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Revision 2

ASME Code Efforts Supporting HTGRs

D. K. Morton

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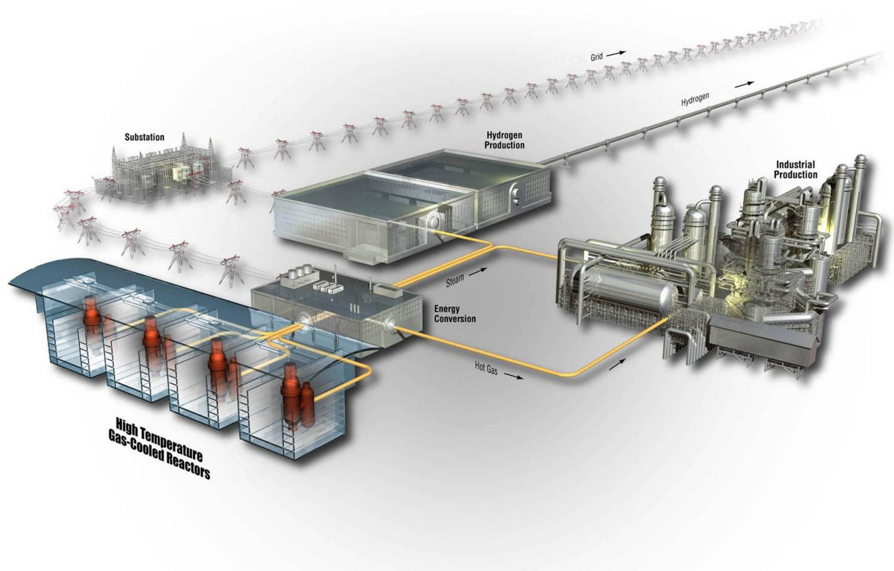

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Approved by:

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ABSTRACT

In 1999, an international collaborative initiative for the development of advanced (Generation IV) nuclear reactors was started. The idea behind this effort was to bring nuclear energy closer to the needs of sustainability, to increase proliferation resistance, and to support concepts able to produce energy (both electricity and process heat) at competitive costs. The U.S. Department of Energy (DOE) has supported the development of a high temperature gas-cooled reactor plant under the Next Generation Nuclear Plant (NGNP) Project in the past and currently under the Very High Temperature Reactor Technology Development Office (VHTR TDO). This support has included research and development of pertinent data, initial regulatory discussions, and engineering support of various codes and standards development. This report discusses the various applicable American Society of Mechanical Engineers (ASME) codes and standards that are being developed to support these high temperature gas-cooled reactors during construction and operation. ASME is aggressively pursuing these codes and standards to support an international effort to build the next generation of advanced reactors for a worldwide benefit. It is the support from the international nuclear industry, including DOE and the VHTR TDO, that helps ASME achieve this worthy goal.

SUMMARY

In 1999, an international collaborative initiative for the development of advanced (Generation IV) nuclear reactors was started. The idea behind this effort was to bring nuclear energy closer to the needs of sustainability, to increase proliferation resistance, and to support concepts able to produce energy (both electricity and process heat) at competitive costs. The U.S. Department of Energy (DOE) has supported this effort by pursuing the development of a high temperature gas-cooled reactor (HTGR) under the Next Generation Nuclear Plant (NGNP) Project in the past and continues this support through the Very High Temperature Reactor Technology Development Office (VHTR TDO). This support has included research and development of pertinent data, initial regulatory discussions, and engineering support of various codes and standards development.

This report has been written to provide a current update on those major American Society of Mechanical Engineers (ASME) codes and standards (both nuclear and non-nuclear) that will most likely be used in the construction of an HTGR plant. After a brief history of ASME codes and standards, the purpose of ASME codes and standards are discussed as well as the organizational structure of ASME committees. Next, additional background information on ASME activities occurring in the last few years [including Fiscal Year (FY) 2011] that have been supporting the development of codes and standards relating to HTGRs is presented. This was done in order to better understand the current FY 2012 accomplishments that ASME has achieved regarding codes and standards for HTGRs. Finally, expectations of what ASME activities might achieve in the near future are presented.

This report cites examples that demonstrate ASME is indeed supporting the development of HTGRs by providing numerous codes and standards from which an HTGR plant can be constructed, licensed, and operated. Not all of the HTGR requirements are currently published, but the codes and standards process is effectively moving forward. A prime example is Section III, Division 5 of the ASME Boiler and Pressure Vessel Code, which was issued on November 1, 2011. The foundation for HTGR codes and standards has already been laid and the development and approval of appropriate codes and standards will continue.

ASME develops and maintains codes and standards that are used internationally to establish rules of safety, develop technology while achieving standardization, and more, all done with groups of volunteers from around the world. DOE and the NGNP Project /VHTR TDO have also supported this effort by funding the Generation IV Reactor Materials Project and supporting national laboratory personnel and subcontractors to participate in developing ASME codes and standards for HTGRs. That effort is beginning to provide substantial benefit. Code revisions based on results from the Generation IV Reactor Materials Project are now being approved by ASME committees and incorporated into the codes and standards. In multiple cases, funded personnel have assumed committee leadership positions, are developing Code revision proposals, and are actively guiding those proposals through the ASME approval process, expediting the issuance of rules for HTGRs.

ACKNOWLEDGEMENTS

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CONTENTS

TABLES

ACRONYMS

ANS	American Nuclear Society
ANSI	American National Standards Institute
ASME	American Society of Mechanical Engineers
ASTM	American Society for Testing and Materials
BNCS	Board on Nuclear Codes and Standards
BPTCS	Board on Pressure Technology Codes and Standards
BPV	Boiler and Pressure Vessel
CFR	U.S. Code of Federal Regulations
DOE	U.S. Department of Energy
FY	Fiscal Year
GCC	Graphite Core Components
HTGR	High Temperature Gas-cooled Reactor
INL	Idaho National Laboratory
LMR	Liquid Metal Reactor
LWR	Light Water Reactor
NDE	Nondestructive Examination
NGNP	Next Generation Nuclear Plant
NRC	U.S. Nuclear Regulatory Commission
OM	Operation and Maintenance
ORNL	Oak Ridge National Laboratory
PRA	Probabilistic Risk Assessment
QME	Qualification of Mechanical Equipment
R&D	Research and Development
RFP	Request For Proposal
RIM	Reliability and Integrity Management
RTNSS	Regulatory Treatment of Non-Safety Systems
SG	Subgroup
ST	Standards Technology
SWG	Special Working Group
TDO	Technology Development Office
VHTR	Very High Temperature Reactor
WG	Working Group

ASME Code Efforts Supporting HTGRs

1. INTRODUCTION

The American Society of Mechanical Engineers (ASME) has been involved in codes and standards activities for over 125 years. Starting with the Society's first performance test code for steam boilers in 1884, ASME has expanded its areas of responsibility to also include codes and standards for pressure vessels, piping, screw threads, steel stacks, geometric dimensioning and tolerancing, cranes and hoists, elevators and escalators, and much more. The goal of ASME is to be the world leader in mechanical and multidisciplinary engineering codes, standards, conformity assessment programs, and related products and services.

The most widely recognized ASME code is the International Boiler and Pressure Vessel (BPV) Code, first originating in 1914. The ASME BPV Code is now adopted in part or in its entirety by all 50 states and numerous municipalities and territories of the United States and all the provinces of Canada. It also has additional use beyond North America. The BPV Code addresses both nuclear and non-nuclear facilities.

The American National Standards Institute (ANSI) is a privately funded federation of business and industry, standards developers, trade associations, labor unions, professional societies, consumers, academia, and government agencies. ASME is an accredited Standards Developing Organization that meets the due process requirements of ANSI. Codes and standards developed under an accredited program may be designated as American National Standards.

ASME has played a vital role in supporting the nuclear power industry since its inception, when ASME codes, standards and conformity assessment programs originally developed for fossil fuel fired power plants were applied to nuclear power plant construction. Today, ASME nuclear codes and standards exist to ensure public safety, support global trade, develop technology, and foster knowledge transfer while easing government's regulatory burden. By uniting technical and quality requirements—enhanced by a time-proven consensus approach to decision-making—ASME standards can be adopted, applied, and accepted universally.

In 1999, an international collaborative initiative for the development of advanced (Generation IV) reactor power plants was started. The idea behind this effort was to bring nuclear energy closer to the needs of sustainability, to increase proliferation resistance, and to support concepts able to produce energy (both electricity and process heat) at competitive costs. The U.S. Department of Energy (DOE) supported the development of a high temperature gas-cooled reactor (HTGR) plant under the Next Generation Nuclear Plant (NGNP) Project. This support included conceptual engineering studies, research and development (R&D) of pertinent data, initial regulatory discussions, and engineering support of various codes and standards development. In Fiscal Year (FY) 2012, the NGNP Project was transitioned to a research and development program, and the codes and standards activities continued under the Very High Temperature Reactor Technology Development Office (VHTR TDO).

This report has been written to provide a current update on those major codes and standards (both nuclear and non-nuclear) that will most likely be used in the construction and operation of an HTGR plant. Some of these codes and standards are adequate as is, some need to be modified, and some need to be newly developed for use with an HTGR. For the purposes of this report (as defined in Section III, Division 1, Subsection NCA-9000 of the ASME BPV Code),[1] the term 'construction' is defined to be "an all-inclusive term comprising materials, design, fabrication, examination, testing, inspection, and certification required in the manufacture and installation of an item." Hence, ASME addresses a broad range of technical issues in many of their codes and standards.

In the following sections of this report, the purpose of ASME codes and standards of interest are discussed as well as the organizational structure of various ASME committees. Next, additional background information on ASME activities occurring over the last few years that have been supporting the development of codes and standards relating to HTGRs is presented. This was done in order to better understand the current FY 2012 accomplishments that ASME has achieved regarding codes and standards for HTGRs. Finally, expectations of what ASME activities might achieve in the near future are presented.

1.1 Purpose of ASME Codes and Standards

Before proceeding further, it is appropriate to clarify just what a standard and a code actually are. According to ASME[2]:

> *A standard can be defined as a set of technical definitions and guidelines, "how to" instructions for designers, manufacturers and users. Standards promote safety, reliability, productivity and efficiency in almost every industry that relies on engineering components or equipment. Standards can run from a few paragraphs to hundreds of pages, and are written by experts with knowledge and expertise in a particular field who sit on many committees. Standards are considered voluntary because they serve as guidelines, but do not of themselves have the force of law. ASME cannot force any manufacturer, inspector, or installer to follow ASME standards. Their use is voluntary. Standards become mandatory when they have been incorporated into a business contract or incorporated into regulations. A code is a standard that has been adopted by one or more governmental bodies and has the force of law.*

The purpose of codes and standards can be summarized in one word: effectiveness. Codes and standards assure that appropriate margins exist for the safe use of the final product. Codes and standards also define the quality levels needed and promote the standardization of components, parts, and items for ease of interchangeability. It is not uncommon for parts (such as a nut and a bolt or piping and a fitting) to be procured in different locations from different companies with the expectation that they will fit together as needed.

The following report subsections identify the specific ASME codes and standards that will be addressed in this report in further detail. Each subsection will address the purpose of the code or standard.

1.1.1 Boiler and Pressure Vessel Code

As mentioned previously, the ASME BPV Code addresses construction rules for both nuclear and non-nuclear facility components. It contains 12 distinct sections. However, for the purposes of this report addressing the major codes and standards that are expected to be used for the construction of an HTGR plant, Sections III, VIII, and XI of the BPV Code will be of primary importance along with the critical supporting sections that include Sections II, V, and IX. The 2010 Edition of the ASME BPV Code with the 2011 Addenda will be used as the basis for defining BPV Code content and status.

The BPV Code typically addresses components that perform the function of a pressure boundary. This means that the construction rules focus on achieving the pressure integrity of these components for safety purposes. As usual, there are exceptions, especially in Section III, for items like supports and core support structures (although core support structures do experience pressure difference loadings). In these exceptions, the ASME committees felt it was appropriate to develop these additional rules in an effort to provide adequate construction rules for those components whose failure could adversely affect pressure retaining components and to provide a complete set of rules for nuclear facility construction.

The *Foreword* to the BPV Code contains many additional insights on what the BPV Code does and does not address. Issues such as the use of engineering judgment, computer analyses, dimensional tolerances, code cases, dates[a] when Code Editions may be used, and more are discussed. It is strongly suggested that persons interested in gaining a better understanding of the ASME BPV Code (or other codes and standards) read the *Foreword*.

1.1.1.1 Section III

Section III of the ASME BPV Code addresses the rules for nuclear facility components. Components include vessels and systems, pumps, valves, piping, component supports, and core support structures. The components and supports covered by Section III are intended to be installed in a nuclear power system that serves the purpose of producing and controlling the output of thermal energy from nuclear fuel and those associated systems essential to the functions and overall safety of the nuclear power system. Section III provides requirements for new construction and includes consideration of mechanical and thermal stresses due to cyclic operation. Deterioration, which may occur in service as a result of radiation effects, corrosion, erosion, or instability of the material, is typically not addressed.

Section III currently has five Divisions. Division 1 provides requirements for the materials, design, fabrication, examination, testing, inspection, installation, certification, stamping, and overpressure protection of nuclear facility components, typically those associated with light water reactors (LWRs). Section III also provides requirements for concrete reactor vessels and containments under Division 2 and containment systems for storage and transport packagings for spent nuclear fuel and high-level radioactive waste under Division 3. Division 4 has been assigned for future rules addressing components for fusion energy devices. Division 5 is a new division recently approved and provides construction rules associated with high temperature reactors, including HTGRs and liquid metal reactors (LMRs), and was published with a November 1, 2011 issuance date (see Sections 2.6.5 and 3.6.5 for further details). Table 1 provides a listing of the divisions and subsections that comprise the current Section III. The BPV Committee on Construction of Nuclear Facility Components (III) is responsible for the contents.

1.1.1.2 Section VIII

Section VIII of the BPV Code addresses the rules needed to construct non-nuclear pressure vessels. Pressure vessels not subject to nuclear requirements (e.g., the balance of plant pressure vessels) can utilize these rules. Section VIII provides requirements applicable to the design, fabrication, inspection, testing, and certification of pressure vessels operating at either internal or external pressures exceeding 15 psig (100 kPa). This pressure may be obtained from an external source or by the application of heat from a direct or indirect source, or any combination thereof. Section VIII includes three divisions:

- Division 1 – Basic Coverage
- Division 2 – Alternative Rules
- Division 3 – Alternative Rules for Construction of High Pressure Vessels.

In comparison to the basic design-by-rule approach of Division 1, the requirements of Division 2 on materials, design, and nondestructive examination (NDE) are more rigorous. However, Division 2 also permits higher allowable design stress intensity limits, resulting in more efficient construction (i.e., thinner vessel walls). The rules of Section VIII allow temperatures greater than those allowed in Section III, Division 1 (with the exception of Subsection NH). Division 3 addresses vessels operating at either internal or external pressures generally above 10,000 psi (69 MPa). The BPV Committee on Pressure Vessels (VIII) is responsible for the contents of Section VIII.

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a. Approval of revisions by the higher ASME Committees does not result in their immediate publication but when the next scheduled publication cycle occurs. Publication (meaning copies available for purchase) occurs near the date of issuance, either before or shortly thereafter. The BPV Code may be used beginning with the date of issuance but compliance is mandatory after the time interval specified in the Foreword of each Section. A similar process exists for other ASME codes.

Table 1. Section III – Rules for Construction of Nuclear Facility Components.

Subsection Identifier	Subsection Title
NCA	General Requirements for Division 1 and 2
Division 1	
NB	Class 1 Components
NC	Class 2 Components
ND	Class 3 Components
NE	Class MC Components
NF	Supports
NG	Core Support Structures
NH	Class 1 Components in Elevated Temperature Service
Appendices	Appendices
Division 2 – Code for Concrete Containments	
CC	Concrete Containments (Prestressed or Reinforced)
Division 3 – Containments for Transportation and Storage of Spent Nuclear Fuel and High Level Radioactive Material and Waste	
WA	General Requirements
WB	Class TC Transportation Containments
WC	Class SC Storage Containments
Division 4 – (reserved for fusion energy devices)	
Division 5 – High Temperature Reactors	
HA	General Requirements
HB	Class A Metallic Pressure Boundary Components
HC	Class B Metallic Pressure Boundary Components
HF	Class A and B Metallic Supports
HG	Class A Metallic Core Support Structures
HH	Class A Non-Metallic Core Support Structures

1.1.1.3 Section XI

Section XI currently provides Division 1 rules for the examination, inservice testing and inspection, and repair and replacement of components and systems in light water-cooled nuclear power plants. Division 2 rules for inservice inspection and testing of components of gas-cooled nuclear power plants were no longer published in the 1995 Edition of the ASME BPV Code (or thereafter) because the need for those requirements ended with the decommissioning of the only gas-cooled reactor (Fort St. Vrain) to which those rules applied. See Sections 2.5 and 3.5 for recent efforts to develop a new Division 2. Division 3 rules for inservice inspection and testing of components in liquid-metal-cooled nuclear power plants were no longer published in the 2004 Edition of the ASME BPV Code (or thereafter) presumably due to little or no identified use of the rules.

Application of Section XI of the BPV Code begins when the requirements of the construction code (typically Section III) have been satisfied. The rules of Section XI constitute requirements to maintain the nuclear power plant while in operation and to return the plant to service, following plant outages, and repair or replacement activities. The rules require a mandatory program of scheduled examinations, testing, and inspections to evidence adequate safety. The method of NDE to be used and flaw-size characterization are also contained within Section XI. The BPV Committee on Nuclear Inservice Inspection (XI) is responsible for the contents of Section XI.

1.1.1.4 Supporting BPV Sections

The major supporting sections of the ASME BPV Code that will most likely be used to support the construction of an HTGR include the following three sections:

- Section II – Materials
  - Part A – Ferrous Material Specifications
  - Part B – Nonferrous Material Specifications
  - Part C – Specifications for Welding Rods, Electrodes, and Filler Metals
  - Part D – Properties.
- Section V – Nondestructive Examination
- Section IX – Welding and Brazing Qualifications.

Section II

Part A is a service book to the other BPV Code sections, providing material specifications for ferrous materials adequate for safety in the field of pressure equipment. These specifications contain requirements for heat treatment, manufacture, chemical composition, heat and product analyses, mechanical test requirements and mechanical properties, test specimens, and methods of testing. They are designated by SA numbers[b] and are derived from American Society for Testing and Materials (ASTM) International 'A' specifications. Part B is also a service book to the other BPV Code sections providing material specifications for nonferrous materials adequate for safety in the field of pressure equipment. Containing the same type of information as Part A, the Part B specifications are designated by SB numbers[a] and are derived from ASTM International 'B' specifications. The Part C service book provides material specifications for the manufacture, acceptability, chemical composition, mechanical usability, surfacing, testing requirements and procedures, operating characteristics, and intended uses for welding rods, electrodes, and filler metals. These specifications are designated by SFA numbers[a] and are derived from American Welding Society specifications. Finally, Part D is a service book to other BPV Code sections providing tables of design stress values, tensile and yield strength values, and tables and charts of metallic material properties. Part D facilitates ready identification of specific materials to specific sections of the BPV Code. Part D also contains appendices that contain criteria for establishing allowable stress and stress intensity values, the bases for establishing external pressure charts, and information required for the approval of new materials. The BPV Committee on Materials (II) is responsible for the contents of Section II.

Section V

Section V contains requirements and methods for NDE, which are referenced and required by the other BPV Code sections. It also includes manufacturer's examination responsibilities, duties of authorized inspectors, and requirements for inspection, examination, and qualification of personnel. Examination methods are intended to detect surface and internal discontinuities in materials, welds, and fabricated parts and components. A glossary of related terms is also included. The BPV Committee on Nondestructive Examination (V) is responsible for the contents of Section V.

Section IX

Section IX contains rules relating to the qualification of welding and brazing procedures as required by the other BPV Code sections for component manufacture. It also covers rules relating to the qualification and requalification of welders, brazers, and welding and brazing operators in order that they

b. ASME designations for specifications regarding ferrous materials (SA), nonferrous materials (SB), or for welding rods, electrodes, and filler metals (SFA).

may perform welding or brazing as required by other BPV Code sections in the manufacture of components. Welding and brazing data cover essential and nonessential variables specific to the welding or brazing process used. The BPV Committee on Welding and Brazing (IX) is responsible for the contents of Section IX.

1.1.2 Probabilistic Risk Assessment Standard

The probabilistic risk assessment (PRA) method plays an important role in ensuring that the operation of a nuclear power plant presents no undue risk to public health and safety. The PRA systematically looks at how the pieces of a complex system work together to ensure safety. The PRA allows analysts to quantify risk and identify what systems could have the most impact on safety. Hence, the use of the PRA method is important to the proper construction of nuclear power plants and the proper execution of a PRA is best utilized when the methodology follows a proper standard.

The Standards Committee on Nuclear Risk Management[c] issued RA-S-2008, "Standard for Level 1/Large Early Release Frequency Probabilistic Risk Assessment for Nuclear Power Plant Applications."[3] This standard sets forth requirements for probabilistic risk assessments used to support risk-informed decisions for commercial LWR plants, and prescribes a method for applying these requirements for specific applications. See Sections 2.2 and 3.2 for more recent development efforts on PRA standards, including a new PRA standard for non-LWRs.

1.1.3 Operation and Maintenance Standard

The Standards Committee on Operation and Maintenance of Nuclear Power Plants has recently issued OM-2009, "Operation and Maintenance of Nuclear Power Plants."[4] This standard establishes the requirements for preservice and inservice testing and examination of pumps, valves, and dynamic restraints (snubbers) to assess their operational readiness in LWR power plants. It identifies the components subject to test or examination, responsibilities, methods, intervals, parameters to be measured and evaluated, criteria for evaluating the results, corrective action, personnel qualification, and record keeping. These requirements apply to: (a) pumps and valves that are required to perform a specific function in shutting down a reactor to the safe shutdown condition, in maintaining the safe shutdown condition, or in mitigating the consequences of an accident; (b) pressure relief devices that protect systems or portions of systems that perform one or more of these three functions; and (c) dynamic restraints used in systems that perform one or more of these three functions.

1.1.4 Qualification of Mechanical Equipment Standard

The Standards Committee on Qualification of Mechanical Equipment Used in Nuclear Facilities has issued the qualification of mechanical equipment (QME) document QME–1-2007, "Qualification of Active Mechanical Equipment Used in Nuclear Power Plants."[5] This standard describes the requirements and guidelines for qualifying active mechanical equipment (including pumps, valves, and dynamic restraints [snubbers]) used in nuclear power plants. The requirements and guidelines presented include the principles, procedures, and methods of qualification.

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c. In 2010, ASME and ANS decided to combine their respective committees responsible for developing PRA standards into a joint committee. This new standards committee was expected to be known as the ASME/ANS Joint Committee on Nuclear Risk Management. The first joint committee meeting occurred in September 2010. The joint committee was organized to have a number of subcommittees and working groups to maintain the standard, answer inquiries, make interpretations, and issue addenda and revisions to the standard. Since 2010, these committees continued their work, assuming the official joint committee would eventually occur. However, a legal agreement between the two societies was not finalized until 2012. Efforts to officially reorganize are underway as of the writing of this report. For FY 2012, the committee will be referred to as the Joint Committee on Nuclear Risk Management per the ASME's C&S Connect Website on September 7, 2012.

1.1.5 B31 Pressure Piping Codes

The B31 Code for Pressure Piping Standards Committee is responsible for the series of B31 Codes that address non-nuclear piping system requirements for a variety of industrial applications, including the balance of plant applications in nuclear facilities. Table 2 indicates the B31 identifier and the type of piping addressed by that specific B31 Code.

Table 2. B31 Pressure Piping Codes.

Identifier	Code Title
B31.1	Power Piping
B31.3	Process Piping
B31.4	Pipeline Transportation Systems for Liquid Hydrocarbons and Other Liquids
B31.5	Refrigeration Piping and Heat Transfer Components
B31.8	Gas Transmission and Distribution Piping Systems
B31.9	Building Services Piping
B31.12	Hydrogen Piping and Pipelines

These B31 Codes have similar requirements that have been adapted to the specific industrial application. However, focusing on those major codes and standards that will most likely be utilized for an HTGR, the piping code used for power plants is B31.1[6]. The most recent B31.1 Code issued in 2012 prescribes minimum requirements for the design, materials, fabrication, erection, test, inspection, operation, and maintenance of piping systems typically found in electric power generating stations, industrial institutional plants, geothermal heating systems, and central and district heating and cooling systems. The B31.1 Code also covers boiler external piping for power boilers and high temperature, high pressure water boilers in which steam or vapor is generated at a gage pressure of more than 15 psig (100 kPa), and high temperature water is generated at gage pressures exceeding 160 psig (1103 kPa) and/or temperatures exceeding 250°F (120°C). Many B31.1 materials have allowable stress values reflecting temperatures up to 1500°F (815°C).

1.2 ASME Organizational Structure

In order to better understand how ASME codes and standards are developed and who develops them, a brief look at the ASME organizational structure is in order.

Supported by ASME staff, the ASME codes and standards are developed and approved by volunteers who follow a specified consensus process proven to be the best path to use for codes and standards development. These volunteers come from a variety of industries, universities, government, and regulatory agencies and have expertise in the various areas to which they are assigned. Volunteers may be on one or more committees. The lower committees (task groups, working groups, or subgroups) work on their respective scope of rules and requirements and, after developing or modifying the requirements and approving them, pass those requirements up to the next higher committee for their respective consideration and approval. The committees identified in Section 1.1 for each of the respective codes or standards discussed are the highest level committee having final determination and responsibility for the content of each code or standard.

ASME's organizational structure focusing on codes and standards begins at the highest level with the Council on Standards and Certification (formerly the Standards and Certification Board of Directors). The Council on Standards and Certification, under the direction of the ASME Board of Governors, will supervise the activities of the Society relating to standards and certification. Nine boards report to the

Council on Standards and Certification. In terms of focusing on those major codes and standards that will be utilized for an HTGR, the Board on Pressure Technology Codes and Standards (BPTCS) and the Board on Nuclear Codes and Standards (BNCS) are clearly the two of most interest and will be discussed in further detail in the following subsections. One can gain a better insight as to which codes and standards each Board is associated with by studying the associated committee listings in Appendix A. The highest level committees report to these Boards.

1.2.1 Committees Reporting to BPTCS

As seen in Appendix A for the BPTCS, the most significant committees of interest to an HTGR construction effort (and hence the associated major codes and standards) include the B31 Code for Pressure Piping Standards Committee, the BPV Committee on Materials (II), the BPV Committee on Welding and Brazing (IX), the BPV Committee of Nondestructive Examination (V), and the BPV Committee on Pressure Vessels (VIII). Hence, the BPTCS has administrative control over the non-nuclear codes and the supporting BPV Code sections (also known as the service books).

1.2.2 Committees Reporting to BNCS

Again, looking at Appendix A for the BNCS, it is evident that the remaining significant committees of interest to an HTGR construction effort (and hence the associated major codes and standards) are the nuclear committees, which include the BPV Committee on Construction of Nuclear Facility Components (III), the BPV Committee on Nuclear Inservice Inspection (XI), the Joint Committee on Nuclear Risk Management, the Standards Committee on Operation and Maintenance of Nuclear Power Plants, and the Standards Committee on Qualification of Mechanical Equipment Used in Nuclear Facilities. Further inspection of the Appendix A listing for BNCS also shows the Standards Committee on Nuclear Quality Assurance, the Standards Committee on Nuclear Air and Gas Treatment, and the Standards Committee on Cranes for Nuclear Facilities as potential committees of interest. These later three committees and their associated standards will obviously be applicable to an HTGR but are believed to be viable in their current state (with continuing minor revisions) and do not need similar major modification or development efforts that the other codes and standards (e.g., see Section 3.6) need in order to support the construction of an HTGR.

1.3 ASME Responsiveness to Industry Needs

ASME codes and standards are continually being revised in order to be responsive to industry needs. If the ASME codes and standards were not responsive, they simply would no longer be used. However, the use of ASME codes and standards continues to increase with more users and more applications and the number of effective ASME codes and standards developed continues to increase to support new applications.

For LWRs (including the advanced light water reactors such as the Advanced Passive 1000 reactor design by Westinghouse, Advanced Boiling Water Reactor design by General Electric, Economic Simplified Boiling Water Reactor design by General Electric – Hitachi Nuclear Energy, Evolutionary Power Reactor design by AREVA Nuclear Power, etc.), the ASME BPV Code has been updating requirements to reflect not only new design approaches and design methods, but also new materials, new environmental fatigue evaluation methods, and new requirements that reflect the latest licensing approaches instituted by the U.S. Nuclear Regulatory Commission (NRC) specified in Title 10 of the Code of Federal Regulations (CFR), Part 52 (10 CFR Part 52).[7]

For advanced reactors such as LMRs and HTGRs, the publication of new code requirements has just begun. The results of past efforts and FY 2012 accomplishments will soon be evident as explained in Sections 2 and 3 of this report with an emphasis on efforts supporting HTGRs.

1.4 Industry Support of ASME

The previous Section 1.2 highlights the critical dependency that ASME has on the volunteers that actually generate, revise, maintain, and approve the codes and standards. How do these volunteers gather together to accomplish this feat? It is through the support of the international nuclear and non-nuclear industries (both public and private businesses, universities, government agencies, and regulatory agencies) that provide funding so that these volunteers can attend the ASME meetings and complete their assignments. Included in this knowledgeable volunteer support group are DOE, NRC, the national laboratories, and programs such as the VHTR TDO effort at the Idaho National Laboratory (INL). This is truly a unique working relationship. Without the volunteers, industry, and ASME combining together in their efforts, today's codes and standards simply would not exist.

2. REVIEW OF RECENT ASME ACTIVITIES SUPPORTING HTGRS

Before addressing the most recent ASME accomplishments over FY 2012, it is necessary to review recent ASME activities that have occurred over the last few years in order to provide appropriate background information. These past accomplishments include planning documents, earlier starts on ASME codes and standards specifically addressing HTGRs or advanced reactors in general, ASME committee reorganizations, and ongoing improvements to the construction rules for nuclear and non-nuclear components. Further details are provided in the subsections below.

2.1 HTGR Roadmap

During the August 2008 ASME BPV Code Week, a planning meeting was held to discuss how ASME could most effectively begin to develop ASME Code rules to support the development of HTGRs. It was agreed that a document was needed that could identify the current problems and needs and make recommendations to ASME on how to proceed with this effort. Over the course of the next 12 months with multiple meetings held during ASME BPV Code Week and outside of Code Week, a guidance document was developed. Funding was provided by the NRC and logistical support was provided by ASME Standards Technology (ST), LLC. This final document was entitled "Roadmap for the Development of ASME Code Rules for High Temperature Gas Reactors". As discussed in the 'Overview and Executive Summary' of the HTGR Roadmap:

> *The Roadmap has been developed as a guide to the R&D and Code development tasks that should be considered in developing rules for High Temperature Gas Cooled Reactors (HTGR). The primary focus of the Roadmap is on the development of a complete set of rules for the design and operating conditions that are being proposed for the Next Generation Nuclear Plant (NGNP) demonstration unit. This is considered to be a Phase I, Part B or intermediate-term activity for the purpose of the Roadmap. However, the Roadmap also mentions a Phase I, Part A activity to modify the existing elevated temperature design rules and to provide a Code Case for graphite and ceramic composite core support structures for interim use for HTGRs. There are some additional tasks that have been identified as a part of the Phase I, Part A effort as noted in the Roadmap. In addition, a Phase II activity is envisioned to develop rules for future generations of HTGRs that are expected to operate at higher temperatures, and for other advanced reactors, such as Liquid Metal (e.g. sodium) Cooled (LMR) designs.*

Revision 8 of this draft document was made available electronically to ASME committee members on ASME's Website on an interim basis until it could be officially published. Revision 8 of the HTGR Roadmap received further review by the NRC (who funded the development effort) and proposed various minor revisions. These NRC review comments were incorporated into the document and the HTGR Roadmap was electronically issued on the ASME Website as ASME ST, LLC report STP-NU-045. BNCS is monitoring the progress Section III has made with regards to construction rules for HTGRs in light of the HTGR Roadmap recommendations.

In light of ASME's approval of Division 5 and other existing HTGR Roadmap assumptions regarding code development goals and aggressive R&D efforts that would not likely be supported, ASME ST, LLC decided it would be appropriate to update the HTGR Roadmap. The goal of this update was to focus more on the probable near-term achievements (within 5 years) so that a more usable Division 5 could be published by 2015 (the next scheduled ASME BPV Code Edition after the upcoming 2013 Edition). To achieve this, a revised HTGR Roadmap needed to better reflect current industry needs, establish more realistic code development goals, discuss improved analysis methodologies to reduce current evaluation conservatisms, and establish credible supporting R&D efforts. The Chair of the Working Group on HTGR (an NGNP Project funded individual) led the revision effort starting in early FY 2011. A draft revision 1 to the HTGR Roadmap was developed. Since funding for the effort was again supplied by the NRC, the revised HTGR Roadmap was submitted to the NRC for review. However, review comments were not received by the end of FY 2011.

2.2 Probabilistic Risk Assessment Standard

In response to a 1987 request by the NRC, ASME formed a project team to develop a standard for PRAs. Subsequently, the Standards Committee on Nuclear Risk Management was formed. Recognizing that developing a standard that covered all modes and aspects of a PRA was too ambitious to achieve in a single step, the committee focused their initial efforts on a PRA scope that covered internal events (plant hardware failures, internal floods, and human errors), Level 1 (required to estimate core damage frequency), and a limited scope of Level 2 (sufficient to estimate the large early release frequency) for currently operating and licensed LWRs. The American Nuclear Society (ANS) then started to develop PRA standards to cover a wider scope including external events, internal fires, and events initiated during non-full power operating modes. ASME published their first standard in 2002 and in the following year, ANS published their first standard.

Under ASME, two working groups were set up in 2006 to address advanced reactors, one for advanced LWRs and another for advanced non-LWRs. The advanced non-LWR working group decided to develop a new stand-alone standard to address non-LWRs such as HTGRs, LMRs, and any reactor that may have issues of applicability with the existing LWR standard. A draft PRA standard for advanced non-LWRs was developed by the Standards Committee on Nuclear Risk Management for public review and comment in October 2008. About 600 comments were received from the U.S. and international PRA community, although many of the comments repeated a much smaller set of issues. Committee members spent many hours developing responses to those comments and revising the draft. The NGNP Project supported committee members in the resolution of comments on the draft PRA standard for advanced non-LWRs.

Furthermore, ASME and ANS decided to integrate the respective LWR PRA standards into a combined ASME/ANS PRA Standard that was first published in 2009 (a full revision addendum to RA-S-2008)[8] and this is the current referenced standard approved by ASME and ANS and endorsed by the NRC in Revision 2 of Regulatory Guide 1.200,[9] dated March 2009.

After addressing the numerous comments received, the revised draft PRA standard for advanced non-LWRs continued to proceed through the multi-tiered ASME balloting and revision process in 2010 and most of 2011. In September 2011, the draft advanced non-LWR PRA standard was balloted at the Subcommittee on Standards Development with the recommendation that it move forward as a “trial use” standard to the Joint Committee on Nuclear Risk Management for approval. An NGNP Project funded individual led the development effort for this non-LWR PRA standard and the effort to guide it through the ASME approval process.

2.3 Operation and Maintenance Standard

The first five series of Operation and Maintenance (OM) Codes published in the early 1980s were originally written to support older existing nuclear power plants that did not always have a capability to perform certain testing procedures. Therefore, the phrase “if not practical” was included in places just for those older plants that could not comply with the requirement being discussed. In 1987, the first five series were combined into a single OM Code.

More recently, the Standards Committee on Operation and Maintenance of Nuclear Power Plants worked hard to combine two separate publications (the OM Code and the OM Standards and Guides) into one document. This was done to ensure that the standards and guides documents were readily available to the users of the OM Code. The latest OM Code, reflecting this consolidation effort, referred to as the 2009 Edition, was published in 2010.

The Standards Committee on Operation and Maintenance of Nuclear Power Plants is continuing to develop a revised OM Code that will provide improved testing and maintenance requirements for all new advanced LWR plants. Since these new plants have not yet been built, the phrase “if not practical” was deleted. Due to the generic statement of the technical requirements, the committee believes that these changes can also work very well for all types of new reactors, including HTGRs. The focus of their recent efforts has been on developing requirements for new issues related to components and to add new requirements for equipment not previously addressed such as squib valves (pyro-technic-actuated valves used for rapid exit of a fluid from a pressurized source). The 2011 Addenda to the 2009 Edition of the OM standard has recently been issued. This Addenda reflects much of the recent development efforts and includes a new Nonmandatory Appendix M and a new Subsection ISTF.

Nonmandatory Appendix M provides guidance for the design of new power plants to better support preservice and inservice testing. The committee recognized that it was essential for the OM Code to keep up with evolving new reactor and advanced reactor technologies. The new Subsection ISTF addresses improved inservice testing methods on pumps for plants receiving either a construction permit or a combined license for construction and operation by the applicable regulatory authority on or following January 1, 2000.

2.4 Qualification of Mechanical Equipment Standard

The first QME-1 standard, “Qualification of Active Mechanical Equipment Used in Nuclear Power Plants,”[10] was published in 1994 and has been regularly updated since then. Most recently, the Committee on Qualification of Mechanical Equipment Used in Nuclear Facilities approved the latest edition of this standard, published in 2007.

There are no specific ongoing activities for HTGRs being pursued in the Committee on Qualification of Mechanical Equipment Used in Nuclear Facilities. This committee believes that they have already written their requirements in such a fashion that their rules are applicable to any reactor design. However, efforts have been ongoing to update this standard by correcting errata, incorporating applicable Code Cases, and addressing new equipment such as squib valves and visco-elastic dampers.

2.5 Section XI – Division 2 Code

In 2004, the Special Working Group High Temperature Gas Cooled Reactors (SWG HTGR) began efforts to revitalize Division 2 of Section XI, which addresses inservice requirements for gas-cooled reactors. The SWG HTGR completely rewrote Section XI, Division 2 from that last published in the 1992 Edition (including 1993 Addenda) of Section XI of the BPV Code. This new Division 2 is based on technology of the new HTGRs reflecting passive system performance but currently addresses metallic components only. It incorporates a risk-informed technology that drives the required periodic examinations. This new methodology is based on the latest draft ASME PRA Standard for advanced non-LWR nuclear power plants. The traditional Inservice Inspection Program has been replaced with the Reliability and Integrity Management (RIM) Program.

The SWG HTGR continued development of the new Division 2 proposal in 2010, including adding requirements for a performance based nondestructive qualification appendix. The first complete draft of the new Section XI, Division 2 was finished in early FY 2011 and received vigorous reviews by the SWG HTGR, NGNP Project personnel, and the NRC and its contractors. Incorporation of those review comments continued through the first half of 2011 and the first complete draft was finalized at the August 2011 BPV Code Week meeting. The proposal then began its formal ASME approval process with the

Section XI lower committees. In 2011, an NGNP Project individual gained membership onto SWG HTGR in order to further support the development and approval of this new Division 2.

2.6 Section III Activities

Many separate and distinct efforts were also accomplished to specifically support the development of HTGR construction rules in Section III. These efforts are discussed in more detail below. Due to the fact that Section III contains rules needed for preliminary and final design efforts, Section III requirements are leading the way to establish many of the approaches regarding construction rules for HTGRs.

2.6.1 ASME ST, LLC Tasks 1 Through 14

In February 2005, ASME met with the DOE to present a proposal for technology development projects addressing research gaps for emerging energy needs. These gaps had been identified from visits and interfaces by ASME delegations with key energy stakeholders in the United States, Canada, Europe, Japan, and South Africa, and included the next generation of commercial nuclear power plants. In support of these advanced (non-LWR) reactors, inclusion of new materials for very high temperatures (graphite, composites and new metallic materials) was discussed.

In June 2005, DOE responded with a draft work scope outlining support of ASME Codes and Standards activities related to the Advance Reactor Program. On June 30, 2005, members of ASME ST, LLC staff met with DOE Headquarters staff to discuss the Generation IV Reactor Materials Project, which included ASME's initial response to the draft work scope. Several exchanges of comments and responses followed. On August 1, 2005, the final proposal was submitted to DOE-Idaho Operations for their evaluation. DOE and ASME ST, LLC signed a cooperative agreement on September 30, 2005. This agreement provided DOE funding to ASME ST, LLC to develop technical basis documents necessary to update and expand ASME codes and standards for application to advanced reactor systems that operate at elevated temperatures. A steering committee was established to provide overall guidance to the process and the NGNP Project placed two consultants on this committee in December 2007 to assure that the near-term needs of the NGNP Project were considered.

The initial results of that cooperative agreement were eleven completed tasks that focused on issues pertinent to HTGRs. Many of these tasks addressed NRC concerns with the current construction rules in Subsection NH and ASME Code Cases. As a result of these completed tasks, improvements to ASME Code rules were implemented. A twelfth task, funded by the NRC but also managed by ASME ST, LLC, focused on NDE development for high temperature service, another issue pertinent to HTGRs. Table 3 lists all twelve completed initial agreement tasks and the subject matter of those efforts.

In 2010, with additional DOE funding, two more tasks were identified that would provide needed support for HTGR development. These two tasks (Tasks 13 and 14) were agreed upon and pursued specifically in order to support the NGNP Project.

Per the ASME ST, LLC Request for Proposal (RFP),[11] Task 13 had three parts:

> *Part 1:*
>
> *Develop a data package for consideration by applicable code committees supporting the use of Alloy 800H for extended time and temperature compatible with NGNP design objectives, primarily associated with in-core structures and heat transport systems exposed to hot gas temperatures. The established objective is 850°C and 500,000 hr. (Note: higher temperatures are desirable as permitted based on available data.)*

Table 3. Generation IV Reactor Materials Project – Completed Initial Agreement Tasks.

Task	Issues	Results
1	• Longer life at higher temperatures than currently in Subsection NH needed for HTGRs • Discrepancies in Alloy 800H and Grade 91 allowable stresses	• 800H data support 600,000 hr to 800ºC – more testing required above 800ºC and above 750ºC for welds • Grade 91 data support 600,000 hr to 650ºC – base stress rupture fractures for welds on current worldwide effort
2	• Identify Subsection NH concerns to be resolved for licensing	• Creep crack growth in weldments and notches, inelastic analysis, and environmental effects are primary issues
3	• Current Subsection NH negligible creep and creep-fatigue damage criteria are overly restrictive	• Modification to current elastic analysis creep-fatigue rules formulated and incorporated negligible creep criteria proposal under consideration
4	• Update core support structure Code Case N-201 for HTGRs	• Survey identified materials and operating conditions of interest • N-201 errors corrected and updated to reflect current provisions of Subsections NG and NH
5	• Assess creep-fatigue data and damage evaluation procedures for Grade 91 and Hastelloy XR	• Subsection NH procedures overly conservative compared to data and other international standards • Potential improvements to Subsection NH reference methodology identified and evaluated • Hastelloy XR data support Subsection NH rules
6	• Minimum creep-rupture stress in Subsection NH inconsistent with Section II, Part D - impacts wall sizing and allowable cyclic life	• Inconsistencies have been quantified for Subsection NH materials - all but Alloy 800H have problems • Assembly and assessment of data bases underway
7	• Intermediate heat exchanger, particularly compact micro channel concepts with unique design features, may require new or modified rules for materials, design, examination, etc.	• Survey completed on current experience base and materials and operating conditions of interest
8	• Creep and creep-fatigue crack growth is top NRC concern • Subsection NH has design factors and procedures but not quantitative assessment of crack initiation and growth	• Available methodologies reviewed and British R5 procedure selected for development of design rules
9	• Current Subsection NH rules based on 1970s and 1980s technology – simplified elastic and inelastic design methods need updating to take advantage of advances in computing technology and understanding of material behavior	• Key elements of "ideal" code developed • Detailed review and comparison of Subsection NH and other national and international standards completed
10	• Creep-fatigue is identified NRC concern and Tasks 3 & 5 identified deficiencies in current methodologies	• Newly proposed (and/or new to Subsection NH) methodologies have been selected for detailed review, which is in progress
11	• Additional material options in Subsection NH advantageous to address unique HTGR requirements - a preliminary assessment is needed to focus future effort	• Survey identified 617, 800H, Grade 91, and 316H of current interest • Additional materials to be added for comparison include 230, 533/508, Grade 22 Class 1, Grade 22V, 316L(N) and 20Cr-25Ni-Mo-Cb-N
12	• NDE development needs for unique HTGR requirements	• Identified potential degradation mechanisms and susceptibility criteria • Introduced nondestructive monitoring • Proposed supporting R&D activities

Part 2:

Develop a data package to support a code case or modification to Code Case N-201 and Subsection NH that supports the use of 800H for infrequent, off-normal temperature excursions compatible with NGNP design objectives. The number of events can tentatively be based on what is specified in Code Case N-499. Allowable stresses should be provided for times and temperatures bounded by times for 100 hours to 10,000 hours and associated temperature limits as high as possible consistent with current data.

Part 3:

Submit draft code rules with data packages developed in Parts 1 and 2 above to the BPV Committee on Materials (II) and the BPV Committee on Construction of Nuclear Facility Components (III) and applicable subgroups for additions to and revisions of material properties information contained in Section II and Section III, Subsection NH. These submittals shall follow the guidance provided in Appendix 4 and 5 of Section II, Part D and are a formal proposed standards action with complete supporting information.

Task 14 was actively pursued due to the potential for creep-rupture failures earlier than projected using BPV Code data and the desire to address this safety issue in order to eliminate any adverse impacts on potential near-term design activities. Per the ASME ST, LLC RFP[11], Task 14 also had three parts:

Part 1:

Assess available data on 304 and 316 stainless steel (note: 316 stainless steel is the primary material of interest in this assessment) to determine if there are restrictions that should be placed on specifications and procurement packages or additional acceptance test and examination requirements, e.g., chemical, physical or mechanical properties or processing variables, that would exclude from use those heats of material that are not representative of the data base from which the currently published allowable stress values were derived. (Note: the effectiveness of existing restrictions and requirements shall also be included in the assessment.)

Part 2:

In the event that the results of the Part 1 assessment do not identify applicable restrictions and/or additional acceptance requirements, the time and temperature limits beyond which the validity of the current allowable stress values cannot be guaranteed shall be determined. Allowable stresses beyond their range of validity should be recommended for deletion.

Part 3:

Based on the results from Parts 1 and 2 above, prepare a submittal of draft code rules with supporting information. Recommendations should include specific Code words or modification to tables implementing the restrictions that are ready for consideration as a code case or BPV Code revision. This submittal shall be a formal proposed standards action and shall have an assigned tracking number.

Work on Tasks 13 and 14 began in August of 2010. In the August 2011 BPV Code Week meeting, final recommendations for both tasks were presented and discussed at various committee meetings. Tasks 13 and 14 were completed with the submittal of draft reports and Code revision proposals to ASME ST, LLC. However, committee deliberations were still not complete.

For Task 13, determining the material property data values for extending both the time of use and the temperature limits for Alloy 800H was completed and Code revision proposals were generated. For the time-independent properties of yield strength and tensile strength as well as the time-dependent minimum stress-to-rupture and the allowable stress intensity values of S_t and S_{mt} , extended limits were established at 1650ºF (900ºC) and 500,000 hours. Progress on obtaining ASME approval on certain aspects of this task was achieved by the end of FY 2011. The time-independent Alloy 800H data was approved by the BPV Committee on Materials (II) and the BPV Committee on Construction of Nuclear Facility Components (III). The minimum stress-to-rupture data was approved by the BPV Committee on Materials (II). The remaining time-dependent Alloy 800H data (S_t and S_{mt}) were just beginning the ASME approval process. These Alloy 800H approval efforts were managed by NGNP Project personnel.

For Task 14, the root cause of the safety concern was determined. While performing material specimen testing at temperatures above 1200ºF (650ºC), researchers at the Japanese National Institute for Materials Science experienced a premature stress rupture failure while testing certain heats of 316 stainless steel material that met ASME BPV Code specifications Additional studies for this task revealed that the material heat that failed had high amounts of aluminum (> 0.04 wt %). Therefore, the need to restrict certain elements such as aluminum were identified in the proposed Code revision to either modify an existing chemical requirements table or to include a new chemical requirements table in Section III, Division 1, Subsection NH implementing the appropriate limitations. No ASME approvals for this task were achieved before the end of FY 2011.

ASME ST, LLC made a formal proposal to DOE, requesting funding to complete seven more tasks (Tasks 15 through 21) under the Generation IV Reactor Materials Project. These tasks were identified as supporting development of the ASME BPV Code and furthering the advancement of HTGR design capabilities. A meeting was held at the INL in July 2011 where the tasks were discussed in detail and then a prioritization of those proposed tasks was established. A final decision was made not to provide funding for these additional tasks.

2.6.2 Establishment of New ASME BPV Code, Section III Committees and Reorganization of ASME BPV Code, Section III Committee Structure

Prior to the August 2008 meeting, there were no official ASME BPV Code, Section III committees to develop construction rules for HTGRs, excluding the Subgroup on Graphite Core Components discussed below. Only minimal progress on ASME construction rules can be accomplished without a structured committee of experts in place. Hence, Section III leadership began the formation of the necessary working groups and subgroups needed to develop these rules. By September 2009, two working groups [Working Group on Nuclear High Temperature Gas-Cooled Reactors (WG HTGR) and the Working Group on Liquid Metal Reactors (WG LMR)] had been created and populated with volunteers. In addition, a Subgroup on High Temperature Reactors (SG HTR), to whom both WG HTGR and WG LMR report, had also been created and populated with volunteers. This creation of an ASME BPV Code, Section III committee infrastructure for advanced non-LWR power plants occurred very rapidly but reflects the desire of the volunteers and their sponsors to move effectively forward to develop construction rules for these reactors.

These committees were initially assigned the responsibility to develop Division 5 under Section III. The committees were organized so that Division 5, Part 1 addressing HTGRs would be developed by the WG HTGR. Part 2 of Division 5 would contain rules for liquid metal reactors that would be developed by the WG LMR.

In 2010, the BPV Committee on Construction of Nuclear Facility Components (III) began a reorganization effort of the Section III committee structure, which continued into the next year. Perhaps the biggest driver behind the reorganization was to assure that all proposed rule changes for all of the

Divisions 1 through 5 receive a proper and thorough review and approval by committees comprising the three major technical areas:

- Design
- Materials, fabrication, and examination
- General requirements.

In order to assure that this goal is achieved and maintained, it was clarified that all technical committees have representation under at least one of the three technical areas (if not more) and that three subcommittees (higher level committees than subgroups) were to be formed that would be responsible for each of these technical areas.

During 2011, the BPV Committee on Construction of Nuclear Facility Components (III) completed its reorganization. As can be seen in Appendix A, the ASME BPV Code, Section III committee structure now has a Subcommittee on Design, a Subcommittee on General Requirements, and a Subcommittee on Materials, Fabrication, and Examination. Other organization changes included efforts to support the development of rules for high density polyethylene pipe as well as expanding international participation in China and Korea with international working groups organized to meet in their respective countries.

2.6.3 Graphite Core Component Rules

The genesis of the Section III Subgroup on Graphite Core Components (initially referred to as the Graphite Project Team on Core Supports) resulted from a workshop at Oak Ridge National Laboratory (ORNL) on October 15 and 16, 2003. The NRC contracted ORNL to hold the ASME Graphite Code Workshop to seek out all potential stakeholders to gauge their level of interest and recruit members for the code committee. The meeting was attended by representatives from graphite manufactures, gas reactor vendors, and DOE national laboratories to build a consensus on how to restart the previous ASME effort[d] of developing a nuclear graphite construction code. The NRC staff also identified a lack of ASTM International material specifications for graphite. As a result of the workshop and discussions between ORNL and ASME, the first meeting of the new graphite committee was held in February 2004. Since that initial meeting, committee members have been actively developing both general requirements and construction rules for graphite core components and the core assembly.

In October 2009, the Subgroup on Graphite Core Components (SG GCC) began the process of balloting the resulting proposed graphite rules to the BPV Committee on Construction of Nuclear Facility Components (III). Approval for both the graphite general requirements and the design and construction rules[e] for graphite by the BPV Committee on Construction of Nuclear Facility Components (III) and the BNCS was achieved in May 2010. Since that time, the SG GCC has continued to improve the graphite rules by submitting additional editorial and technical corrections for approval. NGNP Project personnel have been actively involved in the writing, correction, and approval of these new graphite rules.

At the start of FY 2011, the SG GCC continued to deliberate various editorial and technical revisions to the graphite rules approved in 2010. Regarding the graphite general requirements, the Subgroup GCC revised the rules such that the Designer would also be required to become a Certificate Holder. Rules for the machining, installation, examination, and testing of graphite were revised and began the process of

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d. In 1990, the Section III, Division 2, Subsection CE "Design Requirements for Graphite Core Supports," was submitted to the Section III Subcommittee [now the BPV Committee on Construction of Nuclear Facility Components (III)] for review and comment but little progress was made after that initial submittal.

e. The Subgroup on Graphite Core Components decided to define '*construction*' differently than Division 1 as "All operations required to build the Graphite Core Assembly (manufacture materials, machine Graphite Core Components and install) in accordance with Design Drawings and the Construction Specification." Hence, design and construction are two distinct terms for graphite.

ASME balloting. The SG GCC began consideration of composites (carbon/carbon composites and ceramic matrix composites), discussing how to approach the development of rules for such unique materials. The SG GCC also decided to improve the rules associated with the two-parameter Weibull approach used in the design of graphite core components. One of the issues raised during the approval of the initial graphite rules at the BPV Committee on Construction of Nuclear Facility Components (III) level was the control of contamination during machining. The SG GCC discussed and developed a new Mandatory Appendix on contamination control. Finally, the SG GCC began interacting with the Special Working Group on Non-Metallic Materials [under the BPV Committee on Materials (II)] with the eventual goal of having graphite material properties published in Section II as qualified data becomes available. This is recognized as a long-term goal but the objective is to capture graphite data as it becomes available for the many different graphite grades from the different graphite manufacturers.

2.6.4 Section III Revisions

It must be recognized that committee members are continually involved in efforts to improve the existing Section III rules. These numerous updates and revisions improve the ability of nuclear facilities to comply with the requirements established in the ASME BPV Code. Significant examples of recent changes include:

- Making revisions to permit the 10 CFR Part 52 licensing process to be used
- Updating the listing of national standards for use
- Approving the use of the 2008 Edition of ASME NQA-1 with the 2009 Addenda.

In terms of more directly supporting HTGRs, efforts were made to update Subsection NH in Division 1 of Section III. Although most of these revisions were more of an editorial nature, such revisions improve the user's comprehension of requirements and eliminate typographical errors.

Revising the ASME BPV Code is a continual ongoing process. ASME continues to be responsive to industry needs. In light of the close interaction between Division 1 and Division 5 rules (see Subsection 2.6.5), changes to specific Division 1 rules were also considered in light of their potential affect on high temperature reactors.

2.6.5 Development of Division 5 Rules

The original HTGR Roadmap recommended three phases to develop ASME rules for HTGRs.

- Phase I, Part A was to update existing code rules as necessary for immediate application, such as the HTGR for the NGNP Project.
- Phase I, Part B was to build on that initial effort but develop new rules specifically for HTGRs (to be designed within the next ten years or so) and issue them in a new Division 5 (under Section III) but still reflecting current design conditions for the HTGR being developed by the NGNP Project[f].
- Phase II was to upgrade the new rules in Division 5 to satisfy the needs of stakeholders that would be designing HTGRs designed ten years or more in the future with significantly higher component temperatures than those considered in Phase I, Part B.

Efforts to begin defining the structure of Division 5 were officially initiated at the November 2009 ASME BPV Code Week. However, there was initial opposition in the lower committees to the proposed Division 5 structure based on not following the specific HTGR Roadmap recommendations identified above. However, after initial BPV Committee on Construction of Nuclear Facility Components (III)

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f. Design conditions as defined herein only reflect a reactor outlet temperature of 750-800ºC and a design life of 60 years. These provide minimal guidance for the temperature and time extension needed for Alloy 800H material data.

balloting of the graphite general requirements in December 2009, it was recognized that ASME policy requires a Code or Standard to exist in order to issue general requirements. General requirements cannot be issued using a code case approach. Therefore, at the February 2010 BPV Code Week meeting, the lower committees agreed that it was appropriate to move forward with a Division 5. This meant that the HTGR Roadmap recommendations were still being followed, just that Phase I, Part A and Phase I, Part B were essentially being combined.

The current Division 5 structure was considered to be for short-term use only until the ASME committees have sufficient time to develop new construction rules. Table 4 illustrates the Division 5 structure approved by the BPV Committee on Construction of Nuclear Facility Components (III) and the origin of the rules that were incorporated. This Division 5 structure was intended to support NGNP Project needs. Issues such as higher temperature limits and longer hours of operational use can be more easily incorporated (via code cases or code revisions) with Division 5 already published.

When the initial draft of Division 5 was developed, the established goal was to develop rules just for HTGRs under Part 1. The WG LMR agreed with this approach since they wanted additional time to establish their committee goals and plan their respective Part 2 rules. However, at the May 2010 BPV Code Week meeting, the Chair of the BPV Committee on Construction of Nuclear Facility Components (III) required the SG HTR to consider issuing Division 5 with common rules for both HTGRs and LMRs. Another Division 5 draft was generated reflecting common rules.

Balloting of Division 5 to the lower ASME committees began in July 2010. A total of 21 committees were asked to review the proposed Division 5. Many review comments were received and each comment required a response prior to the closure of the ballot. The ballot for approval to the committees representing the three major technical areas of general requirements, design, and materials, fabrication, and examination was completed on September 22, 2010. No disapprovals were received. The next scheduled step was to submit Division 5 to the BPV Committee on Construction of Nuclear Facility Components (III) by early FY 2011.

The entire Division 5 was balloted under one proposal and was approved by the BPV Committee on Construction of Nuclear Facility Components (III) on the first ballot, with concurrence by BNCS and other higher level ASME committees before the end of January 2011. However, numerous issues had to be addressed in order for ASME to physically publish Division 5. Formatting discussions were held with ASME Staff to explain the approved record and where changes had to be made to adjust to new publication software and formatting requirements. Initial draft proofs then had to be generated and reviewed by members of the Subgroup on Editing and Review, chaired by an NGNP Project individual. After that initial effort and following further detailed reviews by the ASME Editors, the Subgroup on Editing and Review and the SG GCC (with NGNP Project personnel participating) again reviewed the first official proofs of Division 5 to ensure that the Section III requirements approved by the BPV Committee on Construction of Nuclear Facility Components (III) and BNCS were indeed being published correctly. The task of developing Division 5 and guiding it through the ASME approval process was assigned to an NGNP Project individual.

2.7 Section VIII, Supporting BPV Code Sections, and B31 Pressure Piping Codes

Much like the Section III revisions, ongoing revision and updating of Sections II, V, VIII, and IX and the B31 Pressure Piping Codes continued. New materials or updates to material properties were continually being approved for Section II, improvements to NDE procedures were being made in Section V, enhancements to pressure vessel and piping design methodologies were being addressed in Section

Table 4. Division 5 Structure and Origin of Rules Being Incorporated.

Class	Subsection	Subpart	Subsection ID	Title	Code Basis	Scope
—	HA	—	—	**General Requirements**	—	—
A & B		A	HAA	*Metallic Materials*	NCA	Metallic
A		B	HAB	*Graphite Materials*	Part GA*	Graphite
A		C	HAC	*Composite Materials*	TBD	Composites
A	HB	—	—	**Metallic Pressure Boundary Components**	—	Metallic
		A	HBA	*Low Temperature Service*	NB	
		B	HBB	*Elevated Temperature Service*	NH & N-499**	
B	HC	—	—	**Metallic Pressure Boundary Components**	—	Metallic
		A	HCA	*Low Temperature Service*	NC	
		B	HCB	*Elevated Temperature Service*	N-253**, -254**, -257**, & -467**	
A & B	HF	—	—	**Metallic Supports**	—	Metallic
		A	HFA	*Low Temperature Service*	NF	
A	HG	—	—	**Metallic Core Support Structures**	—	Metallic
		A	HGA	*Low Temperature Service*	NG	
		B	HGB	*Elevated Temperature Service*	N-201**	
A	HH	—	—	**Non-Metallic Core Support Structures**	—	—
		A	HHA	*Graphite Materials*	Part GB*	Graphite
		B	HHB	*Composites Materials*	TBD	Composites
A & B	Appendices	None	Appendices	**Mandatory and Nonmandatory Appendices**	TBD	All

* - New graphite rules approved by BPV Committee on Construction of Nuclear Facility Components (III) and BNCS

** - Nuclear Code Case

TBD – to be developed

Notes:

1. Subsection ID has three characters for referencing flexibility to address low / elevated temperature service concerns or varying materials under the same subsection.
2. Two Section III Classes (A and B) were chosen for high temperature reactors, reflecting the classification philosophy described in ANSI/ANS-53.1: (1) safety-related; and (2) non-safety related with special treatment classification categories, respectively, for structures, systems, and components (SSCs). Non-safety related SSCs will use non-nuclear standards such as BPV Code Section VIII, B31, etc. as appropriate.
3. Appendices may be generally applicable (as shown above) or subsection specific and attached to those subsections (like Subsection NF).

VIII and the B31 Pressure Piping Codes, and improved welding procedure requirements were being updated in Section IX for the same reason as Section III was updated. ASME must support the industries for which they develop requirements in order to stay effective and continue their goal of assuring safety.

Preliminary efforts began to have the diffusion welding process be incorporated into the ASME BPV Code. NGNP Project personnel assisted that effort by providing data on the diffusion welding process using Alloy 800H material. The diffusion welding process can be used to fabricate compact heat exchangers, necessary for very high temperature HTGRs. But the first step was for the BPV Committee on Welding and Brazing (IX) to incorporate this advanced joining process into Section IX. This effort began in July 2010 and committee interactions continued through September 2010. With the help of a compact heat exchanger industry vendor and support provided by NGNP Project personnel with diffusion welding process data, efforts to establish diffusion welding in Section IX continued. The BPV Committee on Welding and Brazing (IX) approved the action and BNCS provided their final approval in early FY 2011 for diffusion welding to be incorporated into Section IX. The 2011 Addenda to Section IX contains these requirements for diffusion welding.

3. UPDATE ON ASME ACTIVITIES SUPPORTING HTGRS DURING FY 2012

The previous Section 2 established background information on past ASME activities in order to better understand what specific HTGR supporting activities ASME was able to achieve during FY 2012. The following subsections address these major current year achievements.

Many of these committees have taken organizational steps to address nonmetallic material issues, much like the BPV Committee on Construction of Nuclear Facility Components (III). The BPV Committee on Welding and Brazing (IX) has added a new Subgroup on Plastic Fusing. The BPV Committee on Nuclear Inservice Inspection (XI) has a new Working Group on Non-Metals Repair/Replacement Activities and a new Task Group on Repair by Carbon Fiber Composites. Finally, under the BPTCS, a new Committee on Nonmetallic Pressure Piping Systems has been formed with Subcommittees addressing glass fiber –reinforced thermosetting resin, thermoplastic, and other nonmetallic materials. These changes, as shown in Appendix A, reflect the responsiveness of ASME Codes and Standards to changing industry needs.

3.1 HTGR Roadmap

ASME ST, LLC decided it would be appropriate to update the original issue of the HTGR Roadmap to better reflect changing industry needs, establish more realistic code development goals, discuss improved analysis methodologies to reduce current evaluation conservatisms, and establish credible supporting R&D efforts. A revised draft HTGR Roadmap was developed and review comments from the NRC (who provided the funding) were received in 2012 and incorporated as appropriate. Revision 1 to the HTGR Roadmap document (STP-NU-045-1[12]) was published in June 2012 and can be obtained from the ASME Website at: http://stllc.asme.org/News_Announcements.cfm.

3.2 Probabilistic Risk Assessment Standard

The current committee structure is in transition. The Standards Committee on Nuclear Risk Management will remain for the interim in order to address inquiries and interpretations of existing standards and to fulfill other "historical needs". The Joint Committee on Nuclear Risk Management now officially exists and is expected to move forward with reorganization efforts.

The revised draft PRA standard for advanced non-LWRs was balloted at the Joint Committee on Nuclear Risk Management, starting in mid-March 2012. As expected, a number of comments were received and this first consideration ballot is expected to be disapproved (noted as "pending" at the time this report was written). Due to the significant volume of this proposed standard (nearly 600 pages), it is expected that certain revisions will be made to the draft proposal before resubmitting for a reconsideration ballot.

There are two other codes and standards that will require the application of this advanced non-LWR PRA standard that are of particular interest to HTGRs. The first is standard ANSI/ANS-53.1-2011[g], approved by ANS in December 2011, which describes a risk-informed, performance-based approach for designing and licensing modular HTGRs. It also establishes the component classification system used by ASME BPV Code, Section III, Division 5. DOE sponsored vendor and national laboratory personnel to work on the standard. The NGNP Project, as the first potential user, was directly engaged in facilitating the drafting of the language, supporting outside reviews, balloting, re-writing, and editing the document. The other code is the major rewrite of ASME Section XI, Division 2 (see Section 3.5), which sets forth requirements for inservice inspection of passive metallic components for modular HTGRs.

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g. ANSI/ANS-53.1-2011, "Nuclear Safety Design Process for Modular Helium-Cooled Reactor Plants", released August 2012.

3.3 Operation and Maintenance Standard

The Standards Committee on Operation and Maintenance of Nuclear Power Plants is continuing to develop a revised OM Code that will provide improved testing and maintenance requirements for all existing LWR reactor plants as well as new advanced reactors. Specifically, the Standards Committee on Operation and Maintenance of Nuclear Power Plants has proactively begun new efforts to look at not only the Generation III+ reactor designs but also Generation IV and Small Modular Reactor designs to clarify what requirements may need to be added or revised and what new components may be necessary to address. (See the new O&M Task Group on New Reactor OM Code committee in Appendix A). The Standards Committee on Operation and Maintenance of Nuclear Power Plants has extensive links to both regulators and the nuclear industry in order to be responsive to the needs of all, including international users.

The Standards Committee on Operation and Maintenance of Nuclear Power Plants is currently supporting efforts to issue a 2012 Edition of the OM Code. This new edition is expected to contain requirements for squib valves as well as differentiating rules for existing plants and new (after January 1, 2000) approved plant designs. In terms of future changes and in response to recent seismic events, the Standards Committee on Operation and Maintenance of Nuclear Power Plants is looking at appropriate inspection rules for snubbers that have experienced beyond design basis events. Responding to NRC concerns, the committee is also looking into the issue of regulatory treatment of non-safety systems (RTNSS). Nuclear plants designed with passive safety systems may still have active systems (i.e., systems requiring power to operate) that are designated as non-safety related yet can still provide defense-in-depth capabilities, especially for long-term, post-accident support. [Note that components in these types of systems would be classified as "Class B" components in the ASME BPV Code, Section III, Division 5 if they were in a high temperature reactor plant.]

3.4 Qualification of Mechanical Equipment Standard

There are no specific ongoing activities for HTGRs being pursued in the Committee on Qualification of Mechanical Equipment Used in Nuclear Facilities. This committee believes that they have already written their requirements in such a fashion that their rules are applicable to any reactor design. However, discussions were held at the March 2012 meeting regarding updating QME-1 for certain issues related to new construction plants and expanding the equipment addressed. Efforts have been ongoing to update this standard by correcting errata, incorporating applicable Code Cases, and addressing new equipment such as visco-elastic dampers and squib valves. The committee is supporting efforts to issue a new 2012 Edition of QME-1 (with a new title reflecting nuclear facility application rather than nuclear power plants). However, squib valve or visco-elastic damper rules are still being developed and are not expected to be in the 2012 Edition.

3.5 Section XI – Division 2 Code

At the start of FY 2012, the new Division 2 rewrite effort for HTGRs had generated a complete draft and had already begun the ASME approval process. However, after having discussions with a small modular reactor vendor, the Section XI Executive Committee recognized how the Section XI, Division 2 RIM approach could also support this reactor type and other new reactors. Therefore, the SWG on HTGRs was renamed the Special Working Group on Reliability and Integrity Management Program (SWG on RIM Program) and were directed to expand their draft proposal to apply to all new advanced reactors, including LWRs. This decision has significantly expanded the scope of the Division 2 rewrite. As currently envisioned, the main body of Division 2 will address issues generic to all reactor types with separate mandatory appendices focusing on LWR, HTGR, LMR, and other reactor type specifics. Section XI believes this effort will produce a more diversified set of rules that will better support new reactors, especially the small modular reactors that are of current interest to many.

3.6 Section III Activities

Due to the fact that Section III contains rules needed for preliminary and final design efforts, Section III requirements are leading the way to establish many of the approaches regarding construction rules for HTGRs.

3.6.1 ASME ST, LLC Tasks 13, 14, and 14a

On-going efforts for extending the temperature and time of use for Alloy 800H (ASME ST, LLC Task 13) continued throughout the year. The yield and tensile strength values extended to 1650ºF (900ºC) and the minimum stress-to-rupture properties extended to 1650ºF (900ºC) and 500,000 hours were approved by BNCS and are scheduled to be published in the 2013 Edition of the ASME BPV Code. An issue regarding what limit procedure should be used to establish the time-dependent S_t and S_{mt} values was raised and multiple discussion at multiple meetings were held. A path forward was finally determined at the August 2012 Code Week meeting as was an agreement on how to extend the modulus of elasticity values (another important material property needing to be extended) to the higher temperatures. Therefore, it is expected that these Task 13 actions will proceed to the BPV Committee on Materials (II) for final approval before the end of 2012 with submittal to the BPV Committee on Construction of Nuclear Facility Components (III) following.

ASME ST, LLC Task 14 addresses the premature rupture failure of certain stainless steel heats. The proposed solution was to place more restrictive controls on certain elements in a new table in Nonmandatory Appendix X of Subsection NH. However, concerns over how industry could realistically respond to the number of elements being controlled brought about more discussions. This Subsection NH revision effort is still in the discussion stage but a potential path forward may be to follow the example that ASTM, International has recently implemented for Grade 91 material. At this time, it is not certain when the final implementation of the Task 14 recommendations will officially move forward to the BPV Committee on Materials (II).

Due to the fact that sufficient Generation IV Reactor Materials Project funding remained after the completion of Tasks 13 and 14, ASME ST, LLC initiated Task 14a to achieve additional HTGR code support. Work on this task began in late 2011. Task 14a is making corrections to the stainless steel allowable stress intensity tables in Subsection NH. This effort was discussed at multiple code week meetings and is having difficulty implementing the procedures for establishing stress limits when considering the tertiary creep effects exhibited by certain stainless steels. Expectations are that an acceptable approach can be agreed upon and the proposal can be completed and moved forward to the BPV Committee on Materials (II) before the end of 2012. Once approved, the action will then be forwarded to the BPV Committee on Construction of Nuclear Facility Components (III) for approval.

3.6.2 Reorganization of ASME BPV Code, Section III Committee Structure

Although the BPV Committee on Construction of Nuclear Facility Components (III) recently completed a reorganization, additional considerations prompted a need for discussions regarding another reorganization effort. Although still being discussed, drivers such as the Section III Division committees having more direct control over the content of their subsections, the potential for a new subsection containing nonmetallic material properties, and other enhancements have prompted this new reorganization effort.

3.6.3 Graphite Core Component Rules

The ASME Subgroup on Graphite Core Components (SG GCC) continues to revise and enhance the graphite rules in Division 5. Graphite general requirements continue to be improved along with the rules addressing machining, installation, examination, and testing of graphite. The SG GCC has been editing sections of the code pertaining to contamination control and has finalized a Mandatory Appendix on contamination control at graphite fabrication facilities. These code revisions should be ready to submit to the BPV Committee on Construction of Nuclear Facility Components (III) for approval in 2013.

The SG GCC has adopted a statistical method that will be used to accept graphite billets using mechanical tensile test data. The number and orientation of the tensile test specimens are governed by the ASTM D7219 standard, which is endorsed by the Subgroup. The method uses a statistical method known as Likelihood Ratio Ring to compare the spread in the tensile test strengths to a specified allowable component stress for a given probability of failure. The method also visually compares billets tested from prior furnace charges to determine the repeatability of past production runs in comparison to current manufactured graphite billets.

The committee's debate on the use of the three-parameter Weibull distribution continues. The committee agrees the three-parameter Weibull distribution is not useable for all graphite data. The two-parameter Weibull distribution ease of use makes this distribution superior over the three-parameter. The full assessment construction/design rules rely on a threshold value of strength. The threshold is determined by performing a fit to the three-parameter Weibull distribution from tensile test data. The committee has formed a task group to review the need of the threshold strength in determining probability of failure under the full assessment rules and to ascertain if another method of determining the probability of failure (that does not require the threshold strength) provides comparable results.

In May 2012, the SG GCC was informed by the Chair of the BPV Committee on Construction of Nuclear Facility Components (III) that nonirradiated graphite properties would now reside in ASME BPV Code, Section III (potentially as a new nonmetallic material properties book) and not in ASME BPV Code, Section II. This action was necessary when the Board on Nuclear Codes and Standards ruled that nonmetallic material properties would not be included in Section II where metallic properties now reside. This will necessitate restructuring of a current committee working on high density polyethylene pipe into a committee responsible for nonmetallic properties used in Section III. This will cause a delay in gaining acceptance of nonirradiated graphite properties in the ASME BPV Code. Additional time will be needed while the SG GCC enhances the restructured committee's knowledge on the unique aspects of graphite material properties.

The SG GCC has started work on drafting individual articles of Subsection HH, Subpart B of Division 5, which are the construction/design rules for the carbon/carbon composites. A draft ASTM standard on nuclear grade carbon/carbon composites, developed by SG GCC, was circulated for review by the committee. This standard, intended to define the composition of these materials, will be submitted to the ASTM Committee C28 on Advanced Ceramics for review at their next meeting. This standards is needed by ASME in order to establish a controlled material specification from which applicable construction rules can be written.

3.6.4 Section III Revisions

The ongoing activities as discussed in Section 2.6.4 continued. In light of the close interaction between Division 1 and Division 5 rules (see Subsection 3.6.5), changes to Division 1 rules were also considered in light of their potential affect on high temperature reactors.

3.6.5 Issuance of Division 5 Rules

The biggest achievement for high temperature reactors was the issuance of Section III, Division 5 on November 1, 2011. With an actual published code, not only are personnel more aware of existing ASME BPV Code rules for HTGRs, it also make revisions easier to achieve. New revisions to Division 5 were already approved by the BPV Committee on Construction of Nuclear Facility Components (III) and are expected to achieve full ASME approval by the end of 2012, allowing these changes to be reflected in the new 2013 Edition of the ASME BPV Code.

3.6.6 Stakeholder's Meeting

At the August 2012 BPV Code Week meeting, a High Temperature Reactor Stakeholders meeting was held with the purpose of providing industry a direct means of identifying their needs to the ASME BPV committees. VHTR TDO personnel from the INL supported the BPV Committee on Construction of Nuclear Facility Components (III) Chair in organizing this meeting. The meeting was heavily supported by industry (AREVA, TerraPower, Gen4 Energy, NuScale Power, Mersen, Toyo Tanso, Graf Tech, and more) as well as committee personnel and Department of Energy headquarters personnel. ASME Committee Chairs associated with high temperature reactors provided brief presentations on current status and future plans while industry leaders also provided brief updates on their current efforts on high temperature reactors. Of particular note, AREVA (from the NGNP Industry Alliance) indicated their current focus was to design a 700-750ºC core outlet HTGR and would be using a SA508/SA533 reactor vessel with Alloy 800H and graphite core support structures. [Note that ASME has been effectively moving forward with efforts to support a plant like AREVA's proposal by developing graphite rules and extending Alloy 800H material properties for higher temperatures and longer operating times.] As expected, AREVA indicated no current plans to use Alloy 617 for this lower temperature HTGR. Open and insightful information exchanges were held for the benefit of all attendees. Individuals agreed to continue these meetings at regular future intervals. Therefore, ASME has successfully used the stakeholder's meeting to encourage industry participation on ASME committees and to request industry to provide those committees with specific industry needs and any appropriate information.

3.7 Section VIII, Supporting BPV Code Sections, and B31 Pressure Piping Codes

The ongoing activities as discussed in Section 2.7 continued. Without further details on the normal and off-normal operating conditions of the next HTGR being developed, it is difficult to ensure that these loadings fit within the limits currently established by these codes. Such information, to be determined by the applicant or the applicant's design organization, will need to be evaluated to ascertain if those design parameters fit within the limits of these codes.

4. FUTURE EXPECTATIONS

It is always difficult to accurately predict future achievements, especially regarding the ASME committees since the members are volunteers and have multiple demands on their time. However, the following subsections briefly discuss what the expectations are for ASME codes and standards that address HTGRs.

4.1 HTGR Roadmap

Expectations are that the revised HTGR Roadmap will help support committee efforts to get a more viable and complete Division 5 published in the 2015 Edition of the ASME BPV Code. It will be used as guidance for future Division 5 development and it will be easier for BNCS to judge committee progress toward the identified goals.

4.2 Probabilistic Risk Assessment Standard

The multiple extensions for the first consideration ballot and the concern expressed by committee members that the draft should be editorially reviewed in detail by a task group have delayed pursuing a reconsideration ballot for the draft PRA standard for advanced non-LWRs. However, once this technology neutral proposal has been reviewed and updated, approval by the Joint Committee on Nuclear Risk Management is expected. The Joint Committee on Nuclear Risk Management is also expected to move forward with a reorganization effort to consolidate its lower committee structure.

4.3 Operation and Maintenance Standard

As previously mentioned, due to the generic statement of the technical requirements, the Standards Committee on Operation and Maintenance believes that this standard can apply to many types of reactors, including HTGRs. However, the committee also feels that with more detailed equipment and performance information from HTGRs and small modular reactors, they could better evaluate any need for further changes, especially for those plants that utilize passive systems for their safety function to shutdown the plant and mitigate the consequences of accidents. Lastly, this committee also believes that the OM Standard must further adapt to the global environment, expanding its use throughout the international community. The next 2012 Edition of this standard is expected to be available for use by early 2013.

4.4 Qualification of Mechanical Equipment Standard

There are no specific ongoing HTGR activities being pursued in the Committee on Qualification of Mechanical Equipment Used in Nuclear Facilities. Committee leadership is aware that they may need additional personnel with the appropriate backgrounds in the advanced plant concepts in order to prepare unique qualification requirements for advanced reactors such as HTGRs and LMRs, provided additional equipment details become available. The most significant near-term future goal of the committee is to have a new 2012 Edition available to Code users by early 2013.

4.5 Section XI – Division 2 Code

With the redirection of the SWG on RIM Program, the timeframe to develop a new draft has obviously been extended. However, based on the amount of industry participation, this new draft proposal could be completed and fully approved by ASME in time for the 2015 Edition of the ASME BPV Code. The BPV Committee on Nuclear Inservice Inspection (XI) and the BPV Committee on Construction of Nuclear Facility Components (III) need to ensure proper continuity exists between these two separate sets of requirements for advanced reactors.

4.6 Section III Activities

During the approval balloting for Division 5, a number of commitments were agreed upon. These commitments were to investigate future revisions or changes that the approver felt should be pursued or evaluated but did not feel the issue was sufficient to hold up the initial publication of Division 5. These commitments are being actively worked by many ASME committees. As a result, revisions to Division 5 are already making their way through the ASME approval process, new task groups have been form to address specific issues, or ASME committees have already discussed the issue and have decided not to make any changes. Consideration of Code rules for compact heat exchanges could also begin in the near future but identification of basic design features is needed by the committees before any code development efforts can effectively begin.

The remaining Code revision proposals resulting from Tasks 13, 14 and 14a are expected to continue the ASME approval process well into 2013. Therefore, the expected final publication of these remaining items is the 2015 Edition of the ASME BPV Code. Many other Subsection NH revision proposals are currently in the ASME balloting process and are also expected to be published in the 2015 Edition. Benefitting from the joint efforts of INL and ORNL regarding the elevated temperature testing of Alloy 617, ASME committees should have a substantial database within the next few years from which Alloy 617 can be successfully brought into Section III, Subsection NH, assuming sufficient funding is provided. A task group, recently formed under the Subgroup on Elevated Temperature Design, will focus on developing code revision proposals for the Alloy 617 incorporation effort.

Expectations are that enhancements to the graphite rules would continue as needed simultaneously with the SG GCC continuing efforts to generate rules for composite materials. Any pertinent insights gained from the ongoing graphite testing at INL and ORNL are expected to be brought forth to the SG GCC for their consideration.

4.7 Section VIII, Supporting BPV Code Sections, and B31 Pressure Piping Codes

Without further design details indicating otherwise, the current Section VIII, supporting BPV Code Sections, and B31 Pressure Piping Codes are expected to be adequate as is for construction of an HTGR. Of course, new materials, higher temperature limits, or clarification of advanced NDE needs for certain components may be necessary. However, those issues will be addressed when further details become known.

5. CONCLUSION

ASME has a formidable task. ASME develops and maintains codes and standards that are used internationally to establish rules of safety, develop technology while achieving standardization, and more. This is all done with groups of volunteers from around the world, many of whom have strong opinions on the proper approach to resolve issues. Yet, the ASME consensus process has worked in the past and continues to work effectively today.

DOE and the VHTR TDO continue to support this effort. Supporting the Generation IV Reactor Materials Project and supporting personnel to participate in the committees that develop ASME codes and standards for HTGRs is beginning to provide substantial benefit. In multiple cases, funded personnel have assumed committee leadership positions, are developing proposals, and are actively guiding proposals through the ASME approval process, expediting the issuance of rules for HTGRs. The issuance of the new Section III, Division 5 rules in November 2011 is clear evidence of the impact these funded personnel are having. Division 5 was written, guided through the ASME approval process, and its proper publication supported by funded personnel. But more work is needed to complete this significant and valuable effort. As evidenced in this report, continued funding support will yield more and better ASME codes and standards that can assist in the development of HTGRs.

This report details how ASME is supporting the development of HTGRs by discussing numerous codes and standards from which an HTGR can be constructed, licensed, and operated. Not all of the HTGR requirements are currently published, but the codes and standards process is effectively moving forward. The foundation for HTGR codes and standards has already been laid and the development and approval of probability risk assessment, construction, preservice, operation, and inservice inspection rules will continue. ASME is aggressively pursuing the issuance of these codes and standards to support an international effort to build the next generation of advanced reactor power plants for a worldwide benefit.

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Appendix A

BPTCS and BNCS Charters and Associated Committees of Interest

Appendix A

BPTCS and BNCS Charters and Associated Committees of Interest

Board on Pressure Technology Codes and Standards

Charter:

Management of all ASME activities related to codes, standards, guidelines, and accreditation programs directly applicable to non-nuclear pressure containing equipment.

Associated Committees

ASME/API Joint Committee on Fitness for Service

B16 Standardization of Valves, Flanges, Fittings, and Gaskets Standards Committee

- B16 Materials Technical Committee
- ISO/TC 153 - Valves
- ISO/TC 153/SC 1 - Design, Manufacturing, Marking and Testing
- ISO/TC 153/SC 2 - Valve Actuator Attachment
- ISO/TC 5 Ferrous Metal Pipes and Metallic Fittings
- ISO/TC 5/SC 10 - Metallic Flanges and Their Joints
- ISO/TC 5/SC 5 - Threaded Fittings, Solder Fittings, Welded Fittings, Pipe Threads, Thread Gauge
- Subcommittee B - Threaded Fittings (Except Steel), Flanges, and Flanged Fittings
- Subcommittee C - Steel Flanges and Flanged Fittings
- Subcommittee F - Steel Threaded and Welding Fittings
- Subcommittee G - Gaskets for Flanged Joints
- Subcommittee J - Copper and Copper Alloy Flanges, Flanged Fittings, and Solder Joint Fittings
- Subcommittee L - Gas Shutoffs and Valves
- Subcommittee N - Steel Valves and Face-to-Face and End-to-End Dimensions of Valves

BPV Committee on Materials (II)

- Honorary Members of BPV Committee on Materials
- Materials Database Working Group
- Subgroup on External Pressure
- Subgroup on Ferrous Specifications
- Subgroup on International Material Specifications
- Subgroup on Nonferrous Alloys
- Subgroup on Physical Properties
- Subgroup on Strength of Weldments (BPV II and BPV IX)
- Subgroup on Strength, Ferrous Alloys

BPV Committee on Nondestructive Examination (V)

- Subgroup on General Requirements/Personnel Qualifications and Inquiries
- Subgroup on Surface Examination Methods
- Subgroup on Volumetric Methods
    - Working Group on Acoustic Emissions
    - Working Group on Guided Wave Ultrasonic Testing
    - Working Group on Radiography
    - Working Group on Ultrasonics

BPV Committee on Power Boilers (I)

BPV Committee on Welding and Brazing (IX)

- Subgroup on Brazing
- Subgroup on General Requirements
- Subgroup on Materials
- Subgroup on Performance Qualification
- Subgroup on Plastic Fusing
- Subgroup on Procedure Qualification

B31 Code for Pressure Piping Standards Committee

- B31 Conference Group
- B31 Executive Committee
- B31 Fabrication and Examination Committee
- B31 Forever Medal Nominating Committee
- B31 Honors & Awards Committee
- B31 Materials Technical Committee
- B31 Mechanical Design Technical Committee
- B31 Qualification of Pipeline Personnel Technical Committee
  - B31 Qualification of Pipeline Personnel Technical Committee, India IWG
- B31.1 Power Piping Section Committee
  - B31.1 International Working Group - India
  - B31.1 Subgroup on Design
  - B31.1 Subgroup on Fabrication and Examination
  - B31.1 Subgroup on Materials
  - B31.1 Subgroup on Operation and Maintenance
  - B31.1 Subgroup on Special Assignments
  - B31.1 Subgroup on General Requirements
- B31.12 Hydrogen Piping and Pipelines Section Committee
- B31.3 Process Piping Section Committee
  - B31.3 International Review Group
  - B31.3 Process Piping, India IWG
  - B31.3 Subgroup on Design
  - B31.3 Subgroup on Edit
  - B31.3 Subgroup on Fabrication, Examination and Testing
  - B31.3 Subgroup on General Requirements
  - B31.3 Subgroup on High Pressure Piping
  - B31.3 Subgroup on High Purity Piping
  - B31.3 Subgroup on Inquiry Review
  - B31.3 Subgroup on Materials
  - B31.3 Subgroup on Non-Metallic Piping
- B31.4/11 Liquid and Slurry Piping Transportation Systems
  - B31.4 Liquid and Slurry Transportation Systems, India IWG
  - B31.4/B31.8 Joint Executive Committee
- B31.5 Refrigeration Piping Section Committee
- B31.8 Executive Committee
- B31.8 Gas Transmission and Distribution Piping Systems Section Committee
  - B31.8 Gas Transmission and Distribution Piping Systems, India IWG
  - B31.8 Subgroup on Design, Materials and Construction
  - B31.8 Subgroup on Editorial Review
  - B31.8 Subgroup on Offshore Pipelines
  - B31.8 Subgroup on Operation and Maintenance
- B31.9 Building Services Piping Section Committee

BPV Committee on Transport Tanks (XII)

Bioprocessing Equipment Standards Committee (BPE)

BPV Committee on Fiber-Reinforced Plastic Pressure Vessels (X)

BPV Committee on Heating Boilers (IV)

Committee on Turbine Water Damage Prevention (TWDP)

Pressure Technology Post Construction Committee

BPV Committee on Pressure Vessels (VIII)

- Special Working Group on Bolted Flanged Joints
- Special Working Group on Graphite Pressure Equipment
- Subgroup on Design
  - Working Group on Design-By-Analysis
- Subgroup on Fabrication and Inspection
- Subgroup on General Requirements
  - Task Group on U-2(g)
- Subgroup on Heat Transfer Equipment
- Subgroup on High Pressure Vessels
  - Task Group on Design
  - Task Group on Impulsively Loaded Vessels
  - Task Group on Materials
- Subgroup on Materials
- Subgroup on Toughness (BPV II and BPV VIII)

Pressure Vessels for Human Occupancy (PVHO)

Reinforced Thermoset Plastic Corrosion Resistant Equipment Main Committee (RTP)

Structures for Bulk Solids (SBS)

Technical Oversight Management Committee (TOMC)

Committee on Nonmetallic Pressure Piping Systems

- Subcommittee on Glass Fiber-Reinforced Thermosetting Resin Piping
- Subcommittee on Nonmetallic Materials
- Subcommittee on Thermoplastic Piping

Notes:

1. Lower reporting committees are not shown for those highest level committees that are not of significant interest to the development of rules for HTGRs.
2. BPTCS organization structure on ASME Website as of September 7, 2012.

Board on Nuclear Codes and Standards

Charter:

The charter of the Board on Nuclear Codes and Standards is the management of all ASME activities related to codes, standards and guides directly applicable to nuclear facilities and technology.

Associated Committees

BNCS - Honorary Members

BNCS Committee on Honors and Awards

BNCS Task Group on New Reactors and Globalization

BNCS Task Group on Regulatory Endorsement

Committee on BNCS Operations

Committee on Board (NCS) Strategic Initiatives

Joint Committee on Nuclear Risk Management (JCNRM)

Standards Committee on Nuclear Risk Management (CNRM)

- Addenda Project Team
- CNRM Executive Committee
- CNRM Fire PRA Working Group
- CNRM Interpretations Committee
- CNRM Joint Standard Configuration Control Project Team
- CNRM Subcommittee for ASME/ANS Participation
- CNRM Subcommittee for RISC Participation
- CNRM Subcommittee on Applications
- CNRM Subcommittee on Standards Planning
  - CNRM New Non-Light Water Reactors Plants Working Group (New Non-LWR WG)
- CNRM Subcommittee on Technology
- JCNRM Subcommittee on Standards Development
  - JCNRM SC-SD Advanced Light Water Reactors Writing Group
- JCNRM Subcommittee on Standards Maintenance
  - JCNRM SC-SM Working Group Part 1
  - JCNRM SC-SM Working Group Part 2
  - JCNRM SC-SM Working Group Part 3
  - JCNRM SC-SM Working Group Part 4
  - JCNRM SC-SM Working Group Part 5
- JCNRM Subcommittee on Planning, Implementation, Interface & Interpretations
  - JCNRM SC-PIII Working Group on Implementation
  - JCNRM SC-PIII Working Group on Interface
  - JCNRM SC-PIII Working Group on Interpretation
  - JCNRM SC-PIII Working Group on Planning

Standards Committee on Qualification of Mechanical Equipment Used in Nuclear Facilities

- QME Honorary Members
- Subcommittee on General Requirements
  - Subgroup on Dynamic Qualification
- Subcommittee on Qualification of Active Dynamic Restraints
  - Subgroup on QDR Development
- Subcommittee on Qualification of Pump Assemblies
- Subcommittee on Qualification of Valve Assemblies

BPV Committee on Construction of Nuclear Facility Components (III)

- Executive Committee on Strategy and Project Management
  - Subgroup on Containment Systems for Spent Fuel and High Level Waste Transport Packagings
  - Subgroup on Fusion Energy Devices
  - Subgroup on High Temperature Reactors
    - Working Group on Liquid Metal Reactors
    - Working Group on Nuclear High Temperature Gas–Cooled Reactors
  - Subgroup on Industry Experience for New Plants (BPV III & BPV XI)
  - Subgroup on Editing & Review
  - Subgroup on Polyethylene Pipe
    - Working Group on Research & Development
  - Subgroup on Management Resources
  - BPV III - China International Working Group
  - BPV III – Korea International Working Group
  - Special Working Group for New Advanced Light Water Reactor Plant Construction Issues
- Subcommittee on Design
  - Subgroup on Component Design
    - Working Group on Core Support Structures
    - Working Group on Design of Division 3 Containments
    - Working Group on Piping
    - Working Group on Probabilistic Methods in Design
    - Working Group on Pumps
    - Working Group on Supports
    - Working Group on Valves
    - Working Group on Vessels
    - Special Working Group on Environment Effects
  - Subgroup on Elevated Temperature Construction (BPV I, BPV III, and BPV VIII)
  - Subgroup on Elevated Temperature Design
    - Working Group on Analysis Methods
    - Working Group on Allowable Stress Criteria
    - Working Group on Creep-Fatigue and Negligible Creep
    - Working Group High Temperature Flaw Evaluation (BPV III & BPV XI)
  - Subgroup on Fatigue Strength
  - Subgroup on Graphite Core Components
  - Working Group on Design Methodology
    - Special Working Group on Computational Modeling for Explicit Dynamics
  - Special Working Group on High Density Polyethylene Design of Components
- Subcommittee on Materials, Fabrication & Examination
  - Subgroup on Materials, Fabrication, and Examination
    - Working Group Polyethylene Pipe Materials
- Subcommittee on General Requirements
  - Subgroup on General Requirements
    - Working Group - Duties and Responsibilities
    - Working Group on Quality Assurance, Certification and Stamping
    - Special Working Group on Regulatory Interface
    - Special Working Group General Requirements Graphite & Composite Materials
    - Special Working Group HDPE
- Joint ACI-ASME Committee on Concrete Components for Nuclear Service
  - Working Group Design
  - Working Group Materials, Fabrication and Examination
  - Working Group Modernization

Subgroup on Pressure Relief
Honorary Members of BPV III Standards Committee

BPV Committee on Nuclear Inservice Inspection (XI)

- Executive Committee
- Special Working Group on Reliability and Integrity Management Program
- Special Working Group Nuclear Plant Aging Management
- Special Working Group on Editing and Review
- Subgroup on Evaluation Standards
  - Working Group on Flaw Evaluation
  - Working Group on Operating Plant Criteria
  - Working Group on Pipe Flaw Evaluation
- Subgroup on Nondestructive Examination
  - Working Group on Personnel Qualification and Surface Visual and Eddy Current Examination
  - Working Group on Procedure Qualification and Volumetric Examination
- Subgroup on Repair/Replacement Activities
  - Working Group Non-Metals Repair/Replacement Activities
    - Task Group on Repair by Carbon Fiber Composites
  - Working Group on Design and Programs
  - Working Group on Welding and Special Repair Processes
- Subgroup on Water-Cooled Systems
  - Task Group on High Strength Nickel Alloys Issues
  - Working Group on Containment
  - Working Group on Risk-Informed Activities
  - Working Group on Inspection of Systems and Components
  - Working Group on Pressure Testing
- Working Group on General Requirements

Standards Committee on Operation and Maintenance of Nuclear Power Plants (O&M)

- O&M Executive Committee
- O&M Special Committee on Standards Planning
- O&M Subcommittee on OM Codes
  - O&M Subgroup - ISTA/ISTC
  - O&M Subgroup - ISTB
  - O&M Subgroup - ISTD
  - O&M Subgroup - ISTE
  - O&M Subgroup on Air-Operated Valves
  - O&M Subgroup on Check Valves
  - O&M Subgroup on Diesel Generators
  - O&M Subgroup on Functional Systems
  - O&M Subgroup on Heat Exchangers
  - O&M Subgroup on Loose Parts
  - O&M Subgroup on Motor-Operated Valves
  - O&M Subgroup on OM-29
  - O&M Subgroup on Piping Systems
  - O&M Subgroup on Reactor Internals and Heat Exchangers
  - O&M Subgroup on Relief Valves
  - O&M Subgroup on Rotating Equipment
  - O&M Subgroup on RTDs
  - O&M Task Group on Pump Performance Based IST
  - O&M Working Group on Pneumatically Operated Valves
- O&M Task Group on New Reactor OM Code

Standards Committee on Cranes for Nuclear Facilities
Standards Committee on Nuclear Air and Gas Treatment
Standards Committee on Nuclear Quality Assurance

Notes:
1. Lower reporting committees are not shown for those highest level committees that are not of significant interest to the development of rules for HTGRs.
2. BNCS organization structure on ASME Website as of September 7, 2012 with updated information of the BPV III organization structure provided by ASME Staff.