.

Fig. 3.14. Temperature instrument edits for the transition from normal to standby simulation at 4 bar using the HFIR cold source ATHENA model.

Fig. 3.15. Valve position and T10 edits for the transition from normal to standby simulation at 4 bar using the HFIR cold source ATHENA model.

Fig. 3.16. Valve mass flow rate edits for the transition from normal to standby simulation at 4 bar using the HFIR cold source ATHENA model.

Fig. 3.17. Miscellaneous variable edits for the transition from normal to standby simulation at 4 bar using the HFIR cold source ATHENA model.

**Table 3.9. Sequence of events for transition to standby mode for a
4-bar liquid hydrogen operating condition**

Simulation time (s)	Plot time (s)	Description
0.0	-300.0	Start of simulation; perfect pressure control enabled (at 3 bar)
300.0	0.0	End of steady-state calculation, start of transition from normal to standby mode: HD circulator trips off in 0.1 s, disable perfect pressure control; HD circulator isolation valves close in 0.1 s; feed/bleed system valves open/close in a manner to start purging HD hydrogen and replacing with LD hydrogen.
314.0	14.0	Start LD circulator in 0.1 s, open LD circulator isolation valves in 1.0 s; enable perfect pressure control
316.0	16.0	End of simulation

The pressure at the inlet to the circulator station, P5, shows a similar pattern as in the moderator, but at a lower pressure. Then at 14 s, P5 increases dramatically (because of the initiation and subsequent backflow through the LD circulator) and shows essentially a steady-state value lower than the feed vessel pressure.

Both the feed vessel pressure (P8) and bleed vessel pressure (P9) remain constant at 5 bar and 1 bar, respectively, during the entire simulation. These volumes were temporarily changed to a RELAP5-defined TDV for purposes of simplification of the calculation.

The total modeled mass is shown by the first plot of Fig. 3.14. This plot can be misleading since it also includes the TDVs extraneous to the actual hydrogen loop (required for the model). The important aspect of the plot is that the total mass decreases as the simulation proceeds, which is directly related to the mass error computed by the code. The computed mass error is shown by the next plot in Fig. 3.15. Any ATHENA computation will produce some mass error because of the numerical method used by the code. What is important is the relative mass error with respect to the total mass. In this application, the relative mass error is on the order of 1–2% during the entire simulation. This is an acceptable range of computation error for the given application.

The plot of the heat exchanger tube side (hydrogen) outlet temperature (T3) shows a gradual heat-up, particularly from 2 to 14 s. During this time, there is very little flow in the loop caused by the circulators (off-line). In addition, warmer hydrogen fluid (caused by the moderator heat-up) is being pushed (because of thermal expansion of the fluid via density changes) from the direction of the moderator back into the loop towards the pressurizer and circulator station. In addition, the purging of the loop by the high pressure (5-bar) feed tank adds warmer (ambient temperature, 300 K) hydrogen gas to the loop. The T3 temperature is starting to level off beyond the LD circulator start-up time at 14 s.

The temperature of the inlet and outlet of the moderator, T4 and T5, shows similar patterns. Following the transient initiation, flow through the moderator vessel initially increases but then degrades as warmer fluid from the feed tank reaches the moderator vessel. The subcooled liquid then reaches saturation temperature at about 8 s. Then at 14 s, the flow degrades further as a consequence of the opening of the LD circulator isolation valves. The fluid temperature in the moderator vessel remains at the saturation temperature corresponding to the pressure.

The temperature at the inlet to the pressurizer, T6, shows a gradual decrease until about 8 s, then fluctuates until about 14 s. The 8–14 s fluctuation can be attributed to the transport of the warmer fluid from the moderator region of the loop toward the pressurizer. Then, as in other variables, the T6 temperature shows a dramatic increase at the 14 s mark, because of the backflow of warm feed vessel fluid through the LD circulator.

The temperature at the inlet of the circulator station, T7, shows essentially no change up until the LD circulator start-up at 14 s, when backflow through the LD circulator feeds warm gas at the feed vessel temperature.

The plots for the feed vessel exit vapor temperature, T9, and the hydrogen supply tank vapor temperature (not modeled but retained for consistency with the AEDC experiments), T10, are not shown but remain essentially constant at 300 K for the entire duration of this simulation.

The next page of plots given by Fig. 3.15 shows the position of seven of the valves included in the model. The plot of "vlv1b" shows that the LD circulator exit isolation valve opens in 0.1 s starting at 14 s. The plot of "vlv2b" shows that the HD circulator exit isolation valve closes in 0.1 s starting at 0 s. The plot of "vlv2d" shows that the standby HD circulator exit isolation valve remains closed for the duration of the simulation. The plot of "vlv3" shows that the liquid purge isolation valve opens in 1 s starting at 0 s. The plot of "vlv4" shows that the feed-bleed crossover isolation valve remains closed for the duration of the simulation. The plots of "vlv5" and "vlv6" show that the feed vessel and bleed vessel control valves open in 1 s, starting at 0 s. The plot of "vlv7" is not shown but remains open all the time. All of the control logic required to perform the valve open and close operations is assumed and may change as the design evolves.

The next page of plots given by Fig. 3.16 shows the mass flow rate through the same seven valves discussed in Fig. 3.15. The plot of "vlv1b" shows the mass flow rate through the LD circulator exit isolation valve. The most interesting thing about this plot is that it shows that the flow through the LD circulator is negative after the pump turns on! This is because of the differential pressure created by the combination of feed vessel opening, venting to the bleed tank, and moderator heat-up. It is expected that the flow would eventually arrive at the desired direction after a sufficient time in standby mode with the feed and bleed subsystem isolated from or with minimal interaction with the loop.

The plot of "vlv2a" confirms zero flow through the HD circulator exit isolation valve following the transient initiation. The plot of "vlv2c" shows that the standby HD circulator flow remains zero for the duration of the simulation.

The plot of "vlv3" shows that the mass flow rate through the liquid purge isolation valve becomes nonzero at transient initiation (0 s), fluctuates at relatively small positive values until the 14 s mark, then dramatically increases as the LD circulator is turned on. Hence, the timing of the opening of the LD circulator isolation valves has a significant impact on the ability to purge the HD fluid out of the system. If done too early, then backflow may occur through the LD circulator and negatively impact system purging.

The plot of "vlv4" confirms that the feed-bleed crossover flow rate through the isolation valve remains zero for the duration of the simulation.

The plot of "vlv5" shows that the mass flow rate at the feed vessel exit is very similar to that previously described by the "vlv3" purge line flow rate. Hence, the backflow through the LD circulator temporarily tends to enhance the feed vessel flow. This unanticipated effect demonstrates the importance of timing the start-up of the LD circulator (ramp up) and the LD circulator isolation valves (quick opening). The calculated behavior of loop response in the presence of reverse flow through the LD circulator is not expected to be representative of an actual transition to standby.

The plot of "vlv6" shows a nonzero flow rate at the bleed vessel entrance for the duration of the transient simulation. As in other variables, a dramatic increase in flow rate is seen at the 14 s mark. It is important to mention that very often in this or other simulations, the flow rate into the bleed vessel entrance may become choked because of the significant pressure drop induced by the low bleed vessel pressure (initially vacuum in the AEDC experiments). The choke front can also move from the bleed vessel entrance back up as far as the pressurizer entrance if the flow conditions are right to do so.

The final page of 4-bar transient plots given by Fig. 3.17 shows a number of interesting variables that will enable a better understanding of the phenomena taking place. The volumetric flow rate into the moderator starts (prior to transient initiation, $t < 0$ s) at just under 1 L/s and dramatically increases to about 2 L/s at the transient initiation (hence, the temperature decrease in the moderator vessel exit fluid

temperature seen earlier). This flow rate change is coincident with a change in differential pressure across the moderator and loop caused by both the moderator heat-up and the feed-bleed purging. After the 5-s mark, the density change caused by the fluid temperature increase also causes the volumetric flow rate to increase into the moderator vessel. At the time of LD circulator trip on (14 s), the volumetric flow rate into the moderator has trended upward because of the influence of the purge from the feed vessel. However, when the LD circulator isolation valves are opened at 14 s, backflow through the LD circulator tends to reduce flow into the moderator vessel.

The next plot of interest, Fig. 3-17, shows four metal surface temperatures within the pressurizer vessel: (2-1) the top node of the inside surface of the pressurizer vessel outside wall; (2-2) the bottom node of the inside surface of the pressurizer vessel outside wall; (2-3) the top node of the outside surface of the copper rod; and (2-4) the bottom node of the outside surface of the copper rod. Subplot 2-1 shows a continuous decrease in temperature throughout the transient simulation. Subplot 2-2 also shows a decrease in temperature until the 14 s mark after which the temperature slightly increases because of the surge of warmer fluid entering the pressurizer. The copper rod top and bottom temperatures, shown by subplots 2-3 and 2-4, respectively, indicate a decrease in value until about 10 s after the start of the transient. At about 10 s a dramatic increase in temperature is seen because of a surge of warmer fluid entering the pressurizer from the loop. These same temperatures tend to fluctuate (as one might expect because of the tendency of the copper rod to boil off the surging fluid that is entering) until the 14 s mark. Thereafter, the copper rod temperatures increase because of the dominating surge of warmer fluid leaving the loop inventory and passing through the pressurizer on into the bleed vessel. The plot also reveals a trend toward steady-state during the last half-second of the transient simulation.

The next plot of interest shows the metal surface temperatures of the moderator inside walls at three locations: (1) the moderator entrance, (2) the moderator exit, and (3) midway within the moderator volumes. Both inside and outside walls, with respect to the moderator vessel oriented adjacent to the hydrogen fluid within the reactor beam tube, are overlaid on the plot. The general trend for all the moderator wall surface temperatures prior to about 7.5 s is moderate change. Beyond the 7.5-s mark, the midway surface temperatures show dramatic temperature rise because of boiling and subsequent decrease in heat transfer coefficient at low flow rates. Following the 14-s mark, a subsequent increase is predicted for all moderator wall temperatures. This subsequent rise corresponds to the further degradation of moderator vessel flow because of the backflow of fluid through the LD circulator when the LD circulator is started. Close inspection of the mid-way wall temperatures prior to 14-s suggests that cooling of the moderator vessel wall by the purge flow may be sufficient to preclude threatening wall temperatures. Here again, the importance of timing of when the LD circulator is started is clearly evident.

The plots of HD circulator pump head and rotational velocity confirm a trip on transient initiation (0 s) over an assumed period of 0.1 s, and remaining zero thereafter. The plots of LD circulator pump head and rotational velocity confirm zero contribution until the 14-s mark, and nonzero contribution thereafter, winding up over an assumed period of 0.1 s. The plot of liquid velocity entering the pressurizer from the loop confirms the earlier plots of temperature phenomena within the pressurizer and mass flow rates into the bleed vessel.

In summary, the steady-state results for the 4-bar operating point appears reasonable. The 4-bar steady-state also shows significant voiding in the moderator that may not be real (recall the conservative wall surface temperatures). It appears to be possible that a normal-to-standby transition at the 4-bar operating point could be achieved. This is evidenced by a slowed growth of moderator vessel wall temperatures prior to turning on the LD circulator. The benefit of purging the loop with a higher pressure feed tank is clear. This simulation also shows the importance of timing the initiation of the LD circulator and opening the isolation valves. If the LD circulator/isolation valves are started/opened too soon, a significant portion of the purging flow from the feed tank may bypass the moderator vessel and backflow through the LD circulator into the pressurizer and out of the loop. This reduces flow into the moderator vessel and cooling of the vessel walls. It is clear that if a decision were made to go back to a 4-bar 20 K

operating mode, much more analysis would have to be performed, and several issues would have to be resolved.

3.4.3.2 Results of the Supercritical System at 15-Bar Operation

The analysis results for the reference design supercritical operation at 15 bar are subdivided into steady-state and transient sections. The transient section discusses the simulation of the transition from normal to standby mode, and an HD circulator trip coincident with decrease in secondary heat removal.

Steady-state results

The steady-state calculation for the nominal 15-bar model immediately revealed enhanced numerical stability (over the 4-bar model) by enabling a larger time-step throughout the simulation. This is primarily because of the lack of two-phase interaction in the pressurizer. The calculation was executed to 100-s simulation time with a time-step of 0.001 s, and then extended to 200 s with a larger time-step of 0.002 s. Judging from the minor-edit outputs, by this time it appeared that the system was sufficiently steady to conclude a steady-state had been reached.

The results show a maximum steady-state pressure in the system just downstream of the HD circulator at 15.03 bar. The minimum pressure at the HD circulator inlet is 13.97 bar for a total differential pressure around the loop of 1.06 bar at a volumetric flow into the moderator vessel of 0.96 L/s (recall the loop losses have been tuned to yield approximately 1 bar loss at 1 L/s flow).

The minimum temperature in the hydrogen loop is just downstream of the heat exchanger outlet at exactly 170 K (recall the perfect heat exchanger). The maximum temperature in the loop, at the heat exchanger inlet, is 21.03 K. The temperature at the moderator inlet and outlet is 17.06 K and 20.62 K, respectively. The additional temperature rise between the heat exchanger outlet and the moderator inlet (0.06 K) and the moderator outlet and heat exchanger inlet (0.41 K) can be attributed to the frictional losses in the transfer lines, the work done by the HD circulator (0.34 K by the circulator alone), and the energy deposited into the loop from the copper rod of the pressurizer. The total power deposited into the moderator heat structures from nuclear heating is about 1524.5 W. The corresponding total power deposited into the hydrogen fluid adjacent to the moderator heat structures from both nuclear heating and convection heat transfer is about 2154.9 W (hence, about 630.4 W is deposited directly into the fluid by nuclear heating). This agrees well with the control variable computed flow energy (CV-28) of 2104.6 W.

Perhaps the most challenging part of the model development has been the pressurizer. The steady-state results reveal why this is the case. The copper rod (HS-160) shows an axial temperature profile starting at 291.4 K at that top node (HS-160-10) and dropping down to about 279.3 K at the bottom node, which interfaces with the second fluid node up within the pressurizer (PV-60-2). The top node is external to the vessel and interfaces with a heat structure and single volume combination that allows a constant temperature boundary condition of 300 K. The temperature profile in the radial direction of the copper rod is essentially flat throughout at steady-state since very little energy is deposited into the fluid except during level and/or pressure disruptions. The fluid temperatures of the hydrogen volumes adjacent to the copper rod (PV-60-2 through PV-60-10) are slightly less than the surface temperatures of the copper rod. The volume just below the copper rod is at a temperature of 32.9 K that is several degrees above the nominal loop temperature at that point (20.7 K) and significantly less than the fluid nodes above (~ 300 K). Even though the fluid is not two-phase because it is supercritical, the density difference between the adjacent volumes is quite large (44.9 kg/m³ below the copper rod and 1.3 kg/m³ adjacent to the copper rod). These dramatic changes in fluid properties, coupled with the complex phenomena associated with the reflood option of the code, cause the calculation to be sensitive and less stable numerically than otherwise. Further, because the pressurizer outside wall is essentially the same temperature as the adjacent fluid volumes, it may be important to also include axial conduction in this heat structure. Presently, this axial conduction is not simulated, and no heat from the copper rod is

allowed to conduct into the loop via the pressurizer side wall. This assumption may not be realistic with respect to heat input into the loop. It is hoped that the AEDC experiments will allow this time-saving assumption to be validated.

Transient results: Transition from normal to standby mode

The transient results for the transition from normal to standby are shown by plots of the variables of interest vs time. The layout of variables chosen for display is identical to that shown previously for the 4-bar transient results and are given by Figs. 3.18 through 3.22, respectively.

The essential sequence of events for this transient is provided in Table 3.10. A detailed discussion of each plot follows.

The first plot of Fig. 3.18 is the computer CPU time utilized to produce the calculation. As before, a significant amount of computer work is required to perform this simulation on a modern workstation.

The pressure at the inlet and outlet tube side (hydrogen) of the heat exchanger (P6 and P2, respectively), and the pressure at the inlet and outlet of the moderator vessel (P3 and P4, respectively), all indicate a similar response. Because the HD circulator was assumed to trip and coast down in 15 s, a smooth drop in pressure is expected at all locations downstream of the HD circulators. The only pressure plot upstream of the HD circulators, shown as P5 at the inlet of the circulator station, indicates the pressure will remain essentially constant at that location for the first 30 s of the transient. The pressure within the moderator shows fluctuation starting at about 20 s due to the temperature-induced density changes from the moderator heat-up.

Contrary to what one might expect, the dramatic pressure increase throughout the loop, shown as a peak at about 34 s, is *not* caused solely by the spinup of the LD circulator, which starts at 30 s. This is confirmed by the results of the next transient to be presented, which is similar but does not include the trip-on (spin-up) of the LD circulator. The pressure peak can possibly be explained in the following manner. The moderator vessel experiences an immediate heat-up due to the loss of flow, which causes an

Table 3.10. Sequence of events for transition to standby for supercritical hydrogen operating mode

Simulation time (s)	Plot time (s)	Description
0.0	-200.0	Start of simulation
200.0	0.0	End of steady-state calculation; start of transition from normal to standby; HD circulator trips off in 15 s; perfect pressure control (feed/bleed system valves remain closed)
215.0	15.0	HD circulator coastdown complete
230.0	30.0	Start-up LD circulator in 15 s; open LD circulator isolation valves in 2 s; close HD circulator isolation valves in 2 s; increase secondary side temperature set point from 17 K to 70 K in 15 s
232.0	32.0	LD circulator isolation valves completely open; HD circulator isolation valves completely closed
245.0	45.0	LD circulator start-up complete; secondary side temperature set point change complete
263.0	63.0	End of transient simulation

Fig. 3.18. Pressure instrument edits for the transition from normal to standby simulation at 15 bar using the HFIR cold source ATHENA model.

Fig. 3.19. Temperature instrument edits for the transition from normal to standby simulation at 15 bar using the HFIR cold source ATHENA model.

Fig. 3.20. Valve position and T10 edits for the transition from normal to standby simulation at 15 bar using the HFIR cold source ATHENA model.

Fig 3.21. Valve mass flow rate edits for the transition from normal to standby simulation at 15 bar using the HFIR cold source ATHENA model.

Fig. 3.22. Miscellaneous variable edits for the transition from normal to standby simulation at 15 bar using the HFIR cold source ATHENA model.

overall decrease in density in the fixed-volume loop. The expansion resulting from this density change then causes the pressure to increase. Because a perfect pressure control is enabled, the pressurizer will respond quickly to changes in pressure in the system. As a consequence, when the loop pressure increases, fluid will be expelled from the loop into the pressurizer to maintain the pressure set point.

This fluid expulsion is evidenced in the plot of "velocity entering the pressurizer from the loop" shown in Fig. 3.22 (and also illustrated in the second transient as shown in Fig. 3.27). It is observed that the fluid expulsion is much more pronounced in the case with initiation of the LD circulator. This is because of the enhanced heat transfer to the hydrogen from the helium during the period when the heat exchanger temperature controller is increased from 17 K to 70 K, and the hydrogen flow through the heat exchanger is increased because of the start-up of the LD circulator. For a short period of time, the rate of expansion and pressurization exceeds the rate at which hydrogen can be removed from the pressurizer by the perfect pressurizer controller. After about 46 s, the perfect controller is adequate for controlling pressure.

Both the feed vessel pressure (P8) and bleed vessel pressure (P9) remain constant at 15 bar and 1 bar, respectively, during the entire simulation. These volumes were temporarily changed to a RELAP5-defined TDV for purposes of simplification of the calculation. Furthermore, because the feed/bleed subsystem is not utilized during this simulation, the pressure at the feed vessel exit (P7) also remains constant. Recall the system is being perfectly controlled by use of a TDV connected to the top of the pressurizer and allowing fluid to freely flow in and out of the pressurizer.

The total modeled mass and computed mass error, shown as the first two plots of Fig. 3.19, again indicate an acceptable 1–2 % computed mass error.

The plot of the heat exchanger tube side (hydrogen) inlet (T8) and outlet temperature (T3), and the temperature at the inlet of the circulator station (T7), shows a gradual heat-up between 30 and 45 s. During this time, the temperature set point on the secondary side of the heat exchanger (recall the perfect heat exchanger) has been reset from 17 K to 70 K. This has the effect of heating up the loop from the other end (as opposed to the moderator end). Hence, because there is very little forced flow in the loop, the heat from both ends pushes the hydrogen (due to thermal expansion of the fluid via density changes) from both directions towards the pressurizer and circulator station.

The temperature of the inlet and outlet of the moderator, T4 and T5, shows similar patterns. Following the coastdown of the HD circulator at transient initiation, the flow through the moderator vessel decreases and becomes negative at the inlet causing both T4 and T5 to increase dramatically. During the time between 20 and 30 s, the inlet fluid temperature decreases while the exit fluid temperature continues to increase. At 30 s, the LD circulator is started, resulting in positive flow through the vessel thereafter. The net positive flow increase in the vessel causes both the inlet and outlet fluid temperatures to decrease to a level close to the level existing at the steady-state prior to the transient initiation. However, at the time of completion of the LD circulator start-up at 45 s, the moderator vessel inlet and outlet fluid temperatures continue to increase because of the arrival of warmer hydrogen from the heat exchanger portion of the loop.

The temperature at the inlet to the pressurizer, T6, takes on the temperature of the fluid passed into the pressurizer as the transient evolves, and warmer fluid is generated.

The temperature at the feed vessel exit, T9, and the feed-bleed recycle, T10 (on Fig. 3.20), are meaningless since they are isolated from the loop for this transient.

The next page of plots given by Fig. 3.20 shows the position of seven of the valves included in the model. The plot of "v1v1b" shows that the LD circulator exit isolation valve opens in 2 s starting at 30 s. The plot of "v1v2b" shows that the HD circulator exit isolation valve closes in 2 s starting at 30 s. The plot of all the remaining valves in the system shows that they are closed for the entire simulation.

The next page of plots given by Fig. 3.21 shows the mass flow rate through the same seven valves discussed in Fig. 3.20. The plot of "v1v1a" shows the mass flow rate through the LD circulator exit isolation valve. The flow rate is zero up until 30 s, then shows a dramatic negative flow initially as the valve opens and gradually increases as the LD circulator spins up completely. During the time period of

30–45 s the temperature of the hydrogen in the heat exchanger is increasing from 17 K to 70 K, and the flow through the LD circulator is mostly negative. The flow eventually becomes positive near the end of the simulation.

The plot of “vlv2a” confirms zero flow through the HD circulator exit isolation valve after the 32-s mark. Prior to that time, the flow steadily decreases as the HD circulator coasts down, then hovers around zero until the isolation valves start to close. During the time the isolation valves close, a significant negative flow rate occurs across “vlv2a” because of hydrogen expansion in the heat exchanger caused by heat transfer from the helium on the secondary side and eventually becomes zero when the valve is finally closed.

The remainder of the valve flow plots confirms zero flow through the closed valves for the entire duration of the transient.

The final page of 15 bar normal-to-standby plots is given by Fig. 3.22. The volumetric flow rate into the moderator starts (prior to transient initiation, $t < 0$ s) at just under 1 L/s and decreases below zero as the HD circulator coasts down. Then the flow into the moderator hovers about zero until the LD circulator and temperature set point starts to change at the 30-s mark. The inlet flow rate remains above zero for the duration of the transient, but has not arrived at a steady-state by the end of the simulation.

The next plot of interest, shown as a subplot in Fig. 2.22, shows four metal surface temperatures within the pressurizer vessel: (2-1) the top node of the inside surface of the pressurizer vessel outside wall, (2-2) the bottom node of the inside surface of the pressurizer vessel outside wall, (2-3) the top node of the outside surface of the copper rod, and (2-4) the bottom node of the outside surface of the copper rod. Subplot 2-1 shows a continuous decrease in temperature throughout the transient simulation. Subplot 2-2 also shows a decrease in temperature until the 32-s mark, after which the temperature slightly increases because of the surge of warmer fluid entering the pressurizer. The copper rod top and bottom temperatures, shown by subplots 2-3 and 2-4, respectively, show a decrease in value until about 34 s after the start of the transient. At about 34 s an increase in temperature is seen because of a surge of warmer fluid entering the pressurizer from the loop (due to warm hydrogen from backflow through the LD circulator and forward flow from the moderator vessel). Then after about 45 s, the gradual cool-down of the copper rod continues until the end of the simulation.

The next plot of interest shows the metal surface temperatures of the moderator inside walls at three locations: (1) the moderator entrance, (2) the moderator exit, and (3) midway within the moderator volume. Both inside and outside walls, with respect to the moderator vessel oriented adjacent to the hydrogen fluid within the reactor beam tube, are overlaid on the plot. The general trend for all the moderator wall surface temperatures prior to about 15 s is a moderate change. Beyond the 15-s mark, the midway surface temperatures show a dramatic temperature rise because of a decrease in heat transfer coefficient at low flow rates. Following the 32-s mark, an overall decrease in moderator temperature is shown because of the surge of cooler hydrogen from the portion of the loop between the heat exchanger and the moderator vessel inlet. Then, after about 55 s, the gradual heat-up of the moderator surfaces continues as warmer hydrogen from the heat exchanger reaches the moderator vessel.

The plots of the HD circulator pump head and rotational velocity confirm a trip on transient initiation (0 s) over an assumed period of 15 s and remain at zero rotational velocity thereafter. The plots of the LD circulator pump head and rotational velocity confirm zero contribution until the 30-s mark, and a spin-up over a period of 15 s.

The plot of liquid velocity entering the pressurizer from the loop confirms the earlier plots of temperature phenomena within the pressurizer and loop due to surging of fluid from density changes and forced flow (pumping).

In summary, the steady-state results for the 15-bar operating point appear reasonable. The 15-bar operating point appears able to achieve a normal-to-standby transition in operating mode. However, significant improvements in the modeling of the pressure control system are needed for several reasons including: (1) to prevent pressure excursions of the type presented, (2) to provide for smooth pressure transitions, and (3) to model the actual pressure control system design. For example, the calculated

maximum loop pressure at about 34 s shows very little margin to the rupture disk pressure that would be included in the system for protection against over-pressure events. The final design of the control system will need to provide for smooth pressure transitions rather than the sharp spikes that have appeared here from the simplified perfect control system. The fluid and heat structure variable responses appeared reasonable. The simulation was terminated at approximately 63 s due to fluid property difficulties near the critical point. At this simulated time, the moderator vessel wall temperatures were increasing at a rate of approximately 5 K/s. Therefore, it is not known what the final steady-state value of the vessel wall temperature would be. However, there is nothing to suggest that moderator cooling would become further degraded after the time simulated. This assertion is based on the fact that positive flow has been established in the moderator vessel, and a significant thermal safety margin (about 278 K) with respect to wall stresses remains.

Transient results: HD circulator trip with decrease in secondary heat removal

The transient of HD circulator trip with decrease in secondary heat removal is essentially identical to the normal to standby transient with the exception of not spinning up the LD circulator to establish a standby flow. As before, the results are shown by plots of the variables of interest vs time. The layout of variables chosen for display is identical to those shown previously and are given by Figs. 3.23 through 3.27. The essential sequence of events for this transient is provided in Table 3.11. A discussion of each plot and the difference from the previous normal-to-standby case follows.

The plots of pressure shown in Fig. 3.23 are essentially the same as those from the previous normal-to-standby case given by Fig. 3.18. The exception to this is a less pronounced pressure increase between 45 and 50 s as compared to the normal-to-standby case. The pressure pulse at 39 s is roughly 3/4 that of the normal-to-standby simulation. Therefore, the LD circulator does not dramatically effect the overall loop pressure for this scenario.

Table 3.11. Sequence of events for HD circulator trip transient for supercritical hydrogen operating mode

Simulation time (s)	Plot time (sec)	Description
0.0	-200.0	Start of simulation
200.0	0.0	End of steady-state calculation; start of transition from normal to standby mode; HD circulator trips off in 15 s; perfect pressure control is enabled; feed/bleed system valves remain closed (no purging is performed)
215.0	15.0	HD circulator coastdown complete
230.0	35.0	Keep LD circulator isolation valves closed; keep HD circulator isolation valves open; increase the secondary side temperature set point from 17 K to 70 K in 15 s
245.0	50.0	The change in secondary side temperature set point is complete
263.0	63.0	End of transient simulation

Fig. 3.23. Pressure instrument edits for the high-density circulator trip and decrease in secondary heat removal simulation at 15 bar using the HFIR cold source ATHENA model.

Fig. 3.24. Temperature instrument edits for the high-density circulator trip and decrease in secondary heat removal simulation at 15 bar using the HFIR cold source ATHENA model.

Fig. 3.25. Valve position and T10 edits for the high-density circulator trip and decrease in secondary heat removal simulation at 15 bar using the HFIR cold source ATHENA model.

Fig. 3.26. Valve mass flow rate edits for the high-density circulator trip and decrease in secondary heat removal simulation at 15 bar using the HFIR cold source ATHENA model.

Fig. 3.27. Miscellaneous variable edits for the high-density circulator trip and decrease in secondary heat removal simulation at 15 bar using the HFIR cold source ATHENA model.

The plots of temperature in Fig. 3.24 show a few differences from those in Fig. 3.19. The temperature at the inlet to the moderator, T4, oscillates through a few cycles and does not trend upward. This is likely because of the longer time required before warm hydrogen fluid from the heat exchanger region reaches the moderator vessel inlet. It follows that this longer time period is caused by the lack of forced flow from the LD circulator. The temperature at the outlet of the moderator, T5, is significantly higher at the end of the simulation for the present case also because of the lack of forced flow from the LD circulator.

The temperature at the inlet of the circulator station, T7, shows a drop-off at 60 s that is not present in the previous normal-to-standby case. The peak in temperature at the inlet to the pressurizer, T6, remains present longer than the normal-to-standby case. As expected, the flow produced by the LD circulator tends to keep the loop temperature and moderator vessel wall temperatures lower.

The next page of plots given by Fig. 3.25 shows the position of seven of the valves included in the model. Since all valves remain fixed, no change from their initial position is seen. In particular, the LD and HD circulator isolation valves, "vlv1b" and "vlv2b," do not change position.

The next page of plots given by Fig. 3.26 shows the mass flow rate through the same seven valves discussed in Fig. 3.20. Since all valves but the HD circulator isolation valve (vlv2a) are closed, the corresponding flow rate is zero. Compared to the normal-to-standby case, the "vlv2a" flow rate is essentially the same up until the 35-s mark. Then as the hydrogen temperature set point of the loop heat exchanger increases, a dramatic surge in loop density and pressure takes place. This causes the "vlv2a" flow rate to be dramatically different (more like the LD isolation valve flow rate in the previous normal-to-standby case).

The final page of results is given by Fig. 3.27. The volumetric flow rate into the moderator is very similar to the previous normal-to-standby case except the flow remains near zero beyond the 50-s mark rather than increasing. This is because the LD circulator is turned on in the previous transition to standby while the current simulation does not turn on the LD circulator. As a consequence, a "chugging" of flow is evident in the moderator region that may be attributed to fluid density changes throughout the loop. In the previous simulation, this chugging was not as evident due to the continuously induced positive flow toward the moderator caused by the action of the LD circulator. The importance of the LD circulator is clearly evident in this regard.

The next plot of metal surface temperatures within the pressurizer vessel is very similar to the previous case except that beyond the 50-s mark, the temperatures appear to approach a new steady-state rather than decrease.

The next plot of the metal surface temperatures of the moderator inside walls is also very similar to the previous case except the temperatures trend toward much higher increasing values beyond the 50-s mark (because of the lack of flow).

The plots of the HD circulator pump head and rotational velocity are nearly identical to the previous case. The LD circulator head and rotational velocity are zero for this case.

The plot of liquid (ATHENA nomenclature: supercritical with temperature less than the critical temperature) velocity entering the pressurizer from the loop confirms the earlier plots of temperature phenomena within the pressurizer and loop because of surging of fluid from density changes and forced flow (pumping). The response is similar to the previous case except a trend toward a zero steady-state is seen beyond the 60-s mark. Again, this is because of the lack of forced flow from the LD circulator.

In summary, this last case of an HD circulator trip with decrease in secondary heat removal at 15 bar demonstrates the importance of the LD circulator flow in standby mode.

3.4.4 Conclusions and Recommendations

It has been shown that the ATHENA code will provide the capability to simulate the HFIR cold source thermal-hydraulic system dynamics under both normal and accident conditions. The strength of this approach to the simulation need is the demonstrated history of nuclear light-water safety analysis

with the companion RELAP5 code and the ability to simulate complete systems. The weakness of this approach is the need for validation and some deficiencies in the code with respect to hydrogen simulation via ATHENA.

The model development for the HFIR cold source is well underway. At the time of writing, the design is not finalized and so several areas of model development are incomplete. These areas, which include the pressure control system, transfer line geometry, pressurizer geometry, and heat exchanger geometry presently include assumed values that will be modified as the design evolves.

The ATHENA moderator model yields conservatively high wall surface temperatures adjacent to the hydrogen. It is anticipated that this conservatism will be verified as an outcome of the AEDC experiments. There are currently no plans to relax this conservatism and arrive at a best estimate of temperatures on the moderator wall surfaces.

The pressurizer model does not include axial conduction along the side walls and into the loop. It appears from data obtained in the AEDC experiments that this simplification will not yield acceptable results (it is probable that significant heat is conducted into the loop via the pressurizer side walls). Hence, it is likely that this detail must be incorporated into the model.

Several assumptions were made to obtain the homologous performance curves for the circulators. The particular case of reverse flow through the circulators challenges these assumptions.

The ability to include helium as a second fluid in the model (secondary side of the heat exchanger) has been demonstrated. It is now known from the AEDC experiments that the currently assumed simplistic level of detail shown here is not sufficient for simulating the secondary side effects. In particular, significant changes in helium temperature were observed on the secondary side of the heat exchanger during the AEDC experiments. Therefore, this detail must also be incorporated into the model.

A deficiency presently exists in the ATHENA code when fluid properties are required near the critical point. In particular, the code will often end abruptly and not complete a simulation because of a required speed-of-sound evaluation near the critical point. This occurred for all ATHENA simulations presented in this section. This problem must be corrected to obtain reliable results in the future.

The steady-state results for both the 4-bar and 15-bar operating points appear reasonable. The 4-bar case shows significant voiding in the moderator at steady-state, which may not be real (recall the conservative wall surface temperatures).

The ability to achieve a change in operating mode from normal, full power conditions to a standby mode has been demonstrated to a reasonable level of confidence (given the current state of the model and system design) for the 4-bar and 15-bar operating points.

The demonstration of the normal-to-standby transition from the 4-bar subcritical operating point was not totally successful in that an abnormal ATHENA termination (because of critical-point fluid property difficulties) prevented reaching a new steady-state at standby conditions. However, sufficient progress was made toward this standby state to clearly demonstrate the validity of the purge/blow-through method to quickly reduce the loop hydrogen density sufficient to permit LD circulator operation while at full power.

The 4-bar normal-to-standby analysis also demonstrated the importance of timing when LD circulation should begin. The potential for backflow through the LD circulator during the purging operation because of premature LD circulation initiation is very real. This backflow should be prevented because it provides a pathway for the high pressure purging flow from the feed vessel through the circulator, into the pressurizer, and out of the loop in a manner that bypasses the moderator vessel and the majority of the concentric walled hydrogen transfer line. During the time period prior to LD circulator initiation, a reduction in cooling occurred after the HD circulator was turned off. The purge flow through the moderator vessel was sufficient to slow the rate of growth in the moderator vessel wall temperature that resulted. Once the purging flow bypassed the moderator vessel because of backflow through the LD circulator, forced flow cooling of the moderator vessel decreased and once again accelerated the rate of growth of the moderator vessel wall temperature.

The simulation of the normal-to-standby transition from the 15-bar supercritical operating point was likewise complicated by an abnormal ATHENA termination because of critical-point fluid property difficulties. This prevented the simulation from reaching a new steady-state at standby conditions. However, sufficient progress was made to demonstrate the validity of the heat exchanger induced heat-up method in approaching standby while at full power. As shown in the 4-bar case, the timing of LD circulator initiation after HD circulator shutdown is very important because of the potential for backflow through the LD circulator. This 15-bar simulation also clearly shows that additional work is required in the modeling of the pressure control system to prevent pressure excursions arising when the hydrogen heat-up occurs in the heat exchanger.

The importance of forced flow cooling of the moderator vessel induced by start-up of the LD circulator was demonstrated by a second 15-bar simulation of an HD circulator trip with decrease in secondary heat removal. This additional simulation indicated periods of positive and negative flow in the moderator vessel inlet as a result of chugging in the loop induced by heat transfer and fluid expansion inside the moderator vessel. Such chugging was not as pronounced in the first 15-bar simulation and can be directly attributed to the flow induced by the LD circulator.

The simulations presented in this section are preliminary and based on the design as of approximately May 1997. The model development was frozen at that time in order to produce the preliminary transient simulation results shown here. Since that time, significant design changes have been made. Because the design is evolving, many of the model components are already out of date. In addition, a number of model deficiencies have been identified that are very significant and that require further model development (critical-point fluid property fix, incorporation of the actual control system, improved modeling of the pressurizer side walls, modification of the transfer lines geometry, change of heat exchanger, etc.). Therefore, the reader must be very careful in interpreting these results and not try to draw any absolute conclusions on system behavior with regard to hard numerical results such as the magnitude of vessel wall temperature, rates of change of heat-up, or the severity of pressure transients. It is, however, very appropriate to arrive at general conclusions on the behavior of the system from these results and the ability of the simulation tools chosen to perform the safety analysis tasks ahead.

3.5 EXPECTED RESPONSES FOR OFF-NORMAL EVENTS ASSOCIATED WITH THE HB-4 COLD SOURCE

Off-normal event transients were examined for the current HFIR cold source design. Event definitions and frequencies were taken from a recently completed event assessment report (see ref. 3) for the preconceptual design (and summarized/updated in Sect. 3.3 for the current design), and the expected cold source loop responses were deduced. Events that are recommended as design basis events are explicitly recognized and considered. Cold source loop components and equipment pertinent to each event were identified. Conservative HFIR responses to each event were identified with respect to Requirement 11 of Sect. 1.2.3. The lowest acceptable number of barriers is one, with the underlying objective of this requirement being the segregation of H_2 from O_2 .

In performing this evaluation, it was assumed that there are five standard/normal operating modes for the cold source from which most abnormal events would occur. Detailed procedures will be developed in the future for each of these operating modes (see further discussions in Sect. 4.4.4). A brief introductory description of each mode is presented here. First is the shutdown mode in which the cold source is in thermal equilibrium with the ambient; the cold source pressure is uniform throughout the system at approximately 4 bar; no significant nuclear heat load exists on the moderator vessel wall (the reactor is shut down); and none of the active cold source hydrogen systems are functioning (with the exception of the vacuum blanketing systems that remain operational during all periods when hydrogen is housed in the system).

The second mode is one of transient cooling in which there is no significant nuclear heat load on the moderator vessel wall (the reactor is shut down), and controlled forced flow heat transfer in the heat exchanger is employed to cool the hydrogen from ambient temperature to the normal operating temperature of about 17 K [while taking precautions not to freeze hydrogen (~14 K)]. LD followed by HD circulator operation is used to provide the forced circulation while the pressure control system is used to maintain loop pressure at ~15 bar. Supercritical pressure (> 13 bar) must be maintained throughout the cooling process to retain the advantages of uniphase heat transfer and fluid flow. The rate of cool-down is to be controlled by the operation of the refrigeration system coupled to the hydrogen system pressure and temperature controller. This transient is referred to as the cooling or cool-down mode.

The third operating mode is that of normal steady-state operation at 15 bar with (or without) the full nuclear heat load applied to the moderator vessel after the hydrogen system is fully cooled to ~17 K. Pressure and temperature control are maintained by the loop controller by balancing the heat applied to the hydrogen with the heat removal rate supplied by the refrigeration system. Forced hydrogen flow is provided by an HD circulator. This is the design state in which the cold neutron beam is available for experimental use.

The fourth mode is a constant pressure (15 bar) transient warm-up to approximately 100 K (following the normal mode operating temperature of ~17 K) and a switch to forced circulation provided by the LD circulator instead of the HD circulator. Supercritical pressure (> 13 bar) must be maintained throughout the warm-up process in order to retain the advantages of uniphase heat transfer and fluid flow. (The refrigeration system is neutralized in this mode, meaning that an attempt is made to reduce the rate of heat transfer from the hydrogen inventory to the helium refrigerant by use of a compensation heat load supplied directly into the helium. This technique maintains the refrigeration system near its full heat load rejection rate and is intended to simplify the coordination task of matching the heat load added in the moderator vessel with the heat transfer rate desired in the main heat exchanger.) The hydrogen inventory heat-up may (or may not) be accomplished under full nuclear heat loading in the moderator vessel walls, but must be accomplished under the imposed requirement of no impact on reactor operation. The heat addition rate to the hydrogen while in this mode must be carefully monitored and coordinated with the forced flow circulation and refrigerator operation in order to preclude undesirable pressure transients. This controlled transient mode of operation is referred to as the warming or warm-up mode and is required prior to establishing the standby or shutdown modes following loop operation in the normal mode.

The fifth operating mode is one of steady-state continuous operation using the LD circulator with the nominal hydrogen inventory temperature maintained near 100 K. Full moderator vessel wall nuclear heat load may (or may not) be maintained, but the loop pressure must be kept supercritical near 15 bar in order to maintain the uniphase conditions required for possible subsequent cool-downs and reestablishment of the normal operating mode. Refrigerator operation during this mode is to employ only the LN_2 cooling portion with a bypass of the normally active expander based portion. This steady mode of loop operation is referred to as standby.

As mentioned earlier, most abnormal events will occur during one of the five normal operating modes. These abnormal events will require response actions of the cold source system and the reactor. Formal AOPs and EOPs will be formulated to implement these response measures, and the reader is referred to Sect. 4.4.6 for further discussion. The requirements and goals of such cold source and reactor responses are more fully identified in the abnormal event discussions that follow in this section and depend on the type of threat posed by the abnormal event. Preliminary mitigative responses identified thus far include: (1) the provision of hydrogen system depressurization and subsequent operation at relatively warm temperatures, (2) passive external cooling of the cold source moderator vessel provided by helium flooding of the vacuum tube vacuum space, and (3) reactor shutdown. These response measures will be employed as appropriate in the AOPs or EOPs to maximize the mitigative benefit. Additional response measures will be developed as necessary in the future.

The number of intact HD hydrogen barriers of the cold source loop configuration after the occurrence of each abnormal event are identified and compared with the number of barriers extant during normal operation. If after an event, three levels remained for the cold HD portions or two levels for the warm LD portions, then the threat to safety is minimized. If less than these numbers of barriers remained, then the loop was characterized as having a reduced level of safety. It must be remembered from Requirement #11 in Sect. 1.2.3 that at least one hydrogen barrier must remain after an event. For the HD cold portions of the loop, a maximum of two barriers can fail before this one-barrier limit is imposed. For the warm portions of the loop, only one barrier failure is permitted before reaching this limit.

It is assumed that a decrease in the number of hydrogen barriers below that extant in the normal operating mode because of a leak in the primary or innermost hydrogen barrier generally results in the response measures of reactor shutdown and a transition of cold source loop operation to a low pressure mode at a hydrogen temperature similar to that of the standby mode. There is good reason for these response actions. First, it should be recognized that the overall event frequency of hydrogen leaving the cold source system in an uncontrolled manner depends directly on the simultaneous failure of all barriers of the multibarrier design and upon the failure of the venting measures provided. Thus, even though a single barrier may fail and not leak hydrogen in an uncontrolled manner, the available level of protection remaining in the design after barrier failure is reduced and a follow-on action to enhance the effectiveness of the remaining safety features is prudent.

That loop depressurization is an appropriate response to primary H_2 barrier failure derives from the effectiveness of the cold source design in responding to hydrogen leaks. Such effectiveness depends upon the magnitude of the H_2 leak. (For instance, a small H_2 leak into the vacuum tube vacuum space may be exhausted by the vacuum pump directly into the H_2 ventilation system with no subsequent consequence. However, a catastrophic failure of the moderator vessel may inject sufficient H_2 into the vacuum space that it pressurizes the vacuum space such that the rupture disk cannot protect the double-walled neutron window. The window fails, H_2 gas over-pressurizes the beam tube extension region volume, fails the beam tube window, and flows into the beam room atmosphere.) Because the H_2 leak flow rate depends directly on the pressure of the hydrogen being leaked, and because the effectiveness of controlled venting of such leaks in turn depends on the leak flow rate, it is prudent to attempt to minimize the leak flow rate by depressurization of the hydrogen system.

Note also that cold source operation at low pressure directly reduces moderator vessel wall stresses compared to those arising at standby 15-bar operation. Low pressure operation, combined with the reduced heating following reactor shutdown, increases the margin of safety in the moderator vessel wall stresses compared to that extant at 15 bar.

This low pressure/low heating operational response to hydrogen leaks also accomplishes two additional functions in regard to safety and reliability of cold source loop operation. First, higher temperature operation permits the sole use of the LN_2 based mode of refrigerator operation to remove the moderator vessel heat load. This refrigerator option is more reliable than use of the expander portion of the refrigerator as the use of passive measures is maximized. Second, higher temperature operation of the cold source loop warms the system sufficiently to prohibit condensation or freezing of any air that may come into contact with normally colder (17–20 K) hydrogen containing surfaces after a primary barrier failure/leak. Such cryopumping has the potential to accumulate O_2 (or air) inventory without detection (see Sects. 3.2.2 and 3.2.3) and creates the potential for a H_2/O_2 (or H_2 /air) reaction that may occur if H_2 becomes available to react.

This zero tolerance philosophy in dealing with hydrogen leaks from the primary H_2 barriers of the cold source system will, however, require future assessment and evaluation. The reason for this is because of the definition of what is actually meant by the term " H_2 leak" and the follow-up question of how such leaks are detected and monitored. It is clear that strict imposition of zero leakage must be relaxed because of the existence of molecular diffusion of hydrogen through solid, intact aluminum barrier walls. Thus some nonzero level of hydrogen is always to be expected in either the vacuum space surrounding the cold H_2 or the inert cover blanket (either vacuum or inert gas) surrounding the warm

hydrogen. It should also be noted that the zero tolerance philosophy of hydrogen leaks gives no credit to the venting and exhaust features of the current design.

The low-pressure/high-temperature operation of the cold source loop as discussed above after reactor shutdown contrasts with the previously described standby mode, which simply increases the operating temperature of the H_2 coolant to approximately 100 K, while maintaining loop operating pressure supercritical at 15 bar. Thus the standby mode maintains full operating pressure throughout the H_2 system, while operation at low pressure (after a H_2 leak) attempts to minimize the cumulative mass of H_2 lost. In both standby and low-pressure operation, forced flow cooling of the moderator vessel is to be maintained by the LD circulator. Standby operation must be capable of removing full reactor-power-induced heating of the moderator vessel, while low-pressure operation must remove only the decay heat arising after the mitigative action of reactor shutdown.

Another difference between the low-pressure response mode and the standby mode is the expected method of entry. Warm-up to standby is expected at continuous flow at constant pressure at relatively constant power with a switch from HD to LD circulation. Warm-up to the low-pressure response mode will be at nominally zero power under the highly transient blow-down conditions of hydrogen expulsion to reach the required lower pressure. The blow-down would be accomplished by warm LD high-pressure hydrogen resident in the charging tank pushing the cold HD hydrogen loop inventory into the low-pressure storage tank. Transient two-phase conditions will exist during the blow-down before the establishment of continuous forced gas flow with the LD circulator.

The demonstration of a controlled transition from the normal to the standby operating mode (at 15 bar and full reactor power using only the LN_2 -based cooling feature of the refrigerator) has been (and will be) a large focus of intended systems level experiments (Sect. 4.7) and analytical efforts (see Sect. 3.4). Similarly, successful transitions to the low-pressure response mode from the normal mode of operation also must receive confirmatory experimental and analytical evaluations.

For all abnormal events considered in this evaluation, special interpretation of the barrier level assessment is used if an event is one where helium slowly leaks into a vacuum region. If the vacuum level can be maintained despite detectable helium leaking into it, then the barrier level is judged as if the leak did not exist. This measure is consistent with the intent of Requirement 11 as no H_2 is involved and at least one intact H_2 barrier remains. This approach is not followed if O_2 (or air) leaks (and is detected) into an evacuated space housing very cold H_2 . This is because of the potential for condensation and buildup of solid O_2 and N_2 on the cryogenic (17–20 K) H_2 surfaces (see Sect. 3.2.3). Such condensation (cryopumping) would maintain/enhance the level of vacuum, would render vacuum level monitoring useless for detecting leaks, and would increase the mass of oxidizer available for reaction with any subsequently leaked H_2 .

A measure of the effectiveness of the safety strategy developed for the HFIR cold source is to judge if an event that leads to zero intact hydrogen barriers is preceded by a reactor shutdown. Reactor shutdown must occur prior to such events as hydrogen/air reactions, and the risk of their impact upon HFIR systems must be minimized (Safety Requirement 11, Sect. 1.2.3).

It is assumed that any failure/leak of the beam tube wall or vacuum tube wall that is detected will require reactor shutdown. These structures constitute reactor coolant primary and secondary pressure boundaries (as required for all beam tubes) for which simultaneous failures would most likely result in a pathway into the beam room for reactor coolant. No credit is taken in this simultaneous failure event for the neutron windows (on the vacuum tube and the outer mouth of the beam tube) as full reactor pressure (33 bar = 482 psia) would (by assumption) fail them.

It is also assumed that reactor shutdown will be required if forced-flow cooling of the moderator vessel is threatened. This response measure has been proposed as a means of reducing the heat load on the moderator vessel walls so that passive heat transfer mechanisms (radiation and conduction through a He gas gap between the moderator vessel and the vacuum tube) can prevent overheating and melting of the vessel walls. Such response actions (reactor shutdown and flooding the vacuum tube with helium) are necessary in order to avoid the requirement of alternative/backup forced-flow cooling in the event that

normal forced-flow cooling is lost. Loss of forced-flow cooling could occur under any of three situations. These are (1) loss of all circulators, (2) loss of all refrigeration, or (3) loss of H₂ inventory.

For reactor shutdowns initiated by loss of forced-flow cooling of the moderator vessel, it is further assumed for this evaluation that cold source depressurization and subsequent operation under the low levels of heating associated with reactor shutdown are available as additional mitigative response actions. The benefits and advantages of such low-pressure operation have been discussed previously in connection with H₂ leaks. This response action would be taken in recognition of the residual heating occurring after shutdown and the potential increase in temperature ($T_{\text{wall-max}} \sim 500 \text{ K}$)¹⁸ and loss of strength of the moderator vessel walls. Wall stress reduction would be accomplished by directly decreasing the internal pressure of the moderator vessel (via loop depressurization) and, if required to enhance the external heat transfer from the moderator vessel, directly increasing the external pressure surrounding the moderator vessel by flooding the vacuum tube with helium. [Such cold source operational responses (e.g., depressurization plus helium flooding) after a reactor shutdown because of loss of forced flow cooling of the moderator vessel may not be required for all such loss of cooling initiators. If analytical and experimental evidence indicates continued wall integrity during 15-bar post scram heat-up, then the cold source loop could be maintained at the full 15-bar pressure after reactor shutdown without normal forced flow cooling of the moderator vessel.]

It is assumed that there is a double-walled neutron window at the end of the vacuum tube and that the interstitial space between the walls is filled with helium. The inner wall of the window is adjacent to the high vacuum of the vacuum tube vacuum space, while the outer wall is adjacent to the helium-filled volume of the beam tube extension. The walls of the vacuum tube neutron window are the second and third hydrogen boundaries between the hydrogen inside the moderator vessel and the beam room atmosphere. The purpose of the intra-wall helium region is to provide a means of detecting flaws in the inner wall of the neutron window as helium would flow through the leak into the high vacuum space inside the vacuum tube.

Another important assumption involves the region of the HB-4 beam tube adjacent to the reactor pool/biological shield and at the beam room interface. This portion of the design has not reached maturity and is still being developed. It has been assumed for this evaluation that the reactor pool/biological shield region of the beam tube is filled with helium at subatmospheric pressure. Preliminary analysis has indicated that the effect on the cold neutron beam of the presence of the helium is minimal. It is assumed that the beam tube mouth is closed by a neutron window maximizing cold neutron passage from the gas space into the beam room experiments. If this window were to fail, it is assumed that no flow obstruction would exist between the beam tube helium space adjacent to the reactor pool and the beam room atmosphere.

The presence of helium in the beam tube extension region is desirable. It effectively reduces the concentration of any air that may enter the vacuum tube vacuum space through a failed double-wall neutron window of the vacuum tube. It therefore reduces the inventory of air that might be involved in a subsequent reaction with hydrogen and reduces the quantity of chemically active species (such as ozone produced by irradiation of air) that may spontaneously react even in the absence of hydrogen.

It has been suggested that the helium pressure in the beam tube extension be raised to slightly greater than one atmosphere. This would ensure that if a leak occurred, the initial leakage would always be out of the system, which would further reduce the risk of air entering the system. However, at the writing of this report, the impacts of the higher pressure helium on the cold neutron beam and the stress loads on the aluminum windows have not been evaluated, and thus this higher pressure option is not included in this reference design report. Use of higher pressure helium will be addressed early in the detailed design process.

The double-walled neutron window on the vacuum tube and the single-wall window on the beam tube are assumed to represent effective barriers to hydrogen and to reactor pool water at its hydrostatic pressure, but as mentioned above, not to reactor coolant at 33 bar (482 psia). Detailed design of these neutron windows has not been completed, and so assumptions regarding their pressure loading capability

must be made. However, it is clear that they should be relatively thin in order to maximize cold neutron passage through them, but strong enough to withstand some degree of pressurization. It is also clear that use of two independent walls to form the window at the end of the vacuum tube and a single-wall window at the end of the beam tube results in a total of four barriers between the hydrogen in the moderator vessel and beam room atmosphere. In the following discussions concerning event responses, full credit is taken for the fourth barrier (i.e., the beam tube neutron window). However, in the event of beam tube neutron window failure, no follow-up actions to protect the reactor or cold source are warranted because the three required design barriers of the cold H₂ remain, and no threat exists to any nuclear safety function.

It is assumed that backup electrical power supplies are available only for nuclear safety-related loop components (see Sect. 3.6), the hydrogen circulators (battery backup for a single circulator, either HD or LD), and the pressure/inventory control system. This is undesirable as other cold source loop components will not be powered (e.g., refrigeration system, nonnuclear safety-related vacuum systems, and the charging pump for the high pressure feed tank) and thus will not function under loss-of-power scenarios. This will require reactor shutdown as long as the moderator vessel forced-flow cooling function is degraded.

For events discussed in the following, it is assumed that multiple events do not occur simultaneously because of the assumed independent nature of the failures. Simultaneous failures complicate loop behavior and are highly unlikely. Also, common cause failure modes may induce simultaneous events but are not considered.

The functions, requirements, and responses of cold source equipment, along with expected reactor responses, are deduced for events previously identified (in Sect. 3.3 and ref. 3) and discussed in the following sections.

3.5.1 HB-4 Beam Tube Wall

The beam tube wall represents a portion of the reactor coolant primary pressure boundary. It also represents a portion of the third hydrogen barrier for the cold H₂ inside the hydrogen bearing structures of the vacuum tube. The HB-4 beam tube, both inside the reactor vessel and outside the reactor vessel (e.g., along the beam tube extension) must withstand the potential pressure loading resulting from a maximum credible H₂/air detonation (Sect. 3.2.2) inside the reactor vessel portion of the beam tube (Requirement 19, Sect. 1.2.3). Internal structures along the path of the beam tube such as the vacuum tube, the vacuum tube double-walled neutron window, and the beam tube neutron window are not required to withstand such loadings and would be expected to fail.

The portion of the beam tube outside the reactor vessel is assumed to be filled with subatmospheric helium gas during normal operation of the cold source loop, but is to be flooded with water for radiation shielding during periods of maintenance activities around HB-4 in the beam room. As a result, hydrogen events initiated within this portion of the beam tube are considered beyond design basis. However, as mentioned above, this portion of the beam tube must survive the maximum credible H₂ detonation event initiated inside the reactor vessel portion. Such a requirement is necessary to preclude reactor pool flooding of the beam room.

The reactor vessel portion of the beam tube must not buckle at an external design pressure of 1000 psig and (as is the case of the current in-vessel HB-4 beam tube wall) will be periodically subjected to external pressures of 850 psig as part of the HFIR reactor vessel hydro testing.

Failure of the in-vessel portion of the beam tube is estimated at a frequency of 6×10^{-4} /year (λ_{LBT} from ref. 3). Section 3.3 reports the combined failure rate of the proposed new beam tube wall and rupture disk (intended to provide HB-4 internal volume over pressurization relief) consistent with the current operating HB-4 beam tube design as $\ll 1 \times 10^{-6}$ /year. The current evaluation assumes that the rupture disk does not exist but is replaced by a relief valve in order to conserve reactor coolant inventory after pressure is relieved.

The vacuum tube is designed to withstand the maximum reactor pressure of 690 psig externally (690 psig is the set point of the main reactor coolant system pressure relief devices) and should perform as a backup reactor coolant pressure boundary. The beam tube/vacuum tube annulus relief valve serves to protect the vacuum tube wall and to prevent continuous beam tube leakage (at reactor coolant pressure of 482 psia) after the internal HB-4 beam tube volume pressure falls below the relief valve set point. The annulus is currently proposed to be normally filled with a mixture of H_2O and helium to provide cooling of the vacuum tube wall. The annulus relief valve set point is assumed at a value of 690 psig. (This set point takes credit for the full pressure bearing capability of the vacuum tube and does not permit further discharge of reactor coolant even if the beam tube leaks at normal reactor coolant pressure.) The relief valve line is connected to the vacuum tube mounting flange, and relief valve flow discharges into the reactor pool. A leak in the beam tube wall would be detected by an increase in pressure of the annulus fluid.

Little pressure effect is expected in the reactor coolant system at the time of an in-vessel beam tube wall leak/break. This is because of the presence of the small expansion volume of the filled H_2O/He annulus and the pressure-bearing capacity of the vacuum tube wall and the vacuum tube mounting flange. The system supplying the H_2O/He coolant to the annulus region must also be able to withstand the increase in pressure because of a leak in the beam tube wall. It is expected that the HFIR will be shut down in the event of a detectable in-vessel beam tube leak, because it implies that a break in the reactor coolant primary pressure boundary has occurred. The reactor safety system may scram in the event that the beam tube/vacuum tube annulus relief valve operates. The scram signal would be reactor coolant pressure decrease, identifying a loss-of-coolant accident (LOCA). The flow through the annulus relief valve is permitted to be at a maximum of 100 gpm by technical specification limit (see Requirement 18, Sect. 1.2.3). This limit maintains the break flow from the reactor system within the SBLOCA domain documented in the HFIR SAR. A leak in the beam tube wall is expected to result in a transition of the cold source to the standby mode from the normal mode after reactor shutdown. The number of cold hydrogen barriers decreases to two after a beam tube leak.

The failure frequency of the reactor pool/biological shield portion of the beam tube wall (i.e., the beam tube extension) is estimated to be less than $6e-4$ /year and is not considered in the analysis of Sect. 3.3. The reduced failure/leakage frequency (relative to the in-vessel portion) derives from the lower external pressure exerted by the reactor water pool compared to the in-vessel portion. A failure in the ex-vessel portion would permit H_2O from the reactor pool to enter the subatmospheric helium space. Credit is taken in the ability of the beam tube single-wall neutron window and the vacuum tube double-wall neutron window to withstand the hydrostatic head of the reactor pool.

A failure/leak in the reactor pool portion of the beam tube wall would be detected by the degradation of the cold neutron beam. The response of such a failure of the outer beam tube is to shut down the HFIR, as only one barrier (the thin beam tube neutron window) would remain to prevent loss of pool water and subsequent beam room flooding. The cold source loop is expected to transition to the standby mode after reactor shutdown following an event of this nature.

A leak/failure in the beam tube neutron window is estimated to occur at a frequency of 0.0084/year (Φ_{All} from ref. 3). A failure of the beam tube neutron window would permit beam room air to enter and mix within the beam tube extension helium volume. H_2 /air reaction inside the vacuum tube is still prevented by presence of the double-wall vacuum tube neutron window and by continued integrity of the moderator vessel and transfer line walls. If a continuous purge of the subatmospheric helium region existed, then the air concentration inside the beam tube extension may be reduced. The cold neutron beam will be detrimentally affected as air mixes with helium and the cold source loop transitions into standby mode as the number of cold HD H_2 barriers remains at three. Such operation at 100 K precludes any possibility of O_2 and N_2 condensation onto the cold H_2 structures inside the vacuum tube, and thus enhances the detection capability (via RGA) of any subsequent air leak into the vacuum tube vacuum space. No reactor response is required as the moderator vessel continues to remain cooled, and the reactor water pool primary boundary (beam tube extension wall) remains intact.

It should be noted that gas discharge into the beam room near the mouth of HB-4 requires careful evaluation. This region would receive the gaseous effluent from any uncontrolled mishap inside HB-4. The released gas mixture could contain H_2 , He, and/or H_2O . Any H_2 leaked would require rapid dilution to preclude combustion (H_2 mole fraction must remain less than 4 %). The SBHE and hot off-gas (HOG) systems exhaust the beam room and utilize charcoal filters that may be susceptible to H_2 events in the ducting. An analysis of the beam room and SBHE system consequences of gases released from an HB-4 event is planned as part of the final safety analysis work.

A separate hardened protective duct for a portion of the hydrogen transfer line located in the reactor building is currently being planned. Its purpose is the mechanical support and protection of the multiwalled cold hydrogen transfer line in the pony motor battery room and the electrical equipment room and the isolation of any externally leaked hydrogen (from the housed transfer line) from the atmosphere of these rooms. It has been suggested that the cost/benefit of passing inert gas (N_2) through the duct be examined. This purging through the duct would dilute and inert any controlled amount of H_2 leakage into the duct prior to venting outside. It has also been suggested that the gas streams associated with the vacuum tube/ H_2 transfer line vacuum system exhaust, the vacuum tube rupture disk discharge, and any leakage into the annulus between the transfer line and the support pipe wall be received by this hardened duct. This hardened duct will be independent of the SBHE and HOG systems. The protective duct will consist of a thick-walled pipe within which the multiwalled H_2 transfer line will lie. Such a design further minimizes the possibility of mixing of H_2 with reactor building air and represents a fourth barrier between H_2 and reactor building air for a significant portion of the system inventory of H_2 during normal operation.

3.5.2 Vacuum Tube

The cold source vacuum tube encloses the moderator vessel and two single-wall H_2 lines containing the flowing hydrogen necessary to cool the neutrons originating in the HFIR core and to cool the intensely heated moderator vessel walls. Discussed in this section are several aspects of the cold source vacuum tube. These include the vacuum tube wall, the H_2O/He interstitial region between the vacuum tube and the beam tube, the internal vacuum region, and the vacuum tube neutron window.

As discussed in Sect. 3.5.1, the vacuum tube wall represents a secondary pressure boundary for the reactor coolant. It is designed to withstand an external pressure of 690 psig without buckling. The vacuum tube wall also represents the second barrier between cold H_2 in the moderator vessel and the high-pressure reactor coolant. Its exterior surface is cooled by conduction through a proposed water/helium mixture residing in the annulus between the vacuum tube and the beam tube. The vacuum tube wall and the annulus H_2O/He mixture will conduct heat transferred from the external surface of the moderator vessel to the beam tube wall under accident conditions when the reactor is shut down and the moderator vessel is not internally cooled by the forced flow of hydrogen. In turn, the beam tube wall is cooled by reactor coolant at its outer surface.

The water/helium mixture in the annulus between the vacuum tube and the beam tube is intended to serve two purposes during normal operation. First, it cools the vacuum tube wall, which is most intensely heated in the region nearest the reactor core. Second, the helium is intended to act as a leak detector if the vacuum tube wall leaks. The high vacuum inside the vacuum tube serves to cause dissolution and leakage of helium through even very small flaws so that monitoring of the residual gases in the vacuum space will detect the presence of helium and, therefore, flaws. Helium is the standard monitor gas for such applications because of its small molecular size, and it is not susceptible to cryopumping (see Sect. 3.2.3). No monitoring other than pressure is assumed for the H_2O/He mixture in the annulus.

The vacuum tube is assumed to be protected from excessive external pressurization by a relief valve. The current vacuum tube wall is designed to withstand 690 psig externally without buckling. The vacuum tube wall and the vacuum tube mounting flange represent a portion of the second pressure boundary for

the reactor coolant. A relief valve for this annulus is to be located external to the reactor vessel and will exhaust to the reactor pool. The pressure set point is assumed to be at 690 psig.

The frequency of loss of vacuum tube wall cooling (Φ_{LTC}) is not quantified in Sect. 3.3 (or ref. 3). The cause for such an event may be a failure in the H₂O/He pumping system, a relief valve failure, flow blockage, line break, etc. The response to a loss of vacuum tube wall cooling includes a heat-up in the vacuum tube wall and increased radiation heat transfer to the moderator vessel. The loss of H₂O/He inventory should be detectable by annulus pressure change.

Loss of vacuum tube cooling should initiate a cold source transition (from 15 bar operating pressure) to the standby mode. This directly reduces heat generation in the hydrogen inventory within the moderator vessel and precludes O₂ and N₂ condensation on the exposed H₂ bearing structures within the vacuum tube. Reactor shutdown should not be automatically required as no direct threat to the backup reactor coolant pressure boundary (vacuum tube wall) exists if the annulus relief valve remains operable and the vacuum tube wall remains intact. The barrier level of the cold H₂ remains at level three as long as the moderator vessel walls remain coolable in the standby mode. However, if the vacuum tube fails or leaks because of overheating, then reactor shutdown and cold source transition into the low pressure response mode would be required as the alternate reactor coolant pressure boundary has failed. Such a failure in the vacuum tube wall would be detected by RGA monitoring of the vacuum space as well as by the H₂O/He annulus pressure measurement.

The vacuum region inside the vacuum tube serves two functions. First, it provides thermal insulation to the outer surface of the moderator vessel. Second, the absence of gas in the vacuum tube preserves the cold neutron beam leaving the face of the moderator vessel traveling along the length of the vacuum tube. This region requires a high vacuum pumping system capable of 1e-6 torr pressures. As mentioned earlier, the exhaust of this vacuum pumping system will be routed into the H₂ ventilation system. The intake to the vacuum system pumping station will be monitored for H₂ and He leaks through the primary H₂ boundary, the vacuum tube wall, and the inner wall of the vacuum tube double-walled neutron window. The vacuum tube wall and the inner wall of the vacuum tube neutron window will be protected from over-pressurization from within the vacuum tube by a rupture disk. The rupture disk will be located on the inlet line to the vacuum pumping system and will also be exhausted into the H₂ ventilation system. The loss of the vacuum tube vacuum system is addressed in Sect. 3.5.5.

Failure of the vacuum tube wall is estimated to occur at a frequency of 5.7e-5/year (Φ_{BMP} from Sect. 3.3 and ref. 3) and has been recommended as a design basis event. This event will permit He and perhaps H₂O to enter the vacuum tube vacuum space. If severe, this event may induce failure of the moderator vessel walls by thermal shock as water warms the extremely cold (17–20 K) moderator vessel surface. The increase in heat load to the moderator vessel and the in-tube transfer lines will result in the expansion of hydrogen and might challenge the ability of the pressure control system. The occurrence of a vacuum tube wall leak is detected by the presence of He in the vacuum space and a decrease in the annulus pressure. Failure of the moderator vessel wall upon failure of the vacuum tube wall may be inferred by the simultaneous appearance of H₂, He, and H₂O in the vacuum tube vacuum space, as well as a transitory reduction of pressure in the H₂ loop, an increase in heat load, and a degradation of vacuum in the vacuum tube.

Water leakage into the vacuum tube will have an adverse effect on the cold neutron beam as water freezes onto the moderator vessel. The reactor must be shut down as the alternate pressure boundary for reactor coolant has failed. There would be essentially a LOCA pathway for any high pressure reactor coolant (482 psia) entering the beam tube to pass directly into the beam room as the neutron windows of the vacuum tube and beam tube are not designed to withstand reactor coolant pressure. In addition, a failed vacuum tube wall may not limit reactor coolant leakage to 100 gpm (see Requirement 18, Sect. 1.2.3) if the beam tube wall also fails. The response of the cold source loop to a vacuum tube wall failure (sufficient to permit in-leakage of water) would be to transition to the low-pressure response mode. The cold H₂ barrier level would decrease from three to two in the event of a vacuum tube wall

failure. If the moderator vessel wall also fails because of thermal shock, the barrier level would decrease to one.

As discussed previously, the double-walled vacuum tube neutron window is protected from over-pressurization failure by a vacuum tube rupture disk. The failure frequency of the vacuum tube neutron window is less than 0.0084/year (Φ_{All} from ref. 3) because of the use of a double rather than single wall design. A leak in the inner wall of this window will be detectable as helium residing in the space between the walls expands into the vacuum of the vacuum tube and is detected by the RGA monitoring.

The response to a vacuum tube neutron window failure would be to transition the cold source loop into the standby mode. The increase in heat load on the moderator vessel arising when warm helium from a slow leak comes in contact with the cold moderator vessel wall might be removed by excess refrigeration capacity. The cold neutron beam might be degraded as the cold HD hydrogen in the moderator vessel becomes warmer due to heat transfer permitted by the helium contact with the cold walls. However, the vacuum tube vacuum system would work toward exhausting the helium into the hydrogen ventilation system. The cold H_2 barrier level would decrease from three to two for one-wall failure or from three to one for double-wall failure. (The intact neutron window at the mouth of the beam tube also would serve as a hydrogen barrier so that the minimum number of actual cold hydrogen barriers remaining is two.) In either case, reactor shutdown may not be necessary as neither the vacuum tube wall, the beam tube wall, or the moderator vessel wall are directly threatened. Reactor shutdown and loop transition to the low pressure response mode might be required if cooling of the moderator vessel wall is threatened.

3.5.3 Refrigeration System

The refrigeration system for the HFIR cold source loop has been described earlier in Sect. 2.3.5. Features of the refrigerator include helium as the refrigerant, liquid nitrogen for precooling during normal operation and for primary cooling during the 100 K standby and low pressure response modes, water cooling for secondary heat removal from the compressors, cryogenic valving and tubing, bank of compressors, piston expander box, heat exchanger box, and helium inventory control system.

The cold source refrigeration system is external to the reactor building. Cold helium supplied by the refrigerator is used in the safe room in the H_2/He heat exchanger. The cold box expanders and the cold box heat exchangers are also located outside the reactor building in a building adjacent to the safe room. The refrigerator compressors are housed close to the safe room building outside the reactor building.

Because the cold source operating pressure (15 bar) may exceed the helium refrigerant pressure on the secondary side of the heat exchanger (~ 1 bar), a leak in the heat exchanger could permit hydrogen to enter the refrigeration loop. Hydrogen monitors have been added to the helium inventory control portion of the refrigerator to detect hydrogen (see Requirement 20, Sect. 1.2.3). There is no monitoring of the nitrogen coolant as the nitrogen precooling function is by boil-off. The resulting N_2 gas is vented to the outside atmosphere.

Expander-based loss of the normal cooling function produced by the refrigeration system is expected at a rate of 0.78/year (from Sect. 3.3). This has been recommended as a design basis event. The expanders are the cold source components most likely to malfunction (Sect. 3.3). The quoted frequency does not accurately represent, however, the frequency of the failure to provide an adequate source of hydrogen cooling. As discussed more fully in the information that follows, the refrigeration system has been modified to provide a backup LN_2 -based mode if the expanders have failed. This backup mode is very reliable and is the mode of choice when operating the refrigerator in loop standby mode or in the low-pressure response mode. This LN_2 -based mode of refrigerator operation has not been examined in Sect. 3.3 for frequency of failure.

The impact of a total refrigeration system failure (normal expander piston mode and LN_2 -based mode) is the loss of cooling of the moderator vessel. This leads to a degradation of the cold neutron beam, gradual degradation of the moderator vessel heat transfer, expansion of the loop inventory of

hydrogen, and a challenge to the cold source pressure control system. If the H_2 control system cannot control the pressure of the expanding hydrogen, then the hydrogen inventory will be partially expelled via a relief valve into the H_2 storage tank outside the HFIR reactor building. If the relief valve is insufficient to stop the pressure from increasing, an additional relief valve and a rupture disk are available (at higher pressure set points) to vent additional H_2 inventory directly into the H_2 ventilation system and into the outside atmosphere.

The LN_2 -based backup refrigeration capability has been designed into the cold source refrigeration system to minimize the frequency of total loss of refrigerator function. It requires one (of a maximum of five existing) operating He compressor of the normal refrigerator, electrical power to operate the compressor, a LN_2 supply, the LN_2 precooling heat exchangers of the normal refrigeration system, and the He inventory control portion of the refrigerator. LN_2 is pressure fed into the precooling heat exchangers, and the N_2 vapor produced as a result of heat transfer from the warmer helium is vented to the outside atmosphere. Thus, the backup cooling mode of refrigerator operation is expected to be very reliable.

If the refrigerator backup cooling mode is operable, but the normally active refrigeration mode is not operable, it is expected that the cold source moderator vessel may still be cooled adequately at full reactor power. However, since the boiling temperature of N_2 at atmospheric pressure is 77 K, the cold source H_2 temperature must be sufficiently above 77 K to remove the moderator vessel heat load. Standby operation of the cold source loop at 15 bar pressure and approximately 100 K is considered sufficient. No effect on reactor operation is expected as long as the moderator vessel can remain cooled.

If the backup LN_2 -based refrigeration method is not operable upon loss of normal refrigerator function, the refrigeration system is completely unavailable and the temperature, and pressure of the H_2 inventory of the cold source loop will begin to rise. As described earlier, the inventory control system will attempt to control pressure by releasing cold hydrogen from the pressurizer into the storage vessel. Because the normal operating temperature of the cold H_2 system is 17–20 K, and the high-density and the low-density circulators continue to function, a relatively long period is required to increase the moderator vessel wall temperature to a threatening level. The response in this loss of cooling scenario includes reactor shutdown to reduce heating of the moderator vessel walls and cold source transition into the standby mode. The cold source loop might eventually require transitioning into the low-pressure response mode and flooding the vacuum tube with helium if the moderator vessel is not sufficiently cooled. Recent analysis (ref. 18) indicates that the moderator vessel wall cannot melt ($T_{\text{melt}} \sim 855$ K) under such decay heating conditions.

3.5.4 Hydrogen Circulators

Three circulators are to be available to pump hydrogen around the HFIR cold source loop. Two are available to pump HD hydrogen, while the third is to be used to pump LD hydrogen. During normal operation of the cold source, only one of the HD circulators is in use, with the remaining one available as backup. The LD circulator is used exclusively during the standby and low-pressure response steady-state modes and is used in conjunction with a HD circulator in the transient cooling and warming modes. The cryogenic portions of all three circulators are surrounded by high-level vacuum inside the pump module to reduce heat in-leakage. In turn, the vacuum is encased in an inert helium blanket to prevent in-leakage of air (in particular oxygen and nitrogen) from the outer atmosphere that may cryopump onto the cold H_2 -bearing surfaces exposed to the vacuum. The vacuum space is monitored for hydrogen and helium to detect leaks in either the hydrogen barrier or the helium barrier. The circulators and the pressurizer are located inside the pump module, which in turn is located in the safe room outside the reactor building. Emergency electrical power (from batteries) is assumed to be provided to operate a single circulator (either HD or LD) during loss of normal power.

The loss of circulator function because of loss of the normal and backup HD circulators is estimated at 0.23/year (see Sect. 3.3) and is a recommended design basis event. The loss of both HD circulators combined with a failure of the LD circulator is not addressed in Sect. 3.3.

Detection of loss of forced flow may be directly by a flow measuring device. It will also be detected via diagnostic measurement of circulator performance such as rpm, electric motor power consumption, pressure measurements around the loop, and H_2 temperature exiting the beam tube.

The response to loss of normal mode high-density H_2 circulation at full power conditions is a rapid heating and expansion of H_2 in the moderator vessel. This expanding hydrogen would challenge the pressure control system and might possibly actuate the H_2 relief valve, which would limit the pressure rise by dumping hydrogen from the loop into the storage tank. If the backup high-density circulator cannot be brought on-line to continue forced flow cooling of the moderator vessel, then loop operation would go into standby mode and employ the low-density circulator. This requires that the operating temperature of the hydrogen be increased such that the density is reduced sufficiently so that the gas circulator will pump. The nominal operating temperature of the hydrogen in the standby mode is 100 K. As mentioned in Sect. 3.5.3, it is anticipated that refrigerator operation using only the LN_2 precooling heat exchangers will be sufficient to remove the entire heat load of the cold source loop with the reactor running at full power and the H_2 temperature at a nominal value of 100 K. The moderator vessel heat load in standby mode is expected to be somewhat reduced compared to the normal mode because of the reduction in direct nuclear heating as a result of the density reduction of the hydrogen in the moderator vessel.

In the event that all forced flow circulation (i.e., all three circulators) in the cold source loop is lost, the reactor must be shut down to reduce heat generation in the moderator vessel. The cold source loop might be required to transition to the low-pressure response mode and to flood the vacuum tube with helium if moderator vessel cooling is sufficiently threatened. Loop depressurization in the low pressure response mode would reduce wall stresses in the moderator vessel. Recent analysis has indicated a maximum wall temperature of 500 K which is sufficiently low to preclude melting of the moderator vessel walls during the heat-up associated with decay heating from the reactor core.

3.5.5 Vacuum Systems

This section addresses several vacuum systems associated with the HB-4 cold source. Specifically included are the vacuum systems required to produce the cryogenic and ambient temperature blanketing of the innermost (or primary) hydrogen barriers. The He refrigeration system has its own dedicated vacuum system and is not addressed here.

Two vacuum systems exist in the HB-4 cold source design that are associated with the cold H_2 portions of the system: (a) is for vacuum tube and transfer line evacuation out to the HFIR Building wall and (b) is for evacuation of the vacuum regions of the transfer line between the HFIR Building wall and the pump module, the vacuum region surrounding the structures within the pump module (e.g., the HD and LD H_2 circulators and the pressurizer), the insulation region around heat exchanger, and the H_2 transfer lines to/from the heat exchanger.

Additional vacuum systems might exist to evacuate regions of the inert blanket surrounding the ambient temperature H_2 inventory. These include the feed/bleed H_2 lines between the pressurizer and the high-pressure charging tank and between the pressurizer and the low-pressure storage tank, the low-pressure H_2 storage tank, the higher-pressure charging tank, and the compressor/charging pump required to pressurize the high-pressure charging tank. These vacuum systems would produce only rough vacuums (1e-3 torr) compared to the high-level insulation vacuums (1e-6 torr) of (a) and (b) above.

In addition to the cryogenic and ambient vacuum systems discussed above and associated with the hydrogen, there is another vacuum system associated with the cold source loop that is important to the neutron performance of the cold source. This vacuum system (c) is that associated with the subatmospheric helium region filling the internal volume of the beam tube extension external to the

reactor vessel. Its purpose is to provide a long cold neutron path length such that scattering is minimized as the cold neutrons travel down the length of the tube toward the experiments at the mouth of the beam tube. Unlike the previously described vacuum systems, this system is not required to provide vacuum blanketing of hydrogen in the cold source and is not part of the required multiple barrier concept employed for the cold source. Failure of this system would result only in the degradation of the cold neutron beam delivered to the experiments and would not necessitate either cold source loop or reactor responses. It is not important to either cold source or reactor safety, and therefore it is not discussed further.

Vacuum systems (a) and (b) above that require high vacuum typically employ a roughing vacuum pump and a high-performance turbo-vacuum pump along with fast-acting valves to prohibit gas (air) back-flow into the evacuated space. An RGA monitor is included with each high vacuum system to measure vacuum level (or total pressure), as well as residual gas species partial pressures. The monitor employs a mass spectrometer to identify each individual gas species in the evacuated space. The low vacuum systems require a roughing vacuum pump as well as check valves and a monitor to measure vacuum level. All cold vacuum system components are to be enclosed in a helium blanket to prevent inflow of air from the outside surrounding environment in the event of a leak. Redundancy is provided by the availability of a separate backup unit for each vacuum system associated with the hydrogen system.

Vacuum system (a) will be located inside the HFIR Building. Vacuum system (b) will be located in the safe room outside the reactor building. Vacuum system exhaust for systems (a) and (b), the rough vacuum systems for the ambient parts of the loop, and the vacuum tube rupture disk discharge will all be to the hydrogen-qualified ventilation system located outside the reactor building. The vacuum system exhaust and vacuum tube rupture disk discharge lines will be inert gas blanketed along their entire length both inside the reactor building and outside the reactor building to provide double containment of any hydrogen discharge into these lines. In turn, the hydrogen-qualified ventilation system will be continuously purged with inert gas (N_2) so that any gas discharged into the vent will be rendered noncombustible.

Gross loss of vacuum function because of vacuum pump and vacuum pump check valve failure is a relatively infrequent event at $1.3e-4$ /year/vacuum system ($= \Phi_{LVAC}$ from Sect. 3.3). The most important failures would be the loss of vacuum systems (a) and (b), as the heat load on the cryogenic H_2 bearing structures would increase. Any air drawn into the evacuated space through failed vacuum pump check valves would increase the potential for cryopumping (and cold neutron beam degradation). Vacuum system failures are detected by deterioration of vacuum level for the affected pumping system and the presence of undesirable gases through applicable RGA results.

The response to a failure of vacuum system (a) may include the degradation of the cold neutron beam as gas fills the normally evacuated space. The initial response to failures in systems (a) or (b) should be to switch to the standby vacuum systems provided. If the standby vacuum system cannot be used, then transition the cold source loop to standby mode, especially for small vacuum system leaks. This reduces the nuclear heat load in the moderator vessel even at full reactor power operation and precludes cryopumping of O_2 for systems (a) or (b) onto H_2 bearing surfaces. If vacuum system (a) or (b) leaks, there will be increased heat load on the H_2 system because of the potential loss of vacuum insulation. This might be largely absorbed by the excess cooling capacity of the refrigerator. It is anticipated that no reactor shutdown is necessary as long as the moderator vessel can be cooled. Degradation of the vacuum [vacuum system (a)] inside the vacuum tube because of vacuum system seal leakage is estimated at a frequency of $2.5e-2$ /year and has been recommended as a design basis event (Sect. 3.3).

If the moderator vessel cannot remain cooled in standby mode after a failure/leakage in a vacuum system, then the reactor must be shut down as a threat may exist to the moderator vessel wall. If the moderator vessel still cannot be cooled adequately, the cold source loop should transition to the low-pressure response mode, and if necessary, the vacuum tube vacuum space might require flooding with helium. These actions reduce moderator vessel heating, promote cooling of the moderator vessel walls, and reduce stresses in the moderator vessel walls.

3.5.6 Electrical Power Supply

The cold source electrical supply is normally taken from the HFIR off-site power supply and is routed through the normal HFIR power distribution system. Current cold source design provides nonnuclear safety-related emergency/backup battery-based power to a single hydrogen circulator (either HD or LD), and perhaps to the computer controller to enhance reliability. Electrical backup power must also be made available to nuclear-safety-related systems such as the vacuum tube vacuum system, and, if necessary, the safety-related portions of the H₂ ventilation system.

Important electrically powered cold source components include the two HD circulators, the LD circulator, the refrigeration system, the vacuum pumps for vacuum systems and associated RGAs, the vacuum tube H₂O/He coolant system, the pump/compressor for charging the high-pressure H₂ gas reservoir, the cold source pressure/temperature control and data acquisition computer, cold source loop instrumentation, and ventilation fans for the safe room. It is assumed that there are no fans or dampers requiring electrical power associated with the hydrogen vent system.

The frequency of loss of all off-site power is estimated at 0.44/year (Φ_{LOSP} as per Sect. 3.3) and is the second highest event frequency examined for the cold source system. Unless provided with an uninterruptible supply, loss of electrical power to cold source loop instrumentation tied in with the reactor safety system would result in reactor scram as they are required to fail (either physically or through loss of power) in a way that results in a reactor scram signal (see Requirement 16, Sect. 1.2.3).

Consequences of the loss of off-site power are the loss of hydrogen pressure and temperature control as a result of the failure of the charging compressor, the loss of the refrigeration system, and the loss of some vacuum systems. Reactor power would be reduced as is currently required, and the emergency diesel generators for the reactor would be started. The loss of pressure control in the H₂ system might result in the periodic dumping of some loop inventory of H₂ into the surge tank as the passive loop relief valve limits the system pressure.

In addition to any direct scram associated with loss of power to cold source scram instrumentation, the loss of all cold source refrigeration (both normal and backup) because of the loss of off-site power might require a cold source initiated reactor shutdown as the moderator vessel loses cooling and the loop heats up. (The LN₂ cooling backup mode of refrigerator operation is assumed not available because of lack of an emergency power supply to the helium compressor bank.) If the moderator vessel cannot be adequately cooled after reactor shutdown, then the loop must transition into the low-pressure response mode. It is not expected that the standby mode will be attainable in the absence of normal electrical power as the refrigerator, the pressure control system, and the charging compressor for the high-pressure hydrogen reservoir will not be operable. Flooding of the vacuum tube with helium might be required.

It should be noted that emergency power will be required for any electrically operated device associated with the inert (N₂) purge gas of the hydrogen ventilation system. This is necessary to provide continuous venting capability for leaked H₂ from any cold source equipment in the reactor building (even when the reactor is shut down). It is required that emergency power be available to meet any demands of the vacuum tube H₂O/He cooling system. This is necessary in order to ensure a continuous satisfactory heat sink for radiative and conduction heat transfer from the external surface of the moderator vessel wall.

Provision of backup power such that H₂ can be circulated through the loop during a reactor shutdown (caused by loss of off-site power) will result in nearly uniform heat-up of the entire H₂ system and therefore increases the time required to reach threatening temperatures of the moderator vessel wall. However, the reactor must remain shut down as long as electrical power is unavailable to key cold source systems such as the refrigerator system and the H₂ pressure/inventory control system. In addition, once off-site power is restored, the reactor must remain shut down until such time that the cold source moderator vessel can be cooled in either the normal or the standby mode. This may require a rather long period as restoration of cold source systems (such as vacuum pump-down or hydrogen cool-down) in

their fully powered design operating states might be a rather lengthy process (from AEDC experience, 24 h is representative of a full hydrogen cool-down from ambient temperature).

3.5.7 Hydrogen Barriers

This section discusses expected responses to leaks and failures of H₂ barriers (both cryogenic and ambient) of the HFIR cold source. Of primary interest are the innermost hydrogen barriers of the multibarrier (either three for cold or two for warm) design. Components of the barriers include the moderator vessel wall; transfer lines and couplings; HD and LD circulators and isolation valves and seals; heat exchanger; pressurizer; charging (transfer) pump/compressor and seals; large, low-pressure H₂ gas storage vessel; the smaller H₂ gas high-pressure charging vessel; and the pressure/inventory control valving and lines.

3.5.7.1 Moderator Vessel Walls

Moderator vessel wall failure will most likely be because of the small wall thickness and relatively low strength of 6061-T6 aluminum alloy during loss of forced flow cooling conditions. Reduction in strength at high temperature ($T_{\max} \sim 500$ K from ref. 18) requires the depressurization of the H₂ loop to reduce moderator vessel wall stresses. The frequency of failure is estimated at $1.3\text{e-}4/\text{year}$ (Φ_{BMV} as per Sect. 3.3) and has been recommended as a design basis event. The results of stress analysis at normal steady-state operating temperature ($\sim 17\text{--}20$ K) and pressure (15 bar) indicates a narrow but acceptable margin of safety.

Vessel wall failure because of over-pressurization at normal mode operating temperature requires the malfunction of the relief valve installed on the hydrogen gas transfer line between the pressurizer and the H₂ gas storage tank and the failure of the relief valve/rupture disk combination to dump the loop H₂ inventory into the H₂ venting system. Such a moderator vessel wall failure would be detected by the presence of H₂ in vacuum tube vacuum space.

The response to such failure is that the cold source neutron beam would degrade as neutron scattering is no longer optimized. Hydrogen entering the vacuum tube vacuum space from a small leak would be exhausted from the vacuum space to the hydrogen ventilation system by the vacuum pump. If the break was large enough that the vacuum space vacuum pump could not maintain vacuum, pressure build-up in the vacuum tube vacuum space would eventually burst the rupture disk protecting the vacuum tube neutron window and force GH₂ through the vacuum tube vacuum/relief system into the H₂ ventilation system.

If the H₂ were not sufficiently vented through the rupture disk, the double-wall vacuum tube neutron window and the single-wall beam tube neutron window might be over-pressurized, fail, and lead to gaseous hydrogen (GH₂) injection directly into the beam room at the mouth of HB-4. There, the GH₂ would tend to enter the HOG/SBHE exhaust system and combine with a large volume of air. Hydrogen reaction with air cannot be dismissed a priori in either the beam room or the HOG/SBHE exhaust system and requires further evaluation.

For a detectable leak in the moderator vessel wall, the reactor must be shut down and the cold source loop must transition into the low-pressure response mode. If the moderator vessel wall cannot remain cooled in the presence of leaked hydrogen gas in the vacuum space, then helium gas should be added to enhance heat transfer. These actions are generally in response to the zero tolerance H₂ leak policy, but, as described earlier (Sect. 3.5), require further assessment and evaluation with respect to precise conditions of response. These response actions are necessary to reduce subsequent leakage of H₂ into the vacuum tube vacuum space, reduce moderator vessel wall stresses, and preclude overheating of the moderator vessel walls.

A catastrophic failure of the moderator vessel might lead to direct contact of cold H₂ with the vacuum tube wall, a thermal-shock failure of the vacuum tube wall, and, perhaps, subsequent cold H₂

contact with the HB-4 beam tube wall. As much as 5 kg of H₂ is currently contemplated for use in the HB-4 cold source loop. Such thermal and hydrodynamic interactions will require more detailed evaluations to estimate responses and consequences.

3.5.7.2 Heat Exchanger Walls

A leakage in the heat exchanger boundary is estimated at 1.3e-2/year (Φ_{HXB1} from Sect. 3.3) while the rupture frequency is 7.9e-3/year (Φ_{HXB2} from Sect. 3.3). It appears that, under all operating (15 bar) modes, the H₂ (primary-side) pressure will be greater than the He (secondary side) pressure. A leak will then result in H₂ ingress into the He refrigeration system. Such a failure would be detected by a drop in H₂ loop pressure and the presence of H₂ at the inlet of the He refrigerator. Hydrogen is likely to freeze in the He system expansion pistons and may cause damage to the expander cold box. Requirement 20 (Sect. 1.2.3) imposes H₂ detection and pressure control/ventilation measures to be met for the refrigeration system. The refrigerator controller will most likely shut down the refrigerator if H₂ is identified in the He working fluid.

In response to heat exchanger failure, the neutron beam may or may not be appreciably affected as the loop controller initially attempts to maintain the desired operating system H₂ pressure by adding warm H₂ gas to the loop. A hydrogen leak would eventually require, however, transition into the low-pressure response mode in an effort to reduce H₂ loss and mechanical stresses in the moderator vessel wall. It would require a shutdown of the reactor as the capability to provide forced flow cooling of the moderator vessel might be jeopardized because of loss of H₂ inventory and loss of refrigeration. A large leak might also require that the vacuum tube vacuum space be flooded with He to prevent potential decay-heat-induced overheating of the moderator vessel wall.

3.5.7.3 H₂ Transfer Lines

A leakage within the H₂ transfer lines inside the reactor building is estimated to occur at leak frequency of 2.0e-2/year (from Sect. 3.3) and at a frequency of 7.7e-2/year outside the reactor building. It represents the frequency of failure of the innermost H₂ barrier composing the multiwalled transfer line and does not necessarily imply that HD hydrogen is permitted to enter the atmosphere surrounding the transfer line. Depending on the magnitude of the leak, the H₂ inventory control measures may prevent H₂ from uncontrollably escaping the cold source system. Such a leak within the portion of the transfer line located in the reactor building (failure frequency = 2.0e-2/year, Sect. 3.3) has been recommended as a design basis event.

The transfer line gross failure frequency is 3.0e-6/year inside the reactor building and is 1.5e-6/year outside the reactor building (from Sect. 3.3) and represents the estimated frequency of failure of all three barriers between the HD hydrogen and the atmosphere surrounding the transfer line. Hydrogen would enter the atmosphere surrounding the transfer line. Again, depending on the magnitude and location of the break, venting measures may prevent H₂ from escaping the cold source system in an uncontrolled manner (e.g., operation of the safe room emergency fans). Such a catastrophic failure of the transfer line in the beam room (frequency=3.0e-6/year) has been recommended as a design basis accident (Sect. 3.3).

Because a vacuum layer lies between the cold supercritical HD hydrogen and the He cover gas, an internal transfer line H₂ leak would be identified by H₂ detection in the associated vacuum space. The response to a line leak into a vacuum region would be to exhaust the leaked H₂ into the H₂ ventilation header by the vacuum pumps (small leaks) or by actuation of a rupture disk (large leaks). In response to the zero tolerance H₂ leakage policy, the loop should be transitioned to the low-pressure response mode, and the reactor should be shut down as the cold source moderator vessel cooling may be threatened by loss of H₂ inventory. After reactor shutdown, flooding the vacuum tube with helium may be required to prevent overheating of the moderator vessel walls.

The response to an internal line He leak into the vacuum region of the line would be to continue reactor and cold source operation as long as moderator vessel cooling can be maintained. The H_2 barrier level remains at two for the cold H_2 . The He from small leaks may be exhausted by the vacuum pumps fast enough that the vacuum level may be retained. In addition, the added heat load to the cold hydrogen because of vacuum insulation degradation may be largely removed by excess refrigeration capacity. If the moderator vessel cannot remain cooled because of vacuum insulation degradation, the cold source loop should be transitioned into standby and proper cooling of the moderator vessel should be checked. If adequate cooling is still not provided, the reactor power might be reduced in steps to investigate the adequacy of available moderator vessel cooling. If cooling is not established with staged reactor power reductions, the reactor should be shut down and the cold source loop transitioned into the low pressure response mode. Finally, if insufficient moderator vessel cooling exists, the vacuum tube will be flooded with helium.

Because the vacuum space of the vacuum tube and the vacuum space of the H_2 transfer line within the HFIR Building are currently proposed to be common, the source of helium present in this vacuum space cannot be easily identified. Three immediate sources of helium exist in the current proposed design. They are: (1) the He mixed with H_2O in the gap between the vacuum tube and the beam tube, (2) the He cover gas inside the concentric walled in-building H_2 transfer line, and (3) the helium-filled region between the double walls of the vacuum tube neutron window. Because the origin of the helium for such a leak cannot be identified with certainty, one must assume a He leak into the vacuum region comes from the structure most critical to reactor safety—the vacuum tube wall. The response actions to such a helium leak are as stated above.

For hydrogen leaking outside of the transfer line, the response would be to transition the loop into the low-pressure response mode, shut down the reactor, and flood the vacuum tube vacuum volume with helium. Moderator vessel cooling may be directly threatened, or there may be hydrogen directly entering the beam room atmosphere (or the safe room atmosphere or the atmosphere outside the reactor building).

A leakage of hydrogen externally of the multiwalled transfer line will not be directly detectable except in the safe room where dedicated hydrogen sensors will be located in the fan exhaust line to the hydrogen release stack. However, the possibility of externally leaking H_2 should be detectable as an internal leak because of the design of the transfer line. Thus the H_2 should be directly detectable by the RGA for the corresponding vacuum space prior to entering the atmosphere outside the transfer line and appropriate operational response actions taken. A helium leak outside the transfer line may only be indicated through pressure monitoring of the He space.

External H_2 or helium leakage from the transfer line inside the pony motor battery room or the electrical equipment rooms of the reactor building is very unlikely. However, a special hardened hydrogen protection/mounting duct is being planned to house the transfer line through these two rooms. Such a protective/mounting duct is feasible because of the long length of straight line possible in these rooms. The annulus area between the transfer line and the duct wall is to be sealed from reactor confinement at the interface with the beam room and most likely will be open directly to the outside atmosphere at the interface with the outer wall of the reactor building. This thick-walled duct will serve to protect the line from mechanical threats inside these rooms and will serve as a defacto fourth boundary between H_2 and the atmosphere of these rooms. This protective duct is intended to house safely any combustion event occurring outside the transfer line inside the duct so that absolutely no additional threat will exist to nuclear safety-related equipment located in these rooms.

It is noted that in the current cold source design the length of concentric tube H_2 transfer line in the beam room is not to be located within a hardened hydrogen protective duct. Such a protective duct is considered impractical in the beam room as sharp turns will exist in order to route the line to the HB-4 beam tube mouth area. It is also considered unnecessary from a hydrogen release standpoint as three hydrogen barriers already exist in the design. However, an external leak of H_2 or He from this portion of the transfer line would pass directly into the beam room atmosphere. Reactor building SBHE/HOG ventilation ducts located in this room most likely would draw any leaking gas into the SBHE/HOG

ventilation system, dilute it with building air, pass it through a series of charcoal filters, and exhaust it to the outside atmosphere through the plant stack. Such a line failure into the beam room has been recommended as a design basis event (frequency= 3.0×10^{-6} /year, Sect. 3.3) and requires further evaluation. The operational response to an external break in the beam room portion of the cold H₂ transfer line is the same as that above. Shut down the reactor, transition the loop into the low-pressure response mode, and flood the vacuum tube vacuum volume with helium. There may be a direct threat to beam room equipment and personnel, as well as to moderator vessel wall cooling.

3.5.7.4 Transfer Line Couplings

Leaks may also occur in the transfer line couplings. For the current design, couplings exist in three locations. These are located (tentatively) as follows: inside the vacuum tube near its attachment to the beam tube; inside the reactor building but near its southeastern external wall; and inside the pump module. The purpose of these couplings is to either bring together the separated single-wall HD hydrogen lines into the multiple concentric wall transfer line geometry or to provide the physical interfaces between the various vacuum/vacuum and helium/helium regions along the HD hydrogen boundary represented by the multiwalled transfer line.

A leak of H₂ or He into the vacuum space from the transfer line coupling inside the vacuum tube or pump module is estimated to have a frequency of 1×10^{-2} /year (Φ_{SL2} from ref. 3). It would be detected by RGA monitoring of the affected vacuum space. The response to a H₂ leak would be to shut down the reactor, transition to the cold source low-pressure response mode, and if necessary, flood the vacuum tube vacuum space with helium. The response to a small He leak is to do nothing as long as vacuum is maintained; for a larger He leak that degrades vacuum, the proper response would be to transition into the cold source standby mode for as long as the moderator vessel can remain cooled. Loop cold hydrogen barrier level remains at a minimum of two. If cooling cannot be maintained, transition to the low-pressure response mode, shut down the reactor, and if required to further enhance moderator vessel coolability, flood the vacuum tube vacuum space with helium.

The coupling inside the reactor building joining separate sections of the concentric tube transfer line should be located if possible inside the protective hydrogen transfer line duct so that any leaked gas external to the coupling does not pose a threat. That such a coupling inside the building is necessary is because of the requirement that the helium and vacuum layers surrounding the HD hydrogen inside the line be distinct for the portions inside and outside the reactor building. In turn, the requirement for distinction between these two portions of the transfer line derives from the nuclear safety classifications of cold source equipment inside and outside the reactor building (see Sect. 3.6).

3.5.7.5 Other H₂ Boundary Components

Leaks may develop from other H₂ boundary components. Such locations include the sealing surfaces found in the circulator isolation valves, pressure control valves, circulator seals, etc. All components subject to such leakage are located (by design) outside the reactor building. The overall leak frequency is 0.077/year (from Sect. 3.3). Such leaks would be detected by H₂ gas entering the vacuum space blanketing the cold hydrogen-bearing structure or by H₂ gas entering the inert blanket (either rough vacuum or inert gas) surrounding the ambient temperature hydrogen-bearing structure. All vacuum or inert gas blankets immediately surrounding ambient temperature hydrogen-bearing structures will be monitored for leaks by simple pressure indication. The leaked H₂ is exhausted by the corresponding vacuum pump into the H₂ ventilation header or simply pressurizes the ambient temperature inert gas cover to the point that it actuates the inert blanket pressure relief device and is discharged into the H₂ ventilation system. The hydrogen barrier level is reduced from that at normal operation and the loop is transitioned into the low pressure response mode, the reactor is shut down, and, if needed, the vacuum tube is flooded with helium.

Explicit recognition has been given in Sect. 3.3 to the leakage/failure event frequencies associated with the cold source system storage tank. The leakage (< 5 g/s) frequency is given as $6.0\text{e-}3/\text{year}$, while the large leakage (~ 800 g/s) or failure frequency is given as $6.0\text{e-}6/\text{year}$. This component is of special interest as it is large (15 m^3), is proposed to be located relatively close to the reactor building, and during the cold source shutdown, standby, and low-pressure response modes contains a predominant fraction of the cold source system inventory of hydrogen (about 5 kg). This structure remains pressurized with respect to the outside atmosphere during the steady-state shutdown, standby, and low pressure response modes and during portions of the transient cooling and warming modes. A large scale rupture of the storage vessel has been recommended as a design basis event in Sect. 3.3.

Uncontrolled release of hydrogen from the storage vessel to the outside atmosphere requires the failure of two inventory control measures. These are a rupture disk that directly protects the vessel inner containment from internal over-pressurization and another rupture disk that protects the inert blanket region between the vessel inner and outer walls. Leaks into the blanket region between the two walls will be detected via gas pressure monitoring. The consequences of a worst case detonation involving the entire 5 kg with outside building air near the storage vessel are currently unknown. However, threats might be posed to the reactor building wall, to the exposed reactor secondary coolant line connecting the reactor main heat exchangers to the cooling tower, and to the various buildings housing cold source equipment. In addition, uncontrolled leaks of cold hydrogen gas into the outside atmosphere may be close enough to the air intakes of the electrical building and the air intake of the pony motor/electrical equipment rooms of the reactor building to affect air quality. These possibilities will be investigated more in future assessments.

3.5.8 Seismic and Wind/Tornado Events

The design basis earthquake (DBE) is 0.15 g for new HFIR facilities such as the cold source. The H_2 boundary failure frequency is $3.3\text{e-}5/\text{year}$ because of a DBE (Φ_{SEIS} from ref. 3). Requirement 16 (Sect. 1.2.3) defines conditions associated with a DBE. All reactor coolant pressure boundaries, H_2 barriers located within the reactor building, and auxiliary cold source loop equipment located inside the reactor building must survive an earthquake and remain functional. These include the HB-4 beam tube wall, the vacuum tube wall, pressure relief systems for the vacuum tube vacuum space and the vacuum tube/beam tube annulus, the moderator vessel wall and supporting structures, the in-building H_2 transfer lines and couplings, the vacuum tube $\text{H}_2\text{O}/\text{He}$ cooling system, the vacuum tube/transfer line vacuum system, the vacuum tube helium flooding system, the vacuum tube double-wall neutron window and the beam tube single-wall neutron window, the protective duct enclosing the multiwall hydrogen transfer line in the pony motor battery room and the electrical equipment room, all components of the hydrogen ventilation system (including the inert gas purging system) necessary for venting of in-building H_2 leaks, and any electrical power source required for the above equipment.

It should be noted that the refrigeration system is not subject to this requirement. Thus, the He and backup N_2 cooling portions of the refrigeration system may be permitted to lose He and N_2 inventories during and following a DBE and/or wind/tornado events. In addition, all (with the exception of the portion of the H_2 vent system associated with exhausting cold source gas leaks from within the reactor building to the outside atmosphere) portions of the primary barriers of the H_2 system outside the reactor building are excluded from these seismic and tornado requirements. They may fail and leak H_2 into the outside atmosphere, given the occurrence of such an event. The impact and consequences of such an occurrence will be addressed in future analyses. Of particular concern is the potential impact of a H_2/air detonation outside the reactor building.

The response to a seismic event includes a currently existing reactor scram upon detection of an earthquake. Heat addition to the cold source will diminish after the scram. Several cold source loop process variables will be monitored by the reactor safety system so that the forced flow cooling status of the cold source moderator vessel can be continuously ascertained. If this moderator vessel cooling

function is not maintained following an earthquake or tornado (due to failure of cold source equipment or systems outside the reactor building), an additional reactor shutdown signal will be generated by the cold source such that the reactor will not be allowed to start up. In addition, upon recognition of the loss of moderator vessel cooling function, the cold source loop will transition into the standby, and, if necessary, the low-pressure response mode. Finally, the vacuum tube might be flooded with helium in an effort to prevent overheating and failure of the moderator vessel walls. These actions explicitly recognize the loss of strength of aluminum 6061-T6 material at the elevated temperature that may be reached ($T_{\max} \sim 500$ K estimated from ref. 18) during a loss of moderator vessel cooling event.

It should also be noted that the loop controller and instrumentation are to be designed such that they fail safe. Therefore, if they are disabled during an earthquake or windstorm (computer loop controller located in room outside of the reactor building), the cold source loop will automatically be depressurized and the H_2 inventory will be vented into the storage tank outside the reactor building. Such a transition into the low pressure response mode would be accompanied by a reactor shutdown.

If a small H_2 leak were to occur from cold source equipment located inside the reactor building because of an earthquake, vacuum and ventilation systems have been designed to contain the H_2 such that leakage into the building atmosphere is minimized. By design, such a controlled H_2 leak is vented to the outside atmosphere through an inerted hydrogen-qualified ventilation system designed to withstand the DBE.

3.5.9 Nonoperational Events

Nonoperational events may involve a H_2 /air reaction caused by change out of the H_2 inventory in the cold source system or by inadvertent release of H_2 into the surrounding atmosphere during gas-handling operations. Because the HB-4 cold source system is a closed system (i.e., the hydrogen inventory is continuously housed and recycled within the loop), such change-out operations will be infrequent. However, change-out events cannot be totally eliminated and could be caused by human error in following H_2 change out/handling procedures. The frequency of human error is estimated at $4.7e-3$ /operation (Φ_{GHE} from Sect. 3.3 and ref. 3), and the frequency of air entering the H_2 system during such a gas-handling operation is $2.8e-4$ /year ($=\Phi_{A12}$ from ref. 3).

If a H_2 /air reaction occurs in the safe room during the handling operation, there are no cold source loop consequences other than damage to safe room equipment and possibly to personnel at the time of the reaction. The reactor threat should be minimal as it and the cold source loop would be shut down. However, a large safe room detonation event might pose some threat to the reactor building walls as the safe room building is designed to be frangible and will not provide protection to the reactor building wall. This threat will require further assessment.

If an H_2 /air reaction occurred inside the H_2 system, then the pressure pulse will possibly have an effect on the cold source components within the confines of the reactor building. A worst case pressure pulse might propagate along the transfer line from outside the reactor building to inside the reactor building, fail the moderator vessel inside the vacuum tube, fail the double-wall vacuum tube neutron window, fail the beam tube neutron window, and enter the beam room. However, the reactor and the cold source loop would (most likely) not be operational at the time of the mishap, and so the impact of the event is expected to be limited.

In addition, if air enters the H_2 system and does not immediately react with H_2 at the time of the gas-handling operation (due to O_2 concentration being below the reaction threshold), it is expected that it would subsequently freeze in the heat exchanger and might affect heat exchanger performance.

3.6 CLASSIFICATION OF COLD SOURCE EQUIPMENT: SAFETY, QUALITY, AND SEISMIC

Equipment of the HB-4 cold source has been classified with respect to requirements in three areas, which are safety, quality, and resistance to seismic events. The goal was to define and establish cold source equipment requirements such that acceptable levels of performance, reliability, and safety can be achieved for cold source operation. The approach used in this effort is similar to that used for the cold source facility at BNL.¹⁹

It is assumed that the reactor coolant boundaries modified/impacted by the cold source are also to be included in the assessment. Thus the HB-4 beam tube wall is included in the classification process.

It is also noted that the operational actions of reactor shutdown and flooding of the vacuum tube vacuum space with helium (ref. 18) are sufficient to preclude melting of the moderator vessel wall in the event that all forced flow cooling of the moderator vessel is lost. Additionally, depressurization of the under-cooled moderator vessel from full pressure (15 bar) decay heating conditions might also increase the margin of safety extant in the thin moderator vessel wall. (The transfer path followed for heat generated in the moderator vessel wall under such conditions is conduction and radiation to the vacuum tube wall, conduction through the H₂O/He-filled annulus between the vacuum tube wall and the beam tube wall, conduction through the beam tube wall, and convection to the reactor coolant.) Reactor shutdown is assumed to be taken in the current evaluation for any period when the forced flow cooling function of the moderator vessel normally provided by the three H₂ circulators, or the H₂/He heat exchanger, or both of the refrigerator cooling modes is lost. The actions of helium flooding and loop depressurization are also assumed available as further mitigative measures. These operational actions directly protect the integrity of the cold source hydrogen boundaries and minimize any threat to HFIR posed by the cold source.

That the moderator vessel can be precluded from melting also relieves the cold source cooling systems of serving any nuclear safety function if the HFIR is scrammed upon detection of a loss of moderator vessel cooling event. It admits the possibility that all cold source equipment located outside the HFIR Building (with one exception identified below) may be classified as nonnuclear safety-related. It is therefore assumed for the current classification effort that the HFIR safety system will be modified to be capable of producing a scram in the event of loss of cold source moderator vessel cooling function. It is believed that operational measures designed to provide redundancy in function such as switching to the solely LN₂-cooled mode of refrigerator operation or switching among the three H₂ circulators will minimize the number of reactor scrams required. Nevertheless, a total loss of forced-flow cooling condition may arise if (1) all three H₂ circulators fail, (2) if both (expander based and LN₂-cooling based) modes of refrigerator operation fail, or (3) if there is a loss of H₂ inventory such as would occur in a transfer line break.

If such a reactor shutdown response does not exist, it is clear that two additional avenues must be pursued with respect to cold source design and development. First, additional measures of redundancy must be provided to reduce to an acceptable level the occurrence frequency of total loss of moderator vessel cooling function expected under normal conditions. Second, a fully qualified moderator vessel cooling system must be provided to ensure moderator vessel coolability under all abnormal conditions, including those arising from external events such as seismic, fire, or tornado transients.

Sections 3.6.1, 3.6.2, and 3.6.3 present the guidelines used in determining the safety, quality, and seismic ratings for cold source equipment. Sources for this information include the American Nuclear Society (ANS) standard concerning safety criteria,²⁰ a quality document developed and used at BNL for quality level guidance (ref. 19), and previously identified HFIR cold source equipment safety requirements with respect to seismic considerations (see Sect. 1.2.3).

Section 3.6.4 presents the results of the classification process and the basis for the classification for each piece of cold source equipment.

3.6.1 Safety Classification Guidelines

The following information is excerpted from the ANS standard (ref. 20) concerning recommendations for safety classification of equipment and systems at boiling water reactors (BWRs). The recommendations are for the most part independent of reactor type and have been adopted for application to the HFIR cold source. General criteria are identified for use with each safety class. Three safety classes (SC-1, SC-2, SC-3) are established and are relied upon to accomplish nuclear safety functions. A fourth safety class, nonnuclear safety (NNS), is defined for all safety-related equipment not required to accomplish nuclear safety functions. The NNS class is still a safety class, even though it is a class not relied upon for nuclear safety.

Because the HB-4 cold source uses H_2 as the working fluid, the dangers of H_2 /air reactions are paramount in considering loop, reactor, and personnel safety. Therefore, the safety classification process must heavily consider this particular hazard when using the classification guidelines from the standard.

Before the safety class guidelines are presented, a definition of a nuclear safety function is presented from the standard (ref. 20). A nuclear safety function is one "that is necessary to ensure:

1. The integrity of the reactor coolant pressure boundary,
2. The capability to shut down the reactor and maintain it in a safe shutdown condition, or
3. The capability to prevent or mitigate the consequences of Plant Conditions that could result in potential offsite exposures that are comparable to the guideline exposures of the Code of Federal Regulations, Title 10, 'Energy,' Part 100, 'Reactor Site Criteria'[2]."

The following four sections are taken from the standard (ref. 20) regarding safety class guidelines and serve as the basis used for defining the HFIR HB-4 cold source equipment (nuclear and nonnuclear) safety classifications presented in Sect. 3.6.4.

3.6.1.1 Safety Class 1

Safety Class 1 (SC-1) shall apply to pressure-retaining portions and supports of mechanical equipment that form part of the reactor coolant pressure boundary (RCPB) whose failure could cause a loss of reactor coolant in excess of the reactor coolant normal makeup capability and whose requirements are within the scope of the *ASME Boiler and Pressure Vessel Code*, Section III.

3.6.1.2 Safety Class 2

Safety Class 2 (SC-2) shall apply to pressure-retaining portions and supports of primary containment and other mechanical equipment, requirements for which are within the scope of the *ASME Boiler and Pressure Vessel Code*, Section III, that is not included in SC-1 and is designed and relied upon to accomplish the following nuclear safety functions:

1. Provide fission product barrier or primary containment radioactive material holdup or isolation.
2. Provide emergency heat removal for the primary containment atmosphere to an intermediate heat sink, or emergency removal of radioactive material from the primary containment atmosphere (e.g., containment spray).
3. Introduce emergency negative reactivity to make the reactor subcritical (e.g., standby liquid control).
4. Ensure emergency core cooling where the equipment provides coolant directly to the core (e.g., residual heat removal and emergency core cooling).
5. Provide or maintain sufficient reactor coolant inventory for emergency core cooling (e.g., suppression pool).

3.6.1.3 Safety Class 3

Safety Class 3 (SC-3) shall apply to equipment, not included in SC-1 or -2, that is designed and relied upon to accomplish the following nuclear safety functions:

1. Provide for functions defined in SC-2 where equipment, or portions thereof, is not within the scope of the *ASME Boiler and Pressure Vessel Code*, Section III.
2. Provide secondary containment radioactive material holdup, isolation, or heat removal.
3. Except for primary containment boundary extension function, ensure hydrogen concentration control of the primary containment atmosphere to acceptable limits.
4. Remove radioactive material from the atmosphere of confined spaces outside primary containment (e.g., control room or fuel building) containing SC-1, -2, or -3 equipment.
5. Introduce negative reactivity to achieve or maintain subcritical reactor conditions.
6. Provide or maintain sufficient reactor coolant inventory for core cooling.
7. Maintain geometry within the reactor to ensure core reactivity control or core cooling capability (e.g., core support structures).
8. Structurally load-bear or protect SC-1, -2, or -3 equipment.
9. Provide radiation shielding for the control room or off-site personnel.
10. Ensure required cooling for liquid-cooled stored fuel (e.g., spent fuel storage pool and cooling system).
11. Ensure nuclear safety functions provided by SC-1, -2, or -3 equipment (e.g., provide heat removal for SC-1, -2, or -3 heat exchangers, provide lubrication of SC-2 or -3 pumps, provide fuel oil to the emergency diesel engine).
12. Provide actuation or motive power for SC-1, -2, or -3 equipment.
13. Provide information or controls to ensure capability for manual or automatic actuation of nuclear safety functions required of SC-1, -2, or -3 equipment.
14. Supply or process signals or supply power required for SC-1, -2, or -3 equipment to perform their required nuclear safety functions.
15. Provide a manual or automatic interlock function to ensure or maintain proper performance of nuclear safety functions required of SC-1, -2, or -3 equipment.
16. Provide an acceptable environment for SC-1, -2, or -3 equipment and operating personnel.

3.6.1.4 Nonnuclear Safety

NNS shall apply to equipment that is not included in SC-1, -2, or -3. This equipment is not relied upon to perform a nuclear safety function.

3.6.2 Quality Level Guidelines

The guidelines followed for ensuring adequate quality for HB-4 cold source equipment originated from those presented in the BNL High Flux Beam Reactor (HFBR) quality assurance document (ref. 19). Quality is judged with respect to potential impact on the entire facility, including both personnel and financial resources. Four levels of equipment and component quality are characterized below.

3.6.2.1 Quality Level 1

Quality Level 1 (QL-1) is associated with critical items requiring the highest degree of reliability in performance. It is assigned to mechanical items designated SC-1. Typical Quality Level 1 items include the following:

1. components within the reactor vessel;
2. the primary system pressure boundary;
3. the plant protection system;
4. the scram system; or
5. items not identified as part of reactor safety systems, the failure of which could seriously damage experiments or preclude operation of the facility so that it cannot achieve its programmatic mission.

This quality level is also considered for items that are SC-2, SC-3, or NNS that are judged to have a major programmatic impact on the HFIR.

3.6.2.2 Quality Level 2

Quality Level 2 (QL-2) is associated with major items requiring a high degree of reliability in performance. It is assigned to mechanical systems designated SC-2. Typical Quality Level 2 items include the following:

1. decay heat removal and emergency core cooling systems;
2. the poison water addition system;
3. instrumentation and control systems needed to assure systems in (a) and (b) above are performing properly; or
4. nonsafety systems, the failure of which could damage experiments or significantly affect the programmatic mission of the facility.

This quality level is also considered for items that are SC-3 or NNS that are judged to have a moderate programmatic impact on the HFIR.

3.6.2.3 Quality Level 3

Quality Level 3 (QL-3) is associated with minor items requiring a degree of reliability and performance above that achieved from application of standard industry practices and codes. It is assigned to mechanical items designated SC-3. Typical Quality Level 3 items include the following:

1. the reactor building confinement structures and systems;
2. cooling water systems for residual heat removal and emergency core cooling;
3. instrumentation and control systems that monitor the release of radioactive materials from the primary pressure boundary; or
4. the nuclear incident alarm system.

This quality level is also considered for items that are not directly related to reactor safety (e.g., NNS) that are judged to have a programmatic impact primarily related to cost. In addition, it may be desirable to assign Quality Level 3 to certain mechanical items designated NNS whose function(s) may facilitate operation of the reactor or the mitigation of postulated accidents.

3.6.2.4 Quality Level 4

Quality Level 4 (QL-4) is associated with other items for which the reliability and performance achieved by applying standard industrial codes and practices is acceptable. This quality level is assigned to all reactor facility items that are not related to reactor safety and are not designated as QL-1, -2, or -3.

3.6.3 Seismic Criteria

Section 1.2.3 presents the safety requirements and goals of the HB-4 cold source. One of these safety requirements relates to equipment status during and after a DBE. Currently, the DBE for new facilities at the HFIR is 0.15 g. The cold source seismic safety requirement is repeated as follows:

“Requirement 16: Components that form any of the multiple hydrogen boundaries inside the reactor building shall be seismically qualified or shall be shown by analysis to maintain hydrogen containment integrity following a design basis earthquake (DBE). The hydrogen vent system handling hydrogen leaks from equipment inside the reactor building shall be designed to withstand seismic and wind/tornado events. The vacuum tube helium flooding system shall be seismically qualified. Support structures and systems located within the reactor building that ensure the normal functions of the above cold source systems shall also be seismically qualified. Active components of the hydrogen system and the helium refrigeration system shall not be required to function during and following a DBE and thus do not require seismic qualification. All cold source instrumentation and operation controllers shall be designed to ‘fail safe;’ [i.e., depressurization and H₂ inventory expulsion into the storage tank should result automatically upon any controller failure (including failure due to a DBE)]. In addition, any cold source loop instrumentation interfacing with the HFIR safety system shall be designed to fail (seismic or nonseismic) in a manner such that the resulting signal ensures reactor shutdown.”

In addition to the cold source equipment seismic safety requirement, HFIR Technical Specification 2.1 identifies the safe shutdown earthquake (SSE) safety limit of 0.15 g for reactor coolant boundaries. Thus, requiring the primary (beam tube) and secondary (vacuum tube) reactor coolant pressure boundary to maintain their integrity at a design basis of 0.15 g automatically meets the current seismic safety limit.

The current classification effort examines each component with respect to the above cold source seismic safety requirement and determines if the equipment is subject to the requirement or not subject to the requirement. A seismic-yes or seismic-no response is given in Sect. 3.6.4.

3.6.4 HB-4 Cold Source Equipment And Component Classifications

The equipment and components of the HB-4 cold source system were reviewed for the purpose of identifying the various safety classes (SC), quality levels (QL), and seismic classifications. A discussion of the classification basis for each piece of equipment or component is also presented. A discussion of expected equipment and loop response under off-normal events has been presented in Sect. 3.5 and serves to identify the various safety functions (nuclear and nonnuclear) and HFIR interfaces required.

The presentation is in list format and roughly follows that of Sect. 3.5. A summary of the classification results is given in Table 3.12.

3.6.4.1 HB-4 Beam Tube

In-vessel portion (SC-1, QL-1, and seismic-yes). Its nuclear safety function is to serve as the primary pressure boundary for the reactor coolant. It also protects reactor core structures from the harmful effects of any credible H₂/air detonation. Any device used to mount the in-vessel portion of the beam tube to the reactor vessel wall is also SC-1. This portion of the beam tube also serves as the third design barrier between cold H₂ in the moderator vessel and the reactor coolant.

Reactor pool/bio-shield portion of the beam tube or the beam tube extension (SC-2, QL-1, and seismic-yes). Its nuclear safety function is to prevent drainage of the reactor pool into the beam room in the event of a beam tube H₂/air reaction. It serves to provide isolation of radioactively contaminated pool water from the HFIR containment. It also acts to direct any large amount of H₂ leakage into the beam

Table 3.12. HB-4 cold source equipment classification results

System, component, or structure	Safety class	Quality level	Seismically qualified
HB-4 beam tube wall:			
In-vessel wall	SC-1	QL-1	Yes
Extension region wall	SC-2	QL-1	Yes
Neutron window	SC-3	QL-3	Yes
Water tube	NNS	QL-4	No
Extension region helium fill system	NNS	QL-2	No
Vacuum tube:			
Wall	SC-3	QL-1	Yes
He flooding system	SC-3	QL-3	Yes
Double-wall neutron window and intrawall helium system	SC-3	QL-3	Yes
Rupture disk	SC-3	QL-3	Yes
Rupture disk discharge tubing	SC-3	QL-3	Yes
Beam tube/vacuum tube annulus:			
Relief valve	SC-3	QL-1	Yes
relief valve discharge tubing	SC-3	QL-3	Yes
H ₂ O/He cooling system	SC-3	QL-2	Yes
Primary H ₂ barriers (inside reactor building):			
Moderator vessel wall	SC-3	QL-1	Yes
H ₂ transfer lines (including all support systems for internal vacuum and helium blankets) and couplings	SC-3	QL-1	Yes
Primary H ₂ barriers (outside reactor building):			
H ₂ transfer lines (and support systems) and couplings	NNS	QL-2	No
HD/LD circulators, valves, seals	NNS	QL-2	No
Pressurizer	NNS	QL-2	No
Storage vessel	NNS	QL-2	No
High pressure charging vessel	NNS	QL-2	No
Transfer pump, seals	NNS	QL-2	No
H ₂ inventory control valving/lines	NNS	QL-2	No
H ₂ /He heat exchanger	NNS	QL-2	No
Refrigerator:			
Normal mode:	NNS	QL-2	No
LN ₂ storage and precoolers, expansion pistons, compressors, valving, heat exchangers, pressure/vacuum control			
LN ₂ backup mode:	NNS	QL-2	No
LN ₂ storage, LN ₂ coolers, heat exchanger bypass line, compressor			

Table 3.12 (continued)

System, component, or structure	Safety class	Quality level	Seismically qualified
Vacuum systems (excluding refrigerator): (Includes: suction/exhaust lines, check valves, vacuum pumps, vacuum level, RGA, helium blanket)			
(a) Vacuum tube/H ₂ transfer line	SC-3	QL-2	Yes
(b) Safe room/pump module: HD/LD circulators, pressurizer, H ₂ transfer line, heat exchanger box	NNS	QL-2	No
(c) HB-4 beam tube extension	NNS	QL-2	No
Ventilation system:			
In-building H ₂ transfer line protective duct	SC-3	QL-2	Yes
Ex-building H ₂ release stack	SC-3	QL-2	Yes
Inert gas purging of H ₂ vent	SC-3	QL-2	Yes
Safe room ductwork, fans, dampers	NNS	QL-2	No
refrigerator	NNS	QL-2	No
Remaining cold source equipment and structures located outside reactor building:			
Heating, air-conditioning,	NNS	QL-2 ^a	No
Roof, floor, walls, equipment support structures	NNS	QL-2 ^a	No
Reactor scram-related instrumentation (high/low pressure, high vessel H ₂ temperature)	SC-3	QL-1	No
Computer controller	NNS	QL-2	No
Nonscram loop instrumentation (flow, pressure, temperature, rpm, current)	NNS	QL-2	No
Electrical power supply (N or E)			
Scram-related instruments	NNS	QL-1	No
Nonscram instruments	NNS	QL-2	No
Computer controller	NNS	QL-2	No
Safe room equipment (HD/LD circulator)	NNS	QL-2	No
Refrigeration room, equipment, (compressors)	NNS	QL-2	No
Vacuum systems (vacuum level, RGA, valves, pumps)			
(a) Vacuum tube	SC-3	QL-2	Yes
(b) Pump module	NNS	QL-2	No
(c) Beam tube extension	NNS	QL-2	No

^aThis equipment must be intrinsically safe if it is housed in or is part of a room containing H₂.

room by providing a vent path for gases exiting the vacuum tube through failed neutron windows to the HFIR Building ventilation system. The internal volume serves normally as a monitored subatmospheric helium-filled isolation boundary preventing beam room air from entering undetected into the vacuum tube vacuum region. Finally, it serves as the third design barrier between cold H₂ in the moderator vessel and reactor bay air above the reactor pool.

Beam tube water tube (NNS, QL-4, seismic-no). Its safety function is to provide a water-tight boundary for flooding the reactor pool/bio-shield portion of the beam tube during nonoperational periods when shielding is required for radiation protection at the mouth of the beam tube.

Beam tube extension helium system (NNS, QL-2, seismic-no). This system serves no nuclear safety function. It exists to provide a long path-length for cold neutrons traveling through the beam tube extension region toward the experiments at the mouth of the HB-4. The strict quality level (QL-2) is required in order to ensure that the warm helium in this region does not become contaminated sufficiently to degrade the cold neutron beam passing through. The helium for the beam tube extension region will most likely be maintained by vacuum system (c) of Sect. 3.6.4.7.

Beam tube neutron window (SC-3, QL-3, seismic-yes). Its nuclear safety function is to ensure isolation of the contaminated reactor pool water from the beam room. It serves as a fourth barrier between cold H_2 in the moderator vessel and beam room air. This window acts as a boundary for the subatmospheric helium filling the beam tube extension region and must be constructed to minimize cold neutron beam degradation. It is also assumed to prevent reactor pool flooding of the beam room in the event of beam tube extension wall failure. Therefore, this component must survive an earthquake in order to continue to provide this second radiological barrier function.

3.6.4.2 Vacuum Tube

Vacuum tube wall (SC-3, QL-1, seismic-yes). Its nuclear safety function is to serve as a fully qualified alternate/backup pressure boundary for reactor coolant (it does not bear reactor coolant pressure in normal operation). It must withstand 690 psig externally without buckling. It also serves as the second design barrier between cold H_2 in the moderator vessel and reactor coolant. The vacuum tube wall forms a portion of the vacuum insulation region boundary surrounding the moderator vessel. This component must survive the DBE to maintain a secondary pressure boundary for reactor coolant and to provide a potential heat sink during reactor shutdown for moderator vessel cooling through helium flooding gas in the vacuum tube vacuum space.

Vacuum tube vacuum space helium gas flooding system (SC-3, QL-3, seismic-yes). Its nuclear safety function is to provide emergency helium gas to the vacuum tube vacuum space so that the moderator vessel will remain coolable (due to conduction and radiation heat transfer from the moderator vessel through the helium to the vacuum tube) and not melt in the event that forced-flow cooling of the moderator vessel is lost and the reactor is shut down. It must remain fully functional after the DBE in order to continue to assure cooling capability.

Vacuum tube neutron window intra-wall helium blanket system (SC-3, QL-3, seismic-yes). Its nuclear safety function is to provide leak detection capability for the inner wall of the vacuum tube neutron window. Thus, it supports the nuclear safety functions of the moderator vessel, the vacuum tube wall, and the vacuum tube neutron window. Helium in this intra-wall region will pass through a flaw in the inner wall of the window into the high vacuum region of the vacuum tube surrounding the moderator vessel and be detected by residual gas monitoring of the vacuum space. This system must survive a DBE to continue to provide such leak detection capability.

Double-wall vacuum tube neutron window (SC-3, QL-3, seismic-yes). Its nuclear safety function is to serve as a second and third design barrier between the cold H_2 in the moderator vessel and the beam room atmosphere. The inner wall of the window serves as the second barrier and forms a portion of the vacuum insulation region boundary surrounding the moderator vessel. The outer wall of the neutron window serves as the third cold H_2 barrier and lies adjacent to the subatmospheric helium-filled region of the beam tube extension. This double-wall window must survive the DBE in order to continue to provide controlled venting capability of the vacuum tube vacuum space to the H_2 ventilation header. An intact neutron window after the DBE also ensures the boundary for helium gas flooding of the vacuum space of the vacuum tube upon loss of forced-flow cooling capability.

Vacuum tube vacuum region rupture disk (SC-3, QL-3, seismic-yes). Its purpose is to vent GH_2 from the vacuum tube vacuum region in the event that the moderator vessel experiences a large leak and pressurizes the vacuum space. Its nuclear safety function is to prevent H_2 from entering the beam room atmosphere. It also serves to protect the double-wall vacuum tube neutron window from becoming over-

pressurized by a gas leak into the vacuum tube vacuum space. This component must survive the DBE in order to continue to provide controlled venting capability of the vacuum tube vacuum space to the H₂ ventilation header.

Vacuum tube vacuum region rupture disk discharge tubing (SC-3, QL-3, seismic-yes). Its purpose is to pass vented GH₂ from the vacuum tube vacuum region directly (bypassing the vacuum pump) into the H₂ qualified ventilation system outside the reactor building. Its nuclear safety function is to prevent discharged H₂ from entering the beam room atmosphere. This component must survive the DBE in order to continue to provide venting capability of the vacuum space region.

3.6.4.3 Equipment Servicing the H₂O/He Annulus Between The Beam Tube And The Vacuum Tube

Relief valve (SC-3, QL-1, seismic-yes). Its nuclear safety function is to provide external over-pressurization protection to the vacuum tube wall ($P_{\max} = 690$ psig) due to a leak in the beam tube wall. Relief valve operation also acts to conserve reactor coolant inventory once annulus pressure falls below the pressure relief set point. It acts as an interface between the annulus structures (e.g., SC-1 beam tube and SC-3 vacuum tube walls) and the SC-3 tubing conducting relief valve discharge to the reactor pool. This component must survive the DBE to ensure reactor coolant primary and secondary pressure boundary integrity and the nuclear-safety-related cooling function of the vacuum tube wall.

Relief valve discharge tubing (SC-3, QL-3, seismic-yes). Its purpose is to transport He/H₂O annulus relief valve discharge flow to the reactor pool. Its nuclear safety function is to ensure the integrity of the beam tube and vacuum tube walls and, thus, the primary and backup pressure boundaries of the reactor coolant. This component must survive the DBE in order to continue to ensure reactor coolant primary and secondary pressure boundary integrity and the nuclear-safety-related cooling function of the vacuum tube wall.

H₂O/He mixture vacuum tube cooling system (SC-3, QL-2, seismic-yes). Its purpose is to continuously provide the H₂O/He mixture required to cool the vacuum tube wall (a SC-3 component). Wall cooling is by thermal conduction from the vacuum tube wall through the He/H₂O layer into the beam tube wall. Therefore, its nuclear safety function is to ensure the integrity of the vacuum tube wall. It must withstand a maximum of 690 psig H₂O/He pressure without leaking. Helium suspended in the mixture serves as a leak checking medium for the vacuum tube wall since high-level vacuum exists inside the vacuum tube. It must survive the DBE to ensure continued cooling of the vacuum tube wall. The vacuum tube wall is an intermediate heat sink for reactor shutdown emergency cooling of the moderator vessel once forced-flow hydrogen cooling of the moderator vessel wall ceases.

3.6.4.4 Primary H₂ Barriers Inside the HFIR Building

Moderator vessel wall (SC-3, QL-1, seismic-yes). Its nuclear safety function is to confine H₂ and prevent it from entering into the beam room or the reactor coolant. At reactor power, it must be forced-flow cooled to ensure the nuclear safety functions provided by the vacuum tube wall (SC-3) and the beam tube wall (SC-1). Unmitigated overheating of the moderator vessel will lead to melting, degradation, and H₂ release, which in turn might directly threaten the vacuum tube and beam tube wall integrity. It also serves as the first design barrier between cold H₂ and reactor coolant. This component must survive the DBE to continue to provide a barrier between the H₂ and beam room air.

H₂ transfer lines (and support systems) and couplings inside reactor building (SC-3, QL-1, seismic-yes). The nuclear safety function of the H₂ transfer lines and couplings is to confine H₂ and prevent it from entering into the beam room, the pony motor battery room, or the electrical equipment room. At reactor power, these lines enable the transfer of supercritical H₂ coolant to and from the moderator vessel to prevent overheating and thus ensure the nuclear safety function provided by the vacuum tube wall and the beam tube wall. These multiwalled concentric transfer lines constitute the first,

second, and third hydrogen barriers whose containment functions are monitored via internal blankets of vacuum and helium. Vacuum surrounds the primary (first) and helium surrounds the secondary (second) hydrogen boundary. The support systems required to produce the internal vacuum (see vacuum system "a" of Sect. 3.6.4.7) and helium blankets for the H₂ transfer lines inside the reactor building are included implicitly in this classification. The in-building transfer lines (and their supporting vacuum and helium blanket systems) must survive a DBE to ensure continued segregation of H₂ from the beam room atmosphere.

3.6.4.5 Primary H₂ Barriers Outside the HFIR Building

H₂ transfer lines and couplings outside reactor building, HD/ LD circulators, isolation valves, and seals (NNS, QL-2, seismic-no). The control of full power moderator vessel heat removal is not a nuclear safety function because of proposed trips of the reactor on indication of the loss of forced-flow cooling. Shutdown forced-flow heat removal from the moderator vessel is not a nuclear safety function because it is not essential. Recent calculations (ref. 18) show moderator vessel wall cooling during reactor shutdown conditions via passive conduction/radiation to the vacuum tube wall sufficient to preclude moderator vessel melting. The potential for H₂ detonation inside the reactor building is not a concern in the event of a line break outside the reactor building, because the break will depressurize the H₂ system and reduce the hydrogen inventory to the point where any oxygen ingress is minimal and cannot fail the moderator vessel (**confirmatory calculation to be performed**). The potential for moderator vessel melting due to a break outside the building, or loss of flow, or loss of heat sink for the moderator volume is minimized by proposed trips of the reactor on high temperature of H₂ leaving the moderator vessel, low H₂ system pressure, and low values of loop pressure difference.

In addition to the reactor trips, two additional measures are proposed to reduce the failure potential of the moderator vessel walls when the reactor is shut down because of loss of forced-flow cooling of the moderator vessel. First, the vacuum tube vacuum space could be flooded with helium to enhance heat transfer from the moderator vessel to the vacuum tube wall. Second, the cold source loop could enter the low pressure response mode (see Sect. 3.5). Both actions will reduce wall stresses in the moderator vessel and explicitly address the reduction in strength occurring at elevated wall temperatures such as might develop under loss of forced-flow cooling.

The QL-2 classification arises due to the recognition that failures in such equipment could have a significant impact on programs other than the cold source that also depend directly on operation of the HFIR. Thus, significant effort should be expended to ensure maximum operational reliability of these cold source components so that the impact on HFIR availability because of loss of the cold source forced-flow cooling function is minimized.

Pressurizer, low pressure ambient temperature storage vessel, high pressure charging vessel, transfer pump, seals, pressure/inventory control valving, and lines (NNS, QL-2, seismic-no). The justification is the same as that discussed for the previous component.

He-cooled heat exchanger (NNS, QL-2, seismic-no). The justification is the same as that discussed for the previous component.

3.6.4.6 Refrigerator

The entire refrigeration system is classified as nonnuclear safety-related. This is because failure of the refrigeration system has no impact on nuclear-safety status of the reactor. This is because of (1) location of the system outside the reactor building, (2) a proposal for reactor shutdown based upon a loss of cooling of the moderator vessel because of total refrigerator failure, (3) the provision of a passive means of cooling the moderator vessel under reactor shutdown conditions to prevent melting of the vessel walls, and (4) the provision of a depressurization capability of the H₂ system in the event of an unacceptably high moderator vessel wall temperature following the total loss of forced-flow cooling of

the moderator vessel. The QL-2 designation arises from the same consideration presented in Sect. 3.6.4.5 above.

Normal expansion-based system (including components to support LN₂ precooling, expansion piston bank, compressor bank, control valving, heat rejection, pressure control, vacuum level control) (NNS, QL-2, seismic-no). Its NNS function is to actively remove heat from the cold source loop H₂ inventory through the main heat exchanger in a controlled manner. Part of the heat load might be rejected in the HFIR cooling tower via secondary cooling water through the heat rejection portion of the compressor bank. The remaining heat removed by the refrigerator is rejected either to the outside environment via LN₂ boil-off and venting in the precooling heat exchangers or as external work performed in the expanders. It serves as the second and third barriers between H₂ in the heat exchanger and air in the safe room.

LN₂ backup system (including LN₂ tank, cold box piping, valving, and at least one operational He compressor in the refrigerator) (NNS, QL-2, seismic-no). Its NNS function is to provide an alternative mode of refrigerator operation suitable for removing heat from the hydrogen inventory in the main heat exchanger at a hydrogen temperature of approximately 100 K. In the event the expansion portion of the refrigerator malfunctions, the expansion pistons may be bypassed and the refrigerator operated simply as an LN₂-cooled He heat exchanger with forced He flow provided by a single operational refrigerator compressor. Such a mode of refrigerator operation provides partial redundancy in heat removal function while the cold source loop operates in standby mode at ~100 K. Redundancy in heat removal function is only partial because both modes of refrigerator operation are vulnerable to some common mode failures such as loss of electrical power and helium transfer line failure.

3.6.4.7 Vacuum Systems Excluding the Refrigerator

Vacuum tube/H₂ transfer line vacuum system (a) (including suction and exhaust lines, check valves, RGA and vacuum level monitoring, vacuum pumps, and vacuum pump helium blanket) (SC-3, QL-2, seismic-yes). Its purpose is to provide vacuum-based thermal insulation to the exterior surfaces of the moderator vessel and to the surfaces of the primary hydrogen boundaries inside the length of multiwalled cold H₂ transfer line located within the reactor building. Therefore, its nuclear safety function is to ensure the integrity of the hydrogen boundaries by reducing heat in-leakage and by preventing hydrogen from entering the reactor building atmosphere. It provides means for detection and control of leaks of H₂ from the primary hydrogen boundary and of leaks of helium from the outer boundary of the vacuum region. The vacuum pump exhaust of leaking H₂, He, or H₂O vapor into the H₂ ventilation header maintains isolation of any leaked H₂ from beam room air.

Vacuum system (b) for the pump module (HD/LD circulators and pressurizer), the main heat exchanger box, and the H₂ transfer lines located outside the reactor building boundary (NNS, QL-2, seismic-no). As in vacuum system (a) above, components include suction and exhaust lines, check valves, RGA and vacuum level monitoring, vacuum pumps, and vacuum pump helium blanket. The purpose of this vacuum system is to provide a vacuum thermal insulation boundary to structures housing cold H₂ outside the reactor building. Vacuum level and gas monitoring of the evacuated space will detect leaks, and vacuum pump exhaust into the H₂ ventilation system will maintain isolation of leaked H₂ from safe room air. The justification for the quality level is the same as that discussed under Sect. 3.6.4.5.

HB-4 beam tube extension vacuum system (c) (including suction and exhaust lines, check valves, roughing vacuum pump, vacuum level monitoring, and vacuum pump helium blanket) (NNS, QL-2, seismic-no). Its purpose is to maintain a monitored, long cold neutron path-length region between the vacuum tube neutron window and the beam tube neutron window. It has been proposed that the region be partially evacuated and filled with subatmospheric helium to provide this long path length. The proposed subatmospheric helium region would also maintain isolation of cold H₂ inside the moderator vessel from the air in the beam room, but does so only as a further backup of the helium cover gas region between the two walls of the vacuum tube neutron window. Thus, the helium gas in the beam

tube extension region would serve no necessary nuclear safety function. Other than degrading the cold neutron beam passing through, failure of this vacuum system has no impact on cold source operation and reactor operation. A quality level of QL-2 is assigned to emphasize that the helium must remain uncontaminated such that the cold neutron beam is not degraded. Vacuum level monitoring will indicate leaks in the beam tube extension wall and/or beam tube/vacuum tube neutron windows. Vacuum pump exhaust into the H₂ qualified ventilation system will maintain isolation of exhausted gases and beam room air.

3.6.4.8 H₂ Ventilation System

In-building transfer line protective duct, ex-building H₂ release stack, and any required dampers and fans (SC-3, QL-2, seismic-yes). Its purpose is to maintain isolation and control of H₂ leaks from cold source components located inside and outside the reactor building by dilution with inert gas (N₂) and venting through a H₂ qualified stack outside both the reactor building and the safe room (and refrigerator room). Therefore, the nuclear safety function served by this equipment is the isolation of hydrogen from the reactor building atmosphere. Only the portion of this system associated with venting hydrogen leaks from inside the reactor building through the exhaust stack is to be classified as SC-3. The rest of the hydrogen ventilation system is to be NNS.

This venting system also provides venting of the He cover gas surrounding the vacuum region of vacuum systems (a) and (b) above in Sect. 3.6.4.7 or the inert blanket regions (either He or vacuum) surrounding ambient temperature H₂ bearing components described in Sect. 3.6.4.5. Finally, it provides the venting pathway for gas mixtures exhausted by fans from the safe room.

The in-building H₂ transfer line protective duct serves to provide mechanical protection to the H₂ transfer line housed inside. It also provides further isolation of any externally leaked hydrogen from the atmosphere of the pony motor battery room and the electrical equipment room. Survival after the DBE is required to ensure continued protection of the transfer line inside the reactor building.

The portion of the H₂ ventilation system located outside the reactor building that is nuclear safety related must not be vulnerable to wind loadings nor to failures of equipment or structures associated with the safe/refrigerator rooms. Survival after the DBE is required to ensure continued isolation of cold source hydrogen from the reactor building atmosphere.

Inert gas purging of the H₂ ventilation system (SC-3, QL-2, seismic-yes). This system provides inert dilution gas for leaked H₂ routed to the hydrogen release stack. The inerting function provided by the gas purge prevents H₂/air reactions inside the ventilation system and therefore serves to maintain the integrity of the ventilation duct. Thus, the nuclear safety function served by the inert gas purging system is the prevention of hydrogen/air reactions inside the nuclear-safety-related hydrogen release stack. It must survive the DBE to continue to provide leaked gas inerting capability.

Safe room ventilation components including ductwork, fans, and dampers (NNS, QL-2, seismic-no). Its safety function is to provide venting of leaked H₂ into the safe room atmosphere to the H₂ ventilation system outside. Ventilation machinery must be "intrinsically safe" (see Sect. 3.6.4.9 below). The justification for the quality level is the same as that discussed under Sect. 3.6.4.5 above.

Refrigerator ventilation classification (NNS, QL-2, seismic-no). A special refrigerator ventilation system separate from the hydrogen vent may be required by Requirement 15 in Sect. 1.2.3. This arises from the potential presence of hydrogen in the helium due to a leaking main heat exchanger in the safe room. Detection of H₂ in the helium would occur wherever the He inventory control equipment for the refrigerator is located. The justification for the quality level is the same as that discussed under Sect. 3.6.4.5 above.

3.6.4.9 Remaining Cold Source Equipment Outside The Reactor Building

Portions or all of the following equipment is located outside the reactor building but is classified elsewhere.

Safe room equipment comprising the pump module (e.g., the HD and LD circulators and the pressurizer), main heat exchanger, LH₂ and He transfer lines. These items are discussed in Sect. 3.6.4.5.

Components housing ambient temperature gaseous H₂ including the storage vessel, the charging tank, the transfer pump, the pressure control lines and valves. These items are discussed in Sect. 3.6.4.5.

Refrigeration equipment. This equipment is discussed in Sect. 3.6.4.6.

Ventilation equipment. This equipment is discussed in Sect. 3.6.4.8.

The computer controller. This item is discussed in Sect. 3.6.4.10.

Equipment and structures such as heating, air-conditioning, roof, floor, walls, and equipment support structures in the refrigerator room and in the safe room (NNS, QL-2, seismic-no). The safe room and refrigerator room are located outside the reactor building and serve no nuclear safety function. Special requirements exist, however, in that equipment housing H₂ must be intrinsically safe, {i.e., designed such that static electrical charge and mechanical energy ignition sources are minimized according to industrial standards applicable to hydrogen systems (i.e., < 0.019 mJ)}. In addition, the walls of such safe rooms must incorporate blowout panels to relieve pressure build-up inside the room to protect both equipment and personnel in the event of an H₂/air deflagration. The justification for the quality level is the same as that given in Sect. 3.6.4.5 above.

3.6.4.10 Computer Controller And Loop Instrumentation

Cold source instrumentation associated with the HFIR safety system (SC-3, QL-1, seismic-no). The nuclear safety function is to directly ensure the nuclear safety functions of the beam tube, the vacuum tube, and the moderator vessel by monitoring the cold source loop forced-flow cooling status of the moderator vessel. If the cooling function is threatened, then the reactor is shut down, and a passive means of cooling the moderator vessel (helium flooding) is provided. There is no seismic requirement as the instrumentation is required (upon malfunction) to promote reactor scram. It is required that the instrumentation and the resulting electronic signals from such scram-related instrumentation be isolated (physically and electronically) from cold source equipment, components, and systems located outside the reactor building.

The cold source loop controller (NNS, QL-2, seismic-no). The safety function of the pressure, temperature, and inventory control system is to maintain the moderator vessel in a controlled, stable, single-phase (or uni-phase) mode of forced-flow heat transfer. The existence of adequate forced-flow cooling of the moderator vessel during reactor operation is a nuclear safety function. However, the control of such cooling is NOT a nuclear safety function, as proposed reactor scrams and alternative passive cooling mechanisms render such cooling unnecessary. The justification for the quality level is the same as that discussed in Sect. 3.6.4.5 above.

Nonscram related loop instrumentation (including remaining temperature, pressure, flow, RPM, and motor current) (NNS, QL-2, seismic-no). Process variable signals from the loop instrumentation may be required by the loop controller to maintain the desired loop operating conditions. They also may identify abnormal operating conditions indicative of malfunctioning equipment and may be used by the loop controller to cause the loop to transition from the normal mode to the standby mode or even to the low-pressure response mode (Sect. 3.5) of operation. However, they are not used by the reactor safety system to detect the existence of adequate moderator vessel cooling and, thus, are not nuclear safety related. The justification for the quality level is the same as that discussed in Sect. 3.6.4.5

above. (Note that the RGA and vacuum level monitoring functions are included in Sect. 3.6.4.7 and may be nuclear safety related.)

3.6.4.11 Electrical Power Supply

Power to cold source equipment located in the safe room (HD/LD circulators), or the refrigerator room (compressors) (NNS, QL-2, seismic-no). Cold source equipment located in these rooms does not serve any nuclear safety function. Thus the electrical power supply to this equipment either normal or emergency backup (battery) is not nuclear safety related. Emergency power to this equipment serves only to ensure reliability and performance of the moderator vessel cooling function and will reduce the number of reactor scrams initiated by the cold source.

Power to scram-related instrumentation (NNS, QL-1, seismic-no). Electrical power supply to this instrumentation serves no nuclear safety function as this instrumentation is required to fail (physically or by loss of power) in such a way as to promote reactor shutdown automatically (see Requirement 17 in Sect. 1.2.3). Emergency battery power to this instrumentation enhances reliability and reduces the potential for scrams. The justification for the quality level is the same as that discussed in Sect. 3.6.4.5.

Power to non-scram-related instrumentation and computer controller (NNS, QL-2, seismic-no). Electrical power supply to these components serves no nuclear safety function as they are NNS. Emergency battery power enhances reliability and performance. The justification for the quality level is the same as that discussed in Sect. 3.6.4.5.

Power to vacuum systems (including RGA and vacuum level monitoring) located inside the reactor building for the vacuum tube and H₂ transfer line (SC-3, QL-2, seismic-yes). Electrical power (normal or emergency battery) to these systems ensures the nuclear safety function of these systems. Its availability is required after a DBE to continue the nuclear safety function.

Power to vacuum systems (including vacuum level monitoring) located inside the reactor building for the beam tube extension (NNS, QL-2, seismic-no). Electrical power (normal or emergency battery) to this system is not nuclear safety related because the subatmospheric helium filling the extension region of the vacuum tube serves no nuclear safety function. Failure of the electrical power supply to this vacuum system results only in the eventual degradation of the cold neutron beam passing through. No systems important to reactor safety are affected.

Power to vacuum systems (including RGA and vacuum level monitoring) located outside the reactor building (e.g., the pump module) (NNS, QL-2, seismic-no). The electrical power (normal or emergency battery) to these systems ensures the safety function of these systems. The justification for the quality level is the same as that discussed in Sect. 3.6.4.5.

3.7 REFERENCES FOR CHAPTER THREE

1. D. L. Selby et al., *High Flux Isotope Reactor Cold Source Preconceptual Design Study Report*, ORNL/TM-13136, Martin Marietta Energy Systems, Inc., Oak Ridge Natl. Lab., December 1995.
2. MELCOR 1.8.1 Computer Code Manual, *Volume 1: Primer and Package Users' Guides*, and *Volume 2: Reference Manual and Programmers' Guides*, June 1991.
3. C. T. Ramsey, *Probabilistic Event Assessment of the Liquid Hydrogen Cold Source Proposed for the High Flux Isotope Reactor*, ORNL/RRD/INT-110/R0, Lockheed Martin Energy Research Corp., Oak Ridge Natl. Lab., April 1997.
4. *Final Safety Analysis Report on the Cold Neutron Facility for the Brookhaven HFBR*, Andrew Kevey, ed., revised August 1980.

5. B. A. Hands, ed., *Cryogenic Engineering*, Academic Press, 1986.
6. Walter Frost, ed., *Heat Transfer At Low Temperatures*, Plenum Press, 1975.
7. D. H. Cook et. al., *The High Flux Isotope Reactor Safety Analysis Report* (draft), ORNL/INT-99, Lockheed Martin Energy Systems, Inc., Oak Ridge Natl. Lab., to be published.
8. R. M. Harrington and C. T. Ramsey, *Preliminary Probabilistic Design Basis Accident Evaluation of the Cold Source Facilities of the Advanced Neutron Source*, ORNL/M-4606, Martin Marietta Energy Systems, Inc., Oak Ridge Natl. Lab., August 1995.
9. American Nuclear Society American National Standard, ANSI/ANS-51.1-1983, *Nuclear Safety Criteria for the Design of Stationary Pressurized Water Reactor Plants*.
10. DOE Order 6430.1A, *General Design Criteria*, U.S. Department of Energy, June 1, 1989.
11. *Guidance for Preparation of Safety Analysis Reports*, DOE/OR-901, Rev. 1, U.S. Department of Energy, December 1989.
12. *Database for Probabilistic Risk Assessment of Light Water Nuclear Power Plants: Vol. 5, Human Error Rates*, PLG-0500, PLG, Inc., Newport Beach, California, November 1991.
13. ATHENA [see RELAP5/Mod3 Code Manual, Volumes 1-7 Inclusive, The RELAP5 Code Development Team, NUREG/CR-5535, INEL-95/0174 (formerly EGG-2596), June 1995.]
14. RELAP5/Mod3 Code Manual, Volumes 1-7 Inclusive, The RELAP5 Code Development Team, NUREG/CR-5535, INEL-95/0174 (formerly EGG-2596), June 1995.
15. Randall F. Barron, *Cryogenic Systems*, 2nd ed., ISBN 0-19-503567-4, 1985.
16. *CFX 4.1 Flow Solver User Guide*, Computational Fluid Dynamics Services, Harwell Laboratory, Oxfordshire, United Kingdom, October 1995.
17. Personal communication from J. A. Bucholz, Oak Ridge Natl. Lab., to D. L. Selby, Oak Ridge Natl. Lab., Aug. 16, 1996.
18. C. R. Hyman, *HFIR HB-4 Cold Source Moderator Vessel Post Scram Heat-Up Analysis*, C-HFIR-97-026, March 7, 1997.
19. *Brookhaven National Laboratory Reactor Division Quality Assurance Manual*, Sect. 2.1, "Quality Levels," Rev. 5, Brookhaven National Laboratory, Upton, N.Y.
20. American National Standard Nuclear Safety Criteria for the Design of Stationary Boiling Water Reactor Plants, ANSI/ANS-52.1, 1983.

4. PROJECT PLANS

4.1 HFIR COLD SOURCE CONFIGURATION MANAGEMENT PLAN

The HFIR cold source will consist of a forced circulation of supercritical hydrogen through an aluminum cryostat inserted inside beam tube HB-4. The cold hydrogen in the cryostat will slow neutrons (increasing their wavelengths) for transport to experiment positions. Heat added to the system by neutron and gamma interactions in the cryostat structures will be removed by the hydrogen. A heat exchanger will be used to transfer the heat from the supercritical hydrogen loop to a gaseous helium loop that is, in turn, cooled by a helium refrigeration system. The helium system will reject heat to the atmosphere. The main hydrogen components (hydrogen pumps, heat exchanger, and cryogenic valves) will be contained outside HFIR confinement, and the helium refrigeration system will have its own building or annex, also outside confinement, for the compressors and refrigerator cold box.

4.1.1 Project Organization

This project will be administered by ORNL. RRD is the sole design authority for HFIR. As such, RRD has design authority over all components that can physically affect the safe and reliable operation of the reactor. This means that an RRD-designated staff member will provide interface with all cold source design and project activities, with special review and approval processes required for those specific components or features deemed capable of affecting the safe and reliable operation of the reactor. Detailed design will be performed by a team of RRD staff, other ORNL laboratory staff, and subcontractors. Detailed safety analysis will be coordinated and primarily performed by the HFIR safety analysis group. Installation of the cold source and the development of the cold source operations group will be coordinated by RRD staff with technical support from the cold source design team. The project's QA function will be provided by HFIR's QA department.

4.1.2 Configuration Management Objectives

Configuration management is a management process that identifies the baseline functional and physical characteristics of an operating system throughout its life cycle. Configuration management controls baseline changes, reports change implementation status, and maintains records. Configuration management maintains the integrity and continuity of the systems' technical, schedule, and cost baselines as trade-off decisions are made with respect to proposed changes.

4.1.3 Configuration Management Plan Scope

The scope of the configuration management plan (CMP) provided in this report is for the duration of the project starting with the detailed design phase and continuing on through the construction phases up to the operations phase.

4.1.4 Configuration Control Implementation

During the design phase covered by this report, the principal method of recording and communicating changes has been through bi-weekly cold source meetings. Task leaders or designated representatives attended these meetings and discussed design change considerations. Minutes of these meetings documenting discussions and decisions reached were compiled by the project manager and a record copy retained as an electronic file on the FEDC server. Task leaders were responsible for ensuring that the minutes covering their areas of responsibility were correct and included any significant changes.

The release of this report marks the beginning of the detailed design phase. With the start of this phase, detailed design, construction, and installation configuration management shall be in accordance with the requirements of RRAP-3.2, *Configuration Control of Plant Design Modifications*.¹ The Configuration Control Board (CCB) required under RRAP-3.2 shall consist of the project manager, the lead cryogenic system design engineer, the HFIR interface engineer, the project safety manager, the project operations manager, and the project QA specialist as a minimum. Additional members (e.g., subject matter experts) will be called upon as deemed necessary by any standing member of the CCB. All configuration changes shall be approved by the CCB. Approval of all standing members of the committee or their designated representatives is required to make a change in the reference design. The reference design and associated plans as presented in this report shall serve as the baseline.

4.2 PROCUREMENT PLAN

Procurement of nonsafety class items and fabrication services from sources outside of ORNL will be processed in accordance with ORNL site procedures. Nuclear safety class items procured from outside of ORNL will be processed using RRD procedure, RRAP 1.9, *Procurement of Items and Services*.² A graded approach will be used to evaluate all purchased items. All items determined to be nuclear safety-related will either be procured through NQA-1 qualified vendors (which does not mean they must possess a nuclear stamp) or the dedication process. Dedication is a process whereby design engineering determines the critical characteristics for each item, and methods are established to verify that the procured items meet those characteristics. These dedication methods include activities such as reviews of supplier performance history, commercial-grade supplier surveys, source inspections, and special tests and inspections.

Nonnuclear safety-related items will be evaluated against the requirements of ORNL Standard Practice Procedure, X-GP-16, *Procurement of Critical Applications/Safety Class Items and Services at ORNL*.³ Critical applications and safety-class items include commercial safety-class hardware; personnel safety items; and items with critical environmental, safety, and health applications. The procurement specification for these items will require additional reviews by appropriate groups such as health physics, industrial safety, RRD engineering, and QA, as deemed necessary by the project team. This additional review assures that proper inspection and testing will be specified in the procurement and receiving process. Dedication methods as described above will be used on critical items where appropriate.

A buyer will be assigned to the project to handle all procurements regardless of the point of origin (see Table 4.1 for procurement strategy). The assignment of a single buyer for the project will facilitate uniform handling of each procurement and will simplify communication. Procurement records, including purchase specifications, dedication packages, vendor surveys, and test and inspection reports, will be stored by the applicable procurement organization as quality records. The records of procurement processed using RRD procedure, RRAP-1.9, will be stored at RRD as quality records.

Fabrications inside and outside ORNL will be processed through RRD procedure, RRAP 3.3.1, *Fabrication Control Governing Modifications to Reactor Systems and/or Replacement Hardware*.⁴ This procedure provides for collection, review, and audit of all required fabrication records such as dimensional reports, weld reports, material test reports, cleaning reports, deviation requests, and nonconformances before the item can be put into reactor service. This collection of information is kept in the RRD document control center as quality records and serves as the as-built documentation for fabricated parts. This fabrication file will also serve as the dedication package for the fabrication.

Each piece of equipment/component has been evaluated and assigned a quality level and a safety class (see Table 3.11 in Sect. 3.6 of this report). Note that the safety classification (SC) does not automatically correspond to a particular quality level (QL) requirement. The QL was assigned with respect to the QA requirements that need to be implemented to ensure the appropriate quality of the

Table 4.1. Equipment procurement strategy

Module	Procurement strategy
Moderator plug (including beam tube)	Design – internal Fabrication – outside
Pumps	Design build – outside
Heat exchanger	Design build – outside
Gas-handling system	Design build – internal
Refrigeration unit	Design build – outside
Transfer lines	Design build – internal
Vacuum system	Design build – internal
Inert blanket system	Define requirements for other systems
Control system	Design – internal Fabrication – internal
Buildings	Design build – outside
Vent system	Design build – internal

equipment/component. The SC was assigned based on the amount of safety risk related to the piece of equipment/component.

QL-1 is the highest level and will have the most rigorous QA requirements. These will include implementation of specific QA requirements as stated in the purchase requisition, pre-award QA program review, post-award audit of the QA program, source surveillances, and a job specific fabrication/inspection plan, as a minimum. QL-4 is the least restrictive in regard to implementation of quality assurance requirements on the vendor/supplier. A QL-4 program would only implement the requirement for deviation requests and nonconformances. QL-2 and QL-3 requirements are between QL-1 and QL-4. QL-2 will not be as restrictive as QL-1 but will be more restrictive than QL-3. QL-3 will be more restrictive than QL-4 but less restrictive than QL-2. The minimal requirements for QL-2 and QL-3 will be dependent upon the equipment/component's usage, fabrication process, and safety class to a certain degree. Each will be assessed individually to determine the appropriate QA requirements that must be placed in the procurement specification.

4.3 CONSTRUCTION AND INSTALLATION PLAN

Construction of the cold source support buildings and installation of cold source equipment is anticipated to occur as per the schedule contained in Sect. 1.3 of this report. The major construction activities are to construct support buildings to contain the refrigeration equipment. The major installation activities include the installation of the refrigeration equipment, heat exchanger, hydrogen systems, hydrogen transfer lines, and moderator vessel. Interfaces between the new equipment and the existing systems, such as electrical power to new buildings, are also covered by this plan.

4.3.1 Building Construction

Under the present plan, three or possibly four structures are required: a hydrogen building and a refrigerator building located between the present HFIR electrical building and the HFIR truck bay and an operations (Ops) building east of the truck bay that would contain the cold source control room and cold source staff offices. (It should be noted that for economic reasons the Ops building may be split into a cold source control room building and an office structure.) Figure 4.1 provides a view of the proposed cold source buildings. All construction activities will be accomplished in accordance with governing RRD procedures and requirements. The layout of the cold source structures shall be approved by the HFIR operations manager to ensure that structures do not impede access to existing HFIR facilities. All buildings will be built to the Uniform Building Code and will meet ORNL fire protection requirements. Standard seismic requirements will apply to all structures unless higher requirements are deemed necessary by the cold source safety group. Services to be supplied for the buildings include normal electrical power from both the 294 and 234 switchgear, steam for building heating and water-line tracing, potable water for sanitary usage, and possibly instrument air to operate equipment (gas-operated valves may be operated by pressurized gas or instrument air). Additional information on each structure is provided below.

4.3.1.1 Hydrogen Building

The hydrogen building will be adjacent to the refrigerator building and will normally be accessed from the refrigerator building through an interior entrance (see Fig. 4.1). Both the hydrogen building and the refrigerator building will be at the same grade as the electrical building. The hydrogen building will house the hydrogen-helium heat exchanger, hydrogen valves, and hydrogen circulators. The building will be steam-heated and ventilated by a dedicated hydrogen-safe fan. The atmosphere in the hydrogen building may be replenished once a minute, as needed, by this fan, which will discharge into a dedicated

Fig. 4.1. Cold source buildings.

hydrogen stack running up the east wall of the HFIR. Blowout panels sized to comply with hydrogen safety standards will be designed into the roof section. The building will have a normal entrance through a personnel door, which communicates with the refrigerator building, and a truck door entrance to facilitate equipment maintenance. The entrance through the refrigerator building will require passage through a dress-out area to ensure personnel entering the hydrogen building are dressed appropriately in hydrogen-safe clothing. The building itself will be approximately 16 × 20 ft and will be composed of preengineered, prefabricated structural modules that will require assembly on site.

Construction work will be allocated such that a contractor or Lockheed Martin (LM) provides the site preparation, the pouring of the pad, and the erection of the prefabricated building parts. All construction activities will be scheduled through the HFIR plan-of-the-day process.

4.3.1.2 Refrigerator Building

The refrigerator building will be immediately adjacent to the hydrogen building and will contain the helium expanders, the helium heat exchangers, and the helium compressors (see Fig. 4.1). Most of the building will be heated and air conditioned. The building will have three removable hatches on the roof to expedite component placement and maintenance and will also have a maintenance area to support maintenance of both helium and hydrogen components. Normal entrance and egress into the hydrogen building will be via a change area to allow personnel to change into clothing appropriate for a hydrogen environment. The refrigerator building itself will be approximately 20 × 60 ft and will be composed of preengineered, prefabricated structural modules that will require assembly on site.

Construction work will be allocated such that a contractor or LM provides the site preparation, the pouring of the pad, and the assembly of the prefabricated building parts. All construction activities will be scheduled through the HFIR plan-of-the-day process.

4.3.1.3 Operations Building

The Ops building will contain the cold source control room, operational support equipment, limited spare parts storage, and offices and facilities for the cold source staff. The building will be a preengineered, prefabricated structure anchored to a concrete slab floor. Construction work will be allocated such that a contractor or LM provides the site preparation and the pouring of the pad. Anchoring of the building will be performed by either LM or the contractor supplying the prefabricated structure. All construction activities will be scheduled through the HFIR plan-of-the-day process.

4.3.1.4 Ancillary Equipment

Ancillary tanks will include a liquid nitrogen tank, a hydrogen gas tank, and a helium gas tank. All three tanks will be located adjacent to the refrigerator building. The hydrogen tank will be located so as to minimize possible damage caused by a hydrogen deflagration, but to also minimize hydrogen inventory. The nitrogen tank will be located so as to provide easy access by truck for refilling. The hydrogen and helium tanks will be filled or replenished by gas bottles. All tanks are expected to be prefabricated tanks that will be delivered and installed on a concrete pad. Supply hydrogen will be stored in bottles placed in lockable racks attached to the hydrogen building. Supply helium will be stored in bottle racks in or near the refrigerator building. Strict QA and accountability requirements will be used to ensure that hydrogen introduced into the cold source system meets predefined purity specifications.

Installation work associated with these tanks will be allocated such that a contractor or LM provides pouring of the pads for the tanks, installation of a prefabricated tank, and installation of supporting service lines and conduit.

4.3.2 Equipment Installation

Equipment to be installed includes the moderator plug assembly, the refrigerator equipment, the hydrogen transfer lines, the hydrogen-safe room equipment (including the pump module, heat exchanger, and vacuum systems), the instrumentation and control system, and various gas tanks and gas-handling systems. Installation activities will be accomplished in accordance with all governing RRD procedures and requirements, and all installation activities will be scheduled through the HFIR plan-of-the-day process. Priority will be given to integrating cold source installation activities so as to cause minimal disruption to HFIR operations. More detailed information for each component installation is provided below.

4.3.2.1 Moderator Plug Assembly Installation

The moderator plug assembly will be installed by LM personnel. Installation will coincide with a beryllium reflector changeout of the HFIR, which is tentatively planned for FY 2000. Testing of the moderator plug assembly will be performed on site by LM personnel prior to its installation into the HB-4 beam tube.

4.3.2.2 Refrigerator Installation

The installation of the refrigerator is scheduled to start in the last quarter of FY 1998; therefore, all supporting infrastructure for the helium refrigeration systems must be in place to meet this date. Current plans call for the refrigerator to be installed by the refrigerator subcontractor. Oversight of this installation will be supplied by RRD personnel. All members of the refrigerator vendor installation team will receive the appropriate training needed to do work on the HFIR site.

4.3.2.3 Hydrogen Transfer Line Installation

The hydrogen transfer line from the hydrogen building will penetrate the HFIR Building on the east side, traverse through the electrical equipment room and the battery room, run above the entrance to the subpile room, and interface with hydrogen lines coming out of the HB-4 beam tube. LM personnel will make all connections, install conduit, and provide needed services inside of confinement. Hydrogen lines connecting the hydrogen building to the moderator vessel will be installed by LM personnel. Leak testing of the lines will also be performed by LM personnel.

4.3.2.4 Hydrogen Building/Safe Room Equipment

The hydrogen safe room consists of all the equipment installed in the hydrogen building. This equipment includes the pump module, the heat exchanger, and their associated plumbing and valves.

The pump module assembly contains all pumps and their associated valves and plumbing. This equipment will be preassembled into a single component that will be installed by the pump module vendor and/or LM personnel. The pump module will be connected by transfer line to the heat exchanger and the moderator plug assembly. Testing of the pump module following installation will be performed by LM personnel prior to installation of the cold source moderator plug assembly into the HB-4 beam tube.

The heat exchanger and associated plumbing will also be installed by LM personnel. It will be connected by transfer line to the hydrogen pump system and the moderator plug assembly.

4.3.2.5 Instrumentation and Control System Installation

With the exception of refrigerator equipment instrumentation, all instrumentation will be supplied, tested, and installed by LM personnel. The instrumentation associated with the refrigerator will be installed by the refrigerator vendor.

4.4 OPERATIONS PLAN

The cold source is designed to operate 24 h/day during reactor cycles. Cold neutrons will allow more effective use to be made of HB-4 in support of ORNL's core mission of neutron science.

The cold source consists of a helium loop operating at 17 K, a hydrogen loop operating at 20 K, and supporting ancillary equipment. The helium loop is the heat sink for the hydrogen moderator and is used to remove energy from the hydrogen. The helium is compressed and cooled by a refrigerator, which is a series of compressors and piston expanders. The helium does work on the piston expanders, thereby cooling the helium. A heat exchanger transfers heat from the hydrogen to the helium. Temperature controls add supplemental electric heat through a heater as necessary to maintain the helium loop in a steady-state mode. Extra compressor capability will be automatically brought on line as the system requires it. Two liquid nitrogen heat exchangers provide precooling to the helium and will also maintain the helium loop at 100 K during standby mode. Approximately 550 gal of liquid nitrogen per day is required for precooling. Hydrogen is circulated at 1 L/s by one of two HD circulators for operation between 20–50 K and at 2.25 L/s by a LD circulator for operation between 50 K and ambient temperatures. The HD circulators can be utilized above 50 K if conditions require it but will be inefficient. Hydrogen circulation capability will be supplied from a backup power supply during loss of normal power events. The backup power source and components that need access to it are still undergoing evaluation. Hydrogen transfer lines connect the heat exchanger in the hydrogen building with the moderator inside HB-4. The moderator slows down neutrons from the HFIR reflector and provides a path for the resulting "cold" neutrons to pass through HB-4. All hydrogen lines will be vacuum-insulated to minimize heat transfer and to allow definitive indication of a hydrogen leak. The moderator vessel will be installed inside a vacuum tube installed in the HB-4 beam tube. Demineralized water, which is injected with helium, will slowly circulate around the exterior of the vacuum tube to remove heat from the tube. The helium will function as a leak indicator in case of leakage between the vacuum tube and the moderator vessel (water leaking into the moderator vessel would freeze out at the low temperatures; helium will not). The vacuum surrounding the hydrogen transfer lines will be monitored for hydrogen, and the moderator vacuum region will be monitored for hydrogen and helium.

Ancillary cooling systems are required for the helium compressors and the LD circulator. The compressor cooling system will be a self-contained chiller that dissipates heat to the atmosphere. The LD circulator will use process/potable water that discharges to HFIR's sanitary system or to the process waste system. All buildings with water will be heated.

The cold source will be operated primarily from its control room. Major cold source equipment will be controlled from this location, and process parameters will be displayed. All of the controls and indications will be digital; much of the information and many of the controls will also be available via drop connections in the HFIR control room and at HB-4. Command and control may be transferred to other drop connection locations by following a strict transfer protocol.

The authorization bases for the cold source will be the HFIR SAR and the HFIR technical safety requirements (TSRs). The HFIR SAR will refer to a safety assessment that will provide the safety envelope for the cold source. The authorization bases documents will be kept current by RRD's technical section with funding from the cold source operations budget.

4.4.1 Operations Organization

The cold source operations organization will consist of up to three persons. The cold source manager will report to the HFIR section head and will be responsible for all aspects of cold source operations. Two cold source technicians/operators will report to the cold source manager and will perform daily checks, conduct operational adjustments, and perform maintenance on cold source equipment. During the off-shifts, one of the cold source crew will be on call, will respond to telephone requests for assistance, and will report to work to address cold source problems within a yet-to-be-determined time period.

Selected HFIR reactor operations staff personnel will be provided training that will allow them to recognize and respond to a prescribed set of cold source adjustments and transients. Problems beyond the scope of their training may result in shutdown of the cold source or reduction in power of the HFIR, pending arrival of a cold source crew member.

Cold source operations will be under the supervision of the HFIR shift supervisor, as are all HFIR experiments. In case of an emergency, the RRD emergency response organization will be mobilized to respond to it.

4.4.2 Operations Staffing

Staffing is further subdivided into the functional areas of operations support, maintenance support, training, and technical support.

4.4.2.1 Operations Support:

As previously indicated, the cold source will be operated by a dedicated support crew of up to three persons. Cold source support personnel will primarily work day shifts, but will be available on a call-in basis in case of problems during off shifts. HFIR reactor operations personnel will be trained to recognize and respond to selected problems with the cold source and place it in a stable condition pending arrival of a cold source support person.

4.4.2.2 Maintenance Support:

Cold source operations personnel will maintain equipment and components that can be worked on while the cold source is in operation. During cold source outages (which are expected to coincide with HFIR outages), maintenance will be performed on the cold source systems that cannot be worked on during operational periods. Cold source operations personnel will perform most of the maintenance required by the cold source. Maintenance support from other non-RRD organizations and vendors will be requested as required.

4.4.2.3 Training

The cold source will have a dedicated trainer who will prepare lesson plans and conduct training for cold source operations personnel, HFIR personnel, and technical support personnel. The cold source operators will be required to be trained to criteria defined by DOE Order 5480.20A.⁵ A matrix defining the training plan's degree of conformance with the Order will be submitted to DOE in time to support the schedule for operation of the cold source.

4.4.2.4 Technical Support

Technical support personnel from HFIR's technical section will assist in keeping the SAR updated, reviewing procedures, preparing unresolved safety question determinations (USQDs), preparing TSR revisions, and addressing technical problems.

4.4.3 Operational Budget

The anticipated operational budget for the cold source is approximately \$926K/year. A breakout of costs is included below in Table 4.2.

Table 4.2. Breakdown of normal cold source operation costs

Item	Breakdown (FTEs) ^a	Nominal costs FY 1997 K/\$
Operations support	CS ^b man, 2 CS technicians, 0.2 technical support, 0.2 QA ^c	471
Maintenance support	200 h craft support + HPs ^d	13
Training	0.25 training	31
Utilities	HFIR operating efficiency = 70%; average 11 cycles/year; average outage = 10 days in length. Compressors = 3 100% w/ 4th on 20% when HFIR is running; when HFIR is shut down, 2 compressors for half of the time and 1 compressor for the remaining time	198
Administrative purchases	Copier, computers, office equipment, etc.	10
Spare parts and materials	Costs include replacement parts and chemicals such as nitrogen, helium, etc.	188
Waste management	Waste disposal	15
Total		926

^aFTE = full-time equivalent.

^bCS = cold source.

^cQA = quality assurance.

^dHP = health physicist.

4.4.4 Operations and Maintenance Policy

The HFIR cold source will be operated and maintained by the HFIR section. The crew will have skills unique to equipment operated under vacuum and cryogenic conditions.

The cold source will follow all conduct of operations guidelines currently subscribed to by HFIR. The cold source conduct of operations guidelines will be unique to the cold source and may, therefore, be somewhat different from the HFIR conduct of operations guidelines. The degree of conformance with the DOE conduct of operations order will be the same. Prior to initial operation of the cold source, a set of cold source conduct of operations matrices will be submitted to DOE for approval.

Cold source procedures will be written utilizing the HFIR procedure writers guide and will undergo review and approval through the Plant Operations Review Committee (PORC) process. All approved

procedures will undergo unreviewed safety question determinations by RRD's technical section. The procedural development, approval, and usage process for the cold source will be explicitly described in Chapter XVI of the conduct of operations guidelines applicable to the cold source.

Routine and breakdown maintenance procedures shall be developed, approved, and issued prior to initial reactor operation with the cold source installed inside the beam tube. The operations crew will be trained on the maintenance procedures prior to initial operation with the cold source moderator plug assembly installed inside of HB-4.

4.4.5 Normal Modes of Operation

The cold source has five predefined normal modes of operation: shutdown, cool-down, cold neutron production, warm-up, and standby. These modes of operation are described in the following paragraphs.

4.4.5.1 Shutdown Mode

In this mode, all cold source systems, other than vacuum systems, are completely shut down, and the system pressure is stabilized at 4 bar abs. This mode is entered only when reactor core decay heat has fallen sufficiently for the moderator vessel temperature to stabilize without forced cooling. The refrigerator will be shut down during outages lasting longer than three days. It should be noted that while the hydrogen loop is in the shutdown mode, periodic circulation may be required to remove heat and monitor moderator temperatures.

4.4.5.2 Cool-Down Mode

The cold source will always be started prior to the start-up of the reactor, as heat transfer might not be adequate to cool the vessel material with warm, low-density gas. It should be noted, however, that as part of the detailed design activity, the potential to recool back to the cold neutron production mode from the standby mode under at least partial reactor power will be examined. This capability would allow resumption of cold source operations following a cold source fault condition with minimal impact on reactor operation.

The refrigerator helium system operation is completely automated. Predetermined compressors and expanders will be activated and will start as required to perform normal cool-down functions. To prevent freezing during the early stages of the cool-down, a heater, controlled by the hydrogen exit temperature of the heat exchanger, will ensure hydrogen temperatures do not fall below 14 K. Hydrogen circulation is initially supplied by the LD circulator. At 50 K, the HD circulator is brought on line, and temperature control is shifted to a sensor downstream of the heat exchanger. The temperature control set point is adjusted to 18 K. Hydrogen loop pressure will be maintained supercritical at 14 bar abs (measured upstream of the circulator) to ensure uniphase heat transfer and fluid flow throughout the process. Transition from cool-down mode to cold neutron production mode is completed when system temperature and pressure stabilize at their normal operating conditions. A procedural outline for system cool-down is provided in Table 4.3. A normal cool-down should take approximately 8–10 h.

There is also a special case cool-down in which the hydrogen loop can be cooled from ambient temperature with the refrigerator already running and cold. This would allow the refrigerator to be left running during a short outage of up to two to three days.

Table 4.3. Procedural outline for system cool-down

Procedure step No.	Procedure
1	Start both insulation vacuum stations, and establish a vacuum of about 10^{-6} torr
2	Fill inert blanket sectors with helium gas at about 1.5 bar abs, or draw vacuum to about 10^{-3} torr
3	Fully open all valves in the primary hydrogen circuit (including V-12051). Close V-12058 and V-12405, and evacuate the loop using VP-3 purge pump. Refill with helium gas to atmospheric pressure and re-evacuate. Repeat twice (total of three times); finally pump down to about 10^{-2} torr. Stop vacuum pump, and close V-12606 and all other gas-handling system valves
4	Establish nitrogen flow through vent stack and start-up safe room ventilation fan
5	Using 99.95% pure hydrogen gas bottles, fill the entire hydrogen system through the double valves (V-12059 and V-12060). Use an on-line liquid-nitrogen-cooled cold trap to trap any impurities in the bottled gas. Fill to a pressure of 4 bar abs
6	Close V-12051 and V-12012
7	Isolate the HD circulators and valve in the LD circulator. Set up the refrigerator for operation at a heat load rejection rate of 2.5 kW
8	Set the pressure control to a set point of 14 bar abs (note: this will start the transfer pump, which will raise the loop to 14 bar abs with gas from the main storage tank)
9	Set up the power supplies for the LD and one HD circulator, and confirm switchover point for circulators and temperature control sensor is set at 50 K. Check temperature sensor 1413 set at 14 K, and 1406/7 set at 18 K. Check operation of gas analyzers and adequate nitrogen flow through vent stack
10	Start the refrigerator and the LD circulator with a hydrogen volumetric flow rate of 2.25 L/s
11	Confirm cooling to 50 K. Verify automatic switchover of temperature control sensors. Verify switch from LD circulator to HD circulator
12	Ensure that the main control heater is adding at least 2.5 kW to the helium loop and that temperature of the hydrogen loop is at 18 K. Continue cool-down until steady-state is achieved

4.4.5.3 Cold Neutron Production Mode

In the cold neutron production mode of operation the cold source is operating at temperatures conducive to production of cold neutrons (nominally 14–20 K). Hydrogen loop pressure is maintained in the supercritical region at about 14 bar abs as measured upstream of the circulator. Operating in the supercritical region ensures that potentially detrimental two-phase heat transfer and fluid flow inside the moderator vessel does not develop. Temperature is controlled by the control heater in the helium inlet to the heat exchanger, which responds to a hydrogen temperature sensor downstream of the heat exchanger. This mode of operation is, by design, independent of the reactor power level. If the reactor is not running, the control heater is powered to a level slightly higher than the anticipated reactor heat load. When the reactor starts up, power from the heater is reduced as the nuclear heat load applied to the moderator increases. At full reactor power, the control heater remains energized at a low level and can still provide trimming heat to the refrigerator to compensate for small changes in reactor power. Circulation of the hydrogen is provided by a HD circulator. There are five compressors to provide helium isentropic

compression and circulation; three are required for normal operation and a fourth may cycle on and off if required. The number of compressors on line during steady-state will be controlled by the helium heater. When the heater draws above 900 W of power, a compressor will be shut off automatically by the refrigerator control system.

Normal operational transients, such as reactor downpower, will be automatically handled by the cold source control system during cold neutron production mode. Temperature sensor TE-1400/1 will act to maintain steady-state conditions by supplementing reactor heat with the control heater in the helium flow line. The hydrogen pressure control system will maintain hydrogen loop pressure at the pressure set point.

4.4.5.4 Warm-up Mode

The warm-up mode of operation is entered whenever there is a controlled warm-up of the hydrogen system from the cryogenic conditions of the cold neutron production mode. There are two major operations performed in the warm-up mode: warm-up to enter standby state and warm-up to enter shutdown mode.

4.4.5.4.1 Warm-up to enter standby state

When normal cooling of the cold source moderator vessel is degraded, the cold source may be placed in a standby mode of operation (see description in Sect. 4.4.5.5). The standby mode provides sufficient cooling to the moderator vessel to avoid overheating and subsequently damaging the cold source, even with the reactor operating at full power. Warm-up to this mode is accomplished by switching the refrigerator to passive liquid nitrogen cooling and controlling the hydrogen pressure in the supercritical region (14 bar abs as measured upstream of the circulator). When the hydrogen system temperature reaches 50 K, the LD circulator (operating at half speed) takes over from the HD circulator. At 70–80 K, the LD circulator is automatically increased to normal speed, and the HD circulator is shut off. Transition from the warm-up mode to standby mode is completed when the hydrogen temperature stabilizes at around 100 K at 14 bar abs pressure with a hydrogen flow rate of 2.25 L/s. The reactor may be operated at full power under these conditions.

4.4.5.4.2 Warm-up to shutdown mode

The cold source may be placed in a shutdown mode when the reactor is going to be shut down for more than a few days. Warm-up to the shutdown mode is accomplished after reactor shutdown by switching the refrigerator to passive liquid nitrogen cooling and allowing the system to warm up. Pressure is maintained in the supercritical region at 14 bar abs. When the loop temperature reaches 50 K, the LD circulator (running at half speed) takes over from the HD circulator. At 80 K, the LD circulator is raised to full speed and the HD circulator is shut off. When sufficient time has elapsed for the core decay heat to fall to the point where forced cooling of the moderator vessel is no longer required, the passive refrigerator operation and LD circulation may be stopped. All ancillary equipment (including the pressure control system) may then be shut down, with the exception of the vacuum systems, which continue to operate as long as hydrogen is present in the loop. After shutdown of the pressure control system, the entire hydrogen system together with the storage vessel falls in pressure to 4 bar abs while the temperature gradually increases. Two-phase conditions do not develop during the depressurization due to the high hydrogen temperature. Transition from the warm-up mode to shutdown mode is completed when the gaseous hydrogen system temperature approaches ambient and the pressure stabilizes.

4.4.5.5 Standby Mode

The standby mode of operation allows steady-state operation of the cold source under conditions where the moderator vessel temperature is cooled to 100 K, not to exceed 140 K nominally. The standby mode would be used during a situation where normal cold source cooling is degraded, but reactor operation at full power continues. The standby mode is reached as described in Sect. 4.4.5.4. Thereafter, the hydrogen is circulated at 2.25 L/s by the LD circulator at a supercritical loop pressure of 14 bar abs (as measured at the inlet of the LD circulator). Failure to keep the moderator temperature below 273 K will result in a reactor shutdown.

4.4.6 Annealing of the Cold Source Moderator Vessel

One special condition associated with the expected need to anneal the moderator vessel will be referred to as the anneal mode. Annealing is required to remove undesirable characteristics (silicon defects) from the aluminum moderator vessel caused by radiation embrittlement. It is anticipated that annealing to ambient temperature will be required after every reactor cycle to recover from radiation-induced silicon point defects.

This anneal would be accomplished by entering the standby mode immediately following reactor shutdown for normal refueling. Once the temperature stabilizes at approximately 100 K, the system temperature of the loop would be allowed to slowly rise in a controlled or preprogrammed manner to room temperature (25°C). Following completion of the designated anneal time, the system would be transitioned to the shutdown mode or the cool-down mode, depending on the remaining length of the scheduled reactor shutdown.

4.4.7 Abnormal and Emergency Operating Conditions

As discussed in Sects. 3.3 and 3.5 of this report, off-normal events that the design must consider have been identified based on probabilistic event analyses. These events range from inadequate cooling of the cold source to hydrogen leaks. Although a defense-in-depth approach has been incorporated into the design to minimize the frequency and consequences of these off-normal events, the potential for these events cannot be totally eliminated. Therefore, AOPs and EOPs must be developed and the cold source operations staff trained in their implementation. As discussed in Sect. 3.5 and in the remainder of this section, approaches for dealing with these off-normal events have been identified based on anticipated system response and preliminary analyses. However, insufficient data exist at this time to develop detailed AOPs and EOPs. During the coming year, the data obtained from the AEDC experiments will be extensively analyzed, and detailed analytical models of the cold source system will be developed and used to examine the system response to the various identified transients. The results of this analysis will be used to develop the detailed AOPs and EOPs to be incorporated into the cold source operator training activities. These procedures are expected to be validated with the fully assembled cold source system prior to the installation of the moderator plug assembly into the HB-4 beam tube.

The following paragraphs provide a brief description of the potential responses to certain identified transients. A more detailed discussion of transients is provided in Sect. 3.5. As stated above, it should be clearly understood that the detailed transient analyses have not yet been completed, and thus the responses presented here are simply based on our best knowledge at this time.

4.4.7.1 Loss of Circulator

The cold source loop contains two HD circulators and one LD circulator. During normal operation, one HD circulator is operating and the other HD circulator and the LD circulator are in a standby mode. All circulators will be instrumented so as to detect incipient failures. Detection of problems with the

operating HD circulator will result in the spare HD circulator being placed in service. Loss of both HD circulators will result in transition to the standby mode. Following the trip of the second HD circulator (assuming the other HD circulator is already out of service), the LD circulator is started at one-half speed. As the hydrogen warms up to 100 K, the LD circulator speed will be programmed to ramp upwards. At 80 K (as measured by sensor 1400/1), the hydrogen gas should nominally be circulating at a rate of 2.25 L/s at a temperature of 90 K.

4.4.7.2 Loss of Refrigerator

There are two major initiators for the loss of refrigeration: loss of expanders or loss of compressors. In the event of expander loss, the system will enter a standby mode as described in Sect. 4.4.5.4. Loss of all of the compressors leaves no mechanism to circulate the refrigerant helium and will require rapid response measures. One response being analyzed is to flood the vacuum space with helium gas and immediately scram the reactor to avoid damage to the moderator vessel due to overheating (the cold source compressors draw too much current to be provided back-up power from the HFIR diesels). As a result, the need to maintain electrical power to at least one compressor is important. Electrical power to the compressors is expected to be available from both the 234 and 294 switchgear; therefore, nearly all cold source loss-of-power initiating events will also shut down the reactor. The cold source transformer may be fed from both feeders but is potentially a single point of failure.

4.4.7.3 High Pressure Inside a Hydrogen System

High pressure inside a hydrogen system is a potentially serious situation. High pressure may indicate an exothermic reaction of the hydrogen with another substance, loss of a circulator, or loss of cooling. Pressure relief valves and rupture disks will keep hydrogen pressure within design limits. All rupture disks and relief valves relieve to the dedicated hydrogen vent stack. If a rupture disk actuates, immediate reactor shutdown will be required. Some means of removing heat from the moderator vessel, such as flooding the cavity with helium, must also be employed.

4.4.7.4 Positive Pressure Inside the Moderator Vacuum Vessel (Hydrogen or Helium Leaks)

This area is normally under vacuum. Therefore, positive pressure would be indicative of a helium leak through the vacuum tube, the vacuum tube neutron window, a transfer line internal leak, or possibly a primary hydrogen barrier leak. All helium leaks should be evaluated. If the leak is manageable, continued operation of the cold source is possible, but the implication of operating in this condition needs to be evaluated. If degradation of the hydrogen cooling occurs, the reactor should be shut down immediately.

Evidence of hydrogen migration outside of its primary system will in all cases require an evaluation. If vacuum can be maintained at 10^{-5} – 10^{-6} torr, then no further response may be required. The master control system would be programmed to respond to a predetermined leakage gradient over time to produce an alarm when a response is required. Such a leak would suggest that it would ultimately become impossible to maintain the vacuum, and measures would be taken to provide emergency cooling to the moderator vessel. This casualty would require immediate reactor shutdown and a depressurization of the hydrogen system. Hydrogen in a vacuum line is indicative of a leak that will most likely require component replacement. Such replacement of cold source components will require an extended reactor outage during which other major functions of the HFIR would be unavailable.

Future measures to provide alternative moderator vessel forced-flow cooling (even in the presence of a leak in the primary fluid boundary) using helium instead of hydrogen are being considered as a means to permit reactor start-up and continued operation without the generation of cold neutrons. Such a capability is highly desirable and would provide a method to minimize HFIR's downtime because of

replacement and/or repair of cold source equipment. Such measures will be explored more fully in the detailed design stage of the project.

4.4.7.5 Loss of Electrical Power

Following a loss of off-site electrical power, hydrogen flow will be shifted to the LD circulator, which is supplied with backup power.

The helium gas of the refrigeration system will also need to be circulated using one compressor. The compressors may be fed from the 234 bus or the 294 bus; therefore, loss of all compressors in nearly all cases will have also already shut the reactor down. The exceptions are single-point failures between the feeders and the compressor motor control centers. Backup power for a compressor from the HFIR diesel generator is not an option because of the large power requirements. Loss of all compressors would therefore occur on a loss of off-site power event that will, in turn, require a shutdown of the reactor to avoid overheating of the cold source. A potential source of decay heat removal for the moderator, in this case, is to flood the vacuum tube vacuum space with gaseous helium. Helium will transfer heat away from the moderator to the vacuum tube cooling system.

4.4.7.6 Hydrogen in Refrigerator Helium System

The refrigerant helium system is equipped with three parallel hydrogen sensors to detect migration of hydrogen through the hydrogen/helium heat exchanger into the lower pressure helium system. The hydrogen sensor system will shut down all operating compressors when 50 ppm hydrogen is detected in two out of three detectors. This alarm will also scram the reactor. As discussed in Sect. 4.4.6.5, flooding of the moderator vacuum cavity with helium is a potential source of decay heat removal from the vessel since all compressors would be shut down.

4.4.8 Waste Management

Wastes will be generated both during operations and maintenance periods. During operations, a dedicated hydrogen stack will vent any hydrogen from the hydrogen building up the east wall of the HFIR Building. This stack will also continually discharge gaseous nitrogen as will the exit from the precoolers for the helium compressors. During maintenance of the hydrogen circulators, minor amounts of tritium and activated aluminum corrosion products may be encountered on the hydrogen components. These contaminants must be segregated and contained per health physics procedures and will result in a contaminated waste stream. The LD circulator, when in operation, will discharge either potable or process water as a waste stream. This waste will be routed to either the sanitary system or the process waste system. Several of the buildings will have steam heating or tracing. Waste water from these heaters will be discharged into French drains adjacent to the point of use. The operations building will have a unisex rest room and kitchen facilities, which will discharge sanitary wastes to a holding tank. It will also generate wastes typical of an office/maintenance environment consisting nominally of paper, food preparation wastes, oily rags, etc. Routine production of a Resource Conservation and Recovery Act (RCRA)⁶ material or mixed waste stream from operation of the cold source is not anticipated.

4.5 QUALITY ASSURANCE PLAN

The QA program will be responsive to applicable DOE Orders, ORNL Standard Practice Procedures, RRD Quality Assurance Procedures, and RRD Administrative Policies and Procedures. All quality-related project activities shall be controlled via RRD procedures whenever possible. If an applicable RRD procedure does not exist, an ORNL site procedure shall be utilized to control the activity.

Control mechanisms shall be established to ensure that (1) design inputs are correctly translated into design documents in a timely manner; (2) organizational and physical interfaces are identified and controlled; (3) changes to the design and/or procurement activities are controlled in a manner commensurate with the original requirements; and (4) documentation and records of the design, design verification process, procurement process, and construction activities are maintained in accordance with the QA program established for the project.

QA encompasses all those planned and systematic actions and controls necessary to provide adequate confidence that a structure, system, or component will perform satisfactorily in service. QA includes quality control, which includes actions needed to ensure that the physical characteristics of a material, structure, component, or system meet predetermined requirements.

An adequate QA program provides the following assurances:

- Organization interfaces are identified and controlled.
- The design is adequate and will function as intended.
- A document control system is in place.
- A change control system is in place.

The QA program shall include quality control functions in the following areas:

- The design shall satisfy functional requirements as determined by the customer/user.
- The prepared drawings and construction specifications adequately incorporate QA requirements, design requirements, and applicable codes, standards, and rules. The prepared drawings and construction specification shall be prepared and available in a timely manner.
- Construction can be performed in accordance with the approved design, and any changes made during the construction phase shall be documented and approved.
- Tests, reviews, or inspections confirm the adequacy of the design and quality of the construction and manufactured components, where deemed necessary by the project team and responsible designers.
- Quality status tagging systems are provided for turnover acceptance, maintenance, and system outages as related to the project activities.

As part of the QA program, the design shall be closely coordinated and functionally analyzed to avoid conflicts that could result in costly changes during construction of the design. Prior to initiating detailed design, QA requirements shall be established for the project to be implemented during this phase. Essentially, QA evaluations will be done prior to each major phase of the project (i.e., conceptual, design, and construction phases) to determine what QA requirements shall be implemented. The evaluation will take into consideration the following:

- what the facility is to accomplish;
- the range of operating conditions and the associated risks;
- the required degree of reliability;
- the proposed life span of the facility/equipment; and
- how it can be maintained, repaired, and/or replaced.

Whenever possible, design shall reflect experience gained on similar projects or similar types of construction.

Provisions shall be made for reviewing and checking design calculations, drawings, and construction specifications by qualified personnel other than those responsible for the original design.

Deviations from specified standards and/or approved specifications/requirements shall be identified, and applicable RRD procedures shall be implemented to ensure proper approval and control.

To the extent practical, and particularly in the case of innovative design, the design should be independently reviewed by competent subject matter experts external to the project to confirm the practicality.

Because of the familiarity with RRD and ORNL procedures, the close working relationship of the project team, and continual project update meetings, preparation of a QA plan is not necessary. Quality is an integral part of the project and will be involved in the development of the design, procurement specifications, construction specifications, and all major on-going activities.

4.6 RELIABILITY, AVAILABILITY, AND MAINTAINABILITY PLAN

4.6.1 Goals and Requirements

Two basic goals of the cold source project are to maximize availability of the cold source for neutron scattering experiments and to minimize the impact of any cold source system problems on other reactor operations. This can be accomplished through the development and implementation of a comprehensive reliability, availability, and maintainability (RAM) program, which identifies those functions of the cold source affecting these basic goals, prioritizes the importance of systems and components, and ensures that the design reflects maximum performance. The program outlined below is the basic structure for a RAM plan that should provide for the optimum design of the cold source as well as long-term feedback and improvement of its operation.

4.6.2 Cold Source Systematic Approach to RAM

The optimization of system function and reliability is the result of a systematic and iterative process that examines the demands of system operation and balances those demands against any needs to assure availability of the system. A good functional design does not necessarily lead to a reliable system. However, with a detailed evaluation of the proposed cold source design, early in its design stages, any necessary design changes can be incorporated to optimize both reliability and function with minimal additional costs.

4.6.2.1 System Functions

The first step in the comprehensive RAM plan is to identify the essential goals for the cold source facility. These goals have been identified as:

1. **The overall availability of the cold source facility will be maximized whenever possible.** During the design phase of the cold source, ancillary systems will be optimized in support of minimum cold neutron beam performance requirements and availability goals.
2. **The impact of the cold source facility on the operation of the reactor will be minimized.** The emphasis of the RAM program will be the reduction of inadvertent forced reactor shutdowns over reactor setbacks and power reductions. In addition, emphasis will be placed on ensuring that the cold source is available prior to reactor start-up so as not to impact the predictability of scheduled operation.

Based on these goals, the cold source systems and components are evaluated to identify those essential functions needed to achieve these goals. These functions include all modes of operation including off-normal conditions.

4.6.2.2 Prepare System and Functional Descriptions

The next step is to collect detailed design information on the cold source equipment. The objective for this effort is to obtain a comprehensive understanding of system and component operation, requirements, boundaries and interfaces, support needs, and limitations. Recommended manufacturer maintenance activity will be included as deemed appropriate. This assembly of information is the core input to defining not only "what" preventive maintenance (PM) activity should be done for a given component, but "why" the PM activity is necessary.

4.6.2.3 Identify Equipment Failure Modes and Human Errors

The third step is to perform a systematic evaluation of each component to identify all failure modes that could result in failure of that component to perform essential functions. This can be accomplished using one of several methods. One of the most widely used methods is a FMEA. The product of the FMEA process is the identification of potential failure modes for each component within the cold source system boundaries and a detailed evaluation of each of those failure modes for their impact on the cold source function. The evaluation includes the assessment of human error on equipment. This failure mode information is then used to both screen out equipment failures that do not constitute a significant impact on the goals for the cold source and to identify those components whose high reliability is crucial to overall availability.

4.6.2.4 Evaluate Failure Modes Using Logic Tree Analysis

Failure modes that are not screened out in the FMEA process are then subjected to a further evaluation to determine if the failure can be detected under normal operating conditions, if the failure represents a safety problem (i.e., requires a reactor scram), if it represents an outage problem, or requires transfer to standby operation. The logic tree analysis process provides some measure of importance and priority to the identified failure modes and promotes categorization of events with similar initiating mechanisms and sequence consequences.

If a function or system is shown through FMEA or logic tree analysis to be a potentially high contributor to cold source unreliability, further modeling may be done, such as fault tree techniques, to optimize the design for reliability. Fault tree techniques allow quantifiable consideration of redundant components in assessing the likelihood of a functional loss. The additional information regarding the identification of the failure under normal conditions may be used to justify design modifications to reduce the impact of such failures.

4.6.2.5 Finalize RAM-Based Design

Once the process steps defined in Sects. 4.6.2.2 through 4.6.2.4 have been completed, there should be sufficient information available to finalize a cold source design that provides the optimum balance between operational and RAM requirements. A detailed report will be prepared to document all aspects of the RAM process and plan. This will include the system information, the FMEA, and the logic trees. Any design modifications to enhance reliability and monitoring will be described and justified. This includes changes needed to monitor equipment conditions for those components identified as needing continual monitoring. Preliminary recommendations on preventive and predictive maintenance tasks will be provided with descriptions for their purpose in enhancing reliability.

4.6.3 Maintaining RAM Program Integrity During Final Design and Construction

As the cold source design proceeds to its final configuration and through the procurement process, the components and final design modifications should be reviewed to evaluate if the RAM-based design objectives have been compromised or their effectiveness reduced. This review process should include those changes made during construction as situations in the field dictate changes in the design. Because system reliability can be very sensitive to seemingly minor design changes, RAM considerations should go hand-in-hand with the safety and operability evaluation of design and field change requests. It should be noted that design and procurement of some very long lead procurement items began prior to formulation of this RAM program plan. A prime example of this is the refrigeration system that has already been ordered. It is important to recognize that although a RAM plan was not in place, consideration has been given to reliability, maintainability, availability, and performance through the design and procurement process with the vendor and in each stage of design review. This is reflected in the level of redundancy that has been incorporated into the design. As such, this system should transition smoothly into the RAM program, once it is in place. Items initially purchased for testing purposes only were not scrutinized as much and will have to be evaluated and appropriately qualified if any are to be approved for use in the final system.

4.6.4 Implementing A Long-Term RAM Program

The basis for a long-term RAM program is a monitoring and feedback process that (1) identifies areas of weakness or degradation before it affects the cold source overall availability, (2) acknowledges that the actual performance of the system may not match the projected performance, and (3) is dynamic with the ability to adapt to long-term performance changes and modifications to the system. Monitoring of performance data for decreasing performance trends is critical to the identification of pending failures, especially those that could be hidden from the operations staff during normal operations. Preventive and predictive maintenance activities should be designed around failure modes that are significant to system function. Manufacturer recommendations for preventive maintenance should be critically reviewed to assure they have a measurable positive impact on the performance of the equipment in particular, and the system in general. The objectives of a long-range RAM program are to establish an acceptable level of performance, balanced against available resources; perform those activities necessary to maintain that acceptable performance level; monitor for indications of degrading performance; and quickly identify and correct problems before they significantly affect operation of the cold source or the reactor.

4.7 COLD SOURCE TESTING PLAN

At present, all tests planned for the cold source fall into one of five categories:

- individual component development tests,
- hydrogen loop tests,
- equipment acceptance tests,
- final installation loop tests with the moderator plug assembly mounted outside the reactor building, and
- final installation loop tests with and without the reactor in operation.

It should be noted that some tests have already been performed and the results have been used to support or modify the design. Detailed plans for all remaining tests have not been fully formulated, but will be developed prior to testing based on basic performance validation needs, safety analysis requirements,

review committee recommendations, and QA requirements. A brief general description of the tests in each of the five categories is provided in the remainder of this section.

4.7.1 Component Development Tests

Certain components require special testing of a prototype to demonstrate performance before the design is finalized. The cryogenic valves used to isolate the circulators and the moderator vessel are components that fall into this category.

4.7.1.1 Cryogenic Valve Tests

The cryogenic valves to be used to isolate the high-density circulators are an in-house design that requires development and thorough safety testing to demonstrate performance. Two pairs of these valves are required for either circulator to isolate the operating loop from atmosphere if replacement is required. The planned testing includes tests of the sealing integrity of the seat after at least 100 openings and closures of the valve. A final set of tests will be performed with the cold end of the valve immersed in liquid nitrogen.

4.7.1.2 Moderator Vessel

The moderator vessel will be tested for maximum test pressure integrity, vacuum tightness, and for adequate heat transfer at the design flow rate of supercritical hydrogen. The vessel will be hydrostatically tested to 1.5 times the design pressure, and vacuum tightness will be monitored with the vessel pumped down to at least 10^{-6} torr. Supercritical hydrogen will be prototypically circulated through the moderator vessel at 20 K to validate vessel cooling capabilities. The outside surface of the vessel will be monitored for hot spots that would indicate poor heat transfer between the fluid and the vessel walls.

4.7.2 Integral Hydrogen Loop Tests

The planned integral hydrogen loop tests are now completed. These tests were performed at the AEDC in the Mark I vacuum chamber. The objectives of these tests were (1) to validate correct operation of the system design and demonstrate system stability under normal and faulted conditions of the system, (2) to provide benchmarking data for computer models, and (3) to test the proposed instrumentation and control system concept. A detailed test plan for both a liquid 4-bar hydrogen system and a supercritical hydrogen system were formulated, but only a supercritical system was tested. A summary of the design of the test facility and tests performed is provided in Sect. 2.7.1 of this report.

Although a full analysis of the tests performed is just now getting started, it is believed that almost all of the goals of the test plan were accomplished. It was determined during the test phase that certain tests could not be performed or could not be completed fully because of the limitations of the test facility.

One early indication from the test data is that the cold source control system performed very well. Even when unplanned abnormal conditions developed, the control system acted as designed and carried out automatic actions to stabilize and protect the system. A report on the analysis of these tests is expected to be issued during the coming year.

4.7.3 Acceptance Tests

Specifications for the fabrication of certain cold source components will include detailed acceptance test requirements as appropriate. These tests will in general be performed by the fabricator, but witnessed by a designated project representative as deemed appropriate. The specific acceptance tests to be performed will be identified as part of the fabrication specification development.

As an example, the cold source refrigerator contract has already been awarded. The detailed specification for this key component contains a number of acceptance test requirements. Some of these tests will be performed at the vendor's fabrication site, while others will be performed when equipment is installed at ORNL. The vendor must give ORNL notification of the intent to perform a test at least ten working days prior to the test. The tests to be performed and documented shall include, but not be limited to, the following:

1. Verify system capacity; operation at 100% of rated capacity (with simulated heat loads).
2. Verify stable operation of compressor at 20% rating.
3. Verify vibration limits for compressor and motor are met at 20% and 100% rated compressor capacity.
4. Verify vacuum system isolation and standby start on failure of operating unit.
5. Verify operation of refrigerator at a reduced load of 2 kW.
6. Perform piping system pressure tests in accordance with B31.3.
7. Perform pressure vessel pressure tests in accordance with ASME Section VIII.
8. The compressors and all other tubing and piping will be pressurized to their maximum allowable pressure with helium, and each joint will be examined with a mass spectrometer using a sniffer probe. The sensitivity of the sniffer system shall be 5×10^{-4} std. cc He/s.
9. Verify cold box vacuum system maintains system pressure as specified.
10. Perform cleanliness test of all internal piping.
11. Perform a final acceptance test of the fully installed refrigerator unit based on a final functional test plan approved by ORNL.

These types of component tests are critical to the validation/verification of the performance of the various cold source components before their installation into the cold source loop. These tests will be documented and reviewed prior to the components being integrated into the cold source operating system.

4.7.4 Installed Loop Tests With Moderator Plug Assembly Mounted Outside the HFIR Building

Present plans call for the initial installation of the cold source system to be with the hydrogen transfer lines routed away from the reactor building and connected to a moderator plug assembly housed in a temporary shelter outside the reactor building. A dummy heater would be added to simulate the reactor heat load, and a series of tests would be performed under near prototypic operating conditions. Although the detailed test plan for this series of tests has not yet been developed, it is anticipated that tests to be performed would include, but not be limited to:

1. full cool-down of the system jointly with the refrigerator and full cool-down of the system with the refrigerator already cold,
2. steady-state control with simulated reactor on and off operations,
3. transition to standby mode,
4. confirm optimum refrigerator settings for cool down and operating conditions,
5. optimize control parameters for loop temperatures and pressures, and
6. simulate as many faulted conditions as practical, and establish safe operational procedures.

The objectives of these tests would be to (1) provide a final demonstration of cold source performance before installation into the reactor building, (2) provide feedback on proposed operating procedures, and (3) provide hands-on operational training to the cold source operators in a geometry that poses less risk to the HFIR facility. These tests will be documented and reviewed before final installation of equipment in the HFIR reactor building.

4.7.5 Tests With the Cold Source Fully Installed With and Without Reactor Power

Once the test series described in Sect. 4.7.4 is completed and the project team, HFIR operations, and review committee(s) are satisfied with the results, the hydrogen transfer lines will be installed in the HFIR Building, and the moderator plug assembly will be installed in the HB-4 beam tube hole. Another series of tests will be performed without external heat load with the fully installed geometry to confirm leak tightness and cool down of the system prior to restart of the reactor.

Following the restart of the reactor, a test plan will be implemented to verify the performance of the cold source system under reactor operational conditions. Although the detailed start-up test plan has not been developed yet, it is believed that all tests could be performed during the first cycle of the restart. It is anticipated that optimal testing may require some special requests for modification of the reactor operation during this cycle (i.e., a power setback during the cycle may be requested as well as a possible reactor scram). The details of this start-up test plan will be reviewed and approved by the HFIR reactor operations group prior to its implementation. The successful completion of this testing activity will mark the end of the cold source project, and the cold source operation will be officially turned over to the HFIR operations group.

4.8 REFERENCES FOR CHAPTER FOUR

1. Research Reactors Division Administrative Policies and Procedures Manual RRAP 3.2, Configuration Control of Plant Design Modifications, ORNL/RRD/INT-12, Martin Marietta Energy Systems, Inc., Oak Ridge Natl. Lab., Sept. 28, 1987.
2. Ref. 1, RRAP-1.9, Procurement of Items and Services.
3. ORNL Standard Procedures, X-GP-16, Procurement of Critical Applications/Safety Class Items and Services at ORNL, Lockheed Martin Energy Research Corp., Oak Ridge Natl. Lab., Nov. 12, 1996.
4. Ref. 1, RRAP-3.3.1, Procedure for Fabrication Control Governing Modification of Reactor Systems and/or Replacement of Hardware.
5. DOE Order 5480.20A, Personnel Selection, Qualification, and Training Requirements for DOE Nuclear Facilities, U.S. Department of Energy, Nov. 15, 1994.
6. Resource Conservation and Recovery Act of 1970, Title 42, U.S.C. 6901 et seq.

Appendix A

**ORTHO/PARA-HYDROGEN
PARAMETRIC PHYSICS STUDY**

Appendix A: Ortho/Para-Hydrogen Parametric Physics Study

This appendix contains the details of a parametric study to compare the impact of the ortho/para-hydrogen ratio on the generated cold neutron beam spectrum in different geometries. Previous physics studies for the HFIR cold source project have indicated that changes in this ratio produced only small impacts on the cold neutron beam spectrum. As discussed in Sect. 2.4.4.1, this was found to be inconsistent with similar studies for the NIST hydrogen cold source. Therefore, ten geometries were examined that linked the HFIR cold source geometry to that of the NIST geometry by a series of step-changes in the model geometry. This study provides a qualitative assessment of how the physics (and resulting flux spectra) changes as one moves slowly from a NIST-like geometry to a HFIR cold source geometry. The remaining pages of this appendix provide the results for each of the geometries considered.

Geometry No. 1: NIST-like Geometry

Geometry No. 1, as illustrated to the right, is a simple representation of the NIST geometry that is a 2-cm thickness of hydrogen in a 32-cm diam spherical shell arrangement surrounded by a D_2O reflector. The results from the calculations performed with this geometry are provided in the plot below. As shown, there is a big difference between the 65% ortho-hydrogen cases and the all para-hydrogen case for the energy range of interest (< 2 meV). This is consistent with results previously reported by NIST.

FLUX SPECTRUM WITH D_2O REFL AND 32-CM-DIAM $\times$ 2-CM-THK SHELL OF LH_2
AS SEEN 940.9 CM DOWN A 20-CM-DIAM BEAM TUBE (0.6 DEG), LIKE NIST

Geometry No. 2: Reduced Beam Tube With Beryllium Reflector

Geometry No. 2, as illustrated to the right, was obtained by replacing the D_2O reflector in the previous geometry by beryllium and by reducing the beam tube size from 20 cm to 10 cm. The hydrogen thickness and shell size remained the same. The results, as illustrated in the plot below, are very interesting. The shape of the 65% ortho-hydrogen and the all para-hydrogen cases is significantly changed and in the energy range of interest (< 2 meV), the relative difference between the two curves is increased.

FLUX SPECTRUM WITH BE REFL AND 32-CM-DIAM x 2-CM-THK SHELL OF LH2
AS SEEN 477.5 CM DOWN A 10-CM-DIAM BEAM TUBE (0.6 DEG)

Geometry No. 3: Reduced Shell Diameter

Geometry No. 3, as illustrated to the right, is obtained from geometry No. 2 by reducing the diameter of the hydrogen shell from 32 cm to 26 cm. The thickness of the hydrogen shell remained at 2 cm. A comparison of the plotted results below with those reported for geometry No. 2 indicates that this change had little impact on the spectrum for the two cases.

FLUX SPECTRUM WITH BE REFL AND 26-CM-DIAM x 2-CM-THK SHELL OF LH2
AS SEEN 477.5 CM DOWN A 10-CM-DIAM BEAM TUBE (0.6 DEG)

Geometry No. 4: Additional Shell Diameter Reduction

Geometry No. 4, as illustrated to the right, was obtained from geometry No. 3 by making an additional reduction in the diameter of the shell. The shell was reduced from a diameter of 26 cm to a diameter of 20 cm. As in the discussion for geometry No. 3, the results shown in the plots below indicate that this reduction had little impact on the spectrum for the two scenarios.

FLUX SPECTRUM WITH BE REFL AND 20-CM-DIAM x 2-CM-THK SHELL OF LH2
AS SEEN 477.5 CM DOWN A 10-CM-DIAM BEAM TUBE (0.6 DEG)

Geometry No. 5: Third Reduction in Shell Diameter

Geometry No. 5, illustrated to the right, involves a third reduction of the shell diameter. In this case, the diameter is reduced to 14 cm. This is less than half the diameter of the reference NIST shell geometry. As in the previous shell diameter reductions, there was very little impact on the spectrum for the two scenarios. The results are plotted below.

FLUX SPECTRUM WITH BE REFL AND 14-CM-DIAM x 2-CM-THK SHELL OF LH2
AS SEEN 477.5 CM DOWN A 10-CM-DIAM BEAM TUBE (0.6 DEG)

Geometry No. 6: Fourth Reduction in Shell Diameter with No Hole in the Shell

Geometry No. 6, illustrated to the right, was obtained by reducing the diameter of the shell down to 10 cm and filling in the hole in the shell of Geometry No. 5. The hydrogen region is now a full spherical shell. Comparing the plots below with the plots presented for Geometry No. 5, the large impact of these changes can be noted. For the energy range of interest (< 2 meV), the brightness values calculated for the all para-hydrogen and the 65% ortho-hydrogen cases are much more similar with an average difference in the 10 to 15% range.

Geometry No. 7: Full Sphere Hydrogen Region

Geometry No. 7, illustrated to the right, was obtained by replacing the hydrogen shell of Geometry No. 6 with a full 10 cm sphere of hydrogen. Comparing the plots below with the previous geometry, it is found that this does not have a major effect. The difference between the all para-hydrogen and 65% ortho-hydrogen cases is increased slightly to about 25 to 30% over the energy range of interest.

Geometry No. 8: Dented Sphere

Geometry No. 8, illustrated to the right, was obtained by putting a notch in the solid hydrogen sphere of Geometry No. 7. This gives the hydrogen half-moon appearance. As seen in the plots below, the impact of this geometry change resulted in a decrease in the difference between the all para-hydrogen assumption and the 65% ortho-hydrogen assumption over the range of interest (< 2 meV). In fact, over much of the energy range of interest, the difference between the two is within the statistics of the Monte Carlo calculations.

FLUX SPECTRUM WITH BE REFL AND A 10-CM-DIAM DENTED SPHERE OF LH2
AS SEEN 477.5 CM DOWN A 10-CM BEAM TUBE (0.6 DEG)

Geometry No. 9: Dented Sphere with Hydrogen Inlet and Outlet Flow Wings

Geometry No. 9, illustrated to the right, was obtained by adding a region to simulate the inlet and outlet hydrogen flow regions of the proposed HFIR cold source. These additional hydrogen wings create a partial flux trap that elevates the brightness of the cold neutron beam. In addition, as seen from the plots below, the all para-hydrogen and 65% ortho-hydrogen cases are calculated to produce the same beam over the energy range of interest considering the few percent statistical uncertainty in each calculation.

FLUX SPECTRUM WITH BE REFL AND A 10-CM-DIAM DENTED SPHERE OF LH₂ WITH WINGS, AS SEEN 477.5 CM DOWN A 10-CM BEAM TUBE (0.6 DEG)

Geometry No. 10: HFIR Cold Source Geometry

Geometry No. 10, illustrated by the two figures to the right, is the last step in geometry changes studied to transition from the NIST cold source geometry to the HFIR cold source geometry. This final change is the addition of an aluminum window 200 cm down the beam tube (shown in the lower figure). As expected, this change had very little impact on the relationship between the beams produced from the all para-hydrogen and 65% ortho-hydrogen assumptions. The only difference between the plots below and those presented for Geometry No. 9 is a small decrease in magnitude for each case expected from the additional aluminum in the beam path.

Appendix B
CALCULATION STEPS FOR LH₂ STRUCTURES/AIR
EQUILIBRATION ANALYSIS

Appendix B: Calculation Steps For LH₂ Structures/Air Equilibration Analysis

Step 1. Determine the amount of energy required to raise the aluminum temperature from the initial value (15 K) to the condensation temperature of nitrogen (77 K). Add to this the amount of energy required to boil off an amount of LH₂ required to produce a volume of GH₂ inside the LH₂ structures at the saturation temperature corresponding to 4-bar pressure. This total amount of energy is denoted as Q1. Determine the mass of air at ambient temperature (TAMBNT) such that when cooled to 77 K, it will completely condense all O₂ and N₂ and release an amount of energy equal to Q1. No frozen condensate will result as the freezing temperatures of O₂ and N₂ are below 77 K. Because the saturation temperature of O₂ is 90 K, this end state temperature of 77 K requires the removal of sensible energy from the O₂ liquid once it has formed at 90 K.

The quantities XMO21 and XMN21 represent the end-of-step condensed masses of O₂ and N₂ assumed to exist on the LH₂ structures. The temperature of the condensate and the LH₂ structures is uniform at 77 K. No gas exists in the vacuum tube vacuum volume, VTUBE.

Step 2. Warm the LH₂ aluminum structures from 77 K to 90 K. Add to this the amount of energy required to heat the liquid O₂ mass XMO21 from 77 K to 90 K. Add to this the amount of energy required to boil off the liquid N₂ mass XMN21 at 77 K and to heat this N₂ vapor from 77 K to 90 K. This total amount of energy is denoted as Q2. Determine the mass of ambient temperature air required to be cooled from TAMBNT to 90 K (and completely condensing the O₂ content at 90 K) such that an amount of energy Q2 is released. The quantities XMO22 and XMN22 represent the masses of O₂ and N₂ required to produce this heat exchange.

At the end of Step 2, the mass of N₂ in the vacuum tube is XMN2=XMN21+XMN22 and is all gas (GASN2=XMN21+XMN22) at a temperature of 90 K. The total mass of O₂ in the vacuum tube is XMO2=XMO21+XMO22 and is all liquid (XO2CND=XMO21+XMO22) on the LH₂ structures at 90 K. The O₂ gas mass in the vacuum tube atmosphere remains at zero (GASO2=0.0). The LH₂ structures temperature is also at 90 K.

Evaluate the pressure of the N₂ gas in the vacuum tube vacuum space, P2, using the temperature of 90 K and a gas mass of GASN2. As mentioned previously, all calculations to date have shown that P2 is less than 1 atm. If P2 is less than or equal to 1 atm, then proceed to Step 3. If P2 is greater than 1 atm, then write an error message and stop the calculation. Additional logic must be developed for this situation.

Step 3. Calculate the partial volume (VOLN2) of N₂ gas in the vacuum tube assuming atmospheric pressure and a temperature of 90 K. Determine the volume of ambient temperature gas at 1 atm pressure required to fill the remaining volume of the vacuum tube (VOLAIR=VTUBE - VOLN2). Determine the mass of O₂ and N₂ at ambient temperature (TAMBNT) associated with VOLAIR. Denote these masses as XMO23 and XMN23. Equilibrate gas masses XMO23 and XMN23 at TAMBNT with the cold N₂ gas mass GASN2 at 90 K. The equilibrated gas temperature is denoted by T3. Check that T3 is greater than 90 K. Update the O₂ and N₂ gas inventories (e.g., GASO2 = GASO2+XMO23 and GASN2 = GASN2+XMN23). Determine the gas mixture pressure P3 and check that it is 1 atm. Set the gas temperature TGAS to T3 (e.g., TGAS=T3).

At the end of Step 3, the vacuum tube gas is composed of the masses GASO2 and GASN2 at temperature TGAS=T3 (> 90 K) and pressure P3 (=1 atm). The LH₂ structures remain in thermal equilibrium with a mass of O₂ liquid (XO2CND) at 90 K. Set the iteration counter I to zero for possible use in Step 4 below.

Step 4. Update iteration counter I, I=I+1. Determine the quantity of energy (Q4) required to evaporate the entire mass of liquid O₂ (XO2CND) at 90 K. Determine the final temperature of the gas

mixture GASO₂ and GASN₂ at TGAS assuming that Q₄ is removed. Denote this final gas temperature by T_{4A}.

If T_{4A} is greater than 90 K, then the gas in the vacuum tube at TGAS has enough energy to completely evaporate the O₂ liquid (XO₂CND). Equilibrate the mass of O₂ gas (previously liquid XO₂CND) at 90 K with GASO₂ and GASN₂ at T_{4A}. Denote the new gas temperature as T₄. Set TGAS to T₄ and update the O₂ gas inventory, GASO₂ = GASO₂ + XMO₂₁ + XMO₂₂. Decrement the O₂ liquid remaining on the structures, XO₂CND = XO₂CND - XMO₂₁ - XMO₂₂ = 0.0. Determine the gas pressure of GASO₂ and GASN₂ in the vacuum tube at temperature TGAS. Denote this pressure as P₄. The LH₂ structures remain at 90 K. No iteration is required. Go to Step 5.

If T_{4A} is less than 90 K (and TGAS is greater than 90 K), then the gas mixture GASO₂ and GASN₂ at TGAS does not possess sufficient energy to completely evaporate the liquid O₂ mass XO₂CND. Iteration may be involved. Calculate the energy released from GASO₂ and GASN₂ in cooling from TGAS to 90 K. Denote this energy as Q_{4A} and determine the fraction of the O₂ liquid that can be evaporated from the LH₂ surfaces, FRACT = Q_{4A}/Q₄. Update the O₂ gas inventory and the O₂ liquid inventory (e.g., GASO₂ = GASO₂ + FRACT * XO₂CND and XO₂CND = XO₂CND - FRACT * XO₂CND). The gas temperature TGAS is 90 K and set T₄ to 90 K. Determine the partial volume of GASO₂ and GASN₂ at TGAS at 1 atm pressure and denote as VGAS.

If VGAS is greater than or equal to the vacuum tube free volume, VTUBE, then evaluate the pressure P₄ of GASO₂ and GASN₂ at TGAS and stop the calculation. The maximum amount of O₂ has evaporated from the surface of the LH₂ structures. The LH₂ structures, the remaining O₂ liquid XO₂CND, and the vacuum tube gas (GASO₂ and GASN₂) are at 90 K, with the vacuum tube gas pressure at or above 1 atm. No further equilibration can occur.

If VGAS is less than VTUBE, determine the partial volume of the vacuum tube to be filled by ambient temperature air (VOLAIR) at TAMBNT and 1 atm pressure. Determine the O₂ and N₂ masses associated with VOLAIR and denote by XMO_{2I} and XMN_{2I}. Equilibrate XMO_{2I} and XMN_{2I} at TAMBNT with GASO₂ and GASN₂ at TGAS. Denote the equilibrated temperature as T_I and set TGAS to T_I. Update the vacuum tube total O₂ and N₂ inventories as well as the gas inventories (e.g., XMO₂ = XMO₂ + XMO_{2I}, GASO₂ = GASO₂ + XMO_{2I}, XMN₂ = XMN₂ + XMN_{2I}, and GASN₂ = GASN₂ + XMN_{2I}). If the remaining O₂ liquid (XO₂CND) is zero, then set T₄ = TGAS, evaluate the pressure of GASO₂ and GASN₂ at TGAS occupying volume VTUBE and denote as P₄, and go to Step 5. If the remaining O₂ liquid (XO₂CND) is greater than zero, go to the beginning of Step 4 and perform another iteration.

At the end of Step 4, two possible state conditions exist. First, there may be O₂ liquid left on the LH₂ structures (XO₂CND). The gas (GASO₂ and GASN₂), the O₂ liquid (XO₂CND), and the LH₂ structures are all at 90 K (T₄) and the vacuum tube pressure (P₄) is at or above 1 atm pressure. The equilibration process for this state is complete, and Step 5 below is bypassed.

The second end state for Step 4 is one where all O₂ liquid has been evaporated (XO₂CND = 0.0). All of the tube inventory of O₂ and N₂ is gaseous (GASO₂ and GASN₂) at temperature of TGAS greater than 90 K, and the gas pressure is less than 1 atm. Since the LH₂ structures remain at 90 K and the gas pressure is less than 1 atm, more ambient temperature gas may flow into the vacuum tube and additional thermal equilibration may occur. This is performed in Step 5. The iteration counter I is reset to zero for use in Step 5 iteration. The LH₂ structures remain at 90 K (e.g., T_{AL} = 90 K).

Step 5. Increase the iteration counter I = I + 1. Determine the partial volume (VGAS) of the vacuum tube gas (GASO₂ and GASN₂) at 1 atm pressure and a temperature of TGAS. Check that VGAS is less than VTUBE and calculate the remaining vacuum tube volume (VOLAIR = VTUBE - VGAS) to be occupied by ambient temperature gas at TAMBNT and 1 atm pressure. Determine the O₂ and N₂ inventories associated with the ambient temperature air volume VOLAIR and denote these gas masses as XMO_{2I} and XMN_{2I}. Equilibrate XMO_{2I} and XMN_{2I} at TAMBNT with GASO₂ and GASN₂ at TGAS and denote the resulting temperature as T_I. Set TGAS to T_I and update the whole tube inventories of O₂

and N_2 as well as the gas inventories (e.g., $XMO2 = XMO2 + XMO2I$, $GASO2 = GASO2 + XMO2I$, $XMN2 = XMN2 + XMN2I$, and $GASN2 = GASN2 + XMN2I$). Calculate the gas pressure (PI) of GASO2 and GASN2 at TGAS in volume VTUBE and check that it is 1 atm.

Equilibrate the gas at TGAS with the aluminum structures at TAL. This requires the evaluation of changes in aluminum energy with respect to temperature and takes into account the strong temperature dependence of aluminum specific heat in the low temperature region. The resulting equilibrated gas/structure temperature is denoted as TI. Set TGAS and TAL to TI and determine the gas pressure of GASO2 and GASN2 at TGAS in the vacuum tube VTUBE. This pressure is denoted as PI. If PI is equal to 1 atm, then the equilibration process is complete and the calculation is terminated. If PI remains less than 1 atm, then additional ambient temperature air may flow into the vacuum tube and equilibrate with the LH2 structures. In this case, go to the beginning of Step 5 and perform another iteration.

INTERNAL DISTRIBUTION

1. J. B. Ball	28-30. K. A. Morgan
2. J. A. Bucholz	31. H. A. Mook
3. S. J. Chang	32. R. M. Moon
4. D. H. Cook	33. R. H. Nelson
5. M. B. Farrar	34. L. D. Proctor
6. K. Farrell	35. C. T. Ramsey
7. G. F. Flanagan	36. J. B. Roberto
8. E. C. Fox	37. K. B. Rothrock
9. J. D. Freels	38-44. D. L. Selby
10. M. L. Gildner	45. B. F. Siefken
11-15. R. M. Harrington	46. J. R. Weir III
16. W. E. Hill	47. M. W. Wendel
17. S. A. Hodge	48. C. D. West
18. M. T. Hurst	49. G. D. Wignall
19. C. R. Hyman	50. P. T. Williams
20. M. W. Kohring	51. M. A. Woody
21. J. E. Lee	52. G. L. Yoder
22. R. A. Lillie	53-54. Central Research Library
23. M. A. Linn	55-56. Laboratory Records Dept.
(OSTI)
24-26. A. T. Lucas	57. Laboratory Records Dept. (RC)
27. R. S. McKeehan	

EXTERNAL DISTRIBUTION

58. L. W. Boyd (DOE/ORO), ORNL, Bldg 7964-B, MS-6390, Oak Ridge, Tennessee 37831
59. G. Bauer, Paul Scherrer Institut (PSI), 5232 Villigen PSI, Switzerland
60. K. Böning, Fakultät für Physik E21, Technische Universität München, D-8046 Garching, FR Germany
61. S. K. Chae, Director, HANARO Center, KAERI, P.O. Box 105, Yusong, Taejeon, Korea 305-600
62. F. J. Edeskuty, 913 Tewa Loop, Los Alamos, New Mexico 87544
63. R. Fair (DOE/ORO), ORNL, Bldg 7964-B, MS-6390, Oak Ridge, Tennessee 37831
64. R. F. Lidstone, Atomic Energy of Canada Ltd., Whiteshell Nuclear Research Establishment, Pinawa, Manitoba, ROE 1LO Canada

65. J. M. Rowe, National Institute of Standards and Technology, Quince Orchard Road, Building 235, Rm. A100, Gaithersburg, Maryland 20899
66. D. C. Royer, Deputy Division Manager, Brookhaven National Laboratory, Reactor Division, Building 120, 81 Cornell Ave., Upton, NY 11973
67. E. G. Tourigney, U.S. Department of Energy, Technical Support Branch, Office of Civilian Reactor Development NE443, 1000 Independence Avenue SW, Washington, DC 20585