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REGULATORY SAFETY ISSUES IN THE STRUCTURAL DESIGN CRITERIA OF ASME SECTION III SUBSECTION NH

For Very High Temperatures for VHTR & Gen IV

ASME STANDARDS TECHNOLOGY, LLC

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FOREWORD

This document is the result of work resulting from Cooperative Agreement DE-FC07-05ID14712 between the U.S. Department of Energy (DOE) and ASME Standards Technology, LLC (ASME ST-LLC) for the Generation IV (Gen IV) Reactor Materials Project. The objective of the project is to provide technical information necessary to update and expand appropriate ASME materials, construction, and design codes for application in future Gen IV nuclear reactor systems that operate at elevated temperatures. The scope of work is divided into specific areas that are tied to the Generation IV Reactors Integrated Materials Technology Program Plan. This report is the result of work performed under the regulatory safety area and is entitled "Regulatory Safety Issues in Structural Design Criteria of ASME Section III Subsection NH and for Very High Temperatures for VHTR & Gen IV."

ASME ST-LLC has introduced the results of the project into the ASME volunteer standards committees developing new code rules for Generation IV nuclear reactors. The project deliverables are expected to become vital references for the committees and serve as important technical bases for new rules. These new rules will be developed under ASME's voluntary consensus process, which requires balance of interest, openness, consensus and due process. Through the course of the project ASME ST-LLC has involved key stakeholders from industry and government to help ensure that the technical direction of the research supports the anticipated codes and standards needs. This directed approach and early stakeholder involvement is expected to result in consensus building that will ultimately expedite the standards development process as well as commercialization of the technology.

ASME has been involved in nuclear codes and standards since 1956. The Society created Section III of the Boiler and Pressure Vessel Code, which addresses nuclear reactor technology, in 1963. ASME Standards promote safety, reliability and component interchangeability in mechanical systems.

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ABSTRACT

This Report: 1) identifies the safety issues relevant to the ASME Boiler and Pressure Vessel Code, including Section II, Section VIII, Section III Subsection NH (Class 1 Components in Elevated Temperature Service) and Code Cases that must be resolved in order to support licensing of Generation IV Nuclear Reactors, particularly Very-High-Temperature Gas-Cooled Reactors (VHTRs); 2) describes how Subsection NH addresses these issues; and 3) identifies the needs for additional criteria to cover unresolved safety concerns for very-high-temperature service.

The report also contains a description of the high-temperature structural integrity safety concerns raised by the U.S. Nuclear Regulatory Commission (NRC) and the Advisory Committee on Reactor Safeguards (ACRS) and how these issues are addressed in Subsection NH of the ASME Code.

1 SUMMARY

The U.S. Nuclear Regulatory Commission (NRC) and Advisory Committee on Reactor Safeguards (ACRS) issues which were raised in conjunction with the licensing of the Clinch River Breeder Reactor (CRBR) provide the best early indication of regulatory licensing issues for high-temperature reactors. A construction permit for CRBR was supported by the ACRS with the stipulation that numerous ACRS/NRC technical issues be resolved prior to requesting an operating license. The research and development (R&D) program that was agreed upon to resolve elevated temperature structural integrity licensing issues was never implemented because Congress halted the construction of CRBR. The technical issues included materials, design analysis, weldment integrity, creep ratcheting, creep cracking and creep fatigue—creep rupture damage evaluations. The table in Appendix A lists 25 licensing concerns which the NRC asked the CRBR project to address. This 1983 list provides the most definitive description of NRC elevated temperature structural integrity licensing issues at that time.

Since the 1980s, the ASME Code has made numerous improvements in elevated-temperature structural integrity technology. These advances have been incorporated into Subsection NH of Section III of the Code, "Components in Elevated Temperature Service" [1]. The current need for designs for very high temperature and for Gen IV systems requires the extension of operating temperatures from about 1400˚F (760˚C) to about 1742˚F (950˚C) where creep effects limit structural integrity, safe allowable operating conditions, and design life.

Materials that are more creep and corrosive resistant are needed for these higher operating temperatures. Material models are required for cyclic design analyses. Allowable strains, creep fatigue and creep rupture interaction evaluation methods are needed to provide assurance of structural integrity for such very high temperature applications. Current ASME Section III and NRC design criteria for lower operating temperature reactors are intended to prevent through-wall cracking and leaking.

Section 4 of this report describes the NRC and ACRS work on elevated temperature structural integrity licensing issues subsequent to CRBR. Source documents for the ACRS and NRC sponsored work are References [15] through [21], [23], and [24]) herein of this report. NRC has issued a number of definitive reports (see References [16], [17], [18], [19] and [23] and [24]). Oak Ridge National Lab (ORNL), Argonne National Lab (ANL), and Idaho National Engineering and Environmental Laboratory (INEEL)[1] have performed the other studies and evaluations cited in this report. A list of materials and design basis issues cited in a 1993 ORNL Report is given in section 4.1 where the ten most important issues are identified with an asterisk. Dr. Joseph Muscara of the NRC presented materials engineering needs for advanced high-temperature reactor designs in coordination meetings with the ACRS in 2002. The major issues cited concerned the adequacy of Subsection NH, and Code Cases N-201 and N-499. The safety issues cited are summarized in Section 4.3, along with the relevant NRC research under way in 2002.

ANL provided two reports in 2003 describing a review and assessment of the materials behavior issues, and the Codes and Procedures for high-temperature gas cooled reactors (HTGRs). They state that Subsection NH is considered to be applicable to HTGR components that will operate at relatively low temperatures, and that the scope of Subsection NH needs to be expanded to include materials with higher allowable temperatures and other materials of interest. They cite Alloy 617, 9 Cr-1M0-V steel and Hastelloy X as candidates for core support structures and vessel internals. Their major findings are given in Sections 4.4 and 4.5 of this report.

[1] On February 1, 2005 the Idaho National Engineering and Environmental Laboratory and Argonne National Laboratory-West became the Idaho National Laboratory (INL).

DOE authorized INEEL to conduct a review of technology alternatives for the Next Generation Nuclear Plant (NGNP). Their report—Reference [22]—was published June 30, 2004. While the results of their review, described in Section 4.6, do not necessarily represent the licensing concerns of NRC, their recommendations are certainly relevant to structural design criteria and Code development. The NGNP is to be designed, constructed, licensed and operating by no later than 2020, with a target date for initial operation of 2017. INEEL believes that meeting these objectives will require technology stretch, and warns against attempting too great a stretch. Several nuclear system concepts for a Very-High-Temperature Reactor (VHTR) for the NGNP were reviewed. These include the helium-cooled prismatic reactor, the helium-cooled pebble bed reactor and the molten salt-cooled prismatic reactor. Based on construction time objectives and material development requirements, INEEL recommended that maximum metal temperatures be limited to 900˚C (1652˚F). This corresponds to a maximum core average outlet temperature of 900–950˚C (1652–1742˚F). Even at 900˚F (1652˚F) metal temperature, they state that some reactor core subassemblies might require replacement during the 60-year design plant life.

Stu Rubin of the NRC prepared a presentation on "NGNP Technical Issues Safety Research Needs" in June 2006. His presentation, described in Section 4.7 of this report, discusses the licensing framework, and related components, qualifications, fabrication and testing issues. The major metallic component technical issues he quotes as still requiring resolution include:

- Fatigue, creep and creep fatigue interaction;
- Coolant impurities and crevice concentration impacts;
- Metal carburization, decarburization and oxidation;
- Sensitization of austenitic steels;
- Alloy aging behavior at elevated temperatures;
- The adequacy of in-service inspection; and
- The applicability/adequacy of the ASME Code database.

Appendix B summarizes the current NRC licensing issues for the structural design of VHTR and Gen IV systems. In order to resolve these issues, Subsection NH of Section III of the Code and the Code Cases for elevated- temperature design require further development. The metal temperature limits of the Code need to be extended from 760˚C (1400˚F) to at least 900˚C (1652˚F).

The design lifetime limit of 34 years needs to be extended to 60 years. Additional materials including Alloy 617 and Hastelloy X need to be fully characterized. Environmental degradation effects, especially impure helium and those noted herein, need to be adequately considered. Since cyclic finite element creep analyses will be used to quantify creep rupture, creep fatigue, creep ratcheting and strain accumulations, creep behavior models and constitutive relations are needed for cyclic creep loading. Such strain- and time-hardening models must account for the interaction between the time-independent and time-dependent material response.

The manner in which NRC licensing issues for the structural design of VHTR and Gen IV systems are addressed in the current ASME Subsection NH and Code Cases is described in section 5 of this report. The materials creep behavior, creep fatigue and environmental effects are addressed in Subsection NH and Code Cases largely in terms of design criteria and allowable stress and strain values.

The detailed material properties needed for cyclic finite element creep design analyses are generally not provided in the Code. The minimum strength properties given in the Code are used as anchor values for the more comprehensive material suppliers' average properties. The NRC perspective is that the Code and/or Code Cases currently do not adequately cover the material behavior under cyclic loads in the creep regime, and creep fatigue–creep rupture damage interaction effects.

Subsection NH has rules for the design of welded joints separated into categories A through D. The permissible types of welded joints and their dimensional requirements are specified. Paragraph NH-3353 provides analysis requirements for the design and location of all pressure retaining welds operating at temperatures where creep effects are significant. Reduction factors for creep stress rupture are given as a function of time and temperature. Permissible weld metals are limited and special examination requirements are imposed.

Probably the most restrictive Subsection NH requirements for welds are that the inelastic accumulated strains are limited to one-half the allowable strain limits for the base metal. This has forced designers to keep welds out of high-stress areas. The allowable fatigue at weldments is limited to one-half the design cycles allowed for the base metal. The allowable creep rupture damage at weldments is limited in NH by requiring that the rupture strength be reduced by the weld strength reduction factor when determining the time-to-rupture. The Code also imposes additional examination requirements on Category A thru D welded joints. The adequacy of these and other Code weldment structural design requirements has been questioned by the NRC, even for the temperatures currently covered, which are lower than the VHTR and Gen IV High-Temperature Systems.

Section 6 of this report describes the material models, design criteria and analyses methods which NRC has indicated are remaining needs in the ASME Code to cover Regulatory Issues for Very High Temperature Service. The Code technical committees involved are listed for each of these needs:

1. Material cyclic creep behavior, creep rupture–creep fatigue interaction and environmental effects.
2. The structural integrity of welds
3. The development of extended simplified design analysis methods (to avoid dependence on “black box” finite element analysis (FEA) for cyclic creep)
4. Test verification of 1, 2 and 3

The NRC is currently expanding its staff to deal with their increased licensing workload for Gen III reactors as well as to address Gen IV technical licensing issues. They have expressed concerns about the validity of extending the current technology of Subsection NH to much higher temperatures, and see the need to resolve new corrosion and structural integrity issues for the materials to be used for very-high-temperature applications. Appendix B gives the current (May 2, 2007) NRC Draft for Review, Further Analysis of Elevated Temperature Structural Integrity (Licensing) Issues.

2 INTRODUCTION

The objective of this report is to identify issues relevant to ASME Section III, Subsection NH [1], and related Code Cases that must be resolved for licensing purposes for VHTGRs (Very-High-Temperature Gas Reactor concepts such as those of PBMR Ltd., Areva and General Atomics); and to identify the material models, design criteria, and analysis methods that need to be added to the ASME Code to cover the unresolved safety issues.

Subsection NH was originally developed to provide structural design criteria and limits for elevated-temperature design of Liquid-Metal Fast Breeder Reactor (LMFBR) systems and some gas-cooled systems. The U.S. Nuclear Regulatory Commission (NRC) and its Advisory Committee for Reactor Safeguards (ACRS) reviewed the design limits and procedures in the process of reviewing the Clinch River Breeder Reactor (CRBR) for a construction permit in the late 1970s and early 1980s, and identified issues that needed resolution. In the years since then, the NRC and various contractors have evaluated the applicability of the ASME Code and Code Cases to high-temperature reactor designs such as the VHTGRs, and identified issues that need to be resolved to provide a regulatory basis for licensing.

This report describes:

1. NRC and ACRS safety concerns raised during the licensing process of CRBR
2. How some of these issues are addressed by the current Subsection NH of the ASME Code
3. The material models, design criteria and analysis methods that need to be added to the ASME Code and Code Cases to cover unresolved regulatory issues for very-high-temperature service

3 NRC AND ACRS SAFETY ISSUES IN LICENSING REVIEW OF CRBR

This section describes NRC staff and ACRS safety concerns with regard to the elevated-temperature structural design of LMFBR systems, related to licensing of the CRBR that took place during the late 1970s and early 1980s. The ACRS has statutory responsibilities as described in the Atomic Energy Act of 1954, as amended. The ACRS reviews and advises the Commission with regard to the licensing and operation of production and utilization facilities and related safety issues, the adequacy of proposed reactor safety standards, technical and policy issues related to the licensing of evolutionary and passive plant designs, and other matters referred to it by the Commission.

3.1 Elevated-Temperature Design and Operating Licensing Conditions

In order to assess the relevance of issues identified by the NRC and ACRS licensing reviews of CRBR to the structural design of VHTR and GEN IV systems, it is necessary to consider the specific design and operating conditions of the CRBR.

The Clinch River Breeder Reactor Plant was designed to demonstrate that a liquid-metal fast breeder reactor can operate safely and reliably in an electric utility system. The plant was designed as a 350 MW_e, three-loop system to be located in the Tennessee Valley Authority system at a site on the Clinch River near Oak Ridge, Tennessee. With a reactor vessel outlet temperature of 995˚F (535˚C) it was necessary in the structural design of the plant to take account of loading conditions and component response unique to elevated-temperature service-enhanced thermal transients and gradients, nonlinear deformation and creep of materials, and time-dependent failure modes. With a design life of 30 years it was necessary to take account of material degradation effects due to sustained load and environment, geometry change due to creep, and a potential for loss of function.

Since LMFBR systems operate at low pressure, the sodium containing components (reactor vessel, tanks, piping, heat exchangers, steam generators, pumps, and valves) are relatively thin-walled. Besides steady loads due to pressure, thermal expansion, and dead weight, there are cyclic loads due to thermal transients, pressure changes, and seismic events. Thermally induced stresses become more significant in elevated-temperature systems, so additional attention must be paid to elastic follow-up, strain concentration, and geometrical instability. In contrast with low-temperature design where the response is time-independent, cyclic loads combine with elevated-temperature, time-dependent material behavior making it necessary to follow the actual load history through time and to predict response as a function of time. The ordering of events, as well as the time between events, may have a significant effect on response.

3.2 Structural Integrity Evaluation Approach for Licensing

3.2.1 Modes of Failure to Consider

Elevated-temperature CRBR systems and components were designed to meet the limits of the ASME Boiler and Pressure Vessel Code, Section III, Case N-47 (1981) [2], the forerunner of Subsection NH, which applies for ferritic steels at temperatures above 700˚F (371˚C) and for austenitic stainless steels above 800˚F (427˚C). Failure is prevented by: (1) identifying each possible failure mode, (2) determining the damage criterion for each failure mode, and (3) establishing design rules that appropriately separate design limits from initiation of failure. Other rules rely on control of geometry, design rules to specify details, and design factors based on experience to avoid failure, but do not treat each failure mode explicitly.

Case N-47 is based primarily on design by analysis since it is not possible to develop simple, generally applicable formulas to represent the time-dependent response of complex structures for a 30-year life. However, it did include a number of simplified limits and bounding methods. The latter